TABLE OF CONTENTS

CHAPTER 3.0

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF FIGURES

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a safe 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 10 CFR 100.

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

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

3.1.1 DEFINITION OF SINGLE FAILURE

The single failure criterion is a constraint used in the design of safety systems to improve the reliability of the system to perform its safety function following a design-basis event or design occurrence.

A single failure means an occurrence which results in the loss of the capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming that passive components function properly) nor (2) a single failure of a passive component (assuming that active components function properly) results in a loss of the capability of the system to perform its safety functions.

Single failures are random occurrences imposed upon safety systems that are required to respond to a design basis event. They are postulated despite the fact that the systems were designed to remain functional under the adverse condition imposed by the accident. No mechanism for the cause of the single failure need be postulated. Single

failures of passive components in electrical systems are assumed in designing against a single failure.

3.1.1.1 Active Component

An active component is a device characterized by an expected significant change of state or a discernible mechanical motion in response to an imposed design-basis load demand upon the system. Examples are switches, relays, powered valves, check and safety valves, pressure switches, turbines, transistors, motors, dampers, pumps, analog meters, etc. (see Sections 3.9(B).3.2 and 3.9(N).3.2 for discussions and lists of active pumps and valves).

The definition of an active component for the purpose of supporting the pump and valve operability program includes the Westinghouse nuclear steam supply system (NSSS) check valves. These check valves, although not powered components, meet the definition of having mechanical motion and are therefore included in Table 3.9(N)-11. At the same time, however, they are not considered to be active (powered) components in the Westinghouse design with respect to the Emergency Core Cooling System (ECCS) failure modes and effects analysis (FMEA) of active components or the single active failure analysis for ECCS components. Refer to Section 6.3.2.5.

3.1.1.2 Active Component Failure

An active failure is a failure of an active component to complete its intended function upon demand. Examples of active component failures include the failure of a powered valve to move to its correct position, failure of a pump, fan, or diesel generator to start, failure of a relay to respond, etc.

Certain selected valves that are provided with a power supply for proper system functioning must be prevented from unwanted movement in certain situations. Remote manual power lockout of these valves is provided to preclude unwanted valve motion due to an assumed single electrical failure. The valves are identified in their appropriate sections.

Where the proper active function of a component can be demonstrated despite any reasonable postulated condition, then that component may be considered exempt from active failure. Examples of such components may include code safety valves and check valves. Where such exemption is taken, the basis for the exemption shall be documented in the single failure analysis.

Although Westinghouse NSSS check valves are included in Table 3.9(N)-11, they are not considered to be active components in Table 6.3-5 and 6.3-6. Refer to Section 3.9(N).3.2.1 and Section 6.3.2.5.

3.1.1.3 Passive Component

A passive component is a device characterized by an expected negligible change of state or negligible mechanical motion in response to an imposed design basis demand upon the system. Examples are cables, piping, valves in stationary position, resistors, capacitors, fluid filters, indicator lamps, cabinets, cases, etc.

3.1.1.4 Passive Component Failures

A passive component failure is the structural failure of a static component which limits the component's effectiveness in carrying out its design function. When applied to a fluid system, this means a breach of the pressure boundary is postulated, resulting in abnormal leakage. Such leakage is limited to that which results from a single sprung flange, a single pump seal failure, a single valve stem packing failure, or other single failure mechanisms considered credible by a systematic analysis of system components. The probability of a large break in a piping system (e.g., rupture of ECCS piping), subsequent to the original large LOCA pipe break, is considered to be sufficiently low that it need not be postulated.

Single failures of passive components in electrical systems are assumed in designing against a single failure.

3.1.2 ADDITIONAL SINGLE FAILURE ASSUMPTIONS

In designing for and analyzing for ANS Condition II events (defined in FSAR Section 15.0.1.2), ANS Condition III events (defined in FSAR Section 15.0.1.3), and ANS Condition IV design basis accidents or DBAs (defined in FSAR Section 15.0.1.4), various general and accident sequence-specific analysis assumptions are made. Refer to FSAR Section 15.0 for a discussion of general assumptions including single failure assumptions, initial plant conditions and uncertainties, reactivity coefficients, rod insertion timing and characteristics, reactor trip and ESF actuation setpoints and time delays, available mitigation systems, fission product inventories, decay heat modeling, computer codes, etc. Refer to the specific FSAR Chapter 15 section for a discussion of accident sequence-specific analysis assumptions such as the availability or loss of offsite power, operator actions, etc.

The ANS Condition II and III events analyzed in FSAR Chapter 15 are assumed to not result from a tornado, hurricane, flood, fire, loss of offsite power (except for the FSAR Section 15.2.6 event which may be initiated by a loss of offsite power), or earthquake.

In designing for and analyzing for DBAs (i.e., large break loss-of-coolant accident (LBLOCA), main steam line break, main feedwater line break, rod ejection, locked RCP rotor or RCP shaft break, fuel handling accident, or steam generator tube rupture), the following assumptions (a-f) are made in addition to postulating the initiating event. In designing for and analyzing for an ANS Condition III small break loss-of-coolant accident

(SBLOCA), assumptions (a-e) are made in addition to postulating the initiating event (see also FSAR Sections 6.3.2.5, 6.3.3 Safety Evaluation 9, and 15.6.5).

a. The events are assumed not to result from a tornado, hurricane, flood, fire, loss of offsite power, or earthquake.

b. Any one of the following occurs:

   1. During the short term of an accident, a single failure of any active mechanical component. The short term is defined as less than 24 hours following an accident, or

   2. During the short term of an accident, a single failure of any active or passive electrical component, or

   3. A single failure of passive components associated with long-term cooling capability, assuming that a single active failure has not occurred during the short term. Long-term cooling applies to a time duration greater than 24 hours.

c. No reactor coolant system transient is assumed, preceding the postulated reactor coolant system piping rupture.

d. No operator action is assumed to be taken by plant operators to correct problems during the first 10 minutes following the accident. Although not a design basis accident, operator action times of less than 10 minutes are assumed in the mitigation of an inadvertent ECCS actuation at power event. See Section 15.5.1.

e. All offsite power is simultaneously lost and is restored within 7 days (except that for events postulated to occur during MODE 5, MODE 6, and/or during movement of irradiated fuel assemblies when the plant is in MODE 5 or MODE 6 or with the core fully offloaded, such as a fuel handling accident, a loss of all offsite power is not required to be assumed in addition to a single failure).

f. For a LBLOCA, for additional safety no credit is taken for the functioning of non-seismic Category I components.

In the design and analysis performed for provision of protection of safety-related equipment from hazards and events (tornadoes, floods, missiles, pipe breaks, fires, and seismic events) which could reasonably be expected, the following assumptions were

made (FSAR Section 3.6.1.1 describes the design bases relative to the evaluation of the effects of the pipe failure hazards discussed in Section 3.6.2.):

a. Should the event result in a turbine or reactor trip, loss of offsite power is assumed, and the plant will be placed in a hot standby condition.

b. If required by a Technical Specification limiting condition for operation or if the recovery from the event will cause the plant to be shut down for an extended period of time, the plant will be taken to a cold shutdown (CSD) condition.

c. Redundancy or diversity of systems and components is provided to enable continued operation at hot standby or to cool the reactor to a CSD condition. If required, it is assumed that temporary repairs can be made to circumvent damages resulting from the hazard. All available systems, including non-safety related systems and those systems requiring operator action, may be employed to mitigate the consequences of the hazard.

   In determining the availability of the systems required to mitigate the consequences of a hazard and those required to place the reactor in a safe condition, the direct consequences of the hazard are considered. The feasibility of carrying out operator actions are based on ample time and adequate access to the controls, motor control center, switchgear, etc., associated with the component required to accomplish the proposed action.

d. When the postulated hazard occurs and results in damage to one of two or more redundant or diverse trains, single failures of components in other trains (and associated supporting trains) are not assumed. The postulated hazard is precluded, by design, from affecting the opposite train or from resulting in a DBA. For the situation in which a hazard affects a safety-related component, the event and subsequent activities are governed by Technical Specification requirements in effect when that component is not functional.

e. When evaluating the effects of any earthquake, no other major hazard or event is assumed, and no seismic Category I equipment is assumed to fail as a result of the earthquake. Certain non-seismic Category I components are designed and constructed to ensure that their failure will not reduce the functioning of a safety-related component to an unacceptable safety level.

   This criterion meets the intent of Regulatory Guide 1.29, Position C.2. Evaluation of component failure includes drop impact forces and secondary effects, such as spray and flooding from piping failure.

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

DISCUSSION

The quality assurance programs of SNUPPS and the individual utilities, 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 SNUPPS 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 applicants. Quality assurance is described in Chapter 17.0.

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 FSAR dealing with structures, systems, and components important to safety, e.g., the containment as described in Section 3.8.2.

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

"Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without 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 to fail 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 margin 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 nature and magnitude of the natural phenomena considered in the design of this plant are discussed in Chapter 2.0. Chapter 3.0 discusses the design of the plant in relationship to natural events. Seismic and quality group classifications, as well as other pertinent standards and information, are given in the sections discussing individual structures and components.

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 features systems, and throughout the unit whenever 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 Section 9.5.1. 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 codes, guides, and standards used in the design of the fire protection system and equipment conform to the applicable standards as described in Section 9.5.1.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

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

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 3.0, and the environmental conditions are described in Sections 3.11(B) and 3.11(N).

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 3.0, 5.0, 6.0, 7.0, 9.0, and 10.0. Evaluation of the performance of the safety features is contained in Chapter 15.0.

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

DISCUSSION

Not applicable to single unit site.

3.1.4 PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 10 - REACTOR DESIGN

"The reactor core and associated coolant, control, and protection systems shall be designed with an 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 I) or any transient conditions arising from occurrences of moderate frequency (Condition II) beyond the small fraction of clad defects (1 percent) for which the plant shielding, cleanup, and radwaste systems are designed. Fuel damage, as used here, is defined as penetration of the fission product barrier (i.e., the fuel rod clad). Conditions I and II, as used here, are defined by ANSI N18.2-1973. 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 III event with only a small fraction of the fuel rods damaged, although sufficient fuel damage might occur to preclude the immediate resumption of operation. Condition III, as used here, is defined by ANSI N18.2-1973.

c. The core will remain intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV). Condition IV, as used here, is defined by ANSI N18.2-1973.

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 4.0 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 7.0. This information supports the accident analyses of Chapter 15.0 which show that the acceptable fuel design limits are not exceeded for Condition I and II 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 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 moderator temperature coefficient of reactivity is dependent upon core characteristics, such as fuel loading, the dissolved absorber (boron) concentration, and burnable poisons.

Reactivity coefficients and their effects are discussed in Chapter 4.0.

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 coefficients 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 the Callaway Technical Specifications, 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 4.0. Details of the instrumentation design and logic are discussed in Chapter 7.0.

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 features systems, radiological waste systems, and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the control room in proximity to the controls for maintaining the indicated parameter in the proper range.

The quantity and types of process instrumentation provided ensure safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 6.0, 7.0, 8.0, 9.0, 10.0, 11.0, and 12.0.

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 is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all 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 3.0. The piping is protected from overpressure by means of pressure-relieving devices, as required by 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.

Coolant chemistry is controlled to protect the materials of construction of the reactor coolant pressure boundary from corrosion.

The reactor coolant pressure boundary is accessible for inservice inspections to assess the structural and leaktight integrity. The details are given in Chapter 5.0. For the reactor vessel, a material surveillance program conforming to applicable codes is provided. Chapter 5.0 has additional details.Instrumentation is provided to detect significant leakage from the reactor coolant prssure boundary with indication in the control room, as discussed in Chapter 5.0.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

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

DISCUSSION

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

Additionally, reactor coolant pressure boundary 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, and surveillance samples monitor adherence to expected conditions throughout the plant life.

Multiple safety and relief valves are provided for the reactor coolant system. 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 5.0 discusses the reactor coolant system design.

CRITERION 16 - CONTAINMENT DESIGN

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

DISCUSSION

A steel-lined, prestressed, post-tensioned concrete reactor containment structure encloses the entire reactor coolant system. 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 reactor coolant system 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 reactor containment structure and engineered safety features systems are designed to assure the required functional capability of containing any uncontrolled release of radioactivity. The concrete radiological shielding and the liner within the containment limit the uncontrolled release of radioactivity to the environment.

Refer to Chapters 3.0, 6.0, and 15.0.

CRITERION 17 - ELECTRIC 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 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 a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

"Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated

by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies."

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 8.0, 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 to minimize the vulnerability of the redundant systems to any single credible event.

Two physically independent sources of power provide preferred power to the onsite power system. One preferred circuit is connected to a 13.8/4.16-kV ESF transformer which supplies power normally to its associated 4.16-kV Class 1E bus. The second preferred circuit is connected to one secondary winding of a 3-winding startup transformer which supplies power to a second 13.8/4.16-kV ESF transformer. The second ESF transformer supplies power normally to its associated 4.16-kV Class 1E bus. Each ESF transformer normally supplies power to its associated 4.16-kV Class 1E ac bus, but it can simultaneously supply power to the second 4.16-kV Class 1E bus, if required, by the closure of the circuit breaker. 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.

The onsite ac power is furnished by two diesel generators. Each diesel generator is connected to a Class 1E bus. The ESF loads are divided between the Class 1E busses in a balanced, redundant load grouping. Each diesel generator is capable of supplying sufficient power in sufficient time for the operation of the engineered safety features required for the unit during a postulated loss-of-coolant accident. During a postulated LOCA, both diesel generators start automatically. If preferred power is available to the Class 1E bus following a loss-of-coolant accident, the ESF loads will be started sequentially. However, in the event that preferred power is lost, the load sequencing system will connect the diesel generator to its associated Class 1E bus and sequentially start the ESF equipment. The associated diesel generator is so arranged 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 this system will not impair control of the engineered safety features required to maintain the reactor in a safe condition. Further discussion of GDC-17 is included in Chapter 8 of the Site Addendum.

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, engineered safety feature (ESF) busses, 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 switchyard circuit breakers will be inspected, maintained, and tested on a routine basis without affecting the rest of the system. For details see each Site Addendum. Transmission lines and protective relaying on these lines will be periodically tested.

Any one of the ESF transformers and its circuit to the Class 1E busses can be taken out of service and tested periodically. Each transformer has the capacity to supply power to both group 1 and group 2 Class 1E loads simultaneously. The 4160-V and 480-V circuit breakers and the associated equipment will be tested one at a time only while redundant equipment is operational.

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.

Complete provisions for the testing of Class 1E electric power systems and the standby power supplies (diesel generators) are described in Chapter 8.0. For non-Class 1E systems see Chapter 8 of the Site Addendum.

CRITERION 19 - CONTROL ROOM

"A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions 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 separate control room is provided for the control of each unit from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain in a safe manner under accident conditions, including LOCAs. 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 I requirements, as discussed in Chapter 3.0. Adequate concrete shielding and radiation protection are provided against direct gamma radiation and inhalation doses postulated to result from a TID-14844 release of fission products inside the containment structure. 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 15.0. Fission product removal is provided in the control room recirculation equipment to remove iodine and particulate matter, thereby minimizing the thyroid dose which could result from the accident. The control room habitability features are described in Chapter 6.0.

In the event that the operators are forced to abandon the control room, panel-mounted local instrumentation and controls are provided to achieve and maintain the plant in the hot shutdown condition (see Chapter 7.0). The capability for bringing the plant to a cold shutdown is also provided outside the control room through the use of local controls.

3.1.5 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 intent of 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 engineered safety features 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 engineered safety features 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 high functional reliability and in-service testability. The design employs redundant logic trains and measurement and equipment diversity. The reliability of the system has been verified by analysis which is documented by Reference 1.

The protection system, including the engineered safety features 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 the above 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 reduction 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.)

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 Westinghouse portion of the protection system, refer to Reference 2.

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 so that the environment accompanying any emergency situation in which the components are required to function does not result in the loss of the safety function.

Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. Diverse

protection functions will automatically terminate an accident before intolerable consequences can occur.

Sufficient redundancy and independence is 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 and undervoltage measurements. Trips may also be initiated manually or by a safety injection signal. See Chapter 7.0 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 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(B), 3.10(N), 3.11(B) and 3.11(N) 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 de-energize-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 engineered safety features actuation system provided for actuation of auxiliary feedwater system 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 7.0.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

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

DISCUSSION

The protection system is separate and distinct from the control systems, as described in Chapter 7.0. 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 failure or removal from service of any single protection system component or channel which is common to the control and protection system, leaves intact a system which satisfies the requirements of the protection system. Distinction between channel and train is made in this discussion. The removal of a train from service is allowed only during testing of the train.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

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

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 function 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 15.0. These analyses show that for postulated boron dilution during refueling, startup, manual or automatic operation at power, hot standby, or cold shutdown, 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. Either manual or automatic controls can be used to terminate dilution and initiate boration. 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 (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. Using the rod control system, the operator 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 are presented in Chapter 4.0, and the operation is discussed in Chapter 7.0. The means of controlling the boric acid concentration is described in Chapter 9.0. Performance analyses under accident conditions are included in Chapter 15.0.

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 4.0 and 9.0. Combined use of the rod cluster control system and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full out upon trip for this determination.

CRITERION 28 - REACTIVITY LIMITS

"The reactivity control system 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 reactor coolant system 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 RCCAs and the dilution of the boric acid in the reactor coolant systems are specified in the technical specifications for the facility. The specification includes appropriate graphs that show the permissible withdrawal limits and overlap of the RCCA banks as a function of power. These data on reactivity insertion rates, dilution, and withdrawal limits are also discussed in Chapter 4.0. The capability of the chemical and volume control system to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 9.0. The relationship of the reactivity insertion rates to plant safety is discussed in Chapter 15.0.

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 rigorous 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 4.0 and 7.0.

3.1.6 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 reactor coolant system components are designed, fabricated, inspected, and tested in conformance with the ASME Boiler and Pressure Vessel Code, Section III.

All components are classified according to ANSI-N18.2-1973 and are accorded all the quality measures appropriate to this classification. The design bases and evaluations of the reactor coolant system are discussed in Chapter 5.0.

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 reactor building sump where the level is monitored. Leakage is also detected by measuring the airborne activity of the containment. Indication of containment humidity is also availabe as an indirect indication of leakage. Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank, and coolant drain collection tank provides an accurate indication of integrated leakage. Refer to Chapter 5.0 for complete description of the reactor coolant pressure boundary 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 reactor coolant system to assure that the boundary behaves in a nonbrittle manner. The reactor coolant system materials which are exposed to the coolant are corrosion-resistant stainless steel or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50, Appendix G, "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 test of reactor vessel plate for shear wave 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 program, the evaluation of radiation damage is based on preirradiation and postirradiation testing of Charpy V-notch and tensile specimens. Compact tension (CT) fracture mechanics test specimens, along with the capsules and material left from Charpy V-notch and tensile testing, will be stored by the analyst to support future testing, reconstitution, or reinsertion, unless given NRC approval to discard. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E-185, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and the requirements of 10 CFR 50, 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 reactor coolant system 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 5.0 for details.

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

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

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

DISCUSSION

The design of the reactor coolant pressure boundary 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 vessel shell beneath the nozzles, the top and bottom heads, and external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing the pressure boundary component's integrity. The reactor coolant pressure boundary will be periodically inspected under the provisions of the ASME Code, Section XI.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates forging, weldments, and associated heat treated zones is performed in accordance with 10 CFR 50, 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 not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation will be used to confirm the allowable limits calculated for all operational transients.

The design of the reactor coolant pressure boundary 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. The inservice inspection program is discussed in detail in Chapter 5.2.4.

CRITERION 33 - REACTOR COOLANT MAKEUP

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

DISCUSSION

The chemical and volume control system provides a means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below a preset level. The high-pressure ECCS centrifugal charging pumps provided are capable of supplying the required makeup and reactor coolant seal injection flow when power is available from either onsite or offsite electric power systems. These pumps also serve as high head safety injection pumps. 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 and 9.3 and Chapter 15.0, with details of the electric power system included in Chapter 8.0.

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 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 residual heat removal system functions when temperature and pressure are below approximately 350°F and 400 psig, respectively.

Redundancy of the residual heat removal 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 residual heat removal system redundancy, refer to Section 5.4.7. The residual heat removal 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, four atmospheric relief valves, and the auxiliary feedwater system.

Details of the system design are provided in Section 5.4.7.

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

An emergency core cooling system 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 single failure.

Emergency core cooling is provided even if there should be a failure of any component in the system. A passive system of four accumulators which do not require any external signals or source of power to operate provide the short-term cooling requirements for large reactor coolant pipe breaks. Two independent and redundant high pressure flow and pumping systems, each capable of the required emergency cooling, are provided for

small break protection and to keep the core submerged after the accumulators have discharged following a large break. 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 break 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 is described in Chapter 6.0. The LOCA including an evaluation of consequences, is discussed in Chapter 15.0.

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 non-destructive inservice inspection, as required by the ASME Code, Section XI.

Components outside the containment are accessible for leaktightness inspection during operation of the reactor.

Details of the inspection program for the emergency core cooling system are discussed in Section 6.3.

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

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 may 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/safety injection 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 may be checked during integrated system tests performed during a planned cooldown of the reactor coolant system.

Design provisions include special instrumentation, testing, and sampling lines to perform the tests during plant shutdown to demonstrate proper automatic operation of the ECCS (refer to Appendix 3A for a discussion of Regulatory Guide 1.22). 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 can verify 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 reactor coolant system initially cold and depressurized. The ECCS valving is set to initially simulate the system alignment for plant power operation. Details of the ECCS are found in Chapter 6.0. Performance under accident conditions is evaluated in Chapter 15.0. Surveillance requirements are identified in the Callaway Technical Specifications.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

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

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

DISCUSSION

The containment spray and containment fan cooler systems, in conjunction with the residual heat removal system, 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 residual heat removal system is drawn from the refueling water storage tank. During the later recirculation phase, spray water and reflood water are pumped from the containment sump.

Each of these systems consists of two independent subsystems supplied from separate IE power busses. 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 6.0. Electrical facilities are described in Chapter 8.0. A containment pressure and temperature analysis following a LOCA is given in Chapter 6.0 with additional results found in Chapter 15.0.

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 is outside the containment, except for risers, distribution header piping, spray nozzles, and the containment sump. The containment sump, 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 embedded in concrete and are not accessible for inspection. A portion of the piping from the refueling water storage tank is buried in the ground and 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 Callaway Technical Specifications is met.

For details on the containment air coolers and containment spray system, see Chapter 6.0.

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 leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as 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 containment spray system 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 shutdown, 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 6.0, 8.0, and the Callaway 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 containment spray system serves to remove radioiodine and other airborne particulate fission products from the containment atmosphere following a LOCA. The system consists of two independent systems, each supplied from separate electrical power busses, as described in Chapter 8.0. Either subsystem alone can provide the fission product removal capacity for which credit is taken in Chapter 15.0, in compliance 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 6.0). 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 Section 6.2.5. A hydrogen purge system is provided as a backup. No single failure causes both subsystems to fail to operate.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

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

DISCUSSION

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically, as required. The essential equipment of the containment spray system is outside the containment, except for risers, distribution header piping, and spray nozzles in the containment. The hydrogen purge and monitoring components of the hydrogen control system are located outside the containment. The equipment outside the containment may be inspected during normal power operation. Components of the containment spray system and the hydrogen control system located inside the containment can be inspected during refueling shutdowns. See Chapter 6.0 for details on the containment spray system and details of the hydrogen control system.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the

performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems."

DISCUSSION

The containment spray system 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 Sections 6.2.2 and 6.5.2 for details and Chapter 8.0 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 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 component cooling and essential service water 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.

Complete descriptions of the essential service water system and the component cooling water system are given in Chapter 9.0.

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 integrity and capability of the component cooling water system and portions of the essential service water system are monitored during normal operation by alternating operation of the systems between the redundant system components. Normally, inactive portions of the essential service water system are periodically tested.

The important components are located in accessible areas with the exception of any underground piping for the essential service water system. 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. These systems are discussed in Chapter 9.0.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

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

DISCUSSION

The component cooling system operates continuously during normal plant operation and shutdown, under flow and pressure conditions that approximate the accident conditions. The essential service water system distribution piping utilizes the service water system cooling flow, during normal plant operation, at flows and pressures approximating accident conditions. Provisions are incorporated in the design to allow for periodic starting of the essential service water pumps and verification of the required flowpath at pressure conditions approximating 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.

For a detailed description of the cooling water system, refer to Section 9.2.

3.1.7 REACTOR CONTAINMENT

CRITERION 50 - CONTAINMENT DESIGN BASIS

"The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

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 reactor coolant system 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 features systems minimum safety features. The maximum pressure and temperature reached for a containment design basis accident are presented in Chapter 6.0. Containment design pressure of 60 psig and the design saturation temperature of 320°F provide ample margin to the design basis limits.

See Chapters 3.0 and 6.0 for details.

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

The containment liner plate is a fully silicon kilned, fine-grain practice, normalized plate 1/4-inch thick.

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

Refer to Section 3.8.1 for details.

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 10 CFR 50. Details concerning the conduct of periodic integrated leakage rate tests are included in Chapter 6.0.

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 leaktightness 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 10 CFR 50. Refer to Chapter 6.0.

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 Section 6.2.4. The isolation valve arrangements are discussed in Chapter 6.0.

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 10 CFR 50, as described in Chapter 6.0.

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 - <u>REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT</u>

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

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

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

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

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

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

"Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as

necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs."

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 lines (RHR system) which are part of the reactor coolant pressure boundary 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 Section III, Class 2, specifications and is considered the second passive barrier to fission product release, as described in Chapter 6.0. The arrangement and type of valves utilized are discussed in Chapter 6.0. Containment penetrations are seismic Category I and are protected against possible environmental effects, including missiles.

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

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

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

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

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

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

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

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 6.0.

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 6.0.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

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

DISCUSSION

All containment penetrations are considered to be covered by either GDC-55 or GDC-56. There are no penetrations to which GDC-57 is considered applicable. For a more detailed discussion of containment isolation, refer to Section 6.2.4.

3.1.8 FUEL AND RADIOACTIVITY 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 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 10 CFR 20 during normal operation. The radioactive waste processing system, the design criteria, and the amounts of estimated releases of radioactive effluents to the environment are described in Chapter 11.0.

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 fuel storage 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 fuel storage pool cooling system provides cooling to remove residual heat from the fuel stored in the fuel storage pool. The system is designed with redundancy and testability to assure continued heat removal. The fuel storage pool cooling system is described in Section 9.1.3.

The fuel storage pool is designed so that no postulated accident could cause excessive loss-of-coolant inventory. Accidents are discussed in Chapter 15.0.

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 12.0. Radiation monitoring is provided as discussed in Chapters 11.0 and 12.0.

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

The Independent Spent Fuel Storage Installation (ISFSI) has been designed and licensed under 10 CFR 72 requirements, as appropriate, and is not subject to 10 CFR 50, Appendix A, General Design Criteria.

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 9.0.

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

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

The Independent Spent Fuel Storage Installation (ISFSI) has been designed and licensed under 10 CFR 72 requirements, as appropriate, and is not subject to 10 CFR 50, Appendix A, General Design Criteria.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

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

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 fuel storage pool and discharges it by way of the plant vent. The ventilation system is continuously monitored by gaseous, particulate, and radio-iodine radiation monitors.

If radiation levels reach a predetermined point, there will be an alarm sounded in the control room and the ventilation discharge path will be automatically transferred through filter adsorber units which provides adequate filtration before discharge from the plant vent. See Chapters 7.0, 9.0, and 12.0 for details.

The Independent Spent Fuel Storage Installation (ISFSI) has been designed and licensed under 10 CFR 72 requirements, as appropriate, and is not subject to 10 CFR 50, Appendix A, General Design Criteria.

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

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

DISCUSSION

The containment atmosphere is continually monitored during normal and transient station operations, using the containment particulate, gaseous, and radio-iodine 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 the auxiliary and radwaste buildings are provided to continually monitor 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 station radiation monitoring systems. In addition to the installed detectors, periodic plant environmental surveillance is established. Measurement capability and reporting of effluents will meet the recommendations of Regulatory Guides 4.1 and 1.21. Radiation monitoring systems are discussed in Sections 11.5 and 12.3.4 and Chapter 18.0.

3.1.9 REFERENCES

1. Gangloff, W. C. and Loftus, W. D., "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L(Proprietary) and WCAP-7706 (Non-Proprietary), July, 1971.

2. Katz, D.N., "Solid State Logic Protection System Description," WCAP-7488-L (Proprietary), January, 1971 and WCAP-7672 (Non-Proprietary), June, 1971.

3. Westinghouse Electric Corporation Reference Safety Analysis Report, RESAR-3, Chapter 3.1.1, Pages 3.1-3 and 3.1-2 dated June 1972.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are important to safety because they:

a. Assure the integrity of the reactor coolant pressure boundary.

b. Assure the capability to shut down the reactor and maintain it in a safe condition.

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

d. Contain or may contain radioactive material.

The purpose of this section is to classify structures, systems, and components, according to the importance of the item, in order to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Table 3.2-1 delineates each of the items in the plant which fall under the above-mentioned categories and the respective associated classification that the NRC, ANS, and industrial codes committees have developed. Each of the classification categories in Table 3.2-1 is addressed in the following sections.

For identification of system and subsystem boundaries, Table 3.2-1 is supplemented (i.e., referenced to applicable figures) by piping and instrument diagrams which have been marked to clearly show the limits of the seismic Category I and various quality group classifications on a system. The legend for the piping and instrument diagrams is provided in Figure 1.1-1.

Classification of power supplies, instrumentation and controls, valve operators, supports, hangers, and restraints is not delineated in Table 3.2-1 because of the extensive listing required. Generic listings for piping/valves and ductwork/ dampers are included for completeness, since for some systems these are the only items serving a safety function. Containment penetrations are not included in these generic listings as there is a separate subheading for containment penetrations. The classification for all of these unlisted and generically listed items is consistent with the boundaries shown on the piping and instrumentation drawings. A listing of the piping and instrumentation drawings and their associated FSAR figures is found in Table 1.7-2 and in Section 1.7 of each Site Addendum.

3.2.1 SEISMIC CLASSIFICATION

Seismic classification criteria are set forth in 10 CFR 100 and supplemented by Regulatory Guide 1.29. Clarifications and specific exceptions to Regulatory Guide 1.29 are discussed in Table 3.2-3.

All components classified as Safety Class 1, 2, or 3 (classifications are as defined by Reference 1), are seismic Category I.

Seismic Category I structures, components, and systems are designed to withstand the safe shutdown earthquake (SSE), as discussed in Sections 3.7(B) and 3.7(N), and other applicable load combinations, as discussed in Sections 3.8.1 through 3.8.5. Seismic Category I structures are sufficiently isolated or protected from the other structures to ensure that their integrity is maintained.

Radwaste systems and structures are designated as nonseismic Category I. In accordance with Regulatory Guide 1.143, a simplified seismic analysis is performed for portions of the gaseous radwaste system (which by design are intended to store and delay the release of gaseous radioactive waste), including isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these components from the rest of the system. In addition, a simplified seismic analysis is performed for structures housing radioactive waste management systems in accordance with Regulatory Guide 1.143.

Nonsafety-related structures, systems, and components that must be designed to retain structural integrity during and after an SSE, but do not have to function, are seismically analyzed to ensure that faulted stress limits are not exceeded. These items (for example: piping and piping supports for nonsafety-related piping located over safety-related items) whose continued function is not required are nonseismic Category I and are not controlled by a 10 CFR 50 Appendix B Quality Assurance Program (not Q-listed). The nonseismic Category I Systems Quality Assurance Program is described in Section 17.D of the SNUPPS Quality Assurance Programs for Design and Construction.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

The quality group classification for each water- and steam-containing pressure component is shown in Table 3.2-1. The components are classified according to their importance to safety, as dictated by service and functional requirements and by the consequences of their failure. The quality group classifications and code requirements for the quality of plant process systems meet the intent of Regulatory Guides 1.26 and 1.143. Clarifications and specific exceptions to these guides are discussed in Tables 3.2-4 and 3.2-5, respectively. These tables compare the design to each regulatory position.

The design, fabrication, inspection, and testing requirements of each classification provide the required degree of conservatism in assuring component pressure integrity and operability.

Radioactive waste management systems are designed consistent with Regulatory Guide 1.143, as noted in Tables 3.2-1, 3.2-2 and 3.2-5. The radioactive waste management systems are considered to begin at the interface valve(s) in each line, from other systems provided for collecting wastes that may contain radioactive materials, and to include related instrumentation and control systems. The radioactive waste management systems terminate at the point of controlled discharge to the environment, at the point of recycle back to storage for reuse in the reactor, or at the point of storage of packaged solid wastes prior to shipment offsite to a licensed burial ground. The steam generator blowdown system begins at, but does not include, the outermost isolation valve on the blowdown line and terminates at the point of controlled discharge to the environment, at the point of interface with other liquid waste systems, or at the point of recycle back to the secondary system.

The code requirements applicable to each quality group classification are identified in Table 3.2-2. The quality group classifications and the interfaces between classifications in a system having components of different classifications are indicated on the piping and instrumentation diagram or flow diagram of that system.

3.2.3 SAFETY CLASSES

Table 3.2-1 lists the safety class assigned to applicable systems and components in accordance with ANSI N18.2 (Ref. 1). The criteria (of Ref. 1) are used in the plant design to provide an added degree of assurance that the plant is designed, constructed, and operated without undue risk to the health and safety of the public.

3.2.4 QUALITY ASSURANCE PROGRAM

Quality assurance practices, in accordance with the program outlined in 10 CFR 50, Appendix B, have been applied to activities which influence the ability of items in Safety Classes 1, 2, and 3 to perform their intended safety function. The quality assurance program for design, construction and operation of the plant is described in the Operating Quality Assurance Manual. To fulfill the requirements of the Operating Quality Assurance Manual, those Q-listed items which fall under a quality assurance program are identified in Table 3.2-1.

In addition to the 10 CFR 50, Appendix B, quality assurance program for the safety-related items shown as Q-listed on Table 3.2-1, an augmented quality program is implemented for certain non-safety related items. The quality assurance program for these non-safety items is described in the applicable FSAR section and implemented in administrative procedures.

3.2.5 ENGINEERING CODES AND STANDARDS

The engineering codes and standards are listed in Table 3.2-1. For those components covered by the system quality group classification and the safety classes, the codes and standards employed meet the given classification requirements.

The designs of areas and equipment involving the safety and health of personnel include consideration of the Occupational Safety and Health Administration (OSHA) Requirements, 29 CFR 1910.

3.2.6 LOCATION

Table 3.2-1 identifies the location of each item by building.

3.2.7 REFERENCES

1. "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2, November 1973.

INDEX TO TABLE 3.2-1

INDEX TO TABLE 3.2-1 (Sheet 2)

INDEX TO TABLE 3.2-1 (Sheet 3)

TABLE 3.2-1 CLASSIFICATIONS OF STRUCTURES, COMPONENTS, AND SYSTEMS (15)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
1.0	NSSS AND NUCLEAR AUXILIARY SYSTEMS							
1.1	Reactor Coolant System							
	(Figure 5.1-1)							
	Reactor Vessel and Appurtenances							
	Vessel	Y	A	1	Y-W1	III-1	C	
	Head	Y	A	1	Y-A	III-1	C	
	Studs	Y	A	1	Y-W1	III-1	C	
	Shoes	Y	NA	1	Y-W2	III-1	C	
	Supports	Y	NA	1	Y-B	III-1	C	
	Lower internals structure	Y	NA	2	Y-W3	III/NG	C	
	Upper internals structure	Y	NA	2	Y-W3	III/NG	C	
	Irradiation specimen baskets	Y	NA	2	Y-W3	III/NG	C	
	Irradiation capsules	N	NA	NNS	N	NA	C	
	Irradiation specimens	N	NA	NNS	N	NA	C	
	Fuel assemblies and appurtenances	Y	NA	NA	Y-W3	-	C	
	Control rods	Y	NA	NA	Y-W3	NA	C	
	Primary source rods	Y	NA	NA	Y-W3	NA	C	
	Burnable poison rod assemblies	Y	NA	NA	Y-W3	NA	C	
	Thimble guide tubing	Y	A	1	Y-W2	III-1	C	
	Thimble guide couplings	Y	A	1	Y-W2	III-1	C	
	Thimble seal table and parts	Y	NA	1	Y-W3	III-1	C	
	Flux thimble assembly	Y	B	2	Y-W1	III-2	C	
	Control rod drive mechanism (CRDM), housing only	Y	A(22)	1	Y-A	III-1	C	Non-Class 1E power supply
	CRDM dummy can assemblies	N	NA	NNS	N	NA	C	
	Integrated Head Assembly (IHA) shroud assembly	N	NA	NNS	N	NA	C	The IHA shroud assembly is seismically qualified

TABLE 3.2-1 (Sheet 2)

System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
CRDM seismic support frame assembly, spacer plates and tie rods	Y	NA	1	Y-A	III-1	C	
Thermal sleeves	N	NA	NNS	N	NA	C	
Steam generator							
Tube side - RC	Y	A	1	Y-F	III-1	C	
Shell side - main steam and feedwater	Y	B	2	Y-F	III-2(7)		
Pressurizer	Y	A	1	Y-W3	III-1	C	
Pressurizer heaters	N	NA	1/ NNS (12)	N	NA	C	Power supply is diesel-backed non-Class 1E
Flux mapping frame	N	NA	NNS	N	NA	C	
RC thermowell	Y	A	1	Y-W2	III-1	C	
Pressurizer relief tank	N	D	NNS	N	VIII	C	The PRT is a seismically qualified Section VIII component
RC pump standpipe	N	D	NNS	N	VIII	C	
RC pump:							
Casing and supports	Y	A	1	Y-W3	III-1	C	
in flange	Y	A	1	Y-W3	III-1		
Thermal barrier	Y	A	1	Y-W3	III-1		
Thermal barrier heat exchanger	Y	A	1	Y-W3	III-1		
#1 Seal housing	Y	A	1	Y-W3	III-1		
#2 Seal housing	Y	B	2	Y-W3	III-1		
#3 Seal housing	N	D	NNS	N	NA		
Bolting (Pressure-retaining)	Y	A	1	Y-W3	III-1		
RC Pump Motors							
Shaft coupling	Y	NA	2	Y-W3	NA	C	Power supply is non-Class 1E
Spool piece	Y	NA	2	Y-W3	NA		
Armature	Y	NA	2	Y-W3	NA		

TABLE 3.2-1 (Sheet 3)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Flywheel	Y	NA	2	Y-W3	NA		
	Motor bolting	Y	NA	2	Y-W3	NA		
	Upper oil cooler							
	Tube side-CCW	Y	C	3	Y-W3	III-3		
	Shell side-oil	Y	NA	3	Y-W3	NA		
	Lower oil cooler							
	Tube cooling coil	Y	C	3	Y-W3	III-3		
	Oil reservoir	Y	NA	3	Y-W3	NA		
	Air water coolers	Y	C	3	Y-W3	III-3		
	Motor stand and frame	Y	NA	2	Y-W3	NA		
	Piping/valves	Y	A	1	Y-W1	III-1	C	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	Y	C	3	Y-B	III-3	C	
	Piping/valves	N	D	NNS	N	B31.1	A/C/R	
1.2	Chemical and Volume Control System (Figure 9.3-8)							
	Letdown and Charging Loop							
	Regenerative heat exchanger							
	Tube side - letdown	Y	B	2	Y-W1	III-2/TEMA-R	C	
	Shell side - charging	Y	B	2	Y-W1	III-2/TEMA-R	C	
	Letdown heat exchanger							
	Tube side - letdown	Y	B	2	Y-W1	III-2/TEMA-R	A	
	Shell side - CCW	Y	C	3	Y-W1	III-3/TEMA-R	A	
	Letdown throttle valves	Y	B	2	Y-W2	III-2	A	
	Excess letdown heat exchanger							
	Tube side - letdown	Y	B	2	Y-W1	III-2/TEMA-R	C	
	Shell side - CCW	Y	C	3	Y-W1	III-3/TEMA-R	C	
	Seal water return heat exchanger							
	Tube side - letdown/sealwater	Y	B	2	Y-W1	III-2/TEMA-R	A	

TABLE 3.2-1 (Sheet 4)

System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
Shell side - CCW	Y	C	3	Y-W1	III-3/TEMA-R	A	
Mixed bed demineralizers	N	D(A)	NNS	Y-W2	VIII(7)	A	
Cation bed demineralizers	N	D(A)	NNS	Y-W2	VIII(7)	A	
Boron meter	N	D	NNS	N	B31.1	A	
RC filter	Y	B	2	Y-W1	III-2	A	
Volume control tank	Y	B	2	Y-W1	III-2	A	
ECCS Centrifugal charging pump	Y	B	2	Y-W1	III-2	A	Class 1E power supply. CCW is required.
Normal charging pump	Y	B	2	Y-U	III-2	A	Non-Class 1E power supply.
Seal water injection filter	Y	B	2	Y-W1	III-2	A	
Seal water return filter	Y	B	2	Y-W1	III-2	A	
Boric Acid Makeup Subsystem							
Boric acid tank	Y	C	3	Y-B	III-3	A	
Boric acid transfer pump	Y	C	3	Y-W1	III-3	A	Diesel backed non-Class 1E power supply
Boric acid filter	Y	C	3	Y-W2	III-3	A	
Boric acid batching tank	N	D	NNS	N	VIII	A	
Boron injection makeup pump	N	D	NNS	N	MS	A	
Chemical mixing tank	N	D	NNS	N	VIII	A	
Boron Thermal Regeneration Subsystem							
Moderating HX							
Tube side - letdown	N	D(A)	NNS	N	VIII(7)	A	
Shell side - letdown	N	D(A)	NNS	N	VIII(7)	A	
Letdown chiller HX							
Tube side - letdown	N	D(A)	NNS	N	VIII(7)	A	
Shell side -chilled water	N	D	NNS	N	VIII(7)	A	
Letdown reheat HX							
Tube side - letdown	N	B	2	Y-W1	III-2	A	
Shell side - letdown	N	D(A)	NNS	N	VIII(7)	A	

TABLE 3.2-1 (Sheet 5)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Chiller unit	N	D	NNS	N	NA	A	
	Chiller pump	N	D	NNS	N	MS	A	
	Chiller surge tank	N	D	NNS	N	VIII	A	
	Thermal regeneration demineralizers	N	D(A)	NNS	N	VIII(7)	A	
	Piping/valves	Y	A	1	Y-W1	III-1	C	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	Y	C	3	Y-B	III-3	A/C	
	Piping/valves	N	D	NNS	N	B31.1	A/C	
1.3	Residual Heat Removal System((Figure 5.4-7)							
	RHR Pumps	Y	B	2	Y-W1	III-2	A	Class 1E power supply. CCW required.
	RHR Heat Exchanger							
	Tube side - RC	Y	B	2	Y-W1	III-2	A	
	Shell side - CCW	Y	C	3	Y-W1	III-3		
	Recirculation valve encapsulation	Y	B	2	Y-B	III-2	A	
	Piping/valves	Y	A	1	Y-W1	III-1	C	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	Y	C	3	Y-B	III-3	A	
	Piping valves (16)	N	D	NNS	N	B31.1	A/C	
1.4	Safety Injection System (Figure 6.3-1)							
	Accumulators	Y	B	2	Y-W1	III-2	C	
	Refueling water storage tank	Y	B	2	Y-B	III-2	O	
	Safety injection pumps	Y	B	2	Y-W1	III-2	A	Class 1E power supply. CCW required.
	Piping/valves	Y	A	1	Y-W1	III-1	C	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	Y	C	3	Y-B	III-3	A	

TABLE 3.2-1 (Sheet 6)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Piping valves (16)	N	D	NNS	N	B31.1	A/C	
1.5	Containment Spray System							
	(Figure 6.2.2-1)							
	Containment spray pump	Y	B	2	Y-B	III-2	A	Class 1E power supply
	Containment spray pump eductor	Y	B	2	Y-B	III-2	A	
	Nozzles	Y	B	2	Y-B	III-2	C	
	Recirculation valve encapsulation	Y	B	2	Y-B	III-2	A	
	Containment recirculation sump strainer	Y	NA	2	Y-U	NA	C	
	Loop A & D bioshield debris barriers	Y	NA	2	Y-U	NA	C	
	Loop A & D bioshield debris baskets	Y	NA	2	Y-U	NA	C	
	Bioshield penetration debris barriers	Y	NA	2	Y-U	NA	C	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping valves	N	D	NNS	N	B31.1	A/C	
	TSP-C baskets	N	NA	NA	N	AISC	C	
1.6	Containment Cooling System							
	(Figure 9.4-6)							
	Containment air cooler cooling coil							
	Tube side - ESW	Y	C	3	Y-B	III-3	C	
	Shell side - air	Y	NA	2	Y-B	NA	C	
	Containment air cooler fan	Y	NA	2	Y-B	NA	C	
	Containment air cooler fan motor	Y	NA	2	Y-B	NEMA	C	Class 1E power supply
	Piping/valves	Y	C	3	Y-B	III-3	C	
	Piping (16)	N	D	NNS	N	B31.1	C	
	Ductwork dampers	N	NA	NNS	N	NA	C	
1.7	Containment Isolation							
	Piping	Y	B	2	Y-B	III-2	C/A	
	Flued heads	Y	B	2	Y-B	III-2	C/A	

TABLE 3.2-1 (Sheet 7)

System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
Valves	Y	B	2	Y-B	III-2	C/A	

TABLE 3.2-1 (Sheet 8)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
1.8	Containment Hydrogen Control System							
	(Figures 6.2.5-1 and 9.4-1)							
	Containment hydrogen recombiner	Y	NA	2	Y-B	NEMA	C	Class 1E power supply
	Containment hydrogen mixing fans	Y	NA	2	Y-B	NA	C	
	Containment hydrogen mixing fan motors	Y	NA	2	Y-B	NEMA	C	Class 1E power supply
	Containment hydrogen analyzer	Y	B	2	Y-B	NA	A	Class 1E power supply
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves (16)	N	NA	NNS	N	B31.1	A	
2.0	WATER SYSTEMS							
2.1	Deleted							
2.2	Essential Service Water System (11)							
	(Figure 9.2-2)							
	Essential service water pump	Y	C	3	Y-B	III-3	E	Class 1E power supply
	Essential service water pump prelube storage tank	Y	C	3	Y-B	III-3	E	
	Essential service water self-cleaning strainers	Y	C	3	Y-B	III-3	E	
	ESW prelube storage tank filter	Y	C	3	Y-B	III-3	E	
	Piping/valves	Y	C	3	Y-B	III-3	A/B/C/D/E/F/O	
	Piping/valves	N	D	NNS	N	B31.1	A/B/C/D/E/F/R/T/Z	Vents, drains, etc. and air compressor piping
2.3	Component Cooling Water System							
	(Figure 9.2-3)							
	Component cooling water pump	Y	C	3	Y-B	III-3	A	Class 1E power supply
	Component cooling water heat exchanger							
	Tube side - ESW	Y	C	3	Y-B	III-3/TEMA-R	A	
	Shell side - CCW	Y	C	3	Y-B	III-3/TEMA-R	A	

TABLE 3.2-1 (Sheet 9)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Component cooling water surge tank	Y	C	3	Y-B	III-3	A	
	Component cooling water chemical addition tank	N	D	NNS	N	VIII	A	
	Piping valves	Y	C	3	Y-B	III-3	A/C/F/R	
	Piping valves	N	D	NNS	N	B31.1	A/C/F/R	
2.4	Deleted							
2.5	Fuel Pool Cooling and Cleanup System							
	(Figure 9.1-3)							
	Fuel pool cooling pump	Y	C	3	Y-B	III-3	F	Class 1E power supply
	Fuel pool skimmer pump	N	D	NNS	N	MS	F	
	Fuel pool cleanup pump	N	D	NNS	N	MS	F	
	Fuel pool cooling heat exchanger							
	Tube side - fuel storage pool water	Y	C	3	Y-B	III-3/TEMA-R	F	
	Shell side - CCW	Y	C	3	Y-B	III-3/TEMA-R		
	Fuel pool cleanup demineralizer	N	D	NNS	N	VIII	R	
	Skimmer strainer	N	D	NNS	N	B31.1	F	
	Fuel pool cleanup filter	N	D	NNS	N	VIII	R	
	Skimmer filter	N	D	NNS	N	VIII	R	
	Piping/valves	Y	C	3	Y-B	III-3	C/F	
	Piping/valves	N	D	NNS	N	B31.1	C/F/R	
2.6	Deleted							
2.7	Ultimate Heat Sink							
	(Figure 9.2-2 and Section 9.2.5 of the Site Addendum)							
	Mechanical Draft Cooling Tower							
	Unit	Y	C	3	Y-B	III-3	Z	
	Fans	Y	NA	3	Y-B	NA	Z	
	Fan motors	Y	NA	3	Y-B	NEMA	Z	Class 1E power supply

TABLE 3.2-1 (Sheet 10)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Mechanical draft cooling tower basin retention pond, including slopes and rip rap	Y	NA	3	Y-B	ACI-318-7/ AISC (19)	O	
3.0	FUEL HANDLING AND STORAGE							
	Fuel transfer system							Non-Class 1E
	power supply							
	Conveyor system and controls	N	NA	NNS	N	NA	C/F	
	Remainder of system	N	NA	NNS	N	NA	C/F	
	RCC changing fixture	N	NA	NNS	N	NA	C	
	Fuel transfer							
	Flange	Y	B	2	Y-W2	III/MC	C	
	Tube	Y	B	2	Y-W2	III/MC	C/F	
	Valve	N	D	NNS	N	MS	F	
	Sleeve	Y	B	2	Y-B	III/MC	C/F	
	Fuel storage racks	Y	NA	3	Y-B	NA	F	
	New fuel storage racks	Y	NA	3	Y-W2	NA	F	
	Integrated Head Assembly (IHA) lift tripod	Y	NA	NNS	N	NA	C	The IHA lift tripod is seismically qualified
	Integrated Head Assembly (IHA) missile shield and lifting legs	Y	NA	2	Y-A	III/MC	C	
	Polar crane	S	NA	3	Y-B	NA	C	Non-Class 1E power supply
	Refueling machine	N	NA	NNS	Y-W2 (23)	NA	C	Non-Class 1E power supply
	Cask handling crane	S	NA	3	Y-B	NA	F	Non-Class 1E power supply
	Spent fuel pool bridge crane	S	NA	3	Y-B	NA	F	Non-Class 1E power supply
	Internals lifting device	N	NA	NNS	N	NA	C	
	Spent fuel pool handling tool	Y	NA	3	Y-W2	NA	F	
	VECASP	S	NA(25)	NA	Y-(25)		O	

TABLE 3.2-1 (Sheet 11)

System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
4.0 RADWASTE MANAGEMENT SYSTEMS							
4.1 Boron Recycle System							
(Figure 9.3-11)							
Tanks							
Recycle holdup	N	D(A)	NNS	N	API-650/III-3	R	
Pumps							
Recycle evaporator feed	N	D(A)	NNS	N	MS(7)	R	
Filters							
Recycle evaporator feed	N	D(A)	NNS	N	VIII(7)	R	
Recycle evaporator condensate	N	D(A)	NNS	N	VIII	R	
Miscellaneous							
Recycle evaporator feed demineralizer	N	D(A)	NNS	N	VIII(7)	R	
Recycle evaporator condensate demineralizer	N	D(A)	NNS	N	VIII	R	
Recycle holdup tank vent eductor	N	D(A)	NNS	N	B31.1(7)	R	
Piping/valves	N	D(A)	NNS	N	B31.1	A/R	
Piping/valves	N	D	NNS	N	B31.1	A/R	
4.2 Liquid Radwaste System							
(Figure 11.2-1)							
Tanks							
Laundry and hot shower tank B	N	D(A)	NNS		N	VIII	R
RC drain	N	D(A)	NNS	N	VIII	C	
Floor drain	N	D(A)	NNS	N	VIII	R	
Waste holdup	N	D(A)	NNS	N	VIII	R	

TABLE 3.2-1 (Sheet 12)

System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
Waste monitor	N	D(A)	NNS	N	VIII	R	
Chemical drain	N	D(A)	NNS	N	VIII	R	
Discharge monitor	N	D(A)	NNS	N	API-650	O	
Waste evap. condensate	N	D(A)	NNS	N	VIII	R	
Laundry and hot shower tank A	N	D(A)	NNS	N	VIII	R	
Pumps							
RC drain tank	N	D(A)	NNS	N	MS	R	
Waste evap. feed	N	D(A)	NNS	N	MS	R	
Waste evap. - condensate tank	N	D(A)	NNS	N	MS	R	
Chemical drain tank	N	D(A)	NNS	N	MS	R	
Laundry and hot shower tank B	N	D(A)	NNS	N	MS	R	
Floor drain tank	N	D(A)	NNS	N	MS	R	
Waste monitor tank	N	D(A)	NNS	N	MS	R	
Waste evap. distillate	N	D(A)	NNS	N	MS	R	
Laundry and hot shower tank A	N	D(A)	NNS	N	MS	R	
Discharge monitor tank transfer	N	D(A)	NNS	N	MS	R	
Caustic metering	N	NA	NNS	N	MS	R	
Acid metering	N	NA	NNS	N	MS	R	
Filters							
Waste evap. feed	N	D(A)	NNS	N	VIII	R	
Waste evap. condensate	N	D(A)	NNS	N	VIII	R	
Laundry and hot shower tank A&B	N	D(A)	NNS	N	VIII	R	
Waste monitor tank	N	D(A)	NNS	N	VIII	R	
Floor drain tank	N	D(A)	NNS	N	VIII	R	
Miscellaneous							
RC drain tank heat exchanger							

TABLE 3.2-1 (Sheet 13)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Tube side RC drains	N	D(A)	NNS	N	VIII	C	
	Shell side - CCW	Y	C	3	Y-W1	III-3	C	
	Waste monitor tank demineralizer	N	D(A)	NNS	N	VIII	R	
	Waste evap. condensate demineralizer	N	D(A)	NNS	N	VIII	R	
	Floor drain tank strainer	N	D(A)	NNS	N	NA	R	
	Liquid waste charcoal adsorber	N	D(A)	NNS	N	VIII	R	
	Laundry and hot shower charcoal adsorber	N	D(A)	NNS	N	VIII	R	
	Laundry washing machines	N	NA	NNS	N	MS	A	
	Laundry dryers	N	NA	NNS	N	MS	A	
	Piping/valves	Y	C	3	Y-B	III-3	C	CCW to RCDT Hx
	Piping/valves	N	D(A)	NNS	N	B31.1	A/B/C/R/T	
	Piping/valves	N	D	NNS	N	B31.1	A/B/C/R/T	
4.3	Gaseous Radwaste System (Figure 11.3-1)							
	Waste gas decay tanks	D	D(A)	NNS	N	VIII(7)	R	
	Waste gas compressor package	D	D(A)	NNS	N	MS/VIII(7)	R	
	Catalytic hydrogen recombiner package	D	D(A)	NNS	N	VIII(7)	R	
	Gas traps	D	D(A)	NNS	N	VIII(7)	R	
	Waste gas drain filter	D	D(A)	NNS	N	VIII	R	
	Gas decay tank drain pump	D	D(A)	NNS	N	MS	R	
	Gaseous radwaste drain collection tank	N	D(A)	NNS	N	VIII	R	
	Piping/valves	D	D(A)	NNS	N	B31.1	A/R	
	Piping/valves	N	D	NNS	N	B31.1	A/R	

TABLE 3.2-1 (Sheet 14)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
4.4	Steam Generator Blowdown System							
	(Figure 10.4-8)							
	Tanks							
	Surge tank	N	D(A)	NNS	N	VIII	R	
	Blowdown flash tank	N	D(A)	NNS	N	VIII	T	
	Pumps							
	Discharge	N	D(A)	NNS	N	MS	R	
	Drain	N	D	NNS	N	MS	A	
	Recirculation	N	D	NNS	N	MS	T	
	Miscellaneous							
	Blowdown regenerative heat exchanger	N	D(A)	NNS	N	VIII	T	
	Blowdown nonregenerative heat exchanger	N	D(A)	NNS	N	VIII	T	
	Mixed-bed demineralizer	N	D(A)	NNS	N	VIII(7)	R	
	Filters	N	D(A)	NNS	N	VIII	R	
	Strainers	N	D(A)	NNS	N	VIII	R	
	Recirculation sample cooler	N	D	NNS	N	MS	T	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	N	D(A)	NNS	N	B31.1	B/R/T	
	Piping/valves	N	D	NNS	N	B31.1	B/R/T/A/C	
4.5	Solid Radwaste System							
	(Figure 11.4-1)							
	Caustic addition tank	N	D	NNS	N	VIII	R	
	Solidification package	N	D(A)	NNS	N	VIII	R	
	Spent resin tank (primary)	N	D(A)	NNS	N	VIII(7)	R	
	Spent resin tank (secondary)	N	D(A)	NNS	N	VIII	R	

TABLE 3.2-1 (Sheet 15)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Spent resin sluice pump (primary)	N	D(A)	NNS	N	MS(7)	R	
	Spent resin sluice pump (secondary)	N	D(A)	NNS	N	MS	R	
	Caustic addition metering pump	N	D	NNS	N	MS	R	
	Resin charging tank (CVCS)	N	D	NNS	N	VIII	R	
	Resin charging tank (radwaste)	N	D	NNS	N	VIII	R	
	Spent resin sluice filter (primary)	N	D(A)	NNS	N	VIII	R	
	Spent resin sluice filter (secondary)	N	D(A)	NNS	N	VIII	R	
	Dry waste compactor	N	NA	NNS	N	MS	R	
	Solid radwaste decanting station	N	D(A)	NNS	N	MS/VIII	R	
	Solid radwaste drumming station	N	D(A)	NNS	N	MS	R	
	Solid radwaste cement handling station	N	D	NNS	N	MS/VIII	R	
	Solid radwaste bridge crane	N	NA	NNS	N	NA	R	
	Piping/valves	N	D(A)	NNS	N	B31.1	A/R	
	Piping/valves	N	D	NNS	N	B31.1	A/R	
5.0	SECONDARY CYCLE SYSTEMS							
5.1	Main Steam System							
	(Figure 10.3-1)							
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	Y	C	3	Y-B	III-3	A	TDAFP steam supply
	Piping/valves	N	D	NNS	N	B31.1	A/T	

TABLE 3.2-1 (Sheet 16)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
5.2	Main Feedwater System/Feedwater Heater Extraction, Drains, and Vents							
	(Figure 10.4-6)							
	Feedwater heaters	N	D	NNS	N	VIII/TEMA-C	T	
	Heater drain tank	N	D	NNS	N	VIII	T	
	Heater drain pump	N	D	NNS	N	MS	T	
	Feedwater pump	N	D	NNS	N	MS	T	
	Reheater drain tank	N	D	NNS	N	VIII	T	
	Moisture separator drain tank	N	D	NNS	N	VIII	T	
	Motor driven feedwater pump	N	D	NNS	N	MS	T	
	Main Feedwater Flow Venturi	Y	B	2	Y-U	III-2 (21)	A	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	N	D	NNS	N	B31.1	A/T	Vents, drains, etc. only in aux. bldg.
5.3	Deleted							
5.4	Auxiliary Feedwater System							
	(Figures 10.4-9 and 10.4-10)							
	Motor-driven auxiliary feedwater pump	Y	C	3	Y-B	III-3	A	Class 1E power supply
	Turbine-driven auxiliary feedwater pump	Y	C	3	Y-B	III-3	A	Class 1E power supply
	Piping/valves	Y	B	2	Y-B	III-2	A	
	Piping/valves	Y	C	3	Y-B	III-3	A	
	Piping/valves	N	D	NNS	N	B31.1	A/O	
5.5	Deleted							
5.6	Deleted							
5.7	Deleted							
5.8	Deleted							

TABLE 3.2-1 (Sheet 17)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
5.9	Secondary Liquid Waste System							
	(Figure 10.4-12)							
	SLW charcoal adsorber	N	D	NNS	N	VIII	R	
	SLW demineralizer	N	D	NNS	N	VIII	R	
	SLW oil interceptor	N	D	NNS	N	NA	T	
	SLW drain collector tank	N	D	NNS	N	VIII	T	
	SLW monitor tank	N	D	NNS	N	VIII	R	
	SLW drain collector tank pump	N	D	NNS	N	MS	T	
	SLW discharge pump	N	D	NNS	N	MS	R	
	SLW evaporator feed filter	N	D	NNS	N	VIII	R	
	High TDS transfer tank	N	D	NNS	N	VIII	T	
	High TDS transfer pump	N	D	NNS	N	MS	T	
	High TDS collector tank	N	D	NNS	N	VIII	T	
	High TDS collector tank pump	N	D	NNS	N	MS	T	
	Low TDS transfer tank	N	D	NNS	N	VIII	T	
	Low TDS transfer tank pump	N	D	NNS	N	MS	T	
	Low TDS collector tank	N	D	NNS	N	VIII	T	
	Low TDS collector tank pump	N	D	NNS	N	MS	T	
	Low TDS filters	N	D	NNS	N	VIII	R	
	SLW oil interceptors transfer pump	N	D	NNS	N	MS	T	
	Piping/valves	N	D(A)	NNS	N	B31.1	A/B/R/T	
	Piping/valves	N	D	NNS	N	B31.1	A/B/R/T	
5.10	Deleted							
5.11	Condensate Storage and Transfer System							
	(Figure 9.2-12 and Figure 9.2-17)							
	Condensate storage tank	N	D	NNS	N	API 650	O	
	Hardened condensate storage tank	N	NA	NNS	N	API 650	O	

TABLE 3.2-1 (Sheet 18)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Piping/valves	N	D	NNS	N	B31.1	O/T	
	Non-safety auxiliary feedwater pump	N	D	NNS	N	NA	T	Built to ASME Section III, Class 3; procured as non-safety
6.0	SERVICE SYSTEMS							
6.1	Deleted							
6.2	Standby Diesel Generator Engine (Figures 9.5.5-1, 9.5.6-1, and 9.5.7-1)							
	Lube oil cooler	Y	C	3	Y-B	III-3	D	
	Keep-warm lube oil pump (17)	Y	C	3	Y-U	MS	D	
	Main lube oil strainer (duplex)	Y	C	3	Y-B	III-3	D	
	Fuel oil filter	Y	C	3	Y-B	III-3	D	
	Lube oil heater	Y	C	3	Y-B	III-3	D	
	Compressor aftercooler	N	NA	NNS	N	MS	D	
	Starting air compressor filter	N	NA	NNS	N	MS	D	
	Diesel rocker lube oil filter	Y	NA	NNS	N	MS	D	
	Diesel oil separator	Y	NA	NNS	N	MS	D	
	Motor-driven rocker prelube pump	Y	NA	NNS	N	MS	D	
	Starting air compressor	N	NA	NNS	N	MS	D	
	Starting air dryer	N	NA	NNS	N	MS	D	
	Standby diesel engine	Y	C	3	Y-B	III-3	D	
	Starting air dryer prefilter	N	NA	NNS	N	MS	D	
	Starting air dryer after filter	N	NA	NNS	N	MS	D	
	Starting air instrument dist filter	S	NA	NNS	N	MS	D	
	Lube oil suction strainer	Y	C	3	Y-B	NA	D	
	Engine-driven fuel oil pump	Y	C	3	Y-B	MS	D	
	Engine-driven intercooler pump	Y	C	3	Y-B	MS	D	
	Engine-driven jacket water pump	Y	C	3	Y-B	MS	D	
	Engine-driven lube oil pump	Y	C	3	Y-B	MS	D	
	Engine-driven rocker lube pump	Y	C	3	Y-B	MS	D	

TABLE 3.2-1 (Sheet 19)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Ejector	Y	NA	NNS	N	MS	D	
	Rocker reservoir tank	Y	C	3	Y-B	MS	D	
	Fuel rack supply air tank	Y	C	3	Y-B	III-3	D	
	Starting air pulsation dampener	N	NA	NNS	N	MS	D	
	Lube oil level control tank	Y	C	3	Y-B	(17)	D	
	Lube oil filter	Y	C	3	Y-B	III-3	D	
	Starting air tanks	Y	C	3	Y-B	III-3	D	
	Jacket water heat exchanger	Y	C	3	Y-B	III-3	D	
	Jacket water - expansion tank	Y	C	3	Y-B	III-3	D	
	Keep-warm jacket water pump	Y	C	3	Y-B	III-3	D	
	Intake air filter	Y	NA	NA	Y-B	MS	D	
	Intake air silencer	Y	NA	NA	Y-B	MS	D	
	Exhaust silencer	Y	NA	NA	Y-B	MS	D	
	Engine/generator control panels	Y	NA	NA	Y-B	MS	D	
	Intercooler water heat exchanger	Y	C	3	Y-B	III-3	D	
	Interconnecting piping	Y	C	3	Y-B	III-3	D	
	Fuel oil strainer	Y	C	3	Y-B	III-3	D	
	Auxiliary lube oil tank	Y	C	3	Y-B	III-3	D	
	Jacket water (keepwarm) heater	Y	C	3	Y-B	III-3	D	
	Engine gauge panel	Y	NA	NA	Y-B	MS	D	
	Piping/valves (24)	Y	C	3	Y-B	III-3	D	
	Piping/valves	N	D	NNS	N	B31.1	D	
6.3	Emergency Fuel Oil System (Figure 9.5.4-1)							
	Emergency fuel oil storage tank	Y	C	3	Y-B	III-3	O	
	Emergency fuel oil transfer pump	Y	C	3	Y-B	III-3	O	Class 1E power supply
	Emergency fuel oil day tank	Y	C	3	Y-B	III-3	D	
	Emergency fuel oil strainers	Y	C	3	Y-B	III-3	D	
	Piping/valves	Y	C	3	Y-B	III-3	D/O	

TABLE 3.2-1 (Sheet 20)

System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
Piping/valves	N	D	NNS	N	B31.1	D/O	

TABLE 3.2-1 (Sheet 21)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
6.4	Compressed Air							
	(Figure 9.3-1)							
	Instrument air compressors	N	D	NNS	N	NA	T	
	Air receivers	N	D	NNS	N	VIII	T	
	Emergency accumulators	Y	C	3	YB	III-3	A	
	Piping/valves	Y	C	3	Y-B	III-3	A	
	Piping/valves	N	D	NNS	N	B31.1	A/B/C/D/F/O/R/T	
6.5	Deleted							
6.6	Fire Protection (11)							
	(Section 9.5.1)							
	Standpipes, headers, and valves	N	NA	NNS	N	NFPA	A/B/C/D/E F/O/R/T	
	Sprinkler systems, halogenated extinguishing systems, hose racks, portable extinguishers	N	NA	NA	N	NFPA/UL/ ANI/FM	A/B/C/D/E F/O/R/T	
	Fire detection and alarm system	N	NA	NA	N	NFPA/UL/ ANI/FM	A/B/C/D/E F/O/R/T	
	Main control room fire protection system annunciator and control panel	N	NA	NA	N	MS	B	
	Fire pumps	N	NA	NNS	N	NFPA	U	Non-Class 1E 1 motor driven, 2 diesel
	Piping/valves	Y(20)	NA	NNS	N	NFPA	D	Piping up to deluge valve in diesel generator building is seismically analyzed
6.7	Deleted							
6.8	Deleted							
6.9	Floor and Equipment Drainage System							
	(Figure 9.3-5)							
	General piping, pumps, and sumps	N	NA	NA	N	B31.1	A/B/C/D/F/R/T	
	Auxiliary building isolation valves	Y	C	3	Y-B	III-3	A	

TABLE 3.2-1 (Sheet 22)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
6.10	Nuclear Sampling System							
	(Figures 9.3-2, 9.3-3, 18.2-15)							
	Nuclear sampling panels	N	D	NNS	N	MS	A/R	
	Piping/valves	Y	B	2	Y-B	III-2	A/C	
	Piping/valves	N	NA	NNS	N	B31.1	A/C	
6.11	Deleted							
7.0	HEATING, VENTILATING, AND AIR CONDITIONING							
7.1	Control Building							
7.1.1	Control Room Air Conditioning System							
	(Figure 9.4-1)							
	Control room air conditioning unit							
	Unit	Y	NA	3	Y-B	MS, NEMA (Motor)	B	Class 1E power supply
	Condenser	Y	C	3	Y-B	III-3	B	
	Control room filtration system adsorber train	Y	NA	3	Y-B	ANSI	B	
	Control room filtration fan							
	Fan	Y	NA	3	Y-B	MS	B	
	Motor	Y	NA	3	Y-B	NEMA	B	Class 1E power supply
	Control room pressurization system adsorber train							
	Unit	Y	NA	3	Y-B	ANSI	B	
	Heater	Y	NA	3	Y-B	UL	B	Class 1E power supply
	Control room pressurization fan							
	Fan	Y	NA	3	Y-B	MS	B	
	Motor	Y	NA	3	Y-B	NEMA	B	Class 1E power supply
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.1	B	

TABLE 3.2-1 (Sheet 23)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
7.1.2	Class 1E Electrical Equipment Air Conditioning System							
	(Figure 9.4-1)							
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.1	B	
	Class 1E electric equipment air conditioning system							
	Unit	Y	NA	3	Y-B	MS, NEMA (motor)	B	Class 1E power supply
	Condenser	Y	C	3	Y-B	III-3	B	
7.1.3	Balance of Control Building HVAC Equipment							
	(Figure 9.4-1)							
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.1	B	Control building isolation
	Unit heaters and duct heaters	N	NA	NNS	N	UL	B	Non-Class 1E power supply
	Fans and fan motors	N	NA	NNS	N	NEMA (Motors) MS (Fans)	B	Non-Class 1E power supply
	Fan coil units	N	NA	NNS	N	MS	B	Non-Class 1E power supply
	Booster coils	N	NA	NNS	N	MS	B	Non-Class 1E power supply
	Supply air units	N	NA	NNS	N	MS, NEMA (Motor)	B	Non-Class 1E power supply
	Cooling coils	N	NA	NNS	N	MS	B	Non-Class 1E power supply
	Ductwork/dampers	N	NA	NNS	N	See Sect. 9.4.1	B	
7.2	Fuel Building							
	(Figure 9.4-2)							
7.2.1	Emergency Exhaust System							
	Emergency exhaust fan	Y	NA	3	Y-B	MS	F	
	Emergency exhaust fan motor	Y	NA	3	Y-B	NEMA	F	Class 1E power supply
	Emergency exhaust charcoal adsorber train	Y	NA	3	Y-B	ANSI	F	
	Emergency exhaust electric heater	Y	NA	3	Y-B	NEMA	F	Class 1E power supply
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.2	F	

TABLE 3.2-1 (Sheet 24)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
7.2.2	Pump Room Coolers							
	Pump room cooler							
	Unit	Y	NA	3	Y-B	MS	F	
	Motor	Y	NA	3	Y-B	NEMA	F	Class 1E power supply
	Coil	Y	C	3	Y-B	III-3	F	
7.2.3	Balance of Fuel Building HVAC Equipment							
	Unit heaters	N	NA	NNS	N	MS UL (Elect. Htrs Only)	F	Non-Class 1E power supply
	Supply air units	N	NA	NNS	N	MS	F	Non-Class 1E power supply
	Heating coil units	N	NA	NNS	N	MS	F	Non-Class 1E power supply
	Cooling coils	N	NA	NNS	N	MS	F	Non-Class 1E power supply
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.2	F	Fuel building isolation
	Ductwork/dampers	N	NA	NNS	N	See Sect. 9.4.2	F	
7.3	Auxiliary Building (Figure 9.4-3)							
7.3.1	Pump Room and Penetration Room Coolers							
	Pump/penetration room cooler							
	Unit	Y	NA	3	Y-B	MS	A	
	Motor	Y	NA	3	Y-B	NEMA	A	Class 1E power supply
	Coil	Y	C	3	Y-B	III-3	A	
7.3.2	Balance of Auxiliary Building HVAC Equipment							
	Fans and fan motors	N	NA	NNS	N	MS (Fans) NEMA (Motors)	A	Non-Class 1E power supply
	Unit heaters and duct heaters	N	NA	NNS	N	MS UL (Elect. Htrs Only)	A	Non-Class 1E power supply
	Filter adsorber units	N	NA	NNS	N	ANSI	A	Non-Class 1E power supply
	Supply air units	N	NA	NNS	N	MS	A	Non-Class 1E power supply
	Fan coil units	N	NA	NNS	N	MS	A	Non-Class 1E power supply

TABLE 3.2-1 (Sheet 25)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Exhaust scrubbers	N	NA	NNS	N	MS	A	Non-Class 1E power supply
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.3	A	Auxiliary building isolation
	Ductwork/dampers	N	NA	NNS	N	See Sect. 9.4.3	A	
7.4	Diesel Generator Building Ventilation System (Figure 9.4-7)							
	Diesel generator building ventilation fan	Y	NA	3	Y-B	MS	D	
	Diesel generator building ventilation fan motor	Y	NA	3	Y-B	NEMA	D	Class 1E power supply
	Unit heaters	N	NA	NNS	N	UL	D	Non-Class 1E power supply
7.5	Deleted							
7.6	Essential Service Water Pump House HVAC (Figure 9.4-8)							
	Unit heaters	N	NA	NNS	N	MS	E/Z	Non-Class 1E power supply
	Essential service water pump house fan	Y	NA	3	Y-B	MS	E	
	Essential service water pump house fan motor	Y	NA	3	Y-B	NEMA	E	Class 1E power supply
	UHS cooling tower electrical equipment room fan	Y	NA	3	Y-B	MS	Z	
	UHS cooling tower electrical equipment room fan motor	Y	NA	3	Y-B	NEMA	Z	Class 1E power supply
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.8	E/Z	
7.7	Containment Purge System HVAC (Figure 9.4-6)							
	Supply air units	N	NA	NNS	N	MS	A	Non-Class 1E power supply
	Fans and fan motors	N	NA	NNS	N	NEMA (Motors) MS (Fans)	A	Non-Class 1E power supply
	Filter adsorber unit	N	NA	NNS	N	ANSI	A	
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.6	A	1) Auxiliary building isolation 2) radiation monitor mounting

TABLE 3.2-1 (Sheet 26)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Ductwork/dampers	N	NA	NNS	N	See Sect. 9.4.6	A/C	
7.8	Miscellaneous Buildings HVAC (Figure 9.4-3)							
	Fans and fan motors	N	NA	NNS	N	NEMA (Motors) MS (Fans)	A/C	
	Supply air unit	N	NA	NNS	N	MS	A	
	Unit heaters and duct heaters	N	NA	NNS	N	MS UL (Elect. Htrs Only)	A/C/O/R	
	Ductwork/dampers	Y	NA	3	Y-B	See Sect. 9.4.3	A	Auxiliary building isolation
	Ductwork/dampers	N	NA	NNS	N	See Sect. 9.4.3	A/C	
8.0	CIVIL/ARCHITECTURAL							
8.1	Structures and Buildings							
	Reactor building	Y	NA	2	Y-B	BC-TOP-5A III MC AISC (19)		
	Refueling pool and other internal RB structures	Y	NA	NA	Y-B	ACI-318-71 AISC (19)	C	
	Control building	Y	NA	NA	Y-B	ACI 318-71 AISC (19)		
	Auxiliary building	Y	NA	NA	Y-B	ACI 318-71 AISC (19)		
	Fuel building	Y	NA	NA	Y-B	ACI 318-71 AISC (19)		
	Spent fuel pool	Y	NA	NA	Y-B	ACI 318-71		
	Radwaste building	D	NA	NA	N	ACI 318-71 AISC (19)		
	Solid radwaste storage warehouse	N	NA	NA	N	NA		
	Turbine building	N	NA	NA	N	ACI 318-71 AISC (19)		

TABLE 3.2-1 (Sheet 27)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Essential service water system pumphouse	Y	NA	NA	Y-B	UBC-1973 ACI 318-71 AISC (19)	O	
	Essential service water system electrical duct banks and manholes	Y	NA	NA	Y-B	ACI 318-71		
	Moveable tornado missile barriers	Y	NA	NA	Y-B	ACI 318-71 AISC (19)	O/T	
	Essential service water system discharge structure	Y	NA	NA	Y-B	ACI 318-71	O	
	ESW Supply Lines Yard Vault	Y	NA	NA	Y-U	ACI 318-71	O	
	Diesel generator building	Y	NA	NA	Y-B	ACI 318-71 AISC (19)		
	Supports and foundations for all non-NSSS Category I equipment and tanks	Y	NA	NA	Y-B	ACI 318-71 AISC (19)		
	Refueling water storage tank	Y	B	2	Y-B	III-2	O	
	Access vault for emergency fuel oil tank	Y	NA	NA	Y-B	ACI 318-71 AISC (19)	O	
8.2	Materials for Category I Structures							
	Containment liner plate	Y	NA	NA	Y-B	III - MC VIII	C	Refer to Sections 3.8.1 and 3.8.2 for additional information
	Containment personnel and equipment hatches	Y	NA	NA	Y-B	III - MC	C	
	Watertight doors	Y	NA	NA	Y-B	NA	A	
	Pipe whip restraints	Y	NA	NA	Y-B	NA		
	Missile-resistant doors	Y	NA	NA	Y-B	NA		
	Pressure-resistant doors	Y	NA	NA	Y-B	NA		
	Bullet-resistant doors	Y	NA	NA	N	NA		
	Water stops	N	NA	NA	N	NA	C/A	
	Pool liner plate and gates	N	NA	NA	N	NA	C/F	
	Radiation shielding doors	Y	NA	NA	N	NA		

TABLE 3.2-1 (Sheet 28)

System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
9.0 CONTROL AND INSTRUMENTATION							
(Table 7.1-1)							
BOP engineered safety features actuation system	Y	NA	NA	Y-B	IEEE 279	B	
NSSS engineering safety features actuation and reactor protection system	Y	NA	NA	Y-W3	IEEE 279	A/B/C	
Reactor control system	N	NA	NA	N	CH-7	A/C	
Post-accident containment radiation monitors and safety-related airborne radiation monitors	Y	NA	NA	Y-B	CH-7	F/A/B	
Excore neutron monitoring system	N	NA	NA	N-O	CH-7		
Excore neutron monitor							
Post-accident monitoring system	Y	NA	NA	Y-W3	CH-7	A/B/C/F	
Main control board	Y	NA	NA	Y-B/ W3	CH-7	B	
Safety-related auxiliary control panels	Y	NA	NA	Y-B/ W3	CH-7		
Instrument piping, tubing, fittings, and valves that are connected to quality group Class A or B process systems (9) (10)	Y	B	2	Y-B	III-2	C/A	
Instrument piping, tubing, fittings, and valves that are connected to safety Class 3 process systems (10)	Y	C	3	Y-B	III-3	A/B/C/D/F	
Instrument piping, tubing, fittings, and valves that are connected to NNS process systems	N	D	NNS	N	B31.1	A/B/C/D/F/ I/O/R/T	
10.0 ELECTRICAL POWER SYSTEMS							
10.1 Class 1E Lower Medium Voltage System							
Metal-clad switchgear 4.16 kV	Y	NA	NA	Y-B	IEEE-308, 336	C	
5 kV power cable	Y	NA	NA	Y-B	IEEE-308, 336	C/A/D/I A/I	

TABLE 3.2-1 (Sheet 29)

	System/Component(13)	Seismic Category I (1)	Quality Group Classification (2)	ANS Safety Class (3)	Quality Assurance (4)	Principal Construction Codes and Standards (5)	Location (6)	Remarks
	Large induction motors, 250 hp and larger	Y	NA	NA	Y-B	IEEE-308, 336 NEMA, MG-1		
10.2	Class 1E Low Voltage System							
	Load center unit substations	Y	NA	NA	Y-B	IEEE-308, 336	C/A/I	
	Motor control centers	Y	NA	NA	Y-B	IEEE-308, 336	C/A/D/I	
	600 Volt power and control cable	Y	NA	NA	Y-B	IEEE-308, 336	A/C/D/F/I/R	
	Integral and fractional hp induction motors	Y	NA	NA	Y-B	IEEE-308, 336 NEMA, MG-1	A/C/D/F/I/R	
10.3	Class 1E 125 V DC System							
	Batteries and battery charger	Y	NA	NA	Y-B	IEEE-308, 336	C	
	DC distribution panels	Y	NA	NA	Y-B	IEEE-308, 336	C	
	Emergency lighting dc	Y	NA	NA	Y-B	MS, IEEE-336	C	
10.4	Class 1E Instrument AC Power							
	Vital ac power supply	Y	NA	NA	Y-B	IEEE-308, 336	C	
	120 V ac vital panels	Y	NA	NA	Y-B	IEEE-308, 336	C	
	600 V instrument cable	Y	NA	NA	Y-B	IEEE-308, 336	A/C/D/F/I	
10.5	Reactor Building Cable Penetrations	Y	B	2	Y-B	IEEE-317, 336	A/C	
10.6	Conduit Supports and Tray Supports	Y	NA	NA	Y-B	ASTM	All	
10.7	Raceway Installation	Y	NA	NA	Y-B	IEEE-336	All	
10.8	Load Shedding and Emergency Load Sequencing	Y	NA	NA	Y-B	IEEE-308, 336	B	
10.9	Auxiliary Relay Racks	Y	NA	NA	Y-B	ICEA, NEMA, IEEE-336	A/C	
10.10	Transformers							
	Essential service water	Y	NA	NA	Y-B	IEEE-308	I	
	Regulating	Y	NA	NA	Y-B	IEEE-308	C	
10.11	Status Indicating Systems	Y	NA	NA	Y-B/ W3	IEEE-308, 336	C	
10.12	Local Control Stations	Y	NA	NA	Y-B	IEEE-308, 336	A/D/F	

NOTES TO TABLE 3.2-1

(1) Y - Component is functionally and structurally designed and constructed to meet seismic Category I requirements, as defined in Regulatory Guide 1.29.

S - Category I for structural integrity only.

N - Component is non-Category I. Component is seismically designed and constructed if position C.2 of Regulatory Guide 1.29 applies per Table 3.2-3.

D - Designed and constructed to seismic requirements given in Regulatory Guide 1.143.

(2) A, B, C, D, D(A) - Quality group classification as defined in Regulatory Guide 1.26.
NA - Not applicable to safety classification. Design requirements for components and piping associated with the Quality Group D(A) portions of this system which contain radioactive fluid are augmented by Note 1 of Table 3.2-2.

(3) 1, 2, 3, NNS - Safety classifications as defined in ANSI N18.2.
NA - Not applicable to safety classification.

(4) Quality Assurance Program

All components with Y indicate that the component is subject to utility Quality Assurance Program during plant operation. This includes changes made to components under the utility Design Change Program.

Y-A Component is subject to the Areva Q-listed Quality Assurance Program during design and construction.

Y-B Component is subject to the Bechtel Q-listed Quality Assurance Program during design and construction.

Y-F Component is subject to the Framatome ANP Q-listed Quality Assurance Program during design and construction.

Y-U Component is subject to the utility Q-listed Quality Assurance Program during design and construction.

Y-W1 Component is subject to "Quality Control System Requirements," Westinghouse QCS-1 during design and construction.

Y-W2 Component is subject to "Quality Requirements for Manufacture of Nuclear Plant Equipment," Westinghouse QCS-2 during design and construction.

Y-W3 Component is subject to the quality assurance program of one of the Westinghouse manufacturing divisions during design and construction.

N Component is subject to the requirements of applicable codes and standards and the manufacturer's standard quality assurance program during design and construction.

(5) The principal construction codes and standards are identified as:

I: ASME Boiler and Pressure Vessel Code, Section I

III and 1, 2, 3, MC,NG: ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 1, 2, 3, MC, or NG

VIII: ASME Boiler and Pressure Vessel Code, Section VIII, Division 1

NOTES TO TABLE 3.2-1 (Sheet 2)

B31.1:	ANSI B31.1, Code for Power Piping
TEMA C, R	Tubular Exchanger Manufacturers Association, Class C or Class R
IEEE-279:	Institute of Electrical and Electronics Engineers, Criteria for Protection Systems for Nuclear Power Generating Stations - 1971
IEEE-308:	Institute of Electrical and Electronics Engineers, Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations - 1974
IEEE-317:	Institute of Electrical and Electronics Engineers, Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations - 1976
IEEE-336:	Institute of Electrical and Electronics Engineers, Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations - 1971
NFPA:	National Fire Protection Association
ANI:	American Nuclear Insurers
ARI:	Air Conditioning and Refrigeration Institute
ACI 318-71:	American Concrete Institute, Building Code Requirements for Reinforced Concrete
UBC-1973:	Uniform Building Code (state and/or local building codes may be substituted where they supersede UBC-1973)
ICEA:	Insulated Cable Engineers Association
ASTM:	American Society for Testing and Materials
ANSI:	American National Standards Institute
NEC:	National Electric Code
AISC:	American Institute of Steel Construction, Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th Edition, adopted February 12, 1969, and Supplement Numbers 1, 2, and 3
BC-TOP-5-A	Prestressed Concrete Nuclear Reactor Containment Structures, Revision 3
NEMA:	National Electrical Manufacturers Association
UL:	Underwriters' Laboratories, Inc.
FM:	Factory Mutual
NA:	Design requirements specified by designer with appropriate consideration of the intended service and operating conditions

NOTES TO TABLE 3.2-1 (Sheet 3)

API 650:	American Petroleum Institute, Welded Steel Tanks for Oil Storage - Atmospheric Tanks
MS:	Manufacturer's Standard
CH-7:	Refer to Chapter 7

(6) Location:

A. Auxiliary building

B. Control building/communication corridor

C. Reactor building

D. Diesel generator building

E. Essential service water pumphouse

F. Fuel building

O. Outdoors onsite

R. Radwaste building

T. Turbine building

U. Fire pumphouse

Z. UHS cooling tower and equipment room

(7) Table indicates the required code based on its safety-related importance as dictated by service and functional requirements and by the consequences of their failure. Note that the actual equipment may be supplied to a higher principal construction code than required.

(8) Access for inspection and test required. However, no formal quality program approval is required.

(9) A 3/8-inch restriction is provided for all instrument connections to Quality Group A liquid piping to change the instrument piping Quality Group classification from A to B. A 3/4 instrument connection is used on Quality Group A piping to change the instrument piping quality group classification from A to B.

(10) Requirements of ASME Boiler and Pressure Vessel Code Section III are met, except that the instrument sensing line between the instrument shutoff valve and the instrument is not hydrostatically tested. The instrument sensing line between the process tap and the instrument shutoff valve will be hydrostatically tested in accordance with the Code.

(11) See Site Addendum.

(12) Pressure boundary is Safety Class 1; heaters are electrically NNS.

NOTES TO TABLE 3.2-1 (Sheet 4)

(13) Safety-related instruments and controls are described in SNUPPS FSAR Sections 7.1 to 7.6.

(14) The site drainage system consists of many components including roof drains, site storm drains, culverts and ditches for which no credit is taken in component roof loading or site flooding analyses. However, major modifications to Category I building roofs and the plant railroad spur, roads, and graded surfaces, which are in Zones 1 and 2 of FSAR Addendum Figure 2.4-3, will be evaluated to ensure that such modifications will not result in flooding of Category I structures.

(15) Classification of power supplies, instrumentation and controls, valve operators, supports, hangers, and restraints is not delineated in Table 3.2-1 because of the extensive listing required. Generic listings for piping/valves and ductwork/dampers are included for completeness, since for some systems these are the only items serving a safety function. Containment penetrations are not included in these generic listings as there is a separate subheading for containment penetrations. The classification for all of these unlisted and generically listed items is consistent with the boundaries shown on the piping and instrumentation drawings.

(16) Vents, drains, test connections, etc. only.

(17) These pumps do not carry an N-stamp; however, they are designed and procured with appropriate controls to assure equivalency to ASME Section III Class 3, Seismic Category I, Quality Group C requirements.

(18) See response to Q430.11 for construction code.

(19) Nuclear Construction Issues Group (NCIG)-01, Rev. 2, dated May 7, 1985, "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants." This document provides acceptance criteria for visual inspection of structural welds in nuclear power plants. The development of such acceptance criteria falls within the provisions of the AISC Specifications (Note 5) and AWS D1.1.

(20) Deluge valves KC-XV-0561 and KC-XV-0562 were qualified to seismic Category I requirements as a means to comply to Regulatory Guide 1.29 position C.2.

(21) These components were procured under the guidance of USNRC Generic Letter 89-09-ASME Section III Component Replacements. As such, they are designed and procured with appropriate controls to assure equivalency to ASME Section III Class 2, Seismic Category I, and Quality Group B requirements but are not N/NPT stamped.

(22) Reactor coolant pressure boundary.

(23) The refueling machine gripper, in-mast sipping system, air cylinder, and air cylinder electrical and air lines are not Westinghouse components and QCS-2 is not applicable to these refueling machine components. These components are seismic category II/I components subject to the augmented quality program.

(24) Only those components and piping supplied with the standard diesel engine and which either make up an internal part of the engine or whose design and reliability have been proven through years of previous diesel engine service are not Quality Group C. All other piping, tubing, and components are ASME Section III, Class 3. (See response to Q430.11)

(25) Outriggers of the VECASP are classified as Safety Related.

TABLE 3.2-2 CODE REQUIREMENTS FOR COMPONENTS AND QUALITY GROUPS

QUALITY GROUPS

Component	A	B	C	D(1)(3)
Pressure vessels	ASME B & PV Code Section III, Class 1	ASME B & PV Code Section III, Class 2	ASME B & PV Code Section III, Class 3	ASME B & PV Code Section VIII, Div. 1 or 2, or Section I
Reactor containment pressure vessels (steel)	--	ASME B & PV Code Section III, Class MC	--	--
Pumps	ASME B & PV Code Section III, Class 1	ASME B & PV Code Section III, Class 2	ASME B & PV Code Section III, Class 3	Manufacturer's Standard[2]
Valves	ASME B & PV Code Section III, Class 1	ASME B & PV Code Section III, Class 2	ASME B & PV Code Section III, Class 3	ANSI B31.1.0 Power Piping
Piping	ASME B & PV Code Section III, Class 1	ASME B & PV Code Section III, Class 2	ASME B & PV Code Section III, Class 3	ANSI B31.1.0 Power Piping
015 psig storage tanks	--	ASME B & PV Code Section III, Class 2	ASME B & PV Code Section III, Class 3	API620 or equivalent
Atmospheric storage	--	ASME B & PV Code Section III, Class 2	ASME B & PV Code Section III, Class 3	API650 or API620 or equivalent (Section III for stainless steel) [2]
Heat exchangers	ASME B & PV Code Section III, Class 1 and TEMA "R"	ASME B & PV Code Section III, Class 2 and TEMA "R"	ASME B & PV Code Section III, Class 3 and TEMA "R"	ASME B & PV Code Section VIII, Div. 1 and TEMA "C"

1. Construction of portions of systems identified by Note 2 of Table 3.2-1 use the following augmenting criteria, to the maximum extent possible:
   a. Welded construction. Flanged jointed or suitable rapid disconnect fittings are used only where dictated by maintenance or operational requirements.
   b. Process lines 2-1/2 inches nominal pipe size or above are butt welded (no backing rings are used on resin or evaporator bottom lines). Process lines 2 inches or smaller are socket welded. Instrumentation lines are not considered process lines, and screwed connections may be used. Manual valves are butt welded, except where flanges are dictated.
   c. Material used for construction of pressure-retaining components, primarily carbon steel or austenitic stainless steel, complies with applicable sections of the codes and standards for quality group D (except as delineated in Note 3 below). Malleable wrought or cast iron materials and plastic piping are not used. Manufacturer's material certification of compliance is required.
   d. All welding constituting the pressure boundary of pressure-retaining components is performed by qualified welders employing qualified welding procedures per ASME Code Section IX.
2. No ASME code stamp is required.
3. Two polypropylene line carbon steel 3-inch by 1-inch reducing flanges on each of lines 500-HCD and 501-HCD are fabricated from ASTM A 36 steel plate and not ASTM A 105 steel plate as required by ANSI B32.1.

TABLE 3.2-3 DESIGN COMPARISON TO REGULATORY POSITIONS OF REGULATORY GUIDE 1.29 REVISION 3, DATED SEPTEMBER 1978, TITLED SEISMIC DESIGN CLASSIFICATION

This comparison is presented for the BOP portion of the design. Refer to Appendix 3A for the Westinghouse discussion.

Regulatory Guide 1.29 Position	Union Electric
1. The following structures, systems, and components of a nuclear power plant, including their foundations and supports, are designated as Seismic Category I and should be designed to withstand the effects of the SSE and remain functional. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of these structures, systems, and components.	1. All plant items which are necessary to cope with a LOCA, secondary side break inside containment, or to shut the plant down safely following an SSE in the absence of a LOCA are designed for the SSE. There are, however, some plant items not required following an SSE but which are required to cope with other natural phenomena. For example, a plant item which is required to function only during or following a tornado in order to achieve a safe shutdown must be considered important to safety, but the design of the item for an SSE is unnecessary. Further, there are plant items which serve to mitigate the consequence of certain in-plant occurrences (other than LOCA) which are not considered to occur simultaneously with an SSE. Examples of the latter occurrences are fuel handling or spent fuel cask accidents and loss of control room habitability. Thus, certain items not listed in Regulatory Guide 1.29 are considered important to safety and subject to identification, design and installation requirements, as implemented by Callaway Plant design/modification procedures. Table 3.2-1 itemizes those structures, systems, and components which are designed for a safe shutdown earthquake.

TABLE 3.2-3 (Sheet 2)

Regulatory Guide 1.29 Position	Union Electric
a. The reactor coolant pressure boundary.	a. Complies.
b. The reactor core and reactor vessel internals	b. Complies.
c. Systems* or portions of systems that are required for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system).	c. Complies. See Item 2 below.
d. Systems* or portions of systems that are required for (1) reactor shutdown, (2) residual heat removal, or (3) cooling the spent fuel storage pool.	d. Complies. See Item 2 below.
e. Those portions of the steam systems of boiling water reactors . . .	e. Not applicable to the Callaway Plant.
f. Those portions of the steam and feedwater systems of pressurized water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.	f. Complies with the exception that the words "or remote manual" are considered to be inserted after the word "automatic." This option is included to avoid an unnecessary complication (leading to decreased plant reliability) in the line which is not normally provided with automatic closing valves. Note that valves in lines emanating from the steam generator are for secondary side isolation, not containment isolation.

TABLE 3.2-3 (Sheet 3)

Regulatory Guide 1.29 Position	Union Electric
g. Cooling water, component cooling, and auxiliary feedwater systems* or portions of these systems, including the intake structures, that are required for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, (4) residual heat removal from the reactor, or (5) cooling the spent fuel storage pool.	g. Complies
h. Cooling water and seal water systems* or portions of these systems that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps.	h. Complies.
i. Systems* or portions of systems that are required to supply fuel for emergency equipment.	i. Complies.
j. All electric and mechanical devices and circuitry between the process and the input terminals of the actuator systems involved in generating signals that initiate protective action.	j. Complies.
k. Systems* or portions of systems that are required for (1) monitoring of systems important to safety and (2) actuation of systems important to safety.	k. Complies.
l. The fuel storage pool structure, including the fuel racks.	l. Complies, with the clarification that the pool liner plate and gates are not designated as seismic Category I. (See Section 9.1.2)

TABLE 3.2-3 (Sheet 4)

Regulatory Guide 1.29 Position	Union Electric
m. The reactivity control systems, e.g., control rods, control rod drives, and boron injection system.	m. Complies.
n. The control room, including its associated equipment needed to maintain the control room within safe habitability limits for personnel and safe environmental limits for vital equipment.	n. Complies.
o. Primary and secondary reactor containment.	o. Complies. Note that the Callaway Plant design does not incorporate a secondary containment.
p. Systems*, other than radioactive waste management systems, not covered by items 1.a through 1.o above that contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using meteorology as prescribed by Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors") that are more than 0.5 rem to the whole body or its equivalent to any part of the body.	p. Complies. Note that Regulatory Guide 1.143 provides guidance on radioactive waste management systems and structural seismic design. Table 3.2-1 indicates those systems for which the D (Augmented) design criteria are applied. The dividing line value of 0.5 rem is inappropriate for the types of failures which the guide addresses. Quality Group D or D (Augmented) is applied to such systems unless their failure would result in offsite doses approaching the guide values of 10 CFR Part 100.

TABLE 3.2-3 (Sheet 5)

Regulatory Guide 1.29 Position	Union Electric
q. The Class 1E electric systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for functioning of plant features included in items 1.a through 1.p above.	q. Complies; however, in certain cases Class 1E conduits are supported from non-Category I seismic walls. Although not Category I, these reinforced block walls are analyzed for SSE loads in accordance with position 2 and are subject to the identification, design and installation requirements, as clarified in Paragraph 4 below..
2. Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.q above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure.	2. Complies, including the following clarification: Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature to an unacceptable level included in items 1.a through 1.q above, which is required for safe shutdown of the plant, are designed and constructed so that the SSE will not cause such a failure. Although LOCA or major natural phenomenon or DBE is not postulated to occur at the time of an SSE, in addition to those safety-related items required for safe shutdown all systems required to mitigate the consequences of LOCAs and secondary side breaks inside containment are protected from nonseismic items. Since tornadoes are not postulated to occur with an SSE, the contents of the boric acid tank room are not protected from adverse seismic interactions. This system is only relied upon following a tornado induced loss of the RWST. The system is designed in accordance with position 1.m above.

TABLE 3.2-3 (Sheet 6)

Regulatory Guide 1.29 Position	Union Electric
	For these items, Callaway Plant implementing procedures ensure identification, design and installation is used to meet the intent of Paragraph C.4. See response and clarification to Paragraph C.4 below.
3. Seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of structures, systems, or components that form interfaces between Seismic Category I features should be designed to Seismic Category I requirements.	3. Seismic Category I design analysis requirements are extended to the first seismic restraint beyond the defined boundaries. Since seismic analysis of a piping system requires division of the system into discrete segments terminated by fixed points, this means that the seismic analysis cannot be terminated at a seismic restraint, but is extended to include the interface piping out to the first point in the system which can be treated as an anchor to the plant structure. Inasmuch as the seismic analysis is based upon minimum material properties and documented system hydrostatic and performance tests are made, the nonsafety-related portion of the system (including supports) past the interface boundary valve is not seismic Category I and will not be Q-Listed.
	For these items, Callaway Plant implementing procedures ensure identification, design and installation is used to meet the intent of Paragraph C.4. See response and clarification to Paragraph C.4 below.

TABLE 3.2-3 (Sheet 7)

Regulatory Guide 1.29 Position	Union Electric
4. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of those portions of structures, systems, and components covered under Regulatory Positions 2 and 3 above.	4. The items covered under Regulatory Positions 2 and 3 above are not considered to be seismic Category I and are not considered to be Q-listed. For these items, design aspects such as, design control and design reviews (commensurate with safety considerations and induced loads during an SSE) are carried out in Callaway Plant design development/ modification procedures. Additionally, Seismic Category I analysis incorporates standard commercially available equipment, which can be procured non-safety related for those portions of the analysis which are not Seismic Category I and are not Q-listed. Material specifications are obtained only as required by applicable ASME, ANSI and ASTM Codes when required per design. Callaway Plant work control, installation inspections and performance tests will ensure that equipment and materials conform to the original design/material requirements; e.g. original design form, fit, and functions.

* The system boundary includes those portions of the system required to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.

TABLE 3.2-4 DESIGN COMPARISON TO REGULATORY GUIDE 1.26 REVISION 3, DATED MARCH 1976, TITLED "QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS”

Quality group classifications and standards for plant systems and components meet the intent of Regulatory Guide 1.26. However, certain clarifications and specific exceptions to the guide are necessary.

In Paragraphs A and B of the regulatory guide there is a different usage of the term "important to safety" than that used elsewhere in the regulations and regulatory guides. The guide includes components which fall into quality group D under the definition of "important to safety," which implies that a quality assurance program in accordance with 10 CFR 50, Appendix B, should be applied. These quality assurance requirements are neither applied to quality group D components nor are they applied to quality group D (augmented) components. The definition of the term "important to safety," insofar as quality assurance (Appendix B) is concerned, is considered to be that which appears in the introduction of Regulatory Guide 1.29.

Regulatory Guide 1.26 establishes the quality group classification for steam and water containing components. However, the guidance is also used to establish the quality group classification of other systems. These systems are designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed. Table 3.2-1 itemizes the classification for these systems and components. Sections 3.9(B).3 and 3.9(N).3 discuss design for components not covered by the ASME Code. Below is a comparison of the Callaway Plant design with each of the regulatory guide positions.

Regulatory Guide 1.26 Position	Union Electric
1. The group B quality standards given in Table 1 of the guide should be applied to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves that are either part of the reactor coolant pressure boundary defined in Section 50.2(v) but excluded from the requirements of Section 50.55a pursuant to footnote 2 of that section or not part of the reactor coolant pressure boundary but part of:	1. Complies.

TABLE 3.2-4 (Sheet 2)

Regulatory Guide 1.26 Position	Union Electric
a. Systems or portions of systems important to safety that are designed for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) postaccident fission product removal.	a. Complies.
b. Systems or portions of systems important to safety that are designed for (1) reactor shutdown or (2) residual heat removal.	b. Systems which perform the functions of reactor shutdown and residual heat removal are placed in quality group B, as indicated by the guide. This is limited to include only the minimum of those systems which must function in the performance of an orderly safe shutdown and maintenance of the plant in the safe (hot) shutdown condition. Those systems which may be used in the performance of a normal cold shutdown (such as the reactor coolant pumps) or incidentally in the removal of residual heat from the reactor [i.e., heat removal is not their prime function (such as portions of the CVCS)] are not placed in quality group B.
c. Those portions of the steam systems of boiling water reactors. . .	c. Not applicable to the Callaway Plant.

TABLE 3.2-4 (Sheet 3)

Regulatory Guide 1.26 Position	Union Electric
d. Those portions of the steam and feedwater systems of pressurized water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.	d. Specific exceptions taken to placing portions of main steam and feedwater lines in quality group B are as follows: (1) The words "or remote manual" are considered to be inserted after the word "automatic." This option is included to avoid an unnecessary complication (leading to decreased plant reliability) in lines which would not normally be provided with automatic closing valves. (2) Note that valves in lines emanating from the steam generator are for secondary side isolation, not containment isolation.
e. Systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.	e. Complies.

TABLE 3.2-4 (Sheet 4)

Regulatory Guide 1.26 Position		Union Electric	
2.	The group C quality standards given in Table 1 of the guide should be applied to water-, steam-, and radioactive-waste containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves not part of the reactor coolant pressure boundary or included in quality group B but part of:	2.	Complies as noted below:
a.	Cooling water and auxiliary feedwater systems or portions of these systems important to safety that are designed for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, or (4) residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems). Portions of these systems that are required for their safety functions and that (1) do not operate during any mode of normal reactor operation and (2) cannot be tested adequately should be classified as group B.	a.	Complies.
b.	Cooling water and seal water systems or portions of these systems important to safety that are designed for functioning of components and systems important to safety, such as reactor coolant pumps, diesels, and control room.	b.	Complies.

TABLE 3.2-4 (Sheet 5)

Regulatory Guide 1.26 Position	Union Electric
c. Systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.	c. Complies.
d. Systems, other than radioactive waste management systems, not covered by items 2.a through 2.c above that contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using meteorology as recommended by Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors") that exceed 0.5 rem to the whole body or its equivalent to any part of the body. For those systems located in Seismic Category I structures, only single component failures need be assumed.	d. Complies. Note that Regulatory Guide 1.143 provides guidance on radioactive waste management system design. Table 3.2-1 indicates those systems to which the D (Augmented) design criteria are applied. The dividing line value of 0.5 rem is inappropriate for the types of failures which the guide addresses. Quality Group D [or D (Augmented)] is applied to such systems unless their failure would result in offsite doses approaching the guideline values of 10 CFR Part 100.
However, no credit for automatic isolation from other components in the system or for treatment of released material should be taken unless the isolation or treatment capability is designed to the appropriate seismic and quality group standards and can withstand loss of offsite power and a single failure of an active component.	Radwaste systems, except for portions of the steam generator blowdown system located in the turbine building, are located within a seismically designed building as permitted by Regulatory Guide 1.143, and only single component failures are considered.

TABLE 3.2-4 (Sheet 6)

Regulatory Guide 1.26 Position	Union Electric
3. The group D quality standards given in Table 1 of this guide should be applied to water- and steam-containing components not part of the reactor coolant pressure boundary or included in quality groups B or C but part of systems or portions of systems that contain or may contain radioactive material.	3. Complies. In addition, quality standards for D (Augmented) systems are consistent with Regulatory Guide 1.143.

TABLE 3.2-5 DESIGN COMPARISON TO REGULATORY GUIDE 1.143, FOR COMMENTS DATED JULY, 1978, TITLED "DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS"

Design requirements of this regulatory guide are applied to components, systems, and structures which fall under the D (augmented) classification established by Regulatory Guide 1.26, position C.2.d and Regulatory Guide 1.29, Position C.1.p. The design requirements of this guide are therefore applied to the following systems or portions of systems:

a. Purification portion of CVCS

b. Boron thermal regeneration portion of CVCS

c. Boron recycle system

d. Liquid radwaste system

e. Gaseous radwaste system

f. Deleted

g. Steam generator blowdown system

h. solid radwaste system

The radioactive waste management systems are considered to begin at the interface valve(s) in each line from other systems provided for collecting wastes that may contain radioactive materials and to include related instrumentation and control systems. The radioactive waste management systems terminate at the point of controlled discharge to the environment, at the point of recycle back to storage for reuse in the reactor, or at the point of storage of packaged solid wastes prior to shipment offsite to a licensed burial ground. The steam generator blowdown system begins at, but does not include, the outermost isolation valve on the blowdown line, and terminates at the point of controlled discharge to the environment, at the point of interface with other liquid waste systems, or at the point of recycle back to the secondary systems.

Augmentation requirements do not apply to instrumentation and sampling piping downstream of the system root or isolation valve. Although this exception is not discussed in Rev. 0 of Regulatory Guide 1.143, it was accepted in Rev. 1.

TABLE 3.2-5 (Sheet 2)

Regulatory Guide 1.143 Position	Union Electric
1. Systems Handling Radioactive Materials in Liquids	
1.1 The liquid radwaste treatment system, including the steam generator blowdown system downstream of the second containment isolation valve, should meet the following criteria:	1.1 Applies to the systems identified above.
1.1.1 These systems should be designed and tested to requirements set forth in the codes and standards listed in Table 1, supplemented by the provisions in 1.1.2 and in regulatory position 4 of this guide.	1.1.1 Complies. See Table 3.2-2.
1.1.2 Materials for pressure-retaining components should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications. Manufacturers' material certificates of compliance with material specifications, such as those contained in the codes referenced in Table 1, may be provided in lieu of certified material test reports.	1.1.2 Complies. Carbon steel, stainless steel, or other similar materials compatible with the chemical, physical, and radioactive environment are used for pressure-retaining components. The use of malleable, wrought, or cast iron materials or plastic pipe is not allowed. Material certificates of compliance or certified material test reports are required for the materials.
1.1.3 Foundations and walls of structures that house the liquid radwaste system should be designed to the seismic criteria described in regulatory position 5 of this guide, to a height sufficient to contain the maximum liquid inventory expected to be in the building.	1.1.3 Complies. See Section 3.8.6.4.
1.1.4 Equipment and components used to collect, process, and store liquid radioactive waste need not be designed to the seismic criteria given in regulatory position 5 of this guide.	1.1.4 Complies. Liquid contained sources are not seismically designed.

TABLE 3.2-5 (Sheet 3)

Regulatory Guide 1.143 Position	Union Electric
1.2 All tanks located outside the reactor containment and containing radioactive materials in liquids should be designed to prevent uncontrolled releases of radioactive materials due to spillage (in buildings or from outdoor tanks). The following design features should be included for tanks that may contain radioactive materials:	1.2 See response to 1.2.1 through 1.2.5.
1.2.1 All tanks inside and outside the plant, including the condensate storage tanks, should have provisions to monitor liquid levels. Potential overflow conditions should actuate alarms both locally and in the control room.	1.2.1 Complies. See Table 11.2-2.
1.2.2 All tank overflows and drains and sample lines should be routed to the liquid radwaste treatment system*.	1.2.2 Complies. See Table 11.2-2.
1.2.3 Indoor tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system*.	1.2.3 Complies. See Table 11.2-2.
1.2.4 The design should include provisions to prevent leakage from entering unmonitored systems and ductwork in the area.	1.2.4 Complies. See Sections 9.4 and 11.3.
1.2.5 Outdoor tanks should have a dike or retention pond capable of preventing runoff in the event of a tank overflow and should have provisions for sampling collected liquids and routing them to the liquid radwaste treatment system.	1.2.5 Complies. See Table 11.2-2.
2. Gaseous Radwaste Systems	
2.1 The gaseous radwaste treatment system** should meet the following criteria:	2.1 See response to 2.1.1 through 2.1.3.

TABLE 3.2-5 (Sheet 4)

Regulatory Guide 1.143 Position	Union Electric
2.1.1 The systems should be designed and tested to requirements set forth in the codes and standards listed in Table 1 supplemented by the provisions noted in 2.1.2 and in regulatory position 4 of this guide.	2.1.1 Complies. See Table 3.2-2.
2.1.2 Materials for pressure-retaining components should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications. Manufacturers' material certificates of compliance with material specifications, such as those contained in the codes referenced in Table 1, may be provided in lieu of certified materials test reports.	2.1.2 Complies. Carbon steel, stainless steel, or other similar materials compatible with the chemical, physical, and radioactive environment are used for pressure-retaining components. The use of malleable, wrought, or cast iron materials or plastic pipe is not allowed. Material certificates of compliance or certified material test reports are required for the material.
2.1.3 Those portions of the gaseous radwaste treatment system that are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems, should be designed to the seismic design criteria given in regulatory position 5 of this guide. For the systems that normally operate at pressures above 1.5 atmospheres (absolute), these criteria should apply to isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these components from the rest of the system (e.g., waste gas storage tanks in the PWR) and to the building housing this equipment. For systems that operate near ambient pressure and retain gases on charcoal absorbers, these criteria should apply to the tank support elements (e.g., charcoal delay tanks in a BWR) and the building housing the tanks.	2.1.3 Complies as indicated in response to position 5. The gaseous radwaste system operates above 1.5 atmospheres.

TABLE 3.2-5 (Sheet 5)

Regulatory Guide 1.143 Position	Union Electric
3. Solid Radwaste System	
3.1 The solid radwaste system consists of slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and components and subsystems used to solidify radwastes prior to offsite shipment. The solid radwaste handling and treatment system should meet the following criteria:	3.1 See response to 3.1.1 through 3.1.4.
3.1.1 The system should be designed and tested to the requirements set forth in the codes and standards listed in Table 1 supplemented by the provisions noted in 3.1.2 and in regulatory position 4 of the guide.	3.1.1 Complies. See Table 3.2-2.
3.1.2 Materials for pressure-retaining components should conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Materials should be compatible with the chemical, physical, and radioactive environment of specific applications. Manufacturers' material certificates of compliance with material specifications, such as those contained in the codes referenced in Table 1, may be provided in lieu of certified materials test reports.	3.1.2 Complies. Carbon steel, stainless steel, or other similar materials compatible with the chemical, physical, and radioactive environment are used for pressure-retaining components. The use of malleable, wrought, or cast iron material or plastic pipe is not allowed. Material certificates of compliance or certified material test reports are required for the material
3.1.3 Foundations and adjacent walls of structures that house the solid radwaste system should be designed to the seismic criteria given in regulatory position 5 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building.	3.1.3 Complies, as described in Section 3.8.6.4.
3.1.4 Equipment and components used to collect, process, or store solid radwastes need not be designed to seismic criteria given in regulatory position 5 of this guide.	3.1.4 Complies. Contained sources are not seismically designed.

TABLE 3.2-5 (Sheet 6)

Regulatory Guide 1.143 Position	Union Electric
4. Additional Design, Construction, and Testing Criteria	
In addition to the requirements inherent in the codes and standards listed in Table 1, the following criteria, as a minimum, should be implemented for components and systems considered in this guide:	
4.1 The quality assurance provisions described in regulatory position 6 of this guide should be applied.	4.1 Complies, as described in position 6.
4.2 Process piping systems include the first root valve on sample and instrument lines. Pressure-retaining components of process systems should use welded construction to the maximum practicable extent. Flanged joints or suitable rapid disconnect fittings should be used only where maintenance or operational requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal should not be used, except for instrumentation connections where welded connections are not suitable. Process lines should not be less than 3/4 inch (nominal I.D.). Screwed connections backed up by seal welding, mechanical joints, or socket welding may be used on lines 3/4 inch or larger but less than 2-1/2 inches (nominal I.D.). For lines 2-1/2 inches and above, pipe welds should be of the butt-joint type. Non-consumable backing rings should not be used in lines carrying resins or other particulate material. All welding constituting the pressure boundary of pressure-retaining components should be performed in accordance with ASME Boiler and Pressure Vessel Code Section IX.	4.2 Complies.

TABLE 3.2-5 (Sheet 7)

Regulatory Guide 1.143 Position	Union Electric
4.3 Piping systems should be hydrostatically tested in their entirety, except at atmospheric tank connections where no isolation valves exist. Pressure testing should be performed on as large a portion of the in-place systems as practicable. Testing of piping systems should be performed in accordance with applicable ASME or ANSI codes, but in no case at less than 75 psig. The test pressure should be held for a minimum of 30 minutes with no leakage indicated.	4.3 Complies except that the requirements of ANSI B31.1 will be used in determining whether hydrostatic, pneumatic, or initial service leakage testing will be performed. Hydrostatic testing, when required, will be performed in accordance with ANSI B31.1 at 1.5 times design pressure, rather than 75 psig. Pneumatic and initial service leak tests, when required, will be performed in accordance with ANSI B31.1.
4.4 Testing provisions should be incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.	4.4 Complies. The systems are in intermittent or continuous use, which demonstrates the systems' performance and structural and leaktight integrity.
5. Seismic Design for Radwaste Management Systems and Structures Housing Radwaste Management Systems	
5.1 Gaseous Radwaste Management Systems***	5.1 See 5.1.1 through 5.1.3.

TABLE 3.2-5 (Sheet 8)

Regulatory Guide 1.143 Position	Union Electric
5.1.1 For the evaluation of the gaseous radwaste system described in regulatory position 2.1.3, a simplified seismic analysis procedure to determine seismic loads may be used. The simplified procedure consists of considering the system as a single-degree-of-freedom system and picking up a seismic response value from applicable floor response spectra, after the fundamental frequency of the system is determined. The floor response spectra should be obtained analytically (regulatory position 5.2) from the application of the Regulatory Guide 1.60 design response spectra normalized to the maximum ground acceleration for the operating basis earthquake (OBE), as established in the application, at the foundation of the building housing the gaseous radwaste system. More detailed guidance can be found in Regulatory Guide 1.122, "development of Floor Design Response spectra for Seismic Design of Floor-Supported Equipment or Components."	5.1.1 The gaseous radwaste system is seismically analyzed, considering a single degree of freedom and the floor response spectra discussed in position 5.2.
5.1.2 The allowable stresses to be used for steel system support elements should be those given in "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," adopted in February 1969. The one-third allowable stress increase provisions for combinations involving earthquake loads, indicated in Section 1.5.6 of the specification, should be included. For design of concrete structures, use of ACI 349-76 as endorsed in Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)," is acceptable.	5.1.2 Complies.

TABLE 3.2-5 (Sheet 9)

Regulatory Guide 1.143 Position	Union Electric
5.1.3 The construction and inspection requirements for the support elements should comply with those stipulated in AISC or ACI Codes as appropriate.	5.1.3 Complies.
5.2 Buildings Housing Radwaste Systems	5.2 Complies. Section 3.8.6.4 addresses the requirements of 5.2.1 through 5.2.6.
5.2.1 Input motion at the foundation of the building housing the radwaste systems should be defined. This motion should be defined by normalizing the Regulatory Guide 1.60 spectra to the maximum ground acceleration selected for the plant OBE. A simplified analysis should be performed to determine appropriate seismic loads and floor response spectra pertinent to the location of the system, i.e., an analysis of the building by a several-degrees-of-freedom mathematical model and the use of an approximate method to generate the floor response spectra for radwaste systems and the seismic loads for the buildings. No time history analysis is required.	
5.2.2 The simplified method for determining seismic loads for the building consists of (a) calculating the first several modal frequencies and participation factors for the building, (b) determining modal seismic loads using regulatory position 5.2.1 input spectra, and (c) combining modal seismic loads in one of the ways described in Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."	

TABLE 3.2-5 (Sheet 10)

Regulatory Guide 1.143 Position	Union Electric
5.2.3 With regard to generation of floor response spectra for radwaste systems, simplified methods that give approximate floor response spectra without need for performing a time history analysis may be used.	
5.2.4 The load factors and load combinations to be used for the building should be those given in ACI 349-76 as endorsed in Regulatory Guide 1.142. The allowable stresses for steel components should be those given in the AISC Manual. (See regulatory position 5.1.2)	
5.2.5 The construction and inspection requirements for the building elements should comply with those stipulated in the AISC or ACI Code, as appropriate.	
5.2.6 The foundation media of structures housing the radwaste systems should be selected and designed to prevent liquefaction from the effects of the maximum ground acceleration selected for the plant OBE.	
5.3 In lieu of the criteria and procedures defined above, optional shield structures constructed around and supporting the radwaste systems may be erected to protect the radwaste systems from effects of housing structural failure. If this option is adopted, the procedures described in regulatory position 5.2 need only be applied to the shield structures while treating the rest of the housing structures as nonseismic Category I.	5.3 The criteria and procedures of 5.2 are used.
6. Quality Assurance for Radwaste Management Systems	6. Complies, with the following clarifications:

TABLE 3.2-5 (Sheet 11)

Regulatory Guide 1.143 Position	Union Electric
Since the impact of these systems on safety is limited, a quality assurance program corresponding to the full extent of Appendix B to 10 CFR Part 50 is not required. However, to ensure that systems will perform their intended function, a quality assurance program sufficient to ensure that all design, construction, and testing provisions are met should be established and documented. The following quality assurance program is acceptable to the NRC staff. It is reprinted by permission of the American Nuclear Society from ANSI N199-1976, "Liquid Radioactive Waste Processing System for Pressurized Water Reactor Plants."	Quality assurance aspects, sufficient to ensure design, design control and design reviews, commensurate with safety considerations, are carried out in Plant design development/ modification procedures. Design, construction, and testing provisions for: • systems handling radioactive materials in liquids • gaseous radwaste systems, and • solid radwaste systems, meet the requirements set forth in the applicable Codes and Standards listed in Table 1 of Reg. Guide 1.143, "For Comments", dated July 1978. See response to Section 3.1.1.
"4.2.3 Quality Control. The design, procurement, fabrication, and construction activities shall conform to the quality control provisions of the codes and standards specified herein. In addition, or where not covered by the referenced codes and standards, the following quality control features shall be established.	4.2.3 Complies. See response to Section 6.0.

TABLE 3.2-5 (Sheet 12)

Regulatory Guide 1.143 Position	Union Electric
"4.2.3.1 System Designer and Procurer	
"(1) Design and Procurement Document Control--Design and procurement documents shall be independently verified for conformance to the requirements of this standard by individual(s) within the design organization who are not the originators of the document. Changes of these documents shall be verified or controlled to maintain conformance to this standard.	4.2.3.1(1a) Complies; see responses to Sections 1.0, 2.0, 3.0, 4.0, 5.0 and 6.0. 4.2.3.1(1b) Complies with the following clarification: Design requirements, as specified in the responses to Sections 1.0, 2.0, 3.0, 4.0, 5.0 and 6.0, are specified in procurement documents. Since the design incorporates standard commercially available equipment and materials, procurements can be non-safety related provided the manufacturer's material certificates of compliance, with material specifications, such as those required by the ASME, ANSI and ASTM Codes are obtained. (Reference applicable codes in Table 1 of Reg. Guide 1.143, "For Comments", dated July 1978)
"(2) Control of Purchased Material, Equipment and Services--Measures to ensure that suppliers of materials, equipment, and construction services are capable of supplying these items to the quality specified in the procurement documents shall be established. This may be done by an evaluation or a survey of the suppliers' products and facilities.	4.2.3.1(2) Complies, with the following clarification: Work control and installation inspections will ensure that equipment and materials conform to the original design document/ requirements through Plant implementing procedures; e.g. ensuring the equipment and materials meet the original design form, fit and function. The Callaway Inspection Program will ensure that pressure boundary parts for Group D equipment and materials conform to the original design document/requirements.

TABLE 3.2-5 (Sheet 13)

Regulatory Guide 1.143 Position	Union Electric
"(3) Instructions shall be provided in procurement documents to control the handling, storage, shipping, and preservation of material and equipment to prevent damage, deterioration, or reduction of cleanness.	4.2.3.1(3) Complies via Plant implementing procedures for handling, storage, preservation, etc.
"4.2.3.2 System Constructor	
"(1) Inspection. In addition to required code inspections, a program for inspection of activities affecting quality shall be established and executed by, or for, the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. This shall include the visual inspection of components prior to installation for conformance with procurement documents and the visual inspection of items and systems following installation, cleanness, and passivation (where applied).	4.2.3.2(1) Complies via response and clarification to Section 4.2.3.1(2) above
"(2) Inspection, Test, and Operating Status. Measures should be established to provide for the identification of items which have satisfactorily passed required inspections and tests.	4.2.3.2(2) Complies. See response and clarification to Section 4.2.3.1(2) above.
"(3) Identification and Corrective Action for Items of Nonconformance. Measures should be established to identify items of nonconformance with regard to the requirements of the procurement documents or applicable codes and standards and to identify the action taken to correct such items."	4.2.3.2(3) Complies. Items of nonconformance are addressed through the work control and installation inspection implementing procedures.
In Section 4.2.3.2(3), "items of nonconformance" should be interpreted to include failures, malfunctions, deficiencies, deviations, and defective material and equipment.	

TABLE 3.2-5 (Sheet 14)

Regulatory Guide 1.143 Position	Union Electric
Sufficient records should be maintained to furnish evidence that the measures identified above are being implemented. The records should include results of reviews and inspections and should be identifiable and retrievable.	Complies.

NOTES:

* Retention by an intermediate sump or drain tank designed for handling radioactive materials and having provisions for routing to the liquid radwaste system is acceptable.

** For a BWR, this includes the system provided for treatment of normal offgas releases from the main condenser vacuum system beginning at the point of discharge from the condenser air removal equipment; for a PWR, this includes the system provided for the treatment of gases stripped from the primary coolant.

*** For those systems that require seismic capabilities, as indicated in Regulatory Position 2.1.3.

3.3 WIND AND TORNADO LOADINGS

All standard plant seismic Category I structures which are required for safe shutdown, contain equipment required for safe shutdown, are required to protect reactor coolant system integrity, or which protect stored fuel assemblies are designed to withstand the effects of a tornado and the most severe wind phenomena encountered at any of the sites (see Section 2.3) (GDC-2). These structures are identified in Table 3.3-1.

A tabulation of systems and components and their location by room number, except for the RWST, needed for a safe shutdown and to ensure the integrity of the reactor coolant pressure boundary is provided in Table 3.11(B)-3. All of the components and systems identified in Table 3.11(B)-3, which include those requiring tornado protection, are housed within the protective structures identified in Table 3.5-2. All of those structures are designed to provide tornado protection. The protective structure requirements for the RWST are discussed in Section 6.3.2.2. Since there are no systems or components within the remaining plant structures whose failure could lead to significant offsite radiological consequences, those buildings have not been designed to provide tornado protection for systems contained therein. The structures, systems, and components identified in Appendix A to Regulatory Guide 1.117 have been provided with tornado protection.

BC-TOP-3-A (Ref. 1) defines tornado and extreme wind loadings and criteria, and furnishes data, formulae, and procedures for determining maximum wind loading on structures or parts of structures.

3.3.1 WIND LOADINGS

3.3.1.1 Design Wind Velocity

The design wind velocity for all standard plant seismic Category I structures is 100 mph at 30 feet above ground for a 100-year recurrence interval. Refer to the Site Addenda for the design wind velocity for site-related seismic Category I structures.

The bases for the wind velocity selection and supporting data and wind histories are contained in Section 2.3 of each Site Addendum and in Section 2.0 of BC-TOP-3-A. The design wind velocity envelops all of the site wind conditions.

As referenced in BC-TOP-3-A, ANSI A58.1 (Ref. 2) is used as the basis for determining the vertical velocity distribution and gust factors. The wind pressure values used are those tabulated in Section 6 of ANSI A58.1 for exposure "C," which is flat, open country. Table 5 of ANSI A58.1 is used to determine the effective velocity pressures on buildings and structures. Table 6 of ANSI A58.1 is used to determine the effective velocity pressures on parts and portions of buildings and structures. A basic wind speed of 100 mph is used, and the tables take into account the effects of vertical velocity distribution and gust factors. For site specific information, see each Site Addendum.

3.3.1.2 Determination of Applied Forces

The procedures used to translate the wind velocity into applied forces on the structures are contained in ANSI A58.1 and Sections 2.0 and 4.0 of BC-TOP-3-A. These procedures include the applicable effects of wind force distribution and shape coefficients.

For standard plant seismic Category I structures which are designed for tornado loading, the applied forces due to wind are calculated to determine if they are less severe than the applied forces due to tornado loadings. The applied tornado-force magnitude and distribution are determined, as described in Section 3.3.2.2 below.

Appropriate load combinations, stress levels, and load factors discussed in Section 3.8 are considered in determining the governing loads.

3.3.2 TORNADO LOADINGS

Tornado loadings for structural analysis are obtained in accordance with BC-TOP-3-A. Compliance with Regulatory Guide 1.76 is discussed in Appendix 3A.

3.3.2.1 Applicable Design Parameters

Tornado loads are not assumed to be coincident with any accident condition or earthquake.

Tornado characteristics are established in accordance with Table I of Regulatory Guide 1.76 for tornado intensity region I. A maximum windspeed of 360 mph, which consists of a maximum rotational speed of 290 mph at a radius of 150 feet combined with a maximum translational speed of 70 mph, is used. In order to maximize transit time of the tornado across exposed plant features, a minimum translational speed of 5 mph is used. An atmospheric pressure drop of 3.0 psi, at a linear rate of 2.0 psi per second, is also used.

Tornado-generated missiles are discussed in Section 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

The procedures used to transform the tornado loadings into an effective pressure on exposed surfaces of structures are outlined in Section 3.5 of BC-TOP-3-A. The effects of shape coefficients and pressure distribution are included in these procedures.

All standard plant seismic Category I structures are designed to prevent venting, with the exception of the main steam tunnel (Area 5 of the auxiliary building above El. 2026) and the fuel building. The main steam tunnel and the fuel building are vented to the atmosphere with the exterior walls and roofs designed to resist the full pressure differential (3.0 psi) due to the design basis tornado. The interior walls and slabs are

designed to resist the differential pressures between compartments that occur as a result of venting the structure. The methods employed to determine the differential pressures are found in Section 3.5.2 of BC-TOP-3-A.

The procedures used to transform the tornado-generated missile loadings into effective loads are discussed in Section 3.5.3.

Tornado wind velocity pressure effects, atmospheric pressure change effects, and missile impact effects are combined in accordance with Section 3.4 of BC-TOP-3-A. These combined effects constitute the total tornado effect (W_t), which is then combined with other loads as specified in Section 3.8 (see the Site Addendum).

3.3.2.3 Effect of Failure of Structure or Components not Designed for Tornado Loads

Non-Category I structures are not designed for tornado loads. Non-Category I structures adjacent to seismic Category I structures include the turbine building, RAM Storage Building, and communications corridor. The structural framing of these buildings is designed to preclude gross collapse upon safety-related structures or components under loads imposed by the design basis tornado. Other non-Category I structures are located so that their collapse would not endanger safety-related structures or components.

3.3.3 REFERENCES

1. Tornado and Extreme Wind Design Criteria for Nuclear Power Plants, Bechtel Power Corporation, BC-TOP-3-A, San Francisco, California, Revision 3, August, 1974.

2. American National Standards Institute (ANSI), Building Code Requirements or Minimum Design Loads in Buildings and Other Structures, A58.1-1972.

TABLE 3.3-1 TORNADO-RESISTANT BUILDINGS AND ENCLOSURES

Reactor building

Control building

Fuel building

Auxiliary building

Diesel generator building

Diesel fuel oil storage tank access vaults

Turbine building (for structural framing integrity only)

Communications corridor (for structural framing integrity only)

Independent Spent Fuel Storage Installation (ISFSI)

RAM storage building (for structural framing integrity only)

3.4 WATER LEVEL (FLOOD) DESIGN

The criteria used to establish the design basis flood levels comply with Regulatory Guides 1.59 and 1.102, to the extent described in Appendix 3A.

3.4.1 FLOOD PROTECTION

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

3.4.1.1.1 External Flood Protection

All standard plant seismic Category I structures and the systems they house are designed to withstand the effects of natural phenomena, such as flooding and groundwater level (GDC-2). Flood elevations, including the probable maximum flood (PMF) and the maximum groundwater elevations used in the design of standard plant seismic Category I structures for buoyancy and hydrostatic pressure, are shown in Tables 1.2-1 and 3.4-1 and are discussed in Section 2.4.

Standard plant seismic Category I structures are not protected above grade for flooding because there are no above-grade floods at the structure locations. Safety-related systems located below grade are protected from groundwater inleakage by a combination of a waterproofing system for the structures and the location of safety-related systems in watertight compartments. Refer to Section 1.2 for figures of systems below grade. In addition, an interior floor drainage system, as described in Section 9.3.3, is provided within the structures. For a description of flood protection for site-related seismic Category I structures, refer to Section 3.4 of the Site Addenda.

Although not serving a safety-related function, additional waterproofing is provided below grade by means of waterstops and waterproofing materials. Waterstops are provided at expansion and construction joints located below grade.

An auxiliary waterproofing system is installed on the vertical exterior surfaces of walls below grade of all standard plant seismic Category I structures, except the reactor building. The minimum 5-foot thickness of base mats provides adequate waterproofing of floor areas. The minimum 7-foot-thick vertical wall and internal steel liner plate provide sufficient waterproofing of the reactor cavity and instrumentation tunnel. There is no functional requirement for waterproofing of the tendon gallery.

Below grade penetrations are provided with waterproof seals to protect against postulated groundwater intrusion. Typical waterproofing details are shown in Figure 3.4-1.

Waterstop material is styrene-butadiene synthetic rubber. The auxiliary waterproofing system used consists of one of the following systems:

SYSTEM ONE - This system consists of a two-component thermosetting polyurethane bitumen applied to a minimum dry film thickness of 50 mils.

SYSTEM TWO - This system consists of a surface-applied waterproofing compound in slurry consistency or dry powder form. The waterproofing compound is comprised of chemicals, quartz, sand, and cement. Waterproofing effect is produced by the activated chemicals which penetrate, with moisture as a solution of high salinity under osmotic pressure, into the capillary tracts of the concrete. The chemicals then combine with the free lime in the concrete and form crystals which close the capillaries, tracts, and shrinkage cracks, thus keeping the moisture or water out.

SYSTEM THREE - This system consists of granular bentonite sealed inside a smooth face sheet of corrugated kraft applied as panels.

3.4.1.1.2 Internal Flooding Protection

All safety-related equipment rooms located below grade are protected from back-flooding by the remote location of waste-processing components in the radwaste building. The floor and equipment drains in standard plant seismic Category I buildings drain to sumps in the lowest level of the building in which they are located. These sumps are pumped to the floor drain tank or the waste hold-up tank located in the radwaste building. Should these tanks rupture or leak, flow into safety-related areas will not occur since the tanks are located below the radwaste building flood level.

Equipment and floor drains below the 7-foot flood level of the auxiliary building drain to sumps within the same compartment or are provided with drain caps. Several water-tight areas have been established in the auxiliary building to provide protection of all safety-related equipment. Drainage areas and protection of the safety-related equipment in this area is described in Section 9.3.3 and Figure 9.3-6.

As described in Sections 9.3.3 and 11.2, the drainage and liquid radwaste systems are designed to preclude backflow from occurring in the safety-related equipment in the auxiliary building. Appendix 3B provides an evaluation of the effect of postulated flooding generated within the plant.

3.4.1.2 Permanent Dewatering Systems

No permanent dewatering system is required.

3.4.2 ANALYSIS PROCEDURES

Natural phenomena, such as flood current, wind wave, or hurricane (tsunamis cannot occur at any site), that are associated with dynamic water forces are not applicable to the standard plant seismic Category I structures, since the grades for these structures are located above the probable maximum flood elevations. Design loadings for the

site-related seismic Category I structures which could experience dynamic water forces are described in Section 3.4.2 of the Site Addenda.

Structures as a whole and component parts are designed for the hydrostatic forces due to maximum groundwater level, in accordance with the load factors and loading combinations stated in Section 3.8.

TABLE 3.4-1 PMF, GROUNDWATER, REFERENCE, AND ACTUAL PLANT ELEVATIONS

Site	Structure	Probable Max. Flood Level ft. - msl	Design Ground water Elevation ft. - msl	Reference Plant Grade ft.	Actual Plant Grade ft. - msl
Callaway	Reactor building				
	Control building				
	Fuel building				
	Auxiliary building	559.00	840.00	1999.50	840.00
	Diesel generator building				
	ESWS pumphouses				
	Ultimate heat sink cooling towers				

3.5 MISSILE PROTECTION

Adequate protection is provided to ensure that those portions of the essential structures, systems, or components whose failure would result in the failure of the integrity of the reactor coolant system, reduce the functioning to an unacceptable level of any plant feature required for a safe shutdown, or lead to 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 recommendations of Regulatory Guides 1.13 and 1.115 as they pertain to internally and externally generated missiles are met. The response to Regulatory Guide 1.14 is discussed in Appendix 3A. Discussion of Regulatory Guide 1.27 in regard to missiles is included in Appendix 3A of the Site Addendum.

Appendix 3B provides an evaluation of the effect of postulated missiles generated within the plant. The following sections provide the bases for the selection of the missiles, protection requirements for external missiles, and details of the barrier design.

3.5.1 MISSILE SELECTION AND DESCRIPTIONS

There are four general sources from which missiles are postulated. These are:

a. Rotating component failure

b. Pressurized component failure

c. Tornadoes

d. Missiles associated with activities in the proximity of the site

The locations where the missiles may be generated are categorized as follows:

a. Internally generated missiles

b. Turbine missiles

c. Externally generated (outside the plant building) missiles during tornadoes

3.5.1.1 Internally Generated Missiles (Outside Containment)

There are two general sources of postulated missiles within the plant:

a. Rotating component failures

b. Pressurized component failure

3.5.1.1.1 Rotating Component Failure Missiles

Missiles generated by postulated failures of rotating components, their source and characteristics, and missile protection provided are discussed in Appendix 3B.

Missile selection is based on the following conditions:

a. All rotating components which are operated during normal operating plant conditions are capable of becoming missiles.

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

c. The energy of the missile is sufficient to perforate the protective housing.

3.5.1.1.2 Pressurized Component Failure Missiles

Missiles generated by postulated failures of pressurized components, their source and characteristics, and missile protection provided are discussed in Appendix 3B. The bases for selection are:

a. Pressurized components in systems whose service temperature exceeds 200°F or whose design pressure exceeds 275 psig are evaluated as to their potential for becoming a missile.

b. Temperature 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. Welded dead-end flanges are evaluated as potential missiles if the failure of a single circumferential weld could cause their ejection.

d. Valves of ANSI 900-psig rating and above, constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, are pressure seal, bonnet-type valves. For pressure seal bonnet valves, bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy.

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

e. Most valves of ANSI 600-psig rating and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material by rules set forth in the

ASME Boiler and Pressure Vessel Code, Section III, and by designing flanges in accordance with applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing a simultaneous complete severance failure is very remote. The widespread use of valves with bolted bonnets and the low historical incidence of complete severance valve bonnet failures confirm that bolted valve bonnets need not be considered as credible missiles.

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. Valves with backseats are prevented from becoming missiles by this feature. In addition, air- or motor-operated valve stems will be effectively restrained by the valve operators.

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

3.5.1.2 Internally Generated Missiles (Inside Containment)

Sources of internally generated missiles outside the containment are also applicable to the inside of the containment (see Section 3.5.1.1 for discussion).

3.5.1.3 Turbine Missiles

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external missiles will be released. These ejected missiles may impact various plant structures, including those housing safety-related equipment. The plant layout, as shown in the general arrangement drawings (Section 1.2), is a peninsular arrangement for the turbine generators. This layout minimizes the possibility of a turbine missile impacting the other plant structures and equipment essential for safe shutdown requirements. Section 10.2.3.6 describes the inspection requirements and the testing of valves which prevent turbine overspeed that would cause the missile generation.

The turbine generators for SNUPPS were manufactured by the General Electric Company (GE), and are described in Section 10.2. The high-pressure (HP) and low-pressure (LP) rotors and steam paths were replaced with new designs from Alstom.

GE's experience and calculations (Ref. 1) show that in the improbable event of a rotor fracture the substantial fragments of the high-pressure turbine and generator rotors will be contained within their respective casings. In the low-pressure (LP) turbine design, the energy stored in the hypothetical fragments of the wheels is of the same order of

magnitude as the energy-absorbing capability of the stationary parts, and a potential for generating missiles is dependent on the postulated speed of the LP turbine rotor. Alstom's calculations for the LP turbine (Ref. 2) show that fragments are conservatively assumed to perforate the turbine casing.

Studies of known failures of turbine-generator rotating elements have indicated that they may be classified into two general types:

a. Failure of rotating components at or near normal operating speed.

b. Failure of components that control the admission of steam to the turbine, resulting in excessive shaft rotational speed and consequent mechanical failure.

3.5.1.3.1 Low-Speed Missiles

The cause of failures at or near rated speed has been found to be combinations of severe strain concentrations, developed from hydrogen flaking or nonmetallic inclusions, and relatively brittle metals. Alstom's development programs have considered all aspects of this problem. In particular, careful control of alloy, chemistry, and forging heat-treating cycles has reduced the brittle to ductile transition temperatures of low-pressure wheels to well below startup temperatures. Improved mill practices in vacuum pouring and alloy addition have resulted in forgings which are much more uniform and defect-free. In addition to the above developments, more comprehensive tests involving improved techniques and laboratory investigations have substantially increased the reliability of present-day rotors and reduced the likelihood of burst failures of turbine-generator rotors operating at or near rated (normal) speed. These improved designs, better materials and properties, as well as improved quality control have combined to make the probability of catastrophic failure very small.

The first turbine generator in a nuclear plant with welded LP rotors went into service in 1965. At the end of 2004, there were 277 Alstom Power welded LP rotors in operation in nuclear power plants. To date, there have been no reports of rotor failures and no indications of stress corrosion cracking in the relevant radial-axial plane where they could extend to release a missile. The average operating hours of welded LP rotors, which have been in service for more than 3 years, is greater than 90,000 hours. The last stage rotating blade is Alstom's LP65 design which has been in service at the UL-CHIN Nuclear Power Station for more than 100,000 hours, and at San Onofre units 2 & 3 for more than 25,000 hours.

The application of this design to nuclear units has not resulted in any significant new demands or higher working stresses. The major benefit is the elimination of the "shrunk on discs" which experienced stress corrosion cracking. While stress corrosion cracks were found in some blade root fastenings at Forsmark after more than 130,000 hours of service, Alstom's SCC Threshold Stress design criteria would have predicted it. Alstom's analysis (Ref. 2) has revealed that the probability of missiles occurring at or near running

speed (120% of running speed and below) is 6 x 10^{-9} over an inspection interval of 100,000 operating hours. This probability is well below the required value of 1 x 10^{-5} for plants with an unfavorable equipment orientation.

3.5.1.3.2 High-Speed Missiles

Significant steps in mechanical design have been taken in order to prevent turbine overspeed. The turbine generators for Callaway are provided with an overspeed protection system employing electrohydraulic controls (EHC). Modern electrohydraulic controls incorporate GE's experience gained over a period of 10 years on 140 turbines using EHC.

Table 1 of Reference 1 lists turbines that have experienced bursts of rotating parts. The only turbine in that list that experienced a high-speed burst is the Uskmouth No. 5 turbine designed and built by a British manufacturer. It was equipped with a control system which, in General Electric terminology, is described as mechanical-hydraulic controls (MHC).

There have been no runaways of General Electric turbines equipped with EHC. A probability-of-failure analysis of various EHC components and its effect on the overall probability of overspeed (P_1) is included in Reference 1.

The EHC is a highly reliable system, employing three electrical and one mechanical speed inputs. Logic signals are processed in both electronic and hydraulic channels for redundance. Valve opening actuation is provided by a 1,600-psig hydraulic system which is totally independent of the bearing lubrication system. Valve closing actuation is provided by springs and aided by steam forces following the reduction or relief of hydraulic pressure. The system is designed so that loss of hydraulic fluid pressure leads to valve closing and consequent shutdown.

The main steam turbine inlet valves are provided in series arrangements: a group of stop valves actuated by either of two overspeed-trip signals, followed by a group of control valves modulated by the speed-governing system, and tripped by either overspeed-trip signal. These systems are described in Section 10.2.2.3.2.

The intermediate valves are arranged in series-pairs, with an intermediate stop valve and intercept valve in one casing. The closure of either one of the two valves will close off the corresponding steam line. Thus, a single failure of any component will not lead to destructive overspeed. A multiple failure at the instant of load loss would be required, involving combinations of undetected electronic faults and/or mechanically stuck valves and/or hydraulic fluid contamination. The probability of such joint occurrences is extremely low, due both to the inherently high reliability of the design of the components and frequent inservice testing. For further description and functioning of intercept valves, refer to Section 10.2.2.3.2.

The LP rotors would fail by general ductile yielding at about 175 percent of the normal operating speed. The attainment of this runaway speed is unlikely since a progression of failures would act to disrupt the steam path, limiting the ability to further accelerate the machine and to stop further acceleration by performing work upon the entrained debris. Such a failure is solely dependent on the failure of the control systems. This failure rate has been estimated from actual performance records (see Pages 18 and 19, Ref. 1). The lifetime probability of a missile occurring at runaway speed (127-175 percent) has been estimated to be 1.5×10^{-7}.

3.5.1.3.3 Missile Data

The hypothetical missile data for the 47-inch last-stage bucket, 1,800 rpm low-pressure turbine are given in Section 6 of Reference 2.

3.5.1.3.4 Probability of Damage

The probability of significant damage (P_4) to critical components in the plant due to turbine failure has been assessed by first determining the separate probabilities of turbine failure and missile ejection (P_1), such a missile striking a critical component or entire structure of safety significance (P_2), and significant damage occurring to the component (P_3). Then the overall probability $P_4 = P_1 \times P_2 \times P_3$.

The probabilistic rates for P_1 for turbine-generator failures, which are based upon detailed knowledge of the characteristics and properties of critical components and modeling the event as a sequence of simple events, are soundly based because they reflect pertinent material, stress, and environmental parameters and present techniques of analysis of Reference 2.

P_1 is calculated in Reference 2.

From the standpoint of reactor safety, it is necessary to consider the P_2 probability that, given a turbine-failure missile, the missile will impact a seismic Category I structure, system, or component. For this analysis, P_2 is evaluated such that only major safety-related power block structures and safety-related site structures, as defined in Section 1.2.2.1, are considered as targets.

Orientation of the seismic Category I structures with respect to the missile origin is shown in Section 3.5 of the Site Addendum.

Probability P_3, is the probability that, given a turbine failure, a missile has struck a seismic Category I structure and has perforated through.

The product of the strike and damage probabilities $P_2 \times P_3$ is conservatively assumed to be 1×10^{-2} for the unfavorable plant layout (Reference 2).

The annual probability P_4 of a turbine missile damaging a critical component must be less than 10^{-7}.

Using the above data, the probability P_1 of a missile ejection must be less than 10^{-5} per year. Reference 2 calculates P_1 as a function of time and is used to establish the frequency of turbine inspections. For an operating period of 100,000 hours, the probability P_1 is 6×10^{-9} over the entire period. This value is sufficiently low that no specific protective measures are required.

3.5.1.4 Missiles Generated by Natural Phenomena

Tornado-generated missiles were considered as the limiting natural-phenomena hazard in the design of all structures which are required for safe shutdown. The missiles considered in design are as listed in Table 3.5-1.

Vertical velocities of 70 percent of the indicated horizontal velocities are considered for all missiles, except the 1-inch-diameter steel rod which is critical for penetration and is assumed to have a vertical velocity equal to the horizontal velocity. These design basis missiles are in accordance with Standard Review Plan 3.5.1.4, Revision 1 (Draft).

3.5.1.5 Missiles Generated by Events Near the Sites

As discussed in Section 3.5.1.5 of the Site Addendum, there is no credible basis for anticipating site-proximity missiles at any of the sites.

3.5.1.6 Aircraft Hazards

As discussed in Section 3.5.1.2 of the Site Addendum, there are no aircraft hazards whose probability of occurrence is greater than 10^{-7} per year.

3.5.2 SYSTEMS TO BE PROTECTED

The sources of internal missiles which, if generated, could affect the safety of the plant are considered in Appendix 3B. A tabulation of safety-related equipment is provided in Table 3.11(B)-3.

All safety-related systems and components to be protected from tornado missiles are enclosed within protective structures which meet the requirements of Regulatory

Guide 1.117. Administrative controls for opening and closing missile shields/doors may be utilized for some locations/applications to assure missile protection is provided when required. With the exception of the RWST, all safety-related systems or components which require protection from tornado missiles are located by room number in Table 3.11(B)-3. A tabulation of protective structures, their minimum wall thickness, and concrete strength are given in Table 3.5-2. The protective structure requirements for the RWST are discussed in Section 6.3. Openings to these structures are designed to prevent the entry of the design basis missile when the result would preclude the safety functions of the enclosed system or components. Prevention of missile entry includes the use of missile doors and barriers at openings and adjacent buildings as shields in penetration areas. The missile barriers are designed utilizing the procedures given in Section 3.5.3. Further description of the seismic Category I structures is provided in Section 3.8.1 for the reactor building and Section 3.8.4 for other structures. Other safety-related structures which are site dependent, such as yard pipe, ESW structure, and cooling towers, are given in the Site Addendum.

3.5.3 BARRIER DESIGN PROCEDURES

The plant layout is based on optimizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, in the event a hazard occurs within the plant, there is a minimum effect on other systems or components required for a safe shutdown. Missile-resistant barriers and structures are designed to withstand and absorb missile-impact loads to prevent damage to the protected structures, systems, and components.

3.5.3.1 Tornado Missile Barrier Design Procedures

Tornado-resistant structures may sustain local missile damage, such as partial penetration and local cracking and/or permanent deformation, provided that structural integrity is maintained, perforation is precluded, and the contained seismic Category I systems, components, and equipment are not subjected to damage by secondary missiles, such as from concrete spalling and scabbing.

The wall and roof thicknesses provided to resist the effects of tornado-generated missiles are considered to be more than adequate. It is considered that a thickness of 24 inches for reinforced concrete with a minimum strength of 4,000 psi for the walls and (either 21 inches for the roof with minimum concrete strength of 4,000 psi or 18 inches for the roof with minimum concrete strength of 5,000 psi) roof slabs of seismic Category I structures are adequate to resist the impact of tornado-generated missiles for both penetration and structural response. This is based on the results of the test program, "Missile Impact Testing of Reinforced Concrete Panels," conducted by Calspan Corporation for Bechtel Corporation and reported in Calspan Report No. HC-5609-D-1, January 1975 (Ref. 3) and on the EPRI Report, "Full-Scale Tornado Missile Impact," July 1977 (Ref. 4).

3.5.3.2 Barrier Design Procedures for Internally Generated Missiles

In general, when separation is not feasible, additional protection from internal missiles is provided by barriers. The procedures and calculations employed in the design of missile-resistant barriers for turbine missiles and other internally generated missiles are described in Reference 5. In the design calculations for missile-resistant barriers, ductility ratios never were greater than 10. Therefore additional details are not required here. Appendix 3B discusses the protection required for internally generated missiles.

3.5.4 REFERENCES

1. Hypothetical Turbine Missiles Probability of Occurrence, General Electric Memo Report Dated March 14, 1973.

2. Callaway Plant Unit 1 - 1800 RPM: Missile Analysis, Alstom Power calculation STD0002445, Revision A, January 2005.

3. "Missile Impact Testing of Reinforced Concrete Panels," Calspan Report No. HC-5609-D-1, Calspan Corporation, Buffalo, New York, January 1975.

4. Stephenson, A. E., "Full-Scale Tornado Missile Impact," EPRI Report No. NP-440, July 1977.

5. "Design of Structures for Missile Impact," BC-TOP-9-A, Revision 2, Bechtel Power Corporation, San Francisco, California, September 1974.

TABLE 3.5-1 CHARACTERISTICS OF POSTULATED TORNADO MISSILES

Missile	Weight, lbs	Horizontal Velocity, fps
Wood plank, 4" x 12" x 12' long	115	272
Steel pipe, 6" diameter, schedule 40, 15' long	286	170
Steel rod, 1" diameter, 3' long	9	167
Utility pole, 13.5" diameter, 35' long	1,123	180
Steel pipe, 12" diameter, schedule 40, 15' long	749	154
Automobile, 16.4' x 6.6' x 4.3'	3,991	194

TABLE 3.5-2 STRUCTURES PROVIDING TORNADO MISSILE BARRIER PROTECTION

Structure	Nominal Concrete Thickness	90-Day Strength
Reactor building	4 ft - wall	4,000 psi
	3 ft - dome	4,000 psi
Auxiliary building	2 ft - wall	4,000 psi
	1 1/2 ft - roof	5,000 psi
Control building	2 ft - wall	4,000 psi
	1 1/2 ft - roof	5,000 psi
Diesel generator building	2 ft - wall	4,000 psi
	1 1/2 ft - roof	5,000 psi
Fuel building	2 ft - wall	4,000 psi
	1 1/2 ft - roof	5,000 psi
ISFSI	2 1/2 ft - Top Pad	4,500 psi
	1 3/4 ft - Closure Lid	4,000 psi

TABLE 3.5-3 DELETED

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

Pipe failure protection is provided in accordance with the requirements of 10 CFR 50, Appendix A, GDC 4.

Postulated breaks in the reactor coolant loop (RCL) have been eliminated based on Reference 18, which was reviewed and approved by NRC as discussed in Sections 6.2.1.2.1 and 6.2.1.2.3 item b.1. Subsequent to the GDC 4 final rule change (Reference 23), postulated breaks in two sets of branch lines connected to the RCL (accumulator lines and residual heat removal (RHR) lines) were eliminated by application of leak-before-break (LBB) technology as presented in References 19 and 20. Approval of the elimination of breaks in these branch lines is given in the NRC Safety Evaluation Report for Callaway Amendment No. 161 dated April 12, 2004 (Reference 21). Reference 22 documents the primary loop LBB analysis results after elimination of the steam generator hydraulic snubbers.

In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided to ensure that those portions of the essential structures, systems, or components whose failure could compromise the integrity of the reactor coolant system or reduce the functioning of any plant feature required for a safe shutdown to an unacceptable level are designed, constructed, and protected so as not to fail or cause such a failure.

Appendix 3B, Hazards Analysis, provides several examples of the evaluations made of the effects of postulated pipe failures within the plant. The following sections provide the bases for selection of the pipe failures, the determination of the resultant effects, and details of the protection requirements.

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS INSIDE AND OUTSIDE CONTAINMENT

Table 3.6-1 provides a matrix which indicates high-energy systems, moderate-energy systems, and safety-related systems.

Selection of pipe failure locations for evaluation of the consequences on nearby essential systems, components, and structures, is presented in Section 3.6.2 and, except for the reactor coolant loop, is in accordance with Regulatory Guide 1.46, and NRC BTPs ASB 3-1 and MEB 3-1. For the reactor coolant loop, Reference 1 provides the bases for the selection of pipe breaks.

Reference 1 provides the original criteria for postulating breaks in the reactor coolant loop. Subsequent elimination of postulated pipe breaks in the RCL and Class 1 branch lines is discussed above.

3.6.1.1 Design Bases

The following design bases relate to the evaluation of the effects of the pipe failures determined in Section 3.6.2.

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

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

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

b. Except for the reactor coolant system, the worst case operational plant conditions (including startup, operation at power, hot standby, shutdown, and upset conditions) are used to determine the piping system support/ restraint requirements and to determine blowdown rates and jet impingement loads. For the reactor coolant system, including all Class 1 branch piping, the normal power operation conditions are used as described in Reference 1.

c. Moderate-energy pipe cracks were evaluated for wetting from spray, flooding, and other environmental effects.

d. Each longitudinal or circumferential break in high-energy fluid system piping or leakage crack in moderate-energy fluid system piping was considered separately as a single postulated initial event occurring during normal plant conditions.

e. Offsite power was assumed to be unavailable if a trip of the turbine-generator system or reactor protection system was a direct consequence of the postulated piping failure, unless it was more conservative to assume that offsite power was available (e.g., a feedwater line break with offsite power available leads to a larger inventory of water for flooding considerations).

f. A single active component failure was assumed in systems used to mitigate the consequences of the postulated piping failure and to safely

shut down the reactor, except as noted in Paragraph g below. The single active component failure was 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 and results in damage to one of two or more redundant or diverse safety-related trains, single failures of components in other trains (and associated supporting trains) are not assumed. Postulated failures are precluded, by design, from affecting the opposite train or from resulting in a DBA. The safety-related systems are designed to the following criteria: a) seismic Category I standards, b) powered from both offsite and onsite sources, and c) constructed, operated, and inspected to quality assurance, testing, and in-service inspection standards appropriate for nuclear safety systems.

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

   1. In determining the availability of the systems, account was taken of the postulated failure and its direct consequences, such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions was 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 power or hot standby was assumed as allowed by the plant technical specifications. During this period plant personnel will assess the situation and make repairs.

   2. The use of nonseismic Category I equipment is clarified in the following paragraphs:

      (a) For nonseismic Category I piping failures, it was assumed that a safe shutdown earthquake could be the cause of the failure. Thus, only seismic Category I equipment could be used to bring the plant to a safe shutdown.

      (b) For seismic Category I and seismically supported nonseismic Category I piping failures, it was assumed that the failure was caused by some mechanism other than an earthquake. Thus, nonseismic Category I equipment could be used to bring the plant to a safe shutdown, subject to the power being available to operate such equipment as discussed in Paragraph h(1) above.

i. A whipping pipe was not considered capable of rupturing impacted pipes of equal or greater nominal pipe diameter and equal or greater thickness, assuming that only "piping" was determined to do the impacting. A whipping pipe was considered capable of developing a throughwall leakage crack in a pipe of larger nominal pipe size with thinner wall, assuming that only "piping" was determined to do the impacting. Where the potential existed for valves or other components in the whipping pipe to impact the targets, the above criterion was not utilized and the whipping pipe was not allowed to impact a safety-related component.

j. Pipe whip was 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 was considered to form a plastic hinge and rotate about the nearest rigid restraint, anchor, or wall penetration. 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 was assumed that the pipe comes to rest against that surface, with no pipe whip in other directions.

   If unrestrained, a whipping pipe without a constant energy source (i.e., a break at a closed valve with only one side subject to pressure) was not considered capable of forming a plastic hinge and rotating, provided that its movement could be defined and evaluated.

   Pipe whip restraints are provided wherever postulated pipe breaks have any possibility of affecting any system or component required for the mitigation of that break or safe shutdown of the plant. Unrestrained pipe breaks are limited to those areas of the plant that are physically separated from the systems and components required for pipe break mitigation or safe shutdown.

k. Deleted

l. 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.

m. Initial pipe break events were not assumed to occur in pump and valve bodies because of their greater wall thicknesses.

n. A survey of all potential internal flooding sources was performed for all rooms with safety-related components. This survey determined the worst case internal flooding event for each room. From this survey, calculations were performed to determine the worst case flood level in each of these

rooms. A summary of these flood levels is provided in Table 3.6-6. Additional information on containment flooding is provided in Sections 6.2.2.1.3 and 6.3.2.2. Assumptions used in arriving at the worst case flooding event are as follows:

1. One break or crack occurs at a time

2. Nonseismic lines will experience guillotine breaks during seismic events

3. Drain pipes are assumed to be dry before the break or crack

4. Rooms drain through the floor drain(s). No credit is taken for drainage through uncapped or unsealed equipment drains. Typically, no credit is taken for drainage out under doors.

5. Pipes which are supported II/I and are moderate energy during normal plant operating modes are assumed to develop moderate energy cracks only.

3.6.1.2 Description

Systems, components, and equipment required to safely shut down the plant and mitigate the consequences of postulated piping failures (hereinafter called essential) were reviewed, in order to comply with the design bases, to determine their susceptibility to the failure effects. The break and crack locations were determined in accordance with Section 3.6.2. Figure 3.6-1 and 3.6-3 show the high-energy pipe break locations and break types.

Those essential systems which are subject to the consequences of pipe failure are summarized in Table 3.6-1. The type of hazard (i.e., whipping, jet impingement, spraying, and flooding) is shown. This summary was based on the detailed failure mode analysis discussed in Section 3.6.1.3, Section 3.6.2.5, and Appendix 3B.

The design comparison to Regulatory Guide 1.46 positions, incorporating the comparison to NRC BTP MEB 3-1 and NRC BTP ASB 3-1, is provided in Table 3.6-2.

Pressure response analyses were performed for the subcompartments containing high-energy piping. For a detailed discussion of the line breaks selected, and pressure results, refer to Section 6.2.1.2 and Table 3.6-4 for subcompartments inside the containment and Table 3.6-4 for subcompartments located outside the containment. The methodology used for the pressure response analysis is either done in accordance with BN-TOP-4 (Reference 12) or by use of the GOTHIC 7.2b computer code (Reference 25).

Appendix 3B discusses hazards analysis and Table 3B-1 shows a typical hazards analysis.

There are no high-energy lines in the control building; therefore, there are no effects upon the habitability of the control room from pipe break or pipe whip. Further discussion of the control room habitability systems is provided in Section 6.4.

3.6.1.3 Safety Evaluation

3.6.1.3.1 General

An analysis of postulated pipe failures was performed to identify those safety-related systems, components, and equipment that provide protective actions required to mitigate the consequences of the failure.

By means of protective measures such as separation, barriers, and pipe whip restraints, discussed below, the effects of breaks and cracks are prevented from damaging essential items to an extent that would impair their design function or necessary component operability.

Typical measures used for protecting the essential systems, components, and equipment are outlined below and discussed in detail in Section 3.6.2.4. The ability of specific safety-related systems to withstand a single active failure concurrent with the postulated event is discussed, as applicable.

When the results of the pipe failure effects analysis show that the effects of a postulated high-energy break or moderate-energy crack, on a reasonable basis, are isolated, physically remote, or restrained by protective measures, from essential systems or components, no further dynamic hazards analysis is performed.

3.6.1.3.2 Protection Mechanisms

3.6.1.3.2.1 General

The plant layout arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, in the event a pipe failure occurs within the plant, there is a minimal effect on other essential systems or components which are required for safe shutdown of the plant or to mitigate the consequences of the failure.

The effects associated with a particular high-energy break or moderate-energy crack must be mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the specific measures for protection against actual pipe movement and other associated consequences of postulated failures.

Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, barriers, equipment shields, and physical separation of piping, equipment, and

instrumentation. The precise method chosen depends largely upon considerations such as accessibility, maintenance, and proximity to other pipes.

SEPARATION - The plant arrangement provides separation, to the extent practicable, between redundant safety systems (including their auxiliaries and support systems) in order to prevent loss of safety function as a result of hazards different from those for which the system is required to function, as well as for the specific event for which the system is required to be functional. Separation between redundant safety systems, with their related auxiliary supporting features, therefore, was the basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

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

a. Safety-related systems were located away from high-energy piping, where practicable.

b. Redundant (e.g., "A" and "B" trains) safety systems were located in separate compartments.

c. As necessary, specific components were enclosed to retain the redundancy required for those systems that must function as a consequence of specific piping failure.

d. Drainage systems were reviewed to assure their adequacy for flooding prevention.

BARRIERS, SHIELDS, and ENCLOSURES - Protection requirements were met through the protection afforded by the walls, floors, columns, abutments, and foundations, in many cases. Where adequate protection did not already exist due to separation, additional barriers, deflectors, or shields were provided to meet the functional protection requirements.

Some of the barriers utilized for protection against pipe whip inside the containment are the following: The secondary shield wall serves as a barrier between the reactor coolant loops and the containment liner. In addition, the refueling cavity walls, operating floor, and secondary shield walls minimize the possibility of an accident, which may occur in any one reactor coolant loop from affecting another reactor coolant loop or the containment liner. That portion of the steam and feedwater lines located within the containment was routed behind barriers which separate these lines from all reactor coolant piping. The barriers described above will withstand loadings caused by jet forces and pipe whip impact forces.

Further discussion of barriers and shields is provided in Section 3.6.2.4.

PIPING RESTRAINT PROTECTION - Measures for protection against pipe whip, as a result of high-energy pipe breaks, were provided where, following a single break, the unrestrained pipe movement of either end of the ruptured pipe could damage, to an unacceptable level, any structure, system, or component required to place the plant in a safe shutdown condition or mitigate the consequences of the rupture.

The design criteria for and description of restraints are given in Section 3.6.2.3.

3.6.1.3.3 Specific Protection Considerations

a. Nonessential systems and system components are not required for the safe shutdown of the reactor, nor are they required for the limitation of the offsite release in the event of a pipe rupture. However, while none of this equipment is needed during or following a pipe break event, pipe whip protection is considered where a high-energy nonessential system or component failure could initiate a pipe break event in an essential system or component, or another nonessential system, whose failure could affect an essential system.

b. High-energy containment penetrations are subject to special protection mechanisms. As shown in Figure 3.6-1, isolation restraints are located as close as practical to the containment isolation valves associated with these penetrations. These restraints are provided in order to maintain the operability of the isolation valves and the integrity of the penetration due to a break either upstream or downstream of the penetration and outside the respective isolation restraints.

c. Instrumentation which is required to function following a pipe rupture is protected.

d. High-energy fluid system piping restraints and protective measures are designed so that a postulated break in one piping system cannot, in turn, lead to a rupture of other nearby piping system or components, if the secondary rupture will result in consequences that would be considered unacceptable for the initial postulated break.

e. For any postulated LOCA, the structural integrity of the containment structure is maintained.

f. The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture will not preclude:

    1. Subsequent access to any areas, as required, to cope with the postulated pipe rupture

    2. Habitability of the control room

3. The ability of essential instrumentation, electric power supplies, components, and controls to perform their safety function to the extent necessary to mitigate the consequences of the pipe rupture and achieve and maintain safe shutdown

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

This section describes: the design bases for locating postulated breaks/cracks in high-energy/moderate-energy piping 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; restraint design; and protective assembly design.

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

Except for the reactor coolant loop, accumulator injection lines, and RHR hot leg suction lines, NRC Branch Technical Position (BTP) MEB 3-1 was used as the basis of the criteria for the postulation of high-energy pipe breaks. Specific moderate-energy pipe crack locations were not ascertained and, therefore, they were assumed to occur at any location, except as noted in Section 3.6.2.1.2.4. Full structural weld overlays (FSWOL) have been implemented at some break locations. An FSWOL is a mitigation method for weld locations that are susceptible to cracking, such as Alloy 82/182 welds. FSWOLs will not be assumed to change any postulated break locations.

A postulated 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 development of a sudden longitudinal, uncontrolled crack (i.e., a longitudinal split) and is postulated for a high-energy fluid system only. For moderate-energy fluid systems, pipe failures are confined to postulation of controlled cracks in piping. These cracks affect the surrounding environmental conditions only, and do not result in 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 Section 3.6.2.1.1e, breaks were postulated only in high-energy piping at the following locations:

a. ASME B&PV Code, Section III - Class 1 Piping

    1. In the pressurizer surge line, there are a limited number of locations which are more susceptible to failure by virtue of stress or fatigue than the remainder of the system.

Breaks are eliminated from RCS primary loops and the accumulator and RHR lines. The elimination of these breaks is the result of the application of leak-before-break (LBB) technology (References 18, 19, 20, and 22) allowed by the revised GDC-4 (Reference 23).

The discrete break locations and orientations in the surge line are derived on the basis of stress and fatigue analysis.

The postulated break locations for the pressurizer surge line were determined with the use of a detailed ASME Code NB-3200 piping analysis together with the MEB 3-1 Rev 2, June 1987 break criteria (see Reference 15). The surge line intermediate break locations were deleted (see Reference 16).

The original design basis criteria for the reactor coolant loop (Reference 1) posutlated eleven pipe break locations. Ten of these pipe break locations have subsequently been eliminated from the Callaway structural design basis as a result of the application of LBB technology. The detailed fracture mechanics techniques used in this evaluation are discussed in References 18, 19, 20, and 22. Application of LBB allows the elimination of the dynamic effects of pipe rupture for these ten locations. To provide the high margins of safety required by GDC-4, the non-mechanistic pipe rupture design basis is maintained for containment design and ECCS analyses, and the postulated pipe ruptures are retained for electrical equipment environmental qualification as required by 10 CFR 50.49.

2. Pipe breaks are postulated to occur in the following locations in Class 1 piping runs or branch runs outside the primary reactor coolant loops and pressurizer surge line as follows:

   (a) The terminal ends of the piping or branch run.

   (b) Any intermediate locations between the terminal ends where stresses, calculated using equations (12) and (13) of the ASME B&PV Code, Section III, Subsection NB, exceed 2.4 Sm, where Sm is the design stress intensity, as given in the ASME B&PV Code, and the stress range calculated, using equation (10) of the ASME B&PV Code, exceeds 2.4 Sm.

   (c) Any intermediate locations between terminal ends where the cumulative usage factor, derived from the piping fatigue analysis, under the loadings associated with the OBE and operational plant conditions, exceeds 0.1.

(d) If the stresses and usage factor do not exceed the limits in (b) and (c), intermediate breaks are postulated at points of maximum stresses calculated by using Equation 10 of subarticle NB-3653, ASME B&PV Code, Section III.

b. ASME B&PV Code, Section III - Class 2 and 3 Piping Within Protective Structures

1. Breaks are postulated to occur at terminal ends, including:

(a) Piping-pressure vessel or equipment nozzle intersection

(b) High-energy/moderate-energy boundary

(c) Pipe to anchor intersection

(d) A branch intersection point was not considered a terminal end if: 1) the branch and the main piping systems were modeled in the same static, dynamic, and thermal analyses, 2) the intersection is not rigidly constrained to the building structure, or 3) 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).

2. At intermediate locations between terminal ends where the maximum stress ranges, as calculated by the sum of equations (9) and (10) in Subarticle NC-3652 of the ASME B&PV Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) including an OBE event, exceed 0.8 (1.2 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&PV Code, Section III.

3.(a) In the original analysis of the Class 2 and 3 piping at the Callaway Plant, in the piping systems where the stresses were lower than the limits in 2. above, a minimum of two intermediate break locations were postulated solely on the basis of the highest calculated stress levels. This location may be a pipe to valve weld, pipe to fitting weld, or near clamped support attachment point. Where the piping consisted of a straight run and was shorter than 10 pipe diameters in

length with no fittings, welded attachments, or valves, a minimum of one location was chosen based on the highest stress.

(b) However, Branch Technical Position MEB 3-1, revision 2 issued in 1987 no longer mentions arbitrary intermediate pipe ruptures as described in 3(a) above. Piping stress analyses performed subsequent to the issuance of MEB 3-1 in 1987, do not require arbitrary intermediate break locations if the stresses were lower than the limits in 2. above.

c. ASME B&PV Code, Section III - Class 2 and 3 Piping Not Enclosed Within Protective Structures No Class 2 or 3 high-energy piping is located outside the protective structures.

d. Non-Nuclear Piping (i.e., not ASME Section III Class 1, 2, or 3)

Breaks in non-nuclear piping were postulated at terminal ends of the run and at the following locations:*

1. At all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves), or

2. Breaks are postulated to occur at the locations specified for ASME Section III, Class 2 and 3 piping if the non-nuclear piping is analyzed and supported to withstand SSE loadings.

Leakage cracks in nonseismic Category I piping are postulated in worse case locations.

e. High-Energy Piping in Containment Penetration Areas

The portion of the containment penetration area piping defined above, extending from the outside of the inboard isolation restraint to the outside of the outboard isolation restraint, shall be considered and hereafter referred to as the "no break zone" (NBZ).

"No Break Zone" boundaries are shown on Figure 3.6-1.

---

* With one clarification: On approximately 2.67 feet of pipe on FB-081-HBD-2” and 0.5 feet of pipe on FB-093-HBD-3” between the 8-inch auxiliary steam header and the closed high-energy/moderate-energy boundary valves on these lines, breaks were not postulated. It was judged that the runs were short enough to prevent guillotine breaks and that any breaks that did occur would be in the 8-inch auxiliary steam header. Breaks in the 8-inch header were postulated and evaluated in the vicinity of the connections for lines 081 and 093.

Breaks were not postulated in this area because stresses did not exceed those specified in Section 3.6.2.1.1.b.

The maximum stress in the "no break zone," except within the isolation restraints, did not exceed 1.8 S_h per equation (9), Subarticle NC-3652 of ASME Section III when subjected to the combined loadings of internal pressure, deadweight, and postulated pipe break beyond the "no break zone." The maximum stress within the isolation restraints in the "no break zone" is limited such that no plastic hinge will form in this region.

The number of circumferential and longitudinal piping welds and branch connections was minimized. Welded attachments for pipe supports or other purposes to these portions of piping were avoided, except where detailed stress analyses could be performed to demonstrate compliance with the limits of Section 3.6.2.1.1.

When required for isolation valve operability, structural integrity, or the containment integrity, whip restraints capable of resisting torsional and bending moments produced by a postulated pipe break either upstream or downstream of the "no break zone" were located reasonably close to the isolation valves or penetration.

These restraints do not prevent the access required to conduct inservice inspection of the welds within the restraints specified in Section XI of the ASME Code. Inservice examinations completed during each inspection interval are performed in accordance with the Risk-Informed Break Exclusion Region (RI-BER) Augmented Inservice Inspection program. See Section 6.6 for further discussion of inservice inspection.

Terminal end breaks were not postulated on the main steam, main feedwater, and steam generator blowdown piping at the flued heads inside the containment. The "no break zone" is considered to extend up to and including the pipe to flued head weld inside containment, therefore, the terminal end location falls within the "no break zone" boundary. Inservice examinations, described in Section 6.6, commensurate with the "no break zone" will be performed on the main steam, feedwater, and steam generator blowdown piping inside the containment up to the nearest pipe whip restraint. For postulated breaks beyond the first whip restraint, the stress limit (1.8 S_h) given in Section 3.6.2.1.1.e may be exceeded for the portion of piping from the first pipe whip restraint up to and including the pipe to flued head weld; however, the integrity for this portion of piping is verified.

The restraints outside the containment on the main steam, main feedwater, and steam generator blowdown lines were located as close as possible to the containment to accommodate the design for the auxiliary building

steam tunnel and minimize stresses. The length of the steam tunnel, the location of 5-way restraints in the north wall of the auxiliary building, and the location of isolation restraints just below the floor penetrations, for connecting piping routed to other areas of the auxiliary building, resulted in low stresses considering:

1. Seismic differential building movements

2. Space requirements for safety valves, isolation valves, flued heads, and other piping and components

3. Minimum space for maintenance

4. Maximum accessibility for inservice inspections performed every inspection interval

3.6.2.1.2 Types of Breaks/Cracks Postulated

3.6.2.1.2.1 ASME Section III - Class 1 Reactor Coolant Loop Piping - High-Energy

The types of breaks postulated in the ASME Section III, Class 1 primary reactor coolant loop are discussed in Reference 1.

3.6.2.1.2.2 ASME Section III Piping Other Than Reactor Coolant Loop Piping - High-Energy

The following types of breaks were postulated to occur at the locations determined, in accordance with Section 3.6.2.1.1.

a. Breaks were not postulated in piping where nominal diameter is 1 inch or less.

b. At terminal ends, only circumferential breaks were postulated.

c. At intermediate locations where both the stress and usage factors were less than the limits of Section 3.6.2.1.1, only circumferential breaks were postulated.

d. At intermediate locations where the stress and/or usage factor exceeded the limits of Section 3.6.2.1.1, only circumferential breaks were postulated in piping less than 4-inch nominal pipe diameter but greater than the size exemption stated in a. above. In piping 4 inches and larger, circumferential and longitudinal breaks were postulated. However, if the longitudinal stress was 1.5 times greater than the circumferential stress, only circumferential breaks were postulated. Similarly, if the circumferential

stress was 1.5 times greater than the longitudinal stress, only longitudinal breaks were postulated.

3.6.2.1.2.3 Non-Nuclear Piping - High-Energy

For non-nuclear piping, the following combination of breaks was evaluated:

a. Circumferential breaks in piping larger than 1 inch

b. Longitudinal breaks in piping 4 inches and larger, except at terminal ends.

3.6.2.1.2.4 ASME Section III and Non-Nuclear Piping - Moderate - Energy

Through-wall leakage cracks were postulated in moderate-energy piping larger than 1 inch located within, or outside and adjacent to, protective structures, except as noted in the following:

a. Through-wall leakage cracks were not postulated in those portions of piping between containment isolation valves, since this piping meets the requirements of ASME Code, Section III, Subarticle NE-1120 and is designed so that the maximum stress range does not exceed 0.4 $(1.2\ S_h + S_A)$.

b. Through-wall leakage cracks were not postulated in moderate-energy fluid system piping located in the same area in which a break in high-energy fluid system piping was postulated, provided that such cracks would not result in more limiting environmental conditions than the high-energy pipe break.

c. Through-wall leakage cracks were not postulated in ASME Code, Section III, Class 2 or 3 piping and stress analyzed non-nuclear seismic Category I class piping, provided that the maximum stress range in the piping, as calculated by the sum of EQN(9) and EQN(10) in Subarticle NC-3652 of the ASME Code, Section III, considering normal and upset plant conditions, is less than 0.4 $(1.2\ S_h + S_A)$.

d. Cracks were not postulated when a review of the piping layout and plant arrangement drawings showed that the effects of through-wall leakage cracks at any location in the piping designed to seismic or nonseismic standards were isolated or physically remote from structures, systems, and components required for safe shutdown.

e. Through-wall leakage cracks were not postulated in safety-related, Class 3, high density polyethylene (HDPE) piping provided that the maximum stress range in the piping, as calculated by the sum of the Service Level B Longitudinal Stress Equation and the Alternate Thermal Expansion or

Contraction Evaluation in Reference 26, is less than 0.4 (1.2S + 1100). The Service Level B Longitudinal Stress Equation and Alternate Thermal Expansion and Contraction Evaluation in Reference 26 are equivalent to the EQN(9) and EQN(10) stresses per ASME Section III, Subarticle NC-3652, considering normal and upset conditions, respectively.

Cracks were postulated to occur individually at locations that resulted in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions. Flooding effects were determined on the basis of a conservatively estimated time period required to effect corrective actions. Further discussion of flooding effects is provided in Appendix 3B.

3.6.2.1.3 Break/Crack Configuration

3.6.2.1.3.1 High-Energy Break Configuration

The ends of a circumferentially ruptured pipe were assumed to be displaced laterally by a distance equal to or greater than one pipe diameter until and unless one end was restrained in the lateral direction.

Movement was assumed to be in the direction of the jet reaction initially, and total path controlled by the piping geometry.

The orientation of a longitudinal break, except when otherwise justified by a detailed stress analysis, was considered to cause piping movement normal to the plane of the piping system. The flow area of such a break was equal to the cross-sectional flow area of the pipe. Longitudinal breaks were assumed to be oriented (but not concurrently) at two diametrically opposed points on the piping circumference. Longitudinal and circumferential breaks were not postulated concurrently.

3.6.2.1.3.2 Moderate-Energy Crack Configuration

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

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1 Forcing Functions for Pipe Whip and Jet Impingement

To determine the forcing function, the fluid conditions at the upstream source and at the break exit will dictate the analytical approach and approximations that are used. For most applications, one of the following situations will exist:

a. Superheated or saturated steam

b. Saturated or subcooled water

c. Cold water (non-flashing)

The following three sections describe simplified models that take into account the fluid conditions. Where more complex analysis is warranted, such as for the main steam line, a RELAP4 analysis can be performed, as described in Section 3.6.2.2.1.4. For a discussion of the jet thrust forcing functions from reactor coolant loop breaks, see Section 3.6.2.2.1.5.

3.6.2.2.1.1 Superheated or Saturated Steam Break Analysis

For superheated or saturated steam, steady state thrust forces are calculated from the ideal gas relationship. This relationship has been calculated using Fanno lines, assuming homogeneous flow for superheated steam, in Reference 5, Figure 2-1. When the fluid expands into the wet region, it is treated as having a specific heat ratio of 1.1. Whether the specific heat ratio is 1.1 or 1.3, the values of Figure 2-1 of Reference 5 are used.

The initial value of the thrust is P_oA_e, where P_o is the source pressure in psia and A_e is the exit area in square inches. If the steady state thrust at initial source conditions is higher than P_oA_e, no transient time is calculated, and the steady state thrust is assumed for the entire time frame. Where significant friction results in steady state thrusts below P_oA_e, P_oA_e is applied for the initial transient, and the steady state thrust is applied for the remainder of the time frame.

The unsteady state forces due to time-dependent wave and blowdown force during the initial stages persist for several wave propagations. From Reference 8, time is approximated as time to empty the initial contents of the piping at an average flowrate. For choked flow:

$$t_{ss} = \frac{2\rho_o A_e L}{144(W_i + W_f)} = \frac{2\rho_o L}{144\left[\frac{W_i}{A_e} + \frac{W_f}{A_e}\right]} = \frac{2\rho_o L}{G_i + G_f}$$

$$G_i = 144\left[\frac{W_i}{A_e}\right] = C_o\rho_o\left[\frac{2}{k+1}\right]^{\left[\frac{k+1}{k+1}\right]};\ G_f = 144\left[\frac{W_f}{A_e}\right] = \left[\frac{G_f}{G_{max}}\right]\cdot G_{max}$$

Where:

t_{ss} = time to reach steady state, sec

W_i = initial flowrate, lbm/sec

W_f = final flowrate, lbm/sec

A_e = break area, square inches

L = length of pipe from break to source, ft

G_i = initial mass flowrate per square foot, lbm/sec-ft^2

G_f = final mass flowrate per square foot, lbm/sec-ft^2

ρ_o = source density, lbm/ft^3

C_o = source sonic velocity, ft/sec

k = effective specific heat ratio

G_{max} = maximum mass flowrate per square foot, lb/sec-ft^2

For jet impingement forces, the Moody expansion model is coupled with the Reference 5 steady state thrust to determine jet pressure.

For pressure/temperature (P/T) analysis, the blowdown rate is based on steady state flow and is determined from Figure 14 of the ASME steam tables (Ref. 9), or calculated using the perfect gas law.

This analysis method is based on a converging nozzle at the entrance to the pipe. If a flow restriction is included, it is assumed that a shock wave exists immediately downstream, and the resultant force will be lower than as calculated above (see Figure 2-2 of Reference 5).

3.6.2.2.1.2 Saturated or Subcooled Water Break Analysis

For subcooled or saturated water, steady state thrust forces are calculated, using the Henry/Fauske model for frictionless flow. As with steam, the initial value of the thrust is P_oA_e. However, since frictionless flow is used, the steady state thrust always exceeds P_oA_e and the steady state thrust is applied for the entire time frame, except where upstream restrictions are present as noted in 3.6.1.1.1.

For jet impingement forces, the Moody expansion model defined in Reference 5 is coupled with the steady state thrust to determine jet pressure.

3.6.2.2.1.3 Cold Water Break Analysis

For cold water, steady state thrust is calculated, using Reference 5, Equation 7, coupled with the frictional effects, as demonstrated below:

$$\frac{F}{P_o A_e} = \frac{2 - 2(P_a / P_o)}{1 + f(L / D)}$$

Where:

F	=	steady state thrust, lbf
P_o	=	source pressure, psia
A_e	=	break area, in^2
P_a	=	ambient pressure, psia
f	=	Darcy's friction factor
L/D	=	equivalent length of a resistance in pipe diameters

The initial value of the thrust is P_oA_e. If the steady state thrust at initial source conditions is higher than P_oA_e, no transient time is calculated, and the steady state thrust is assumed for the entire time frame. Where significant friction results in steady state thrusts below P_oA_e, P_oA_e is applied for the initial transient, and the steady thrust is applied for the remainder of the time frame.

The unsteady state forces due to time-dependent wave and blowdown forces during the initial stages persist for several wave propagations. From Reference 8, time is approximated as:

$$t_{ss} = \frac{L}{C_o} \cdot \frac{1}{2} \cdot \frac{1}{V_i / V_{ss}} \cdot \ln\left[\frac{199(1 - V_i / V_{ss})}{1 + V_i / V_{ss}}\right]$$

$$V_i = (144)(32.2)(P_o - P_a)\frac{\gamma_o}{C_o}; \; V_{ss} = \sqrt{\frac{(P_o - P_a)(\gamma_o)(2)(32.2)(144)}{1 + f(L / D)}}$$

Where:

t_{ss}	=	time to reach steady state, sec
L	=	length of pipe from break to source or upstream restriction, ft
γ_o	=	specific volume, ft^3/lbm
V_i	=	initial velocity, ft/sec
V_{ss}	=	steady state velocity, ft/sec

Jet impingement forces are calculated using the relations of Section 2.3 of Reference 5, assuming a 10 degree expansion throughout the entire jet expansion.

For flooding analysis, the blowdown rate is based on the extended Bernoulli Theorem. Derivations of this theorem satisfy the principle methodology; however, consideration is given to the particular application when determining which derivation to use.

3.6.2.2.1.4 RELAP4 Analysis

RELAP4 (Ref. 10) is a computer program developed primarily to describe the thermal-hydraulic transient behavior of water-cooled nuclear reactors subjected to a loss of coolant. This code was used to describe transients resulting from breaks in both main steam and feedwater lines, when necessary. Other codes were also developed and approved to analyze pipe break forces.

For main steam and feedwater systems, only pipe breaks inside containment were analyzed for the effects of pipe whip and jet impingement, although breaks outside of containment may have been analyzed for special case dynamic analyses and responses. Component separation areas and "No-Break Zones" were employed in other areas. In addition, only circumferential or double-ended guillotine break types were considered where seamless piping was installed.

For the calculation of the loading history for RELAP4 analyses, resulting from the above breaks on the piping elbows, the approach suggested in Ref. 8 and 11 was followed. For this purpose, the fluid properties calculated by RELAP4 were utilized.

3.6.2.2.1.5 Time Functions of Jet Thrust Force on Ruptured and Intact Reactor Coolant Loop Piping

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

The hydraulic model represents the behavior of the coolant fluid within the entire RCS. Key parameters calculated by the hydraulic model are pressure, mass flow rate, and density. These are supplied to the thrust calculation, together with plant layout information, to determine the time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure and

momentum flux terms are dominant. The inertia and gravitational terms are taken into account in 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 (Ref. 2) was developed with a capability to provide this information.

The MULTIFLEX 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 required flexibility 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 loads resulting from a LOCA.

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

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

Which includes both the static and dynamic effects. The symbols and units are:

F	=	force, lbf
A	=	aperture area, ft^2
P	=	system pressure, psia
$\dot{m}$	=	mass flow rate, lbm/sec
ρ	=	density, lbm/ft^3
g_c	=	gravitational constant (32.174 $ft\text{-}lbm/lbf\text{-}sec^2$)
A_m	=	mass flow area, ft^2

In the model used to compute forcing functions, the reactor coolant loop system is represented by a model similar to that employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each node is fully described by: 1) blowdown hydraulic information and 2) the orientation of the streamlines of the force nodes in the system, which includes flow areas, and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume, with one 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, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

The THRUST Code (which uses MULTIFLEX results as input) calculates forces exactly the same way as the STHRUST Code (which uses SATAN-IV [Ref. 3] results as input).

The STHRUST Code is described in Reference 4.

3.6.2.2.2 Response Models

3.6.2.2.2.1 Response Model for Other Than Reactor Coolant Loop

The dynamic analysis of system piping is described in Section 3.9(B).

3.6.2.2.2.2 Response Model of the Reactor Coolant Loop Piping, Equipment Supports, and Pipe Whip Restraints

The dynamic analysis of the reactor coolant loop piping for LOCA loadings is described in Section 3.9(N) and Reference 1.

3.6.2.3 <u>Dynamic Analysis Methods to Verify Integrity and Operability</u>

3.6.2.3.1 Dynamic Analysis Methods to Verify Integrity and Operability for Other Than Reactor Coolant Loop

The analytical methods of Reference 5, with the amplifying clarifications and assumptions discussed in Section 3.6.2.2, were used to determine the jet impingement effects and loading effects applicable to components and systems resulting from postulated pipe breaks and cracks.

This information was then used in the protection evaluation described in this section, Section 3.6.2.3. This section describes the design of restraints used to protect the essential systems, components, and equipment from the effects of pipe whip.

3.6.2.3.2 Dynamic Analysis Methods to Verify Integrity and Operability for the Reactor Coolant Loop

3.6.2.3.2.1 General

A LOCA is assumed to occur for a branch line break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3.6-2) on outgoing* and down to and including the second check valve (Case III in Figure 3.6-2) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line 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 will not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6-2), a LOCA is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the reactor coolant loop (RCL) are defined as "large" for the purpose of this criteria and as having an inside diameter greater than 4 inches up to the largest connecting line, generally the pressurizer surge line. Rupture of these lines results in a rapid blowdown from the 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" if they have an inside diameter equal to or less than 4 inches. This size is such that emergency core cooling system analyses, using realistic assumptions, show that no clad damage is expected for a break area of up to 12.5 square inches, corresponding to 4-inch inside diameter piping.

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a 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 have been designed to provide protection for a reactor coolant system pipe rupture of a size up to and including a double-ended severance of a reactor coolant loop.

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

---

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

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

b. The containment leaktightness is not decreased below the design value, if the break leads to a LOCA.*

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.

   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 Reactor Coolant System Piping

Propagation of damage resulting from the rupture of a reactor coolant loop is permitted to occur but must not exceed the design basis for calculating containment and subcompartment pressures, loop hydraulic forces, reactor internals reaction loads, primary equipment support loads, or emergency core cooling system performance.

Large branch line piping, as defined in Section 3.6.2.3.2.1, is restrained to meet the following criteria, in addition to items a through c of Section 3.6.2.3.2.1, for a pipe break resulting in a LOCA.

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

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

3.6.2.3.2.3 Small Branch Lines

In the unlikely event that one of the small pressurized lines, as defined in Section 3.6.2.3.2.1, should 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 Section 3.6.2.3.2.1.

---

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

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.

b. Propagation of the break in the affected leg is permitted but must be limited to a total break area of 12.5 square inches (4 inches 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. Damage to the high head safety injection lines connected to the other leg of the affected loop or to the other loops is prevented.

d. 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.

3.6.2.3.2.4 Design and Verification of Adequacy of RCL Components and Supports

The methods described below are used in the Westinghouse design and verification of the adequacy of primary reactor coolant loop (RCL) components and supports. It is emphasized that these methods are used only to determine jet impingement loads on RCL components and supports.

The design basis postulated pipe rupture locations for the reactor coolant loop piping are determined, using the criteria given in Section 3.6.2. These design basis ruptures are used here as the rupture locations for consideration of jet impingement effects on primary equipment and supports.

The dynamic analysis, as discussed in Section 3.6.2.2.2, is used to determine maximum piping displacements at each design basis rupture location. These maximum piping displacements are used to compute the effective rupture flow area at each location.

This area and rupture orientation are then used to determine the jet flow pattern and to identify any primary components and supports which are potential targets for jet impingement.

The jet thrust at the point of rupture is based on the fluid pressure and temperature conditions occurring during normal (100 percent) steady state operating conditions of the plant. At the point of rupture, the jet force is equal and opposite to the jet thrust. The force of the jet is conservatively assumed to be constant throughout the jet flow distance. The subcooled jet is assumed to expand uniformly at a half angle of 10 degrees, from which the area of the jet at the target and the fraction of the jet intercepted by the target structure can be readily determined.

The shape of the target affects the amount of momentum change in the jet and thus affects the impingement force on the target. The target shape factor is used to account for target shapes which do not deflect the flow 90 degrees away from the jet axis.

The method used to compute the jet impingement load on a target is one of the following:

a. The dynamic effect of jet impingement on the target structure is evaluated by applying a step load whose magnitude is given by:

$$F_j = K_o P_o A_{mB} R S$$

where:

F_j = jet impingement load on target, lbf

K_o = dimensionless jet thrust coefficient based on initial fluid conditions in the broken loop

P_o = initial system pressure, lb/in.^{2}

A_{mB} = calculated maximum break flow area, in.^{2}

R = fraction of jet intercepted by target

S = target shape factor

Discharge flow areas for limited flow area circumferential breaks are obtained from reactor coolant loop analyses performed to determine the axial and lateral displacements of the broken ends as a function of time. A_{mB} is the maximum break flow area occurring during the transient, and is calculated as the total surface area through which the fluid must pass to emerge from the broken pipe. Using geometrical formulations, this surface area is determined to be a function of the pipe separation (axial and transverse) and the dimensions of the pipe (inside and outside diameter).

If simplified static analysis is performed instead of a dynamic analysis, the above jet load (F_j) is multiplied by a dynamic load factor. For an equivalent static analysis of the target structure, the jet impingement force is multiplied by a dynamic load factor of 2.0. This factor assumes that the target can be represented as essentially a one degree of freedom system, and the impingement force is conservatively applied as a step load.

The calculation of the dimensionless jet thrust coefficient and break flow area is discussed in Section 3.6.2.5.

b. The dynamic effect of jet impingement is evaluated by applying the following time-dependent load to the target structure.

$$F_j = K\, P\, A_B\, RS$$

where the system pressure P is a function of time; the jet thrust coefficient K is evaluated as a function of system pressure and enthalpy; and the break flow area A_B is a function of time.

3.6.2.3.3 Types of Restraints

3.6.2.3.3.1 Restraints Other Than Reactor Coolant Loop Restraints

To satisfy varying requirements of available space, permissible pipe deflection, and equipment operability, the restraints are generally located as close as possible to the postulated breaks, in order to limit whipping of the failed pipe in a direction away from the break. Where necessary, guides were used to prevent uncontrolled motion of the pipe in a direction other than that caused by the primary motion generated by the blowdown force. A typical example is shown in Figure 3.6-4.

Restraints identified as isolation restraints are located to protect an essential portion of a piping system from postulated leaks either upstream or downstream of the protected area. These restraints limit pipe motion in all directions. A typical example of an isolation restraint is shown in Figure 3.6-5.

The restraints are of three design types. These include two types of large gap restraints and one type of close gap restraint.

a. Large gap restraints

In order to account for dynamic and gap effects, restraints utilizing either energy absorbing honeycomb material (EAHM) or stainless steel upset rods are used. Large gap restraints are employed where the resulting piping motion may be tolerated without causing a rupture elsewhere in the piping system.

EAHM restraints are the large gap restraints most frequently used. This type of restraint consists of substructures which are allowed to behave plastically within acceptable ductility ratios and have an energy dissipating material (stainless steel honeycomb) between the pipe and the substructure. A typical example of an EAHM restraint is shown in Figure 3.6-6.

The upset rod restraint prevents uncontrolled pipe motion by using its capacity to undergo considerable plastic deformation, thereby absorbing

the kinetic energy of the whipping pipe. A typical example of a rod restraint is shown in Figure 3.6-7.

b. Close Gap Restraints

Close gap restraints are installed where large piping motions permitted by large gap restraints could not be tolerated. The primary purpose of close gap restraints is to limit pipe stresses in areas which are designated as no-break zones. A typical example of a close gap restraint is shown in Figure 3.6-8.

3.6.2.3.3.2 Restraints for Reactor Coolant Loop

Pipe restraints and locations are discussed in Section 5.4.14.

3.6.2.3.4 Analytical Methods

3.6.2.3.4.1 Restraints Other Than Reactor Coolant Loop Restraints

a. Location of restraints

For purposes of locating restraints, the collapse moment of the pipe is determined in the following manner:

$M_p = kS_yS$ for stainless steel pipe

where:

k = 2.5

S_y = yield stress at pipe operating temperature

S = elastic modulus of pipe

$$M_p = 1.07S_u \frac{R_o^{3.14} - R_i^{3.14}}{R_o^{0.14}}$$ for carbon steel pipe (Ref. 13)

where:

S_u = jultimate stress at pipe operating temperature

R_o = outside radius of pipe

R_i = inside radius of pipe

Restraints (with the exception of isolation restraints) are located as close to the postulated break as practicable. Restraints located so that a collapse

moment will not form in the pipe require no further evaluation because the pipe whip is limited by the rigidity of the piping. If, due to physical limitations, restraints are located so that collapse mechanisms in the pipe may form, the consequences of the whipping pipe and the jet impingement effect are further investigated. Guides are provided where necessary to control pipe motion.

b. Design of Restraints

One of the following three methods, depending upon the type of restraint, is used to determine the response of the piping/restraint/supporting structure to the jet thrust developed by the postulated pipe rupture. These methods are energy balance, jet thrust with dynamic load factor of 2, and dynamic analysis using a lumped parameter model. All methods address the following effects, as appropriate:

1. Stiffness characteristics of the piping system, restraint system, major components, and supporting walls and structures

2. Transient forcing functions acting on the piping system, and jet thrusts on structures

3. Elastic and inelastic deformation of piping and/or restraints

4. Insulation thickness

5. Seismic and thermal movements (for determination of clearance values)

The energy balance method of analysis is discussed in Section 3.0 of Reference 5. This method is the primary method used for large gap restraints as described below:

Forcing Function - obtained from Reference 5.

Resistance Response of Piping System - the resistance of piping system (load-deflection response) was achieved by a static analysis (by inputting the force at the postulated pipe break location). The displacement obtained for a corresponding force gave the force-deflection response of the piping system in the elastic range. A perfectly plastic response for the piping system was assumed when the intensified stress (due to the stress intensification factor of the fitting) at the first elbow beyond the pipe whip restraint reached yield stress of the material.

Restraint Response:

EAHM Restraints - This is basically an energy dissipating material which is supported by a substructure. This substructure is allowed to behave plastically within acceptable ductility ratios as defined in BC-TOP-9A. The kinetic energy of the impacting pipe is absorbed by the collapse of the crushable honeycomb core. The substructure, in turn, is designed to absorb the sudden, impulsive dynamic loading created by the crushing EAM (Energy Absorbing Material). The properties as a function of cell size and web size of the honeycomb core were obtained by test by the manufacturer for the specific material used.

The EAHM restraint resistance R_r was determined from equation (1) below:

$$FY = R_r(Y - Y_g) + R_p \frac{Y_p}{2} + R_p(Y - Y_p) \tag{1}$$

$$R_r = \frac{FY - R_p\left(Y - \frac{Y_p}{2}\right)}{Y - Y_g}$$

where:

- F = pipe jet thrust
- Y = total pipe displacement
- Y_g = gap between pipe and restraint
- R_p = maximum pipe resistance
- Y_p = elastic displacement of pipe

where:

$$R_r < A_m P_c \text{ and } \alpha\ t \geq (Y - Y_g) \tag{2}$$

where:

- A_m = cross sectional area of energy absorbing honeycomb material
- P_c = crushing strength of the energy absorbing honeycomb material
- α = allowable deformation in percent of total thickness (t)
- t = total thickness of the energy absorbing honeycomb material

For a suitable value of P_c, A_m is determined from equation (2). Where crushable honeycomb energy absorbing material is used, the material will not experience a deflection in excess of that which is defined by the horizontal portion of its load deflection curve as determined by test, under designed loads.

Upset Rod Restraints

The analytical procedures used to size the upset rod restraint are based on an energy balance method similar to that used for the EAHM restraint design. These are illustrated using a simplified example. Assuming the jet thrust force as constant with time, the strain energy absorbed by the rod in deflecting from its initial configuration to the maximum allowable strain (50% ultimate strain) is equal to the work generated by jet thrust force. In equation form this becomes:

$$W = F(Y_g + L_e e) = \frac{2n\Pi d^2}{4} \quad L_e(u)$$

where:

where:	W	=	total strain energy
	L_e	=	effective length of the restraint determined by test
	n	=	number of upset rods per restraint
	d	=	diameter of rod
	u	=	strain energy per unit volume conservatively idealized to represent the material properties
	e	=	maximum strain allowed

Assuming a plastic collision between the pipe and the restraint and ignoring the energy absorbed by the pipe (in this example) the rod can be sized by solving for (d).

Substructures for both the EAHM and upset rod restraints are allowed to behave plastically throughout a postulated pipe break event. Ductility ratios are in accordance with BC-TOP-9A. A ductility ratio of three is used for anchor bolts and welded studs, based on test data. Design methods are in accordance with Sections 3.8.3 and 3.8.4.

For some close-gap restraints, the simplified jet thrust with load factor method was used. Briefly, the force on the restraint was taken as equal to the jet thrust (pressure x area x thrust coefficient) multiplied by a dynamic load factor. This load factor was conservatively assumed to be 2, the largest possible for a restraint which was virtually in contact with the pipe. (If the clearance between pipe and restraint was large enough to permit the whipping pipe to attain significant velocity before contacting the restraint, thus causing impact effects, other analytical methods were used.)

As an alternate to the energy balance method of analysis, a dynamic analysis of the isolation restraints using a lumped parameter model is employed. The model is shown in Figure 3.6-9.

To calculate the isolation restraint design loads, resulting from a postulated piping failure, a dynamic analysis is performed. PIPE RUP (see 3.9(B).7, Ref. 4) is used to perform this analysis. The isolation restraint is designed such that in the event of a postulated piping failure, inside or outside containment, the "no break zone" criteria per Section 3.6.2.1.1e is met.

3.6.2.3.4.2 Reactor Coolant Loop Restraints

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

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

3.6.2.4 Protective Assembly Design Criteria

3.6.2.4.1 Jet Impingement Barriers and Shields

Barriers and shields, which may be either of steel or concrete construction, are provided to protect essential equipment from the effects of jet impingement resulting from postulated pipe breaks. Barriers differ from shields in that they may also accept the impact of whipping pipes. Barriers and shields include walls and floors and structures specifically designed to provide protection from postulated pipe breaks. Barrier and shield design is based on the methods of Reference 5, Section 3.0, and the elastic-plastic methods for dynamic analysis included in Reference 14. Design criteria and loading combinations are in accordance with Sections 3.8.3 and 3.8.4.

3.6.2.4.2 Auxiliary Guardpipes

The use of guardpipes has been minimized by plant arrangement and routing of high-energy piping. Where they are used, guardpipes are designed to withstand all environmental, jet impingement, and impact effects of postulated breaks of the enclosed pipe. Design criteria, loading combinations, and methods of analysis are similar to those for barriers and shields described in Section 3.6.2.4.1.

3.6.2.5 Material to be Submitted for the Operating License Review

3.6.2.5.1 Piping Systems Other Than Reactor Coolant Loop

Pipe break locations were obtained in accordance with the criteria of Section 3.6.2.1. Pipe crack locations were postulated to occur at any location, as stated in Section 3.6.2.1.

High-energy piping with break locations identified are provided in isometric drawings, Figure 3.6-1. Break types are also shown (i.e., circumferential or longitudinal). The stress results which were utilized to determine the break types and locations are given in Table 3.6-3. If there are changes in the pipe stress analysis, the stress tables will be updated only when those changes affect the break locations shown on the figures previously mentioned. Associated stress nodes are shown in Figure 3.6-1. High-energy pipe break effects analysis is discussed room-by-room in Table 3.6-4.

Each piping isometric (Figure 3.6-1) references the appropriate sheet of Table 3.6-4 by which the effects analysis is discussed for all breaks on that isometric drawing. Table 3.6-4 discussion includes pipe whip, jet impingement, flooding, room pressurization, temperature effect, and humidity effects.

Moderate-energy piping crack locations are defined in Section 3.6.2.1.2.4. Evaluation of the effects of moderate-energy cracks is discussed in Appendix 3B.

The augmented inservice inspection plan is discussed in Section 6.6.8.

Pipe whip restraints are designed in accordance with Section 3.6.2.3. Restraint locations and orientation for each high-energy break are shown in Figure 3.6-1. Barriers and shields are designed in accordance with the criteria of Section 3.6.2.4. Jet thrust and impingement forces were determined in accordance with Section 3.6.2.2. Thrust forces for each break are presented in Figure 3.6-1.

3.6.2.5.2 Reactor Coolant Loop

a. Figure 3.6-3 identifies the design basis break locations and orientations for the reactor coolant loops.

   The primary plus secondary stress intensity ranges and the fatigue cumulative usage factors at the design break locations are specified in Reference 1 for a reference fatigue analysis. The reference analysis has been prepared to be applicable for many plants. It uses seismic umbrella moments which are higher than those used in Reference 1, in which the primary stress is equal to the limits of equation 9 in NB-3650 (Section III of the ASME Boiler and Pressure Vessel Code) at many locations in the system, where in Reference 1 one location was at the limit. Therefore, the results of the reference analysis may differ slightly from Reference 1, but

the philosophy and conclusions of Reference 1 are valid. There are no other locations in the model used in the reference fatigue analysis, consistent with Reference 1, where the stress intensity ranges and/or usage factors exceed the criteria of 2.4 S_m and 0.2, respectively.

The Code limit for actual plant fatigue damage, measured by cumulative usage factors, is satisfied at all locations on the reactor coolant loop piping. The maximum cumulative usage factor obtained for each leg is shown in Reference 17. The primary-plus-secondary stress intensity and fatigue usage factor evaluation of the reactor coolant loop confirms that breaks other than those identified in Reference 1 need not be postulated.

b. Pipe whip restraints associated with the main reactor coolant loop are described in Section 5.4.14.

c. The methods and analysis procedures used to determine jet impingement loads associated with the rupture of the reactor coolant loop piping are discussed in Section 3.6.2.3. These loads are used to determine the adequacy of the primary equipment and supports.

d. Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Section 3.9(N).1.4. Pipe rupture loads include not only the jet thrust forces acting on the piping but also jet impingement loads on the primary equipment and supports.

3.6.3 REFERENCES

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

2. Takeuchi, K., et al., "MULTIFLEX-A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708-P-A, Volumes 1 and 2 (Proprietary) and WCAP-8709-A, Volumes 1 and 2 (Non-Proprietary), February 1976.

3. Bordelon, F. M., "A Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant (SATAN-IV Digital Code)," WCAP-7750, August 1971.

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

5. "Design for Pipe Break Effects," BN-TOP-2, Revision 2, Bechtel Power Corporation, May 1974.

6. NRC Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," November 24, 1975.

7. NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," November 24, 1975.

8. Moody, F. J., "Fluid Reaction and Impingement Loads," presented at the ASCE Specialty Conference, Chicago, Ill., December 1973.

9. American Society of Mechanical Engineers, "Thermodynamic and Transport Properties of Steam Comprising Tables and Charts for Steam and Water," 1967 Edition.

10. Aerojet Nuclear Company, "RELAP4/MOD5 - A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," Volumes I-III, ANCR-NUREG-1335, September 1976.

11. Moody, F. J., "Time-Dependent Pipe Forces Caused by Blowdown and Flow Stoppage," ASME Paper No. 73-FE-23, June 1973.

12. "Subcompartment Pressure Analyses," BN-TOP-4, Revision 1, Bechtel Power Corporation, October 1977.

13. Gerber, T. L., "Plastic Deformation of Piping Due to Pipe-Whip Loading," ASME Paper No. 74-NE-1, June 1974.

14. Biggs, J. M., Introduction to Structural Dyanmics, McGraw-Hill Book Company, New York, 1964.

15. NRC Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," June 1987.

16. "Structural Evaluation of the Wolf Creek and Callaway Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," WCAP-12893, Rev. 0, March 1991.

17. "Structural Analysis of the Reactor Coolant Loop for Standard Nuclear Unit Power Plant System," WCAP-9728 Volume 1 Revision 4, July 2004.

18. WCAP-10691, “Technical Basis for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Callaway and Wolf Creek Plants,” October, 1984.

19. “Technical Justification for Eliminating 10-inch Accumulator Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant,” WCAP-16019-P (Proprietary) and WCAP-16019-NP (Non-Proprietary), February 2003.

20. “Technical Justification for Eliminating 12-inch Residual Heat Removal (RHR) Lines Rupture as the Structural Design Basis for Callaway Nuclear Power Plant,” WCAP-16020-P (Proprietary) and WCAP-16020-NP (Non-Proprietary), February 2003.

21. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 161 to Facility Operating License No. NPF-30 Union Electric Company Callaway Plant, Unit 1 Docket No. 50-483, dated April 12, 2004.

22. WCAP-14059, “Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Callaway and Wolf Creek Plants after Elimination of SG Snubbers,” August 1994 (Westinghouse Proprietary Class 2).

23. Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures -- Final Rule (Broad Scope), 52 FR 41288, October 27, 1987.

24. NRC Generic Letter 87-11, “Relaxation in Arbitrary Intermediate Pipe Rupture Requirements,” June 19, 1987.

25. George, TL, et al., GOTHIC Containment Analysis Package User Manual, Version 7.2b (QA), NAI 8907-02, Rev. 18, Numerical Applications, Inc., Richland, WA, March 2009.

26. APA-ZZ-00662, Appendix F, “Requirements for High Density Polyethylene (HDPE) piping for Nuclear Service,” Relief Request 13R-10, originally submitted for NRC review as ULNRC-05517 Enclosure 5, revised by ULNRC-05529 Attachment 1, and approved by NRC October 31, 2008.

TABLE 3.6-1 SAFETY-RELATED SYSTEMS AND HIGH AND MODERATE ENERGY SYSTEMS

SYSTEM	ESF: TOTAL SYSTEM	ESF: PORTIONS OF SYSTEM	ESF: CONTAINMENT ISO. ONLY	LOCATED IN SAFETY RELATED STRUCTURES: CONTAINMENT BUILDING	LOCATED IN SAFETY RELATED STRUCTURES: AUXILIARY BUILDING	LOCATED IN SAFETY RELATED STRUCTURES: FUEL BUILDING	LOCATED IN SAFETY RELATED STRUCTURES: CONTROL BUILDING	HIGH/MODERATE ENERGY: HIGH ENERGY SYSTEM †	HIGH/MODERATE ENERGY: PIPE WHIP	HIGH/MODERATE ENERGY: MODERATE ENERGY†	HIGH/MODERATE ENERGY: COMPARTMENT PRESSURE	HIGH/MODERATE ENERGY: JET IMPINGEMENT	HIGH/MODERATE ENERGY: FLOODING
MAIN STEAM		●		●	●			●	●		●	●	●
MAIN TURBINE													
CONDENSATE													
FEEDWATER		●		●	●			●	●		●	●	●
FEEDWATER HEATER EXTRACTION, DRAINS AND VENTS													
CONDENSATE POLISHER													
AUXILIARY FEEDWATER	●				●					●			●
DEMINERALIZED WATER MAKEUP STORAGE AND TRANSFER					●	●				●			●
CONDENSATE TRANSFER AND STORAGE					●					●			●
CONDENSATE AND FEEDWATER CHEMICAL CONTROL													
REACTOR COOLANT	●			●				●	●		●	●	●
CHEMICAL AND VOLUME CONTROL		●		●	●			●	●	●	●	●	●
REACTOR MAKEUP WATER			●	●	●	●				●			●
STEAM GENERATOR BLOWDOWN			●	●	●			●	●	●	●	●	●
BORATED REFUELING WATER STORAGE	●				●					●			●
STEAM SEALS													
MAIN TURBINE LUBE OIL													
GENERATOR HYDROGEN AND CARBON DIOXIDE													
GENERATOR SEAL OIL													
STATOR COOLING WATER													
LUBE OIL STORAGE, TRANSFER & PURIFICATION													
CONDENSER AIR REMOVAL													
MAIN TURBINE CONTROL OIL													
CIRCULATING WATER										●			●
SERVICE WATER					●					●			●
CLOSED COOLING WATER										●			●
FUEL POOL COOLING & CLEANUP	●			●	●	●				●			●
ESSENTIAL SERVICE WATER*	●			●	●	●	●			●			●
COMPONENT COOLING WATER	●			●	●	●				●			●
RESIDUAL HEAT REMOVAL***	●			●	●			●	●	●	●	●	●
HIGH PRESSURE COOLANT INJECTION***	●			●	●			●	●	●	●	●	●
CONTAINMENT SPRAY	●			●	●					●			●
ACCUMULATOR SAFETY INJECTION	●			●				●	●		●	●	●
AUXILIARY STEAM GENERATOR													
AUXILIARY STEAM					●			●	●		●	●	●
AUXILIARY TURBINES		●			●			●	●		●	●	●
PLANT HEATING				●	●	●	●			●			●
CENTRAL CHILLED WATER				●	●	●	●			●			●
ESSENTIAL SERVICE WATER PUMP HOUSE BUILDING HVAC*	●												
TURBINE BUILDING HVAC													
MISC. BUILDING HVAC		●			●								
FUEL HANDLING BUILDING HVAC		●				●							
RADWASTE BUILDING HVAC													
CONTROL BUILDING HVAC		●					●						
AUXILIARY BUILDING HVAC		●			●	●							
DIESEL BUILDING HVAC**	●												
CONTAINMENT COOLING		●		●									
CONTAINMENT ATMOS. CONTROL				●									
CONTAINMENT HYDROGEN CONTROL	●			●	●								
CONTAINMENT PURGE		●		●	●								
GASEOUS RADWASTE					●					●			
LIQUID RADWASTE				●	●			●	●	●		●	●
SOLID RADWASTE										●			
DECONTAMINATION			●	●	●	●				●			●
BORON RECYCLE					●					●			●
SECONDARY LIQUID WASTE SYSTEM					●					●			●
EMERGENCY FUEL OIL **	●									●			●
COMPRESSED AIR		●	●	●	●	●	●	●		●			
FIRE PROTECTION				●	●	●	●			●			●
DOMESTIC WATER					●	●	●			●			●
FUEL HANDLING, FUEL STORAGE & REACTOR VESSEL SERVICE		●		●		●							
SERVICE GAS (CO_2, N_2, H_2, AND O_2)				●	●	●	●	●		●			
STANDBY DIESEL ENGINE**	●							●		●			
NUCLEAR SAMPLING			●	●	●	●		●		●			
SANITARY DRAINAGE							●						
CHEMICAL AND DETERGENT WASTE					●								
OILY WASTE				●									
FLOOR & EQUIPMENT DRAINS			●	●	●	●	●						

● - Yes
(blank) - No

* Located in a safety-related area
** Located in diesel building
*** High pressure associated with reactor coolant pressure boundary
† located in a safety-related area

TABLE 3.6-2 DESIGN COMPARISON TO REGULATORY POSITIONS OF REGULATORY GUIDE 1.46, REVISION 0, DATED MAY 1973, TITLED "PROTECTION OF PIPE WHIP INSIDE CONTAINMENT"

The basis for compliance to Regulatory Guide 1.46 is the implementation of NRC Branch Technical Position (BTP) MEB 3-1, NRC BTP ASB 3-1, WCAP-8082-P-A, and WCAP-8172-A. The following provides a summary of the compliance with MEB 3-1 and ASB 3-1.

BTP ASB 3-1 Position	Union Electric Compliance
B.1 Plant Arrangement	
Protection of essential systems and components against postulated piping failures in high- or moderate-energy fluid systems that operate during normal plant conditions and that are located outside of containment should be provided by one of the following plant arrangement considerations:	B.1. Complies. See Section 3.6.1.3.
B.1.a. Plant arrangements should separate fluid system piping from essential systems and components. Separation should be distances between essential systems and components and fluid system piping such that the effects of any postulated piping failure therein (e.g., pipe whip, jet impingement, and the environmental conditions resulting from the escape of contained fluids as appropriate to high- or moderate-energy fluid system piping) cannot impair the integrity or operability of essential systems and components.	
B.1.b. Fluid system piping or portions thereof not satisfying the provisions of B.1.a should be enclosed within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or compartments designed to withstand the effects of postulated piping failures in nearby fluid systems.	
B.1.c. Plant arrangements or system features that do not satisfy the provisions of either B.1.a or B.1.b should be limited to those for which the above provisions are impractical because of the stage of design or construction of the plant; because the plant design is based upon that of an earlier plant accepted by the staff as a base plant under the Commission's standardization and replication policy; or for other substantive reasons such as particular design features of the fluid systems. Such cases may arise, for example, (1) at interconnections between fluid systems and essential systems and components, or (2) in fluid systems having dual functions (i.e., required to operate during normal plant conditions as well as to shut down the reactor). In these cases, redundant design features that are separated or otherwise protected from postulated piping failures, or additional protection, should be provided so that the effects of postulated piping failures are shown by the analyses and guidelines of B.3 to be acceptable. Additional protection may be provided by restraints and barriers or by designing or testing essential systems and components to withstand the effects associated with postulated piping failures.	

TABLE 3.6-2 (Sheet 2)

BTP ASB 3-1 Position	Union Electric Compliance
B.2 Design Features	B.2.a. Complies, as described in Table 3.2-3.
B.2.a. Essential systems and components should be designed to meet the seismic design requirements of Regulatory Guide 1.29.	
B.2.b. Protective structures or compartments, fluid system piping restraints, and other protective measures should be designed in accordance with the following:	
(1) Protective structures or compartments needed to implement B.1 should be designed to seismic Category I requirements. The protection structures should be designed to withstand the effects of a postulated piping failure (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with loadings associated with the operating basis earthquake and safe shutdown earthquake within the respective design load limits for structures. Piping restraints, if used, may be taken into account to limit effects of the postulated piping failure.	B.2.b.(1) Complies. See Sections 3.8.3 and 3.8.4 for loading combinations.
(2) High-energy fluid system piping restraints and protective measures should be designed such that a postulated break in one pipe cannot, in turn, lead to rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break. An unrestrained whipping pipe should be considered capable of (a) rupturing impacted pipes of smaller nominal pipe sizes and (b) developing through wall leakage cracks in larger nominal pipe sizes with thinner wall thickness, except where experimental or analytical data for the expected range of impact energies demonstrate the capability to withstand the impact without failure.	B.2.b.(2) Complies. See Section 3.6.1.1i.

TABLE 3.6-2 (Sheet 3)

BTP ASB 3-1 Position	Union Electric Compliance
B.2.c. Fluid system piping in containment penetration areas should meet the following design provisions:	B.2.c. All high-energy fluid system and selected moderate-energy fluid piping in the containment penetration areas comply with the following criteria:
(1) Portions of fluid system piping between the required restraints located inside and outside containment beyond the isolation valves of single barrier containment structures (including any rigid connection to the containment penetration) that connect, on a continuous or intermittent basis, to the reactor coolant pressure boundary, or the steam and feedwater systems of PWR plants, should be designed to the stress limits specified in B.1.b or B.2.b of Branch Technical Position (BTP) MEB 3-1, attached to Standard Review Plan 3.6.2. These portions of high-energy fluid system piping should be provided with pipe whip restraints that are capable of resisting bending and torsional moments produced by a postulated piping failure either upstream or downstream of the containment isolation valves. The restraints should be located reasonably close to the containment isolation valves and should be designed to withstand the loadings resulting from a postulated piping failure beyond these portions of piping so that neither isolation valve operability nor the leaktight integrity of the containment will be impaired.	B.2.c.(1) High-energy (H-E) piping systems associated with the steam tunnel, i.e., main steam, feedwater, and steam generator blowdown, are provided with isolation restraints which protect the penetration piping in the steam tunnel. For further discussion of the main steam, feedwater and steam generator blowdown piping penetration areas, see Section 3.6.2.1.1.e. For all other H-E piping penetrations, isolation restraints have been provided reasonably close to the containment isolation valves to protect the "no break zone" piping, protect the integrity of the penetration, and protect the operability of the isolation valves (when present), assuming a rupture at the postulated intermediate breakpoints or terminal ends outside the regions defined as "no break zone." For further discussions see Section 3.6.2.1.1.e.
(2) Portions of fluid system piping between the required restraints located inside and outside containment beyond the isolation valves of dual barrier containment structures should also meet the design provisions of B.2.c.(1). In addition, those portions of piping that pass through the containment annulus, and whose postulated failure could affect the leaktight integrity of the containment structure or result in pressurization of the containment annulus beyond the design limits should be provided with an enclosing protective structure. For the purpose of establishing the design parameters (i.e., pressure, temperature) of the enclosing protective structure, a full flow area opening should be assumed in that portion of piping within the enclosing structure and vent areas should be taken into account, if provided, in the enclosing structure. Where guard pipes for individual process pipes are used as an enclosing protective structure, such guard pipes should be designed to meet the requirements specified in B.1.b(6) of BTP MEB 3-1.	B.2.c.(2) Not applicable to Callaway.
(3) Terminal ends of the piping runs extending beyond these portions of high-energy fluid system piping should be considered to originate at a point adjacent to the required pipe whip restraints located inside and outside containment.	B.2.c.(3) Terminal ends of H-E piping fall within the "no break zone" boundary; therefore, no terminal and breaks are postulated except to calculate the design load for the isolation restraint.
(4) Piping classification as required by Regulatory Guide 1.26 should be maintained without change until beyond the outboard restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable.	B.2.c(4) Complies.

TABLE 3.6-2 (Sheet 4)

BTP ASB 3-1 Position	Union Electric Compliance
B.2.d. Inservice examination and related design provisions should be in accordance with the following:	
(1) The protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI, division 1, "Rules for Inspection and Testing of Components in Light-Water Cooled Plants."	B.2.d.(1) Inservice Examinations are in accordance with the Risk-Informed Break Exclusion Region (RI-BER) Augmented Inservice Inspection Program. See Sections 6.6 and 3.6.2.1.1.e.
(2) For those portions of fluid system piping identified in B.2.c, includes piping running from inboard to outboard restraints in containment penetration areas, the extent of inservice examinations completed during each inspection interval (IWA-2400, ASME code, Section XI) should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.	B.2.d.(2) 25% of the Risk Category 1, 2, and 3 welds and 10% of the Risk Category 4 and 5 welds are selected for examination in accordance with the RI-BER Augmented Inservice Inspection Program. See Sections 6.6 and 3.6.2.1.1e.
(3) For those portions of fluid systems piping enclosed in guard pipes, inspection ports should be provided in guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.	B.2.d.(3) Callaway has no guard pipes located in the penetration areas. Guard pipes utilized in other areas comply with this position.
(4) The areas subject to examination should be defined in accordance with Examination Categories C-F and C-G for Class 2 piping welds in Tables IWC-2520.	B.2.d.(4) See Section 6.6.
B.3. Analyses and Effects of Postulated Piping Failures	
B.3.a. To show that the plant arrangement and design features provide the necessary protection of essential systems and components, piping failures should be postulated in accordance with BTP MEB 3-1, attached to Standard Review Plan 3.6.2. In applying the provisions of BTP MEB 3-1, each longitudinal or circumferential break in high-energy fluid system piping or leakage crack in moderate-energy fluid system piping should be considered separately as a single postulated initial event occurring during normal plant conditions. An analysis should be made of the effects of each such event, taking into account the provisions of BTP MEB 3-1 and of the system and component operability considerations of B.3.b. below. The effects of each postulated piping failure should be shown to result in offsite consequences within the guidelines of 10 CFR Part 100 and to meet the provisions of B.3.c and d below.	B.3.a Complies. See Section 3.6.1.1d, 3.6.1.1k, and Table 3.6-4.
B.3.b. In analyzing the effects of postulated piping failures, the following assumptions should be made with regard to the operability of systems and components:	
(1) Offsite power should be 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.	B.3.b.(1) Complies. See Section 3.6.1.1e.

TABLE 3.6-2 (Sheet 5)

BTP ASB 3-1 Position		Union Electric Compliance	
(2)	A single active component failure should be assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in B.3.b.(3) 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.	B.3.b.(2)	Complies. See Section 3.6.1.1f.
(3)	Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system, i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure, single failures of components in the other train or trains of that system only need not be assumed provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that may, in some plant designs, qualify as dual-purpose essential systems are service water systems, component cooling systems, and residual heat removal systems.	B.3.b.(3)	Complies. Section 3.6.1.1g defines a train to include those systems which support its function. Note that the criteria is also applied to single-purpose and high-energy systems, since the same quality, design, construction, and inspection standards are used. The only applicable H-E piping system is CVCS charging.
(4)	All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account should be 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 should be judged on the basis of ample time and adequate access to equipment being available for the proposed actions.	B.3.b.(4)	Complies. See Section 3.6.1.1h.
B.3.c.	The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, should not preclude habitability of the control room or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure.	B.3.c.	Complies.
B.3.d.	A postulated failure of piping not designed to seismic Category I standards should not result in any loss of capability of essential systems and components to withstand the further effects of any single active component failure and still perform all functions required to shut down the reactor and mitigate the consequences o the postulated piping failure.	B.3.d.	Complies. See Section 3B.2.1.

TABLE 3.6-2 (Sheet 6)

BTP MEB 3-1 Position	Union Electric Compliance
B.1. High-Energy Fluid System Piping	
B.1.a. Fluid Systems Separated from Essential Systems and Components For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of Branch Technical Position BTP ASB 3-1, a review of the piping layout and plant arrangement drawings should clearly show the effects of postulated piping breaks at any location are isolated or physically remote from essential systems and components. At the designer's option, break locations as determined from 1.c. and 1.d of this position may be assumed for this purpose.	B.1.a. Complies. See Section 3.6.1.3.2.
B.1.b. Fluid System Piping In Containment Penetration Areas Breaks need not be postulated in those portions of piping identified in B.2.c of BTP ASB 3-1 provided they meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the following additional design requirements: (1) The following design stress and fatigue limits should not be exceeded. For ASME Code, Section III, Class 1 Piping	B.1.b. Complies.
(a) The maximum stress range should not exceed $2.4S_m$ (b) The maximum stress range between any two load sets (including the zero load set) should be calculated by Eq. (10) in Paragraph NB-3653, ASME Code, Section III, for normal and upset plant conditions and an operating basis earthquake (OBE) event transient. If the calculated maximum stress range of Eq. (10) exceeds the limit of B.1.b(1)(a) but is not greater than $3S_m$, the limit of B.1.b(1)(c) should be met. If the calculated maximum stress range of Eq. (10) exceeds $3S_m$, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 should meet the limit of B.1.b(1)(a) and the limit of B.1.b(1)(c). (c) The cumulative usage factor should be less than 0.1 if consideration of fatigue limits is required according to B.1.b(1)(b).	B.1.b.(1)(a)-(d) There is no Class 1 piping in containment penetration areas on Callaway.

TABLE 3.6-2 (Sheet 7)

BTP MEB 3-1 Position	Union Electric Compliance
(d) The maximum stress, as calculated by Eq. (9) in Paragraph NB-3652 under the loadings resulting from a postulated piping failure beyond these portions of piping should not exceed $2.25S_m$ except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirements specified in SRP 3.9.3. Primary loads include those which are deflection limited by whip restraints.	
For ASME Code, Section III, Class 2 Piping	
(e) The maximum stress ranges as calculated by the sum of Eq. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event should not exceed $0.8(1.2S_h + S_A)$.	B.1.b.(1)(e) Complies.
(f) The maximum stress, as calculated by Eq. (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping should not exceed $1.8S_h$. Primary loads include those which are deflection limited by whip restraints. The exceptions permitted in (d) may also be applied provided that when the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ANSI B31.1 (see ASB 3-1 B.2.c.[4]), the piping shall either be of seamless construction with full radiography of all circumferential welds, or all longitudinal and circumferential welds shall be fully radiographed.	B.1.b.(1)(f) Complies. For further discussion see Section 3.6.2.1.1.e.
(2) Welded attachments, for pipe supports or other purposes, to these portions of piping should be avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of B.1.b(1).	B.1.b.(2) Welded attachments to these portions of the piping are minimized. Attachments for welded pipe supports are reviewed separately for local stresses and the limits B 1.b(1) will be met. Stress analysis is performed to demonstrate the Eq. (9) and (10) stresses do not exceed $0.8(1.2 S_h + S_A)$.
(3) The number of circumferential and longitudinal piping welds and branch connections should be minimized. Where guard pipes are used, the enclosed portion of <u>fluid system</u> piping should be seamless construction unless specific access provisions are made to permit inservice volumetric examination of the longitudinal welds.	B.1.b.(3) Complies. Guard pipes are not used in the containment penetration areas.

TABLE 3.6-2 (Sheet 8)

BTP MEB 3-1 Position	Union Electric Compliance
(4) The length of these portions of piping should be reduced to the minimum length practical.	B.1.b(4) See compliance statement to BTP ASB 3-1 position B.2.c.(1).
(5) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) should not require welding directly to the outer surface of the piping (e.g., flued integrally-forged pipe fittings may be used) except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of B.1.b(1).	B.1.b.(5) All high-energy containment penetrations are flued integrally-forged piped fittings. Pipe whip restraints do not require welding directly to the outer surface of the piping, except where examination in accordance with the Risk-Informed Break Exclusion Region (RI-BER) Program and a review for local stresses are performed. The main steam and main feedwater lines outside the containment have flued integrally-forged pipe fitting whip restraints.
(6) Guard pipes provided for those portions identified in B.2.c(2) of BTP ASB 3-1 should be constructed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III, where the guard pipe is part of the containment boundary. In addition, the entire guard pipe assembly should be designed to meet the following requirements and tests:	B.1.b.(6) Callaway has no guard pipes located in the containment penetration areas.
(a) The design pressure and temperature should not be less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.	
(b) The design stress limits of Paragraph NE-3131(c) should not be exceeded under the loading associated with containment design pressure and temperature in combination with the safe shutdown earthquake.	
(c) Guard pipe assemblies should be subjected to a single pressure test at a pressure not less than its design pressure.	
B.1.c. Fluid Systems Enclosed Within Protective Structures	
(1) With the exceptions of those portions of piping identified in B.1.b, breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run within a protective structure or compartment designed to satisfy the plant arrangement provisions of B.1.b or B.1.c of BTP ASB 3-1.	
(a) At terminal ends of the run if located within the protective structure. Terminal ends are identified in ASB 3-1 B.2.c(3).	B.1.c.(1)(a) Complies. See Section 3.6.2.1.1b. and compliance statement to BTP ASB 3-1 position B.2.c.(3).

TABLE 3.6-2 (Sheet 9)

BTP MEB 3-1 Position	Union Electric Compliance
(b) At intermediate locations selected by one of the following criteria: (i) At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping within the protective structure. A terminal end, as determined by B.1.c(1)(a), may be considered as one of these extremes. (ii) At each location where the stresses 1) exceed $0.8(1.2S_h + S_A)$ but at not less than two separated locations chosen on the basis of highest stress. 2) Where the piping consists of a straight run without fittings, welded attachments, and valves, and all stresses are below $0.8(1.2S_h + S_A)$, a minimum of one location chosen on the basis of highest stress.	B.1.c.(1)(b) Complies. Intermediate breaks are selected solely on the basis of highest calculated stress (i.e., breaks may not be separated by a change in direction of the piping run or located at a weld).
(2) Breaks in non-nuclear class piping should be postulated at the following locations in each piping or branch run: (a) At terminal ends of the run if located within the protective structure. (b) At each intermediate pipe fitting, welded attachment, and valve. If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.	B.1.c.(2) Break postulation in non-nuclear class piping complies. See Section 3.6.2.1.1d. Non-nuclear, high-energy pipes will either be refrained from impacting or affecting the separating structure or the separating structure will be designed for full effects.
(3) Applicable to (1) and (2) above: If a structure separates a high-energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.	B.1.c.(3) Separating structures are analyzed to withstand the dynamic effects of the postulated pipe breaks as defined in B.1.c.(1) and B.1.c.(2) above.

TABLE 3.6-2 (Sheet 10)

BTP MEB 3-1 Position	Union Electric Compliance
B.1.d. Fluid Systems Not Enclosed Within Protective Structures	
(1) With the exceptions of those portions of piping identified in B.1.b, breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run routed outside of, but alongside, above, or below, a protective structure or compartment containing essential systems and components and designed to satisfy the plant arrangement provisions of B.1.b or B.1.c or BTP ASB 3-1. Such piping should be considered as located adjacent to a protective structure if the distance between the piping and structure is insufficient to preclude impairment of the integrity of the structure from the effects of a postulated piping failure assuming the piping is unrestrained. (a) At terminal ends of the run if located adjacent to the protective structure. Terminal ends are identified in ASB 3-1 B.2.c.(3). (b) At intermediate locations selected by one of the following criteria: (i) At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure. (ii) At each location where the stresses 1) exceed $0.8(1.2S_h + S_A)$ but at not less than two separated locations chosen on the basis of highest stress. 2) Where the piping consists of a straight run without fittings, welded attachment, or valves, and all stresses are below $0.8(1.2S_h + S_A)$, a minimum of one location chosen on the basis of highest stress.	B.1.d.(1) No Class 2 or 3 high-energy piping is located outside of the protective structures.
(2) Breaks in non-nuclear class piping should be postulated at the following locations in each piping or branch run: (a) At terminal ends of the run if located adjacent to the protective structure. (b) At each intermediate pipe fitting, welded attachment, and valve.	B.1.d.(2) Complies. With one clarification: On approximately 2.67 feet of pipe on FB-081-HBD-2" and 0.5 feet of pipe on FB-093-HBD-3" between the 8-inch auxiliary steam header and the closed high-energy/moderate-energy boundary valves on these lines, breaks were not postulated. It was judged that the runs were short enough to prevent guillotine breaks and that any breaks that did occur would be in the 8-inch auxiliary steam header. Breaks in the 8-inch header were postulated and evaluated in the vicinity of the connections for lines 081 and 093.

TABLE 3.6-2 (Sheet 11)

BTP MEB 3-1 Position	Union Electric Compliance
(3) Applicable to (1) and (2) above: If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.	B.1.d.(3) Complies.
B.1.e. The designer should identify each piping run he has considered to postulate the break locations required by B.1.c and B.1.d above. In complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer should identify and include all such piping within a designated run in order to postulate the number of breaks required by these criteria.	B.1.e. Complies. See Section 3.6.2.5.
B.2. Moderate-Energy Fluid System Piping	
B.2.a. Fluid Systems Separated from Essential Systems and Components For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of BTP ASB 3-1, a review of the piping layout and plant arrangement drawings should clearly show that the effects of through-wall leakage cracks at any location in piping designed to seismic and non-seismic standards are isolated or physically remote from essential systems and components.	B.2.a. Complies. See Section 3.6.1.3 and Appendix 3B.
B.2.b. Fluid System Piping Between Containment Isolation Valves Leakage cracks need not be postulated in those portions of piping identified in B.2.c. of (BTP) ASB 3-1 provided they meet the requirements of the ASME Code, Section III, Subarticle NE-1120, and are designed such that the maximum stress range does not exceed 0.4 $(1.2S_h + S_A)$ for ASME Code, Section III, Class 2 piping.	B.2.b. Complies. See Section 3.6.2.1.2.4.
B.2.c. Fluid Systems Within, or Outside and Adjacent to, Protective Structures	B.2.c. Where the maximum stress range in Class 3 high density polyethylene (HDPE) piping designed to Reference 26 in Section 3.6.3 is less than 0.4 (1.2S + 1100) a moderate energy crack need not be postulated. See Section 3.6.2.1.2.4.
i. Through-wall leakage cracks should be postulated in seismic Category I fluid system piping located within, or outside and adjacent to, protective structures designed to satisfy the plant arrangement provisions of B.1.b. or B.1.c of BTP ASB 3-1, except (1) where exempted by B.2.b and B.2.d, or (2) where the maximum stress range in these portions of Class 2 or 3 piping (ASME Code, Section III), or non-nuclear piping is less than $0.4(1.2S_h + S_A)$. The cracks should be postulated to occur individually at locations that result in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions developed.	

TABLE 3.6-2 (Sheet 12)

BTP MEB 3-1 Position	Union Electric Compliance
ii. Through-wall leakage cracks should be postulated in fluid system piping designed to non-seismic standards as necessary to satisfy B.3.d of BTP ASB 3-1.	
B.2.d. Moderate-Energy Fluid Systems in Proximity to High-Energy Fluid Systems Cracks need not be postulated in moderate-energy fluid system piping located in an area in which a break in high-energy fluid system piping is postulated, provided such cracks would not result in more limiting environmental conditions than the high-energy piping break. Where a postulated leakage crack in the moderate-energy fluid system piping results in more limiting environmental conditions than the break in proximate high-energy fluid system piping, the provisions of B.2.c should be applied.	B.2.d. Complies. See compliance statement to B.2.b above.
B.2.e. Fluid Systems Qualifying as High-Energy or Moderate-Energy Systems Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods[3] but qualify as moderate-energy fluid systems for the major operational period.	B.2.e. Complies. See Section 3.6.1.1a.
B.3 Type of Breaks and Leakage Cracks in Fluid System Piping	
B.3.a. Circumferential Pipe Breaks The following circumferential breaks should be postulated in high-energy fluid system piping at the locations specified in B.1 of this position:	
(1) Circumferential breaks should be postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch, except where the maximum stress range[1] exceeds the limits specified in B.1.c(1) (b) (ii) and B.1.d (1) (b) (ii) but the circumferential stress range is at least 1.5 times the axial stress range. Instrument lines, one inch and less nominal pipe or tubing size should meet the provisions of Regulatory Guide 1.11.	B.3.a.(1) Complies. See Section 3.6.2.1.2.2.
(2) Where break locations are selected without the benefit of stress calculations, breaks should be postulated at the piping welds to each fitting, valve, or welded attachment. Alternatively, a single break location at the section of maximum stress range may be selected as determined by detailed stress analyses (e.g., finite element analyses) or tests on a pipe fitting.	B.3.a.(2) Complies. All high-energy Class 1, 2, and 3 piping is analyzed by stress calculations. Non-nuclear class high-energy piping breaks are postulated at all welds, fittings, welded attachments, etc.
(3) Circumferential breaks should be assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).	B.3.a.(3) Complies.

TABLE 3.6-2 (Sheet 13)

BTP MEB 3-1 Position	Union Electric Compliance
(4) The dynamic force of the jet discharge at the break location should be based on the effective cross-sectional flow area of the pipe on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.	B.3.a.(4) See Section 3.6.2.2.1.
(5) Pipe whipping should be assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.	B.3.a.(5) Complies. See Section 3.6.1.1j.
B.3.b. Longitudinal Pipe Breaks The following longitudinal breaks should be postulated in high-energy fluid system piping at the locations of the circumferential breaks specified in B.3.a:	
(1) Longitudinal breaks in fluid system piping and branch runs should be postulated in nominal pipe sizes 4-inch and larger, except where the maximum stress range[1] exceeds the limits specified in B.1.c(1)(b)(ii) and B.1.d(1)(b)(ii)) but the axial stress range is at least 1.5 times the circumferential stress range.	B.3.b.(1) Complies. See Section 3.6.2.1.2.2.
(2) Longitudinal breaks need not be postulated at: (a) Terminal ends provided the piping at the terminal ends contains no longitudinal pipe welds (if longitudinal welds are used, the requirements of B.3.b(1) apply). (b) At intermediate locations where the criterion for a minimum number of break locations must be satisfied.	B.3.b.(2) Per Section 3.6.2.1.2.2, only circumferential breaks are postulated at terminal ends, even if a longitudinal pipe weld is present at that point. At intermediate locations, the exception of this position was complied with.
(3) Longitudinal breaks should be assumed to result in an axial split without pipe severance. Splits should be oriented (but not concurrently) at two diametrically-opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).	B.3.b.(3) Complies. See Section 3.6.2.1.3.1.

TABLE 3.6-2 (Sheet 14)

BTP MEB 3-1 Position	Union Electric Compliance
(4) The dynamic force of the fluid jet discharge should be based on circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.	B.3.b.(4) See Section 3.6.2.2.1
(5) Piping movement should be assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis.	B.3.b.(5) Complies.
B.3.c. Through-Wall Leakage Cracks The following through-wall leakage cracks should be postulated in moderate-energy fluid system piping at the locations specified in B.2 of this position:	
(1) Cracks should be postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch.	B.3.c.(1) Complies.
(2) Fluid flow from a crack should be based on a circular opening of area equal to that of a rectangle one-half pipe-diameter in length and one half pipe wall thickness in width.	B.3.c.(2) Complies.
(3) The flow from the crack should be assumed to result in an environment that wets all unprotected components within the compartment, with the consequent flooding in the compartment and communicating compartments. Flooding effects should be determined on the basis of a conservatively estimated time period required to effect corrective actions.	B.3.c.(3) Complies.

TABLE 3.6-2 (Sheet 15)

BTP MEB 3-1 Position (footnotes)	Union Electric Compliance
1 Stresses under normal and upset plant conditions, and an OBE event as calculated by Eq. (9) and (10), Para. NC-3652 of the ASME Code, Section III.	
2 Select two locations with at least 10% difference in stress, or, if stresses differ by less than 10%, two locations separated by a change of direction of the pipe run.	
3 An operational period is considered "short" if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is about 2 percent of the time that the system operates as a moderate-energy fluid system (e.g., systems such as the reactor decay heat removal system qualify as moderate-energy fluid systems; however, systems such as auxiliary feedwater systems operated during PWR reactor startup, hot standby, or shutdown qualify as high-energy fluid systems).	

TABLE 3.6-3 HIGH-ENERGY PIPE BREAK INITIAL STRESS ANALYSIS RESULTS

SYSTEM - MAIN STEAM SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 1 (AB01)

Prob. No. P-001
Issue - 8

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) $0.8\ (S_A + 1.2S_h)$
1*	†	†	†	†
5B Bend	Break deleted per Reference 24			
5M Bend	Break deleted per Reference 24			
20B Bend	Break deleted per Reference 24			
40B Bend	Break deleted per Reference 24			
50B Bend	Break deleted per Reference 24			
50E	Break deleted per Reference 24			
80B Bend	Break deleted per Reference 24			
80E	Break deleted per Reference 24			
90B Bend	Break deleted per Reference 24			
90M Bend	Break deleted per Reference 24			
101*	†	†	†	†

* - Indicates Terminal End
† - Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 2)

SYSTEM - MAIN STEAM SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 1 (AB01)

Prob. No. P-001A
Issue - 8

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (ksi) $0.8 (S_A + 1.2S_h)$
1*	†	†	†	†
5B Bend	Break deleted per Reference 24			
5M Bend	Break deleted per Reference 24			
20B Bend	Break deleted per Reference 24			
20M	Break deleted per Reference 24			
40B Bend	Break deleted per Reference 24			
40E	Break deleted per Reference 24			
50B Bend	Break deleted per Reference 24			
50E	Break deleted per Reference 24			
80B Bend	Break deleted per Reference 24			
80M Bend	Break deleted per Reference 24			
90B Bend	Break deleted per Reference 24			
90M Bend	Break deleted per Reference 24			
101*	†	†	†	†

* - Indicates Terminal End
†- Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 3)

SYSTEM - MAIN STEAM SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 1 (AB01)

Prob. No. P-002
Issue - 8

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (ksi) $0.8 (S_A + 1.2S_h)$
1*	†	†	†	†
5B Bend	Break deleted per Reference 24			
5M	Break deleted per Reference 24			
20B Bend	Break deleted per Reference 24			
20M Bend	Break deleted per Reference 24			
40B Bend	Break deleted per Reference 24			
40E	Break deleted per Reference 24			
60B Bend	Break deleted per Reference 24			
60E	Break deleted per Reference 24			
101*	†	†	†	†

* - Indicates Terminal End
† - Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 4)

SYSTEM - MAIN STEAM SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 1 (AB01)

Prob. No. P-002A
Issue - 8

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) $0.8\,(S_A + 1.2S_h)$
1*	†	†	†	†
5B Bend	Break deleted per Reference 24			
5M	Break deleted per Reference 24			
20B Bend	Break deleted per Reference 24			
20M	Break deleted per Reference 24			
40B Bend	Break deleted per Reference 24			
40E	Break deleted per Reference 24			
101*	†	†	†	†

* - Indicates Terminal End
† - Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 5)

SYSTEM - MAIN FEEDWATER SYSTEM INSIDE CONTAINMENT
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 2 (AE04)

Prob. No. P-003
Issue - 5

Node	Stress (ksi) Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) $0.8\ (S_A + 1.2S_h)$
5*	†	†	†	†
20E	Break deleted per Reference 24			
20M	Break deleted per Reference 24			
27B Bend	Break deleted per Reference 24			
27M	Break deleted per Reference 24			
35M	Break deleted per Reference 24			
35E	Break deleted per Reference 24			
75M	Break deleted per Reference 24			
95M	Break deleted per Reference 24			
95E	Break deleted per Reference 24			
100	Break deleted per Reference 24			
125*	†	†	†	†

* - Indicates Terminal End
† - Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 6)

SYSTEM - MAIN FEEDWATER SYSTEM INSIDE CONTAINMENT Prob. No. P-003A
PIPE BREAK ISOMETRIC NO.: Issue - 8
Figure 3.6-1, Sheet 2 (AE04)

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) $0.8 (S_A + 1.2S_h)$
5*	†	†	†	†
20E	Break deleted per Reference 24			
20M	Break deleted per Reference 24			
27B Bend	Break deleted per Reference 24			
27M	Break deleted per Reference 24			
35M	Break deleted per Reference 24			
35E	Break deleted per Reference 24			
75M	Break deleted per Reference 24			
95M	Break deleted per Reference 24			
95E	Break deleted per Reference 24			
100	Break deleted per Reference 24			
125*	†	†	†	†

* - Indicates Terminal End
† - Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 7)

SYSTEM - MAIN FEEDWATER SYSTEM Prob. No. P-004
PIPE BREAK ISOMETRIC NO.: Issue - 5
Figure 3.6-1, Sheet 3 (AE05)

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (ksi) $0.8 (S_A + 1.2S_h)$
10*	†	†	†	†
20B Bend	Break deleted per Reference 24			
20M	Break deleted per Reference 24			
30M	Break deleted per Reference 24			
30E	Break deleted per Reference 24			
45B Bend	Break deleted per Reference 24			
45M	Break deleted per Reference 24			
71M	Break deleted per Reference 24			
71E	Break deleted per Reference 24			
90E	Break deleted per Reference 24			
95	Break deleted per Reference 24			
100*	†	†	†	†

* - Indicates Terminal End
† - Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 8)

SYSTEM - MAIN FEEDWATER SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 3 (AE05)

Prob. No. P-004A
Issue - 7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) 0.8 (S_A + 1.2S_h)
10*	†	†	†	†
20B Bend	Break deleted per Reference 24			
20M	Break deleted per Reference 24			
30M	Break deleted per Reference 24			
30E	Break deleted per Reference 24			
45B Bend	Break deleted per Reference 24			
45M	Break deleted per Reference 24			
71M	Break deleted per Reference 24			
71E	Break deleted per Reference 24			
90E	Break deleted per Reference 24			
95	Break deleted per Reference 24			
100*	†	†	†	†

* - Indicates Terminal End
† - Break as required by MEB 3-1

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 9)

SYSTEM - HIGH PRESSURE COOLANT INJECTION	Prob. No. P-21
PIPE BREAK ISOMETRIC NO.:	Issue - 5
Figure 3.6-1, Sheet 37 (EM02)

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
5 TNGT	9,870	8,286	18,156	39,448
30 TNGT	6,465	15,943	22,408	39,448
50* TNGT	6,905	2,214	9,119	39,448
164 TNGT	10,870	1,806	12,676	39,448
180* TNGT	7,065	2,669	9,734	39,448
67	8,481	2,353	10,834	39,448
100M Bend	7,440	2,849	10,289	39,448
116*	8,440	1,732	10,172	39,448
255E	5,421	6,884	12,305	39,448
266 TNGT	13,353	1,280	14,633	39,448
320B Bend	3,975	24,019	27,994	39,448
340*	5,264	2,834	8,098	39,448

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 10)

SYSTEM - MAIN STEAM-AUXILIARY BUILDING
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 1 (AB01)

Prob. No. P-026
Issue -7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
5	9,170	16,814	25,984	37,800
25	7,063	3,446	10,509	37,800
33	8,902	9,195	18,097	37,800
45F	17,924	0	17,924	38,700
60	12,519	2,299	14,818	37,800
83	6,924	2,309	9,233	37,800
300	8,922	18,909	27,831	37,800
294	8,722	9,636	18,408	37,800
291	9,004	5,890	14,894	37,800
289	8,768	6,571	15,339	37,800
287	11,351	1,228	12,579	37,800
282	9,313	978	10,291	37,800

Note: This problem meets no break zone criteria.

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 11)

SYSTEM - MAIN STEAM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 1 (AB01)

Prob. No. P-27BY
Issue - 7

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
100	16,223	12,234	28,457	32,400
105	11,667	8,978	20,645	32,400
106	12,783	10,526	23,309	32,400
160	15,292	10,316	25,608	32,400
170	15,535	9,142	24,677	32,400
185	14,959	5,561	20,520	32,400
200	11,764	14,004	25,768	32,400
202	9,127	6,284	15,411	32,400
210	9,000	15,582	24,582	32,400
215	10,651	20,208	30,859	32,400
145	16,002	14,908	30,910	32,400
142	12,292	6,952	19,244	32,400
190	9,239	20,320	29,559	32,400
205	9,322	17,696	27,018	32,400

Note: This problem meets no break zone criteria.

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 12)

SYSTEM - MAIN FEEDWATER SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 2 (AE04)

Prob. No. P-028
Issue - 8

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 ($S_A + 1.2S_h$)
675	10,325	2,511	12,836	32,400
720	10,658	5,742	16,400	32,400
775	9,992	9,295	19,287	32,400
820	10,239	5,684	15,923	32,400
575	9,941	7,507	17,448	32,400
620	10,007	6,673	16,680	32,400
875	10,415	3,109	13,524	32,400
920	10,028	6,765	16,793	32,400

Note: This problem meets no break zone criteria.

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 13)

SYSTEM - CVCS - LETDOWN TO REHEAT HEAT EXCHANGER Prob. No. P-29B1
PIPE BREAK ISOMETRIC NO.: Issue - 5
Figure 3.6-1, Sheet 23 (BG11)

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
815*	4,568	4,363	8,931	37,712
840M Bend	6,486	19,775	26,261	37,712
860M Bend	7,607	14,689	22,296	37,712
980M Bend	6,722	11,961	18,683	37,712
878*	4,607	1,121	5,728	37,712
840B Bend	6,402	18,665	25,067	37,712
860B Bend	7,478	14,535	22,013	37,712

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 14)

SYSTEM - CVCS LETDOWN TO REHEAT
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 23 (BG11)

Prob. No. P-29B2
Issue - 4

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
716*	3,232	739	3,971	37,710
774B Bend	8,047	16,567	24,614	37,710
774M Bend	8,568	16,075	24,643	37,710
774E Bend	9,078	13,917	22,995	37,710
778E Bend	8,635	13,483	22,118	37,710
778E Bend	8,377	13,614	21,991	37,710
804M Bend	4,296	9,459	13,755	37,710
818*	4,269	1,444	5,713	37,710
752M Bend	5,935	14,275	20,210	37,710

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 15)

SYSTEM - CVCS LETDOWN FLOW - AUX BLDG
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 23 (BG11)
Sheet 20 (BG03), Sheet 25 (BG22)

Prob. No. P-29B3
Issue - 6

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
450*	4,860	2,692	7,553	37,685
415M Bend	4,961	8,844	13,805	37,685
395	11,182	6,518	17,700	37,685
390	13,332	16,192	29,524	37,685
385*	15,129	13,726	28,855	37,685
705E Bend	5,497	19,064	24,501	37,685
507	11,814	7,909	19,723	37,685
515	12,289	11,476	23,764	37,685
485	12,871	13,855	26,727	37,685
415	4,961	8,844	13,805	37,685

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 16)

SYSTEM - CVCS LETDOWN TO REHEAT BLDG | Prob. No. P-29B3
PIPE BREAK ISOMETRIC NO.: | Issue - 6
Figure 3.6-1, Sheet 23 (BG11)
Sheet 20 (BG03), Sheet 25 (BG22)

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
818*	7,489	5,231	12,720	37,685
834M Bend	4,085	24,439	28,523	37,685
838B Bend	4,027	24,530	28,557	37,685
815*	6,347	2,843	9,190	37,685
790M Bend	4,089	21,269	25,357	37,685
720B	5,567	17,335	22,902	37,685
868	6,653	13,257	19,910	37,685

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 17)

SYSTEM - CHEMICAL AND VOLUME CONTROL
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet (BG09)

Prob. No. P-31
Issue - 7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
780	12,017	3,373	15,390	39,657
785*	7,297	3,256	10,553	39,657
805	8,187	4,923	13,110	39,657
810M Bend	8,024	7,057	15,081	39,657
815	7,414	6,423	13,837	39,657
874M Bend	6,676	7,385	14,061	39,657
875T	8,333	748	9,081	39,657
873M Bend	6,395	5,078	11,473	39,657
903	6,688	906	7,594	39,657
906*	6,825	1,108	7,933	39,657
891*	6,451	2,228	8,679	39,657
995*	6,653	45	6,698	39,657
932*	6,712	69	6,781	39,657

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 18)

SYSTEM - CHEMICAL AND VOLUME CONTROL — Prob. No. P-33
PIPE BREAK ISOMETRIC NO.: — Issue - 7
Figure 3.6-1, Sheet 21 (BG09)

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
140*	4,876	9,159	14,035	39,680
130 TNGT	5,265	12,715	17,980	39,680
95 TNGT	10,483	3,872	14,355	39,680
385M Bend	9,739	7,415	17,154	39,680
85T	11,178	5,113	16,291	39,680
465	8,537	3,706	12,243	39,680
425**	12,930	9,416	22,346	39,680
505**	12,091	12,432	24,523	39,680
580**	14,075	11,878	25,953	39,680
25T**	11,131	17,068	28,199	39,680

* - Indicates Terminal End

** - Meets No Break Zone Criteria

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 19)

SYSTEM - CHEMICAL AND VOLUME CONTROL
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 21 (BG09)

Prob. No. P-33A
Issue - 6

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
140*	3,367	619	3,986	39,448
165M Bend	5,056	13,586	18,642	39,448
240	5,328	28,213	33,541	39,448
305*	5,813	4,716	10,529	39,448
327M Bend	4,377	13,334	17,711	39,448
380*	5,709	931	6,640	39,448
235	5,559	20,010	25,569	39,448

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 20)

SYSTEM - CVCS LETDOWN - AUX BLDG
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 20 (BG03)

Prob. No. P-36
Issue -6

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
80*	4,394	1,730	6,124	40,376
163	6,929	1,850	8,779	40,376
175T	7,826	1,264	9,089	40,376
180M Bend	8,684	567	9,251	40,376
190*	6,946	1,002	7,948	40,376
255*	13,840	2,304	16,144	40,376
225	6,976	943	7,918	40,376

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 21)

SYSTEM - AUXILIARY FEEDWATER

Prob. No. P-43
Issue - N/A

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$

DELETED

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 22)

SYSTEM - TURBINE DRIVEN AUXILIARY FEEDWATER PUMP
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 49 (FC01),
Figure 3.6-1, Sheet 1 (AB01)

Prob. No. P-060
Issue - 10

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
15*	4,140	2,006	6,146	32,400
30M Bend	4,433	2,839	7,272	32,400
35E Bend	5,288	3,945	9,233	32,400
35M Bend	4,968	3,732	8,700	32,400
48T	7,820	10,955	18,775	32,400
50*	8,759	15,837	24,596	32,400
215T**	4,374	4,294	8,668	32,400
260**	6,260	8,444	14,704	32,400
275**	6,295	12,807	17,102	32,400
285**	5,861	16,742	22,603	32,400
410E** Bend	4,265	9,537	14,162	32,400
410M** Bend	4,208	9,496	13,704	32,400

* - Indicates Terminal End

** - Meets No Break Zone Criteria

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 23)

SYSTEM - AUXILIARY FEEDWATER

Prob. No. P-068
Issue - 3

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$

DELETED

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 24)

SYSTEM - CHEMICAL AND VOLUME CONTROL SYSTEM | Prob. No. P-069
PIPE BREAK ISOMETRIC NO.: | Issue - 5
Figure 3.6-1, Sheet 19 (BG02), Sheet 22 (BG10),
Sheet 21 (BG09), Sheet 37 (EM02)

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
170*	3,824	16,258	20,082	39,610
93 TNGT	4,806	273	5,079	39,610
140*	5,252	2,086	7,338	39,610
955 E	5,283	5,649	10,932	39,610
900	5,606	2,438	7,932	39,610
870* TNGT	4,596	4,181	8,777	39,610
650*	4,746	10	4,756	39,610
310*	4,928	362	5,290	39,610
266	5,445	14,584	20,029	39,610
270 M Bend	4,679	14,656	19,335	39,610
730 TNGT	13,020	5,658	18,678	39,610
A75 TNGT	13,736	3,119	16,855	39,610
745 B Bend	4,927	1,325	6,252	39,610
155 M	4,025	9,259	13,284	39,610
50	5,912	8,091	14,003	39,610
970*	5,361	11,601	16,962	39,610

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 25)

SYSTEM - CHEMICAL AND VOLUME CONTROL SYSTEM Prob. No. P-069
PIPE BREAK ISOMETRIC NO.: Issue - 5
Figure 3.6-1, Sheet 19 (BG02), Sheet 22 (BG10),
Sheet 21 (BG09), Sheet 37 (EM02)

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
95F	13,338	22,072	33,410	39,610
TNGT				
95C	16,645	17,209	35,854	39,610
885M	5,173	3,685	8,858	39,610
620*	5,310	541	5,851	39,610
625	10,642	5,560	16,202	39,610
TNGT				
604	5,119	14,293	19,412	39,610
E				
601	5,615	17,256	22,871	39,610
TNGT				
641	12,488	2,385	14,873	39,610
626	10,120	5,459	15,579	39,610
B Bend				
574	11,868	1,861	13,729	39,610
M				
573*	8,461	1,409	9,870	39,610
TNGT				
545	10,951	10,004	20,955	39,610
TNGT				
62A*	7,741	1,667	9,408	39,610
75	9,631	7,767	17,398	39,610
91	14,254	7,365	21,619	39,610
A92	8,117	11,489	19,609	39,610
425B	4,506	11,901	16,407	39,610

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 26)

SYSTEM - CHEMICAL AND VOLUME CONTROL SYSTEM — Prob. No. P-069
PIPE BREAK ISOMETRIC NO.: — Issue - 5
Figure 3.6-1, Sheet 19 (BG02), Sheet 22 (BG10),
Sheet 21 (BG09), Sheet 37 (EM02)

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
C92B*	4,924	1,952	6,876	39,610
D92* TNGT	13,774	6,720	20,494	39,610
575	13,044	1,384	14,428	39,610

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 27)

SYSTEM - AUXILIARY FEEDWATER

Prob. No. P-70
Issue - 3

Node	Stress (ksi) Primary	Stress (ksi) Secondary	Stress (ksi) Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$

DELETED

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 28)

SYSTEM - AUXILIARY FEEDWATER

Prob. No. P-070
Issue - 3

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$

DELETED

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 29)

SYSTEM - CVCS MINIMUM CHGNG FLOW - AUX BLDG
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 18 (BG01),
Sheet 19 (BG02)

Prob. No. P-73A
Issue - 6

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
130	5,489	3,051	8,540	39,564
315*	5,415	2,124	7,539	39,564
5*	5,327	786	6,113	39,564
90*	8,432	1,567	9,999	39,564
290*	4,799	1,087	5,886	39,564

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 30)

SYSTEM - CVCS MINIMUM CHGNG FLOW - AUX BLDG
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 18 (BG01),
Sheet 21 (BG09)

Prob. No. P-73B
Issue - 7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
22* TNGT	9,762	506	10,268	39,564
27 TNGT	7,186	654	7,840	39,564
28 TNGT	8,006	5,505	13,511	39,564
48	6,423	6,406	12,829	39,564
74A**	10,028	2,180	12,208	39,564
86** M Bend	9,314	5,494	14,808	39,564
193**	15,745	6,730	22,475	39,564
995*	5,079	31	5,110	39,564
834	8,204	7,578	15,782	39,564
846 TNGT	9,650	5,852	15,502	39,564
68M	4,801	2,329	7,130	39,564
56M	4,410	6,788	11,198	39,564

* - Indicates Terminal End
** - Meets No Break Zone Criteria

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 31)

SYSTEM - AUXILIARY FEEDWATER

Prob. No. P-90
Issue - 3

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$

DELETED

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 32)

SYSTEM - CHEMICAL AND VOLUME CONTROL SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 25 (BG22),

Prob. No. P-119
Issue - 6

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
45BB	5,349	13,958	19,307	37,244
47M	5,017	19,116	24,133	37,244
49	18,238	15,476	34,395	37,244
60T	15,515	7,989	23,504	37,244
145M	9,021	19,464	28,485	37,244
160M	10,232	21,632	31,864	37,244
220M	4,066	18,211	22,277	37,244
245E	4,087	19,998	24,085	37,244
270*	3,980	3,553	7,533	37,244

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 33)

SYSTEM - CHEMICAL AND VOLUME CONTROL
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 24 (BG21), Sheet 27 (BG24)

Prob. No. P-139
Issue - 5

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
20*	10,062	4,510	14,572	37,240
90 TNGT	8,579	10,707	19,286	37,240
100*	9,888	9,415	19,303	37,240
240M Bend	8,047	15,979	24,026	37,240
297*	5,667	3,973	9,640	37,240
215 TNGT	11,695	9,709	21,404	37,240
225M Bend	9,151	4,170	13,321	37,240
250B Bend	7,073	12,527	19,600	37,240

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 34)

SYSTEM - CHEMICAL AND VOLUME CONTROL
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 24 (BG21), Sheet 27 (BG24)

Prob. No. P-139
Issue - 5

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
280M Bend	7,140	6,327	13,467	37,240
444*	8,452	17,122	25,574	37,240
440 Bend	7,123	11,789	18,912	37,240
405 Bend	8,583	6,581	15,164	37,240
400T	14,109	13,752	27,861	37,240

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 35)

SYSTEM - CVCS AUXILIARY SPRAY
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 27 (BG24)

Prob. No. P-140
Issue - 5

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
444T* TNGT	7,205	8,198	15,403	37,240
450M Bend	5,176	11,886	17,062	37,240
670 Bend	5,129	14,730	19,859	37,240
735M Bend	6,303	15,643	21,946	37,240
770E	4,961	2,141	7,102	37,240
771*	8,247	12,883	21,130	37,240
645	9,178	5,768	14,946	37,240
621	6,981	23,470	30,451	37,240
715A	7,972	20,565	28,537	37,240

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 36)

SYSTEM - CHEMICAL AND VOLUME CONTROL
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 25 (BG22)

Prob. No. P-145
Issue - 5

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
5* TNGT	9,781	333	10,114	37,200
77	7,599	1,377	8,976	37,200
25	13,788	19,220	33,008	37,200
40	17,437	20,592	38,029	37,200
45B Bend	7,419	5,020	12,439	37,200
105*	6,218	885	7,103	37,200
90	7,484	1,246	8,730	37,200
10	7,334	1,087	8,421	37,200

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 37)

SYSTEM - CHEMICAL AND VOLUME CONTROL
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 25 (BG22)

Prob. No. P-146
Issue - 5

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
5*	5,469	5,201	10,670	37,648
30T	7,194	23,858	31,052	37,648
35T	8,722	13,492	22,214	37,648
40T	8,741	10,360	19,101	37,648
44T	10,526	12,356	22,882	37,648
48	11,182	8,537	19,719	37,648
80T	12,578	6,878	19,456	37,648
102T	12,189	22,635	34,824	37,648
106	12,288	10,262	22,550	37,648
130T	15,138	6,132	21,270	37,648
202T	11,671	8,697	20,368	37,648
401*	15,986	22,849	38,835	37,648
315*	7,077	630	7,707	37,648

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 38)

SYSTEM - CVCS CHGNG AND EXCESS LETDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 26 (BG23)

Prob. No. P-147
Issue - 4

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
5*	5,987	754	6,741	40,240
150M Bend	7,001	4,015	11,016	40,240
150E Bend	6,379	4,240	10,619	40,240
157	8,399	4,473	12,872	40,240
160E Bend	6,617	4,062	10,679	40,240
250M** Bend	5,993	2,700	8,693	40,240
300**	10,369	3,212	13,581	40,240

* - Indicates Terminal End

** - Meets No Break Zone Criteria

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 39)

Prob. No. P-196(1)
Issue - 2
Prob. No. P-196(2)
Issue - 1

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 29 (BM01)

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
5 TNGT	6,814	5,314	12,128	32,400
35	5,529	330	5,859	32,400
50	5,468	206	5,674	32,400

Note: This problem meets no break zone criteria.

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 40)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 29 (BM01)

Prob. No. P-197(1)
Issue - 2
Prob. No. P-197(2)
Issue - 1

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
5T	6,020	8,200	14,220	32,400
15	5,799	7,290	13,089	32,400
20	6,796	3,440	10,236	32,400
35	5,394	472	5,866	32,400
50	5,323	306	5,629	32,400

Note: This problem meets no break zone criteria.

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 41)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 29 (BM01),
Sheet 35 (BM20), Sheet 31 (BM03)

Prob. No. P-219
Issue - 7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) $0.8\ (S_A + 1.2S_h)$
C40*	7.802	26.468	34.270	32.400
E48*	15.888	6.986	22.874	32.400
1776*	4.506	29.403	33.909	32.400
240*	3.614	0.383	3.997	32.400

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 42)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 29 (BM01),
Sheet 25 (BM20), Sheet 31 (BM03)

Prob. No. P-219, BM-5-002
Issue - 7

	Stress (ksi)			Pipe Break
Node	Primary	Secondary	Total	Stress Limit (ksi) $0.8 (S_A + 1.2S_h)$

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 43)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 29 (BM01),
Sheet 31 (BM03), Sheet 32 (BM17)

Prob. No. P-220, BM-S-003
Issue - 6

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (ksi) 0.8 (S_A + 1.2S_h)
146A*	11.153	27.451	38.604	32.400
470*	9.371	27.395	36.766	32.400
645A*	6.328	15.582	21.910	32.400
195*	6.471	1.921	8.392	32.400

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 44)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 30 (BM02),
Sheet 33 (BM18), Sheet 18 (BM03)

Prob. No. P-221
Issue - 7

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (ksi) $0.8 (S_A + 1.2S_h)$
105*	6.871	32.236	39.107	32.400
554*	16.952	17.774	34.726	32.400
850*	6.602	12.630	19.232	32.400
106†	6.841	31.656	38.497	32.400

* - Indicates Terminal End
† - Indicates Intermediate Break Point

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 45)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 20 (BM02)
Sheet 33 (BM18), Sheet 21 (BM03)

Prob. No. P-221
Issue - 7

Node	Stress (ksi) Primary	Secondary	Total	Pipe Break Stress Limit (ksi) $0.8\ (S_A + 1.2S_h)$

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 46)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 20 (BM02),
Sheet 33 (BM18), Sheet 21 (BM03)

Prob. No. P-221
Issue - 7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) $0.8\ (S_A + 1.2S_h)$
640*	8.409	1.733	10.142	32.400

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 47)

SYSTEM - STEAM GENERATOR BLOWDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 30 (BM02),
Sheet 31 (BM03), Sheet 34 (BM19)

Prob. No.
P-222,
BM-S-005
Issue - 7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (ksi) $0.8\,(S_A + 1.2S_h)$
C77A*	5.653	26.608	32.261	32.400
F50*	9.090	27.326	36.416	32.400
255*	14.692	4.872	19.564	32.400
F40†	5.895	27.026	32.921	32.400
590*	4.001	2.906	6.907	32.400

* - Indicates Terminal End
† - Indicates Intermediate Break Point

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 48)

SYSTEM - MAIN STEAM ATMOSPHERIC DUMP LINE
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 1 (AB01),

Prob. No. P-225
Issue - 6

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
520B Bend	7,816	19,035	26,851	32,400
545T	8,980	5,552	14,532	32,400
555B Bend	10,257	8,012	18,269	32,400
575	4	0	4	32,400
580T	7,945	10,808	18,753	32,400

Note: This problem meets no break zone criteria.

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 49)

SYSTEM - CHEMICAL AND VOLUME CONTROL
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 24 (BG21),

Prob. No. P-254A
Issue - 3

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
175*	7,033	3,674	10,707	37,244
170M Bend	7,523	3,240	10,763	37,244
170B Bend	7,214	2,869	10,083	37,244
155	6,784	6,413	13,197	37,244
140	6,907	13,563	20,470	37,244
130M Bend	5,388	21,301	26,689	37,244
105	6,526	9,440	15,966	37,244
95*	7,931	1,664	9,595	37,244

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 50)

SYSTEM - REACTOR COOLANT SYSTEM - REACTOR BLDG
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 13 (BB08),

Prob. No. P-276
Issue - 5

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + 1.2S_h)
40*	5,489	1,169	6,658	39,656
85**	6,745	12,128	18,873	39,656
125**	6,469	22,604	29,073	39,656
70	7,585	8,461	16,046	39,656
100**	8,657	6,634	15,291	39,656
50B	7,387	6,998	14,385	39,656
55E	7,389	6,983	14,372	39,656
75E	5,140	8,352	13,492	39,656
80B	5,899	6,720	12,619	39,656

* - Indicates Terminal End

** - Meets No Break Zone Criteria

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 51)

SYSTEM - REACTOR COOLANT SYSTEM
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 14 (BB09)

Prob. No. P-277
Issue - 7

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8\ (S_A + 1.2S_h)$
35*	5,702	239	5,941	39,610
70M** Bend	6,432	5,898	12,330	39,610
70E** Bend	6,343	5,520	11,863	39,610
75	7,694	2,933	10,627	39,610
47E Bend	6,357	772	7,129	39,610
105T**	6,168	826	6,994	39,610
65B	6,467	2,251	8,718	39,610
55E	5,786	2,750	8,536	39,610

* - Indicates Terminal End

** - Meets No Break Zone Criteria

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 52)

SYSTEM - REACTOR COOLANT SYSTEM — Prob. No. P-278
PIPE BREAK ISOMETRIC NO.: — Issue - 6
Figure 3.6-1, Sheet 15 (BB11),

Node	Primary	Stress (ksi) Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
20*	6,348	3,176	9,524	39,610
165	8,832	1,393	10,225	39,610
135	6,781	1,050	7,831	39,610

* - Indicates Terminal End

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 53)

SYSTEM - ACCUMULATOR SAFETY INJECTION (Loop 1)
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 40 (EP01), Sheet 51 (HB27)

Prob. No. P-234
Issue - N/A
See SNP-6566

Node No. West. Anal.	Node No. Orig. P-234	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9106	15	Terminal End Break, Break as required by MEB 3-1			
3055	30	16.8	40.9	0.24	40.4
3160	115	55.0	36.5	0.98	46.4
5000	450	Terminal End Break, Break as required by MEB 3-1			
5003	455	Deleted per Reference 24			
5070	485	53.6	47.6	0.33	46.4
5100	495	Terminal End Break, Break as required by MEB 3-1			
4100	665	Terminal End Break, Break as required by MEB 3-1			
3370	210	Terminal End Break, Break as required by MEB 3-1			
6500	955	Terminal End Break, Break as required by MEB 3-1			
6515	960	Deleted per Reference 24			
6525	975	Terminal End Break, Break as required by MEB 3-1			

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 54)

SYSTEM - ACCUMULATOR SAFETY INJECTION (Loop 4)
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 40 (EP01), Sheet 51 (HB27)

Prob. No. P-235
Issue - N/A
See SNP-6566

Node No. West. Anal.	Node No. Orig. P-235	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9406	15	Terminal End Break, Break as required by MEB 3-1			
3080	35	18.8	40.9	0.24	40.4
3590	65	42.2	35.0	0.98	46.4
4060	348	Terminal End Break, Break as required by MEB 3-1			
3850	860	Terminal End Break, Break as required by MEB 3-1			
4320	360	39.5	58.9	0.33	46.4
4180	365	Break deleted per Reference 24			
4210	720	Terminal End Break, Break as required by MEB 3-1			
3530	300	Terminal End Break, Break as required by MEB 3-1			
5780	405	Terminal End Break, Break as required by MEB 3-1			
5790	410	Break deleted per Reference 24			
5830	425	Break deleted per Reference 24			
5850	430	Terminal End Break, Break as required by MEB 3-1			

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 55)

SYSTEM - ACCUMULATOR SAFETY INJECTION (Loop 3)
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 41 (EP02), Sheet 51 (HB27)

Prob. No. P-236
Issue - N/A
See SNP-6566

Node No. West. Anal.	Node No. Orig. P-236	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9306	15	Terminal End Break, Break as required by MEB 3-1			
3050	35	24.4	40.9	0.24	40.4
3160	85	41.2	30.4	0.98	46.4
4030	525	Terminal End Break, Break as required by MEB 3-1			
4060	450	Terminal End Break, Break as required by MEB 3-1			
5012	535	Break deleted per Reference 24			
5040	550	55.3	59.5	0.33	46.4
5080	610	Terminal End Break, Break as required by MEB 3-1			
3380	205	Terminal End Break, Break as required by MEB 3-1			
3365	955	Terminal End Break, Break as required by MEB 3-1			
6515	960	Break deleted per Reference 24			
6530	972	Break deleted per Reference 24			
6545	975	Terminal End Break, Break as required by MEB 3-1			

See WCAP-9728 for stress and usage factor values..

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 56)

SYSTEM - ACCUMULATOR SAFETY INJECTION (Loop 2)
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 41 (EP02), Sheet 51 (HB27)

Prob. No. P-237
Issue - N/A
See SNP-6566

Node No. West. Anal.	Node No. Orig. P-237	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9206	12	Terminal End Break, Break as required by MEB 3-1			
3055	25	14.3	40.9	0.24	40.4
3160	110	45.2	42.5	0.98	46.4
4045	485	Terminal End Break, Break as required by MEB 3-1			
4060	445	Terminal End Break, Break as required by MEB 3-1			
5020	500	Break deleted per Reference 24			
5050	508	40.9	58.5	0.33	46.4
5070	570	Terminal End Break, Break as required by MEB 3-1			
3430	220	Terminal End Break, Break as required by MEB 3-1			
6500	905	Terminal End Break, Break as required by MEB 3-1			
6515	910	Break deleted per Reference 24			
6540	925	Break deleted per Reference 24			
6555	930	Terminal End Break, Break as required by MEB 3-1			

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 57)

SYSTEM - AUXILIARY PRESSURIZER SPRAY
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 9 (BB04), Sheet 27 (BG24)

Prob. No. P-242
Issue - N/A
See SNP-6566 (see Note 2)

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS ($2.4S_m$) KSI
615**	9.8	45.8	0.08	42.0
See Note 1				
771*	†	†	†	†

Note 1: Class 1 equations and allowables used although this break is on the Class 2 portion of the line.

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Note 2: FSAR numbering of nodes differs from numbering used in SNP-6566.

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 58)

SYSTEM - AUXILIARY PRESSURIZER SPRAY
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 9 (BB04), Sheet 27 (BG24)

Prob. No. P-242
Issue - N/A
See SNP-6566

Node No. .	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	ALLOWABLE STRESS $(2.4S_m)$ KSI
520	Terminal End Break, Break as required by MEB 3-1			
10	Terminal End Break, Break as required by MEB 3-1			
310	Terminal End Break, Break as required by MEB 3-1			
580**	7.0	51.1	0.72	42.0
270	Terminal End Break, Break as required by MEB 3-1			
270**	14.4	45.7	0.168	40.4
285**#	37.0	40.5	0.25	38.8
285 to 305	16.3	12.9	0.50	38.8
305**	37.0	18.3	0.40	38.8
600**	98	45.9	0.02	42.0

**Intermediate Break
Results envelop both elbow and reducer
***Results envelop both elbow and butt weld

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 59)

SYSTEM - PRESSURIZER RELIEF
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 8 (BB02)

Prob. No. P-243A&B
Issue - N/A
See SNP-6566 (see Note 1)

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS ($2.4S_m$) KSI
175*	†	†	†	†
170**	†	†	†	†
165	47.4	25.4	0.20	38.6
165 to				
160**	46.4	24.3	0.40	38.6
160 to				
150**	40.4	29.4	0.975	38.6
150 to				
145**	46.4	35.8	0.70	38.6
145*	†	†	†	†
285*	†	†	†	†
280**	†	†	†	†
275**	47.4	25.4	0.20	38.6
275 to				
270**	46.4	24.3	0.40	38.6
270 to				
260**	46.4	29.4	0.975	38.6
260 to				
255**	46.4	35.8	0.70	38.6
255*	†	†	†	†
5*	†	†	†	†
10**	†	†	†	†
15**	47.4	25.4	0.20	38.6
15 to 20**	46.4	24.3	0.40	38.6
20 to 30**	46.4	29.4	0.975	38.6

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 60)

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Note 1: FSAR numbering of nodes differs from numbering used in SNP-6566.

Changes due to Steam Generator Replacement:: The above break points have not changed, see WCAP-9728 for stresses and usage factor values..

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 61)

SYSTEM - PRESSURIZER RELIEF
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 8 (BB02)

Prob. No. P-243A&B
Issue - N/A
See SNP-6566

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS $(2.4S_m)$ KSI
30 to 35**	46.4	35.8	0.70	38.6
35*	†	†	†	†
450*	†	†	†	†
415**	29.1	38.8	0.85	38.6
395 to 375**	37.5	28.4	0.17	38.6
375 to 340**	46.8	39.7	0.97	38.6
340**	†	†	†	†
415 to 465**	37.5	28.4	0.17	38.6
465 to 500**	46.8	39.7	0.97	38.6
500*	†	†	†	†

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 62)

SYSTEM - CVCS EXCESS LETDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 26 (BG23), Sheet 50 (HB24)

Prob. No. P-244
Issue - N/A
See SNP-6566

Node No. West. Anal.	Node No. Orig. P-244	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
459	5	Terminal End Break, Break as required by MEB 3-1			
4060	20	44.6	28.3	0.191	40.4
4130 & 4140	30 40	Break deleted per Reference 24			
6030	200	Break deleted per Reference 24			
6050	205	Terminal End Break, Break as required by MEB 3-1			
159	415	Terminal End Break, Break as required by MEB 3-1			
3040	410	Break deleted per Reference 24			
3070	400	Terminal End Break, Break as required by MEB 3-1			
4000	15	24.1	45.7	0.099	40.4

** Intermediate Break

See WCAP-9728 for stress and usage factor values..

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 63)

SYSTEM - CVCS LETDOWN
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 25 (BG22), Sheet 50 (HB24)

Prob. No. P-245
Issue - N/A
See SNP-6566

NODE NO.	EQUATION 12 STRESS (PSI)	EQUATION 13 STRESS (PSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS ($2.4S_m$) PSI
5*	†	†	†	†
10**	Break deleted per Reference 24			
30**	42.1	43.5	0.95	40.4
50**	23.4	33.4	0.10	39.4
100**	Break deleted per Reference 24			
205*	†	†	†	†
440*	†	†	†	†
435**	Break deleted per Reference 24			
430*	†	†	†	†
195**	42.4	43.5	0.95	40.4
125*	†	†	†	†

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 64)

SYSTEM - HPCI TO COLD LEG

Prob. No. P-247
Issue -

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS (2.4Sm) KSI

DELETED

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 65)

SYSTEM - BIT LOOPS 1, 2, 3, and 4
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 38 (EM03)

Prob. No. P-247, 247A
Issue - N/A
See SNP-6566

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS ($2.4S_m$) KSI
305*†	†	†	†	†
310**	Break deleted per Reference 24			
320**	Break deleted per Reference 24			
325*	†	†	†	†
210*†	†	†	†	†
215**	Break deleted per Reference 24			
225**	Break deleted per Reference 24			
235*	†	†	†	†
105*†	†	†	†	†
120**	Break deleted per Reference 24			
125**	Break deleted per Reference 24			
130*	†	†	†	†
10*††	†	†	†	†
25**	Break deleted per Reference 24			
30**	Break deleted per Reference 24			
35*	†	†	†	†

*Terminal End Break
**Intermediate Break
† † - includes 3 x 1-1/2” reducer
†Break as Required by MEB 3-1

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 66)

SYSTEM - SI HOT LEG LOOPS 2 & 3
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 38 (EM03)

Prob. No. 248A
Issue - N/A
See SNP-6566

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS ($2.4S_m$) KSI
175*	†	†	†	†
170**	Break deleted per Reference 24			
165*	†	†	†	†
290*	†	†	†	†
285**	Break deleted per Reference 24			
280*	†	†	†	†

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values..

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 67)

SYSTEM - SEAL INJECTION (LOOP 4)
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 12 (BB07)

Prob. No. 249
Issue - N/A
See SNP-6566

Node No. West. Anal	Node No. Orig. P-249	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9460	10	Terminal End Break, Break as required by MEB 3-1			
4050	35	Break deleted per Reference 24			
3260	65	Break deleted per Reference 24			

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 68)

SYSTEM - RCP SEAL INJECTION (LOOP 1)
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 13 (BB08)

Prob. No. 250
Issue - N/A
See SNP-6566

Node No. West. Anal	Node No. Orig. P-250	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9160	7	Terminal End Break, Break as required by MEB 3-1			
3160	20	Break deleted per Reference 24			
3240	40	Break deleted per Reference 24			
3580	140	Terminal End Break, Break as required by MEB 3-1			

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 69)

SYSTEM - RCP SEAL INJECTION (LOOP 3) Prob. No. 251
PIPE BREAK ISOMETRIC NO.: Issue - N/A
Figure 3.6-1, Sheet 14 (BB09) See SNP-6566

Node No. West. Anal	Node No. Orig. P-251	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9360+	7	Terminal End Break, Break as required by MEB 3-1			
3070	15	Break deleted per Reference 24			
3100	25	Break deleted per Reference 24			
4140	130	Terminal End Break, Break as required by MEB 3-1			

+ - includes break at 2 x 1-1/2” reducer and socket welded flange

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 70)

SYSTEM - RCP SEAL INJECTION (LOOP 2)
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 15 (BB11)

Prob. No. 252
Issue - N/A
See SNP-6566

Node No. West. Anal	Node No. Orig. P-252	EQ. 12 Str. (KSI)	EQ. 13 Str. (KSI)	Cum. Usage Factor	Allowable Str. (2.4Sm) KSI
9260	7	Terminal End Break, Break as required by MEB 3-1			
3270	20	Break deleted per Reference 24			
3210	25	Break deleted per Reference 24			
3470	110	Terminal End Break, Break as required by MEB 3-1			

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 71)

SYSTEM - CVCS
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 24 (BG21)

Prob. No. 253
Issue - N/A
See SNP-6566

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS ($2.4S_m$) KSI
5*	†	†	†	†
10** & 20**	24.9	40.0	0.91	40.4
50**	18.8	40.0	0.91	40.4
30**	29.6	33.5	0.93	40.4
45**	29.6	33.5	0.93	40.4
60** See Note 1	18.8	40.0	0.91	40.4
105* See Note 2	†	†	†	†

Note 1: Class 1 equations and allowables used although this break is on the Class 2 portion of the line.

Note 2: Class 2 break

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 72)

SYSTEM - CVCS
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 24 (BG21)

Prob. No. 254
Issue - N/A
See SN\-6566

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS $(2.4S_m)$ KSI
10*	†	†	†	†
35** & 45**	28.2	40.0	0.91	40.4
55**	28.2	40.0	0.91	40.4
17**	9.2	31.8	0.90	40.4
52**	31.7	31.8	0.93	40.4
65** (Note 1)	28.2	40.0	0.91	40.4
95* (Note 2)	†	†	†	†

Note 1: Class 1 equations and allowables used although this break is on the Class 2 portion of the line.

Note 2: Class 2 break

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

<u>Changes due to Steam Generator Replacement</u>: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 73)

SYSTEM - RHR
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 36 (EJ04), Sheet 39 (EM05)

Prob. No. 255
Issue - N/A
See SNP-6566

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS $(2.4S_m)$ KSI
10*	†	†	†	†
15**	Break deleted per Reference 24			
20**	Break deleted per Reference 24			
195*	†	†	†	†
40*	†	†	†	†
200**	Break deleted per Reference 24			
205**	(previously analyzed as terminal end. Intermediate Break deleted per Reference 24)			
4080	32.14	46.46	0.661	38.88
4110**	13.28	45.56	0.01	38.88
4178*	†	†	†	†

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 74)

SYSTEM - RHR LOOP 4
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, Sheet 36 (EJ04), Sheet 39 (EM05)

Prob. No. 256
Issue - N/A
See SNP-6566

NODE NO.	EQUATION 12 STRESS (KSI)	EQUATION 13 STRESS (KSI)	CUM. USAGE FACTOR	ALLOWABLE STRESS ($2.4S_m$) KSI
10*	†	†	†	†
15**	Break deleted per Reference 24			
20**	Break deleted per Reference 24			
195*	†	†	†	†
45*	†	†	†	†
200**	Break deleted per Reference 24			
220*	†	†	†	†

*Terminal End Break
**Intermediate Break
†Break as Required by MEB 3-1

Changes due to Steam Generator Replacement: The above break points have not changed, see WCAP-9728 for stresses and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-3 (Sheet 75)

SYSTEM - REACTOR COOLANT SYSTEM PRIMARY LOOP
PIPE BREAK ISOMETRIC NO.:
Figure 3.6-1, (BB01)

Prob. No. 257
Issue - N/A
See SNP-6566

Node No. West. Anal.	Node No. Orig. P-257	Reason for Break
9417	10	Terminal End Break, Break as required by MEB 3-1
3530	150	Terminal End Break, Break as required by MEB 3-1

See WCAP-9728 for stress and usage factor values.

Calculated stress values in this Table are updated when break locations change. Cited Problem Numbers (Prob. No.) which may be reflected on plant hanger location drawings. Refer to original stress calculation numbers (Prob. No.) which may be superseded and referenced to current stress calculations.

TABLE 3.6-4 HIGH-ENERGY PIPE BREAK EFFECTS ANALYSIS RESULTS.

Room No. 1101 — Elev. 1974'-0" General Floor Area No. 1

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 44, 45

II. Effects Analysis

A. Room 1101; non-LOCA Breaks.

1. General: Breaks at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves) in lines FB-032-HBD-8" having an auxiliary steam supply source and FB-050-HBD-3" having a condensate return source. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: 8-inch and 3-inch auxiliary steam piping restrained per Figure 3.6-1, Sheets 44, 45 such that no whipping occurs.

4. Jet impingement: Auxiliary steam and condensate jets impact safety-related targets required for safe shutdown. Function of the essential targets is ensured.

5. Room pressurization: Breaks in the auxiliary steam supply header will result in peak local pressures greater than 0.2 psid; however, no structures, systems, or components required for safe shutdown will be adversely affected due to the short duration of the blowdown.

6. Temperature and humidity: Humidity is 100 percent following the breaks. The transient temperature is harsh and provides a limiting case for equipment qualification.

TABLE 3.6-4 (Sheet 2)

Room No. 1102 Elev. 1974'-0" Chiller and Surge Tank Area

I. Sheet of Figure 3.6-1 showing high-energy (H-E) piping in this room: 44, 45

II. Effects Analysis

A. Room 1102; non-LOCA Breaks.

1. General: Breaks at all intermediate fittings (e.g., elbows, tees, reducer, welded attachments, and elbows) as follows: FB-032-HBD-8" with auxiliary steam supply source and FB-050-HBD-3" with condensate return source. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Nonsafety-related auxiliary steam piping whips such that no safety-related items are impacted. Whip restraints are, therefore, not required.

4. Jet impingement: Jets do not impact any safety-related equipment in the area.

5. Room pressurization: Breaks in the auxiliary steam supply header will result in peak local pressures greater than 0.2 psid; however, no structures, systems, or components required for safe shutdown will be adversely affected due to the short duration of the blowdown.

6. Temperature and humidity: Humidity is 100 percent following the breaks. The transient temperature is harsh and provides a limiting case for equipment qualification.

TABLE 3.6-4 (Sheet 3)

Room No. 1104 Elev. 1974'-0" Letdown Reheat Heat Exchanger Room

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 23

II. Effects Analysis

A. Room 1104; non-LOCA Breaks.

1. General: Breaks BG11-07, 08 are non-LOCA breaks. BG11-07 has sources from CVCS letdown off Loop 3 and from letdown reheat HX. BG11-08 has letdown reheat HX source with no loop source because of check valve 7039. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Nothing in this room is required for safe shutdown. Therefore, pipes are unrestrained and free to whip.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Adequate vent area is provided to ensure the integrity of all structures, systems, and components.

6. Temperature and humidity: No equipment in this room is required for safe shutdown; however, these breaks result in limiting temperature and humidity conditions for equipment qualification.

TABLE 3.6-4 (Sheet 4)

Room No. 1105	Elev. 1974'-0" Auxiliary Heat Exchanger Valve Compartment

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 23

II. Effects Analysis

A. Room 1105; non-LOCA Breaks.

1. General: Break BG11-06 is a non-LOCA break having sources from CVCS letdown off Loop 3 and from letdown reheat HX. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Whip targets are uniquely associated with CVCS letdown flow path. Redundant letdown path available to ensure safe shutdown. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Adequate vent area is provided to ensure the integrity of all structures, systems, and components.

6. Temperature and humidity: No equipment in this room is required for safe shutdown; however, these breaks result in limiting temperature and humidity conditions for equipment qualification.

TABLE 3.6-4 (Sheet 5)

Room No. 1107	Elev. 1974'-0" ECCS Centrifugal Charging Pump Room B

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 19, 21, 22, 37

II. Effects Analysis

A. Room 1107; non-LOCA Breaks.

1. General: Breaks BG02-04, 12 have B-train CCP (PBG05B) source; no source from A-train CCP (PBG05A) because of check valve 8481B. Breaks BG09-31, 32, 33 have one source from CCP A/CCP B and no other source due to check valve BG-V589. Breaks BG02-13 and BG09-33 have sources from CCP B and from CCP A. Breaks BG10-04, 05 on miniflow line have CCP B source; downstream source is moderate energy. Break EM02-07 has CCP B/CCP A source and moderate energy source downstream. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe Whip: All equipment in this room is uniquely associated with ECCS CCP B. Redundant charging path is available through ECCS CCP A. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Cold water breaks only, P/T analysis not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 6)

Room No. 1114 Elev. 1974'-0" Centrifugal Charging Pump Room A

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 19, 21, 22

II. Effects Analysis

A. Room 1114; non-LOCA Breaks.

1. General: Breaks BG02-01 and BG09-35, 36 have A-train CCP (PBG05A) source; no source from B-train CCP (PBG05B) because of check valves 8481A and V590, respectively. Break BG02-18 has both CCP A and CCP B source. Breaks BG10-01, 03 on miniflow line have CCP A source; downstream source is moderate energy. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: All equipment in this room is uniquely associated with ECCS CCP A. Redundant charging path is available through ECCS CCP B. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Cold water breaks only, P/T analysis not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 7)

Room No. 1115	Elev. 1974'-0" Normal Charging Pump (NCP) Room

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 18, 19, 21

II. Effects Analysis

A. Room 1115; non-LOCA Breaks.

1. General: Break BG01-01 has the NCP source with no ECCS CCP A/B source because of check valves 8497 and V645. Break BG01-06 has a NCP source with a moderate energy source downstream of valve HV 8109. Breaks BG01-08, 09, 11, and 14 have both NCP and ECCS CCP A/B sources with no regenerative HX source because of check valve 8381. Breaks BG02-07 and 10 and the downstream break on BG09-34 have both NCP and ECCS CCP A/B sources. The upstream break on BG09-34 has a NCP source only. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: All equipment in this room is uniquely associated with NCP. Redundant charging path is available through either ECCS CCP path. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Cold water breaks only, P/T analysis not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 8)

Room No. 1116 Elev. 1974'-0" Boric Acid Tank Room A

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 44, 45

II. Effects Analysis

A. Room 1116; non-LOCA Breaks.

Room 1116 openly communicates with room 1117. The previously analyzed HELB in room 1117 from BG-212-HBD-2” is no longer required to be postulated because the steam supply to the room has been isolated by the closure of valve FBV0147. Therefore, no effects analysis is required for room 1116.

TABLE 3.6-4 (Sheet 9)

Room No. 1117	Elev. 1974'-0" Boric Acid Tank Room B
I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room	44, 45

II. Effects Analysis

A. Room 1117; non-LOCA Breaks.

The previously analyzed HELB in room 1117 from BG-212-HBD-2” is no longer required to be postulated because the steam supply to the room has been isolated by the closure of valve FBV0147. Therefore, no effects analysis is required for room 1116.

TABLE 3.6-4 (Sheet 10)

Room No. 1122 | Elev. 1974'-0" General Floor Area No. 3

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 37, 45

II. Effects Analysis

A. Room 1122; non-LOCA Breaks.

1. General: Breaks at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves) in lines FB-095-HBD-3" and FB-050-HBD-3" with condensate return source. No restrictions are considered in calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: No items required for safe shutdown are impacted. Whip restraints are, therefore, not required.

4. Jet impingement: An 8-inch ESW line to the auxiliary feedwater system, et al, is impacted. Function of this essential line is ensured.

5. Room pressurization: Breaks in condensate return lines will not pressurize the area.

6. Temperature and humidity: Humidity is 100 percent following the breaks. The transient temperature is harsh and provides a limiting case for equipment qualification.

TABLE 3.6-4 (Sheet 11)

Room No. 1124 Exchanger | Elev. 1974'-0" Letdown Heat Valve Compartment

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 20

II. Effects Analysis

A. Room No. 1124; non-LOCA Breaks.

1. General: Breaks BG03-01, 02 and the branch break on BG03-03 have a combined source from CVCS letdown/ letdown. Break BG03-12 and the upstream and downstream breaks on BG03-03 have one source from CVCS letdown and one limited source from the letdown HX. Breaks BG03-09, 10, 11, 13 and 16 have one source only - from CVCS letdown. The downstream source is moderate energy. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: All equipment in this room is uniquely associated with normal letdown. Redundant letdown is available for shutdown. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Breaks in the CVCS letdown lines will result in peak local pressures greater than 0.2 psid; however, no structures, systems, or components will be adversely affected due to the short duration of the blowdown.

6. Temperature and humidity: No safe shutdown equipment is in the area; however, these breaks result in limiting temperature and humidity conditions for equipment qualification.

TABLE 3.6-4 (Sheet 12)

Room No. 1125 Elev. 1974'-0" Letdown Heat Room Exchanger

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 20, 23

II. Effects Analysis

A. Room No. 1125; non-LOCA Breaks.

1. General: Break BG03-05 has one source from CVCS letdown and one limited source from the letdown HX. Break BG03-15 has one combined source from CVCS letdown/letdown HX. Break BG03-06 has one source from CVCS letdown. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: All equipment in this room is uniquely associated with normal letdown. Redundant letdown is available for shutdown. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Breaks in the CVCS letdown supplied lines will result in peak local pressures greater than 0.2 psid; however, no structures, systems, or components will be adversely affected due to the short duration of the blowdown.

6. Temperature and humidity: No safe shutdown equipment is in the area; however, these breaks do not result in limiting temperature and humidity conditions for equipment qualification.

TABLE 3.6-4 (Sheet 13)

Room No. 1126 Elev. 1974'-0" Boron Injection Room

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 37

II. Effects Analysis

A. Room 1126, non-LOCA Breaks.

1. General: Breaks EM02-06, 16 have CCP A source; EM02-05 has CCP B source. For all breaks, the Boron Injection source downstream is moderate energy. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks EM02-05 and 06 are restrained per Figure 3.6-1, Sheet 37, such that whipping is prevented.

4. Jet impingement: All equipment in this room is uniquely associated with Boron Injection and redundant means of boration exist for shutdown. Therefore, since any high-energy break in the room will flood all the essential Boron Injection equipment, jet impingement is not applicable.

5. Room pressurization: Cold water breaks only, P/T analysis not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 14)

Room No. 1127 Elev. 1974'-0" Stairwell A-2

I. Sheet of Figure 3.6-1 showing high-energy (H-E) piping in this room — 45

II. Effects Analysis

A. Room No. 1127; non-LOCA Breaks.

1. General: Breaks at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves) in lines FB-050-HBD-3" and FB-095-HBD-3" having source from condensate return. The calculation of thrust forces is not required, since these condensate lines are open to atmospheric pressure.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: See Item 1 above.

4. Jet impingement: See Item 1 above.

5. Room pressurization: Breaks in the condensate return lines will not pressurize the area.

6. Temperature and humidity: Condensate water in these lines will not adversely affect any safety-related equipment required for safe shutdown nor generate a harsh temperature or humidity environment.

TABLE 3.6-4 (Sheet 15)

Room No. <u>1128</u> Elev. 1974'-0" General Area No. 5

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 45

II. Effects Analysis

A. Room No. 1128; non-LOCA Breaks.

1. General: Breaks at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves) in lines FB-050-HBD-3" and FB-095-HBD-3" having source from condensate return. The calculation of thrust forces is not required, since these condensate lines are open to atmospheric pressure.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: See Item 1 above.

4. Jet impingement: See Item 1 above.

5. Room pressurization: Breaks in the condensate return lines will not pressurize the area.

6. Temperature and humidity: Condensate water in these lines will not adversely affect any safety-related equipment required for safe shutdown.

TABLE 3.6-4 (Sheet 16)

Room No. 1129	Elev. 1974'-0" Auxiliary Steam Condensate Recovery and Storage Tank Room

I. Sheet of Figure 3.6-1 showing high-energy (H-E) piping in this room: 43, 45, 46, 47, 48

II. Effects Analysis

A. Room No. 1129; non-LOCA Breaks.

1. General: Breaks at all terminal ends, including piping, pressure vessel, or equipment nozzle intersections, as follows: line FB-110-HBD-2", lines FB-050, 095, 116-HBD-3", line FB-078-HBD-4", and lines FB-051, 052, 053-HBD-6" have condensate sources. Lines FB-001, 054, 055-HBD-4" and lines FB-056, 057-HBD-2" have source from auxiliary steam deaerator feed pumps. The calculation of thrust forces is not required, since these lines carry condensate at low or atmospheric pressure.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: See Item 1 above.

4. Jet impingement: See Item 1 above.

5. Room pressurization: Breaks in the condensate return lines will not pressurize the area.

6. Temperature and humidity: Condensate water in these lines will not adversely affect any safety-related equipment required for safe shutdown.

TABLE 3.6-4 (Sheet 17)

Room No. 1130 Elev. 1974'-0" North Corridor

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 44, 45

II. Effects Analysis

A. Room 1130; non-LOCA Breaks.

1. General: Breaks at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves) as follows: FB-032-HBD-8" with auxiliary steam supply source, FB-095-HBD-3" and FB-050-HBD-3" with condensate return source. No restrictions are used in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: No essential equipment is impacted. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required for safe shutdown.

5. Room pressurization: Breaks in the auxiliary steam supply header will result in peak local pressures greater than 0.2 psid; however, no structures, systems, or components required for safe shutdown will be adversely affected, due to the short duration of the blowdown.

6. Temperature and humidity: Humidity is 100 percent following the breaks. The transient temperature is harsh and provides a limiting case for equipment qualification.

TABLE 3.6-4 (Sheet 18)

High-energy lines which were formerly in Room 1201 (Sheet 17) have been declassified to moderate energy.

TABLE 3.6-4 (Sheet 19)

Room No. 1202

Elev. 1988'-0"
Area: Access Area & Chiller Surge Tank Area

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 44

II. Effects Analysis

A. Room 1202; non-LOCA Breaks,

1. General: Break points at every fitting, valve, welded attachment, and terminal end for FB-093-HBD-3”, with Auxiliary Steam supply source.

2. Criteria: The non-LOCA break criteria have been met. (See Note C)

3. Pipe Whip: Non-safety related auxiliary steam piping whips such that no safety-related items are impacted. Whip restraints are, therefore, not required.

4. Jet Impingement: Jets do not impact any safety-related equipment in the area.

5. Room Pressuriztion: Breaks in the auxiliary steam line will result in peak local pressures equal to 0.02 psid: however, no structures, systems, or coponents required for safe shutdown will be adversely affected due to the short duration of the blowdown.

6. Temperature and Humidity: Humidity is 100 percent following the breaks. The transient temperature is harsh and provides a limiting case for equipment qualification.

TABLE 3.6-4 (Sheet 20)

Room No. 1203 Elev. 1988'-0" Pipe Space B

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 19, 20, 21, 22, 23

II. Effects Analysis

A. Room No. 1203; non-LOCA Breaks.

1. General: Breaks BG02-14, 15 have a CCP B source and a CCP A source. Breaks BG09-06, 07, 08, 19, 20, 23, 29, and 30; Break BG09-38, which is a Callaway only break; and Breaks BG09-41 and 42, which are Wolf Creek only breaks; have a charging pumps source only, since check valves BB-V118, V148, V178, and V208 are between the breaks and downstream source. Breaks BG11-02, 03, 04, 05, and 13 have a CVCS letdown from Loop 3 source and a limited source from the letdown reheat heat exchanger. Breaks BG09-01, 02, 12, 13 have CCP A and CCP B sources. Breaks BG11-09, 10, 11, 12 have a CVCS letdown from Loop 3 source only. There is no source in the opposite direction due to closed valve TCV-381A. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks BG02-14, 15; BG09-01, 02, 12, 13; and BG11-02, 03, 04, 10, 11, 12, 13 and the downstream break on BG11-09 are restrained per Figure 3.6-1, Sheets 21 and 23, such that no essential equipment is impacted.

4. Jet impingement: Two CVCS lines to the seal water injection filters, a CVCS CCP charging line, a CVCS CCP miniflow line, an RHR heat exchanger discharge line, and an RHR SI suction line are impacted by jets. Function of all these essential lines is ensured.

5. Room pressurization: Breaks in the CVCS letdown line will result in pressures greater than 0.2 psid. However, no safe shutdown equipment will be adversely affected due to the short duration of the blowdown.

TABLE 3.6-4 (Sheet 21)

6. Temperature and humidity: Humidity is 100 percent following the breaks. The transient temperature is harsh and provides a limiting case for equipment qualification.

TABLE 3.6-4 (Sheet 22)

Room No. 1204 Elev. 1988'-0" Pipe Space A

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 18, 19, 21, 37

II. Effects Analysis

A. Room No. 1204; non-LOCA Breaks.

1. General: Break BG02-16 is located at a 3-inch tee. The sources for the three break points are as follows: upstream - CCP B and CCP A, Branch (B) - CCP A and CCP B, downstream - CCP A/CCP B. Break BG02-17 has one CCP A/CCP B source. Break BG09-11 has two CCP A/CCP B combined sources. Breaks EM02-08, 09, 10, and 11 have a CCP B source with a moderate-energy source downstream. Break BG09-40 is a Wolf Creek only break with a combined CCP A/CCP source from both directions. Breaks EM02-12, 13, 14, and 15 have a CCP A source with a moderate-energy source downstream. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: All three breaks on BG02-16, the upstream break on BG02-17, and break BG09-11 are restrained per Figure 3.6-1, Sheets 19, 21, and 23, such that no essential equipment is impacted.

4. Jet impingement: A CVCS CCP charging line, a CVCS CCP miniflow line, an ESW room cooler return line, and a CCW room cooler return line are impacted by jets. Function of all these essential lines is ensured.

5. Room pressurization: Cold water breaks only, P/T analysis not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 23)

Room No. 1207 Elev. 1989'-0" Pipe Chase

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 43, 46, 47

II. Effects Analysis

A. Room No. 1207; non-LOCA Breaks.

All previously analyzed HELBs in room 1207 are no longer required to be postulated due to the re-evaluation of the associated piping. Therefore, no effects analysis is required for room 1207.

TABLE 3.6-4 (Sheet 24)

Room No. 1301 Elev. 2000'-0" Corridor No. 1

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 44, 45

II. Effects Analysis

A. Room No. 1301; non-LOCA Breaks.

1. General: Breaks at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves) as follows: FB-032-HBD-8" has an auxiliary steam supply source, and FB-095-HBD-3" has a condensate return source. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The 8-inch auxiliary steam piping whips into non-safety-related equipment. Whip restraints are, therefore, not required.

4. Jet impingement: No essential equipment is impacted by jets.

5. Room pressurization: Breaks in the auxiliary steam supply header will result in peak local pressures greater than 0.2 psid; however, no structures, systems, or components required for safe shutdown will be adversely affected due to the short duration of the blowdown.

6. Temperature and humidity: Humidity is 100 percent following the breaks. The transient temperature is harsh and provides a limiting case for equipment qualification.

TABLE 3.6-4 (Sheet 25)

Room No. 1302 Elev. 2000'-0" Filter Compartments - (5)

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 21

II. Effects Analysis

A. Room No. 1302; non-LOCA Breaks.

1. General: Breaks BG09-09, 10, 14, and 15 have a charging pump source only since check valves BB-V118, V148, V178, and V208 are between the breaks and the downstream source. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: All equipment in each compartment is uniquely associated with the seal water filters and a redundant path through CCW is available to the seals. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required to ensure safe shutdown.

5. Room pressurization: Cold water breaks only, P/T analysis not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 26)

Room No. 1304 Elev. 2013'-6" Auxiliary Feedwater Pipe Chase

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room NA

II. Effects Analysis

Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

1. General: No breaks are postulated in this room as noted above.

7. Criteria: N/A

2. Pipe whip: N/A

3. Jet impingement: N/A

4. Room pressurization: The subcompartment pressurization analysis for the auxiliary feedwater valve compartments and pipe chases is based on a maximum break size of 3/4-inch nozzle on a back-up gas accumulator tank pressurized with N_2 gas.

   The results of the analysis indicate that the existing vent area is adequate to limit the room pressure to the design value of 1.5 psig.

5. Temperature and humidity: No extreme temperature or humidity environments are experienced as a result of the back-up gas accumulator tank nozzle break.

TABLE 3.6-4 (Sheet 27)

Room No. 1305 Elev. 2013'-6" Auxiliary Feedwater Pipe Chase

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room NA

II. Effects Analysis

Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

1. General: No breaks are postulated in this room as noted above.

2. Criteria: N/A

3. Pipe whip: N/A

4. Jet impingement: N/A

5. Room pressurization: The subcompartment pressurization analysis for the auxiliary feedwater valve compartments and pipe chases is based on a maximum break size of 3/4-inch nozzle on a back-up gas accumulator tank pressurized with N_2 gas.

   The results of the analysis indicate that the existing vent area is adequate to limit the room pressure to the design value of 1.5 psig.

6. Temperature and humidity: No extreme temperature or humidity environments are experienced as a result of the back-up gas accumulator tank nozzle break.

TABLE 3.6-4 (Sheet 28)

Room No. 1306 | Elev. 2000'-0" Filter Valve Compartments - (5)

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 21

II. Effects Analysis

A. Room No. 1306; non-LOCA Breaks.

1. General: Break BG09-39 is a Callaway only break and has a charging pump source only since check valves BB-V118, V148, V178, and V208 are located between the break and the downstream source. No restrictions are considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: All equipment in each compartment is uniquely associated with the seal water filters and a redundant path through CCW is available to the seals. Whip restraints are, therefore, not required.

4. Jet impingement: No jet targets are required to ensure safe shutdown.

5. Room pressurization: Cold water breaks only, P/T analysis not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 29)

Room No. 1321 Elev. 2000'-0"
Area: Vestibule

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 44

II. Effects Analysis

A. Room No. 1321; non-LOCA Breaks.

The previously analyzed HELB in room 1321 from FB-093-HBD-3" is no longer required to be postulated because the steam supply to the room has been isolated by the closure of valve FBV0146. Therefore, no effects analysis is required for room 1321.

TABLE 3.6-4 (Sheet 30)

Room No. 1322 (No Break Zone) - Elev. 2000'-0"
Pipe Penetration Room B

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 20, 21, 14, 12, 13, 15

II. Effects Analysis

A. Room No. 1322; No Postulated Breaks.

1. General: This area is a designated no break zone. (See Section 3.6.2.1.1e)

2. Criteria: NA

3. Pipe whip: None, no postulated breaks.

4. Jet impingement: Analysis is not applicable in "no break zone."

5. Room pressurization: No breaks, cold water cracks only, therefore P/T analysis is not applicable.

6. Temperature and humidity: See Section 5 above.

TABLE 3.6-4 (Sheet 31)

Room No. 1323 (No Break Zone) Elev. 2000'-0" Pipe Penetration Room A

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 18, 37

II. Effects Analysis

A. Room No. 1323; No Postulated Breaks.

1. General: This area is a designated no break zone. (See Section 3.6.2.1.1e)

2. Criteria: NA

3. Pipe whip: None, no postulated breaks.

4. Jet impingement: Analysis is not applicable in "no break zone."

5. Room pressurization: No breaks, cold water cracks, P/T analysis is not applicable.

6. Temperature and humidity: See 5 above.

TABLE 3.6-4 (Sheet 32)

Room No. 1324 — Elev. 2000'-0" Auxiliary Feedwater Pumps Valve Compartment No. 1

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — NA

II. Effects Analysis

Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

1. General: No breaks are postulated in this room as noted above.

2. Criteria: N/A

3. Pipe whip: N/A

4. Jet impingement: N/A

5. Room pressurization: The subcompartment pressurization analysis for the auxiliary feedwater valve compartments and pipe chases is based on a maximum break size of 3/4-inch nozzle on a back-up gas accumulator tank pressurized with N_2 gas.

   The results of the analysis indicate that the existing vent area is adequate to limit the room pressure to the design value of 1.5 psig.

6. Temperature and humidity: No extreme temperature or humidity environments are experienced as a result of the back-up gas accumulator tank nozzle break.

TABLE 3.6-4 (Sheet 33)

Room No. 1325 Elev. 2000'-0" Auxiliary Feedwater Pump Room B

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: NA

II. Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

TABLE 3.6-4 (Sheet 34)

Room No. 1326 — Elev. 2000'-0" Auxiliary Feedwater Pump Room A

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — NA

II. Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

TABLE 3.6-4 (Sheet 35)

Room No. 1327 — Elev. 2000'-0" Auxiliary Feedwater Pump Valve Component No. 2

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: NA

II. Effects Analysis

Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

1. General: No breaks are postulated in this room as noted above.

2. Criteria: N/A

3. Pipe whip: N/A

4. Jet impingement: N/A

5. Room pressurization: The subcompartment pressurization analysis for the auxiliary feedwater valve compartments and pipe chases is based on a maximum break size of 3/4-inch nozzle on a back-up gas accumulator tank pressurized with N_2 gas.

   The results of the analysis indicate that the existing vent area is adequate to limit the room pressure to the design value of 1.5 psig.

6. Temperature and humidity: No extreme temperature or humidity environments are experienced as a result of the back-up gas accumulator tank nozzle break.

TABLE 3.6-4 (Sheet 36)

Room No. 1328 | Elev. 2000'-0" Auxiliary Feedwater Pump Valve Compartment No. 3

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: NA

II. Effects Analysis

Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

1. General: No breaks are postulated in this room as noted above.

2. Criteria: N/A

3. Pipe whip: N/A

4. Jet impingement: N/A

5. Room pressurization: The subcompartment pressurization analysis for the auxiliary feedwater valve compartments and pipe chases is based on a maximum break size of 3/4-inch nozzle on a back-up gas accumulator tank pressurized with N_2 gas.

   The results of the analysis indicate that the existing vent area is adequate to limit the room pressure to the design value of 1.5 psig.

6. Temperature and humidity: No extreme temperature or humidity environments are experienced as a result of the back-up gas accumulator tank nozzle break.

TABLE 3.6-4 (Sheet 37)

Room No. 1329 Elev. 2000'-0" Vestibule

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 43

II. Effects Analysis

A. Room 1329; Non-LOCA Breaks.

The previously analyzed HELB in room 1329 from FB-001-HBD-4” is no longer required to be postulated due to re-evaluation of the associated piping. Therefore, no effects analysis is required for room 1329.

TABLE 3.6-4 (Sheet 38)

Room No. 1330 | Elev. 2000'-0" Auxiliary Feedwater Pump Valve Compartment No. 4

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: NA

II. Effects Analysis

Safety-related piping in this room is associated with portions of the auxiliary feedwater system which experience high-energy conditions less than 1 percent of the plant operation time. This piping is therefore considered moderate energy per Section 3.6.1.1a, and high-energy line breaks are not applicable.

1. General: No breaks are postulated in this room as noted above.

2. Criteria: N/A

3. Pipe whip: N/A

4. Jet impingement: N/A

5. Room pressurization: The subcompartment pressurization analysis for the auxiliary feedwater valve compartments and pipe chases is based on a maximum break size of 3/4-inch nozzle on a back-up gas accumulator tank pressurized with N_2 gas.

   The results of the analysis indicate that the existing vent area is adequate to limit the room pressure to the design value of 1.5 psig.

6. Temperature and humidity: No extreme temperature or humidity environments are experienced as a result of the back-up gas accumulator tank nozzle break.

TABLE 3.6-4 (Sheet 39)

Room No. 1331 | Elev. 2000'-0" Auxiliary Feedwater Pump Room C

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 49, 46

II. Effects Analysis

A. Room 1331; Non-LOCA Breaks.

1. General: Breaks FC01-01, 02, 09, and 10 have main steam supply to turbine AFP source; source from auxiliary steam supply is considered moderate energy. Breaks at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves) in FB-078-HBD-4" have condensate return source. No restrictions are considered in the calculation of thrust forces for the FC breaks. No thrust force calculations are required on the FB line since it is at atmospheric pressure.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks FC01-02, 09 and the upstream break on FC01-10 are restrained per Figure 3.6-1, Sheet 49, such that whipping is prevented.

4. Jet impingement: No jet targets are required to ensure safe shutdown.

5. Room pressurization: See Appendix 3B, Section 3.B.4.1.

6. Temperature and humidity: See Appendix 3B, Section 3.B.4.1.

TABLE 3.6-4 (Sheet 40)

Room No. 1407 Elev. 2026'-0" Boric Acid Batching Tank

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 45

II. Effects Analysis

A. Room 1407; Non-LOCA Breaks.

Room 1407 openly communicates with room 1117. The previously analyzed HELB in room 1117 from BG-212-HBD-2" is no longer required to be postulated because the steam supply to the room has been isolated by the closure of valve FBV0147. Therefore, no effects analysis is required for room 1407.

TABLE 3.6-4 (Sheet 41)

Room No. <u>1411</u>	(No Break Zone) - Elevation 2026'-0" Main Steam/Main Feedwater Isolation Valve Compartment

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 1, 2, 3, 29, 30

II. Effects Analysis

A. Room 1411; No Postulated Breaks.

1. General: This area is a designated no break zone. (See Section 3.6.2.1.1e)

2. Criteria: NA

3. Pipe whip: There is no pipe whip because there are no postulated breaks in the no break zone.

4. Jet impingement: See 3 above.

5. Room pressurization: See Appendix 3B, Section 3B.4.2.

6. Temperature and humidity: See Appendix 3B, Section 3B.4.2.

TABLE 3.6-4 (Sheet 42)

Room No. 1412	(No Break Zone) - Elev. 2026'-0" Main Steam/Main Feedwater Isolation Valve Compartment

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 1, 2, 3, 29, 30, 49

II. Effects Analysis

A. Room 1412; No Postulated Breaks.

1. General: This area is a designated no break zone. (See Section 3.6.2.1.1e)

2. Criteria: NA

3. Pipe whip: There is no pipe whip because there are no postulated breaks in the no break zone.

4. Jet impingement: See 3 above.

5. Room pressurization: See Appendix 3B, Section 3B.4.2.

6. Temperature and humidity: See Appendix 3B, Section 3B.4.2.

TABLE 3.6-4 (Sheet 43)

Room No. 2000	Main Steam
I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room	1

II. Effects Analysis

A. Problem No. 001, Steam Generator A, Secondary Systems Breaks.

1. General: Break AB01-01 has sources from steam generator A and turbine building. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 1, such that no whipping occurs.

4. Jet impingement: The jets from these breaks do not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

7. Flooding: See Section 6.3.2.2

B. Problem No. 001A, Steam Generator B, Secondary Systems Breaks.

1. General: Break AB01-05 has sources from steam generator B and turbine building. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 1, such that no whipping occurs.

TABLE 3.6-4 (Sheet 44)

4. Jet impingement: The jets from these breaks do not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 002, Steam Generator D, Secondary Systems Breaks.

1. General: Break AB01-13 has sources from steam generator D and turbine building. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 1, such that no whipping occurs.

4. Jet impingement: The jets from these breaks do not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 002A, Steam Generator C, Secondary Systems Breaks.

1. General: Break AB01-09 has sources from steam generator C and turbine building. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 1, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 45)

Room No. 2000 Main Feedwater

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 2

II. Effects Analysis

A. Problem No. 003, Steam Generator A, Secondary Systems Breaks.

1. General: Break AE04-01 has sources from steam generator A and feedwater heaters. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 2, such that no whipping occurs.

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are the containment cooler C supply and return lines. Function of these essential systems is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 003A, Steam Generator B, Secondary Systems Breaks.

1. General: Break AE04-04 has sources from steam generator B and feedwater heaters. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 2, such that no whipping occurs.

TABLE 3.6-4 (Sheet 46)

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are containment cooler C supply and return lines. Function of these essential systems is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 004A, Steam Generator C, Secondary Systems Breaks.

1. General: Break AE05-01 has sources from steam generator C and feedwater heaters. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 2, such that no whipping occurs.

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are containment cooler A and C essential service water supply and return lines, RCP-B thermal barrier cooling coil inlet and outlet lines, and component cooling water supply and return header to RCP-B and C. Function of these essential systems is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 47)

Room No. <u>2000</u> Main Feedwater

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 3

II. Effects Analysis

A. Problem No. 004, Steam Generator D, Secondary Systems Breaks

1. General: Breaks AE05-04 and 05 have sources from steam generator D and feedwater heaters. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 3, such that no whipping occurs.

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are component cooling water supply and return header to RCP-A and D. Function of these essential systems is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 48)

Room No. 2000 Reactor Coolant System -Pressurizer Relief

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 8

II. Effects Analysis

A. Problem No. 234A, Pressurizer-LOCA Breaks.

1. General: Breaks BB02-01, 02, 03, 04, 05, 06, 07, 08, 09, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 25, 27, 29, and 30 are LOCA breaks having an H-E source from the pressurizer. The downstream source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large-LOCA break criteria has been met. (See Note A)

3. Pipe whip: Whipping occurs. However, no essential systems are impacted. Whip restraints are not required.

4. Jet impingement: The jets from these breaks do not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 49)

Room No. 2000 Pressurizer Spray

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 9

II. Effects Analysis

A. Problem No. 242, Loops No. 1 and 2, LOCA Breaks.

1. General: Break BB04-05 is a large LOCA break having sources from the RCS cold leg, Loops No. 1 and 2, the pressurizer, and the regenerative HX. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The break is restrained per Figure 3.6-1, Sheet 9, such that no whipping occurs.

4. Jet impingement: The jet from this break does not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

7. Flooding: See Section 6.3.2.2

B. Problem No. 242, Loops No. 1 and 2, LOCA Breaks.

1. General: Breaks BB04-01, 02, 07, 08, 09, 10, 11, 12, and 13 are LOCA breaks having sources from RCS cold leg, Loops No. 1 and 2, the pressurizer and the regenerative HX. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 9. Whipping occurs for some breaks. However, no essential systems are impacted.

TABLE 3.6-4 (Sheet 50)

4. Jet impingement: The only target essential to mitigating the consequences of the break is a 2-inch-high head safety-injection line to RCS hot leg Loop No. 1. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 51)

Room No. 2000 RCP-D Seal Injection

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 12

II. Effects Analysis

A. Problem No. 249, Loop No. 4 - LOCA Breaks.

1. General: Break BB07-01 is a LOCA break having sources from the RCS, Loop No. 4, and charging pumps. The thrust force calculation takes into account the fact that the charging pump source is restricted by a throttle valve in the injection line. No other restrictions are considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: whipping occurs; however, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 52)

Room No. <u>2000</u> RCP-A Seal Injection

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 13

II. Effects Analysis

A. Problem No. 250, Loop No. 1 - LOCA Breaks.

1. General: Break BB08-09 is a LOCA break having sources from the RCS Loop No. 1 and the charging pumps. The thrust force calculation takes into account the fact that the charging pump source is restricted by a throttle valve in the injection line. No other restrictions are considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 13. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The jets from these breaks do not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 250, Loop No. 1 - Non-LOCA Breaks.

1. General: Break BB08-04 has a source from the charging pumps only. No source available from RCP A due to double check valves BB-V120 and V121 located between the break and RCP A. The thrust force calculation takes into account the fact that the charging pump source is restricted by a throttle valve in the injection line.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Pipe not capable of whipping due to low thrust force.

TABLE 3.6-4 (Sheet 53)

4. Jet impingement: The jet from this break does not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 276, Loop No. 1 - Non-LOCA Breaks.

1. General: Breaks BB08-03, 12, and 13 are non-LOCA breaks having source from the charging pumps only. No source available from RCP A due to double check valves BB-V120 and V121 located between the breaks and RCP A. The thrust force calculation takes into account the fact that the charging pump source is restricted by a throttle valve in the injection line.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Pipe not capable of whipping due to low thrust force.

4. Jet impingement: The jets from these breaks do not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 54)

Room No. 2000 RCP-C Seal Injection

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 14

II. Effects Analysis

A. Problem No. 251, Loop No. 3 - LOCA Breaks.

1. General: Break BB09-09 is a LOCA break having sources from the RCS Loop No. 3 and the charging pumps. The thrust force calculation takes into account that the charging pump source is restricted by a throttle valve in the injection line. No other restrictions are considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: Pipe geometry prevents whipping.

4. Jet impingement: The jets from these breaks do not impact any essential systems.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 251, Loop No. 3 - Non-LOCA Breaks.

1. General: Break BB09-04 has source from the charging pump only. No source available from RCP A due to double check valves BB-V180 and V181 located between break and RCP C. The thrust force calculation takes into account that the charging pump source is restricted by a throttle valve in the injection line.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Pipe not capable of whipping due to low thrust force.

4. Jet impingement: No essential systems are impacted.

TABLE 3.6-4 (Sheet 55)

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 277, Loop No. 3 - Non-LOCA Breaks.

1. General: Breaks BB09-03, 12, and 13 are non-LOCA breaks having source from the charging pumps only. No source available from RCP C due to double check valves BB-V180 and V181 located between breaks and RCP C. The thrust force calculation takes into account that the charging pump source is restricted by a throttle valve in the injection line.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Pipe not capable of whipping due to low thrust force.

4. Jet impingement: The targets essential to mitigating the consequences of the accident are seal injection to RCP-D and component cooling water injection (CCW) from the excess letdown heat exchanger. Function of these systems is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 56)

Room No. 2000 RCP-B Seal Injection

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room 15

II. Effects Analysis

A. Problem No. 252, Loop No. 2 - LOCA Breaks.

1. General: Break BB11-11 is a LOCA break having sources from the RCS Loop No. 2 and the charging pumps. The thrust force calculation takes into account that the charging pump source is restricted by a throttle valve in the injection line. No other restrictions are considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: Whipping occurs; however, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 252 Loop No. 2 - Non-LOCA Breaks.

1. General: Break BB11-05 has source from the charging pump only. No source available from RCP B due to double check valves BB-V150 and V151 located between break and RCP B. The thrust force calculation takes into account that the charging pump source is restricted by a throttle valve in the injection line.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Pipe not capable of whipping due to low thrust force.

4. Jet impingement: No essential systems are impacted.

TABLE 3.6-4 (Sheet 57)

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 278, Loop No. 2 - Non-LOCA Breaks.

1. General: Breaks BB11-04, are non-LOCA breaks having source from the charging pumps only. No source available from RCP B due to double check valves BB-V150 and V151 located between breaks and RCP B. The thrust force calculation takes into account that the charging pump source is restricted by a throttle valve in the injection line.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Pipe not capable of whipping due to low thrust force.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 58)

Room No. 2000 CVCS - Normal and Alternate Charging - Loops No. 1 and 4

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 24

II. Effects Analysis

A. Problem No. 254, Loop No. 1-LOCA Breaks.

1. General: Breaks BG21-18, 22, and 23 are LOCA breaks having sources from the RCS cold leg Loop No. 1 and regenerative heat exchanger. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small-LOCA break criteria has been met. (See Note B)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 24. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 254, Loop No. 1 - Non-LOCA Breaks.

1. General: Breaks BG21-24 and 25 have a source from the regenerative heat exchanger only. No source available from RCS Cold Leg Loop No. 1 due to double check valves BB-8378A and 8378B located between the breaks and Loop No. 1. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 24. Whipping occurs for some breaks. However, no essential systems are impacted.

TABLE 3.6-4 (Sheet 59)

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 254A, Loop 1 - Non-LOCA Breaks.

1. General: Breaks BG21-08, 09, 10, and 11 have a source from the regenerative heat exchanger only. No source available from RCS cold leg Loop No. 1 due to double check valves BB-8378A and 8378B located between the breaks and Loop No. 1. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 24, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 253, Loop No. 4 - LOCA Breaks.

1. General: Breaks BG21-12, 14, and 15 have sources from the RCS cold leg Loop No. 4 and regenerative heat exchanger. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small-LOCA break criteria has been met. (See Note B)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 24. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The only target essential to mitigating the consequences of the break is the hot leg safety-injection line. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 60)

E. Problem No. 253, Loop No. 4 - Non-LOCA Breaks.

1. General: Breaks BG21-16 and 17 have a source from the regenerative heat exchanger only. No source is available from RCS cold leg Loop No. 4 due to double check valves BB-8379A and 8379B located between the breaks and Loop No. 4. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 24. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is the 12-inch RHR pump suction, Loop No. 4. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

F. Problem No. 139, Loops 1 and 4 - Non-LOCA Breaks.

1. General: Breaks BG21-01, 02, 04, 05, 06, and 07 have a source from the regenerative heat exchanger only. No source available from RCS cold leg Loops No. 1 and 4 due to double check valves BB-8379A and 8379B located between the breaks and the loops. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 24, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 61)

Room No. 2000	CVCS - Letdown
I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room	25

II. Effects Analysis

A. Problem No. 245, Loop No. 3 - LOCA Breaks.

1. General: Breaks BG22-24, 26, 27, and 28 are LOCA breaks having sources from RCS crossover leg Loop No. 3 and the regenerative heat exchanger. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 25. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is seal injection to RCP-B. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 245, Loop No. 3 - Non-LOCA Breaks.

1. General: Break BG22-18 is a non-LOCA break having source from regenerative heat exchanger. No source available from RCS crossover leg due to closure of one of two isolation valves, BG-LCV 459 and LCV 460. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Break is restrained per Figure 3.6-1, Sheet 25, such that no whipping occurs.

TABLE 3.6-4 (Sheet 62)

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 145, Loop No. 3 - Non-LOCA Breaks.

1. General: Breaks BG22-01, 02, 03, and 04 are non-LOCA breaks having source from regenerative heat exchanger. No source available from RCS crossover leg due to closure of one of two isolation valves, BG-LCV459 and LCV460. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 25, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 146, Loop No. 3 - Non-LOCA Breaks.

1. General: Breaks BG22-05, 06, 07, 08, 09, and 13 are non-LOCA breaks having sources from regenerative heat exchanger and letdown heat exchanger. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 25, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 63)

E. Problem No. 119, Loop No. 3 - Non-LOCA Breaks.

1. General: Breaks BG22-10, 12, and 14 are non-LOCA breaks having sources from regenerative heat exchanger and letdown heat exchanger. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 25. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are seal injection to RCP-B and C, and CCW from the excess letdown HX and the excess letdown line. Function of these essential systems is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 64)

Room No. 2000 CVCS Charging and Excess Letdown

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 26

II. Effects Analysis

A. Problem No. 244, Loop No. 4 - LOCA Breaks.

1. General: Breaks BG23-04, 06, 07, and 11 are LOCA breaks having source from RCS crossover leg. The downstream sources are moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: Whipping occurs; however, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

B. Problem No. 147, Non-LOCA Breaks.

1. General: BG23-01, 02, and 03 are non-LOCA breaks having sources from the regenerative heat exchanger and the charging pumps. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 26. Whipping occurs for some breaks. However, no essential systems are impacted.

TABLE 3.6-4 (Sheet 65)

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is a CCW line from RCP-B thermal barrier. Function of the essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 66)

Room No. 2000 CVCS Auxiliary Spray

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 27

II. Effects Analysis

A. Problem No. 242, LOCA Break.

1. General: Break BG24-14 has sources from the RCS Cold Leg Loops No. 1 and 2 and the regenerative heat exchanger. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note B)

3. Pipe whip: Break is restrained per Figure 3.6-1, Sheet 27. Whipping occurs; however, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 242, Non-LOCA Breaks.

1. General: Breaks BG24-08 and 09 have a regenerative heat exchanger source only since the breaks are located between the regenerative heat exchanger and check valve BBV084. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 27, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 67)

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 140, Non-LOCA Breaks.

1. General: Breaks BG24-03, 07, 15 and 17 (Callaway only), 16 and 18 (Wolf Creek only) have a regenerative heat exchanger source only, since the breaks are located between the regenerative heat exchanger and check valve BBV084. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 27. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are seal injection to RCP-D, CCW from RCP-D thermal barrier cooling coil, and CCW line to the excess letdown HX and the excess letdown line. Function of these essential systems is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 139, Non-LOCA Breaks.

1. General: Breaks BG24-01 and 02 have a regenerative heat exchange source only, since the breaks are located between the regenerative heat exchanger and check valve BBV084. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 27, such that there are no whip targets.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 68)

Room No. 2000 Steam Generator A&D Blowdown

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 29

II. Effects Analysis

A. Problem No. 219, Secondary Systems Breaks.

1. General: Break BM01-04 and 06 have a steam generator D source and a turbine building source. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Whipping occurs; however, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 220, Secondary Systems Breaks

1. General: Breaks BM01-01 and 05 have a steam generator A source and a turbine building source. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 29, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 69)

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 70)

Room No. 2000 Steam Generator B&C Blowdown

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 30

II. Effects Analysis

A. Problem No. 221, Secondary Systems Breaks.

1. General: Breaks BM02-04, 05, and 07 have a steam generator B source and a turbine building source. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Break is restrained per Figure 3.6-1, Sheet 30 such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 222, Secondary Systems Breaks.

1. General: Breaks BM02-01, 06,and 08 have a steam generator C source and a turbine building source. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Break is restrained per Figure 3.6-1, Sheet 30 such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 71)

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 72)

Room No. 2000 Steam Generator A, B, C, D Blowdown

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 31

II. Effects Analysis

A. Problem No. 220, Secondary Systems Breaks.

1. General: Break BM03-07 has a H-E source from steam generator A. Downstream source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 31, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 221, Secondary Systems Breaks.

1. General: Break BM03-01 has a H-E source from steam generator B. Downstream source is moderate energy. No restrictions were considered in calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Break is restrained per Figure 3.6-1, Sheet 31, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 73)

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 222, Secondary Systems Breaks.

1. General: Break BM03-02 has a H-E source from steam generator C. Downstream source is moderate energy. No restrictions were considered in calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 31; such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 219, Loop No. 4, Secondary Systems Breaks.

1. General: Break BM03-04 has a H-E source from steam generator D. Downstream source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 31, such that no whipping occurs.

4. Jet impingement: No essential targets are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 74)

Room No. <u>2000</u> Steam Generator A Sample and Tube Sheet Drain

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 32

II. Effects Analysis

A. Problem No. 220, Secondary Systems Breaks.

1. General: Break BM17-06 has sources from steam generator A. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 32, such that no whipping occurs.

4. Jet impingement: The target essential to mitigating the consequences of the breaks is the seal injection to RCP-A. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 75)

Room No. 2000 Steam Generator B Sample and Tube Sheet Drain

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 33

II. Effects Analysis

A. Problem No. 221, Secondary Systems Breaks.

1. General: Break BM18-01 has sources from steam generator B. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 33. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is the feedwater line to steam generator C. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 76)

Room No. 2000 Steam Generator C Sample and Tube Sheet Drain

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 34

II. Effects Analysis

A. Problem No. 222, Secondary Systems Breaks.

1. General: Break BM19-01 has sources from steam generator C. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 34. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 77)

Room No. 2000 Steam Generator D Sample and Tube Sheet Drain

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 35

II. Effects Analysis

A. Problem No. 219, Secondary Systems Breaks.

1. General: Break BM20-01 has sources from steam generator D. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The secondary systems break criteria has been met. (See Note D)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 35. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 78)

Room No. 2000 | Residual Heat Removal, Loops No. 1 and 4

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 36

II. Effects Analysis

A. Problem No. 255, Loop No. 1, LOCA Breaks.

1. General: Breaks EJ04-06, 09, and 10 have a H-E source from the RCS Hot Leg, Loop No. 1. Sources from the RHR and S.I. pumps are moderate energy.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 36. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

7. Flooding: See Section 6.3.2.2

B. Problem No. 256, Loop No. 4, LOCA Breaks.

1. General: Breaks EJ04-01, 02, and 05 have a H-E source from the RCS hot leg, Loop No. 4. Sources from the RHR and S.I. pumps are moderate energy.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 36. Whipping occurs for some breaks. However, no essential systems are impacted.

TABLE 3.6-4 (Sheet 79)

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is seal injection to RCP-B. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

7. Flooding: See Section 6.3.2.2

TABLE 3.6-4 (Sheet 80)

Room No. 2000 — High Pressure Coolant Injection - Loops 2 and 3

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 38

II. Effects Analysis

A. Problem No. 248A, Loop No. 2 - LOCA Breaks.

1. General: EM03-08 and 29 are LOCA breaks having a H-E source from the RCS hot leg Loop No. 2. Sources from the hot leg recirculation line and S.I. pumps are moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 38, such that there are no whip targets.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 248A, Loop No. 3, LOCA Breaks.

1. General: Breaks EM03-05 and 27 are LOCA breaks having a H-E source from the RCS hot leg Loop No. 3. Sources from the hot leg recirculation line and S.I. pumps are moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 38, such that there are no whip targets.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 81)

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 247A, Loop No. 1, LOCA Breaks.

1. General: Breaks EM03-15 and 18 are LOCA breaks having a H-E source from the RCS cold leg Loop No. 1. Source from the boron injection header is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 38, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 247A, Loop No. 2, LOCA Breaks.

1. General: Breaks EM03-09 and 12 are LOCA breaks having a H-E source from the RCS cold leg Loop No. 2. Source from the boron injection header is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 38, such that no whipping occurs.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is boron injection to RCS cold leg Loop No. 3. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 82)

E. Problem No. 247A, Loop No. 3, LOCA Breaks.

1. General: Breaks EM03-01 and 04 are LOCA breaks having a H-E source from the RCS cold leg Loop No. 3. Source from the boron injection header is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 38. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

F. Problem No. 247A, Loop No. 4, LOCA Breaks.

1. General: Breaks EM03-19 and 22 are LOCA breaks having a H-E source from the RCS cold leg loop No. 4. Source from the boron injection header is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 38. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 83)

Room No. 2000 High Pressure Coolant Injection - Loops 1 & 4

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 39

II. Effects Analysis

A. Problem No. 255, Loop No. 1, LOCA Breaks

1. General: Breaks EM05-04, 05, 06, and 07 have a H-E source from the RCS hot leg, Loop No. 1. Source from the S.I. pump is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large-LOCA break criteria has been met for breaks EM05-03, 04, and 05. (See Note A) The small LOCA break criteria has been met for breaks EM05-06 and EM05-07. (See Note B)

3. Pipe whip: Whipping occurs; however, no essential systems are impacted.

4. Jet impingement: No essential targets are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 256, Loop No. 4, LOCA Breaks.

1. General: Break EM05-01 has a H-E source from the RCS hot leg, Loop No. 4. Source from the S.I. pump is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large-LOCA break criteria has been met. (See Note A)

3. Pipe whip: Whipping occurs; however, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

TABLE 3.6-4 (Sheet 84)

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 85)

Room No. <u>2000</u> Accumulator Injection, Loops 1 & 4

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 40

II. Effects Analysis

A. Problem No. 234, Loop No. 1 - LOCA Breaks.

1. General: Breaks EP01-01 and 04 are LOCA breaks having sources from the RCS cold leg Loop No. 1 and accumulator tank A. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 40, such that no whipping occurs.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 234, Loop No. 1 - Non-LOCA Break.

1. General: Breaks EP01-05, 07, 18, 19, 20, and 22 have a H-E source from accumulator tank A only. No source available from Loop No. 1 due to check valve BB-8948B located between the breaks and Loop No. 1. Sources from the RHR and SI pumps are moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 40. Whipping occurs for some breaks. However, no essential systems are impacted.

TABLE 3.6-4 (Sheet 86)

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is the cold leg Loop No. 2 safety-injection line. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 235, Loop 4 - LOCA Breaks.

1. General: Breaks EP01-10 and 13 are LOCA breaks having sources from the RCS cold leg Loop No. 4 and accumulator tank D. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 40, such that no whipping occurs.

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are seal injection to RCP-C and RCP-B. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 235, Loop 4 - Non-LOCA Breaks.

1. General: Breaks EP01-08, 14, 15, 16, 17, and 28 have a H-E source from accumulator tank D only. No source available from Loop No. 4 due to check valve BB-8948D located between the breaks and Loop No. 4. Sources from the RHR and SI pumps are moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 40, such that the whip targets are not required for safe shutdown or to mitigate the consequences of the accident.

4. Jet impingement: The targets essential to mitigating the consequences of the breaks are seal injection to RCP-A, B, and C, component cooling water to excess letdown HX, and to the thermal

TABLE 3.6-4 (Sheet 87)

barrier cooling coil, RCP-D, and the excess letdown heat exchanger discharge line. Function of these essential targets is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 88)

Room No. 2000 — Accumulator Injection - Loops 2 and 3

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room — 41

II. Effects Analysis

A. Problem No. 237, Loop No. 2 - LOCA Breaks.

1. General: Breaks EP02-01 and 04 are LOCA breaks having sources from the RCS cold leg, Loop No. 2 and accumulator tank B. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 41, such that no whipping occurs.

4. Jet impingment: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 237, Loop No. 2 - Non-LOCA Breaks.

1. General: Breaks EP02-05, 06, 16, 17, 18, and 20 have a H-E source from accumulator tank B only. No source available from Loop No. 2 due to check valve BB-8948B located between the breaks and Loop No. 2. Sources from the RHR & S.I. pumps are moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 41, such that no whipping occurs.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is component cooling water from the

TABLE 3.6-4 (Sheet 89)

thermal barrier cooling coil, RCP-B. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 236, Loop No. 3 - LOCA Breaks.

1. General: Breaks EP02-08 and 11 are LOCA breaks having sources from the RCS cold leg Loop No. 3 and accumulator tank C. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large LOCA break criteria has been met. (See Note A)

3. Pipe whip: The breaks are restrained per Figure 3.6-1, Sheet 41, such that no whipping occurs.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is seal injection to RCP-B. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

D. Problem No. 236, Loop 3 - Non-LOCA Breaks.

1. General: Breaks EP02-07, 12, 13, 14, 15, and 22 have a H-E source from accumulator tank C only. No source available from Loop No. 3 due to check valve BB-8948C located between the breaks and Loop No. 3. Sources from the RHR and S.I. pumps are moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Breaks are restrained per Figure 3.6-1, Sheet 41. Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 90)

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 91)

Room No. <u>2000</u>	Loop Drains
I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room	50

II. Effects Analysis

A. Problem No. 245, Loop No. 2 - LOCA Breaks.

1. General: Breaks HB24-03, and 04 have a H-E source from the crossover leg, Loop No. 2. The downstream source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small-LOCA break criteria has been met. (See Note B)

3. Pipe whip: Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

B. Problem No. 245, Loop No. 3 - LOCA Breaks.

1. General: Break HB24-05 has a H-E source from the crossover leg, Loop No. 3. The downstream source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small-LOCA break criteria has been met. (See Note B)

3. Pipe whip: Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 92)

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 244, Loop No. 1 - LOCA Breaks.

1. General: Breaks HB24-01 and 02 have a H-E source from the crossover leg, Loop No. 1. The downstream source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The small-LOCA break criteria has been met. (See Note B)

3. Pipe whip: Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is the safety injection to RCS hot leg Loop No. 1. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 93)

Room No. 2000 Liquid Radwaste

I. Sheets of Figure 3.6-1 showing high-energy (H-E) piping in this room: 51

II. Effects Analysis

A. Problem No. 234, Non-LOCA Breaks.

1. General: Breaks HB27-01, 02, and 09 have a H-E source from the accumulator tank A. The reactor coolant drain tank pump source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

B. Problem No. 235, Non-LOCA Breaks.

1. General: Breaks HB27-07 and 08 have a H-E source from the accumulator tank D. The reactor coolant drain tank pump source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

TABLE 3.6-4 (Sheet 94)

6. Temperature and humidity: See Section 6.2.1.1.3a

C. Problem No. 236, Non-LOCA Breaks.

1. General: Breaks HB27-05 and 06 have a H-E source from the accumulator tank C. The reactor coolant drain tank pump source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

D. Problem No. 237, Non-LOCA Breaks.

1. General: Break HB27-04 has a H-E source from the accumulator tank B. The reactor coolant drain tank pump source is moderate energy. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The non-LOCA break criteria has been met. (See Note C)

3. Pipe whip: Whipping occurs for some breaks. However, no essential systems are impacted.

4. Jet impingement: No essential systems are impacted.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 95)

Room No. 2000 Pressurizer Surge Line

I. Pressurizer surge line pipe breaks are shown on Figure 3.6-3

II. Effects Analysis

A. Pressurizer Surge Line, Loop 4 - LOCA Breaks.

1. General: Breaks BB01-09 and 15 are LOCA breaks having sources from the RCS hot leg Loop 4 and the pressurizer. No restrictions were considered in the calculation of thrust forces.

2. Criteria: The large-LOCA break criteria has been met. (See Note A)

3. Pipe whip: Breaks are restrained per Figure 3.6-3. The whip targets are not required for safe shutdown or to mitigate the consequences of the accident.

4. Jet impingement: The only target essential to mitigating the consequences of the breaks is accumulator safety injection, Loop No. 1. Function of this essential system is ensured.

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 96)

Room No. 2000 Reactor Coolant Loops

I. Reactor coolant loop pipe breaks are shown on Figure 3.6-3

II. Effects Analysis

A. Reactor Coolant Loops 1, 2, 3, and 4 - LOCA Breaks.

1. General - Reactor coolant loop breaks are no longer postulated due to adoption of Leak-Before-Break analysis. Pressurization, Temperature and humidity are now a result of LOCA analysis considering Hot Leg, Cold Leg, and Pump Suction Guillotine breaks.

2. Deleted

3. Deleted

4. Deleted

5. Room pressurization: See Section 6.2.1.1.3a

6. Temperature and humidity: See Section 6.2.1.1.3a, c

TABLE 3.6-4 (Sheet 97)

NOTES TO TABLE 3.6-4 (Sheet 1)

A. LARGE LOCA BREAK CRITERIA

1. The effects of large LOCA breaks must be limited to the following:

   a. Containment integrity must be maintained.

   b. Propagation to the secondary system is not allowed.

   c. No break propagation to the three remaining intact LOOPS is allowed.

   d. For branch line breaks, break propagation in the affected LOOP must be limited to an increase of 20 percent of the initial break area.

   e. For main coolant loop pipe breaks, break propogation limits are stated in PIP Vol. 1-3, Tab 10.

2. The following "ESSENTIAL" functions are required for mitigation of the pipe break via the ECCS systems.

   a. Accumulator safety injection to the three intact loops.

   b. Low head (RHR) safety injection to the three intact loops.

   c. Reactor coolant system equipment supports must maintain their functions.

3. The following other systems located inside the containment must maintain their design redundancy:

   a. Containment Spray (EN)

   b. Containment Cooling (GN)

   c. Containment Hydrogen Control (GS)

   d. Containment Isolation

4. The following safety actuation signals must be capable of being generated from instrumentation within the containment.

   a. Reactor Trip

TABLE 3.6-4 (Sheet 98)

b. Safety Injection Signal

c. Containment Isolation Phase A and Phase B

d. Containment Spray Actuation

5. All safety-related equipment located outside of the containment is operable and subject to single failure criteria.

6. No non-safety related equipment either inside or outside the containment is required for mitigation of the high energy line break effects of this LOCA (i.e., pipe whip, jet impingement, flooding, room pressurization, temperature and humidity effects).

B. <u>SMALL LOCA BREAK CRITERIA</u>

1. The effects of small LOCA breaks must be limited to the following:

   a. Containment integrity must be maintained.

   b. Rupture of steam-feedwater lines must be prevented.

   c. Break propagation must be limited to the affected leg.

   d. Break propagation in the affected leg must be limited to 12.5 square inches (4 inches ID).

   e. Damage to the high head safety injection lines connected to the other leg of the affected loop or to the other loops must be prevented.

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

2. The following ESSENTIAL functions are required for mitigation of the pipe break:

   a. High head safety injection via the ECCS systems.

   b. Boration via one of the following paths:

      1. Boration via the boron injection path to the four loops

TABLE 3.6-4 (Sheet 99)

2. Boration via the RCP seals for all four loops

3. Boration via normal charging

c. Reactor coolant system equipment supports and restraints must maintain their functions.

3. The following other systems located inside the containment must maintain their design redundancy:

a. Containment Spray (EN)

b. Containment Cooling (GN)

c. Containment Hydrogen Control (GS)

d. Containment Isolation

4. The following safety actuation signals must be capable of being generated from instrumentation within the containment:

a. Reactor Trip

b. Safety Injection Signal

c. Containment Isolation Phase A and Phase B

d. Containment Spray Actuation

5. All safety-related equipment located outside the containment is operable and subject to single failure criteria.

6. No non-safety related equipment either inside or outside containment is required for mitigation of the high energy line break effects of this LOCA (i.e., pipe whip, jet impingement, flooding, room pressurization, temperature and humidity effects).

C. <u>NON-LOCA BREAK CRITERIA</u>

1. The effects of non-LOCA breaks must be limited to the following:

a. Containment integrity must be maintained.

b. A non-LOCA break must not cause a loss of coolant or secondary systems line break.

TABLE 3.6-4 (Sheet 100)

c. The essential functions required for safe shutdown due to a non-LOCA break must be maintained. (See Appendix 5.4A)

2. The following other systems located inside containment must maintain their design redundancy:

   a. Containment Cooling (GN)

   b. Containment Isolation

3. The following safety actuation signals must be capable of being generated from instrumentation within the containment:

   a. Reactor Trip

   b. Containment Isolation

   c. Safety Injection Signal

4. No non-safety-related equipment either inside or outside containment is required for safe shutdown due to a non-LOCA pipe break.

D. SECONDARY SYSTEMS BREAK CRITERIA

1. The effects of secondary systems breaks must be limited to the following:

   a. Containment integrity must be maintained.

   b. Propagation to the primary system is not allowed.

   c. The essential functions required for safe shutdown due to a secondary systems break must be maintained. (see Sections 15.1.5 and 15.2.8 and Appendix 5.4A)

2. The following other systems located inside the containment must maintain their design redundancy:

   a. Containment Spray (EN)

   b. Containment Cooling (GN)

   c. Containment Isolation

3. The following safety actuation signals must be capable of being generated from instrumentation within containment:

TABLE 3.6-4 (Sheet 101)

a. Reactor Trip

b. Safety Injection Signal

c. Containment Isolation

d. Containment Spray

4. No non-safety related equipment either inside or outside containment is required for safe shutdown due to a secondary systems pipe break.

TABLE 3.6-5 DELETED

TABLE 3.6-6 SUMMARY OF FLOOD LEVELS IN ALL SAFETY-RELATED ROOMS

AUXILIARY BUILDING

Room No.	Flood Level Above Floor Elevation
1101	2' 9.72"
1102	2' 9.72"
1103	2' 9.72"
1104	2' 9.72"
1105	2' 9.72"
1106	2' 9.72"
1107	0' 0"
1108	0' 0"
1109	6' 4"
1110	6' 4"
1111	6' 4"
1112	6' 4"
1113	0' 0"
1114	0' 0"
1115	2' 9.72"
1116	2' 9.72"
1117	2' 9.72"
1119	2' 9.72"
1120	2' 9.72"
1121	9' 9.72"
1122	2' 9.72"
1123	2' 9.72"
1124	2' 9.72"
1125	2' 9.72"
1126	2' 9.72"
1127	2' 9.72"
1128	2' 9.72"
1129	2' 9.72"
1130	2' 9.72"
1201	0' 0"
1202	0' 0"
1203	0' 7.32"
1203A	0' 7.32"
1204	0' 0"

AUXILIARY BUILDING (Cont.)

Room No.	Flood Level Above Floor Elevation
1206	1' 6.07"
1207	1' 6.07"
1301	0' 7.87"
1302	0' 0"
1304	0' 0"
1305	0' 0"
1306	0' 0"
1307	0' 7.87"
1308	0' 0"
1309	0' 0"
1310	0' 0"
1311	0' 0"
1312	0' 0"
1313	0' 7.87"
1314	0' 7.87"
1315	0' 7.87"
1316	0' 0"
1317	0' 0"
1318	0' 7.87"
1320	0' 7.87”
1322	0' 0"
1323	0' 0"
1324	0' 0"
1325	0' 0"
1326	0' 0"
1327	0' 0"
1328	0' 0"
1329	0' 0.07"
1330	0' 0"
1331	0' 0"
1401	0' 6"
1402	0' 6"
1403	0' 0"

TABLE 3.6-6 (Sheet 2)

AUXILIARY BUILDING (Cont.)

Room No.	Flood Level Above Floor Elevation
1406	0' 0"
1408	0' 6"
1409	0' 0"
1410	0' 0"
1411	2' 2"
1412	2' 2"
1413	0' 0"
1501	0' 0.2"
1502	0' 0"
1503	0' 0"
1504	0' 0"
1505	0' 0"
1506	0' 0"
1507	0' 0"
1508	0' 0"
1509	0' 0"
1512	0' 0.2"
1513	0' 0"

REACTOR BUILDING

Room No.	Flood Level Elevation
2000	
LOCA	2004'-8"
MSLB	2004'-4"

CONTROL BUILDING

Room No.	Flood Level Above Floor Elevation
3101	2' 4.5"
3104	0' 0"
3301	0' 0"
3302	0' 0"
3403	0' 0"
3404	0' 0"

CONTROL BUILDING

Room No.	Flood Level Above Floor Elevation
3405	0' 0"
3407	0' 0"
3408	0' 0"
3409	0' 0"
3410	0' 0"
3411	0' 0"
3413	0' 0"
3414	0' 0"
3415	0' 0"
3416	0' 0"
3501	0' 0"
3601	0' 0"
3605	0' 0"
3609	0' 0"
3801	0' 0"

FUEL BUILDING

Room No.	Flood Level Above Floor Elevation
6102	0' 9.66"
6104	0' 9.66"
6105	0' 9.66"
6203	0' 0"
6303	0' 0"
6304	0' 0"

DIESEL BUILDING

Room No.	Flood Level Above Floor Elevation
5201	0' 2.61"
5203	0' 2.46"

TABLE 3.6-6 (Sheet 3)

RWST VALVE HOUSE

Room No.	Flood Level Above Floor Elevation
9102	See Note 1

ESW PUMP HOUSE

Room No.	Flood Level Above Floor Elevation
U104	0' 1.75"
U105	0' 1.75"

ESW COOLING TOWER

Room No.	Flood Level Above Floor Elevation
U301	See Note 1
U302	See Note 1
U303	0' 0"
U304	See Note 1
U305	See Note 1
U306	0' 0"
U307	0' 0"

Note 1: The flood elevation in this room is not calculated; the flood reaches a height sufficient to damage safety-related equipment.

Note 2: If this table is revised, review FSAR SP Table 3.11(B)-6 for potential impact.

3.7(B) SEISMIC DESIGN

In addition to the steady-state loads imposed on the system under normal operating conditions, the design of equipment and equipment supports requires that consideration also be given to abnormal loading conditions, such as earthquakes. Seismic loadings are considered for earthquakes of two magnitudes: Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE). The SSE is defined as the maximum vibratory ground motion at the plant site that can 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.

For Westinghouse-supplied items and Westinghouse-supplied items replaced by others, refer to Section 3.7(N).

The following material is in addition to Section 3.7(N) and applies to structures, systems, and components not supplied by Westinghouse. This section describes the techniques and discusses the parameters used to develop seismic loadings and criteria for seismic Category I structures, systems, and components.

The seismic responses of the major seismic Category I structures (containment, auxiliary/control, diesel generator, and fuel building) were originally generated for four sites (Callaway, Wolf Creek, Sterling, and Tyrone). Seismic design envelopes were developed by use of the most restrictive site conditions imposed by any one of the four original sites or by generic design criteria which are conservative for each of the sites. With the cancellation of the Tyrone plant, however, the four site enveloping approach was modified, for work not yet completed, to include only the remaining three sites. The seismic design envelopes were not revised to reflect the cancellation of the Sterling plant; therefore, since the design of all power block structures, systems, and components is based on the responses for three or four sites, the power block design is conservative for the remaining two sites. A further discussion of the multiple-site enveloping criteria, as applied to the seismic design of the SNUPPS power block, is contained in Section 3.7(B).2.2.

3.7(B).1 SEISMIC INPUT

3.7(B).1.1 Design Response Spectra

The site design response spectra in compliance with Regulatory Guide 1.60 are illustrated in Figures 3.7(B)-1 and 3.7(B)-2, in both the horizontal and vertical directions for the SSE. For the OBE, the design response spectra values were taken as 60 percent of the SSE. The values shown are for the site with maximum amplification. Section 2.5.2 of each Site Addendum and

Section 2.5 of BC-TOP-4-A (Ref. 3) discuss the effects of focal and epicentral distances from the site, depths between the focus of the seismic disturbances and the site, existing earthquake records, and the associated amplification of the response spectra.

Earthquake duration influences only the number of loading cycles on equipment because the equipment is designed for the elastic range in accordance with the analytical procedures outlined in BC-TOP-4-A. A 20.48-second duration is considered to be adequate for the time-history type of analysis used for the structures and equipment.

The design response spectra and earthquake time-histories are applied in the free field at finished grade for all sites.

3.7(B).1.1.1 Bases for Site Dependent Analysis

Section 2.5.2 of each Site Addendum and BC-TOP-4-A, Sections 2.4 and 2.5, describe the bases for specifying the vibratory ground motion for design use.

3.7(B).1.2 Design Time History

Synthetic earthquake time-histories were generated because the response spectra of recorded earthquake motions do not necessarily envelope any of the sites' design spectra. Figures 3.7(B)-3 and 3.7(B)-4 show the synthetic earthquake time-history motions in the horizontal and vertical directions, respectively. The time-histories shown were truncated to 20.48 seconds for use in the FLUSH finite element analyses discussed in Section 3.7(B).2.4.2. Figures 2-13, 2-14, 2-17, and 2-18 of BC-TOP-4-A show that the response spectra of the synthetic time-histories for the horizontal and vertical directions envelope the corresponding design spectra for 1 percent, 2 percent, 5 percent, 7 percent, and 10 percent damping. Section 2.5.1 of BC-TOP-4-A describes the generation of a typical synthetic earthquake time-history.

Typical foundation-level, free-field acceleration response spectra for each of the three sites are presented in Figures 3.7(B)-9A through D. Their envelope is presented in Figure 3.7(B)-10. All curves overlay the SNUPPS 60 percent design response spectra.

Due to site amplification of the seismic input, deconvolution of the SNUPPS control motion applied at grade will inevitably show an attenuation of the foundation level response relative to grade-level input motion. Attenuation is maximized at frequencies corresponding to the soil deposit fundamental frequencies. Hence, at particular frequencies, the computed foundation-level, free-field response spectra for the individual sites can be expected to and do fall below the SNUPPS 60 percent design spectra at some frequencies, similar to the ground spectrum and as shown by the Humboldt Bay results (Ref. 1).

3.7(B).1.3 Critical Damping Values

For seismic Category I structures, systems, and components not supplied by Westinghouse, the range of damping values (in percent of critical) is shown in Table 3.7(B)-1, is discussed in Sections 2.2 and 3.2.1 of BC-TOP-4-A, and is in compliance with Regulatory Guide 1.61. The applicable allowable stress values are

given in Section 3.8 for the various loading combinations, which include seismic loadings.

The testing of cable tray systems, as discussed in Section 3.10(B).3, clearly demonstrates that a substantial amount of energy is absorbed by friction between the adjacent moving cables and through friction between cables and the cable tray. This phenomenon was also observed to be amplitude dependent. That is, the greater the input level the more pronounced were these losses. Equating these losses during the test program resulted in predicted equivalent viscous damping of up to 50 percent in some cases. After tabulating the results of the several hundred earthquake-type vibration tests and cable tray systems, the allowable damping as a function of the level of seismic input motion was determined. A maximum value of 15 percent of critical was used for cable tray damping. Damping of supports for conduit is 7 percent of critical, regardless of input level.

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

In the FLUSH finite element analyses, the containment building at each of the three sites was supported on stabilized backfill down to a depth of 25 feet below grade. Also in the analyses, the auxiliary/control building at each of the three sites, along with the diesel generator building and the fuel building at the Sterling site, were founded directly on in-situ material. The diesel generator and fuel buildings for Callaway and Wolf Creek analyses were supported on crushed rock. The crushed rock extended from the bottom of the base mats down to a depth below grade of 29.5 feet in the Callaway analyses and 13 feet in the Wolf Creek analyses.

Descriptions of the supporting media at the Callaway and Wolf Creek sites are provided in Section 2.5 of each Site Addendum.

A list of the major seismic Category I structures and the depth of the soil and/or backfill deposits over the bedrock for each structure at each site is given in Table 3.7(B)-2.

The foundation embedment depth and minimum base dimension for each seismic Category I structure are provided in Table 3.7(B)-3, along with the method of seismic analysis utilized for each structure.

3.7(B).2 SEISMIC SYSTEM ANALYSIS

3.7(B).2.1 Seismic Analysis Methods

Seismic Category I structures, systems, and components were classified in accordance with NRC Regulatory Guide 1.29, as shown in Section 3.2. These structures, systems, and components were analyzed for two earthquake conditions, the SSE and the OBE.

The analytical methods utilized for the analysis of the different seismic Category I structures are summarized in Table 3.7(B)-3.

Lumped-mass models were developed for the containment, auxiliary/control, diesel generator, and fuel buildings, following the techniques discussed in Section 3.2 of BC-TOP-4-A. Figures 3.7(B)-17 through 3.7(B)-20 present the models developed for these structures. Mass and cross-sectional properties were calculated for the two principal normal horizontal directions and the vertical direction. The lumped-mass models of the major seismic Category I structures were incorporated, along with models of the significant non-Category I structures, into finite element models, of which Figure 3.7(B)-13 is typical. Time history analyses were performed using these finite element models, following procedures described in Section 3.7(B).2.4.2.

The results obtained from these analyses included maximum accelerations, inertia forces, shears, axial forces, moments, and floor response spectra. It was not possible to obtain displacements directly from the finite element analyses. Consequently, the procedure outline in Section 3.7(B).2.4.2 was used to determine building displacements.

The other seismic Category I structures (refueling water storage tank and valve house, emergency fuel oil storage tanks, and associated access vaults) are small compared to the major structures and are not directly adjacent to the major structures.

Consequently, structure-to-structure interaction between the major seismic Category I structures and these remaining seismic Category I structures is considered to be minimal. Therefore, the remaining structures were not included in the main finite element models.

3.7(B).2.2 Natural Frequencies and Response Loads

A summary of significant natural frequencies for the major seismic Category I structures is provided in Table 3.7(B)-4. The seismic responses generated for these structures, including accelerations, inertia forces, shears, axial forces, moments, and displacements are provided in Table 3.7(B)-5 through 3.7(B)-8. Typical floor response spectra are presented in Figures 3.7(B)-14 and 3.7(B)-15 for the polar crane and upper steam generator support locations, respectively.

All seismic responses were originally generated for the four SNUPPS sites, using the average soil properties for each site. As discussed previously, the responses, from either three or four sites, were enveloped and used in the design of all structures. Likewise, all subsystems and components were designed using either the three or four site envelopes of the floor response spectra of the site specific spectra. The effects of soil property variation on seismic responses was accounted for by the multiple-site enveloping procedures detailed above.

3.7(B).2.3 Procedure Used for Modeling

3.7(B).2.3.1 Lump Mass Modeling

A description of the procedure used to locate lumped masses for the seismic system analyses for seismic Category I structures and equipment is provided in Section 3.2 of BC-TOP-4-A. A similar discussion for piping systems is provided in Section 3.2 of BP-TOP-1 (Ref. 4).

3.7(B).2.3.2 Finite Element Modeling

Procedures used for finite element analysis modeling in seismic system analyses of seismic Category I structures is in accordance with the FLUSH computer program criteria, Reference 2.

3.7(B).2.4 Soil/Structure Interaction

Foundation embedment depth below grade, minimum base dimension, and method of analysis are given in Table 3.7(B)-3. The effect of soil-structure interaction was taken into account by coupling the structural model with the foundation medium.

3.7(B).2.4.1 Lumped Parameter Representation

A seismic analysis utilizing a lumped mass model on an elastic half space with strain independent soil properties was performed for comparison with the FLUSH finite element results. The purpose of this comparison was to provide a check on the FLUSH analysis. Figure 3.7(B)-12 shows the soil-structure model developed for the containment building. The response spectrum curves obtained by utilizing elastic half space analytical techniques compared favorably with the envelope curves developed for design use on the SNUPPS project.

3.7(B).2.4.2 Finite Element Representation

The finite element method of analysis was used to determine the seismic responses of the four major seismic Category I structures and the emergency fuel oil storage tanks. Additionally, displacement of the four major Category I structures was determined by using the DISCOM computer program (see Section 3.8(A).1.24 along with time histories from the finite element analysis. Figure 3.7(B)-13 shows a finite element model typical of the ones used to analyze to major power structures. The analytical model is provided with transmitting boundaries on both the left and right sides. The model also consists of two types of elements--displacement-compatible isoparametric quadrilateral elements (solid elements) and linear bending elements (beam elements). Usage of transmitting boundaries, elements, and analytical techniques are described in Reference 2. The computer program FLUSH, of the same reference, was used to perform the analysis.

Models, typically shown in Figure 3.7(B)-13, were used to perform soil-structure interaction analyses for all three sites. For each site, the site dependent soil properties were used. The vertical dimension of each soil element is equal to or less than Cs/5f, where Cs is the lowest soil element shear wave velocity reached during iterations and f is the highest frequency of interest to be transmitted through the soil profile. The highest frequency used was 25 Hz. In the analyses for the same buildings with site dependent soil parameters, the structural elements remained unchanged.

The site dependent soil properties consisted of strain dependent damping and modulus relationships for each material. In general, the soil properties are nonlinear in character. An iterative process was used to obtain equivalent linear properties which are strain dependent. The methods generally used for such an analysis are included in the computer program FLUSH.

3.7(B).2.5 Development of Floor Response Spectra

Acceleration time-histories obtained from the FLUSH finite element analyses were used in computing the floor response spectra for the major seismic Category I structures. The spectra were generated following the procedures outlined in Section 5.2 of BC-TOP-4-A, using the SPECTRA computer program (see subparagraph 3.8A.12).

3.7(B).2.6 Three Components of Earthquake Motion

Procedures for considering the three components of earthquake motion in determining the seismic response of structures, systems, and components follow the recommendations of Regulatory Guide 1.92 and are described in Section 4.3 of BC-TOP-4-A and Section 5.1 of BP-TOP-1.

3.7(B).2.7 Combination of Modal Responses

Combination is done according to the criterion of "the square-root-of-the-sum-of-the-squares" (SRSS).

Section 4.2.1 of BC-TOP-4-A describes the techniques used to combine modal responses for structures and equipment. For piping systems, closely spaced modes were determined per NRC Regulatory Guide 1.92, Equation 4.

3.7(B).2.7.1 Significant Dynamic Response Modes

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. Multiple degree-of-freedom systems which may have had frequencies in the resonance region of the amplified response spectra curves were analyzed by using a static load of 1.5 times the peak acceleration or the applicable floor response spectra to account for the contribution of higher modes. Multiplication factors less than 1.5 were not used.

Multiplication factors were not used in the equivalent static load method of analysis of conduit and cable tray supports which were multiple-degree-of-freedom, simple span, or cantilever beams. In these cases, other conservatisms such as lumping of masses (i.e., at the center of the simple beam span or at the end of the cantilever beam), consideration of mode shapes, and/or verification by dynamic analysis precludes the need for the use of multiplication factors.

Components which can adequately be characterized as a single-degree-of-freedom system were analyzed by using directly the seismic acceleration from the applicable floor response spectra.

For piping, refer to BP-TOP-1, Section 2.3.2, and Appendix D.

3.7(B).2.8 Interaction of Non-seismic Category I Structures With Seismic Category I Structures

With the use of the computer program FLUSH (see Table 3.7(B)-3), seismic analyses of all seismic Category I structures included the effects of adjacent, significant nonseismic Category I structures.

In addition, neither structural failure nor interference causing displacements during an SSE were permitted.

Elastic analyses have been performed to assure that the non-seismic Category I structures will not collapse onto seismic Category I structures when subjected to an SSE and will be allowed to reach 0.9 fy or 0.9 of any failure mode. Section 3.4 of BP-TOP-1 describes the techniques used to consider the interaction of seismic Category I piping with nonseismic Category I piping.

3.7(B).2.9 Effects of Parameter Variations on Floor Response Spectra

Section 5.2 of BC-TOP-4-A describes the effects on floor response spectra due to expected variations of structural properties, dampings, soil properties, foundation-structure interaction, etc.

3.7(B).2.10 Use of Constant Vertical Static Factors

Constant vertical load factors were not used for the analysis of seismic Category I structures, systems, and components. The methodology for vertical seismic analysis of structures is discussed in Sections 3.0, 4.0, and 5.0 of BC-TOP-4-A. The methodology for vertical seismic considerations for equipment is in accordance with IEEE 344, as amended in Section 3.10(B).

3.7(B).2.11 Method Used to Account for Torsional Effects

Torsional effects, if significant, were included in the horizontal models at locations of major mass and/or structure eccentricity. Section 3.2 and Appendix C of BC-TOP-4-A show the techniques used to account for torsional effects.

3.7(B).2.12 Comparison of Responses

Not applicable, since only the time-history method of analysis is used on major seismic Category I structures.

3.7(B).2.13 Determination of Seismic Category I Structure Overturning Moments

The effects of overturning moments were evaluated by the simplified, conservative static application of forces caused by the SSE. The more sophisticated energy methods shown in Section 4.4 of BC-TOP-4-A were used when the static method indicated unrealistic results. This section also includes a description of the methods used to compute foundation reactions and to account for vertical earthquake effects.

3.7(B).2.14 Analysis Procedure for Damping

The analysis procedure employed to account for damping in different elements of the model of a coupled system is described in Sections 3.2 and 3.3 of BC-TOP-4-A. The criteria used to account for composite damping in the coupled system with different elements are included. The analysis is based on the use of seismic Category I structural models which include a simplified version of the NSSS model provided by the NSSS supplier.

3.7(B).3 SEISMIC SUBSYSTEM ANALYSIS

3.7(B).3.1 Seismic Analysis Methods

Also see Section 3.7(B).2.1.

Section 2.0 and Appendix D of BP-TOP-1 describe the basis for the simplified dynamic analysis technique used in lieu of response spectrum analyses for piping. Simplified dynamic analysis was not used for seismic Category I structures, systems, and components other than piping.

3.7(B).3.2 Determination of Number of Earthquake Cycles

Fatigue analysis, where required by the codes, was performed by the supplier as part of the stress report. The earthquake transients are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination was separate and distinct from those transients resulting from fluid pressure and

temperature. The fluid pressure and temperature transients are given in Section 3.9(N).1.1. A description of the procedures followed in fatigue evaluations is given in Section 3.7(N).3.2.

The procedures used to determine the number of earthquake cycles for piping during one seismic event are discussed in Section 6.2 of BP-TOP-1. Equipment was designed on the basis of analytical results. The design criteria for equipment assumed elastic behavior. Therefore, the number of loading cycles need not be considered in the design. Fatigue was not considered in the design of seismic Category I structures, because the occurrence of full design earthquake loads is too infrequent to warrant consideration of fatigue design, and the calculated stresses and strains are below yield.

3.7(B).3.3 Procedure Used for Modeling

See Section 3.7(B).2.3.

3.7(B).3.4 Basis for Selection of Frequencies

Fundamental frequencies of subsystems and components were calculated in accordance with the procedures outlined in Section 4.2.1 of BC-TOP-4-A. To avoid resonance, the fundamental frequencies of subsystems and components were, where possible, selected in such a way as to avoid excessive load amplifications. If the subsystem's or component's frequencies fell within the amplified region of the forcing functions, the subsystems or components were adequately designed for the applicable loads.

3.7(B).3.5 Use of Equivalent Static Load Method of Analysis

See Section 3.7(B).2.7.1.

3.7(B).3.6 Three Components of Earthquake Motion

See Section 3.7(B).2.6.

3.7(B).3.7 Combination of Modal Responses

The seismic design of the piping and equipment included the effect of the seismic response of the supports, equipment, structures, and components. The system and equipment response was determined, using three earthquake components--two horizontal and one vertical. The design ground response spectra specified in Section 3.7(B).1 were the bases for generating these three input components. The input may be the floor time-history motions or floor response spectra. These floor time-history motions and/or floor response spectra are generated for two perpendicular horizontal directions (i.e., N-S and E-W), and the vertical direction. System and equipment analysis was performed with these input components applied in the N-S, E-W, and vertical directions. The damping values used in the analysis were those given in Table 3.7(B)-1.

In computing the system and equipment response by modal analysis, the square root of the sum of the squares of the modal contributions was used to combine all significant modal responses in each direction (see Section 3.7(B).2.7).

The combined total response was calculated, also using the SRSS formula applied to the resultant unidirectional responses. For instance, for each item of interest, such as displacement, force, stresses, etc., the total response is obtained by applying the above-described method.

This method can be written in equation form. The resultant response at a given node point for the item of interest, for example, s, is

$$\sigma = \left(\sum_{i=1}^{3} \sigma_i^2 \right)^{1/2} \qquad \text{3.7(B)-1}$$

where σ_i is the response in the i-th direction defined as

$$\sigma_i = \left(\sum_{j=1}^{N} \sigma_{ij}^2 \right)^{1/2} \qquad \text{3.7(B)-2}$$

with subscripts i and j in Equations 3.7(B)-1 and 3.7(B)-2 representing the i-th direction of input and the j-th mode (for a total of N significant modes). The term σ_{ij} is the maximum response in the j-th mode for input in the i-th direction, as determined by response spectrum model analysis.

The system and equipment response can also be determined, using time-history analyses.

3.7(B).3.8 Analytical Procedures for Piping

Section 2 of BP-TOP-1 describes the analytical techniques applicable to piping systems outside of the Westinghouse scope. Section 4 of BP-TOP-1 discusses the effect of differential building movements on piping.

3.7(B).3.9 Multiple Supported Equipment and Components With Distinct Inputs

See Section 3.7(B).3.8.

3.7(B).3.10 Use of Constant Vertical Static Factors

See Section 3.7(B).2.10.

3.7(B).3.11 Torsional Effects of Eccentric Masses

The significant torsional effects of valves and other eccentric masses are taken into account in the seismic piping analyses by the techniques discussed in Section 3.2 of BP-TOP-1.

3.7(B).3.12 Buried Seismic Category I Piping Systems and Tunnels

Procedures are defined in Section 6.0 of BC-TOP-4-A. All buried components are designed to remain functional after a seismic event by limiting the calculated stresses under all loading combinations, including earthquakes.

3.7(B).3.13 Interaction of Other Piping With Seismic Category I Piping

Section 3.4 of BP-TOP-1 describes the techniques used to consider the interaction of seismic Category I piping with non-Seismic Category I piping.

3.7(B).3.14 Seismic Analyses for Reactor Internals

See Section 3.7(N).3.14.

3.7(B).3.15 Analysis Procedure for Damping

See Section 3.7(B).2.15.

3.7(B).4 SEISMIC INSTRUMENTATION

3.7(B).4.1 Comparison with Regulatory Guide 1.12, Rev. 1 (April, 1974)

The seismic instrumentation program for the Standard Plant complies with Regulatory Guide 1.12, Rev. 1, except for the items listed below:

a. Response spectrum recorders are not supplied as discrete instruments except on the containment base mat. A spectrum analyzer permanently installed in the control room presents more complete information than that presented by response spectrum recorders. Data from the strong motion accelerometers are fed into the spectrum analyzer to produce earthquake spectra immediately following an earthquake. All locations where response spectrum recorders are required by the regulatory guide are monitored by strong motion accelerometers. This system achieves the intent of Regulatory Guide 1.12, Rev. 1.

b. Seismic triggers designed for use on the containment base slab have an actuated level adjustable over a minimum range of 0.01g to 0.03g, in lieu of the minimum sensitivity level of 0.005g specified in ANSI N18.5. Triggering levels below 0.01g are likely to produce spurious triggering due to normal plant vibrations. Additionally, the range specified in ANSI N18.5 cannot be applied to seismic triggers located at elevations greater than the base slab, due to normal amplification in structures. Each seismic trigger has an actuation level adjustable over a minimum range which is appropriate for that trigger location, such that all triggers actuate at approximately the same earthquake severity.

c. The strong motion accelerometer recording system has a minimum recording time of 15 minutes. Since the strong motion of major earthquakes rarely exceeds 30 seconds, 15 minutes provides sufficient recording time.

A comparison with the regulatory guide recommendations is provided in Table 3.7(B)-9.

3.7(B).4.2 Location and Description of Instrumentation

A seismic instrumentation program is provided to monitor the effect of earthquakes at the plant site and to collect data necessary to evaluate the safety impact of an earthquake on seismic Category I structures, systems, and components. Detailed location for all sensors is chosen to coincide with significant points in the seismic model. All seismic instrumentation is designed to seismic Category I requirements, including the battery emergency power supply. Power for normal operation and for maintaining the charge on the emergency power supply batteries is provided from the non-Class 1E 120V ac instrument bus.

3.7(B).4.2.1 Strong Motion Accelerometer (SMA)

Triaxial SMAs are installed at appropriate locations to provide data on the frequency, amplitude, and phase relationship of the seismic response of the containment structure and the seismic input to other seismic Category I structures, systems, and components.

One SMA (0-SG-AE-1) is located on the containment base, such that it will measure the input vibratory motion on the base slab. A second SMA is installed in the containment building, at the operating floor level (0-SG-AE-2), above and axially aligned with the SMA on the base slab.

SMAs are also provided in the auxiliary building, near the control room air filters (E1. 2047'-6") (0-SG-AE-5); on the auxiliary/control building base slab (0-SG-AE-4); on the outside wall of the reactor support structure (0-SG-AE-3), at the point where the 270-degree radial line from the reactor vessel center line intersects the outside wall (E1. 2012); and in the free field (0-SG-AR-1).

The function of a response spectrum recorder is provided by analyzing the strong motion data with equipment permanently installed in the control room.

3.7(B).4.2.2 Seismic Trigger

Triaxial seismic triggers are provided to start the SMA recording and playback system (0-SG-AR-11). Two seismic triggers are provided internal to 0-SG-AR-11, driven by 0-SG-AE-1 and 0-SG-AE-2. Triaxial seismic triggers are engineered to initiate recording at horizontal and vertical acceleration levels slightly higher than the expected background level, including induced vibrations from sources such as traffic, elevators, people, and machinery.

3.7(B).4.2.3 Peak Recording Accelerograph (PRA)

A PRA is a self-contained instrument which records the peak acceleration experienced at its location. PRAs are located on the radwaste building base slab (0-SG-AR-2), in the auxiliary building mounted on major seismic Category I piping or equipment (0-SG-AR-6), at the essential service water pump structure (0-SG-AR-4), in the control room (0-SG-AR-3), on the containment structure at a high elevation (0-SG-AR-5), and within the containment structure mounted on the steam generator C support (0-SG-AR-8) and reactor coolant system Loop 2 crossover leg piping (0-SG-AR-7).

Where a PRA is installed to specifically measure the acceleration experienced by a major seismic Category I component, it is oriented so that one axis corresponds to a principal equipment axis.

3.7(B).4.2.4 Seismic Switches (SS)

The seismic switches are an integral part of the SMA system. They are driven by the triaxial accelerometer sensors located at the containment base slab and the containment operating floor.

The seismic switches activate a plant annunciator in the control room if their setpoint accelerations are exceeded.

A pair of seismic switches associated with the SMA system are set to sense the operating base earthquake (OBE), and another pair are set to sense the safe shutdown earthquake (SSE).

3.7(B).4.2.5 Recording and Playback System (0-SG-AR-11)

Equipment located in the control room provides the recording, playback, and calibration functions which are used in conjunction with the SMA sensors to provide a time-history record of the earthquake. Also provided is signal conditioning and analysis equipment which performs the function of the response spectrum recorder.

3.7(B).4.2.6 Passive Response Spectrum Recorder (PRSR)

The passive response spectrum recorder (0-SG-ARS-1), located on the containment base slab, is an instrument which has the capability of sensing motion and permanently recording spectral accelerations at specified frequencies. The PRSR provides immediate control room indication when the specified spectral accelerations have been exceeded.

3.7(B).4.3 Control Room Operator Notification

An annunciation in the main control room is actuated whenever the SMA system has been triggered, calling the operator's attention to the fact that an event has occurred. Additionally, the seismic switches, provided as a backup to the SMA system, actuate an independent annunciator in the main control room in the event that the zero period OBE acceleration level has been exceeded.

Following a seismic event, all accessible data will be processed for an initial determination of the earthquake level. At sites where the site-related safety items (ultimate heat sink, etc.) are designed to an OBE less than the power block OBE, the unit will be shut down and site related items examined when an event of site OBE magnitude or greater occurs. If no evidence of damage is detected, the unit will be returned to service and the NRC notified.

3.7(B).4.4 Comparison of Measured and Predicted Responses

If the OBE has been exceeded, a response spectrum will be calculated for the instrument location. This spectrum will then be compared to the design seismic spectrum and the seismic loading calculated.

3.7(B).5 REFERENCES

1. "Seismic Soil-Structure Interaction Effects at Humboldt Bay Power Plant," Journal of the Geotechnical Engineering Division, Vol. 103, No. GT10, October 1977.

2. Lysmer, J., et al., "Efficient Finite Element Analysis of Seismic Structure-Soil-Structure Interaction," Earthquake Engineering Research Center, University of California, Berkeley, Cal., Report No. EERC 75-34, November, 1975.

3. Seismic Analyses of Structures and Equipment for Nuclear Power Plants, BC-TOP-4-A, Revision 3, Bechtel Power Corporation, San Francisco, California, November 1974.

4. Seismic Analysis of Piping Systems, BP-TOP-1, Revision 3, Bechtel Power Corporation, San Francisco, California, January 1976.

5. "Nuclear Reactors and Earthquakes," TID-7024, U.S. Atomic Energy Commission, Division of Technical Information, August 1963.

TABLE 3.7(B)-1 DAMPING VALUES FOR SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (PERCENT OF CRITICAL DAMPING)

Structure or Component	Operating Basis* Earthquake	Safe Shutdown Earthquake
Equipment and large-diameter piping systems**, pipe diameter greater than 12 in.***	2	3
Small-diameter piping systems, diameter equal to or less than 12 in.***	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7

* In the dynamic analysis of active components, as defined in Regulatory Guide 1.48, these values should also be used for the SSE.

** Includes both material and structural damping. If the piping system consists of only one or two spans with little structural damping, then use the values for small-diameter piping.

*** Code Case N-411-1, Alternate Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1, may also be applied subject to the conditions imposed by the NRC staff in Regulatory Guide 1.84.

TABLE 3.7(B)-2 DEPTH OF SOIL DEPOSITED OVER BEDROCK MAJOR SEISMIC CATEGORY I STRUCTURES

Site	Structure	Elev. of Bottom of Base Mat	Average Elev. of Top of Rock	Depth of Soil Over Rock (feet)
Wolf Creek	Reactor building	1088'-6"	1065'-0"	23.5
	Control building	1068'-0"	1065'-0"	3.0
	Fuel building	1093'-6"	1063'-0"	30.5
	Auxiliary building	1068'-0"	1065'-0"	3.0
	Diesel generators building	1089'-6"	1065'-0"	24.5
Tyrone Energy Park	Reactor building	828'-6"	815'-0"	13.5
	Control building	808'-0"	850'-0"	*
	Fuel building	833'-6"	815'-0"	18.5
	Auxiliary building	808'-0"	830'-0"	*
	Diesel generators building	829'-6"	850'-0"	*
Sterling	Reactor building	254'-0"	210'-0"	44.0
	Control building	233'-6"	215'-0"	18.5
	Fuel building	259'-0"	213'-0"	46.0
	Auxiliary building	233'-6"	217'-0"	16.5
	Diesel generators building	255'-0"	221'-0"	34.0
Callaway	Reactor building	829'-0"	809'-6"	19.5
	Control building	808'-6"	809'-6"	*
	Fuel building	834'-0"	809'-6"	24.5
	Auxiliary building	808'-6"	809'-6"	*
	Diesel generators building	830'-0"	809'-6"	20.5

* Base mat is on rock.

TABLE 3.7(B)-3 FOUNDATION DEPTH BELOW GRADE, MINIMUM BASE DIMENSION AND METHOD OF ANALYSIS FOR SEISMIC CATEGORY I STRUCTURES ALL SITES

Structure	Foundation Embedment Depth Below Grade (feet)	Minimum Base Dimension (feet)	Ratio of Embedment Depth to Minimum Base Dimension	Method of Analysis (1)
Reactor building	11	154	0.071	a
Control and auxiliary building	31.5	222	0.142	a
Fuel building	6	91	0.066	a
Diesel generators building	10	66.3	0.151	a
Foundation for refueling water storage tank	5.5	42.7	0.129	e
RWST valve house	13	13.1	0.992	b
Emergency fuel oil storage tanks (EFOST)	-	-	-	d
Vaults for EFOST	6	13.7	0.438	c

(1) Method of analysis

a. Finite-element method, FLUSH computer program

b. Response spectrum modal analysis technique

c. Single lumped mass-spring method - vaults are buried below grade with top at grade.

d. Finite element method in conjunction with the techniques for buried structures outlined in Section 6.0 of Reference 3.

e. Method outlined in Chapter 6.0 of Reference 5.

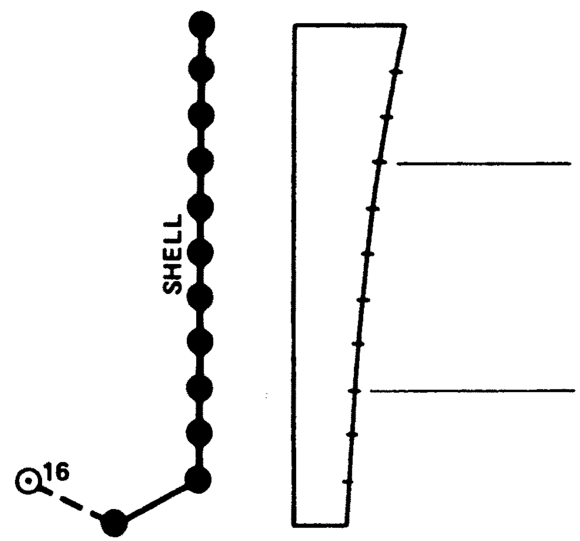
TABLE 3.7(B)-4 SUMMARY FIRST MODE NATURAL FREQUENCIES (HERTZ)

		SSE			OBE		
Building	Site	N-S	E-W	Vert.	N-S	E-W	Vert.
Containment	Callaway	3.8	3.8	10.0	3.8	3.8	10.5
	Sterling	4.4	4.4	13.0	4.8	4.4	13.0
	Wolf Creek	4.4	4.4	13.0	4.4	4.4	13.0
Aux./Control	Callaway	7.0	6.3	3.6	7.5	6.8	4.0
	Sterling	8.5	7.3	6.0	9.0	7.5	6.8
	Wolf Creek	9.0	9.0	3.6	9.0	8.0	6.8
Fuel	Callaway	5.2	2.8	7.0	5.8	3.4	8.0
	Sterling	7.2	6.2	10.0	8.0	6.5	13.0
	Wolf Creek	7.0	5.0	9.5	6.7	5.0	9.5

TABLE 3.7(B)-5A RESPONSE ACCELERATIONS (G'S) CONTAINMENT BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.423	0.727	0.707	0.727
2170'-9"	0.376	0.650	0.626	0.650
2135'-0"	0.322	0.558	0.528	0.558
2119'-0"	0.298	0.516	0.484	0.516
2100'-0"	0.270	0.466	0.429	0.466
2080'-0"	0.242	0.425	0.376	0.425
2056'-6"	0.208	0.373	0.338	0.373
2051'-2"	0.201	0.361	0.329	0.361
2039'-0"	0.185	0.335	0.310	0.335
2028'-0"	0.169	0.313	0.293	0.313
2013'-5"	0.157	0.287	0.270	0.287
2000'-0"	0.160	0.267	0.250	0.267
2090'-4"	0.202	0.324	0.283	0.324
2060'-0"	0.175	0.301	0.270	0.301
2047'-6"	0.169	0.294	0.265	0.294
2034'-0"	0.234	0.286	0.261	0.286
2022'-6"	0.155	0.279	0.258	0.279
2012'-0"	0.155	0.274	0.254	0.274
2000'-0"	0.160	0.267	0.250	0.267



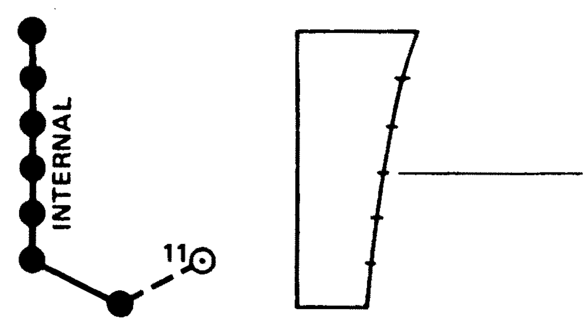
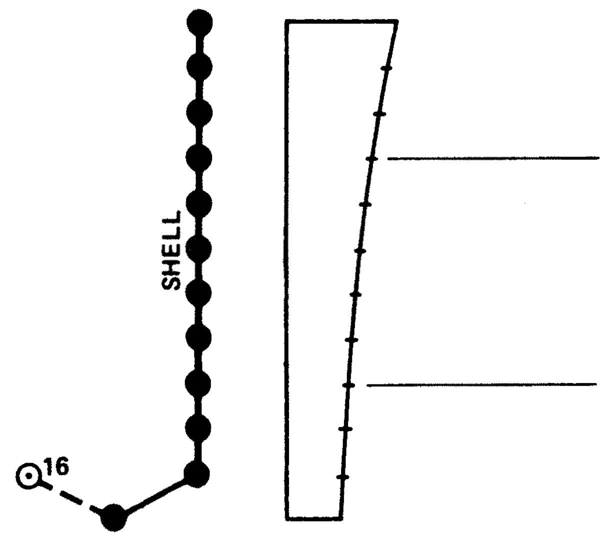


TABLE 3.7(B)-5B RESPONSE ACCELERATIONS (G'S) CONTAINMENT BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.403	0.891	0.814	0.891
2170'-9"	0.357	0.781	0.706	0.781
2135'-0"	0.310	0.651	0.576	0.651
2119'-0"	0.291	0.593	0.521	0.593
2100'-0"	0.265	0.528	0.452	0.528
2080'-0"	0.238	0.457	0.394	0.457
2056'-6"	0.205	0.374	0.355	0.374
2051'-2"	0.198	0.362	0.347	0.362
2039'-0"	0.182	0.335	0.327	0.335
2028'-0"	0.176	0.311	0.304	0.311
2013'-5"	0.175	0.283	0.281	0.283
2000'-0"	0.173	0.263	0.256	0.263
2090'-4"	0.364	0.432	0.373	0.432
2060'-0"	0.248	0.312	0.292	0.312
2047'-6"	0.227	0.301	0.283	0.301
2034'-0"	0.204	0.288	0.276	0.288
2022'-6"	0.186	0.277	0.271	0.277
2012'-0"	0.175	0.270	0.264	0.270
2000'-0"	0.173	0.263	0.256	0.263



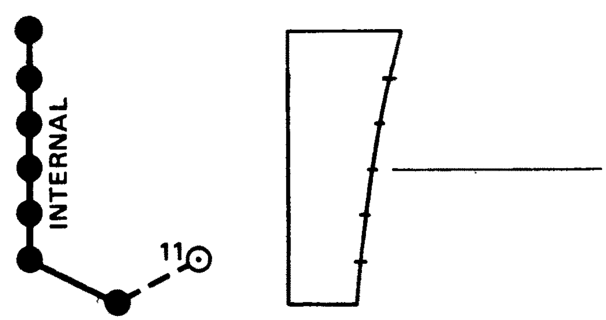
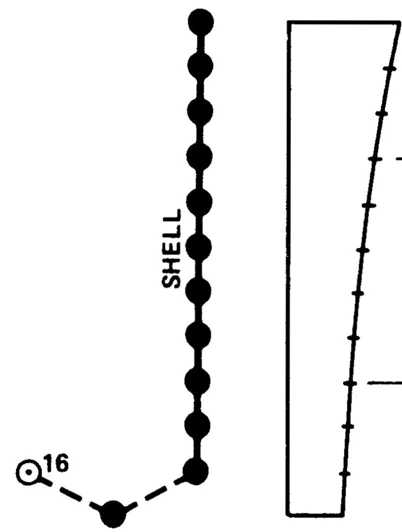


TABLE 3.7(B)-5C RESPONSE ACCELERATIONS (G'S) CONTAINMENT BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.330	0.466	0.388	0.466
2170'-9"	0.323	0.456	0.378	0.456
2135'-0"	0.303	0.428	0.349	0.428
2119'-0"	0.293	0.414	0.335	0.414
2100'-0"	0.278	0.392	0.314	0.392
2080'-0"	0.266	0.365	0.289	0.365
2056'-6"	0.264	0.328	0.282	0.328
2051'-2"	0.264	0.319	0.280	0.319
2039'-0"	0.262	0.302	0.275	0.302
2028'-0"	0.261	0.294	0.271	0.294
2013'-5"	0.259	0.282	0.265	0.282
2000'-0"	0.257	0.273	0.259	0.273
2090'-4"	0.262	0.280	0.267	0.280
2060'-0"	0.261	0.279	0.265	0.279
2047'-6"	0.261	0.278	0.264	0.278
2034'-0"	0.260	0.277	0.263	0.277
2022'-6"	0.259	0.276	0.262	0.276
2012'-0"	0.258	0.275	0.261	0.275
2000'-0"	0.257	0.273	0.259	0.273



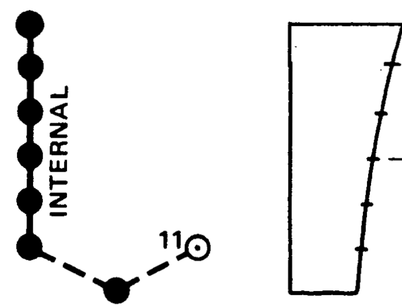
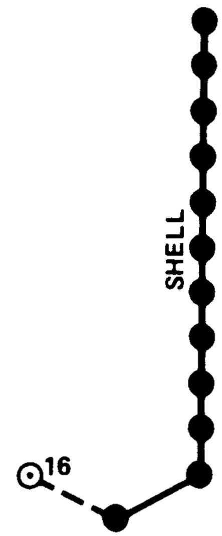


TABLE 3.7(B)-5D RESPONSE INERTIA FORCES (KIPS) CONTAINMENT BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	1590	2700	2630	2700
2170'-9"	2830	4840	4660	4840
2135'-0"	1930	3330	3150	3330
2119'-0"	1780	3050	2860	3050
2100'-0"	1490	2530	2370	2530
2080'-0"	1470	2480	2260	2480
2056'-6"	830	1360	1220	1360
2051'-2"	480	800	700	800
2039'-0"	580	940	830	940
2028'-0"	590	960	800	960
2013'-5"	590	1180	770	1180
2000'-0"	—	—	—	
2090'-4"	270	440	390	440
2060'-0"	270	460	420	460
2047'-6"	1290	2300	1930	2300
2034'-0"	530	1000	850	1000
2022'-6"	630	1200	1160	1200
2012'-0"	1190	2100	1640	2100
2000'-0"	—	—	—	



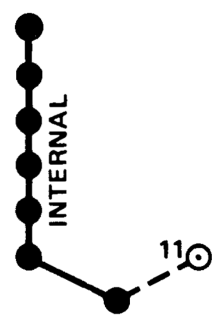
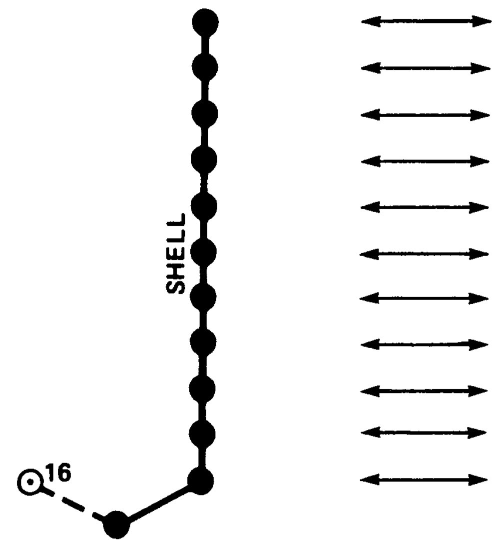


TABLE 3.7(B)-5E RESPONSE INERTIA FORCES (KIPS) CONTAINMENT BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	1520	3310	3040	3310
2170'-9"	2630	5820	5270	5820
2135'-0"	1860	3880	3460	3880
2119'-0"	1720	3500	3080	3500
2100'-0"	1450	2850	2480	2850
2080'-0"	1450	2730	2330	2730
2056'-6"	840	1450	1210	1450
2051'-2"	490	880	710	880
2039'-0"	590	1020	780	1020
2028'-0"	610	960	770	960
2013'-5"	620	940	700	940
2000'-0"	—	—	—	
2090'-4"	410	470	430	470
2060'-0"	380	480	450	480
2047'-6"	1590	2430	1910	2430
2034'-0"	720	1020	750	1020
2022'-6"	870	1210	1440	1440
2012'-0"	1620	1930	1470	1930
2000'-0"	—	—	—	



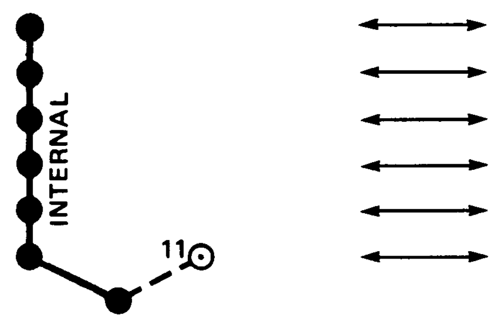


TABLE 3.7(B)-5F RESPONSE INERTIA FORCES (KIPS) CONTAINMENT BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	1220	1760	1400	1760
2170'-9"	2380	3440	2740	3440
2135'-0"	1790	2590	2020	2590
2119'-0"	1710	2480	1920	2480
2100'-0"	1500	2170	1650	2170
2080'-0"	1560	2250	1700	2250
2056'-6"	930	1350	980	1350
2051'-2"	550	790	580	790
2039'-0"	680	990	700	990
2028'-0"	710	1010	710	1010
2013'-5"	740	1040	710	1040
2000'-0"	—	—	—	
2090'-4"	360	390	370	370
2060'-0"	400	420	410	420
2047'-6"	1250	1340	1280	1340
2034'-0"	900	950	910	950
2022'-6"	1630	1740	1650	1740
2012'-0"	1690	1800	1720	1800
2000'-0"	—	—	—	

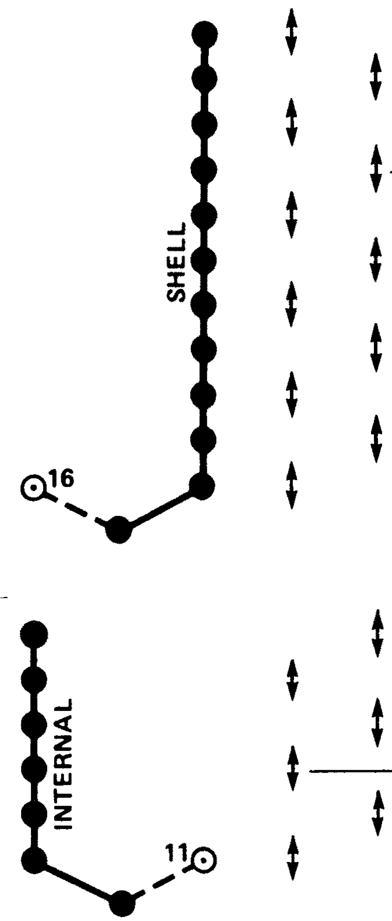
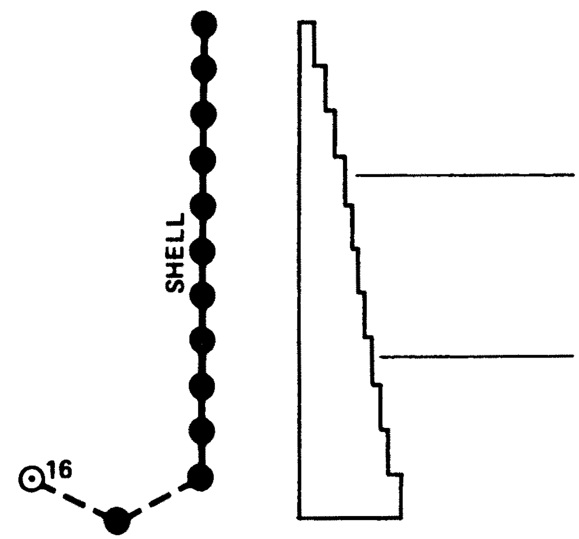


TABLE 3.7(B)-5G RESPONSE SHEAR FORCES (KIPS) CONTAINMENT BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"				
	1590	2700	2630	2700
2170'-9"				
	4420	7540	7290	7540
2135'-0"				
	6350	10,870	10,440	10,870
2119'-0"				
	8130	13,920	13,300	13,920
2100'-0"				
	9620	16,450	15,670	16,450
2080'-0"				
	11,090	18,930	17,930	18,930
2056'-6"				
	11,920	20,290	19,150	20,290
2051'-2"				
	12,400	21,090	19,850	21,090
2039'-0"				
	12,980	22,030	20,680	22,030
2028'-0"				
	13,570	22,990	21,480	22,990
2013'-5"				
	14,160	24,170	22,250	24,170
2000'-0"				
2090'-4"				
	270	440	390	440
2060'-0"				
	540	900	810	900
2047'-6"				
	1830	3200	2740	3200
2034'-0"				
	2360	4200	3590	4200
2022'-6"				
	2990	5400	4750	5400
2012'-0"				
	4180	7500	6390	7500
2000'-0"				



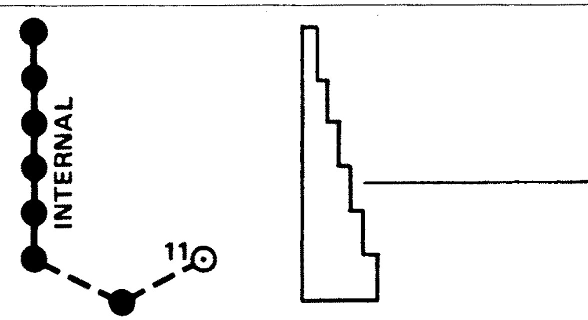
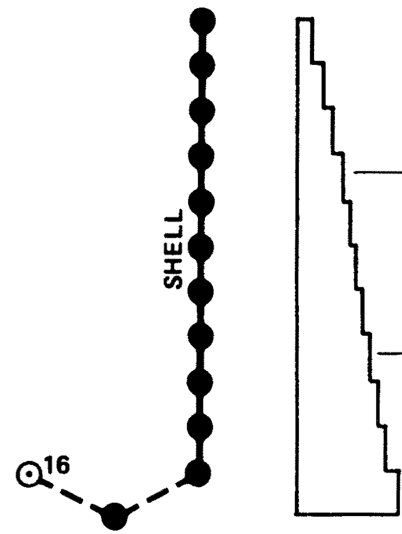


TABLE 3.7(B)-5H RESPONSE SHEAR FORCES (KIPS) CONTAINMENT BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"				
	1520	3310	3040	3310
2170'-9"				
	4150	9130	8310	9130
2135'-0"				
	6010	13,010	11,770	13,010
2119'-0"				
	7730	16,510	14,850	16,510
2100'-0"				
	9180	19,360	17,330	19,360
2080'-0"				
	10,630	22,090	19,660	22,090
2056'-6"				
	11,470	23,540	20,870	23,540
2051'-2"				
	11,960	24,420	21,580	24,420
2039'-0"				
	12,550	25,440	22,360	25,440
2028'-0"				
	13,160	26,400	23,130	26,400
2013'-5"				
	13,780	27,340	23,830	27,340
2000'-0"				
2090'-4"				
	410	470	430	470
2060'-0"				
	790	950	880	950
2047'-6"				
	2380	3380	2790	3380
2034'-0"				
	3100	4400	3540	4400
2022'-6"				
	3970	5610	4980	5610
2012'-0"				
	5590	7540	6450	7540
2000'-0"				



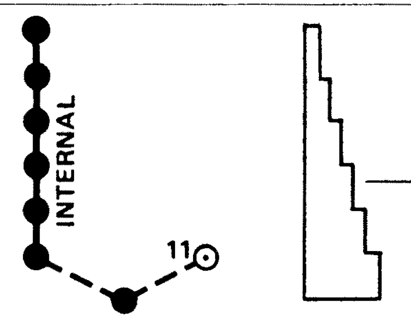
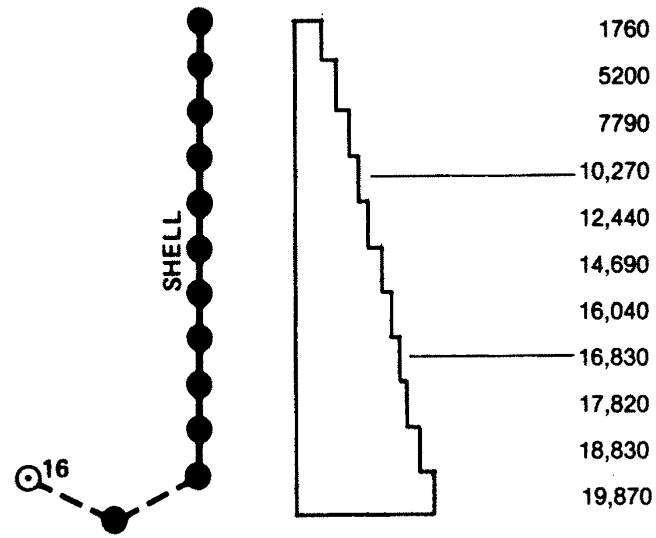


TABLE 3.7(B)-5I RESPONSE AXIAL FORCES (KIPS) CONTAINMENT BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE		
	CALLAWAY	STERLING	WOLF CREEK
2206'-6"			
	1220	1760	1400
2170'-9"			
	3600	5200	4140
2135'-0"			
	5390	7790	6160
2119'-0"			
	7100	10,270	8080
2100'-0"			
	8600	12,440	9730
2080'-0"			
	10,160	14,690	11,430
2056'-6"			
	11,090	16,040	12,410
2051'-2"			
	11,640	16,830	12,990
2039'-0"			
	12,320	17,820	13,690
2028'-0"			
	13,030	18,830	14,400
2013'-5"			
	13,770	19,870	15,110
2000'-0"			
2090'-4"			
	360	390	370
2060'-0"			
	760	810	780
2047'-6"			
	2010	2150	2060
2034'-0"			
	2910	3100	2970
2022'-6"			
	4540	4840	4620
2012'-0"			
	6230	6640	6340
2000'-0"			

ENVELOPE



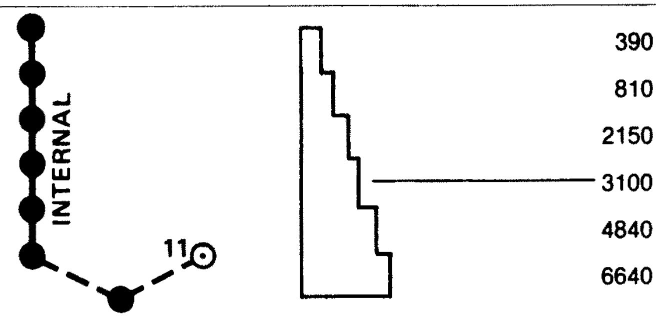
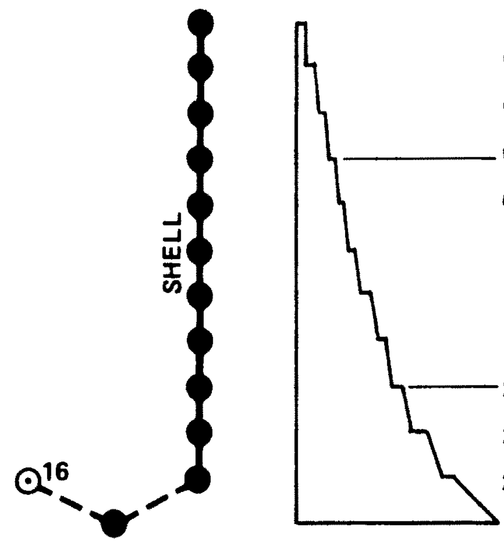


TABLE 3.7(B)-5J RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) CONTAINMENT BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0/0.0084	0/0.0159	0/0.0107	0/0.0159
2170'-9"	0.0622/0.0802	0.1051/0.1389	0.1030/0.1299	0.1051/0.1389
2135'-0"	0.2362/0.2534	0.4000/0.4267	0.3906/0.4189	0.4000/0.4267
2119'-0"	0.3552/0.3712	0.6005/0.6252	0.5860/0.6121	0.6005/0.6252
2100'-0"	0.5259/0.5414	0.8896/0.9132	0.8651/0.8898	0.8896/0.9132
2080'-0"	0.7336/0.7503	1.2421/1.2644	1.2032/1.2289	1.2421/1.2644
2056'-6"	1.0108/1.0210	1.7111/1.7252	1.6503/1.6652	1.7111/1.7252
2051'-2"	1.0845/1.0903	1.8334/1.8416	1.7673/1.7760	1.8334/1.8416
2039'-0"	1.2413/1.2488	2.0972/2.1070	2.0168/2.0286	2.0972/2.1070
2028'-0"	1.3916/1.3992	2.3500/2.3598	2.2560/2.2658	2.3500/2.3598
2013'-5"	1.5972/1.6052	2.6950/2.7028	2.5774/2.5872	2.6950/2.7028
2000'-0"	1.7954	3.0223	2.8851	3.0223
2090'-4"	0/0.0020	0/0.0031	0/0.0027	0/0.0031
2060'-0"	0.0043/0.0048	0.0069/0.0072	0.0061/0.0063	0.0069/0.0072
2047'-6"	0.0113/0.0137	0.0185/0.0200	0.0164/0.0176	0.0185/0.0200
2034'-0"	0.0357/0.0382	0.0622/0.0636	0.0534/0.5470	0.0622/0.0636
2022'-6"	0.0635/0.0678	0.1117/0.1151	0.0958/0.0989	0.1117/0.1151
2012'-0"	0.0980/0.1018	0.1718/0.1737	0.1483/0.1502	0.1718/0.1737
2000'-0"	0.1482	0.2636	0.2268	0.2636



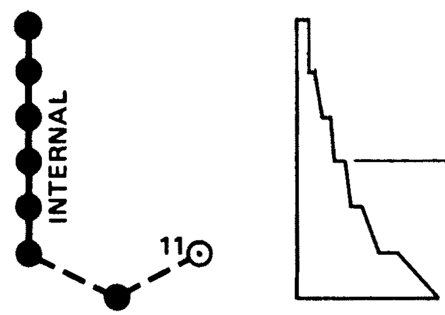
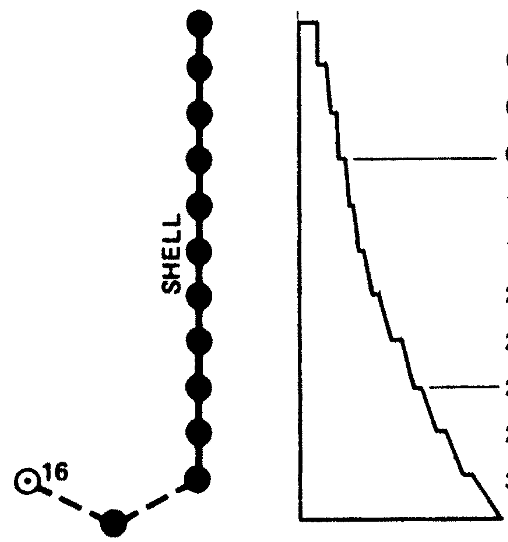


TABLE 3.7(B)-5K RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) CONTAINMENT BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0/0.0103	0/0.0147	0/0.0118	0/0.0147
2170'-9"	0.0614/0.0821	0.1304/0.1663	0.1206/0.1559	0.1304/0.1663
2135'-0"	0.2293/0.2513	0.4925/0.5302	0.4532/0.4898	0.4925/0.5302
2119'-0"	0.3438/0.3642	0.7381/0.7730	0.6782/0.7121	0.7381/0.7730
2100'-0"	0.5018/0.5212	1.0868/1.1196	0.9943/1.0261	1.0868/1.1196
2080'-0"	0.6887/0.7089	1.5068/1.5410	1.3728/1.4055	1.5068/1.5410
2056'-6"	0.9526/0.9614	2.0600/2.0796	1.8677/1.8865	2.0600/2.0796
2051'-2"	1.0225/1.0276	2.2050/2.2168	1.9972/2.0090	2.2050/2.2168
2039'-0"	1.1731/1.1795	2.5127/2.5264	2.2716/2.2834	2.5127/2.5264
2028'-0"	1.3175/1.3242	2.8048/2.8185	2.5304/2.5421	2.8048/2.8185
2013'-5"	1.5161/1.5231	3.2026/3.2144	2.8792/2.8910	3.2026/3.2144
2000'-0"	1.7079	3.5790	3.2124	3.5790
2090'-4"	0/0.0033	0/0.0039	0/0.0033	0/0.0039
2060'-0"	0.0069/0.0080	0.0079/0.0095	0.0068/0.0087	0.0079/0.0095
2047'-6"	0.0179/0.0191	0.0203/0.0223	0.0184/0.0213	0.0203/0.0223
2034'-0"	0.0512/0.0518	0.0665/0.0677	0.0545/0.0556	0.0665/0.0677
2022'-6"	0.0875/0.0891	0.1182/0.1216	0.0964/0.0988	0.1182/0.1216
2012'-0"	0.1308/0.1308	0.1804/0.1819	0.1503/0.1517	0.1804/0.1819
2000'-0"	0.1978	0.2722	0.2285	0.2722



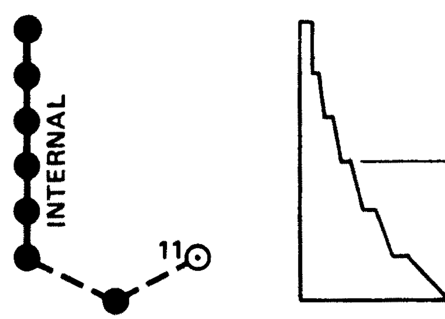
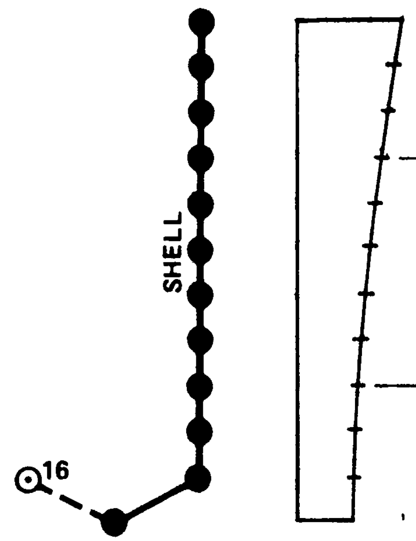


TABLE 3.7(B)-5L RESPONSE DISPLACEMENT (INCHES) CONTAINMENT BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.423	0.371	0.370	0.423
2170'-9"	0.382	0.322	0.321	0.382
2135'-0"	0.335	0.262	0.262	0.335
2119'-0"	0.314	0.235	0.236	0.314
2100'-0"	0.288	0.201	0.203	0.288
2080'-0"	0.260	0.165	0.167	0.260
2056'-6"	0.227	0.122	0.125	0.227
2051'-2"	0.220	0.112	0.115	0.220
2039'-0"	0.203	0.090	0.094	0.203
2028'-0"	0.188	0.071	0.076	0.188
2013'-5"	0.169	0.046	0.052	0.169
2000'-0"	0.152	0.023	0.031	0.152
2090'-4"	0.225	0.072	0.080	0.225
2060'-0"	0.204	0.055	0.063	0.204
2047'-6"	0.196	0.049	0.057	0.196
2034'-0"	0.181	0.042	0.050	0.181
2022'-6"	0.167	0.036	0.044	0.167
2012'-0"	0.160	0.030	0.038	0.160
2000'-0"	0.152	0.023	0.031	0.152



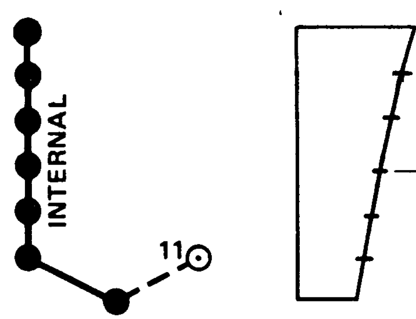
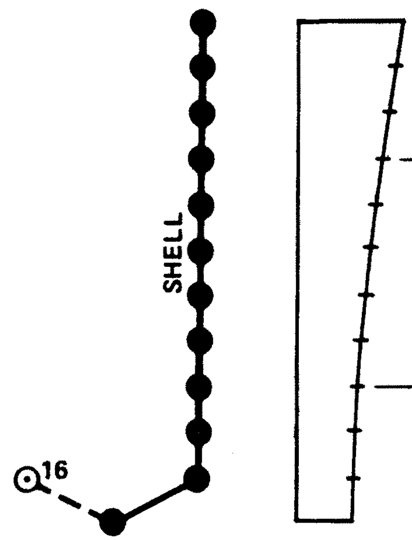


TABLE 3.7(B)-5M RESPONSE DISPLACEMENT (INCHES) CONTAINMENT BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.440	0.471	0.439	0.471
2170'-9"	0.397	0.406	0.379	0.406
2135'-0"	0.349	0.328	0.308	0.349
2119'-0"	0.327	0.293	0.277	0.327
2100'-0"	0.301	0.250	0.238	0.301
2080'-0"	0.273	0.203	0.196	0.273
2056'-6"	0.240	0.148	0.146	0.240
2051'-2"	0.233	0.136	0.135	0.233
2039'-0"	0.216	0.108	0.110	0.216
2028'-0"	0.201	0.083	0.088	0.201
2013'-5"	0.181	0.052	0.061	0.181
2000'-0"	0.164	0.025	0.036	0.164
2090'-4"	0.234	0.098	0.104	0.234
2060'-0"	0.209	0.073	0.081	0.209
2047'-6"	0.200	0.064	0.072	0.200
2034'-0"	0.190	0.053	0.062	0.190
2022'-6"	0.181	0.043	0.053	0.181
2012'-0"	0.173	0.035	0.045	0.173
2000'-0"	0.164	0.025	0.036	0.164



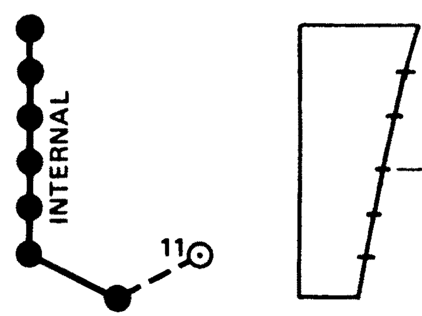
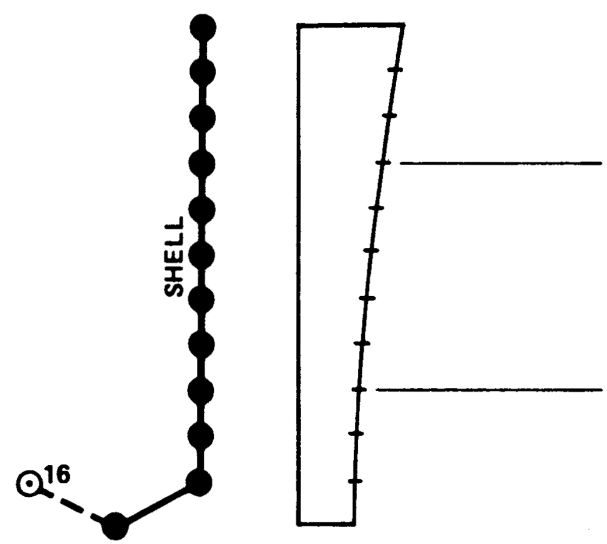


TABLE 3.7(B)-5N RESPONSE DISPLACEMENT (INCHES) CONTAINMENT BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.0207	0.0260	0.0213	0.0260
2170'-9"	0.0202	0.0253	0.0206	0.0253
2135'-0"	0.0186	0.0230	0.0186	0.0230
2119'-0"	0.0178	0.0219	0.0176	0.0218
2100'-0"	0.0166	0.0201	0.0160	0.0201
2080'-0"	0.0149	0.0178	0.0140	0.0178
2056'-6"	0.0126	0.0147	0.0112	0.0147
2051'-2"	0.0120	0.0139	0.0105	0.0139
2039'-0"	0.0106	0.0119	0.0089	0.0119
2028'-0"	0.0092	0.0101	0.0073	0.0101
2013'-5"	0.0072	0.0075	0.0052	0.0075
2000'-0"	0.0053	0.0049	0.0030	0.0053
2090'-4"	0.0075	0.0073	0.0053	0.0075
2060'-0"	0.0071	0.0068	0.0049	0.0071
2047'-6"	0.0069	0.0066	0.0047	0.0069
2034'-0"	0.0067	0.0063	0.0044	0.0067
2022'-6"	0.0063	0.0060	0.0041	0.0063
2012'-0"	0.0059	0.0056	0.0037	0.0059
2000'-0"	0.0053	0.0049	0.0030	0.0053



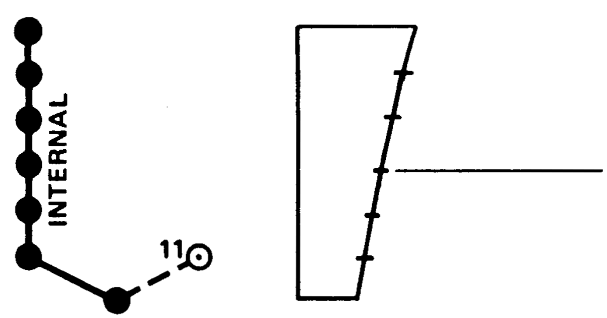
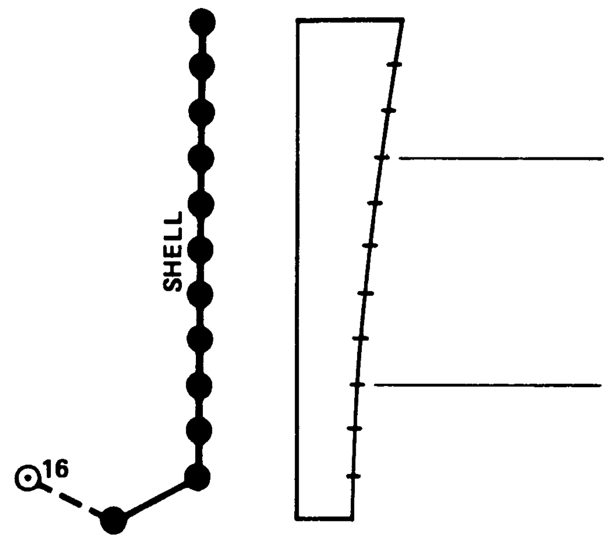


TABLE 3.7(B)-5O RESPONSE ACCELERATIONS (G'S) CONTAINMENT BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.232	0.509	0.411	0.509
2170'-9"	0.211	0.439	0.365	0.439
2135'-0"	0.187	0.356	0.310	0.356
2119'-0"	0.176	0.321	0.285	0.321
2100'-0"	0.163	0.284	0.253	0.284
2080'-0"	0.148	0.253	0.220	0.253
2056'-6"	0.129	0.216	0.181	0.216
2051'-2"	0.125	0.207	0.174	0.207
2039'-0"	0.116	0.186	0.162	0.186
2028'-0"	0.110	0.171	0.154	0.171
2013'-5"	0.107	0.152	0.142	0.152
2000'-0"	0.103	0.139	0.132	0.139
2090'-4"	0.125	0.177	0.146	0.177
2060'-0"	0.113	0.157	0.141	0.157
2047'-6"	0.112	0.154	0.139	0.154
2034'-0"	0.108	0.149	0.137	0.149
2022'-6"	0.104	0.145	0.135	0.145
2012'-0"	0.104	0.142	0.133	0.142
2000'-0"	0.103	0.139	0.132	0.139



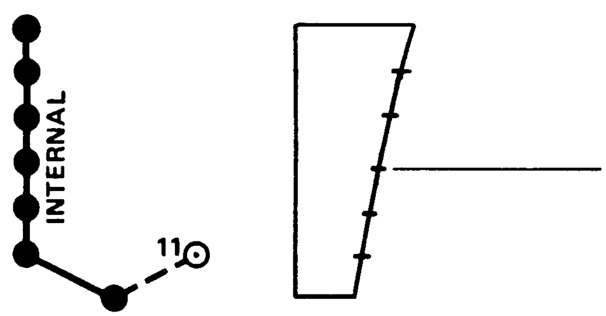
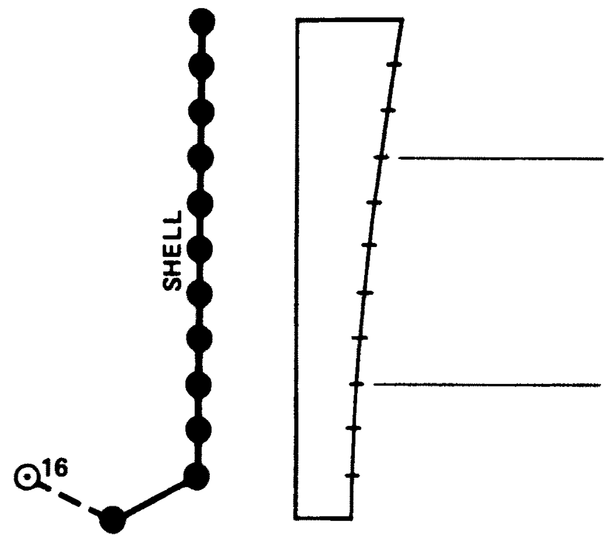


TABLE 3.7(B)-5P RESPONSE ACCELERATIONS (G'S) CONTAINMENT BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.240	0.524	0.546	0.546
2170'-9"	0.217	0.462	0.474	0.474
2135'-0"	0.187	0.394	0.389	0.394
2119'-0"	0.178	0.363	0.351	0.363
2100'-0"	0.164	0.324	0.305	0.324
2080'-0"	0.148	0.281	0.258	0.281
2056'-6"	0.128	0.227	0.204	0.227
2051'-2"	0.124	0.215	0.192	0.215
2039'-0"	0.113	0.187	0.166	0.187
2028'-0"	0.106	0.163	0.156	0.163
2013'-5"	0.101	0.142	0.143	0.143
2000'-0"	0.098	0.134	0.131	0.134
2090'-4"	0.177	0.251	0.223	0.251
2060'-0"	0.131	0.159	0.151	0.159
2047'-6"	0.122	0.154	0.147	0.154
2034'-0"	0.111	0.146	0.144	0.146
2022'-6"	0.102	0.139	0.141	0.141
2012'-0"	0.100	0.135	0.136	0.136
2000'-0"	0.098	0.134	0.131	0.134



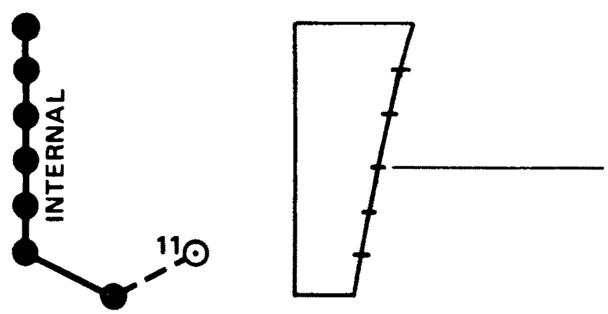
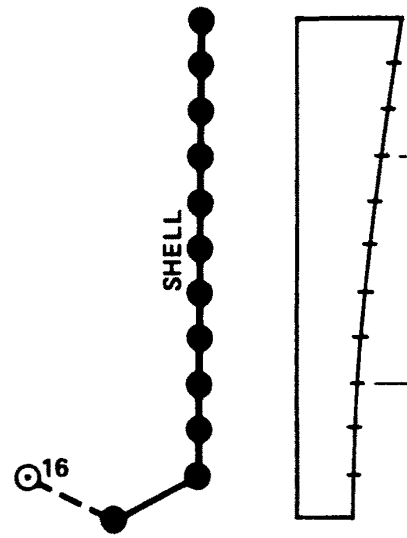


TABLE 3.7(B)-5Q RESPONSE ACCELERATIONS (G'S) CONTAINMENT BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.219	0.269	0.223	0.269
2170'-9"	0.214	0.264	0.218	0.264
2135'-0"	0.200	0.248	0.202	0.248
2119'-0"	0.193	0.240	0.194	0.240
2100'-0"	0.182	0.228	0.182	0.228
2080'-0"	0.169	0.213	0.171	0.213
2056'-6"	0.151	0.192	0.157	0.192
2051'-2"	0.146	0.187	0.156	0.187
2039'-0"	0.140	0.175	0.152	0.175
2028'-0"	0.139	0.164	0.148	0.164
2013'-5"	0.137	0.150	0.143	0.150
2000'-0"	0.136	0.144	0.139	0.144
2090'-4"	0.139	0.150	0.144	0.150
2060'-0"	0.139	0.149	0.143	0.149
2047'-6"	0.138	0.149	0.142	0.149
2034'-0"	0.138	0.148	0.142	0.148
2022'-6"	0.137	0.147	0.141	0.147
2012'-0"	0.137	0.145	0.140	0.145
2000'-0"	0.136	0.144	0.139	0.144



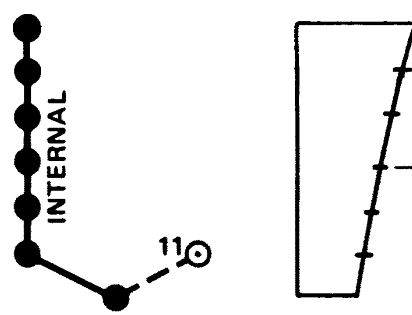
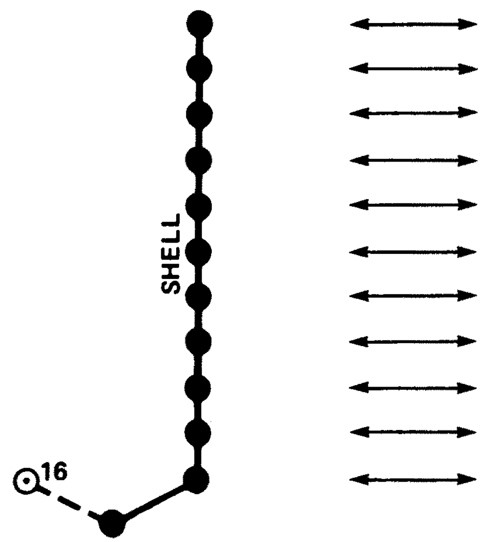


TABLE 3.7(B)-5R RESPONSE INERTIA FORCES (KIPS) CONTAINMENT BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	850	1920	1510	1920
2170'-9"	1530	3320	2700	3320
2135'-0"	1110	2140	1830	2140
2119'-0"	1030	1910	1680	1910
2100'-0"	890	1510	1390	1510
2080'-0"	910	1380	1350	1380
2056'-6"	540	700	740	740
2051'-2"	340	390	430	430
2039'-0"	380	440	500	500
2028'-0"	410	380	500	500
2013'-5"	440	570	470	570
2000'-0"	—	—	—	
2090'-4"	160	220	200	220
2060'-0"	170	240	220	240
2047'-6"	890	1140	1050	1140
2034'-0"	330	390	520	520
2022'-6"	390	690	580	690
2012'-0"	850	1100	940	1100
2000'-0"	—	—	—	



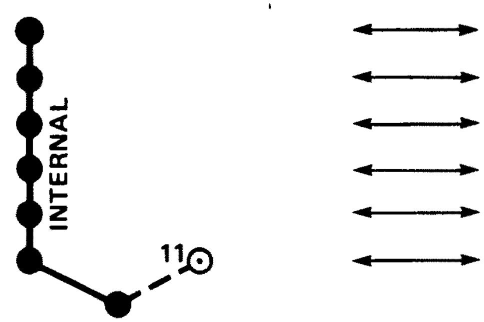
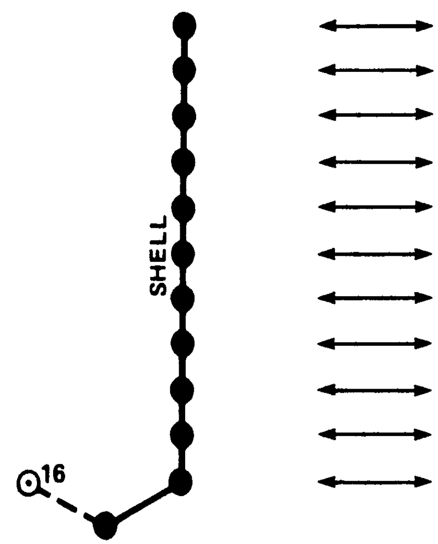


TABLE 3.7(B)-5S RESPONSE INERTIA FORCES (KIPS) CONTAINMENT BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	890	1940	2020	2020
2170'-9"	1620	3390	3530	3530
2135'-0"	1140	2350	2320	2350
2119'-0"	1060	2140	2070	2140
2100'-0"	900	1780	1680	1780
2080'-0"	910	1730	1570	1730
2056'-6"	530	930	810	930
2051'-2"	310	550	470	550
2039'-0"	380	630	540	630
2028'-0"	380	610	520	610
2013'-5"	400	570	470	570
2000'-0"	—	—	—	
2090'-4"	210	250	240	250
2060'-0"	190	240	230	240
2047'-6"	1180	1470	1050	1470
2034'-0"	370	350	360	370
2022'-6"	350	730	820	820
2012'-0"	900	3860	860	3860
2000'-0"	—	—	—	



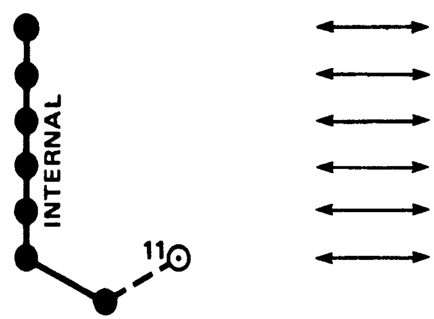


TABLE 3.7(B)-5T RESPONSE INERTIA FORCES (KIPS) CONTAINMENT BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	820	1010	830	1010
2170'-9"	1600	1990	1610	1990
2135'-0"	1190	1500	1190	1500
2119'-0"	1140	1430	1140	1430
2100'-0"	990	1260	990	1260
2080'-0"	1030	1310	1020	1310
2056'-6"	610	780	590	780
2051'-2"	360	460	350	460
2039'-0"	440	580	430	580
2028'-0"	450	590	500	590
2013'-5"	470	610	530	610
2000'-0"	—	—	—	
2090'-4"	200	200	200	200
2060'-0"	210	230	210	230
2047'-6"	670	700	680	700
2034'-0"	490	510	490	510
2022'-6"	870	910	880	910
2012'-0"	910	950	910	950
2000'-0"	—	—	—	

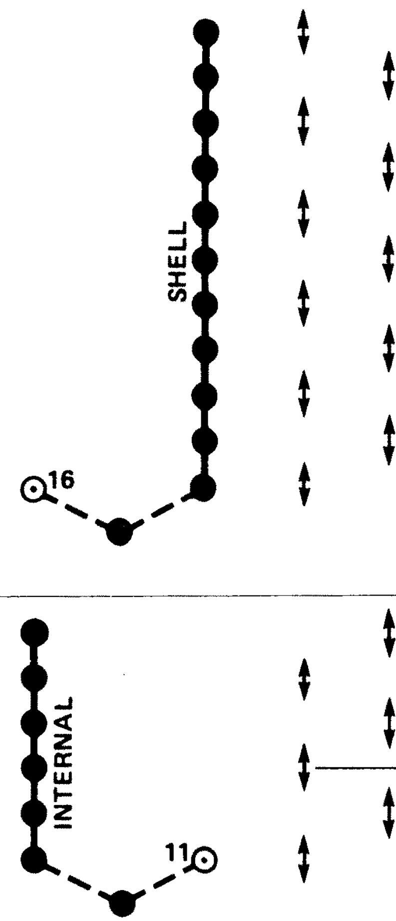
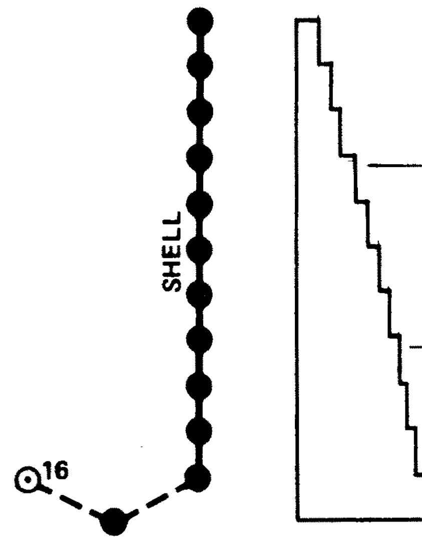


TABLE 3.7(B)-5U RESPONSE SHEAR FORCES (KIPS) CONTAINMENT BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"				
	850	1920	1510	1920
2170'-9"				
	2380	5240	4210	5240
2135'-0"				
	3490	7380	6040	7380
2119'-0"				
	4520	9290	7720	9290
2100'-0"				
	5410	10,800	9110	10,800
2080'-0"				
	6320	12,180	10,460	12,180
2056'-6"				
	6860	12,880	11,200	12,880
2051'-2"				
	7200	13,270	11,630	13,270
2039'-0"				
	7580	13,710	12,130	13,710
2028'-0"				
	7990	14,090	12,630	14,090
2013'-5"				
	8430	14,660	13,100	14,660
2000'-0"				
2090'-4"				
	160	220	200	220
2060'-0"				
	330	460	420	460
2047'-6"				
	1220	1600	1470	1600
2034'-0"				
	1550	1990	1990	1990
2022'-6"				
	1940	2680	2570	2680
2012'-0"				
	2790	3780	3510	3780
2000'-0"				



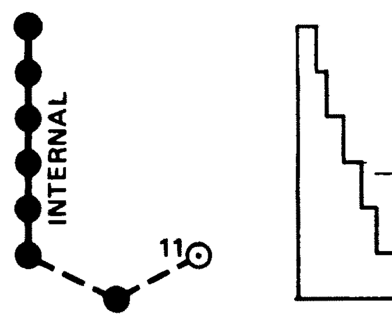
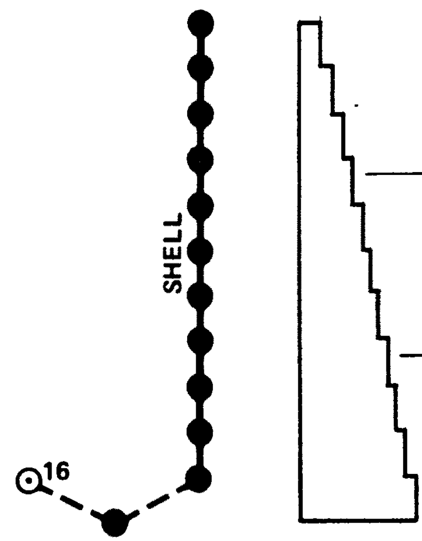


TABLE 3.7(B)-5V RESPONSE SHEAR FORCES (KIPS) CONTAINMENT BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"				
	890	1940	2020	2020
2170'-9"				
	2510	5330	5550	5550
2135'-0"				
	3650	7680	7870	7870
2119'-0"				
	4710	9820	9940	9940
2100'-0"				
	5610	11,600	11,620	11,620
2080'-0"				
	6520	13,330	13,190	13,330
2056'-6"				
	7050	14,260	14,000	14,260
2051'-2"				
	7360	14,810	14,470	14,810
2039'-0"				
	7740	15,440	15,010	15,440
2028'-0"				
	8120	16,050	15,530	16,050
2013'-5"				
	8520	16,620	16,000	16,620
2000'-0"				
2090'-4"				
	210	250	240	250
2060'-0"				
	400	490	470	490
2047'-6"				
	1580	1960	1520	1960
2034'-0"				
	1950	2310	1880	2310
2022'-6"				
	2300	3040	2700	3040
2012'-0"				
	3200	6900	3560	6900
2000'-0"				



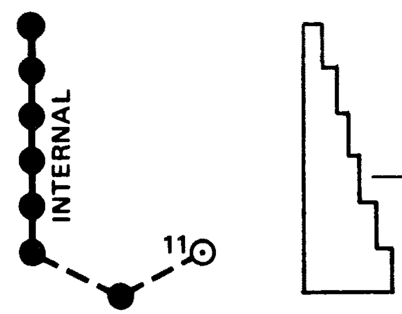
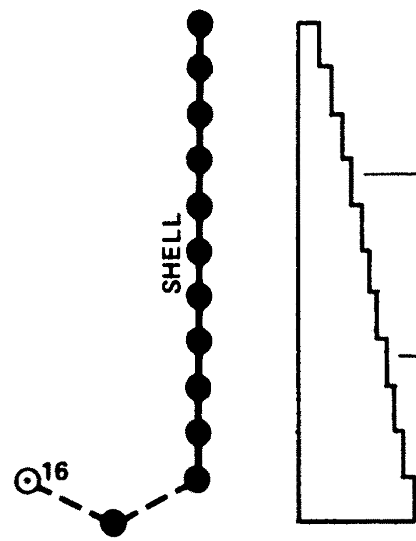


TABLE 3.7(B)-5W RESPONSE AXIAL FORCES (KIPS) CONTAINMENT BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"				
	820	1010	830	1010
2170'-9"				
	2420	3000	2440	3000
2135'-0"				
	3610	4500	3630	4500
2119'-0"				
	4750	5930	4770	5930
2100'-0"				
	5740	7190	5760	7190
2080'-0"				
	6770	8500	6780	8500
2056'-6"				
	7380	9280	7370	9280
2051'-2"				
	7740	9740	7720	9740
2039'-0"				
	8180	10,320	8150	10,320
2028'-0"				
	8630	10,910	8650	10,910
2013'-5"				
	9100	11,520	9180	11,520
2000'-0"				
2090'-4"				
	200	200	200	200
2060'-0"				
	410	430	410	430
2047'-6"				
	1080	1130	1090	1130
2034'-0"				
	1570	1640	1580	2550
2022'-6"				
	2440	2550	2460	3500
2012'-0"				
	3350	3500	3370	
2000'-0"				



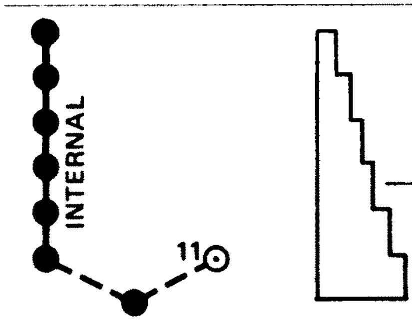
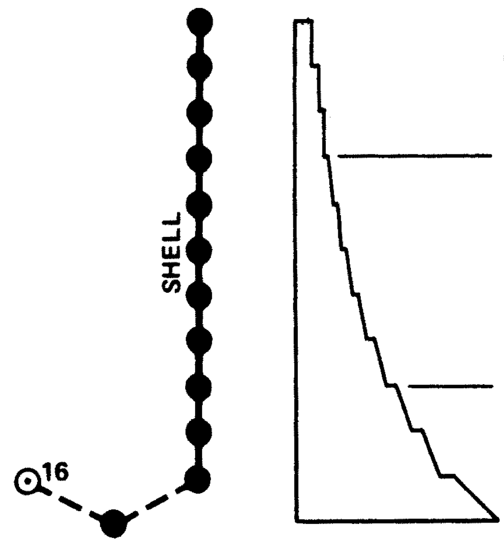


TABLE 3.7(B)-5X RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) CONTAINMENT BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0/0.0039	0/0.0109	0/0.0063	0/0.0109
2170'-9"	0.0323/0.0390	0.0766/0.0999	0.0590/0.0758	0.0766/0.0999
2135'-0"	0.1241/0.1310	0.2873/0.3114	0.2240/0.2395	0.2873/0.3114
2119'-0"	0.1868/0.1932	0.4296/0.4520	0.3361/0.3504	0.4296/0.4520
2100'-0"	0.2791/0.2852	0.6284/0.6492	0.4971/0.5106	0.6284/0.6492
2080'-0"	0.3936/0.4000	0.8651/0.8865	0.6929/0.7070	0.8651/0.8865
2056'-6"	0.5486/0.5525	1.1729/1.1850	0.9528/0.9610	1.1729/1.1850
2051'-2"	0.5892/0.5913	1.2538/1.2609	1.0206/1.0263	1.2538/1.2609
2039'-0"	0.6787/0.6815	1.4224/1.4306	1.1668/1.1725	1.4224/1.4306
2028'-0"	0.7650/0.7680	1.5813/1.5894	1.3058/1.3114	1.5813/1.5894
2013'-5"	0.8844/0.8875	1.7950/1.8020	1.4955/1.5008	1.7950/1.8020
2000'-0"	1.0004	1.9953	1.6766	1.9953
2090'-4"	0/0.0012	0/0.0017	0/0.0014	0/0.0017
2060'-0"	0.0026/0.0029	0.0035/0.0037	0.0031/0.0033	0.0035/0.0037
2047'-6"	0.0070/0.0080	0.0093/0.0102	0.0084/0.0093	0.0093/0.0102
2034'-0"	0.0234/0.0243	0.0293/0.0301	0.0286/0.0291	0.0293/0.0301
2022'-6"	0.0416/0.0432	0.0525/0.0540	0.0515/0.0526	0.0525/0.0540
2012'-0"	0.0630/0.0652	0.0821/0.0830	0.0789/0.0790	0.0821/0.0830
2000'-0"	0.0951	0.1261	0.1212	0.1261



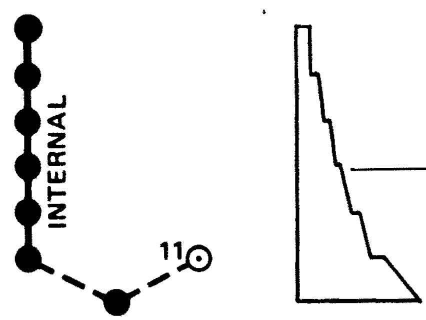
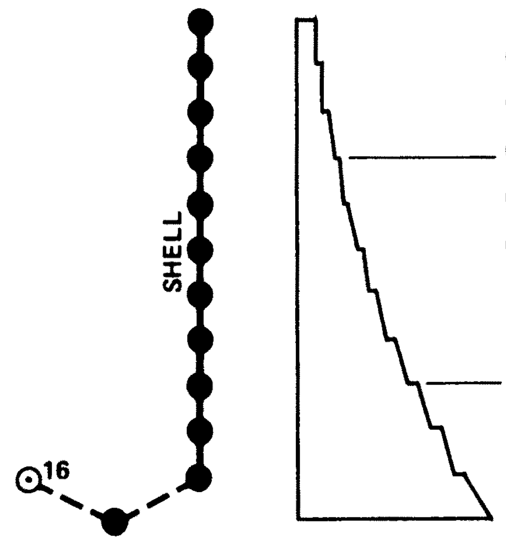


TABLE 3.7(B)-5Y RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) CONTAINMENT BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0/0.0048	0/0.0092	0/0.0078	0/0.0092
2170'-9"	0.0345/0.0422	0.0766/0.0987	0.0802/0.1034	0.0802/0.1034
2135'-0"	0.1319/0.1402	0.2885/0.3116	0.3016/0.3259	0.3016/0.3259
2119'-0"	0.1985/0.2064	0.4322/0.4535	0.4518/0.4741	0.4518/0.4741
2100'-0"	0.2958/0.3032	0.6345/0.6546	0.6631/0.6840	0.6631/0.6840
2080'-0"	0.4155/0.4234	0.8834/0.9024	0.9165/0.9383	0.9165/0.9383
2056'-6"	0.5766/0.5815	1.2156/1.2270	1.2481/1.2609	1.2481/1.2609
2051'-2"	0.6192/0.6219	1.3030/1.3095	1.3355/1.3428	1.3355/1.3428
2039'-0"	0.7115/0.7148	1.4898/1.4976	1.5190/1.5276	1.5190/1.5276
2028'-0"	0.7999/0.8034	1.6676/1.6754	1.6927/1.7011	1.6927/1.7011
2013'-5"	0.9220/0.9257	1.9094/1.9171	1.9277/1.9357	1.9277/1.9357
2000'-0"	1.0402	2.1403	2.1501	2.1501
2090'-4"	0/0.0016	0/0.0021	0/0.0020	0/0.0021
2060'-0"	0.0034/0.0040	0.0043/0.0054	0.0040/0.0053	0.0043/0.0054
2047'-6"	0.0090/0.0100	0.0111/0.0124	0.0107/0.0128	0.0111/0.0124
2034'-0"	0.0313/0.0318	0.0362/0.0384	0.0297/0.0307	0.0362/0.0384
2022'-6"	0.0542/0.0561	0.0609/0.0641	0.0517/0.0535	0.0609/0.0641
2012'-0"	0.0786/0.0791	0.0930/0.0951	0.0789/0.0795	0.0930/0.0951
2000'-0"	0.1175	0.1549	0.1220	0.1549



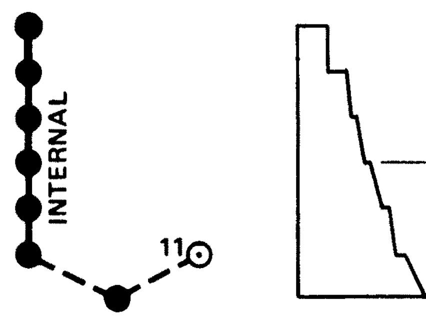
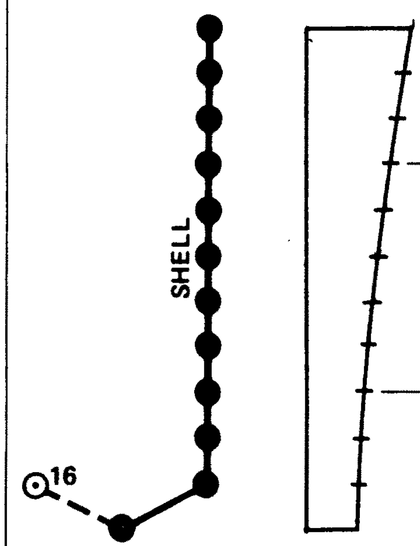


TABLE 3.7(B)-5Z RESPONSE DISPLACEMENTS (INCHES) CONTAINMENT BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.190	0.247	0.213	0.247
2170'-9"	0.170	0.212	0.185	0.212
2135'-0"	0.147	0.170	0.151	0.170
2119'-0"	0.137	0.152	0.136	0.152
2100'-0"	0.124	0.129	0.117	0.129
2080'-0"	0.110	0.104	0.096	0.110
2056'-6"	0.093	0.075	0.071	0.093
2051'-2"	0.089	0.069	0.065	0.089
2039'-0"	0.080	0.054	0.052	0.080
2028'-0"	0.072	0.041	0.041	0.072
2013'-5"	0.062	0.025	0.027	0.062
2000'-0"	0.053	0.012	0.014	0.053
2090'-4"	0.080	0.042	0.044	0.080
2060'-0"	0.071	0.031	0.033	0.071
2047'-6"	0.067	0.027	0.029	0.067
2034'-0"	0.063	0.023	0.025	0.063
2022'-6"	0.060	0.019	0.021	0.060
2012'-0"	0.057	0.016	0.018	0.057
2000'-0"	0.053	0.012	0.014	0.053



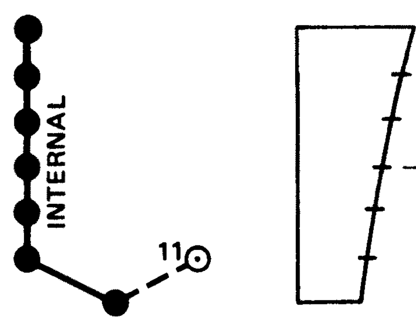
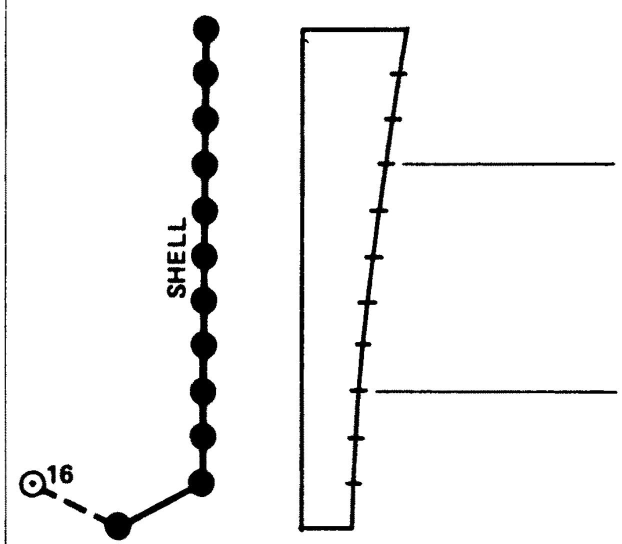


TABLE 3.7(B)-5AA RESPONSE DISPLACEMENTS (INCHES) CONTAINMENT BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.201	0.271	0.274	0.274
2170'-9"	0.180	0.234	0.236	0.236
2135'-0"	0.156	0.190	0.191	0.191
2119'-0"	0.145	0.170	0.171	0.171
2100'-0"	0.132	0.145	0.146	0.146
2080'-0"	0.118	0.118	0.119	0.119
2056'-6"	0.101	0.086	0.088	0.101
2051'-2"	0.097	0.079	0.081	0.097
2039'-0"	0.088	0.063	0.065	0.088
2028'-0"	0.080	0.049	0.051	0.080
2013'-5"	0.070	0.031	0.033	0.070
2000'-0"	0.060	0.015	0.017	0.060
2090'-4"	0.093	0.057	0.059	0.093
2060'-0"	0.081	0.042	0.045	0.081
2047'-6"	0.077	0.037	0.039	0.077
2034'-0"	0.072	0.031	0.033	0.072
2022'-6"	0.068	0.025	0.028	0.068
2012'-0"	0.064	0.020	0.023	0.064
2000'-0"	0.060	0.015	0.017	0.060



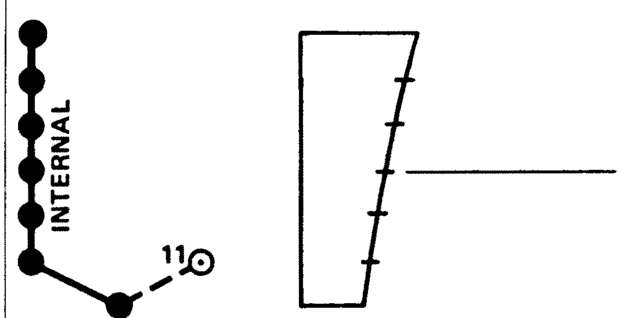
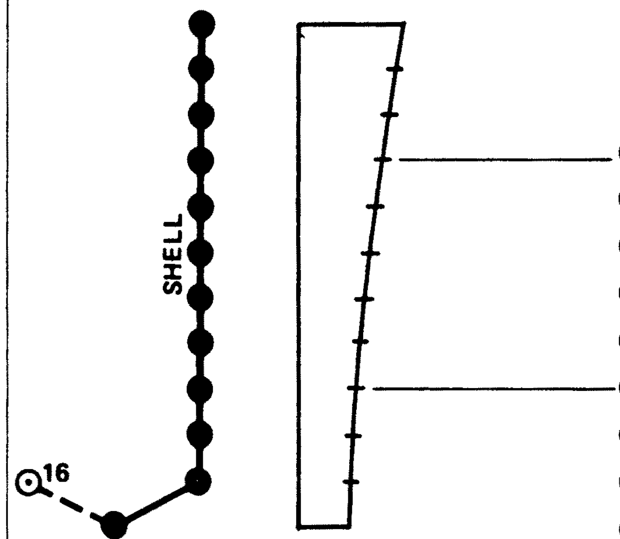


TABLE 3.7(B)-5AB RESPONSE DISPLACEMENTS (INCHES) CONTAINMENT BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-17

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2206'-6"	0.0115	0.0163	0.0125	0.0163
2170'-9"	0.0112	0.0158	0.0120	0.0158
2135'-0"	0.0103	0.0144	0.0108	0.0144
2119'-0"	0.0098	0.0136	0.0102	0.0136
2100'-0"	0.0090	0.0125	0.0093	0.0125
2080'-0"	0.0081	0.0110	0.0081	0.0110
2056'-6"	0.0067	0.0090	0.0065	0.0090
2051'-2"	0.0064	0.0085	0.0061	0.0085
2039'-0"	0.0056	0.0073	0.0051	0.0073
2028'-0"	0.0048	0.0062	0.0042	0.0062
2013'-5"	0.0037	0.0045	0.0029	0.0045
2000'-0"	0.0028	0.0030	0.0016	0.0030
2090'-4"	0.0039	0.0052	0.0027	0.0052
2060'-0"	0.0036	0.0043	0.0025	0.0043
2047'-6"	0.0035	0.0041	0.0024	0.0041
2034'-0"	0.0034	0.0038	0.0023	0.0038
2022'-6"	0.0032	0.0036	0.0021	0.0036
2012'-0"	0.0030	0.0033	0.0019	0.0033
2000'-0"	0.0026	0.0030	0.0016	0.0030



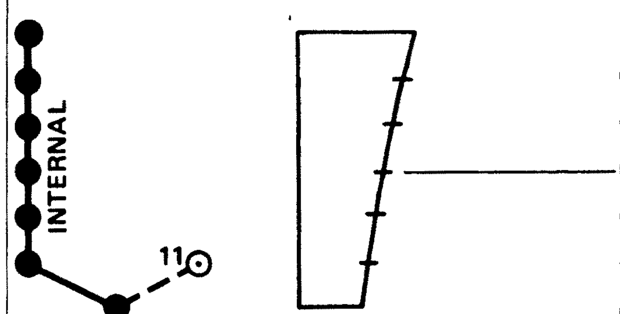


TABLE 3.7(B)-6A RESPONSE ACCELERATIONS (G'S) FUEL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.606	0.635	0.461	0.635
2083'-6"	0.504	0.521	0.362	0.521
2047'-6"	0.331	0.330	0.270	0.331
2026'-0"	0.260	0.287	0.255	0.287
2000'-0"	0.202	0.271	0.252	0.271

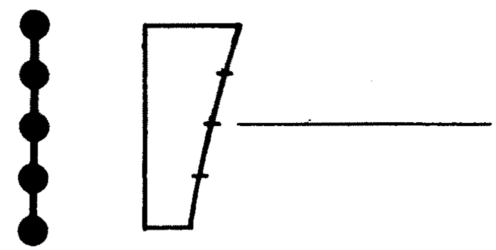


TABLE 3.7(B)-6B RESPONSE ACCELERATIONS (G'S) FUEL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

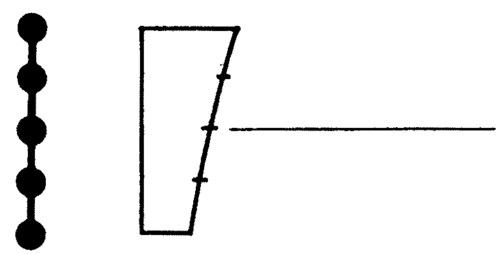
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.418	0.793	0.586	 0.793
2083'-6"	0.371	0.644	0.529	0.644
2047'-6"	0.307	0.367	0.426	0.426
2026'-0"	0.272	0.282	0.375	0.375
2000'-0"	0.242	0.265	0.318	0.318

TABLE 3.7(B)-6C RESPONSE ACCELERATIONS (G'S) FUEL BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.328	0.330	0.389	0.389
2083'-6"	0.327	0.321	0.382	0.382
2047'-6"	0.322	0.295	0.362	0.362
2026'-0"	0.320	0.284	0.353	0.353
2000'-0"	0.316	0.276	0.337	0.337

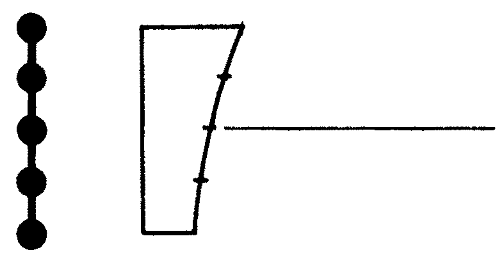


TABLE 3.7(B)-6D RESPONSE INERTIA FORCES (KIPS) FUEL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	3100	3150	2340	3150
2083'-6"	2300	2340	1640	2340
2047'-6"	3300	3400	2150	3400
2026'-0"	4100	4030	3910	4100
2000'-0"	—	—	—	

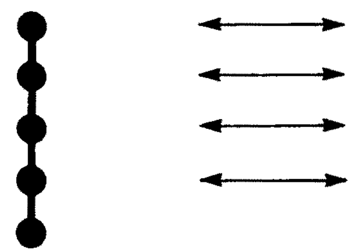


TABLE 3.7(B)-6E RESPONSE INERTIA FORCES (KIPS) FUEL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	2090	3950	2950	3950
2083'-6"	1650	2890	2380	2890
2047'-6"	3010	3820	4400	4400
2026'-0"	4430	4100	6180	6180
2000'-0"	—	—	—	

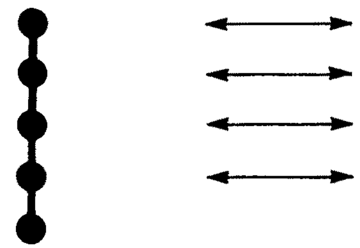


TABLE 3.7(B)-6F RESPONSE INERTIA FORCES (KIPS) FUEL BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

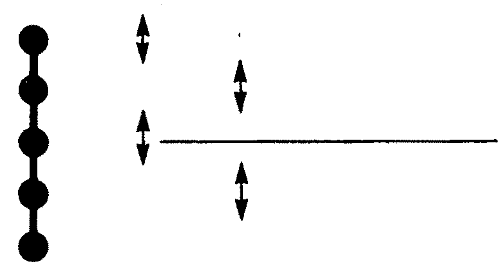
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	1630	1670	2000	 2000
2083'-6"	1450	1460	1770	1770
2047'-6"	3280	3070	3860	3860
2026'-0"	5170	4700	5980	5980
2000'-0"	—	—	—	

TABLE 3.7(B)-6G RESPONSE SHEAR FORCES (KIPS) FUEL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"				
	3100	3150	2350	3150
2083'-6"				
	5400	5490	3980	5490
2047'-6"				
	8700	8890	6130	8890
2026'-0"				
	12,800	12,920	10,040	12,920
2000'-0"				

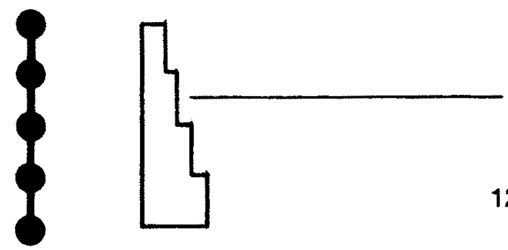


TABLE 3.7(B)-6H RESPONSE SHEAR FORCES (KIPS) FUEL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"				
	2090	3950	2950	3950
2083'-6"				
	3740	6840	5330	6840
2047'-6"				
	6750	10,660	9730	10,660
2026'-0"				
	11,180	14,760	15,910	15,910
2000'-0"				

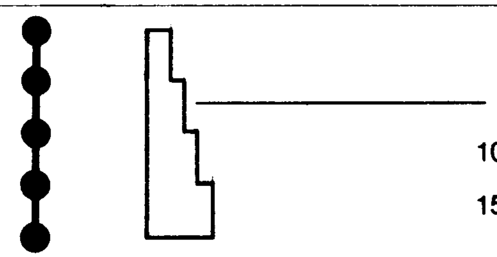


TABLE 3.7(B)-6I RESPONSE AXIAL FORCES (KIPS) FUEL BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"				
	1630	1670	2000	2000
2083'-6"				
	3080	3130	3770	3770
2047'-6"				
	6360	6200	7630	7630
2026'-0"				
	11,530	10,900	13,610	13,610
2000'-0"				

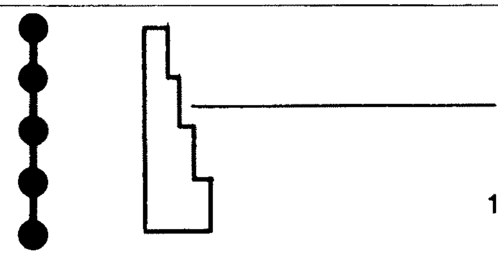


TABLE 3.7(B)-6J RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) FUEL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

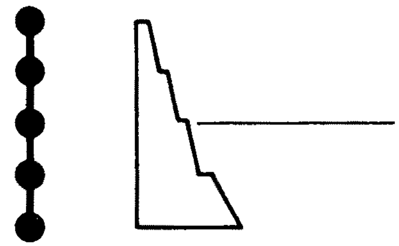
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2106'-6"	0 /0.0269	0/0.0240	0/0.0230		0/0.0269
2083'-6"	0.0981/0.1317	0.0965/0.1257	0.0767/0.1050		0.0981/0.1317
2047'-6"	0.3247/0.3776	0.3234/0.3661	0.2483/0.2916		0.3247/0.3776
2026'-0"	0.5635/0.6161	0.5574/0.5972	0.4212/0.4631		0.5635/0.6161
2000'-0"	0.9383	0.9330	0.6718		0.9383

TABLE 3.7(B)-6K RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) FUEL BUILDING SSE
EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

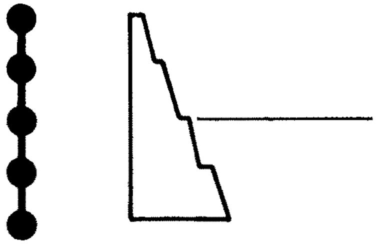
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2106'-6"	0/0.0149	0/0.0215	0/0.0240		0/0.0240
2083'-6"	0.0576/0.0762	0.1124/0.1436	0.0855/0.1204		0.1124/0.1436
2047'-6"	0.2051/0.2239	0.3896/0.4279	0.2899/0.3285		0.3896/0.4279
2026'-0"	0.3640/0.3828	0.6571/0.6938	0.5099/0.5279		0.6571/0.6938
2000'-0"	0.6601	1.0776	0.9415		1.0776

TABLE 3.7(B)-6L RESPONSE DISPLACEMENTS (INCHES) FUEL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.311	0.124	0.115	0.311
2083'-6"	0.275	0.101	0.092	0.275
2047'-6"	0.216	0.060	0.056	0.216
2026'-0"	0.186	0.041	0.039	0.186
2000'-0"	0.154	0.021	0.022	0.154

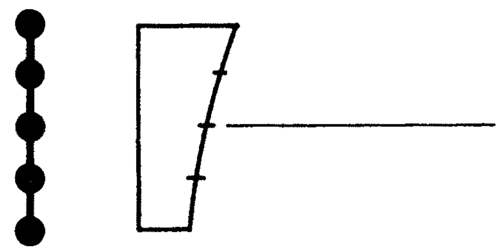


TABLE 3.7(B)-6M RESPONSE DISPLACEMENTS (INCHES) FUEL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

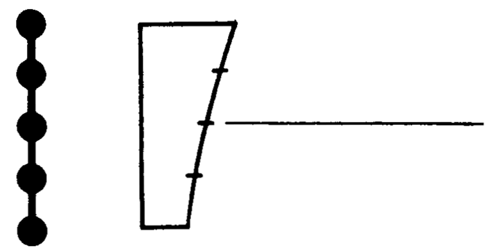
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.473	0.212	0.293	 0.473
2083'-6"	0.393	0.172	0.245	0.393
2047'-6"	0.260	0.097	0.160	0.260
2026'-0"	0.184	0.065	0.117	0.184
2000'-0"	0.095	0.031	0.067	0.095

TABLE 3.7(B)-6N RESPONSE DISPLACEMENTS (INCHES) FUEL BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.050	0.017	0.045	0.050
2083'-6"	0.049	0.016	0.044	0.049
2047'-6"	0.047	0.013	0.041	0.047
2026'-0"	0.046	0.012	0.039	0.046
2000'-0"	0.044	0.010	0.036	0.044

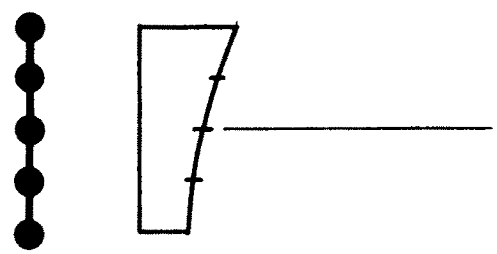


TABLE 3.7(B)-6O RESPONSE ACCELERATIONS (G'S) FUEL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

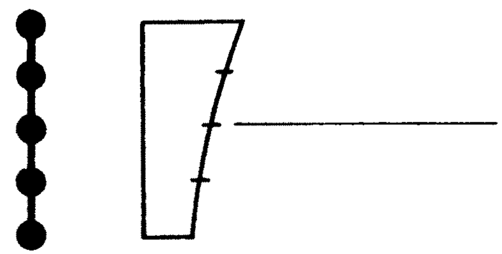
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.313	0.363	0.269	 0.363
2083'-6"	0.249	0.297	0.214	0.297
2047'-6"	0.172	0.188	0.144	0.188
2026'-0"	0.133	0.156	0.138	0.156
2000'-0"	0.109	0.142	0.134	0.142

TABLE 3.7(B)-6P RESPONSE ACCELERATIONS (G'S) FUEL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.252	0.462	0.384	0.462
2083'-6"	0.221	0.372	0.319	0.372
2047'-6"	0.173	0.205	0.207	0.207
2026'-0"	0.149	0.151	0.187	0.187
2000'-0"	0.132	0.134	0.163	0.163

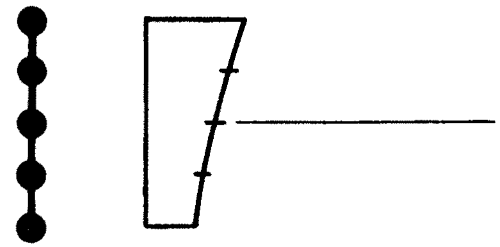


TABLE 3.7(B)-6Q RESPONSE ACCELERATIONS (G'S) FUEL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

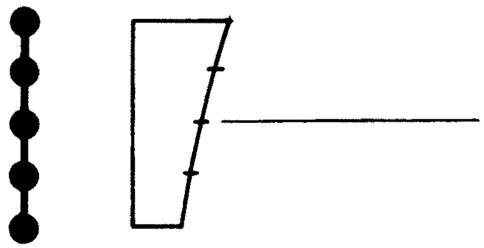
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.191	0.168	0.214	 0.214
2083'-6"	0.188	0.163	0.210	0.210
2047'-6"	0.180	0.152	0.200	0.200
2026'-0"	0.176	0.149	0.195	0.195
2000'-0"	0.169	0.143	0.186	0.186

TABLE 3.7(B)-6R RESPONSE INERTIA FORCES (KIPS) FUEL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	1540	1790	1350	1790
2083'-6"	1140	1340	970	1340
2047'-6"	1670	1930	1250	1930
2026'-0"	2010	2290	1890	2290
2000'-0"	—	—	—	

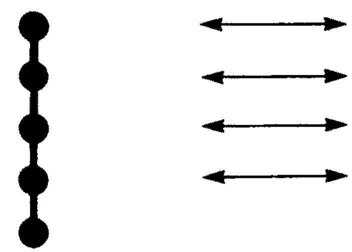


TABLE 3.7(B)-6S RESPONSE INERTIA FORCES (KIPS) FUEL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	1280	2310	1490	2310
2083'-6"	1000	1660	1890	1890
2047'-6"	1780	2130	2100	2130
2026'-0"	2350	2240	2490	2490
2000'-0"	—	—	—	

TABLE 3.7(B)-6T RESPONSE INERTIA FORCES (KIPS) FUEL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	1000	860	1070	1070
2083'-6"	800	740	950	950
2047'-6"	1800	1560	2060	2060
2026'-0"	2900	2400	3190	3190
2000'-0"	—	—	—	

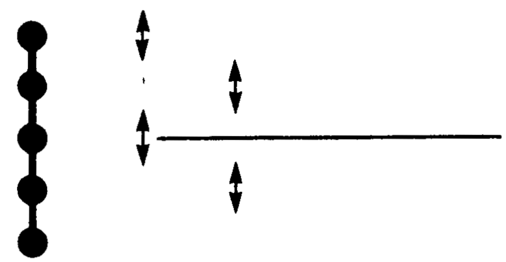


TABLE 3.7(B)-6U RESPONSE SHEAR FORCES (KIPS) FUEL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

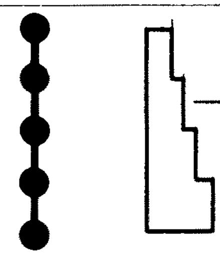
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"				
	1540	1790	1350	1790
2083'-6"				
	2680	3130	2320	3130
2047'-6"				
	4350	5060	3570	5060
2026'-0"				
	6360	7350	5460	7350
2000'-0"				

TABLE 3.7(B)-6V RESPONSE SHEAR FORCES (KIPS) FUEL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

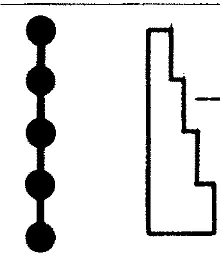
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"				
	1280	2310	1490	2310
2083'-6"				
	2280	3970	3380	3970
2047'-6"				
	4060	6100	5480	6100
2026'-0"				
	6410	8340	7970	8340
2000'-0"				

TABLE 3.7(B)-6W RESPONSE AXIAL FORCES (KIPS) FUEL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

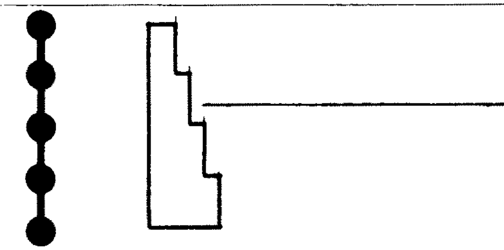
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"				
	1000	860	1070	1070
2083'-6"				
	1800	1600	2020	2020
2047'-6"				
	3600	3160	4080	4080
2026'-0"				
	6500	5560	7270	7270
2000'-0"				

TABLE 3.7(B)-6X RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) FUEL BUILDING OBE
NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0/0.0144	0/0.0139	0/0.0126	0/0.0144
2083'-6"	0.0483/0.0657	0.0550/0.0717	0.0437/0.0593	0.0550/0.0717
2047'-6"	0.1608/0.1860	0.1847/0.2085	0.1426/0.1665	0.1847/0.2085
2026'-0"	0.2795/0.3043	0.3175/0.3394	0.2431/0.2661	0.3175/0.3394
2000'-0"	0.4662	0.5306	0.3915	0.5306

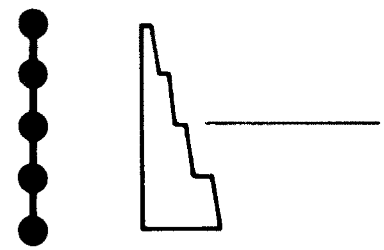


TABLE 3.7(B)-6Y RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) FUEL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

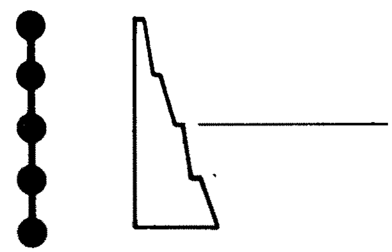
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2106'-6"	0/0.0083	0/0.0127	0/0.0151		0/0.0151
2083'-6"	0.0348/0.0445	0.0657/0.0841	0.0585/0.0805		0.0657/0.0841
2047'-6"	0.1247/0.0.1352	0.2272/0.2493	0.1931/0.2206		0.2272/0.2493
2026'-0"	0.2225/0.2335	0.3805/0.4015	0.3269/0.3451		0.3805/0.4015
2000'-0"	0.4001	0.6183	0.5522		0.6183

TABLE 3.7(B)-6Z RESPONSE DISPLACEMENT (INCHES) FUEL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.119	0.069	0.060	0.119
2083'-6"	0.104	0.056	0.048	0.104
2047'-6"	0.079	0.033	0.028	0.079
2026'-0"	0.066	0.023	0.019	0.066
2000'-0"	0.052	0.011	0.009	0.052

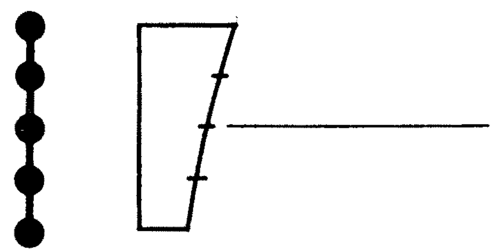


TABLE 3.7(B)-6AA RESPONSE DISPLACEMENT (INCHES) FUEL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.206	0.114	0.154	0.206
2083'-6"	0.171	0.092	0.127	0.171
2047'-6"	0.111	0.050	0.078	0.111
2026'-0"	0.079	0.033	0.054	0.079
2000'-0"	0.040	0.014	0.027	0.040

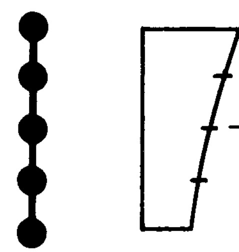


TABLE 3.7(B)-6AB RESPONSE DISPLACEMENT (INCHES) FUEL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-18

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2106'-6"	0.026	0.008	0.020	0.026
2083'-6"	0.025	0.008	0.020	0.025
2047'-6"	0.024	0.006	0.018	0.024
2026'-0"	0.023	0.005	0.017	0.023
2000'-0"	0.022	0.004	0.016	0.022

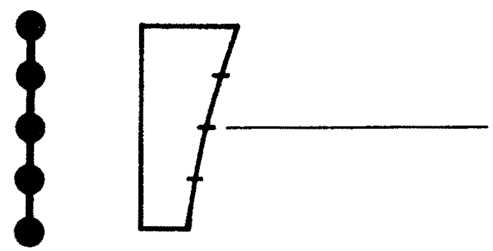


TABLE 3.7(B)-7A RESPONSE ACCELERATIONS (G'S) AUXILIARY/CONTROL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.467	0.499	0.322	0.499
2090'-0"	0.430	0.463	0.304	0.463
2087'-2"	0.417	0.450	0.299	0.450
2073'-6"	0.386	0.420	0.285	0.420
2065'-0"	0.358	0.391	0.273	0.391
2047'-6"	0.306	0.320	0.244	0.320
2032'-0"	0.262	0.286	0.227	0.286
2026'-0"	0.244	0.282	0.223	0.282
2016'-0"	0.216	0.275	0.219	0.275
2000'-0"	0.196	0.261	0.216	0.261

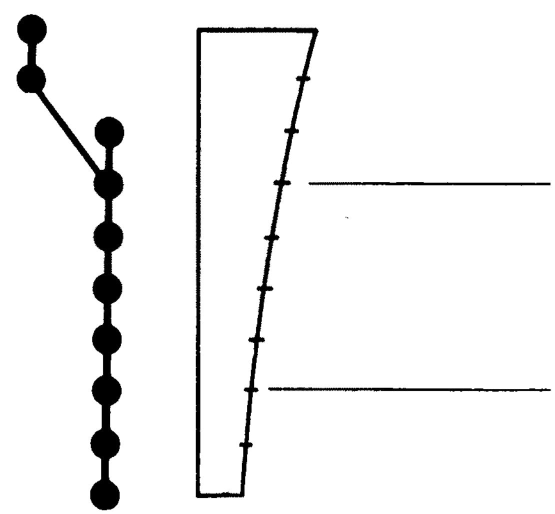


TABLE 3.7(B)-7B RESPONSE ACCELERATIONS (G'S) AUXILIARY/CONTROL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.506	0.552	0.364	0.552
2090'-0"	0.475	0.523	0.342	0.523
2087'-2"	0.453	0.496	0.320	0.496
2073'-6"	0.409	0.448	0.283	0.448
2065'-0"	0.378	0.416	0.257	0.416
2047'-6"	0.333	0.336	0.246	0.336
2032'-0"	0.291	0.295	0.239	0.295
2026'-0"	0.273	0.290	0.236	0.290
2016'-0"	0.242	0.281	0.231	0.281
2000'-0"	0.209	0.267	0.226	0.267

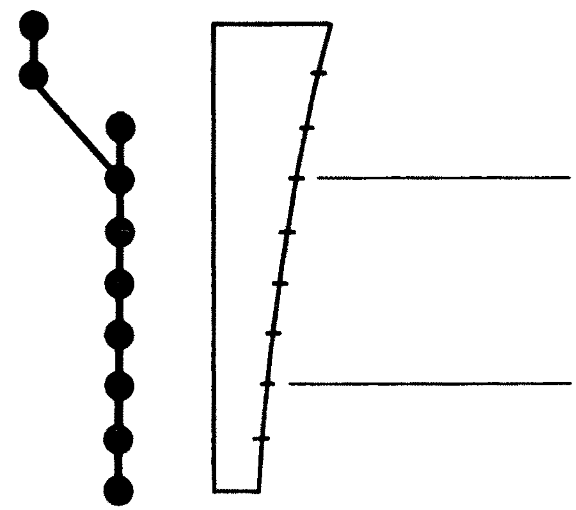


TABLE 3.7(B)-7C RESPONSE ACCELERATIONS (G'S) AUXILIARY/CONTROL BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.256	0.275	0.258	0.275
2090'-0"	0.255	0.273	0.258	0.273
2087'-2"	0.255	0.271	0.257	0.271
2073'-6"	0.254	0.267	0.256	0.267
2065'-0"	0.253	0.262	0.256	0.262
2047'-6"	0.250	0.252	0.253	0.253
2032'-0"	0.247	0.250	0.250	0.250
2026'-0"	0.246	0.248	0.249	0.249
2016'-0"	0.244	0.246	0.247	0.247
2000'-0"	0.240	0.245	0.244	0.245

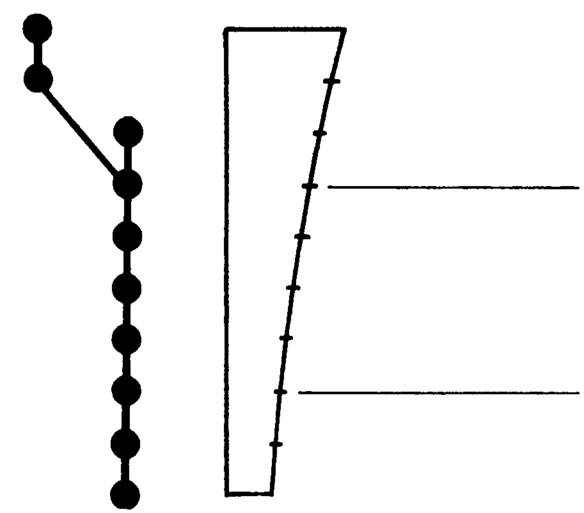


TABLE 3.7(B)-7D RESPONSE INERTIA FORCES (KIPS) AUXILIARY/CONTROL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	230	240	160	240
2090'-0"	640	680	450	680
2087'-2"	1280	1370	920	1370
2073'-6"	3520	3810	2590	3810
2065'-0"	1730	1860	1300	1860
2047'-6"	3780	4110	3150	4110
2032'-0"	1550	1670	1400	1670
2026'-0"	3040	2940	2600	3040
2016'-0"	2250	2130	2010	2250
2000'-0"	—	—	—	

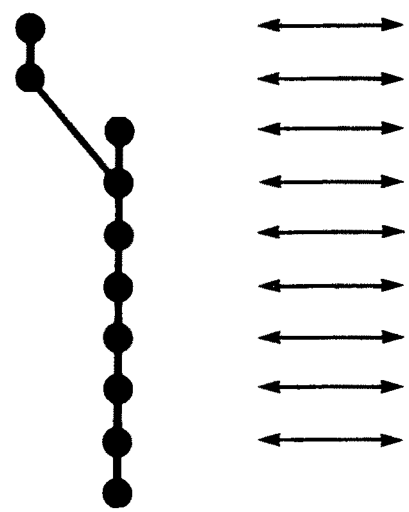


TABLE 3.7(B)-7E RESPONSE INERTIA FORCES (KIPS) AUXILIARY/CONTROL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	250	270	180	270
2090'-0"	700	780	510	780
2087'-2"	1390	1520	980	1520
2073'-6"	3730	4090	2590	4090
2065'-0"	1820	2000	1230	2000
2047'-6"	3980	4330	2600	4330
2032'-0"	1770	1760	1460	1770
2026'-0"	3390	3100	2880	3390
2016'-0"	2530	2250	2390	2530
2000'-0"	—	—	—	

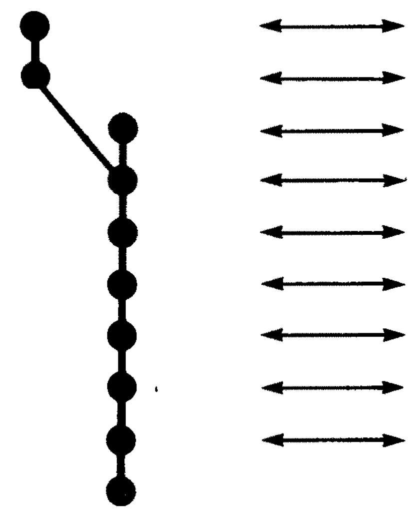
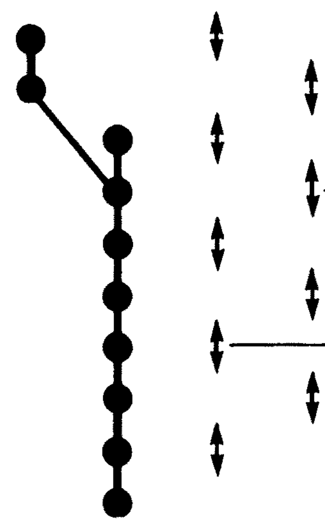


TABLE 3.7(B)-7F RESPONSE INERTIA FORCES (KIPS) AUXILIARY/CONTROL BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19



MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	130	130	120	130
2090'-0"	380	390	380	390
2087'-2"	790	800	770	800
2073'-6"	2330	2330	2290	2330
2065'-0"	1230	1210	1210	1230
2047'-6"	3240	3110	3180	3240
2032'-0"	1600	1530	1580	1600
2026'-0"	3070	3010	3020	3070
2016'-0"	2560	2510	2520	2560
2000'-0"	—	—	—	

TABLE 3.7(B)-7G RESPONSE SHEAR FORCES (KIPS) AUXILIARY/CONTROL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"				
	230	240	160	240
2090'-0"				
	870	920	610	920
2087'-2"				
	1280	1370	920	1370
2073'-6"				
	5670	6100	4120	6100
2065'-0"				
	7400	7960	5420	7960
2047'-6"				
	11,180	12,070	8570	12,070
2032'-0"				
	12,730	13,740	9970	13,740
2026'-0"				
	15,770	16,680	12,570	16,680
2016'-0"				
	18,020	18,810	14,580	18,810
2000'-0"				

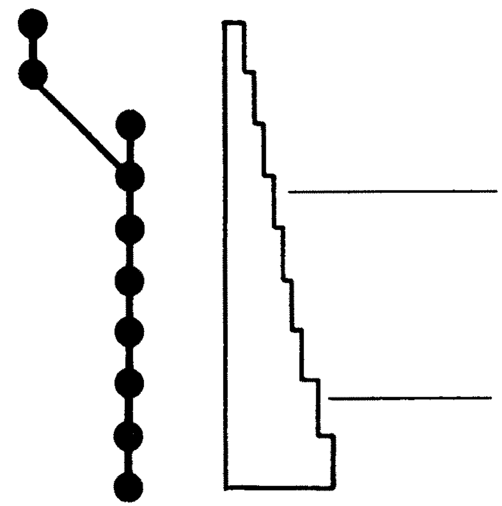


TABLE 3.7(B)-7H RESPONSE SHEAR FORCES (KIPS) AUXILIARY/CONTROL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"				
	250	270	180	270
2090'-0"				
	950	1050	690	1050
2087'-2"				
	1390	1520	980	1520
2073'-6"				
	6070	6660	4260	6660
2065'-0"				
	7890	8660	5490	8660
2047'-6"				
	11,870	12,990	8090	12,990
2032'-0"				
	13,640	14,750	9550	14,750
2026'-0"				
	17,030	17,850	12,430	17,850
2016'-0"				
	19,560	20,100	14,820	20,100
2000'-0"				

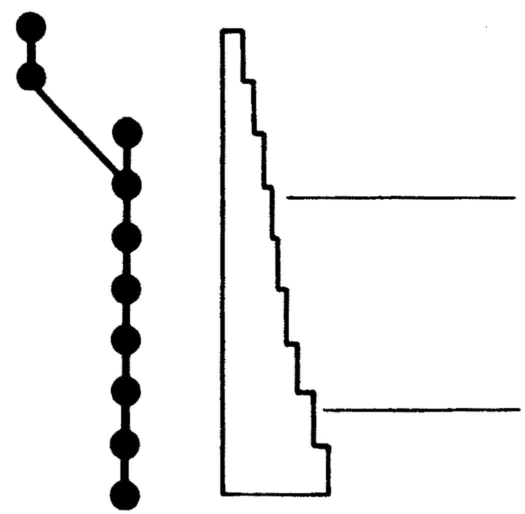


TABLE 3.7(B)-7I RESPONSE AXIAL FORCES (KIPS) AUXILIARY/CONTROL BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"				
	130	130	120	130
2090'-0"				
	510	520	500	520
2087'-2"				
	790	800	770	800
2073'-6"				
	3630	3650	3560	3650
2065'-0"				
	4860	4860	4770	4860
2047'-6"				
	8100	7970	7950	8100
2032'-0"				
	9700	9500	9530	9700
2026'-0"				
	12,770	12,510	12,550	12,770
2016'-0"				
	15,330	15,020	15,070	15,330
2000'-0"				

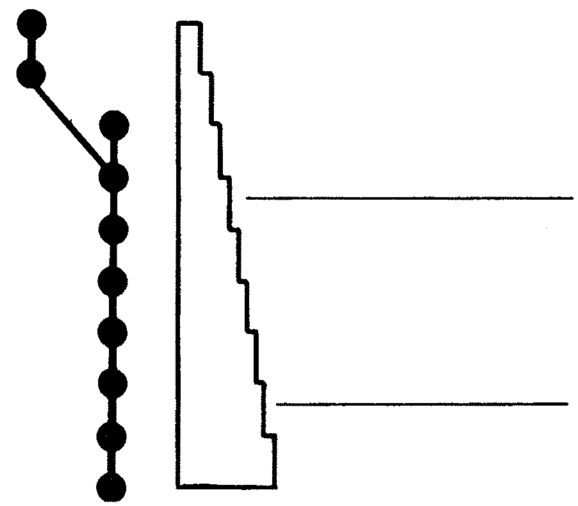


TABLE 3.7(B)-7J RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) AUXILIARY/CONTROL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0/0.0029	0/0.0031	0/0.0020	0/0.0031
2090'-0"	0.0033/0.0177	0.0034/0.0186	0.0022/0.0123	0.0034/0.0186
2087'-2"	0/0.0041	0/0.0036	0/0.0030	0/0.0041
2073'-6"	0.0216/0.0795	0.0218/0.0707	0.0142/0.0525	0.0218/0.0795
2065'-0"	0.1277/0.1585	0.1225/0.1455	0.0820/0.1028	0.1277/0.1585
2047'-6"	0.2879/0.3598	0.2847/0.3370	0.1844/0.2280	0.2879/0.3598
2032'-0"	0.5332/0.5652	0.5240/0.5463	0.3420/0.3544	0.5332/0.5652
2026'-0"	0.6416/0.7013	0.6287/0.6694	0.4120/0.4347	0.6416/0.7013
2016'-0"	0.8571/0.9032	0.8360/0.8633	0.5523/0.5682	0.8571/0.9032
2000'-0"	1.1914	1.1644	0.7824	1.1914

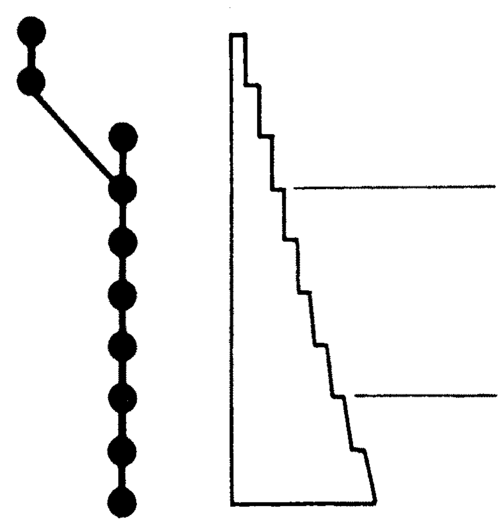


TABLE 3.7(B)-7K RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) AUXILIARY/CONTROL BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.0002/0.0034	0.0002/0.0036	0.0002/0.0024	0.0002/0.0036
2090'-0"	0.0041/0.0198	0.0042/0.0214	0.0029/0.0142	0.0042/0.0214
2087'-2"	0/0.0026	0/0.0019	0/0.0016	0/0.0026
2073'-6"	0.0216/0.0808	0.0227/0.0721	0.0144/0.0506	0.0227/0.0808
2065'-0"	0.1325/0.1682	0.1288/0.1541	0.0867/0.1061	0.1325/0.1682
2047'-6"	0.3062/0.3740	0.3057/0.3524	0.2023/0.2380	0.3062/0.3740
2032'-0"	0.5577/0.5919	0.5537/0.5765	0.3622/0.3795	0.5577/0.5919
2026'-0"	0.6727/0.7169	0.6649/0.6938	0.4332/0.4551	0.6727/0.7169
2016'-0"	0.8805/0.9187	0.8722/0.8963	0.5607/0.5788	0.8805/0.9187
2000'-0"	1.2141	1.2179	0.7656	1.2179

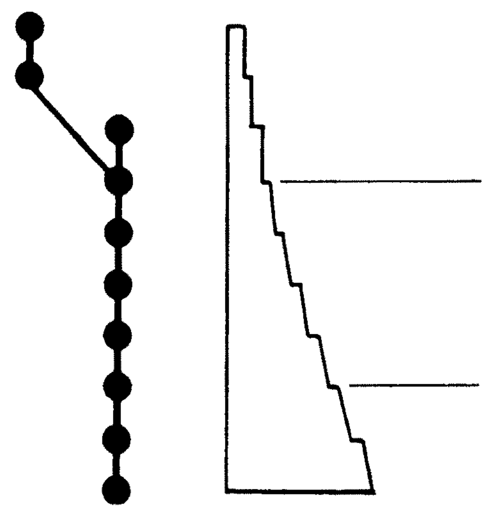


TABLE 3.7(B)-7L RESPONSE DISPLACEMENTS (INCHES) AUXILIARY/CONTROL BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.255	0.061	0.078	0.255
2090'-0"	0.250	0.057	0.075	0.250
2087'-2"	0.249	0.055	0.075	0.249
2073'-6"	0.245	0.051	0.072	0.245
2065'-0"	0.241	0.047	0.070	0.241
2047'-6"	0.231	0.037	0.063	0.231
2032'-0"	0.223	0.028	0.057	0.223
2026'-0"	0.219	0.024	0.054	0.219
2016'-0"	0.213	0.018	0.050	0.213
2000'-0"	0.203	0.007	0.043	0.203

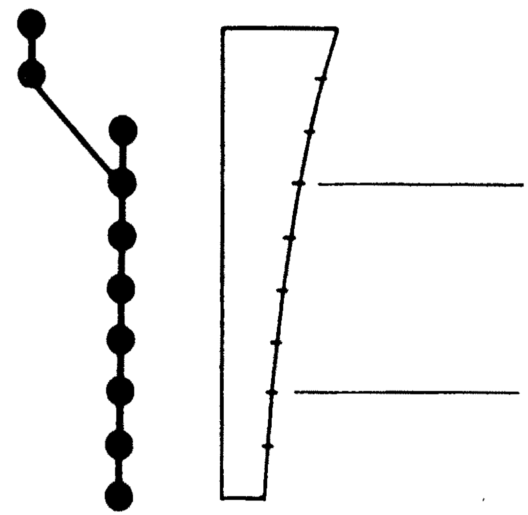


TABLE 3.7(B)-7M RESPONSE DISPLACEMENTS (INCHES) AUXILIARY/CONTROL BUILDING SSE
EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.254	0.080	0.080	0.254
2090'-0"	0.249	0.076	0.077	0.249
2087'-2"	0.246	0.073	0.075	0.246
2073'-6"	0.239	0.066	0.071	0.239
2065'-0"	0.234	0.061	0.068	0.234
2047'-6"	0.221	0.048	0.059	0.221
2032'-0"	0.210	0.037	0.052	0.210
2026'-0"	0.205	0.032	0.049	0.205
2016'-0"	0.197	0.025	0.044	0.197
2000'-0"	0.185	0.012	0.036	0.185

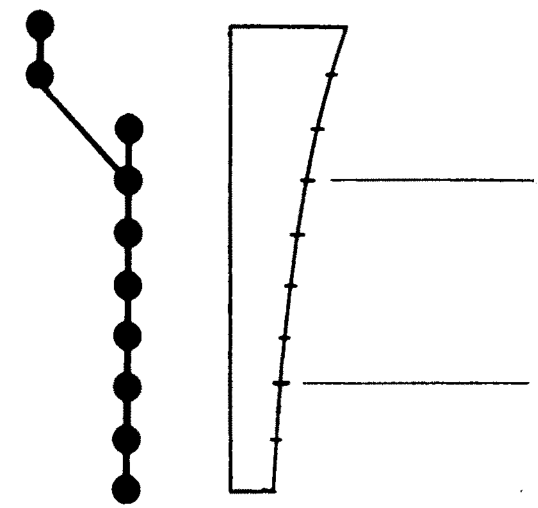


TABLE 3.7(B)-7N RESPONSE DISPLACEMENTS (INCHES) AUXILIARY/CONTROL BUILDING SSE
VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.018	0.007	0.012	0.018
2090'-0"	0.018	0.007	0.012	0.018
2087'-2"	0.018	0.007	0.012	0.018
2073'-2"	0.017	0.006	0.012	0.017
2065'-0"	0.017	0.006	0.012	0.017
2047'-6"	0.016	0.005	0.011	0.016
2032'-0"	0.015	0.004	0.010	0.015
2026'-0"	0.015	0.003	0.009	0.015
2016'-0"	0.014	0.003	0.008	0.014
2000'-0"	0.013	0.001	0.007	0.013

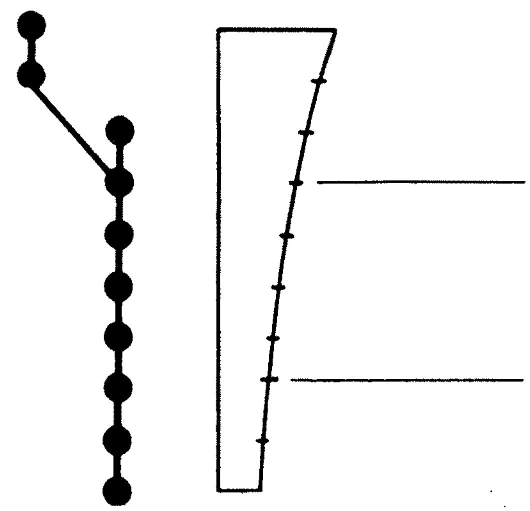


TABLE 3.7(B)-7O RESPONSE ACCELERATIONS (G'S) AUXILIARY/CONTROL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.265	0.284	0.170	0.284
2090'-0"	0.249	0.264	0.157	0.264
2087'-2"	0.243	0.257	0.156	0.257
2073'-6"	0.228	0.240	0.150	0.240
2065'-0"	0.215	0.224	0.145	0.224
2047'-6"	0.181	0.185	0.131	0.185
2032'-0"	0.153	0.163	0.126	0.163
2026'-0"	0.141	0.158	0.125	0.158
2016'-0"	0.125	0.152	0.122	0.152
2000'-0"	0.106	0.140	0.118	0.140

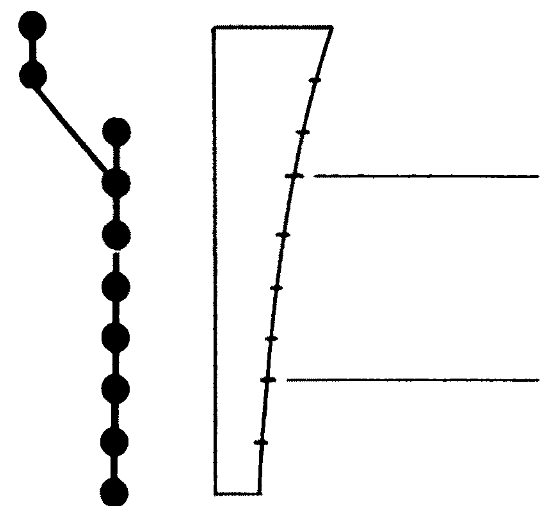


TABLE 3.7(B)-7P RESPONSE ACCELERATIONS (G'S) AUXILIARY/CONTROL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.270	0.326	0.207	0.326
2090'-0"	0.258	0.309	0.195	0.309
2087'-2"	0.250	0.292	0.186	0.292
2073'-6"	0.230	0.264	0.170	0.264
2065'-0"	0.215	0.244	0.157	0.244
2047'-6"	0.188	0.199	0.137	0.199
2032'-0"	0.173	0.166	0.132	0.173
2026'-0"	0.166	0.159	0.130	0.166
2016'-0"	0.153	0.152	0.127	0.153
2000'-0"	0.130	0.141	0.123	0.141

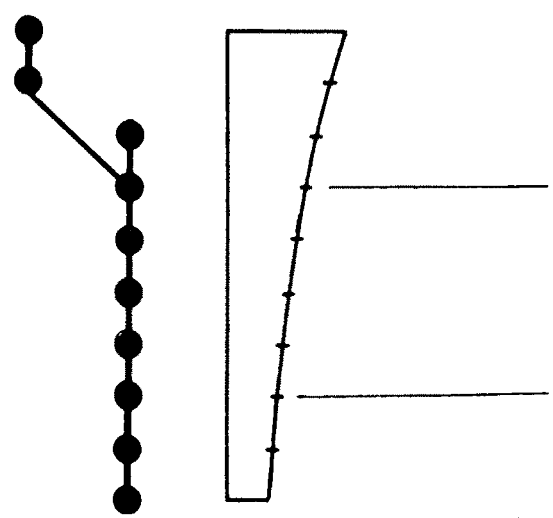


TABLE 3.7(B)-7Q RESPONSE ACCELERATIONS (G'S) AUXILIARY/CONTROL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.134	0.159	0.133	0.159
2090'-0"	0.134	0.158	0.132	0.158
2087'-2"	0.134	0.157	0.132	0.157
2073'-6"	0.134	0.154	0.132	0.154
2065'-0"	0.133	0.151	0.132	0.151
2047'-6"	0.132	0.142	0.131	0.142
2032'-0"	0.130	0.134	0.130	0.134
2026'-0"	0.130	0.131	0.129	0.131
2016'-0"	0.129	0.130	0.128	0.130
2000'-0"	0.127	0.127	0.127	0.127

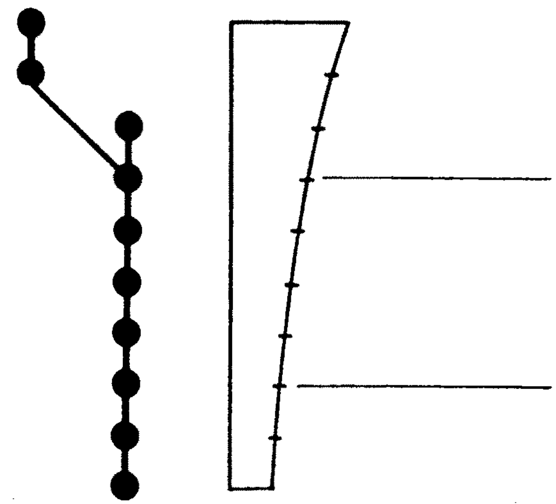


TABLE 3.7(B)-7R RESPONSE INERTIA FORCES (KIPS) AUXILIARY/CONTROL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	130	140	80	140
2090'-0"	370	390	230	390
2087'-2"	740	780	480	780
2073'-6"	2080	2190	1380	2190
2065'-0"	1030	1080	690	1080
2047'-6"	2340	2390	1700	2390
2032'-0"	980	980	770	980
2026'-0"	1770	1740	1410	1770
2016'-0"	1300	1270	1110	1300
2000'-0"	—	—	—	

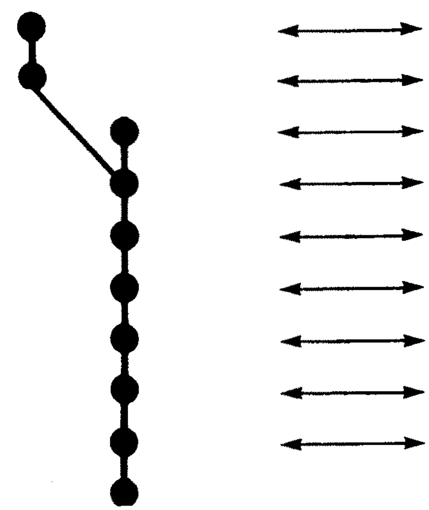


TABLE 3.7(B)-7S RESPONSE INERTIA FORCES (KIPS) AUXILIARY/CONTROL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	130	160	100	160
2090'-0"	390	460	290	460
2087'-2"	770	890	570	890
2073'-6"	2110	2400	1530	2400
2065'-0"	1040	1180	760	1180
2047'-6"	2320	2530	1620	2530
2032'-0"	970	1020	660	1020
2026'-0"	1900	1800	1380	1900
2016'-0"	1610	1310	1310	1610
2000'-0"	—	—	—	

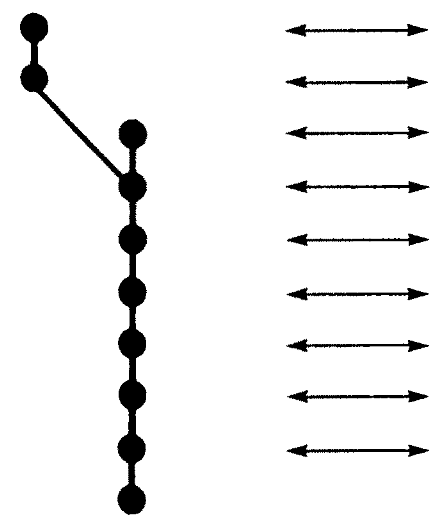


TABLE 3.7(B)-7T RESPONSE INERTIA FORCES (KIPS) AUXILIARY/CONTROL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	70	80	60	80
2090'-0"	190	220	200	220
2087'-2"	410	470	400	470
2073'-6"	1210	1360	1200	1360
2065'-0"	640	700	620	700
2047'-6"	1680	1780	1670	1780
2032'-0"	840	830	820	840
2026'-0"	1590	1590	1580	1590
2016'-0"	1330	1270	1320	1330
2000'-0"	—	—	—	

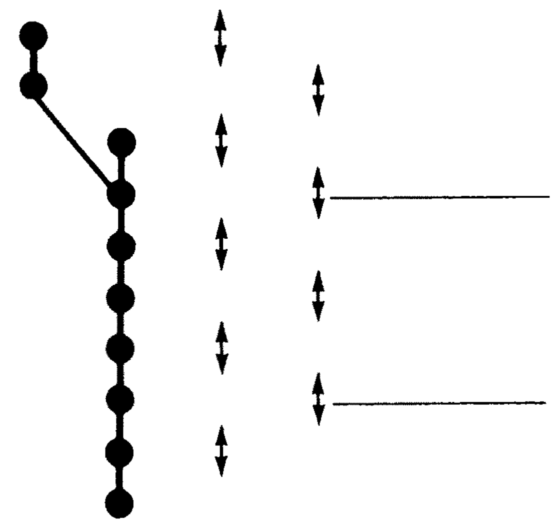


TABLE 3.7(B)-7U RESPONSE SHEAR FORCES (KIPS) AUXILIARY/CONTROL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

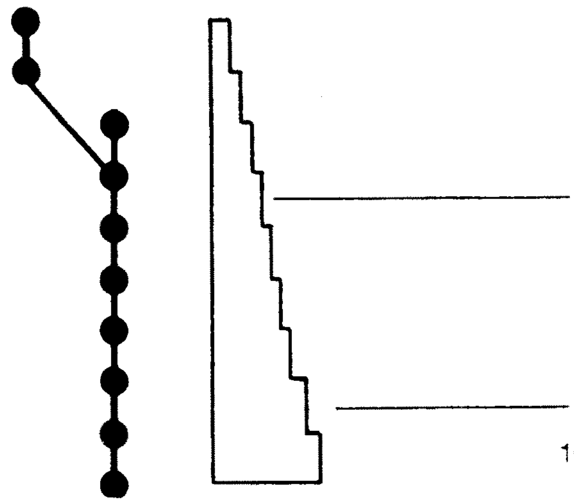
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2102'-6"					140
	130	140	80		
2090'-0"					530
	500	530	310		
2087'-2"					780
	740	780	480		
2073'-6"					3500
	3320	3500	2170		
2065'-0"					4580
	4350	4580	2860		
2047'-6"					6970
	6690	6970	4560		
2032'-0"					7950
	7670	7950	5330		
2026'-0"					9690
	9440	9690	6740		
2016'-0"					10,960
	10,740	10,960	7850		
2000'-0"					

TABLE 3.7(B)-7V RESPONSE SHEAR FORCES (KIPS) AUXILIARY/CONTROL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

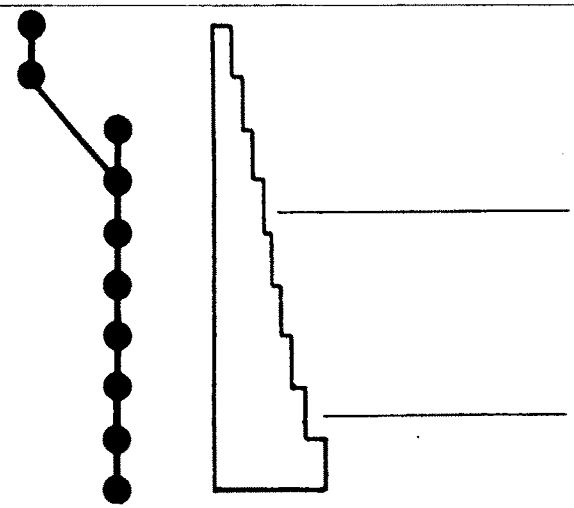
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"				
	130	160	100	160
2090'-0"				
	520	620	390	620
2087'-2"				
	770	890	570	890
2073'-6"				
	3400	3910	2490	3910
2065'-0"				
	4440	5090	3250	5090
2047'-6"				
	6760	7620	4870	7620
2032'-0"				
	7730	8640	5530	8640
2026'-0"				
	9630	10,440	6910	10,440
2016'-0"				
	11,240	11,750	8220	11,750
2000'-0"				

TABLE 3.7(B)-7W RESPONSE AXIAL FORCES (KIPS) AUXILIARY/CONTROL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"				
	70	80	60	80
2090'-0"				
	260	300	260	300
2087'-2"				
	410	470	400	470
2073'-6"				
	1880	2130	1860	2130
2065'-0"				
	2520	2830	2480	2830
2047'-6"				
	4200	4610	4150	4610
2032'-0"				
	5040	5440	4970	5440
2026'-0"				
	6630	7030	6550	7030
2016'-0"				
	7960	8300	7870	8300
2000'-0"				

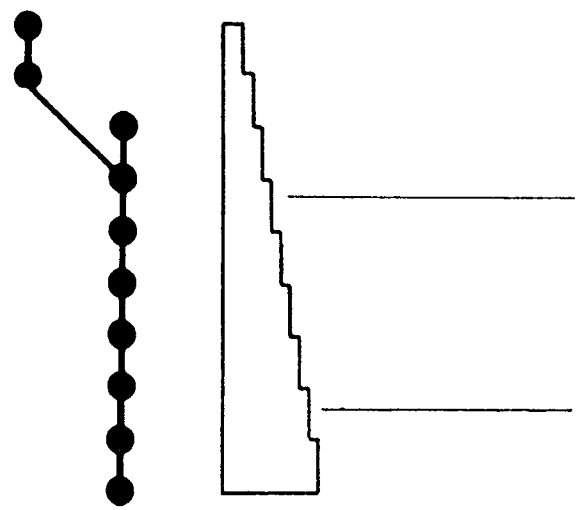


TABLE 3.7(B)-7X RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) AUXILIARY/CONTROL BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0/0.0016	0/0.0018	0/0.0011	0/0.0018
2090'-0"	0.0018/0.0100	0.0019/0.0107	0.0012/0.0062	0.0019/0.0107
2087'-2"	0/0.0018	0/0.0022	0/0.0020	0/0.0022
2073'-6"	0.0119/0.0393	0.0125/0.0402	0.0078/0.0333	0.0125/0.0402
2065'-0"	0.0675/0.0807	0.0700/0.0829	0.0489/0.0634	0.0700/0.0829
2047'-6"	0.1568/0.1879	0.1630/0.1924	0.1043/0.1369	0.1630/0.1924
2032'-0"	0.2916/0.3055	0.3006/0.3130	0.1865/0.2003	0.3006/0.3130
2026'-0"	0.3514/0.3772	0.3608/0.3834	0.2209/0.2458	0.3608/0.3834
2016'-0"	0.4717/0.4898	0.4804/0.4958	0.2837/0.3003	0.4804/0.4958
2000'-0"	0.6617	0.6711	0.4020	0.6711

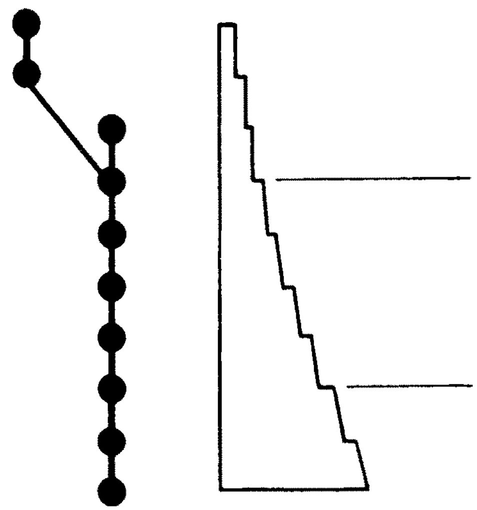


TABLE 3.7(B)-7Y RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) AUXILIARY/CONTROL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.0001/0.0018	0.0001/0.0021	0.0001/0.0014	0.0001/0.0021
2090'-0"	0.0021/0.0106	0.0025/0.0127	0.0016/0.0080	0.0025/0.0127
2087'-2"	0/0.0011	0/0.0011	0/0.0001	0/0.0011
2073'-6"	0.0115/0.0365	0.0133/0.0424	0.0085/0.0283	0.0133/0.0424
2065'-0"	0.0654/0.0785	0.0757/0.0905	0.0486/0.0588	0.0757/0.0905
2047'-6"	0.1561/0.1808	0.1795/0.2067	0.1154/0.1338	0.1795/0.2067
2032'-0"	0.2856/0.2979	0.3248/0.3378	0.2094/0.2184	0.3248/0.3378
2026'-0"	0.3444/0.3602	0.3898/0.4066	0.2518/0.2633	0.3898/0.4066
2016'-0"	0.4548/0.4684	0.5110/0.5248	0.3301/0.3396	0.5110/0.5248
2000'-0"	0.6395	0.7128	0.4593	0.7128

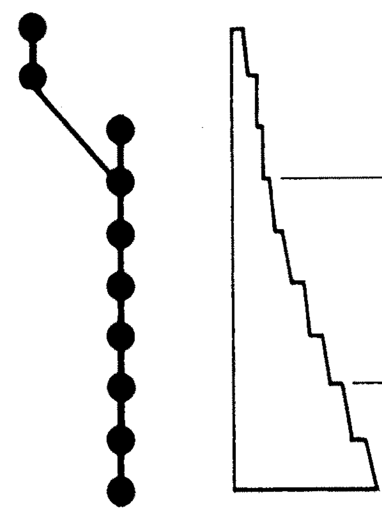


TABLE 3.7(B)-7Z RESPONSE DISPLACEMENTS (INCHES) AUXILIARY/CONTROL BUILDING OBE
NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-19

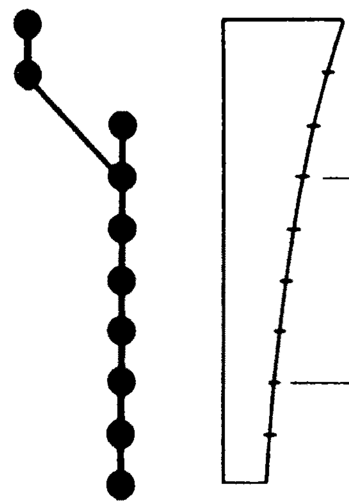
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2102'-6"	0.108	0.035	0.036		0.108
2090'-0"	0.106	0.033	0.034		0.106
2087'-2"	0.105	0.032	0.034		0.105
2073'-6"	0.103	0.030	0.032		0.102
2065'-0"	0.100	0.027	0.031		0.100
2047'-6"	0.095	0.022	0.028		0.095
2032'-0"	0.090	0.016	0.024		0.090
2026'-0"	0.088	0.014	0.023		0.088
2016'-0"	0.084	0.011	0.021		0.084
2000'-0"	0.079	0.005	0.016		0.079

TABLE 3.7(B)-7AA RESPONSE DISPLACEMENTS (INCHES) AUXILIARY/CONTROL BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-19

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.106	0.048	0.039	0.106
2090'-0"	0.103	0.046	0.038	0.103
2087'-2"	0.102	0.044	0.037	0.102
2073'-6"	0.098	0.040	0.034	0.098
2065'-0"	0.095	0.037	0.032	0.095
2047'-6"	0.089	0.029	0.027	0.089
2032'-0"	0.083	0.023	0.022	0.083
2026'-0"	0.080	0.020	0.020	0.080
2016'-0"	0.076	0.015	0.017	0.076
2000'-0"	0.070	0.008	0.013	0.070

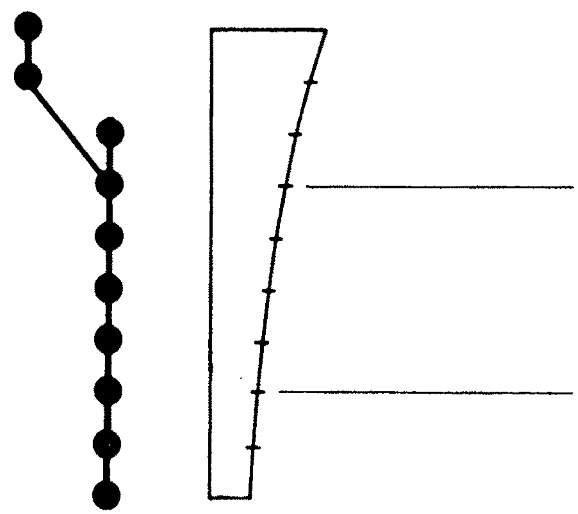


TABLE 3.7(B)-7AB RESPONSE DISPLACEMENTS (INCHES) AUXILIARY/CONTROL BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-19

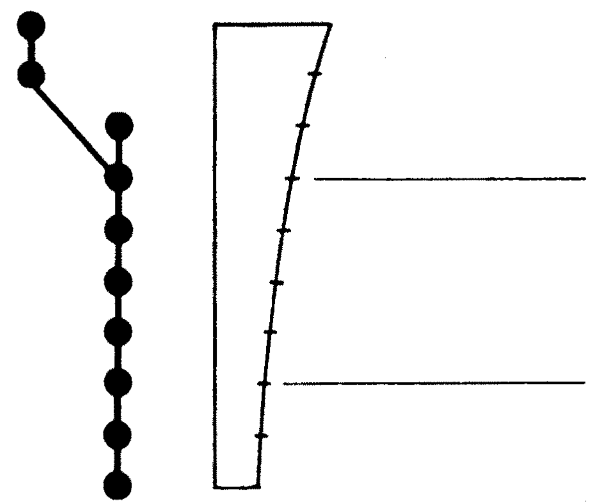
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2102'-6"	0.007	0.004	0.006	 0.007
2090'-0"	0.007	0.004	0.006	0.007
2087'-2"	0.007	0.004	0.005	0.007
2073'-6"	0.007	0.004	0.005	0.007
2065'-0"	0.007	0.004	0.005	0.007
2047'-6"	0.006	0.003	0.005	0.006
2032'-0"	0.006	0.003	0.004	0.006
2026'-0"	0.005	0.002	0.004	0.005
2016'-0"	0.005	0.002	0.003	0.005
2000'-0"	0.004	0.001	0.003	0.004

TABLE 3.7(B)-8A RESPONSE ACCELERATIONS (G'S) DIESEL GENERATOR BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

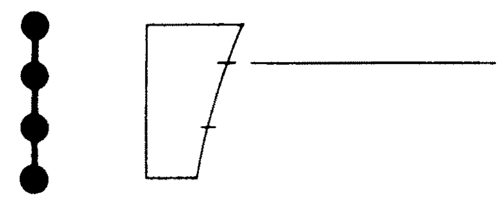
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.581	0.470	0.375	 0.581
2047'-2"	0.477	0.399	0.325	0.477
2027'-6"	0.374	0.317	0.290	0.374
2000'-0"	0.256	0.246	0.255	0.256

TABLE 3.7(B)-8B RESPONSE ACCELERATIONS (G'S) DIESEL GENERATOR BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.585	0.524	0.516	0.585
2047'-2"	0.468	0.457	0.423	0.468
2027'-6"	0.318	0.317	0.287	0.318
2000'-0"	0.184	0.258	0.243	0.258

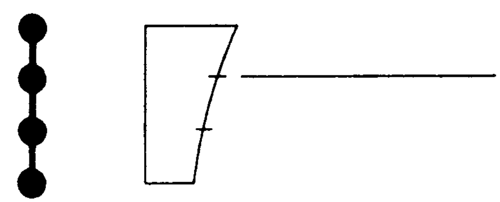


TABLE 3.7(B)-8C RESPONSE ACCELERATIONS (G'S) DIESEL GENERATOR BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

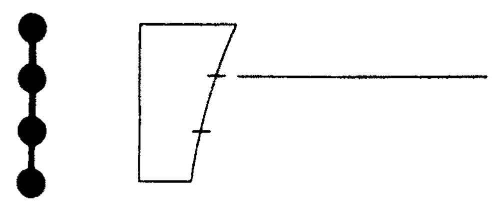
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.263	0.251	0.266	 0.266
2047'-2"	0.263	0.250	0.266	0.266
2027'-6"	0.261	0.247	0.264	0.264
2000'-0"	0.258	0.244	0.260	0.260

TABLE 3.7(B)-8D RESPONSE INERTIA FORCES (KIPS) DIESEL GENERATOR BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	670	550	440	670
2047'-2"	1450	1230	1010	1450
2027'-6"	670	580	490	670
2000'-0"	—	—	—	

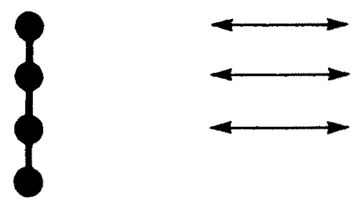


TABLE 3.7(B)-8E RESPONSE INERTIA FORCES (KIPS) DIESEL GENERATOR BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	710	620	600	710
2047'-2"	1470	1400	1300	1470
2027'-6"	580	580	510	580
2000'-0"	—	—	—	

TABLE 3.7(B)-8F RESPONSE INERTIA FORCES (KIPS) DIESEL GENERATOR BUILDING SSE
VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	300	290	310	310
2047'-2"	800	750	790	800
2027'-6"	460	440	460	460
2000'-0"	—	—	—	

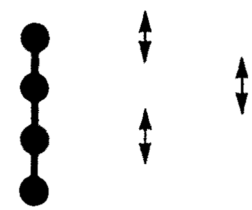


TABLE 3.7(B)-8G RESPONSE SHEAR FORCES (KIPS) DIESEL GENERATOR BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

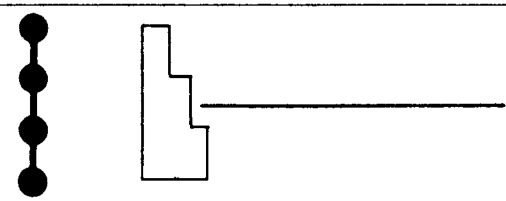
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"				
	670	550	440	670
2047'-2"				
	2120	1780	1450	2120
2027'-6"				
	2790	2360	1940	2790
2000'-0"				

TABLE 3.7(B)-8H RESPONSE SHEAR FORCES (KIPS) DIESEL GENERATOR BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

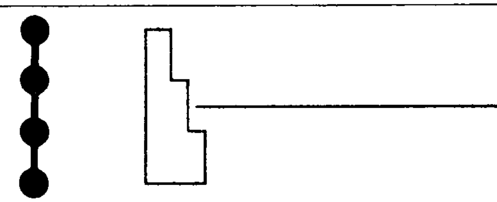
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"				
	710	620	600	710
2047'-2"				
	2180	2020	1900	2180
2027'-6"				
	2760	2600	2410	2760
2000'-0"				

TABLE 3.7(B)-8I RESPONSE AXIAL FORCES (KIPS) DIESEL GENERATOR BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

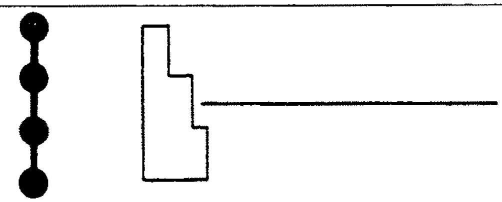
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"				
	300	290	310	310
2047'-2"				
	1100	1040	1100	1100
2027'-6"				
	1560	1480	1560	1560
2000'-0"				

TABLE 3.7(B)-8J RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) DIESEL GENERATOR BUILDING SSE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

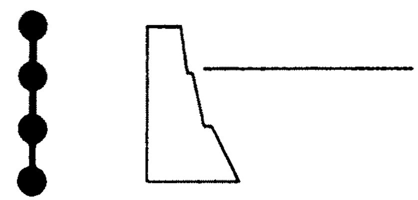
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0/0.0007	0/0.0004	0/0.0004	 0/0.0007
2047'-2"	0.0134/0.0212	0.0108/0.0145	0.0087/0.0120	0.0134/0.0212
2027'-6"	0.0629/0.0698	0.0496/0.0525	0.0405/0.0434	0.0629/0.0698
2000'-0"	0.1460	0.1175	0.0967	0.1460

TABLE 3.7(B)-8K RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) DIESEL GENERATOR BUILDING SSE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0/0.0032	0/0.0015	0/0.0023	0/0.0032
2047'-2"	0.0163/0.0235	0.0131/0.0166	0.0138/0.0195	0.0163/0.0235
2027'-6"	0.0664/0.0706	0.0564/0.0582	0.0570/0.0601	0.0664/0.0706
2000'-0"	0.1464	0.1296	0.1264	0.1464

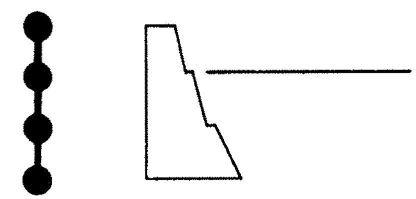


TABLE 3.7(B)-8L RESPONSE DISPLACEMENTS (INCHES) DIESEL GENERATOR BUILDING SSE
NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

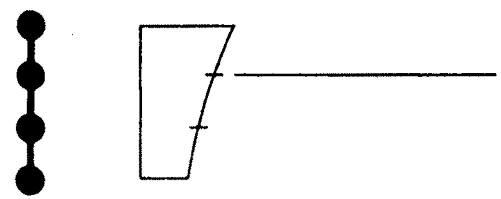
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2066'-0"	0.278	0.043	0.048		0.278
2047'-2"	0.248	0.034	0.037		0.248
2027'-6"	0.215	0.022	0.024		0.215
2000'-0"	0.168	0.007	0.005		0.168

TABLE 3.7(B)-8M RESPONSE DISPLACEMENTS (INCHES) DIESEL GENERATOR BUILDING SSE
EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

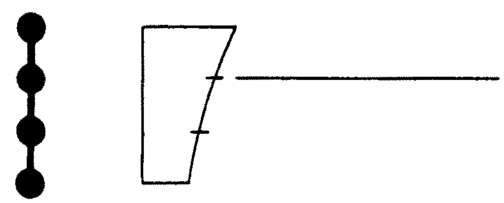
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.347	0.090	0.124	 0.347
2047'-2"	0.304	0.077	0.102	0.304
2027'-6"	0.245	0.049	0.067	0.245
2000'-0"	0.160	0.009	0.015	0.160

TABLE 3.7(B)-8N RESPONSE DISPLACEMENTS (INCHES) DIESEL GENERATOR BUILDING SSE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.010	0.004	0.007	0.010
2047'-2"	0.010	0.004	0.007	0.010
2027'-6"	0.008	0.003	0.006	0.008
2000'-0"	0.006	0.001	0.004	0.006

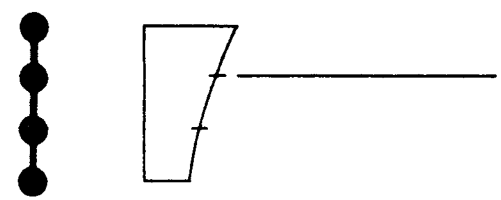


TABLE 3.7(B)-8O RESPONSE ACCELERATIONS (G'S) DIESEL GENERATOR BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.261	0.259	0.205	0.261
2047'-2"	0.221	0.218	0.177	0.221
2027'-6"	0.180	0.168	0.149	0.180
2000'-0"	0.130	0.129	0.134	0.134

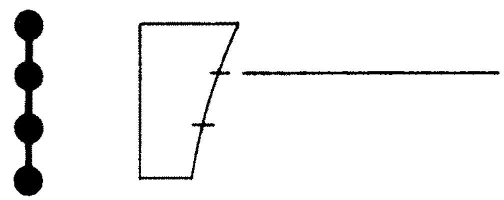


TABLE 3.7(B)-8P RESPONSE ACCELERATIONS (G'S) DIESEL GENERATOR BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

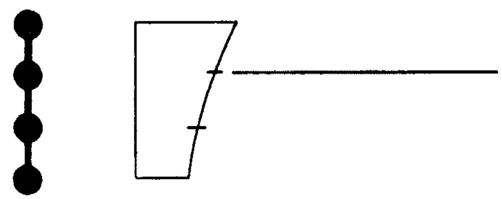
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.291	0.320	0.291	 0.320
2047'-2"	0.238	0.282	0.241	0.282
2027'-6"	0.158	0.197	0.163	0.197
2000'-0"	0.102	0.136	0.130	0.136

TABLE 3.7(B)-8Q RESPONSE ACCELERATIONS (G'S) DIESEL GENERATOR BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

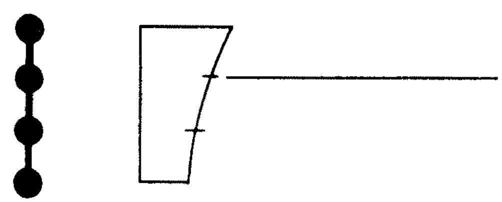
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2066'-0"	0.142	0.131	0.140		0.142
2047'-2"	0.142	0.130	0.139		0.142
2027'-6"	0.140	0.129	0.136		0.140
2000'-0"	0.137	0.127	0.131		0.137

TABLE 3.7(B)-8R RESPONSE INERTIA FORCES (KIPS) DIESEL GENERATOR BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

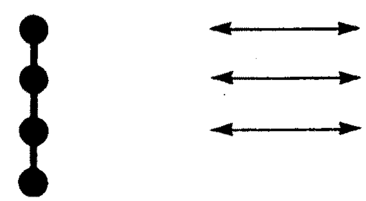
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2066'-0''	300	310	240		310
2047'-2''	670	670	550		670
2027'-6''	320	310	310		320
2000'-0''	—	—	—		

TABLE 3.7(B)-8S RESPONSE INERTIA FORCES (KIPS) DIESEL GENERATOR BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

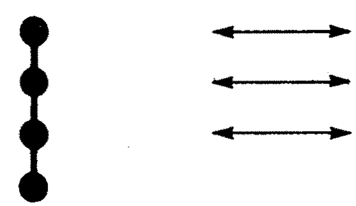
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	350	380	350	 380
2047'-2"	730	860	750	860
2027'-6"	280	360	290	360
2000'-0"	—	—	—	

TABLE 3.7(B)-8T RESPONSE INERTIA FORCES (KIPS) DIESEL GENERATOR BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

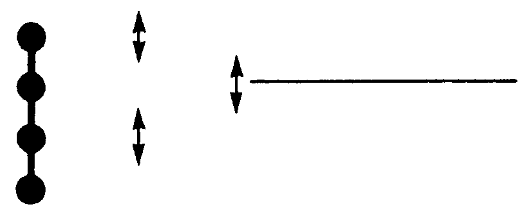
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0''	170	150	160	 170
2047'-2''	430	400	420	430
2027'-6''	250	230	230	250
2000'-0''	—	—	—	

TABLE 3.7(B)-8U RESPONSE SHEAR FORCES (KIPS) DIESEL GENERATOR BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

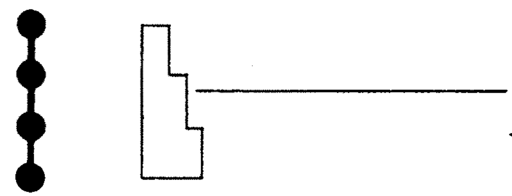
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"				 310
	300	310	240	
2047'-2"				980
	970	980	790	
2027'-6"				1290
	1290	1290	1100	
2000'-0"				

TABLE 3.7(B)-8V RESPONSE SHEAR FORCES (KIPS) DIESEL GENERATOR BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

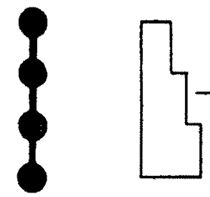
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"				
	350	380	350	380
2047'-2"				
	1080	1240	1100	1240
2027'-6"				
	1360	1600	1390	1600
2000'-0"				

TABLE 3.7(B)-8W RESPONSE AXIAL FORCES (KIPS) DIESEL GENERATOR BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

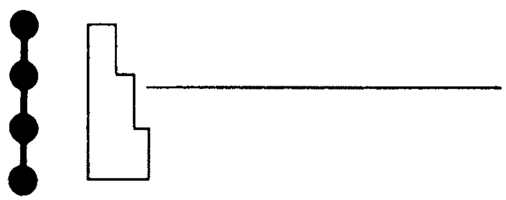
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"				
	170	150	160	170
2047'-2"				
	600	550	580	600
2027'-6"				
	850	780	810	850
2000'-0"				

TABLE 3.7(B)-8X RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) DIESEL GENERATOR BUILDING OBE NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0/0.0003	0/0.0002	0/0.0002	0/0.0003
2047'-2"	0.0059/0.0087	0.0060/0.0082	0.0048/0.0066	0.0060/0.0087
2027'-6"	0.0278/0.0302	0.0275/0.0293	0.0221/0.0237	0.0278/0.0302
2000'-0"	0.0655	0.0647	0.0525	0.0655

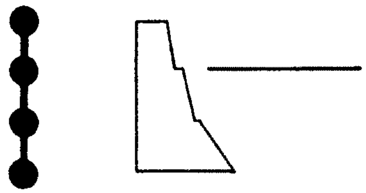


TABLE 3.7(B)-8Y RESPONSE BENDING MOMENTS (MILLIONS OF KIP-FEET) DIESEL GENERATOR BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

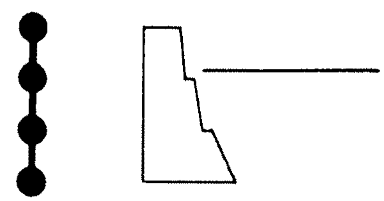
MASS POINT EL.	SITE			ENVELOPE	
	CALLAWAY	STERLING	WOLF CREEK		
2066'-0''	0/0.0013	0/0.0008	0/0.0012		0/0.0013
2047'-2''	0.0078/0.0111	0.0079/0.0099	0.0078/0.0107		0.0079/0.0111
2027'-6''	0.0323/0.0341	0.0343/0.0353	0.0323/0.0338		0.0343/0.0353
2000'-0''	0.0716	0.0793	0.0720		0.0793

TABLE 3.7(B)-8Z RESPONSE DISPLACEMENTS (INCHES) DIESEL GENERATOR BUILDING OBE
NORTH-SOUTH DIRECTION

REF. FIGURE 3.7(B)-20

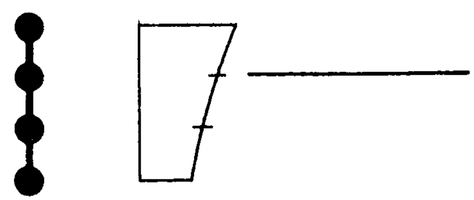
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.101	0.022	0.025	 0.101
2047'-2"	0.090	0.017	0.019	0.090
2027'-6"	0.078	0.011	0.013	0.078
2000'-0"	0.061	0.003	0.003	0.061

TABLE 3.7(B)-8AA RESPONSE DISPLACEMENTS (INCHES) DIESEL GENERATOR BUILDING OBE EAST-WEST DIRECTION

REF. FIGURE 3.7(B)-20

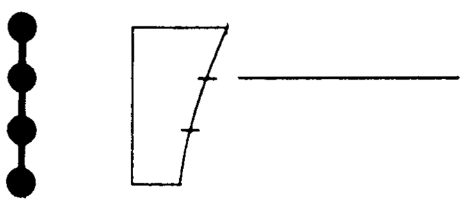
MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.132	0.054	0.064	 0.132
2047'-2"	0.115	0.046	0.053	0.115
2027'-6"	0.090	0.029	0.034	0.090
2000'-0"	0.055	0.005	0.007	0.055

TABLE 3.7(B)-8AB RESPONSE DISPLACEMENTS (INCHES) DIESEL GENERATOR BUILDING OBE VERTICAL DIRECTION

REF. FIGURE 3.7(B)-20

MASS POINT EL.	SITE			ENVELOPE
	CALLAWAY	STERLING	WOLF CREEK	
2066'-0"	0.005	0.002	0.004	0.005
2047'-2"	0.005	0.002	0.003	0.005
2027'-6"	0.004	0.002	0.003	0.004
2000'-0"	0.003	0.001	0.002	0.003

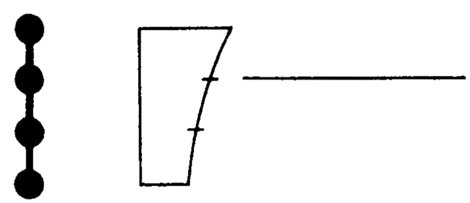


TABLE 3.7B-9 DESIGN COMPARISON WITH R.G. 1.12, REVISION 1, DATED APRIL 1974, TITLED INSTRUMENTATION FOR EARTHQUAKES

Regulatory Guide 1.12 Position	Union Electric	Instr. Tag No.
C. REGULATORY POSITION		
Earthquake instrumentation specified in ANSI N18.5, "Earthquake Instrumentation Criteria for Nuclear Power Plants," is acceptable to the regulatory staff for satisfying the seismic instrumentation requirements indicated in Paragraph VI (a)(3) of Appendix A to 10 CFR Part 100 for assuring the safety of nuclear power plants, subject to the following:	Complies as described in Section 3.7(B).4.1.	
1. The instrumentation called for in Section 4.1 of the Standard should be applied to nuclear power plants with a safe shutdown earthquake maximum foundation acceleration of less than 0.3g, as supplemented by the following:	1. Refer to ANSI 18.5-1974 Section 4.1 below.	
a. Instead of the locations specified in Section 4.1.2 of the Standard, one triaxial peak accelerograph should be provided at one location of each of the following:		
(1) A selected location on the reactor equipment	(1) Complies.	0-SG-AR-8
(2) A selected location on the reactor piping.	(2) Complies.	0-SG-AR-7
(3) The most pertinent location on one of the following outside of the containment structure:	(3) Complies. (b)	0-SG-AR-6
(a) SeismicCategory I equipment		
(b) Seismic Category I piping		

TABLE 3.7B-9 (Sheet 2)

Regulatory Guide 1.12 Position	Union Electric	Instr. Tag No.
b. One triaxial response-spectrum recorder cable of measuring both horizontal motions and the vertical motion and capable of providing signals for immediate control room indication should be provided at the containment foundation.	Complies.	0-SG-AE-1 0-SG-ARS-1
c. One triaxial response-spectrum recorder capable of measuring both horizontal motions and the vertical motion should be provided at one location of each of the following:		
(1) A selected location on the reactor equipment or piping supports.	(1) Complies.	0-SG-AE-3
(2) The most pertinent location on one of the following outside of the containment structure:	(2)(a) Complies.	0-SG-AE-5
(a) A Seismic Category I equipment support or appropriate floor location.		
(b) A Seismic Category I piping support or appropriate floor location.		
(3) At the foundation of an independent Seismic Category I structure where the response is different from that of the reactor containment structure.	(3) Complies.	0-SG-AE-4
2. Section 4.2 of the Standard should not be used.	Complies.	
3. The instrumentation specified in Section 4.3 of the Standard should be applied to nuclear power plants with a safe shutdown earthquake maximum foundation acceleration of 0.3g or greater as supplemented by Regulatory Positions 1 and 2 above, and the following:	N/A	

TABLE 3.7B-9 (Sheet 3)

Regulatory Guide 1.12 Position	Union Electric	Instr. Tag No.
a. Instead of the locations specified in Section 4.3.2 of the Standard, one triaxial time-history accelerograph should be provided at the most pertinent location on one of the independent Seismic Category I structures where the response is different from that of the reactor containment structure.	N/A	
b. Instead of the locations specified in Section 4.3.3 of the Standard, one triaxial peak accelerograph should be provided at the most pertinent location on Seismic Category I equipment or piping in an independent Seismic Category I structure where the response is different from that of the reactor containment structure.	N/A	
c. In addition to the locations specified in Regulatory Positions 1.b., 1.c.(1), and 1.c.(3) above, one triaxial response-spectrum recorder should be provided at one location on both items specified in Regulatory Positions 1.c(2)(a) and 1.c.(2)(b) above.	N/A	
d. Instead of the locations specified in Section 4.3.4 of the Standard, one triaxial seismic switch should be provided at a selected location on reactor equipment supports or piping supports.	N/A	
4. The response-spectrum recorders should have the following specifications:		
a. Dynamic Range--50:1 zero to peak (such as 0.02g to 1.0g).	Complies.	
b. Frequency Range--minimum coverage from 1 Hz to 30.0 Hz.	Complies.	

TABLE 3.7B-9 (Sheet 4)

Regulatory Guide 1.12 Position		Union Electric	Instr. Tag No.
c. Damping--not less than nominal 2 percent nor more than nominal 5 percent of critical damping, controlled to ±0.15 of nominal. The actual amount of damping is to be consistent with the OBE-based design damping for the supported structure or equipment.		Complies.	
5. Instead of the dynamic range specified in Section 5.3.5 of the Standard, a range of 100:1 should be used.		Complies.	
ANSI 18.5-1974, Section 4.1			
4.1 Safe Shutdown Earthquake Maximum Ground Acceleration of Less Than 0.2g			
4.1.1 One triaxial time-history accelerograph shall be provided at one location of each of the following:			
(a) "Free field." See note to (b).	(a)	Complies.	0-SG-AR-1
(b) Containment foundation. Note: If soil-structure interaction is negligible, a single instrument may be located on the "free field" or the containment foundation.	(b)	Complies.	0-SG-AE-1
(c) Containment structure or reactor building.	(c)	Complies.	0-SG-AE-2
4.1.2 One triaxial peak accelerograph shall be provided at one location of each of the following: (a) "Free field." (b) Reactor equipment. (c) One of the following: (1) Containment structure or reactor building, or (2) Reactor piping.		Section 4.1.2 is not used, in accordance with R.G. 1.12 regulatory position 1.a.	

TABLE 3.7B-9 (Sheet 5)

Regulatory Guide 1.12 Position	Union Electric	Instr. Tag No.
4.1.3 One triaxial seismic switch shall be provided at one location of the containment foundation.	Complies.	0-SG-AS-1 0-SG-AS-2

APPENDIX 3.7(B)A - IMPEDANCE FUNCTIONS FOR A RIGID CIRCULAR FOUNDATION ON A LAYERED VISCOELASTIC MEDIUM

A.1 FORMULATION OF THE PROBLEM

A.1.1 Statement of the Problem

In what follows, a study is made of the forced harmonic vibrations of a rigid circular footing of radius a placed on the surface of a layered viscoelastic medium. The layered medium consists of N-1 parallel layers resting on a viscoelastic half-space. Both the layers and the elastic half-space are assumed to be homogeneous and isotropic with densities ρ_i, shear moduli G_i, and Poisson's ratios $\sigma_i (i = 1, 2, \ldots N)$, respectively. In addition, depending on the type of internal friction considered, the relative viscosity coefficient (G_i' / G_i) (for Voigt type dissipation) or the hysteretic damping coefficient $\xi_i = \omega G_i' / 2G_i$ (for hysteretic type dissipation) are assumed to be known for each one of the media forming the soil deposit. The geometry of the model and the coordinate systems used are shown in Figure 3.7(B)A-1.

A welded type of contact is assumed to exist between adjacent layers. Thus, the stresses and displacements are continuous across each interface. The contact between the foundation and the surface of the top layer is assumed to be relaxed, i.e., the contact is frictionless for vertical and rocking vibrations and pressureless for horizontal vibrations.

The boundary conditions at z = 0 expressed in terms of displacement and stress components in cylindrical coordinates are the following:

a. Vertical Vibrations

$$u_z(r, \theta, 0) = \Delta_v e^{i\omega t} \qquad 0 \le r \le a \qquad \text{(A-1.a)}$$

$$\sigma_{zz}(r, \theta, 0) = 0 \qquad r > a \qquad \text{(A-1.b)}$$

$$\sigma_{zr}(r, \theta, 0) = \sigma_{z\theta}(r, \theta, 0) = 0 \qquad 0 < r < \infty \qquad \text{(A-2)}$$

b. Rocking Vibrations

$$u_z(r, \theta, 0) = \alpha r \cos\theta e^{i\omega t} \qquad 0 \le r \le a \qquad \text{(A-3.a)}$$

$$\sigma_{zz}(r, \theta, 0) = 0 \qquad r > a \qquad \text{(A-3.b)}$$

$$\sigma_{zr}(r, \theta, 0) = \sigma_{z\theta}(r, \theta, 0) = 0 \qquad 0 < r < \infty \qquad \text{(A-4)}$$

c. Horizontal Vibrations

$$\left.\begin{array}{c} (u_r(r, \theta, 0) = \Delta_H \cos\theta e^{i\omega t}) \\ 0 \le r \le a \\ u_\theta(r, \theta, 0) = -\Delta_H \sin\theta e^{i\omega t} \end{array}\right\} \qquad \text{(A-5)}$$

$$\sigma_{zr}(r, \theta, 0) = \sigma_{z\theta}(r, \theta, 0) = 0 \qquad r > a \qquad \text{(A-6)}$$

$$\sigma_{zz}(r, \theta, 0) = 0 \qquad 0 < r < \infty \qquad \text{(A-7)}$$

In the equations above, Δ_v is the amplitude of the vertical displacement of the center of the rigid foundation, a is the amplitude of the rocking angle about the y-axis ($\theta = \pi/2$), Δ_H is the amplitude of the horizontal displacement of the foundation in the direction of the x-axis ($\theta = 0$), and w is the frequency of the steady-state vibrations.

The continuity conditions at the interface z = H_i are:

$$u_r^i(r, \theta, H_i) = u_r^{i+1}(r, \theta, H_i) \qquad \text{(A-8.a)}$$

$$u_\theta^i(r, \theta, H_i) = u_\theta^{i+1}(r, \theta, H_i) \qquad \text{(A-8.b)}$$

$$u_z^i(r, \theta, H_i) = u_z^{i+1}(r, \theta, H_i), \qquad (i = 1, 2, \ldots N) \qquad \text{(A-8.c)}$$

$$u_{zr}^i(r, \theta, H_i) = u_{zr}^{i+1}(r, \theta, H_i) \qquad \text{(A-9.a)}$$

$$u_{z\theta}^i(r, \theta, H_i) = u_{z\theta}^{i+1}(r, \theta, H_i) \qquad \text{(A-9.b)}$$

$$u_{zz}^{i}(r, \theta, H_i) = u_{zz}^{i+1}(r, \theta, H_i), \quad (i = 1, 2, \ldots N) \tag{A-9.c}$$

where the superscript i indicates the i^{th} layer. In addition, the displacement and stress components in the underlying half-space must tend to zero as $(r^2 + z^2)$ tends to infinity.

A.1.2 Types of Energy Dissipation

In this study, two types of energy dissipation are considered, namely the Voigt viscous model and the hysteretic model.

The stress-strain relationships for harmonic vibrations of a solid with Voigt type damping are of the form (Ref. A-1)

$$\sigma_{zz} = (\lambda + i\omega\lambda')\textcircled{H} + 2(\mu + i\omega\mu')\varepsilon_{zz} \tag{A-10.a}$$

$$\sigma_{zx} = 2(\mu + i\omega\mu')\varepsilon_{xz} \tag{A-10.b}$$

where

$$\textcircled{H} = \varepsilon_{xx} + \varepsilon_{yy} + \varepsilon_{zz} \tag{A-10.b}$$

In equations (A-10.a) and (A-10.b), ω is the frequency of the excitation, λ and μ are Lame's constants, and λ', μ' are the viscosities. It is clear from equations (A-10.a) and (A-10.b) that the viscoelastic problem may be solved if the solution of the corresponding purely elastic problem is known by substituting in the elastic solution λ and μ by the complex moduli

$$\lambda^* = \lambda(1 + i\omega\lambda'/\lambda) \tag{A-11.a}$$

$$\mu^* = \mu(1 + i\omega\mu'/\mu) \tag{A-11.b}$$

In order to simplify the problem, it is assumed that

$$\frac{\lambda'}{\lambda} = \frac{\mu'}{\mu} \tag{A-12}$$

In this case, the remaining complex constants are given by:

$$E^* = \frac{(3\lambda^* + 2\mu^*)\mu^*}{\lambda^* + \mu^*} = E(1 + i\omega\mu'/\mu) \quad \text{(A-13.a)}$$

$$k^* = \lambda^* + \frac{2}{3}\mu^* = k(1 + i\omega\mu'/\mu) \quad \text{(A-13.b)}$$

$$\sigma^* = \frac{\lambda^*}{2(\lambda^* + \mu^*)} = \sigma \quad \text{(A-13.c)}$$

where E, k, and σ are the Young's modulus, the bulk modulus, and Poisson's ratio, respectively. The assumption given by equation (A-12) has the advantage that the Poisson's ratio for the viscoelastic medium is real and equal to the Poisson's ratio of the corresponding elastic medium. One disadvantage, however, is the fact that the bulk modulus is complex, and consequently there are losses associated with changes of volume.

Equation (A-10.b) indicates that for shear deformations the stress-strain relationship could be described by an ellipse.

The energy loss per cycle is given by the area of the ellipse and the corresponding “specific loss” is

$$\frac{\Delta W}{W} = 2\pi\frac{\omega\mu'}{\mu} \quad \text{(A-14)}$$

where W is the elastic energy stored when the strain is a maximum. Equation (A-14) indicates that for a Voigt solid the "specific loss," or the energy loss per cycle, is proportional to the frequency of the excitation. The elliptical stress-strain loop in this case is a direct result of the viscosity of the medium.

Laboratory tests on soils indicate that the "specific loss" $\Delta W/W$ is independent of the frequency of the excitation and that the stress-strain loop is not an ellipse (Ref. A-2 - A-6). It appears then that the mechanism of energy loss in soils is not of the viscous type but rather is a direct result of the anelastic behavior of soils. In spite of this anelastic behavior, an approximate approach is to assume that the soil may be treated in a similar way as a viscoelastic medium, except that in this case the complex shear modulus μ^* and the "specific loss" are taken to be equal to

$$\mu^* = \mu(1 + 2i\xi) \quad \text{(A-15)}$$

$$\frac{\Delta W}{W} = 4\pi\xi \qquad \text{(A-16)}$$

where ξ is a damping constant independent of frequency. This model of internal damping is also called constant hysteretic type damping. The damping constant ξ is analogous to the percentage of critical damping under resonant conditions, or during free vibrations (Ref. A-3). The hysteretic damping constant ξ is strain dependent: values for low strain may be less than 0.02, while for high strains ξ may reach values of 0.15 or 0.20.

In what follows, the shear modulus μ is designated by G, and the shear viscosity μ' is designated by G'.

A.1.3 Integral Representation

A solution of the equations of motion in cylindrical coordinates satisfying the conditions at the interface between layers, as well as the conditions at infinity, may be obtained by application of the correspondence principle to a representation derived by Sezawa and reported in references A-7 and A-8.

The displacement and stress components of interest on z = 0 are given by

$$u_r\ (r, \theta, 0) = a\ u_r^*\ (r')\cos(n\theta)$$

$$u_\theta\ (r, \theta, 0) = a\ u_\theta^*\ (r')\cos(n\theta) \qquad \text{(A-17)}$$

$$u_z\ (r, \theta, 0) = a\ u_z^*\ (r')\cos(n\theta)$$

$$u_{zr}\ (r, \theta, 0) = G_1\sigma_{zr}^*\ (r')\cos(n\theta)$$

$$u_{z\theta}\ (r, \theta, 0) = G_1\sigma_{z\theta}^*\ (r')\cos(n\theta) \qquad \text{(A-18)}$$

$$u_{zz}\ (r, \theta, 0) = G_1\sigma_{zz}^*\ (r')\cos(n\theta)$$

where n = 0 for vertical vibrations, n = 1 for rocking and horizontal vibrations, r' = r/a, and

$$u_r^*(r') \pm u_\theta^*(r') = \mp 2\int_o^\infty \{k[\Delta_{11}(k)C_1(k) + \Delta_{12}(k)C_2(k)]/\Delta_R \qquad \text{(A-19)}$$

$$\mp\Delta_{33}C_3(k)/\Delta_L\}J_{n\pm 1}(a_o kr')dk$$

$$u_z^*(r') = 2\int_o^\infty \{\{k[\Delta_{21}(k)C_1(k) + \Delta_{22}(k)C_2(k)]/\Delta_R\}J_n(a_o kr')\}dk \tag{A-20}$$

$$\sigma_{zr}^*(r') \pm \sigma_{z\theta}^*(r') = \mp 2a_o\int_o^\infty [kC_1(k) \mp C_3(k)]J_{n\pm 1}(a_o kr')dk \tag{A-21}$$

$$\sigma_{zz}^*(r') = 2a_o\int_o^\infty kC_2(k)J_n(a_o kr')dk \tag{A-22}$$

In equations (A-19) - (A-22), $= a_o = \omega a/\beta_1$ is a dimensionless frequency defined in terms of the shear wave velocity β_1 of the top layer. The functions $\Delta_{ij}(i, j = 1, 2)$, Δ_R, Δ_{33}, and Δ_L appearing in equations (A-19) - (A-22) depend on the properties of the soil column, and are given in Appendix 3.7(B)B. The functions $C_1(k)$, $C_2(k)$, and $C_3(k)$ are to be determined by the boundary conditions on z = 0. The term $J_n(a_o kr')$ is an infinite series known as the Bessel function of the first kind of order n while the term $J_{n+1}(a_o kr')$ is of the order n+1. For vertical and rocking vibrations, equations (A-2) and (A-4) together with equation (A-21) imply that

$$C_1(k) = C_3(k) = 0 \tag{A-23}$$

Similarly, for horizontal vibrations, equations (A-7) and(A-22) imply that

$$C_2(k) = 0 \tag{A-24}$$

Before imposing the remaining boundary conditions, it is convenient to introduce the following substitutions (Ref. A-7, A-9)

a. Vertical Vibrations

$$C_2(k) = -\left[\frac{\Delta_v k_1^2}{\pi a(1-\sigma_1)}a_o\right]\int_o^1 \phi_v(t)\cos(a_o kt)dt \tag{A-25}$$

b. Rocking Vibrations

$$C_2(k) = -\left[\frac{2\alpha\, k_1^2}{\pi(1-\alpha_1)}a_o\right]\int_o^1 \phi_R(t)\sin(a_okt)dt \qquad \text{(A-26)}$$

c. Horizontal Vibrations

$$C_1(k) = -\left[\frac{2\Delta_H\, k_1^2}{\pi a(2-\sigma_1)}a_o\right]\int_o^1\{\phi_1(t)\cos(a_okt) \qquad \text{(A-27)}$$

$$-\phi_2(t)[\cos(a_okt) - \sin(a_okt)/a_okt]\}dt$$

$$C_3(k) = -\left[\frac{2\Delta_H\, k_1^2}{\pi a(2-\sigma_1)}a_ok\right]\int_o^1\{-\phi_1(t)\cos(a_okt) \qquad \text{(A-28)}$$

$$-(1-\sigma_1)\phi_2(t)[\cos(a_okt)$$

$$-\sin(a_okt)/a_okt]\}dt$$

where $\phi_V(t)$, $\phi_R(t)$, and $\phi_1(t)$, $\phi_2(t)$ are functions to be determined by equations (A-1), (A-3), and (A-5), respectively. Also, $k_1^2 = (1 + i\omega G'_1/G_1)^{-1}$ for Voigt-type damping and $k_1^2 = (1 + 2i\xi_1)^{-1}$ for hysteretic-type damping. The substitutions indicated above satisfy directly the stress boundary conditions prescribed in equations (A-1), (A-3), and (A-6).

A.2 INTERNAL EQUATIONS AND IMPEDANCE FUNCTIONS

Substitutions from equations (A-25) - (A-28), together with equations (A-23) and (A-24), into equations (A-17), (A-19), and (A-20), and imposition of the remaining displacement boundary conditions leads to the following integral equations for the unknown functions $\phi_V(t)$, $\phi_R(t)$, and $\phi_2(t)$:

a. Vertical Vibrations

$$\phi_v(t) + \int_o^1 K(t, t')\phi_v(t')dt' = 1 \qquad (0 \le t \le 1) \qquad \text{(A-29)}$$

where

$$K(t, t') = L_1(t - t') + L_1(t + t') \qquad \text{(A-30)}$$

$$L_1(t) = -\frac{a_o}{\pi} \int_o^\infty \left[\frac{k\Delta_{22}}{(1 - \sigma_1)\Delta_R k_1^2} + 1 \right] \cos(a_o kt) dk \qquad \text{(A-31)}$$

b. Rocking Vibrations

$$\phi_R(t) + \int_o^1 K(t, t')\phi_R(t')dt' = t \qquad (0 \le t \le 1) \qquad \text{(A-32)}$$

where

$$K(t, t') = L_1(|t - t'|) - L_1(t + t') \qquad \text{(A-33)}$$

The function $L_1(t)$ in equation (A-33) is defined by equation (A-31).

c. Horizontal Vibrations

$$\phi_1(t) + \int_o^1 [K_{11}(t, t')\phi_1(t') + K_{12}(t, t')\phi_2(t')]dt' = 0 \qquad (0 \le t \le 1) \qquad \text{(A-34)}$$

$$(1 - \sigma_1)\phi_2(t) + \int_0^1 [K_{21}(t, t')\phi_1(t') + K_{22}(t, t')\phi_2(t')]dt' = 0 \qquad (0 \le t \le 1) \qquad \text{(A-35)}$$

where

$$K_{11}(t,t') = \frac{2a_o}{\pi}\left(\frac{1}{2-\sigma_1}\right)\int_o^{\infty}[(1-\sigma_1)H_1(k) + H_2(k)]\cos(a_okt)\cos(a_okt')dk \tag{A-36}$$

$$K_{12}(t,t') = \frac{2a_o}{\pi}\left(\frac{1-\sigma_1}{2-\sigma_1}\right)\int_o^{\infty}[H_1(k)-H_2(k)]\cos(a_okt)\left[\cos(a_okt') - \frac{\sin(a_okt')}{a_okt}\right]dk \tag{A-37}$$

$$K_{21}(t,t') = \frac{2a_o}{\pi}\left(\frac{1-\sigma_1}{2-\sigma_1}\right)\int_o^{\infty}[H_1(k)-H_2(k)]\left[\cos(a_okt') - \frac{\sin(a_okt)}{a_okt}\right]\cos(a_okt')dk \tag{A-38}$$

$$K_{22}(t,t') = \frac{2a_o}{\pi}\left(\frac{1-\sigma_1}{2-\sigma_1}\right)\int_o^{\infty}[H_1(k)+(1-\sigma_1)H_2k]\left[\cos(a_okt) - \frac{\sin(a_okt)}{a_okt}\right]\left[\cos(a_okt') - \frac{\sin(a_okt')}{a_okt'}\right]dk \tag{A-39}$$

and

$$H_1(k) = \frac{k}{k_1^2(1-\sigma_1)}\frac{\Delta_{11}}{\Delta_R} - 1 \tag{A-40}$$

$$H_2(k) = \frac{k\Delta_{33}}{k_1^2\Delta_L} - 1 \tag{A-41}$$

The integral equations (A-29), (A-32), (A-34), and (A-35) are of the Fredholm type and have a form suitable for numerical solution. Once these integral equations have been solved, the entire displacement and stress field may be evaluated by substitution from equations (A-25) - (A-28) into equations (A-19) - (A-22). In particular, the total vertical load V, the rocking moment about the y-axis M, and the total horizontal load in the x-direction H may be found to be given by

$$V = \frac{4G_1 a \Delta_v e^{i\omega t}}{(1-\sigma_1)k_1^2} \int_o^1 \phi_v(t)dt \qquad \text{(A-42)}$$

$$M = \frac{8G_1 a^3 \alpha e^{i\omega t}}{(1-\sigma_1)k_1^2} \int_o^1 t\phi_R dt \qquad \text{(A-43)}$$

$$H = \frac{8G_1 a \Delta_h e^{i\omega t}}{(2-\sigma_1)k_1^2} \int_o^1 \phi_1(t)dt \qquad \text{(A-44)}$$

Equations (A-42), (A-43), and (A-44) constitute the force-displacement relationship for the circular foundation. It should be mentioned that in deriving these equations, the terms coupling the horizontal and rocking vibrations have been neglected.

It is convenient to write equations (A-42) - (A-44) in the following form:

$$V = \frac{4G_1 a}{1-\sigma_1} [k_{VV}(a_o) + ia_o c_{VV}(a_o)]\Delta_v e^{i\omega t} \qquad \text{(A-45)}$$

$$M = \frac{8G_1 a^3}{3(1-\sigma_1)} [k_{MM}(a_o) + ia_o c_{MM}(a_o)]\alpha e^{i\omega t} \qquad \text{(A-46)}$$

$$H = \frac{2G_1 a}{2-\sigma_1} [k_{HH}(a_o) + ia_o c_{HH}(a_o)]\Delta_H e^{i\omega t} \qquad \text{(A-47)}$$

where,

$$k_{VV}(a_o) = \int_o^1 \mathrm{Re}\left[\phi_v(t)/k_1^2\right]dt \ ,$$

$$c_{VV}(a_o) = \frac{1}{a_o} \int_o^1 \mathrm{Im}\left[\phi_v(t)/k_1^2\right]dt \qquad \text{(A-48)}$$

$$k_{MM}(a_o) = 3 \int_o^1 \mathrm{Re}\left[t\phi_R(t)/k_1^2\right]dt,$$

$$c_{MM}(a_o) = \frac{3}{a_o} \int_o^1 \mathrm{Im}\left[t\phi_R(t)/k_1^2\right]dt \qquad \text{(A-49)}$$

$$k_{HH}(a_o) = \int_o^1 \mathrm{Re}\left[\phi_1(t)/k_1^2\right]dt,$$

$$c_{HH}(a_o) = \frac{1}{a_o} \int_o^1 \mathrm{Im}\left[\phi_1(t)/k_1^2\right]dt \qquad \text{(A-50)}$$

The terms inside the square brackets in equations (A-45), (A-46), and (A-47) are the normalized impedance functions for vertical, rocking, and horizontal vibrations; the factors outside the parentheses correspond to the static values (a_o = 0) of the impedance functions for an elastic half-space having the properties of the top layer. The functions $k_{VV}(a_o)$, $k_{MM}(a_o)$, and $k_{HH}(a_o)$, corresponding the real part, Re, of the impedance functions $c_{VV}(a_o)$, $c_{MM}(a_o)$, and $c_{HH}(a_o)$, proportional to the imaginary part, Im of the impedance functions, will be designed here as damping coefficients. Both the stiffness and damping coefficients are functions not only of the dimensionless frequency a_o but also depend on the properties of the different media forming the soil column.

In solving the problem of the horizontal vibrations, a further approximation has been introduced by assuming that $\phi_2(t)$ is sufficiently small so that the integral equations (A-34) and (A-35) may be reduced to

$$\tilde{\phi}_1(t) + \int_o^1 K_{11}(t, t')\tilde{\phi}_1(t')dt' = 1 \qquad (0 \le t \le 1) \qquad \text{(A-51)}$$

where the kernal $K_{11}(t, t')$ is given by equation (A-36). The basis for this approximation is that for the case of a uniform half-space, the function $\phi_2(t)$ is much smaller than $\phi_1(t)$, in particular, for the static case $\phi_2(t)$ = 0. The above approximation is equivalent to the

requirement that σ_{zy} = 0 under the foundation and thus corresponds to a further relaxation of the boundary conditions.

A.3 NUMERICAL SOLUTION

The numerical procedure used to solve the integral equations (A-29), (A-32), and (A-51) consists of reducing these equations to a system of algebraic equations that are solved by standard methods. A key step in this procedure is the evaluation of the kernels K(t, t') given by equations (A-30), (A-33), and (A-36). In the case of a medium with no internal friction, the functions Δ_R and Δ_L have zeroes for real values of k and, consequently, the integrands in equations (A-31) and (A-36) are singular at these points. This situation complicates the numerical evaluation of the kernels. However, if there is internal friction then the zeroes of Δ_R and Δ_L are complex, and consequently the numerical evaluation of the kernels is simplified. The kernels are evaluated numerically by use of Filon's method of integration up to a sufficiently large value of k, the rest is evaluated analytically by using the asymptotic forms of the integrands for large k.

A.4 REFERENCES

A-1. Kolsky, H., Stress Waves in Solids, Dover Publications, Inc., New York, 1963.

A-2. Richart, F. E., Hall, J. R., and Woods, R. D., Vibration of Soils and Foundations, Prentice-Hall, Inc., New Jersey, 1970.

A-3. Dobry, R., "Damping in Soils: Its Hysteretic Nature and the Linear Approximation," Research Report R70-14, Massachusetts Institute of Technology, Department of Civil Engineering, Cambridge, Mass., 1970.

A-4. Krizek, R. J., and Franklin, A. G., "Energy Dissipation in Soft Clay," Proc. Int. Symp. on Wave Propagation and Dynamic Properties of Earth Materials, University of New Mexico Press, Albuquerque, 1967, pp. 797-807.

A-5. Seed, H. B., and Idriss, I. M., "Soil Moduli and Damping Factors for Dynamic Response Analysis," Report EERC 70-10, University of California, Berkeley, 1970.

A-6. Hardin, B. O., and Drnevich, V. P., "Shear Modulus and Damping in Soils: Measurements and Parameter Effects," Proc. Am. Soc. Civ. Engs., Vol. 98, No. SM6, 1972, pp. 603-624.

A-7. Luco, J. E., and Westmann, R. A., "Dynamic Response of Circular Footings," Journal of the Engineering Mechs. Div., ASCE, Vol 97, No. EM5, 1971, pp. 1381-1395.

A-8. Bycroft, G. N., "Forced Vibrations of a Rigid Circular Plate on a Semi-Infinite Elastic Space or on an Elastic Stratum," Phil. Trans., Royal Soc. of London, Vol. 248, 1956, pp. 327-368.

A-9. Luco, J. E., "Impedance Functions for a Rigid Foundation on a Layered Medium", Nuclear Engineering and Design, 1974.

APPENDIX 3.7(B)B - SOIL DEPENDENT DISPLACEMENT FUNCTIONS FOR THE SOLUTION OF THE EQUATIONS OF MOTION

The functions $\Delta_{ij}(k)$ $(i, j = 1, 2)$ and $\Delta_R(k)$ entering in equations (A-19) and (A-20) are defined by

$$\begin{bmatrix} \Delta_{11}(k) & \Delta_{12}(k) \\ \Delta_{21}(k) & \Delta_{22}(k) \end{bmatrix} = (T_{11}*A + T_{12}*B)\mathrm{adj}(T_{21}*A + T_{22}*B) \qquad \text{(B-1)}$$

and,

$$\Delta_R = \det(T_{21}*A + T_{22}*B) \qquad \text{(B-2)}$$

where the matrices [A] and [B] are given by

$$[A] = \begin{bmatrix} -k & \nu'_N \\ \nu_N & -k \end{bmatrix} \qquad \text{(B-3)}$$

$$[B] = \frac{G_N*}{G_1}\begin{bmatrix} -2\upsilon\nu_N k & \left(2k^2 - k_N^2\right) \\ -\left(2k^2 - k_N^2\right) & 2\nu'_N k \end{bmatrix} \qquad \text{(B-4)}$$

and $T*_{ij}$ $(i, j = 1, 2)$ are the submatrices of the total transfer matrix $T*$ associated with the set of layers overlying the base half-space. The total transfer matrix $T*$

$$[T*] = \left[\begin{array}{c|c} T*_{11} & T*_{12} \\ \hline T*_{21} & T*_{22} \end{array}\right] \qquad \text{(B-5)}$$

may be obtained in terms of the transfer matrices for each layer T_j(j = 1, N - 1) by means of the following product:

$$[T*] = [T_1][T_2]\ldots[T_j]\ldots[T_{N-1}] \qquad \text{(B-6)}$$

The transfer matrix for the j^{th} layer is in turn given by

$$[T_j] = \begin{bmatrix} T^j_{11} & T^j_{12} \\ T^j_{21} & T^j_{22} \end{bmatrix} \tag{B-7}$$

where,

$$T^j_{11} = -\frac{1}{k_j^2}\begin{bmatrix} -2k^2CH_j + \left(2k^2 - k_j^2\right)CHP_j & -k\left(2_k^2 - k_j^2\right)SH_j + 2k\nu_j'^2SHP_j \\ 2k\nu_j^2SH_j - k\left(2k^2 - k_j^2\right)SHP_j & \left(2k^2 - k_j^2\right)CH_j - 2k^2CHP_j \end{bmatrix}$$

$$T^j_{12} = -\left(\frac{\rho_1}{\rho_j}\right)\begin{bmatrix} -k^2SH_j + \nu_j'^2SHP_j & k(CH_j - CHP_j) \\ k(CH_j - CHP_j) & -\nu_j^2SH_j + k^2SHP_j \end{bmatrix} \tag{B-8}$$

$$T^j_{21} = -\frac{1}{k_j^4}\left(\frac{\rho_i}{\rho_1}\right)\begin{bmatrix} -4\nu_j^2k^2SH_j + \left(2k^2 - k_j^2\right)^2SHP_j & -2k\left(2k^2 - k_j^2\right)(CH_j - CHP_j) \\ -2k\left(2k^2 - k_j^2\right)(CH_j - CHP_j) & -\left(2k^2 - k_j^2\right)^2SH_j + 4\nu_j^{22}k^2SHP_j \end{bmatrix}$$

$$T^j_{22} = -\frac{1}{k_j^2}\begin{bmatrix} -2k^2CH_j + \left(2k^2 - k_j^2\right)CHP_j & 2k\nu_j^2SH_j - k\left(2k^2 - k_j^2\right)SHP_j \\ -k\left(2k^2 - k_j^2\right)SH_j + 2\nu_j'^2kSHP_j & \left(2k^2 - k_j^2\right)CH_j - 2k^2CHP_j \end{bmatrix}$$

The different terms entering in equations (B-3) to (B-8) are defined by

$$\nu_j = \left(k^2 - \gamma_j^2k_j^2\right)^{1/2} \qquad \nu_j^i = \left(k^2 - k_j^2\right)^{1/2}$$

$$\gamma_j^2 = (1 - 2\sigma_j)/2(1 - \sigma_j) \qquad k_j^2 = G_1\rho_j/G_j{*}\rho_1$$

$$G_j{*} = G_j\left(1 + i\omega G_j'/G_j\right), \text{ or, } G_j{*} = G_j(1 + 2i\xi_j)$$

$$SH_j = \sinh(a_o\nu_j\lambda_j)/\nu_j \qquad SPH_j = \sinh(a_o\nu_j'\lambda_j)/\nu_j' \qquad \text{(B-9)}$$

$$CH_j = \cosh(a_o\nu_j\lambda_j) \qquad CHP_j = \cosh(a_o\nu_j'\lambda_j)$$

$$\lambda_j = h_j/a$$

$$a_o = \omega a/\beta_1$$

where σ_j, ρ_j, G_j, G_j'/G_j, and h_j, respectively, are the Poisson's ratio, density, shear modulus, relative viscosity, and thickness of the j^{th} layer. In the last two equations of (B-9), a is the radius of the circular foundation, ω is the frequency of the steady-state vibrations, and β_1 is the shear wave velocity of the top layer. The first form of G_j^* corresponds to the Voigt-type damping, while the second corresponds to the hysteretic-type damping, ξ_j being the hysteretic damping constant for the j^{th} layer. The functions $\Delta_{33}(k)$ and $\Delta_L(k)$ entering in equation (A-19) are defined by

$$\Delta_{33}(k) = L_{11}{*} + L_{12}{*}\nu_N' G_N{*}/G_1 \qquad \text{(B-10)}$$

$$\Delta_L(k) = L_{21}{*} + L_{22}{*}\nu_N' G_N{*}/G_1 \qquad \text{(B-11)}$$

where L_{ij}^* $(i, j = 1, 2)$ are the elements of the transfer matrix $L*$. The transfer matrix $L*$

$$[L*] = \begin{bmatrix} L_{11}^* & L_{12}^* \\ L_{21}^* & L_{22}^* \end{bmatrix} \qquad \text{(B-12)}$$

is defined in terms of the transfer matrices for each layer by

$$[L*] = [L_1]\cdot[L_2]\ldots[L_j]\ldots[L_{N-1}] \qquad \text{(B-13)}$$

in which,

$$[L_j] = \begin{bmatrix} CHP_j & (G_1/G_j^*)SHP_j \\ (G_j^*/G_1)\nu_j'^2 SHP_j & CHP_j \end{bmatrix} \qquad \text{(B-14)}$$

3.7(N) SEISMIC DESIGN

For the OBE loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. The design for the SSE is intended to ensure:

a. That the integrity of the reactor coolant pressure boundary is not compromised;

b. That the capability to shut down the reactor and maintain it in a safe condition is not compromised; and

c. That the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100 is not compromised.

It is necessary to ensure that required critical structures and components do not lose their capability to perform their safety function. Not all critical components have the same functional safety requirements. For example, a safety injection pump must retain its capability to function normally during the SSE. Therefore, the deformation in the pump must be restricted to appropriate limits in order to ensure its ability to function. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

The seismic requirements for safety-related instrumentation and electrical equipment are covered in Sections 3.10(N) and (B). The safety class definitions, classification lists, operating condition categories, and the methods used for seismic qualification of mechanical equipment are given in Section 3.2.

3.7(N).1 SEISMIC INPUT

3.7(N).1.1 Design Response Spectra

Refer to Section 3.7(B).1.1.

3.7(N).1.2 Design Time History

Refer to Section 3.7(B).1.2.

3.7(N).1.3 Critical Damping Values

The damping values given in Table 3.7(N)-1 are used for the systems analysis of Westinghouse equipment and for the component analysis of the Integrated Head Assembly (IHA) and replaced steam generators (SGs). These are consistent with the damping values recommended in Regulatory Guide 1.61, Rev. 0, except in the case of the primary coolant loop system components and large piping (excluding reactor

pressure vessel internals) for which the damping values of 2 percent and 4 percent are used as established in testing programs reported in Reference 1 and the IHA as noted below. The damping values for control rod drive mechanisms (CRDMs) and the fuel assemblies of the nuclear steam supply system, when used in seismic system analysis, are in conformance with the values for welded and/or bolted steel structures (as appropriate) listed in Regulatory Guide 1.61, Rev. 0.

Tests on fuel assembly bundles justified conservative component damping values of 7 percent for OBE and 10 percent for SSE to be used in the fuel assembly component qualification. Documentation of the fuel assembly tests is found in Reference 2.

The damping values used in component analysis of CRDMs and their seismic supports were developed by testing programs performed by Westinghouse. These tests were performed during the design of the CRDM support; the support was designed so that the damping in Table 3.7(N)-1 could be conservatively used in the seismic analysis. The CRDM support system is designed with plates at the top of the mechanism and gaps between mechanisms. These are encircled by a box section frame which is attached by tie rods to the refueling cavity wall. The test conducted was on a full-size CRDM complete with rod position indicator coils, attachment to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support. The internal pressure of the CRDM was 2,250 psi, and the temperature on the outside of the pressure housing was 400°F.

The program consisted of transient vibration tests in which the CRDM was deflected a specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variations in the drive shaft axial position, upper seismic support clearance, and initial deflection amplitude was investigated.

The upper support clearance had the largest effect on the CRDM damping, with the damping increasing with increasing clearance. With an upper clearance of 0.06 inch, the measured damping was approximately 8 percent. The clearance in a typical upper seismic CRDM support is a minimum of 0.10 inch. The increasing damping with increasing clearances trend from the test results indicated that the damping would be greater than 8 percent for both the OBE and the SSE, based on a comparison between typical deflections during these seismic events to the initial deflections of the mechanisms in the test. Component damping values of 5 percent are, therefore, conservative for both the OBE and the SSE.

These damping values are used and applied to CRDM component analysis by response spectra techniques.

The dampling values for the Integrated Head Assembly (IHA) are also given in Table 3.7(N)-1. These damping values are based on the note from Regulatory Guide 1.61 Revision 1, Table 1, allowing the use of calculated “weighted average” damping value for a structure with a combination of different connection types, for the design-basis Safe

Shutdown Earthquake (SSE). The note from Table 1 was also applied to Table 2 of Regulatory Guide 1.61 Revision 1 to determine the IHA design-basis Operating Basis Earthquake (OBE) damping value. This methodology was approved in the NRC Safety Evaluation Report dated January 14, 2014 issued for Amendment 207 to the Callaway Operating License.

3.7(N).1.4 Supporting Media for Seismic Category I Structures

Refer to Section 3.7(B).1.4.

3.7(N).2 SEISMIC SYSTEM ANALYSIS

This section describes the methods of seismic analysis performed for safety-related components and systems within Westinghouse's scope, unless noted otherwise.

3.7(N).2.1 Seismic Analysis Methods

Those components and systems that must remain functional in the event of the SSE (seismic Category I) are identified by applying the criteria of Section 3.2.1.

In general, the dynamic analyses are performed, using a modal analysis plus either the response spectrum analysis or integration of the uncoupled modal equations as described in Sections 3.7(N).2.1.3 and 3.7(N).2.1.4, respectively, or by direct integration of the coupled differential equations of motion described in Section 3.7(N).2.1.5.

3.7(N).2.1.1 Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dashpots suitable for mathematical analysis. The essence of this step is to select a model so that the displacements obtained will be a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure or component. Some typical modeling techniques are presented in Reference 3.

Equations of Motion

Consider the multidegree of freedom system shown in Figure 3.7(N)-1. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r \ddot{y}_r + \sum^{i} c_{ri} \dot{u}_i + \sum^{i} k_{ri} u_i = 0 \qquad (3.7(N)\text{-}1)$$

where:

m_r = the value of the mass or mass moment of rotational inertia at mass point r

$\ddot{y}_r$ = absolute translational or angular acceleration of mass point r

c_{ri} = damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i, maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity

$\dot{u}_i$ = translational or angular velocity of mass point i relative to the base

k_{ri} = stiffness coefficient - the external force (moment) required at mass point r to produce a unit deflection (rotation) at mass point i, maintaining zero displacement (rotation) at all other mass points.

Force (moment) is positive in the direction of positive displacement (rotation)

u_i = displacement (rotation) of mass point i relative to the base

As an example, note that Figure 3.7(N)-1 does not attempt to show all of the springs (and none of the dashpots) which are represented in Equation 3.7(N)-1.

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad (3.7(N)\text{-}2)$$

where:

$\ddot{y}_s$ = absolute translational (angular) acceleration of the base

$\ddot{u}_r$ = translational (angular) acceleration of mass point r relative to the base

Equation 3.7(N)-1 can be written as:

$$m_r\ddot{u}_r + \sum^{i} c_{ri}\dot{u}_i + \sum^{i} k_{ri}u_i = -m_r\ddot{y}_s \qquad (3.7(N)\text{-}3)$$

For a single degree of freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad (3.7(N)\text{-}4)$$

3.7(N).2.1.2 Modal Analysis

Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which are determined by eigen solution of Equation 3.7(N)-3. The right hand side and the damping term as set equal to zero for this purpose, as illustrated in Reference 4 (Pages 83 through 111). Thus, Equation 3.7(N)-3 becomes:

$$m_r\ddot{u}_r + \sum^{i} k_{ri} u_i = 0 \qquad (3.7(N)\text{-}5)$$

The equation given for each mass point r in Equation 3.7(N)-5 can be written as a system of equations in matrix form as:

$$[M]\{\Delta\} + [K]\{\Delta\} = 0 \qquad (3.7(N)\text{-}6)$$

where:

$[M]$ = mass and rotational inertia matrix

$\{\Delta\}$ = column matrix of the general displacement and rotation at each mass point relative to the base

$[K]$ = square stiffness matrix

$\{\Delta\}$ = column matrix of general translational and angular accelerations at each mass point relative to the base, $d^2\{\Delta\}/dt^2$

Harmonic motion is assumed, and the $\{\Delta\}$ is expressed as:

$$\{\Delta\} = \{\delta\}\sin\,\omega t \qquad (3.7(N)\text{-}7)$$

where:

$\{\delta\}$ = column matrix of the spatial displacement and rotation at each mass point relative to the base

ω = natural frequency of harmonic motion in radians per second

The displacement function and its second derivative are substituted into Equation 3.7(N)-6 and yield:

$$[K]\{\delta\} = \omega^2[M]\{\delta\} \qquad (3.7(N)\text{-}8)$$

The determinant $\left|[K] - \omega^2[M]\right|$ is set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation 3.7(N)-8. This yields n natural frequencies and mode shapes where n equals the number of dynamic degrees of freedom of the system. The mode shapes are all orthogonal to each other and are sometimes referred to as normal mode vibrations. For a single degree of freedom system, the stiffness matrix and mass matrix are single terms and the determinant $\left|[K] - \omega^2[M]\right|$ when set equal to zero yields simply:

$$k - \omega^2 m = 0$$

or (3.7(N)-9)

$$\omega = \sqrt{\frac{k}{m}}$$

where ω is the natural angular frequency in radians per second.

The natural frequency in cycles per second is, therefore:

$$f = \frac{1}{2\pi}\sqrt{\frac{k}{m}} \qquad (3.7(N)\text{-}10)$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n can be substituted in Equation 3.7(N)-8.

Modal Equations

The response of a structure or component is always some combination of its normal modes. Good accuracy can usually be obtained by using only the first few modes of

vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as (Ref. 4, Pages 116 through 125):

$$\ddot{A}_n + 2\omega_n p_n \dot{A}_n + \omega_n^2 A_n = -\Gamma_n \ddot{y}_s \qquad (3.7(N)\text{-}11)$$

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u_{rn}, by the equation:

$$u_{rn} = A_n \phi_{rn} \qquad (3.7(N)\text{-}12)$$

where:

ω_n = natural frequency of mode n in radians per second

p_n = critical damping ratio of mode n

Γ_n = modal participation factor of mode n given by:

$$\Gamma_n = \frac{\Sigma^n m_r \phi'_{rn}}{\Sigma^n m_r \phi_{rn}^2} \qquad (3.7(N)\text{-}13)$$

where:

ϕ'_{rn} = value of ϕ_{rn} in the direction of the earthquake

The essence of the modal analysis lies in the fact that Equation 3.7(N)-11 is analogous to the equation of motion for a single degree of freedom system that will be developed from Equation 3.7(N)-4. Dividing Equation 3.7(N)-4 by m gives:

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \qquad (3.7(N)\text{-}14)$$

The critical damping ratio of the single degree of freedom system, p, is defined by the equation:

$$p \equiv \frac{c}{c_c} \qquad (3.7(N)\text{-}15)$$

where the critical damping coefficient is given by the expression:

$$c_c = 2m\omega \qquad (3.7(N)\text{-}16)$$

Substituting Equation 3.7(N)-16 into Equation 3.7(N)-15 and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \qquad (3.7(N)\text{-}17)$$

Subsituting this expression and the expression for k/m given by Equation 3.7(N)-9 into Equation 3.7(N)-14 gives:

$$\ddot{u} + 2\omega p u + \omega^2 u = -\ddot{y}_s \qquad (3.7(N)\text{-}18)$$

Note the similarity of Equations 3.7(N)-11 and 3.7(N)-18. Thus each mode may be analyzed as though it were a single degree of freedom system, and all modes are independent of each other. By this method, a fraction of critical damping, i.e., c/c_c, may be assigned to each mode, and it is not necessary to identify or evaluate individual damping coefficients, i.e., c. However, assigning only a single damping ratio to each mode has a drawback. There are three ways used to overcome this limitation when considering a slightly damped structure (e.g., steel) supported by a massive moderately damped structure (e.g., concrete).

The first method is to develop and analyze separate mathematical models for both structures, using their respective damping values. The massive, moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures is used as a forcing function for the subsequent detailed analysis. The second method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion. The third method is to use the Rayleigh damping method based on computed modal energy distribution.

3.7(N).2.1.3 Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (Ref. 5, Pages 24 through 51) (displacement, velocity, and acceleration) of a single degree of freedom system versus its natural frequency of vibration when subjected to a time-history motion of its base.

The response spectrum concept can be best explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree of freedom system with a given damping and natural frequency is calculated for a given base motion.

The variations in response are established, and the maximum absolute value of each is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency in sufficient detail to establish the complete curve. Other curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis, and the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with the base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$s_{a_n} = \omega_n s_{v_n} = \omega_n^2 s_{d_n} \qquad (3.7(N)\text{-}19)$$

There are two types of response spectra that must be considered. If a given building is shown to be rigid and to have a hard foundation, the ground response spectrum or ground time-history is used. It is referred to as a ground response spectrum. If the building is flexible and/or has a soft foundation, the ground response spectrum is modified to include these effects. The response spectrum at various support points must be developed. These are called floor response spectra.

3.7(N).2.1.4 Integration of Modal Equations

This method can be separated into the following two basic parts:

a. Integration procedure for the uncoupled modal Equation 3.7(N)-11 to obtain the modal displacements and accelerations as a function of time.

b. Using these modal displacements and accelerations to obtain the total displacements, accelerations, forces, and stresses.

<u>Integration Procedure</u>

Integration of these uncoupled modal equations is done by step-by-step numerical integration. The step-by-step numerical integration procedure consists of selecting a suitable time interval, Δt, and calculating modal acceleration, $\ddot{A}_n$, modal velocity, $\dot{A}_n$, and modal displacement, A_n, at discrete time stations Δt apart, starting at t = 0 and continuing through the range of interest for a given time-history of base acceleration.

Total Displacements, Accelerations, Forces, and Stresses

From the modal displacements and accelerations, the total displacements, accelerations, forces, and stresses can be determined as follows:

a. Displacement of mass point r in mode n as a function of time is given by Equation 3.7(N)-12 as:

$$u_{rn} = A_n \phi_{rn} \qquad (3.7(N)\text{-}20)$$

with the corresponding acceleration of mass point r in mode n as:

$$\ddot{u}_{rn} = \ddot{A}_n \phi_{rn} \qquad (3.7(N)\text{-}21)$$

b. The displacement and acceleration values obtained for the various modes are superimposed algebraically to give the total displacement and acceleration at each time interval.

c. The total acceleration at each time interval is multiplied by the mass to give an equivalent static force. Stresses are calculated by applying these forces to the model or from the deflections at each time interval.

3.7(N).2.1.5 Integration of Coupled Equations of Motion

The dynamic transient analysis is a time-history solution of the response of a given structure to known forces and/or displacement forcing functions. The structure may include linear or nonlinear elements, gaps, interfaces, plastic elements, and viscous and Coulomb dampers. Nodal displacements, nodal forces, pressure, and/or temperatures may be considered as forcing functions. Nodal displacements and elemental stresses for the complete structure are calculated as functions of time.

The basic equations for the dynamic analysis are as follows:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} \qquad (3.7(N)\text{-}22)$$

where the terms are as defined earlier and {F(t)} may include the effects of applied displacements, forces, pressures, temperatures, or nonlinear effects such as plasticity

and dynamic elements with gaps. Options of translational accelerations input to a structural system and the inclusion of static deformation and/or preload may be considered in the nonlinear dynamic transient analysis. The option of translational input such as uniform base motion to a structural system is considered by introducing an intertia force term of $-M\{\ddot{z}\}$ to the right hand side of the basic Equation 3.7(N)-22, i.e.,

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} - [M]\{\ddot{z}\} \qquad (3.7(N)\text{-}23)$$

The vector $\{\ddot{z}\}$ is defined by its components $\ddot{z}_i$ where i refers to each degree of freedom of the system. $\ddot{z}_i$ is equal to a_1, a_2, or a_3 if the i-th degree for freedom is aligned with the direction of the system translational acceleration a_1, a_2, or a_3, respectively. $\ddot{z}_i = 0$ if the i-th degree of freedom is not aligned with any direction of the system translational acceleration. Typical application of this option is a structural system subjected to a seismic excitation of a given ground acceleration record. The displacement $\{x\}$ obtained from the solution of Equation 3.7(N)-23 is the displacement relative to the ground.

The option of the inclusion of initial static deformation or preload in a nonlinear transient dynamic structural analysis is considered by solving the static problem prior to the dynamic analysis. At each state of integration in transient analysis, the portion of internal forces due to static deformation is always balanced by the portion of the forces which is statically applied. Hence, only the portion of the forces which deviates from the static loads will produce dynamic effects. The output of this analysis is the total result due to static and dynamic applied loads.

One available method for the numerical integration of Equations 3.7(N)-22 and 3.7(N)-23 is the Newmark Beta integration scheme proposed by Chan, Cox, and Benfield (Ref. 6). In this integration scheme, Equations 3.7(N)-22 and 3.7(N)-23 are replaced by:

$$\begin{aligned} &\frac{1}{(\Delta t)^2}[M]\{x_{n+2} - 2x_{n+1} + x_n\} + \frac{1}{2(\Delta t)}\{x_{n+2} - x_n\}[C] \\ &+ [K]\{\beta x_{n+2} + (1-2\beta)x_{n+1} + \beta x_n\} \qquad (3.7(N)\text{-}24) \\ &= \{\beta F_{n+2} + (1-2\beta)F_{n+1} + \beta F_n\} \end{aligned}$$

where:

n, n+1, n+2 = past, present, and future (updated)values of the variables

β = parameter to be selected on the basis of numerical stability and accuracy

F = the total right hand side of the equation of motion (Equation 3.7(N)-22 or 3.7(N)-23)

Δt = $t_{n+2} - t_{n+1} = t_{n+1} - t_n$

The value of β is chosen equal to 1/3 in order to provide a margin of numerical stability for nonlinear problems. Since the numerical stability of Equation 3.7(N)-24 is mostly determined by the left hand side terms of that equation, the right hand side terms were replaced by F_{n+2}. Furthermore, since the time increment may vary between two successive time substeps, Equation 3.7(N)-24 may be modified as follows:

$$\frac{2}{(\Delta t + \Delta t_1)} [M]\left\{\frac{x_{n+2} - x_{n+1}}{\Delta t} - \frac{x_{n+1} - x_n}{\Delta t}\right\}$$
$$+ \frac{1}{(\Delta t + \Delta t_i)} [C]\{x_{n+2} - x_n\} + \frac{1}{3}[K] \qquad (3.7(N)\text{-}25)$$
$$\{x_{n+2} + x_{n+1} + x_n\} = F_{n+2}$$

By factoring x_{n+2}, x_{n+1}, and x_n, and rearranging terms, Equation 3.7(N)-26 is obtained as follows:

$$\begin{aligned} &\{C_5[M] + C_3[C] + (1/3)[K]\}\{x_{n+2}\} \\ &= \{F_{n+2}\} + \{C_7[M] - (1/3)[K]\}\{x_{n+1}\} \\ &+ \{-C_2[M] + C_3[C] - (1/3)[K]\}\{x_n\} \end{aligned} \qquad (3.7(N)\text{-}26)$$

where:

$$C_2 = \frac{2}{\Delta t_1(\Delta t + \Delta t_1)}$$

$$C_3 = \frac{1}{\Delta t + \Delta t_1}$$

$$C_5 = \frac{2}{\Delta t(\Delta t + \Delta t_1)}$$

$$C_7 = C_2 + C_5$$

The above set of simultaneous linear equations is solved to obtain the present values of nodal displacements $\{x_t\}$ in terms of the previous (known) values of the nodal displacements. Since $[M]$, $[C]$, and $[K]$ are included in the equation, they can also be time or displacement dependent.

3.7(N).2.2 Natural Frequencies and Response Loads

Refer to Section 3.7(B).2.2.

3.7(N).2.3 Procedures Used for Modeling

Procedures used for modeling are discussed in Section 3.7(N).2.1.1.

3.7(N).2.4 Soil/Structure Interaction

Refer to Section 3.7(B).2.4.

3.7(N).2.5 Development of Floor Response Spectra

Refer to Section 3.7(B).2.5.

3.7(N).2.6 Three Components of Earthquake Motion

The seismic design of the piping and equipment includes the effect of the seismic response of the supports, equip-ment, structures, and components. The system and equipment response is determined, using three earthquake components--two horizontal and one vertical. The design ground response spectra are the bases for generating these three input components. Floor response spectra are generated for two perpendicular horizontal directions (i.e., N-S, E-W) and the vertical direction. System and equipment analysis is performed with these input components applied in the N-S, E-W, and vertical direction. The damping values used in the analysis are those given in Table 3.7(N)-1.

In computing the system and equipment response-by-response spectrum modal analysis, the methods of Section 3.7(N).2.7 are used to combine all significant modal responses to obtain the combined unidirectional responses.

The combined total response is then calculated, using the square root of the sum of the squares formula applied to the resultant unidirectional responses. For instance, for each item of interest such as displacement, force, stresses, etc., the total response is obtained

by applying the above-described method. The mathematical expression for this method (with R as the item of interest) is:

$$R_C = \left[\sum_{T=1}^{3} R_T^2 \right]^{1/2} \qquad (3.7(N)\text{-}27)$$

where:

$$R_T = \left[\sum_{i=1}^{N} R_{Ti}^2 \right]^{1/2} \qquad (3.7(N)\text{-}28)$$

where:

R_C	=	total combined response at a point
R_T	=	value of combined response of direction T
R_{Ti}	=	absolute value of response for direction T, mode i
N	=	total number of modes considered

The subscripts can be reversed without changing the results of the combination.

Again, for the case of closely spaced modes, R_T in Equation 3.7(N)-28 shall be replaced with R_T as given by Equation 3.7(N)-29 in Section 3.7(N).2.7.

3.7(N).2.7 Combination of Modal Response

The total unidirectional seismic response is obtained by combining the individual modal responses, utilizing the square root of the sum of the squares method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen so that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10 percent of the lower frequency. Groups are formed, starting from the lowest frequency and working toward successively higher frequencies. No one frequency is in more than one group. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes the product of the responses of the modes in each

group of closely spaced modes and a coupling factor ε. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + \sum_{j=1}^{S} \sum_{K=M_j}^{N_j-1} \sum_{l=K+1}^{N_j} R_K R_l \varepsilon_{Kl} \qquad (3.7(N)\text{-}29)$$

where:

R_T = total unidirectional response

R_i = absolute value of response of mode i

N = total number of modes considered

S = number of groups of closely spaced modes

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

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

ε_{Kl} = coupling factors with:

$$\varepsilon_{Kl} = \left\{1 + \left[\frac{\omega'_K - \omega'_l}{(\beta'_K \omega_K + \beta'_l \omega_l)}\right]^2\right\}^{-1} \qquad (3.7(N)\text{-}30)$$

and

$$\omega'_K = \omega_K[1 - (\beta'_K)^2 1/2] \qquad (3.7(N)\text{-}31)$$

$$\beta'_K = \beta_K + \frac{2}{\omega_K t_d} \qquad (3.7(N)\text{-}32)$$

where:

ω_K = frequency of closely spaced mode K

β_K = fraction of critical damping in closely spaced mode K

t_d = duration of the earthquake

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Frequency	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20

There are two groups of closely spaced modes, namely with modes {2,3,4} and {6, 7}. Therefore:

S	=	2	number of groups of closely spaced modes
M_1	=	2	lowest modal number associated with group 1
N_1	=	4	highest modal number associated with group 1
M_2	=	6	lowest modal number associated with group 2
N_2	=	7	highest modal number associated with group 2
N	=	8	total number of modes considered

The total response for this system is, as derived from the expansion of Equation 3.7(N)-29:

$$R_T^2 = R_1^2 + R_2^2 + R_3^2 + \dots + R_8^2 + 2 R_2 R_3 \varepsilon_{23} + 2 R_2 R_4 \varepsilon_{24} \qquad (3.7(N)\text{-}33)$$

$$+ 2R_3R_4\varepsilon_{34} + 2R_6R_7\varepsilon_{67}$$

3.7(N).2.8 Interaction of Non-Category I Structures With Seismic Category I Structures

Refer to Section 3.7(B).2.8.

3.7(N).2.9 Effects of Parameter Variations on Floor Response Spectra

Refer to Section 3.7(B).2.9.

3.7(N).2.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used as the vertical floor response load for the seismic design of safety classed systems and components within Westinghouse's scope of responsibility. All such systems and components are analyzed in the vertical direction.

3.7(N).2.11 Methods Used to Account for Torsional Effects

Refer to Section 3.7(B).2.11.

3.7(N).2.12 Comparison of Responses

Refer to Section 3.7(B).2.12.

3.7(N).2.13 Methods for Seismic Analysis of Dams

Refer to Section 3.7(B).2.13.

3.7(N).2.14 Determination of Seismic Category I Structure Overturning Moments

Refer to Section 3.7(B).2.14.

3.7(N).2.15 Analysis Procedure for Damping

In instances under the standard scope of Westinghouse supply and analysis, either the lowest damping value associated with the elements of the system is used for all modes, or an equivalent modal damping value is determined by testing programs, such as was done for the reactor coolant loop (Ref. 5).

As noted in Table 3.7(N)-1, the IHA damping values have been determined per the recommendations of Regulatory Guide 1.61, Rev. 1.

3.7(N).3 SEISMIC SUBSYSTEM ANALYSIS

This section describes the seismic analysis performed on subsystems within Westinghouse's scope of responsibility.

3.7(N).3.1 Seismic Analysis Methods

Seismic analysis methods for subsystems within Westinghouse's scope of responsibility are given in Section 3.7(N).2.1.

3.7(N).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 OBEs must be evaluated to assure resistance to cyclic loading.

The OBE is conservatively assumed to occur 20 times over the life of the plant. The number of maximum stress cycles for each occurrence depends on the system and component damping values, complexity of the system and component, and duration and frequency contents of the input earthquake. A precise determination of the number of maximum stress cycles can only be made, using time-history analysis for each item which is not feasible. Instead, a time-history study has been conducted to arrive at a realistic number of maximum stress cycles for all Westinghouse systems and components and for the IHA and replacement SGs.

To determine the conservative equivalent number of cycles of maximum stress associated with each occurrence, an evaluation was performed, considering both equipment and its supporting building structure as single degree of freedom systems. The natural frequencies of the building and the equipment are conservatively chosen to coincide. The damping in the equipment and building is equivalent to the damping values in Table 3.7(N)-1.

The results of this study indicate that the total number of maximum stress cycles in the equipment having peak acceleration above 90 percent of the maximum absolute acceleration did not exceed 8 cycles.

If the equipment was assumed to be rigid in a flexible building, the number of cycles exceeding 90 percent of the maximum stress was not greater than 3 cycles.

This study was conservative since it was performed with single degree of freedom models which tend to produce a more uniform and unattenuated response than a complex interacting system. The conclusions indicate that 10 maximum stress cycles for flexible equipment (natural frequencies less than 33 Hz) and 5 maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz) for each of 20 OBE occurrences should be used for fatigue evaluation of Westinghouse systems and components.

3.7(N).3.3 Procedure Used for Modeling

Refer to Section 3.7(N).2.1 for modeling procedures for subsystems in Westinghouse's scope of responsibility and for the IHA and replacement SGs.

3.7(N).3.4 Basis for Selection of Frequencies

The analysis of equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports, based upon the mass and stiffness characteristics of the system will determine the seismic accelerations which the equipment must withstand.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

a. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low period region of the floor response spectra.

b. If the equipment is very flexible, relative to the structure, the equipment will show very little response.

c. If the periods of the equipment and supporting structure are nearly equal, response occurs and must be taken into account.

In all cases, equipment under earthquake loadings is designed to be within Code allowable stresses.

Also, as noted in Section 3.7 (N).3.2, rigid equipment/support systems have natural frequencies greater than 33 Hz.

3.7(N).3.5 Use of Equivalent Static Load Method of Analysis

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component number by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as single degree of freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multidegree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50 percent to account conservatively for the increased modal participation.

3.7(N).3.6 Three Components of Earthquake Motion

Methods used to account for three components of earthquake motion for subsystems in Westinghouse's scope of responsibility and for the IHA and replacement SGs are given in Section 3.7(N).2.6.

3.7(N).3.7 Combination of Modal Responses

Methods used to combine modal responses for subsystems in Westinghouse's scope of responsibility and for the IHA and replacement SGs are given in Section 3.7(N).2.7.

3.7(N).3.8 Analytical Procedures for Piping

The Class 1 piping systems are analyzed to the rules of the ASME Code, Section III, NB-3650. When response spectrum methods are used to evaluate piping systems supported at different elevations,the following procedures are used. The effect of differential seismic movement of piping supports is included in the piping analysis, according to the rules of the ASME Code, Section III, NB-3653. According to ASME definitions, these displacements cause secondary stresses in the piping system. The response quality of interest induced by differential seismic motion of the support is computed statically by considering the building response on a mode-by-mode basis.

In the response spectrum dynamic analysis for evaluation of piping systems supported at different elevations, the most severe floor response spectrum corresponding to the support locations is used. Westinghouse does not have in their scope of analysis any piping systems interconnected between buildings.

3.7(N).3.9 Multiple Supported Equipment Components with Distinct nputs

When response spectrum methods are used to evaluate reactor coolant system primary components interconnected between floors, the procedures of the following paragraphs are used. There are no components in the Westinghouse scope of analysis which are connected between buildings. The primary components of the reactor coolant system are supported at no more than two floor elevations.

A dynamic response spectrum analysis is first made, assuming no relative displacement between support points. The response spectra used in this analysis is the most severe floor response spectra.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis. The results of the building analysis are reviewed on a mode-by-mode basis to determine the differential motion in each mode. Per ASME Code rules, the stress caused by differential seismic motion is clearly secondary for piping (NB-3650) and component supports (NF-3231). For components, the differential motion will be evaluated as a free end displacement, since, per NB-3213.19, examples of a free end displacement are motions "that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping." The effect of the differential motion is to impose a rotation on the component from the building. This motion, then, being a free end displacement and being similar to thermal expansion

loads, will cause stresses which will be evaluated with ASME Code methods, including the rules of NB-3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7(N).3.10 Use of Constant Vertical Static Factors

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

3.7(N).3.11 Torsional Effects of Eccentric Masses

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

3.7(N).3.12 Buried Seismic Category I Piping Systems and Tunnels

Refer to Section 3.7(B).3.12.

3.7(N).3.13 Interaction of Other Piping with Seismic Category I Piping

Refer to Section 3.7(B).3.13.

3.7(N).3.14 Seismic Analyses for Reactor Internals

Fuel assembly component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling.

The time-history floor response based on a standard seismic time-history normalized to SSE levels is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The component seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in Reference 2.

Fuel assembly lateral structural damping obtained experimentally is presented in Reference 2 (Figure B-4). The data indicates that no damping values less than 10 percent were obtained for fuel assembly displacements greater than 0.11 inch for the SSE.

The distribution of fuel assembly amplitudes decreases as one approaches the center of the core. The average amplitude for the minimum displacement fuel assembly is well above 0.11 inch for the SSE.

Fuel assembly displacement time-history for the SSE seismic input is illustrated in Reference 2 (Figure 2-3).

The CRDMs are seismically analyzed to confirm that system stresses under the combined loading conditions, as described in Section 3.9(N).1, do not exceed allowable levels, as defined by the ASME Code, Section III for "Upset" and "Faulted" conditions. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation, and the resultant seismic bending moments along the length of the CRDM are calculated. The corresponding stresses are then combined with the stresses from the other loadings required, and the combination is shown to meet ASME Code, Section III requirements.

3.7(N).3.15 Analysis Procedure for Damping

Analysis procedures for damping for subsystems in Westinghouse's scope of responsibility and for the IHA and replacement SGs are given in Section 3.7(N).2.15.

3.7(N).4 SEISMIC INSTRUMENTATION

Refer to Section 3.7(B).4.

3.7(N).5 REFERENCES

1. “Damping Values of Nuclear Power Plant Components," WCAP-7921-AR, May, 1974.

2. Gesinski, L. T. and LeBastard, G., "Safety Analysis of the 8-Grid 17 x 17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236, Addendum 1 (Proprietary), and WCAP-8288, Addendum 1 (Non-Proprietary), March 1974.

3. Lin, C. W., "How to Lump the Masses - A Guide to the Piping Seismic Analysis," ASME Paper 74-NE-7 presented at the Pressure Vessels and Piping Conference, Miami, Florida, June, 1974.

4. Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill, New York, 1964.

5. Thomas, T. H., et al., "Nuclear Reactors and Earthquakes," TID-7024, U. S. Atomic Energy Commission, Washington, D. C., August, 1963.

6. Chan, S. P., Cox, H. L., and Benfield, W. A., "Transient Analysis of Forced Vibration of Complex Structural-Mechanical Systems," J. Royal Aeronautical Society, July, 1962.

TABLE 3.7(N)-1 DAMPING VALUES USED FOR SEISMIC SYSTEMS ANALYSIS FOR WESTINGHOUSE SUPPLIED EQUIPMENT, REPLACEMENT SGS, AND IHA

Item	Damping (Percent of Critical) Upset Conditions (OBE)	Faulted Condition (SSE, DBA)
Primary coolant loop system components and large piping* **	2	4
Small piping**	1	2
Welded steel structures	2	4
Bolted and/or riveted steel structures	4	7
Integrated Head Assembly (IHA)	4.500***	6.25***

* Applicable to 12-inch or larger diameter piping.

** Code Case N-411-1, Alternate Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1, may also be applied subject to the conditions imposed by the NRC staff in Regulatory Guide 1.84.

*** Conservative damping values for the IHA are based on the recommendations in Regulatory Guide 1.61 Revision 1, Tables 1 and 2, using a weighted average for “Welded Steel or Bolted Steel with Friction Connections” and “Bolted Steel with Bearing Connections,” as approved by the NRC via Operating License Amendment 207 for Callaway.

3.8 DESIGN OF CATEGORY I STRUCTURES

This section provides information on the containment structure and its internal structures, other standard plant seismic Category I structures, and their foundations and supports.

3.8.1 CONCRETE CONTAINMENT

The containment structure is designed to house the reactor coolant system and is referred to as the reactor building in the following sections. The reactor building is part of the containment system designed to control the release of airborne radioactivity following postulated design basis accidents (DBAs) and to provide shielding for the reactor core and the reactor coolant system.

This section describes the structural design considerations for the reactor building. Section 6.2 describes the functional design of the containment to minimize leakage following a LOCA. Bechtel Topical Report BC-TOP-5-A provides additional structural information relative to the design, construction, testing, and surveillance of the prestressed concrete reactor building.

3.8.1.1 Description of the Reactor Building

3.8.1.1.1 General

The reactor building consists of a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome and a conventionally reinforced concrete base slab with a central cavity and instrumentation tunnel to house the reactor vessel. A continuous peripheral tendon access gallery below the base slab is provided for the installation and inspection of the vertical post-tensioning system. Figures 3.8-1 through 3.8-7 illustrate this configuration and also show the relationship between the shell and its interior compartment walls and floors. The internal structures are isolated from the shell by means of an isolation gap to minimize interaction. In addition, the connections used to provide for vertical support of the structural steel floor framing at the shell allow for independent horizontal movement. Figure 1.2-1 shows the relationship between the reactor building and the surrounding structures. As shown, the shell is separated from its surrounding structures by a minimum 3-inch isolation gap to avoid interaction. In some instances, the gap is filled with a fireproof compressible material.

The base slab, cylinder, and dome are reinforced by bonded reinforcing steel, as required by the design loading conditions. Additional reinforcing is provided at discontinuities in the structure and at major penetrations in the shell. The main reinforcing patterns for the base slab, cylinder wall, and dome are illustrated in Figures 3.8-8 through 3.8-14.

The interior of the reactor building is lined with carbon steel plates welded together to form a barrier which is essentially leak tight. A post-tensioning system is used to prestress the cylindrical shell and dome.

Principal nominal dimensions of the reactor building are as follows:

Interior diameter	140 ft
Interior height	205 ft
Height to spring line	135 ft
Base slab thickness	10 ft
Cylinder wall thickness	4 ft
Dome thickness	3 ft
Liner plate thickness	0.25 in.
Internal free volume	2.5×10^6 cubic ft

3.8.1.1.2 Post-Tensioning System

The tendon system employed to post-tension the cylindrical shell and dome of the reactor building is shown in Figure 3.8-15. The system uses unbonded tendons, each consisting of approximately 170 one-quarter-inch-diameter high strength steel wires and anchorage components consisting of stressing washers. The prestressing load is transferred by cold-formed button heads on the ends of the individual wires, through stressing washers, to the steel bearing plates embedded in the structure. The ultimate strength of each tendon is approximately 1,000 tons.

The unbonded tendons are installed in tendon ducts (sheathing) and tensioned in a predetermined sequence. The ducts, which form voids through the concrete between the anchorage points, consist of galvanized, spiral-wrapped, semirigid corrugated steel tubing. They are designed to retain their shape and resist the construction loads. The inside diameter of the ducts is sufficiently large to permit the installation of the tendons with minimum difficulty. Trumpets, which are enlarged ducts attached to the bearing plates, allow the wires to spread out at the anchorage to suit washer hole spacing and facilitate field cold formed button heading of the ends of the wires.

The tendon duct provides an enclosed space surrounding each tendon. After stressing, a petroleum-based corrosion inhibitor is pumped into the duct.

The vertical tendons consist of 86 inverted U-shaped tendons, which extend through the full height of the cylindrical wall over the dome and are anchored at the bottom of the base slab. The cylinder circumferential (hoop) tendons consist of 135 tendons anchored at three buttresses equally spaced around the outside of the reactor building. Each tendon is anchored at buttresses located 240 degrees apart. Three adjacent tendons, anchored at alternate buttresses, result in two complete hoop tendons. Refer to Figures 3.8-16 through 3.8-18 for tendon and buttress arrangement.

Prestressing of the hemispherical dome is achieved by a two-way pattern of the inverted U-shaped tendons and 30 hoop tendons, which start at the springline and continue up to an approximate 45-degree vertical angle from the springline. Figure 3.8-16 illustrates the arrangement of the tendons in the dome.

3.8.1.1.3 Liner Plate System

A carbon steel liner plate covers the entire inside surface of the reactor building (excluding penetrations). The liner is 1/4-inch thick but is thickened locally around the penetrations, large brackets, and major attachments. The liner plate, including the thickened plate, is anchored to the concrete structure. The vertical and dome liner plates are also used as forms for concrete placement. Typical details of the liner plate system are shown in Figures 3.8-19 through 3.8-22. In addition to the carbon steel liner plate, the containment normal sumps have an additional ¼" stainless steel liner plate installed over the top of the carbon steel liner plate for corrosion protection.

Refer to Section 3.8.2.1 for a description of the penetrations, including the equipment and personnel access hatches, piping penetration sleeves, electrical penetration sleeves, fuel transfer tube penetration sleeve, and purge line penetration sleeves.

Attachments to the liner plate which transfer loads through the liner plate to the base slab include equipment support anchors and reinforcing steel for the support of the internal structures. Refer to Figures 3.8-23 through 3.8-25 for typical details.

Major structural attachments to the wall which penetrate the liner plate include polar crane brackets, floor beam brackets, and pipe support brackets. Refer to Figures 3.8-26 and 3.8-27 for typical details.

Major structural attachments to the dome include various pipe support brackets. Refer to Figure 3.8-28 for typical details.

Miscellaneous thickened plates, which form a part of the liner plate, are provided and anchored in the concrete to provide supports. Leak chase channels and angles are also attached at seam welds where the welds are inaccessible to nondestructive examination after construction. Refer to Figure 3.8-29 for typical details for these items.

3.8.1.1.4 Shell Discontinuities

The significant discontinuities in the shell structure are at the wall-to-base-slab connection, the buttresses, and the large penetration openings.

The shell wall interface at the base slab incorporated a straight wall-to-slab joint. Refer to Figure 3.8-10 for details of the lower wall configuration.

Buttresses project out from the exterior surface of the shell wall and dome to provide adequate space for the hoop tendon anchorage and tendon-stressing equipment. The

anchorage surfaces of the buttress are normal to the tangent line of the anchored hoop tendons. Details are shown in Figure 3.8-30.

The concrete shell around the equipment hatch opening is thickened by the method shown in Figures 3.8-31 and 3.8-32.

3.8.1.1.5 Special Reinforcing Requirements

Special reinforcing is required in such areas as the major penetrations. Refer to Figures 3.8-31 through 3.8-35 for typical details in these areas.

3.8.1.2 Applicable Codes, Standards, and Specifications

The following codes, regulations, standards, and specifications are utilized in the reactor building design.

3.8.1.2.1 Regulations

a. 10 CFR 50, "Licensing of Production and Utilization Facilities"

b. 10 CFR 100, "Reactor Site Criteria"

3.8.1.2.2 Codes

a. American Concrete Institute, Building Code Requirements for Reinforced Concrete (ACI-318-71)

b. American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th Edition, adopted February 12, 1969, and Supplement Numbers 1, 2, and 3 (See FSAR Table 3.2-1, Note 19)

c. ASME Boiler and Pressure Vessel Code - 1974 Edition or later

   Section II - Material Specifications

   Section III, Division 1 - Nuclear Power Plant Components

   Section V - Nondestructive Examination

   Section VIII - Pressure Vessels

   Section IX - Welding and Brazing Qualifications

d. American Welding Society, Structural Welding Code (AWS D1.1-75) (See FSAR Table 3.2-1, Note 19)

e. Acceptable ASME Code cases per Regulatory Guides 1.84 and 1.85, as addressed in Appendix 3A

3.8.1.2.3 Standards and Specifications

Industry standards, such as those published by the American Society for Testing and Materials (ASTM) and the American Association of State Highway and Transportation Officials (AASHTO), are used whenever possible to describe material properties, testing procedures, fabrication, and construction methods. The applicable standards used are listed in Section 3.8.1.6.

Structural specifications are prepared to cover the areas related to the design of the reactor building. These specifications are prepared specifically for the SNUPPS project. These specifications emphasize the important points of the industry standards for the reactor building and reduce the options that would otherwise be permitted by the industry standards. These specifications cover the following areas:

a. Concrete material properties

b. Mixing, placing, and curing of concrete

c. Reinforcing steel and splices

d. Post-tensioning system

e. Liner plate system

3.8.1.2.4 Design Criteria

The following design criteria form the basis for the reactor building design. Specifically, the criteria contained in Appendix C of the Bechtel Power Corporation Topical Report BC-TOP-5-A are used in the design of the reactor building. Appendix C of BC-TOP-5-A presents a detailed description of compliance with Article CC-3000 of the proposed ASME Code, Section III, Division 2.

a. 10 CFR 50, Appendix A - GDC for Nuclear Power Plants (Compliance is discussed in Section 3.1) GDC Numbers 2, 4, 16, and 50

b. Bechtel Power Corporation topical reports, as referenced in Section 1.6

3.8.1.2.5 NRC Regulatory Guides

NRC Regulatory Guides 1.10, 1.15, 1.18, 1.35, 1.55, 1.84, 1.85, 1.94, and 1.103 are applicable to the design and construction of the reactor building. Specific editions and the extent of compliance with these guides are discussed in Appendix 3A.

3.8.1.3 Loads and Loading Combinations

The applicable loads and loading combinations used in the design and analysis of the reactor building structure, components, and localized areas are those listed in BC-TOP-5-A, Appendix C.

DESIGN ACCIDENT PRESSURE LOAD - Transients resulting from the DBA and other lesser accidents are presented in Section 6.2.1 and serve as the basis for the reactor building design pressure of 60 psig.

PRESTRESSING FORCES - The prestressing forces are related to the design pressure by selection of a level of prestress, as discussed in Section 6.2.1 of BC-TOP-5-A.

THERMAL LOADS - The temperature gradients through the reactor building wall are shown in Figure 3.8-36 for the operating condition and for the postulated DBA condition.

WIND AND TORNADO LOADS - The wind and tornado loads are in accordance with Section 3.3.

EARTHQUAKE LOADS - Earthquake loads are in accordance with Section 3.7.

HYDROSTATIC LOADS - Hydrostatic loads are in accordance with Section 3.4.

EXTERNAL PRESSURE LOAD - External pressure loading with a differential of 3 psig from outside to inside is considered. The external design pressure has conservatively been assumed to account for barometric pressure differentials after the reactor building is sealed. The reactor building is designed to be cooled below 50°F from the operating temperature of 120°F. The inadvertent actuation of the containment spray headers, which induce an external pressure load, is discussed in Section 6.2.1.

MISSILE AND POSTULATED PIPE RUPTURE EFFECTS - The internal and external missile and postulated pipe rupture loads are in accordance with Sections 3.5 and 3.6, respectively.

TEST PRESSURE LOAD - The structure is designed for a Structural Integrity Test pressure load of 69 psig.

POST-LOCA FLOODING - The post-LOCA flooding of the reactor building for the purpose of fuel recovery is not a design condition. Although there are no special provisions incorporated in the structural design of the reactor building or its interior structures for the purpose of fuel recovery after a LOCA, there is sufficient time following a LOCA for the plant operators and/or consultants to assess the extent of the damage to the reactor coolant system, the interior structures of the reactor building, and refueling equipment and to make the necessary provisions, including any additional equipment required, for the recovery of the fuel.

3.8.1.4 Design and Analysis Procedures

The procedures utilized in the analysis and design of the reactor building are in accordance with Sections 6.0 and 7.0 and Appendices B and C of BC-TOP-5-A.

Computer programs are relied upon to perform many of the computations required for the reactor building analysis. However, in many cases, classical methods and manual techniques are used for the analysis of localized areas of the reactor building and for preliminary proportioning. Manual calculations are generally used for (a) the initial proportioning of the dome, wall, and base slab, (b) evaluation of the effects of locally applied loads, such as pipe rupture or crane loads, (c) the preparation of input for the computer analyses, and (d) areas which do not lend themselves to computer applications. Section 7.0 of BC-TOP-5-A describes the analytical methods in more detail.

The design methods incorporate several phases, as described in Section 6.0 of BC-TOP-5-A. They involve the initial proportioning of structures, using the results of preliminary analyses documented in BC-TOP-5-A. Experience based on the completed design or parametric studies of other structures of a similar nature is used as well.

The final design phase incorporates and refines information gained in the earlier phases. It also incorporates closer approximations of the equipment and piping and related loads, based on the completion of the detailed engineering design. Improved assumptions regarding material properties, including the effects of creep, shrinkage, and the cracking of concrete, are used.

3.8.1.4.1 Overall Analysis

The reactor building is considered to be an axisymmetric structure for the overall analysis. Although there are deviations from this ideal shape, such as penetrations and buttresses, these deviations are sufficiently localized so as not to affect the overall analysis and are addressed by special local analyses.

The overall analysis of the reactor building for axisymmetric loads is performed by using the FINEL finite element computer program described in Appendix 3.8A for combinations of the individual loading cases of dead, live, thermal, pressure, and prestress loads. The entire reactor building is modeled with one finite element mesh consisting of the dome, shell, base slab, reactor cavity, and soil. The concrete structure is modeled by continuously interconnected elements. The liner plate is modeled by a layer of elements attached to the interior surfaces of the concrete structure. The finite element mesh is extended into the soil to account for the elastic nature of the foundation material and its effect on the structure. Since the SNUPPS reactor building design is used at sites with different foundation properties, the analyses are performed taking into account the range of geotechnical parameters of the foundation media at all the sites. The tendon access gallery and instrumentation tunnel are analyzed as separate structures. The finite

element model used for the analysis of the reactor building for axisymmetric loads is shown in Figures 3.8-37 through 3.8-39.

The overall analysis of the reactor building for nonaxisymmetric loads (i.e., seismic) is performed, using the SAP three-dimensional finite element computer program described in Appendix 3.8A. One-half of the reactor building is modeled, without the dome, about an axis of symmetry of the structure in plan. Appropriate boundary conditions are simulated at the top of the shell and along the axis of symmetry to provide for strain compatibility. The shell, base slab, and reactor cavity are modeled with one finite element mesh. Soil springs are provided below the structure to account for the effect of the foundation material on the structure. Since the SNUPPS reactor building design is used at sites with different foundation properties, the analyses are performed, taking into account the range of geotechnical parameters of the foundation media at all the sites. The upper portion of the shell, dome, tendon access gallery, and instrumentation tunnel are analyzed separately. The finite element model used for the analysis of the reactor building for nonaxisymmetric loads is shown in Figure 3.8-40.

3.8.1.4.2 Local Analysis

3.8.1.4.2.1 Large Penetration Openings

Large penetrations are defined as those having an inside diameter equal to or greater than 10 feet (2.5 times the reactor building nominal shell wall thickness). The equipment hatch and personnel lock fall into this category.

Local analyses of the reactor building shell in the area of large penetrations are performed, using the SAP three-dimensional finite element computer program. The analytical models consist of a one-quarter segment mesh that follows the axes of symmetry of the penetration opening. The points defining the outermost boundary of the model are located at approximately two penetration diameters beyond the edge of the opening, so that the behavior of the model at the boundaries is compatible with that of the undisturbed cylindrical shell. Boundary conditions along the axes of symmetry and the boundaries of the model are specified to provide for strain compatibility.

The SAP finite element models used for analyses of the equipment hatch and personnel lock are shown in Figures 3.8-41 through 3.8-43.

3.8.1.4.2.2 Small Penetration Openings

Small penetration openings are defined as those having an inside diameter of less than 10 feet (2.5 times the reactor building nominal shell wall thickness). The local analysis of the shell in the area of small penetration openings is discussed in Sections 6.5 and 7.4 of BC-TOP-5-A.

3.8.1.4.2.3 Buttress and Tendon Anchorage Zones

Analysis and design of tendon anchorage zones and reinforcement in buttresses are discussed in Section 6.6 of BC-TOP-5-A and in BC-TOP-7 and BC-TOP-8.

3.8.1.4.3 Creep, Shrinkage, and Cracking of Concrete

In the design of the reactor building post-tensioning system, conservative values of creep and shrinkage for the concrete are utilized, based on past experience. The values used are verified by the evaluation of the tests performed on the concrete which is used in the reactor building shell. In establishing these values, the tests are performed on concrete that is used at each of the SNUPPS sites, and consideration is given to the differences in the environment between the test samples and the actual concrete in the structure.

The moments, forces, and shears are obtained on the basis of an uncracked section for all load combinations. However, in sizing the reinforcing steel required, the concrete is not relied upon for resisting tension. Thermal moments are modified by a cracked section analysis, using analytical techniques.

3.8.1.4.4 Tangential Shear

The design and analysis procedures for tangential shear are in accordance with Appendix C of BC-TOP-5-A.

3.8.1.4.5 Variation in Physical Material Properties

In the design and analysis of the reactor building, consideration is given to the effects of possible variations in the physical properties of materials on the analytical results. The variations in physical properties are accounted for by using allowable stress levels, below ultimate strength, for design of the structure under full service and factored load conditions.

3.8.1.4.6 Steel Liner Plate and Anchors

The analysis and design procedures utilized for the liner plate system are in accordance with BC-TOP-1 and Sections 6.8, 7.5, and Appendix C of BC-TOP-5-A.

3.8.1.4.7 Computer Programs

The computer programs used in the analysis and design of the reactor building are described in Appendix 3.8A.

3.8.1.5 <u>Structural Acceptance Criteria</u>

The fundamental acceptance criterion for the completed reactor building is successful completion of the Structural Integrity Test where measured responses are required to be

within the limits predicted by analyses. The limits are based on test load combinations and code values for stress, strain, or gross deformation for the range of material properties and construction tolerances specified, as described in Section 3.8.1.6.

The limits for allowable stresses and strains are given in Appendix C of BC-TOP-5-A and are compatible with nationally recognized codes of practice. In this way, the margins of safety associated with the design and construction of the reactor building are, as a minimum, the accepted margins associated with nationally recognized codes of practice.

The Structural Integrity Test is planned to yield information on both the overall response of the reactor building and the response of localized areas. This information, together with the test information documented in BC-TOP-7 and BC-TOP-8, provides direct experimental evidence that the containment structure can withstand the design internal pressure.

The design and analysis methods, as well as the type of construction and construction materials, are chosen to allow assessment of the capability of the structure throughout its service life. Additionally, surveillance testing provides further assurances of the continuing ability of the structure to meet its design functions.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

This section contains information relating to the materials, quality control program, and special construction techniques used in the fabrication and construction of the reactor building.

3.8.1.6.1 Concrete

3.8.1.6.1.1 Materials

Cement is Type II, conforming to the Specification for Portland Cement (ASTM C150). The sum of tricalcium silicate and tricalcium aluminate does not exceed 58 percent. The cement contains no more than 0.60 percent by weight of alkalies calculated as Na_2O plus 0.658 K_2O. The limitation of the alkali content of the cement may be waived provided that the aggregates pass required laboratory tests and have no history of alkali aggregate incompatibility. Certified copies of material test reports showing the chemical composition and physical properties are obtained for each load of cement delivered.

All aggregates conform to the Specification for Concrete Aggregate (ASTM C33). For concrete with 1-1/2-inch maximum size aggregate, the coarse aggregate is a combination of 1-1/2-inch and 3/4-inch aggregate. The potential reactivity of the aggregate is established in accordance with ASTM C289. A petrographic examination of the aggregate is performed in accordance with ASTM C295. In addition to the specified gradation, the fine aggregate (sand) has a fineness modulus of not less than 2.5 nor more than 3.1. During normal concrete production, at least four of five successive test samples do not vary more than 0.20 from the average. Coarse aggregate is rejected if

the loss, when subjected to the Los Angeles Abrasion Test (ASTM C131) using Grading A, exceeds 40 percent by weight at 500 revolutions. The particle shape of the coarse aggregate is generally rounded or cubical and does not contain thin, flat, and elongated particles in excess of 15 percent by weight in any nominal size group. A thin, flat, and elongated particle is defined as a particle having a maximum dimension in excess of four times the minimum dimension.

Water and ice used in mixing concrete are free of injurious amounts of oil, acid, alkali, organic matter, and other deleterious substances and are tested in accordance with AASHTO T-26. When tested according to AASHTO T-26, the water does not cause unsoundness in the autoclave test, and does not change the final setting time by more than 1 hour, and the 7- and 28-day compressive strength of ASTM C109 cubes is not reduced by more than 10 percent when compared with results obtained with distilled water. Water is tested for pH, chlorides, and sulfates and does not contain more than 250 ppm of chlorides as Cl, nor more than 1,000 ppm of sulfates as SO_4.

The concrete also contains an air-entraining admixture and a water-reducing admixture. The air-entraining admixture is in accordance with the Specification for Air Entraining Admixtures for Concrete (ASTM C260). It is capable of entraining 3 to 6 percent air, is completely water soluble, and is completely dissolved when it enters the batch. The water reducing and retarding admixture conforms to the Specification for Chemical Admixtures for Concrete (ASTM C494), Types A and D. Type A is used when concrete temperature is below 70°F. Type D is used when concrete temperature is 70°F and above, except for floor slabs where its use is optional. Pozzolans, if used, conform to the Specification for Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete (ASTM C618).

3.8.1.6.1.2 Concrete Mix Design

Structural concrete used in the construction of the reactor building shell and dome has a compressive strength, fc', of 6,000 psi at 90 days. Structural concrete used in the construction of the reactor building base slab, reactor cavity, instrumentation tunnel, and tendon access gallery has a compressive strength, fc', of 5,000 psi at 90 days.

Structural specifications are prepared specifically for the SNUPPS project to identify the required concrete material properties and tests. Concrete conforms to the Specification for Ready-Mixed Concrete (ASTM C94), as modified herein. In lieu of the maximum water content specified in ASTM C94, the concrete is mixed so as to be placed at the specified slumps. The mix proportions are established in accordance with Paragraph 3.8 of ACI 301, Method 1. The required average strength is in accordance with Paragraph 3.8.2.3 of ACI 301. In lieu of the requirements in Paragraph 18.2 of ASTM C94, conformance to ASTM E329, with the exception of Paragraph 4 as it pertains to concrete, is required.

3.8.1.6.1.3 Examination

During construction, concrete materials are regularly sampled and tested to ensure quality control. Table 3.8-1 shows the procedures used and the frequency of testing for the concrete materials used.

3.8.1.6.1.4 Placement

Conveying and placement of concrete are performed in accordance with the following codes and standards to the extent described:

a. ACI 301 - Specifications for Structural Concrete for Buildings, Chapters 4, 6, 8, 9, 10, 11, 12, 13, 14, and 15 are used, except as noted below:

    1. In lieu of the requirements for the removal of forms specified in Paragraph 4.5.4, the following applies:

       Forms for columns, walls, sides of beams, slabs, girders, and other parts not supporting the weight of the concrete are removed as soon as practicable in order to avoid delay in curing and repairing surface imperfections. Wood forms or insulated steel forms for members over 3 feet in thickness are stripped within 24 hours or kept in place for a minimum of 7 days. If forms are stripped within 24 hours, the surfaces are cured by moist curing or membrane curing as specified in ACI 301, Chapter 12.

    2. In lieu of the requirements for the placing of mass concrete specified in Paragraph 14.4.1, the following applies:

       Slump is specified for particular locations and degree of congestion rather than holding a 2-inch maximum. An inadvertency margin for maximum slump above the stated maximum average value is included in the job standards.

    3. In lieu of the requirements for curing and protection of mass concrete specified in Paragraph 14.5.1, the following applies:

       The minimum curing period is 7 days for heavily reinforced massive sections.

b. ACI 304, Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete, Chapters 5 and 6, are used without exception.

c. ACI 305, Recommended Practice for Hot-Weather Concreting, is used without exception.

d. ACI 306, Recommended Practice for Cold-Weather Concreting, is used without exception.

e. ACI 318, Building Code Requirements for Reinforced Concrete, Chapters 5 and 6 are used, except as noted below:

In place of Paragraph 6.3.2.4, the specific provisions of the applicable codes that govern the system of which the embedded piping is a part shall apply. Examples of such applicable codes are: for nuclear piping, ASME Boiler and Pressure Vessel Code, Section III; and for nonnuclear piping, ANSI B31.1, Power Piping.

The purpose of ACI 318, Paragraph 6.3.2.4, is to avoid the removal of concrete if a leak is developed in the pipe wall or joints. The testing requirements of Paragraph 6.3.2.4 are valid in the case of noncode piping; they are not valid for the piping that is required to conform to the acceptable industry codes such as the ASME B&PV Code for nuclear piping, ANSI B31.1 for nonnuclear power piping, and the applicable state or local plumbing codes. Where no such codes or code cases govern a particular pipe embedded in structural concrete, the requirements of ACI 318, Paragraph 6.3.2.4, are implemented.

f. ACI 347, Recommended Practice for Concrete Formwork, is used without exception.

g. ACI SP2, Manual of Concrete Inspection, applicable provisions relating to conveying and placement are used without exception.

h. ASTM C94, Specification for Ready-Mixed Concrete, applicable provisions relating to conveying and placement are used without exception.

The placement of concrete complies with the requirements of Regulatory Guide 1.55 to the extent described in Appendix 3A. No aluminum pipe or other conveying equipment containing aluminum that will be in contact with fresh concrete is used for conveying concrete to the point of placement.

3.8.1.6.2 Reinforcing Steel and Splices

3.8.1.6.2.1 Materials

Reinforcing bars for concrete are deformed bars meeting the requirements of the Specification for Deformed and Plain Billet-Steel Bars for Concrete Reinforcement (ASTM A 615), Grade 60. For each heat or mill shipment, whichever is less, certified copies of the material test reports covering the chemical and mechanical properties of the reinforcing bars are obtained.

Mechanical splices, when used, consist of T-series and B-series Cadweld-type splices. Tubing used for splice sleeves conforms to the Specification for Seamless Carbon and Alloy Mechanical Tubing (ASTM A519), Grades 1018 or 1026. Certified copies of material test reports showing the results of the chemical and mechanical tests of material from each lot of splice sleeves are obtained. In addition, certification is obtained for each lot showing for each lot that the chemical composition of the powdered metal and the chemical and mechanical properties of the resulting filler material conform to the manufacturer's standards.

3.8.1.6.2.2 Examination

During fabrication and construction, reinforcing steel and mechanical splices are regularly sampled and tested to ensure quality control. The examination methods, frequency, and acceptance standards in Regulatory Guide 1.10 for mechanical splices and Regulatory Guide 1.15 for reinforcing steel are used. Refer to Appendix 3A for a description of the extent of compliance with these regulatory guides.

3.8.1.6.2.3 Erection Tolerances

The reinforcing steel is placed in accordance with the tolerances specified in Paragraph 7.3.2 of ACI 318, except as noted below:

a. Inplace reinforcing steel cover tolerances for the containment shall be within the following limits:

Base slab	-0", +1 1/2"
Exterior walls	-0", +1 1/2"
Dome	-1", +1"

Exterior wall tolerances are maintained, except for local areas adjacent to some recesses on the exterior surface of the containment shell where a gradual sweep of the continuous reinforcing steel to clear the recesses could result in these cover tolerances being exceeded.

However, the resulting cover is within the design allowable specified in BC-TOP-5-A, Appendix C, except for the two electrical penetration banks. The electrical penetration banks, centered at azimuth 222°-30', El. 2035'-3" and azimuth 319°-30', El. 2035'-3" have the outside face of concrete recessed 8 inches. The two banks are approximately 15 feet (vertical) by 48 feet (horizontal) and 15 feet by 39 feet, respectively. The transition zone where the continuous reinforcing sweeps gradually inward to clear the recess extends as much as 16 feet-8 inches away from the outside edge of the recess. Although the reinforcing steel in this area is generally within

the limits indicated above, there are a few instances where, including placing tolerances, the cover can be as much as 13-3/4 inches.

b. Cadwelds and other connectors are not considered as reinforcing steel.

c. In no case is the cover reduced by more than one-third of the minimum specified design cover.

d. Minimum splice lengths and minimum embedment lengths are maintained to a tolerance of minus 2 inches. These minimum lengths may be exceeded without limit, provided that the other requirements for cover and clearances are not violated.

e. The variation in spacing is $\pm$ 2 bar diameters, except that the minimum clear distance specified in Paragraphs 3.3.2 and 7.4 of ACI 318 is maintained. The total number of bars in any nominal 10-foot segment is maintained.

f. For longitudinal location of bends and ends of bars that are mechanically spliced, a tolerance of minus 2 inches at the discontinuous end of the member in which the splice occurs is acceptable. Conversely, the cover for this situation may be increased by 2 inches.

3.8.1.6.3 Prestressing System

3.8.1.6.3.1 Materials

The prestressing system consists of load carrying and nonload carrying components. The load carrying components include the prestressing wires which make up the tendons, and anchorage components composed of bearing plates, anchor heads, and shims. Nonload carrying components include the tendon sheathing (including trumpet assemblies, couplers, vent and drain nipples, and other appurtenances), and corrosion prevention material.

The prestressing wire is cold-drawn, of the intermediate relaxation or stabilized type, and conforms to the Specification for Uncoated Stress-Relieved Wire for Prestressed Concrete (ASTM A421), Type BA. The materials used for the anchorage components are compatible with the tendon system.

Tendon sheathing consists of galvanized, spiral-wrapped, semirigid, corrugated tubing conforming to the requirements of the Specification for Steel Sheet, Zinc-Coated (Galvanized) by the Hot-Dip Process, Lock Forming Quality (ASTM A527) or the Specification for Steel Sheet, Zinc-Coated (Galvanized) by the Hot-Dip Process, Drawing Quality (ASTM A528), 22-gauge cold rolled carbon steel. Trumpet material conforms to the Specification for Electric Resistance-Welded Carbon and Alloy Steel Mechanical Tubing (ASTM A513), Grades MT1010 to 1029, or the Specification for Welded and

Seamless Steel Pipe (ASTM A53), Grade B. Couplers and mending sections conform to ASTM A527 or ASTM A528. Vent and drain nipples consist of noncorrosive metal galvanized pipe, or equal.

After fabrication, a thin film of temporary corrosion protection material is applied to the prestressing steel. This material is compatible with the permanent corrosion prevention material and is removable with the use of a nonchlorinated petroleum solvent to permit the installation of attached anchorages.

The permanent corrosion-prevention coating applied to tendons is a petrolatum or microcrystalline wax-base material, containing additives to enhance the corrosion-inhibiting and wetting properties, as well as to form a bond with the tendon steel. The coating has the following properties for the lifetime of the structure and for the anticipated range of the temperature:

a. Freedom from cracking and brittleness

b. Continuous self-healing film over the coated surfaces

c. Chemical and physical stability

d. Nonreactivity with the surrounding and adjacent materials, such as concrete, tendons, and ducts

e. Moisture displacing characteristic

Each batch of coatings is analyzed for the presence of water soluble chlorides, nitrates, and sulphides.

3.8.1.6.3.2 Examination

Prior to construction, a number of tests are performed on the load-carrying components of the prestressing system to ensure that the performance requirements of the system are satisfied and quality control is maintained. In addition to the tests described below, an in-service surveillance program of the prestressing system is carried out, as discussed in Section 3.8.1.7.

All load-carrying components are subject to tensile tests. Materials produced to an ASTM specification are sampled and tested as required by that specification. Materials not produced to an ASTM specification are sampled and tested at the rate of one test for every 20 tons, or fraction thereof, produced from each heat of steel. The tensile strength, yield strength, elongation, and other pertinent data are reported on the Certified Materials Test Report.

The stress-relaxation properties of the wire, determined in accordance with the Recommended Practice for Stress-Relaxation Tests for Materials and Structures

(ASTM E328), are obtained from the manufacturer for a minimum of three relaxation tests of 1,000 hours duration. In addition to those required by ASTM E328, the manufacturer's reports of the test include detailed test method, initial stress, final stress, test time, temperature limits, and mathematical tools used to interpret the test results.

Anchorage components are subjected to hardness tests. For anchorhead assemblies, the Method of Tests for Rockwell Hardness and Rockwell Superficial Hardness of Metallic Materials (ASTM E18) and the Method of Test for Brinell Hardness of Metallic Materials (ASTM E10) are conducted on 10 percent of the parts from each lot (after heat treatment) on a random basis. If the hardness requirement is not met by any single part relative to acceptance standards set by design documents, then all parts from the lot are tested. Only those parts meeting the requirements are used.

The following tests are performed by the tendon manufacturer in order to qualify his system for use in the reactor building:

a. A static tensile test is conducted to destruction to obtain information on yield strength, tensile strength, and compliance with the following performance requirements:

   A full-capacity tendon complete with anchorages will develop an ultimate strength equal to 100 percent of the minimum specified ultimate tensile strength of the prestressing steel, without exceeding the anticipated set of the anchorage elements.

   The total elongation under ultimate load of the tendon will not be less than 2 percent, measured in a minimum gauge length of 100 inches.

b. A high-cycle dynamic tensile test is conducted to ensure that the tendon can withstand, without failure, 500,000 cycles of stress variation from 60 to 66 percent of the tendon minimum specified ultimate tensile strength. A load cycle is defined as an increase from the lower load to the higher load and return. This test is performed on specimens having at least 10 percent of the full-sized prestressing steel area of one production tendon.

c. A low-cycle dynamic tensile test is conducted to ensure that the tendon can withstand, without failure, 50 cycles of stress variation from 40 to 80 percent of the tendon minimum specified ultimate tensile strength. This test is performed on specimens having at least 10 percent of the full-sized prestressing steel area of a production tendon.

3.8.1.6.3.3 Erection Tolerances

The following are the erection tolerances from the theoretical location of the sheathing in the cylindrical wall:

a. Vertical sheathing

±2 inches in the circumferential direction

±½ inch in the radial direction when measured from the liner plate or ±1½ inches when measured from the reactor building theoretical centerline

±6 inches in elevation for points of tangency between the curved and straight sections

±2 inches per 10 feet - 0 inches for variation from the plumb, not cumulative

b. Horizontal sheathing

±2 inches in elevation

±½ inch in the radial direction when measured from the liner plate or ±1½ inches when measured from the reactor building theoretical centerline

±6 inches in the circumferential direction for points of tangency between the curved and straight sections

c. Requirements at penetrations:

The general criterion for placing sheathing in the area of penetrations is to achieve a smooth configuration without sharp bends which would impair the insertion of the tendons or create undesirable loading combinations. The sheathing is also placed to meet the clear distance between any point on the sheathing and a penetration nozzle as well as the minimum distance between sheathing as given on the tendon placement drawings.

The following are the erection tolerances from the theoretical location of the sheathing in the dome:

a. Meridional sheathing

±2 inches in the circumferential direction

±½ inch in the radial direction when measured from the liner plate or ±1.5 inches when measured from the reactor building theoretical centerline

±6 inches in the meridional direction for points of tangency between the curved and straight sections

b. Horizontal sheathing

±2 inches in the meridional direction

±½ inch in the radial direction when measured from the liner plate or ±1.5 inches when measured from the reactor building theoretical centerline

±6 inches in the circumferential direction for points of tangency between the curved and straight sections

3.8.1.6.4 Liner Plate System

The reactor building is lined with welded steel plates, as outlined below, to ensure low leakage. These materials have been chosen on the basis that they have sufficient strength and ductility to resist the expected strains from design criteria loading and, at the same time, preserve the required leaktightness of the reactor building. They are readily weldable by all commercially available arc and gas welding processes.

3.8.1.6.4.1 Materials

The ¼-inch-thick liner plate material conforms to the requirements of the Specification for Low and Intermediate Tensile Strength Carbon Steel Plates for Pressure Vessels (ASME SA 285), Grade A. Thickened liner plates, ranging from ½-inch to 2 inches in thickness, are used at penetrations, brackets, and embedded assemblies and conform to the requirements of the Specification for Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service (ASME SA516), Grade 70. In the event that significant loads are to be transmitted through the thickness dimension of the liner, nondestructive tests are performed to determine the capability of the liner materials used in these locations.

Materials for the containment normal sumps conform to the requirements of the specification for Austenitic Stainless Steel (ASTM A240, Type 304 or approved equivalent). Materials for penetration sleeves conform to the requirements of the following specifications and are impact tested in accordance with Paragraph NE-2300 of Section III of the ASME Code at a temperature no greater than 0°F:

a. Seamless penetration sleeves conform to the Specification for Seamless and Welded Steel Pipe for Low-Temperature Service (ASME SA333), Grade 6.

b. Welded penetration sleeves conform to the Specification for Electric-Fusion Welded Steel Pipe for High Pressure Service (ASME SA155), KCF70, or pipe in accordance with ASME Code Class MC Vessels.

c. Penetration sleeve reinforcing plates conform to the Specification for Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service (ASME SA516), Grade 70.

d. Penetration rods conform to the Specification for Carbon Steel Forgings for Piping Components (ASME SA105).

Materials used for the liner plate anchors and embedments conform to the Specification for Structural Steel (ASTM A36) or the Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service (ASTM A516), Grade 70.

Materials used for test piping, fittings, plates, and shapes conform to the following:

a. Specification for Welded and Seamless Steel Pipe (ASTM A53)

b. Specification for Forgings, Carbon Steel, for Piping Components (ASTM A105)

c. Specification for Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for General Service (ASTM A181)

d. Specification for Piping Fittings for Wrought Carbon Steel and Alloy Steel for Moderate and Elevated Temperatures (ASTM A234)

e. Specification for Low and Intermediate Tensile Strength Carbon Steel Plates for Pressure Vessels (ASME SA285)

f. Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service (ASTM A516), Grade 70

g. Specification for Structural Steel (ASTM A36)

h. Specification for Low and Intermediate Tensile Strength Carbon Steel Plates of Structural Quality (ASTM A283)

Materials used for Cadweld sleeves conform to the Specification for Seamless Carbon and Alloy Mechanical Tubing (ASTM A519), Grades 1018 or 1026.

Materials used for pipe anchors conform to the Specification for Seamless Carbon Steel Pipe for High Temperature Service (ASTM A106), Grade B.

Welding electrode materials are selected on the basis of the welding process used and the type of materials to be joined and in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III. Written control procedures for welding materials are required, which define the measures used to control the use of the materials throughout all welding operations. Such controls provide for the complete traceability of welding materials used in the liner plate seams to all tests and examinations and to the welder.

Materials for machine bolts conform to the Specification for Carbon Steel Externally and Internally Threaded Standard Fasteners (ASTM A307). Materials for high-strength bolts conform to the Specification for High Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers (ASTM A325).

Materials for weld studs conform to the Specification for Steel Bars, Carbon, Cold-Finished, Standard Quality (ASTM A108), Grades 1010, 1015, 1016, 1017, 1018, or 1020.

Materials used for weld backing strips are compatible with the materials being welded.

Where ASTM specifications are referenced, equivalent ASME materials may be used.

Certificates of Compliance are obtained from the manufacturer for bolts, weld studs, weld backing strips, and welding fluxes. Certified copies of material test reports are obtained for all other liner plate system materials which include the actual results of all required chemical analyses, physical tests, mechanical tests, and examinations.

3.8.1.6.4.2 Examination

Nondestructive examination of the liner plate welds complies with Regulatory Guide 1.94 to the extent described in the OQAM.

3.8.1.6.4.3 Erection Tolerances

The liner plate and penetration assemblies are erected to the following tolerances requirements:

a. General Liner Plate

1. The radial location at any point on the liner plate shall not vary from the design radius by more than ±3 inches. Measurements shall be made at 30-degree spacings for each 10 feet of rise.

   The radius of the hemispherical dome for all elevations between the as-built springline and 15 feet above it shall be within ±3 inches of the design radius.

   The radius of the hemispherical dome for all points above a plane parallel to and 15 feet higher than the plane of the as-built springline shall not exceed the design radius plus 8 inches or be less than the design radius minus 12 inches.

2. Plates to be joined by butt welding shall be matched and retained in position during the welding. Misalignment in completed joints shall not exceed the limits shown in Table 3.8-2.

3. A 15-foot-long template curved to the required radius shall not show deviations of more than 1 inch when placed against the completed surface of the shell within a single plate section and not closer than 12 inches at any point to a welded seam. When the template is placed across one or more welded seams, the deviation shall not exceed 1½ inches. The effect of change in plate thickness or of weld reinforcement shall be disregarded when determining deviations.

4. A 15-inch-long template curved to the required radius shall not show deviations of more than 1/8 inch inward or 3/8 inch outward when placed against the completed surface of the shell within a single plate section and not closer than 12 inches to a weld seam.

   A 30-inch-long template, curved to the required radius, shall not show deviations of more than ¼ inch when placed against the completed surface of the shell within a single plate section.

5. The deviation from the true vertical for any 10-foot plate shall not vary by more than ¾ inch. Plates of other depths shall be checked for linearly varying tolerances. The overall out-of-plumbness of the shell shall not exceed 3 inches.

6. A 10-foot straightedge held vertically shall not show deviations greater than ±¾ inch in the horizontal direction between seam welds.

7. Local bends that deviate from the design radius or a vertical straightedge by an offset of more than ½ inch in 1 foot shall not be accepted. The template used to measure the local deviations shall be only 1 to 2 feet longer than the area of the deviation itself.

b. Penetration Assemblies

1. Items 1, 3, 5, and 7 in part "a" above also control the tolerance requirements for penetration assemblies.

2. Alignment of the axes of penetrations, as erected, shall not vary from the alignment shown on the design drawings by more than 2 degrees for pipes 12 inches in diameter or less and by more than one degree for pipes over 12 inches in diameter. Individual penetrations and penetration assemblies shall be located within ±1 inch of their design elevations and circumferential locations, at the cylindrical shell.

3.8.1.6.5 Quality Control

In addition to the quality control measures discussed in Sections 3.8.1.6.1, 3.8.1.6.2, 3.8.1.6.3, and 3.8.1.6.4, the construction quality control program is discussed in Chapter 17.0.

3.8.1.6.6 Special Construction Techniques

The reactor building is constructed of concrete and steel, using proven methods common to heavy industrial construction. No special, new, or unique construction techniques are used.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Structural Integrity Test

Following construction, the reactor building was proof-tested at 115 percent of the design pressure. During this test, deflection measurements and concrete crack inspections were made to determine that the actual structural response is within the limits predicted by the design analyses.

The test procedure complied with the requirements of Regulatory Guide 1.18 to the extent described in Appendix 3A. The associated leak rate test procedure is described in Section 6.2.6. Section 9.0 of BC-TOP-5-A also describes test results obtained using a typical procedure as well as those obtained from early tests where a substantial amount of strain information was collected.

3.8.1.7.2 Long-Term Surveillance

The long-term surveillance program consists of evaluating the general conditions of the post-tensioning system. Data on wire corrosion levels and tendon lift-off forces are obtained and analyzed. The surveillance tendons are designated by the engineer as part of the surveillance program which conforms with Subsection IWL of Section XI, Division 1 of the ASME Boiler and Pressure Vessel Code as limited and modified by 10 CFR 50.55a.

3.8.2 CONTAINMENT SYSTEM STEEL ITEMS

This section describes the major penetrations and portions of penetrations intended to resist pressure which are not backed by structural concrete.

3.8.2.1 Description of Steel Items

The steel items that are part of the containment pressure boundary include access openings, such as the equipment hatch and personnel hatches, piping penetration

sleeves, fuel transfer tube penetration sleeves, electrical penetration sleeves, and the purge line penetration sleeves.

3.8.2.1.1 Equipment and Personnel Access Hatches and Penetration Sleeves

The equipment hatch, shown in Figure 3.8-44, is a welded steel assembly with a double-gasketed, flanged, and bolted cover. Provision is made for leak testing of the flangegasket combination by pressurizing the space between the gaskets.

One personnel hatch and one auxiliary hatch, both of which are welded steel assemblies, are provided as shown in Figures 3.8-45 and 3.8-46. Each hatch has two doors with double gaskets in series. In order to assure leaktightness, provision is made to pressurize the space between the gaskets. The doors are mechanically interlocked to ensure that one door cannot be opened unless the second door is sealed. Provisions are made for deliberately overriding the interlock by the use of special tools and procedures. Each door is equipped with quick-acting valves for equalizing the pressure across the doors. The doors are not operable unless the pressure is equalized. Pressure equalization is possible from every point at which the associated door can be operated. The valves for the two doors are properly interlocked so that only one valve can be opened at one time and only when the opposite door is closed and sealed. Each door is designed so that, with the other door open, it will withstand and seal against design and testing pressure of the containment vessel. There is visual indication outside each door showing whether the opposite door is open or closed. Provision is made outside each door for remotely closing and latching the opposite door so that in the event that one door is accidentally left open it can be closed by remote control. The access hatch barrels have nozzles which permit pressure testing of the hatch at any time. The hatches are protected from tornado missiles by enclosure structures or shields. A moveable missile shield is provided on the outside of the reactor building to protect the equipment hatch. The personnel hatch is enclosed within the auxiliary building. The auxiliary hatch is enclosed within an exterior tornado-resistant concrete structure.

The personnel and auxiliary access hatch barrels are designated as ASME Section III, Class MC components.

The hatch penetration sleeves project into the reactor building and are used to support the hatches. These items are made from carbon steels and conform to the requirements of ASME Section III, Subsection NE.

3.8.2.1.2 Piping Penetration Sleeves

Piping penetrations are divided into three general groups:

a. Type 1: Flued head penetrations used for most high energy piping. Examples of Type 1 penetrations are the main steam and main feedwater lines.

b. Type 2: Closure plate penetrations used for some high-energy, all moderate-energy, and all low-energy general piping. The use of this type of penetration for high energy piping is limited to only those cases where an analysis based on combination of pressure, temperature, and line size has demonstrated the adequacy of the design.

c. Type 3: Spare penetrations reserved for future use.

Typical details of the three types of piping penetrations are shown in Figure 3.8-47.

Type 1 piping penetrations consist of the following major steel items:

a. Process Pipe: This pipe, which is made of welded or seamless carbon or stainless steel and is welded to the flued head, conforms to the requirements of ASME Section III, Subsection NC.

b. Flued Head: This item is made from forged carbon or stainless steel and conforms to the requirements of ASME Section III, Subsection NC. It is designed to contain the full pressure of the process fluid and full reactor building pressure in parts adjoining the pipe sleeve. The connecting process pipes and the flued heads are designed and analyzed to be capable of carrying loads resulting from the failure of the process pipe, as described in Sections 3.6 and 3.9(B).

c. Pipe Sleeve: This steel item consists of the portion which projects into the reactor building and supports the flued head. It conforms to ASME Section III, Subsection NE, except that authorized inspection and stamping are not performed.

Type 2 piping penetrations consist of the following major steel items:

a. Process Pipe: This pipe, which is made of welded or seamless carbon or stainless steel and is welded to the closure plate, conforms to the applicable requirements of ASME Section III, Subsection NC.

b. Closure Plate: This item is made from carbon or stainless steel plate and conforms to the requirements of ASME Section III, Subsection NC.

c. Pipe Sleeve: This steel item consists of the portion which projects into the reactor building and supports the closure plate. It conforms to ASME Section III, Subsection NE, except that authorized inspection and stamping are not performed.

Type 3 spare penetrations consist of the following major items:

a. Solid Closure Plate of Pipe Cap: This item is made from carbon steel and conforms to the requirements of ASME Section III, Subsection NC.

b. Pipe Sleeve: This steel item consists of the portion which projects into the reactor building. It conforms to ASME Section III, Subsection NE, except that authorized inspection and stamping are not performed.

3.8.2.1.3 Fuel Transfer Tube Penetration Sleeve

The fuel transfer tube penetration is provided to transfer fuel between the refueling canal and the spent fuel pool during refueling operations of the reactor. The penetration consists of a 20-inch-diameter stainless steel pipe installed inside a 26-inch sleeve. The steel sleeve which projects into the reactor building conforms to ASME Section III, Subsection NE, except that authorized inspection and stamping are not performed. The inner pipe acts as the transfer tube. The sleeve is designed to provide integrity of the reactor building, allow for differential movement between structures, and prevent leakage through the fuel transfer tube in the event of an accident. Figure 3.8-48 shows details of the fuel transfer tube penetration.

3.8.2.1.4 Electrical Penetration Sleeves

Steel sleeves, which form a portion of the containment pressure boundary, are provided for electrical penetrations. The electrical penetration header plates are designed as discussed in Section 8.1. The sleeve consists of the portion which projects out of the reactor building and supports the electrical assembly. It conforms to ASME Section III, Subsection NE, except that authorized inspection and stamping are not performed. Figure 3.8-49 shows the details of the electrical penetrations.

3.8.2.1.5 Purge Line Penetration Sleeves

The steel sleeves, which are embedded in the reactor building wall concrete, are welded to the purge line piping and form a part of the ASME Section III, Class 2 purge line piping system, as shown in Figure 3.8-50. The sleeves conform to ASME Section III, Subsection NC.

3.8.2.2 <u>Applicable Codes, Standards, and Specifications</u>

The following codes, regulations, standards, and specifications are utilized in the design of the steel portions of the reactor building that are intended to resist pressure but are not backed by structural concrete.

3.8.2.2.1 Regulations

a. 10 CFR 50, "Licensing of Production and Utilization Facilities"

3.8.2.2.2 Codes

a. ASME Boiler and Pressure Vessel Code - 1974 Edition and Later

Section II - Material Specifications

Section III, Division 1 - Nuclear Power Plant Components

Section V - Nondestructive Examination

Section IX - Welding and Brazing Qualifications

b. Acceptable ASME Code cases per Regulatory Guides 1.84 and 1.85, as addressed in Appendix 3A

3.8.2.2.3 Standards and Specifications

Nationally recognized industry standards, such as those published by the ASTM and IEEE, are used whenever possible to define material properties, testing procedures, fabrication, and construction methods. Applicable ASTM standard specifications for materials are those permitted by Article NE-2000 of Section III of the ASME Code. Applicable ASTM standard specifications for nondestructive methods of examination are those referenced in Appendix X, Article X-3000 of Section III of the ASME Code.

Structural specifications are prepared to cover the areas related to the design of steel portions of the containment pressure boundary. These specifications are prepared specifically for the SNUPPS Project. These specifications emphasize the important points of the industry standards for these items and reduce the options that would otherwise be permitted by the industry standards. These specifications cover the following areas:

a. Equipment and personnel access hatches

b. Piping penetration sleeves

c. Fuel transfer tube penetration sleeve

d. Electrical penetration sleeves

e. Purge line penetration sleeves

3.8.2.2.4 Design Criteria

a. 10 CFR 50, Appendix A - General Design Criteria for Nuclear Power Plants (Compliance is discussed in Section 3.1) GDC 2, 4, 16, 50, and 53

b. 10 CFR 50, Appendix J - Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors

c. Bechtel Power Corporation topical reports, as referenced in Section 1.6

3.8.2.2.5 NRC Regulatory Guides

NRC Regulatory Guides 1.29, 1.57, 1.60, 1.61, 1.63, 1.84, and 1.85 are applicable to the design and construction of the steel portions of the reactor building that are intended to resist pressure but are not backed by structural concrete. Specific editions and the extent of compliance with these guides is discussed in Appendix 3A.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 Dead Loads (D)

The dead loads consist of the following typical loads:

a. Weight of the steel item

b. Weight of attached items

c. Weight of electrical connections, mechanisms, ladders, and platforms supported by the containment vessel shell

3.8.2.3.2 Live Loads (L)

The live loads consist of the following typical loads:

a. Live load on the personnel access hatch floor of 200 pounds per square foot

b. Operating fluid weight in attached piping

c. Live load on the equipment hatch floor, using an AASHO (American Association of State Highway Officials) HS-20-44 loading

3.8.2.3.3 Test Pressure Load (P_t)

The structure is designed for a structural integrity test pressure of 69 psig.

3.8.2.3.4 Test Temperature Thermal Load (T_t)

The thermal load associated with a temperature of 100°F is considered as a design basis for the structural integrity test. Testing may proceed at any temperature below this.

3.8.2.3.5 Thermal Loads (T_o, T_e, T_a)

a. Thermal loads produced by the presence of radial and axial temperature gradients during startup, normal, and shutdown conditions (T_o)

b. Thermal conditions causing external pressure (T_e)

c. Thermal conditions generated by the postulated DBA, including $T_o(T_a)$

3.8.2.3.6 Pipe Loads (R_o, R_e, R_a)

The following pipe loads, determined in accordance with procedures described in Section 3.9, are utilized in the design of steel items:

a. Pipe reactions produced during startup, normal, or shutdown conditions (R_o)

b. Pipe reactions under thermal conditions, causing external pressure (R_e)

c. Pipe reactions under thermal conditions generated by the postulated DBA, including $R_o(R_a)$

3.8.2.3.7 Seismic Loads (E, E')

The seismic loads used in the dynamic analysis of the steel items are developed by the use of either a response spectra or time history. The development of this response spectra and/or time history for the SSE and the OBE is discussed in Section 3.7(B) and (N).

3.8.2.3.8 External Pressure Load (P_e)

The design external pressure differential is 3 psig. Refer to Section 3.8.1.3 for a description of this load.

3.8.2.3.9 Pressure Loads (P_a)

Pressure equivalent static load generated by the postulated design basis accident.

3.8.2.3.10 Design Basis Accident (DBA) Loads (Y_r, Y_j, Y_m)

In addition to P_a, T_a and R_a, the following loads are considered:

a. Equivalent static load generated by the reaction on the broken pipe during the design basis accident (Y_r)

b. Jet impingement equivalent static load generated by the broken pipe during the design basis accident (Y_j)

c. Missile impact equivalent static load generated by or during the design basis accident, such as pipe whipping (Y_m)

3.8.2.3.11 Loading Combinations

The following loading combinations are considered:

a. $D + L + P_t + T_t$

b. $D + L + T_o + R_o$

c. $D + L + T_o + R_o + E$

d. $D + L + T_a + R_a + P_a + E$

e. $D + L + T_e + R_e + P_e + E$

f. $D + L + T_a + R_a + P_a + E'$

g. $D + L + T_e + R_e + P_e + E'$

h. $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$

The post-LOCA flooding of the reactor building for the purpose of fuel recovery is not a design loading condition. Refer to Section 3.8.1.3 for a further discussion.

3.8.2.4 Design and Analysis Procedure

Except for the purge line penetration sleeves, the steel items described in Section 3.8.2.1 are designed and analyzed in accordance with Article NE-3000 of Subsection NE of the ASME Code, Section III, Division I and as augmented by the applicable provisions of Regulatory Guide 1.57.

The purge line penetration sleeves are analyzed and designed in accordance with ASME Section III, Subsection NC.

The following paragraphs provide individual descriptions of the design and analysis procedures performed to verify the structural integrity of the steel items.

3.8.2.4.1 Equipment and Personnel Access Hatches

The equipment and personnel access hatches described in Section 3.8.2.1.1 are supported entirely by the concrete shell of the reactor building. The barrels of the personnel hatches are welded to sleeves embedded in concrete which, in turn, are welded at the periphery to the liner plate. The liner plate in the vicinity of the penetration is thickened. The additional thickness in both the barrel and liner plate is provided to satisfy the area reinforcement requirements as well as to resist the external moments and shears due to the cantilevered construction. The discontinuity stresses induced by the combination of external dead and live loads, including the effects of seismic loadings, are evaluated.

The required analyses and limits for the resulting stress intensities are in accordance with Articles NE-3130 and NE-3200 of Section III of the ASME Code.

The doors for both ends of the personnel hatches are of a flat or dished type. The respective analyses are in accordance with Articles NE-3325 and NE-3326 of Section III of the ASME Code. The required analyses and the stress intensity limits are in accordance with Articles NE-3130 and NE-3200 of Section III of the ASME Code. The cover with the bolting flange is designed in accordance with Article NE-3326 of Section III of the ASME Code.

3.8.2.4.2 Piping and Electrical Penetration Sleeves

The penetration sleeves are welded to the thickened areas of the liner plate and are anchored to the reactor building concrete shell.

Penetration sleeves are subjected to various combinations of mechanical, thermal, and seismic loadings. The resulting forces due to these various combinations of loadings are combined with the effects of external and internal pressures. The areas within discontinuities are evaluated to determine the primary and secondary stress intensities.

If the penetration sleeves are subjected to cyclic service, the associated peak stress intensities are also evaluated. The required analysis and associated stress intensity limits are in accordance with Articles NE-3130 and NE-3200 of Section III of the ASME Code.

3.8.2.4.3 Purge Line Penetration Sleeves

The design and analysis of the purge line penetration sleeves are similar to that described in Section 3.8.2.4.2 with stress intensity limits in accordance with ASME Section III, Subsection NC.

3.8.2.4.4 Fuel Transfer Tube Penetration Sleeve

The design and analysis of the fuel transfer tube penetration sleeve are as described in Section 3.8.2.4.2.

3.8.2.4.5 Computer Programs

The computer programs used in the analysis and design of the steel portions of the reactor building intended to resist pressure but not backed by concrete are described in Appendix 3.8A.

3.8.2.5 Structural Acceptance Criteria

The fundamental acceptance criterion for the completed reactor building is successful completion of the structural integrity test.

The structural acceptance criteria for steel items include allowable stress values, deformation limits, and factors of safety, and are established in accordance with ASME Section III, Subsection NC and NE, as applicable, and as augmented by the requirements of Regulatory Guide 1.57. No permanent deformations are allowed under any loading condition.

The steel items, which are an integral part of the reactor building pressure boundary, are designed to meet minimum leakage rate requirements. The leakage rate shall not exceed the acceptable value indicated in the applicable technical specification.

The design and analysis methods, as well as the type of construction materials, are chosen to allow assessment of the steel items' capability throughout the plant life. Additionally, surveillance testing provides further assurances of the steel items' continuing ability to meet their design functions. Surveillance requirements are discussed in Section 3.8.2.7.

The stress limits used for the design of the purge line penetration sleeves are in accordance with Subsection NC of Section III of the ASME Code. The stress limits used for the design of all other steel items are in accordance with Subsection NE of Section III of the ASME Code as augmented by Regulatory Guide 1.57 and are shown in Table 3.8-3 for the load combinations stated in Section 3.8.2.3.11.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The purge line penetration sleeves are fabricated from materials that meet the requirements specified in ASME Section III, Article NC-2000, except as modified by applicable, acceptable ASME Code cases in accordance with Regulatory Guides 1.84 and 1.85. All other steel items are fabricated from materials that meet the requirements specified in Article NE-2000 of Section III of the ASME Code, except as modified by applicable, acceptable ASME Code cases. Specific information relating to materials

used for penetration sleeves is discussed in Section 3.8.1.6.4.1. Details of erection tolerances, quality control, and special construction techniques are provided in Sections 3.8.1.6.4, 3.8.1.6.5, and 3.8.1.6.6.

3.8.2.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance for the steel items consists of leakage testing of the containment. The leakage tests and associated acceptance criteria are discussed in Section 6.2.6.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

3.8.3.1 Description of the Internal Structures

The internal structures consist of the following major components:

a. Reactor support system

b. Steam generator support system

c. Reactor coolant pump support system

d. Primary shield wall and reactor cavity

e. Secondary shield walls

f. Pressurizer support system

g. Refueling canal walls

h. Operating floor

i. Intermediate floors, platforms, and hatches

j. Deleted

k. Polar crane support system

l. Deleted

Descriptions of the supports for the reactor pressure vessel, steam generators, reactor coolant pump, pressurizer, and loop piping are further described in Section 5.4.14.

3.8.3.1.1 Reactor Support System

The general arrangement and principal features of the reactor support system are provided in Figures 3.8-51 and 3.8-52. The reactor vessel is supported by steel assemblies under alternate nozzles of the vessel. These assemblies are designed, furnished, and fabricated by the NSSS manufacturer (refer to Section 5.4.14). The supporting assemblies interface with structural steel built-up members that are almost entirely embedded in the primary shield wall. The reactor vessel is supported to resist normal-operating loads, seismic loads, and loads induced by postulated pipe rupture, including the loss-of-coolant accident. The support system limits the movement of the reactor vessel to within allowable limits under the applicable combinations of loadings, and is designed to minimize resistance to the thermal movements expected during operation.

3.8.3.1.2 Steam Generator Support System

The general arrangement and principal features of the steam generator support system are provided in Figures 3.8-53 through 3.8-55. The four steam generators are located in the loop compartments and are supported by steel assemblies which are designed, furnished, and fabricated as required by the NSSS manufacturer (refer to Section 5.4.14). Four vertical columns beneath each steam generator transfer vertical loads to the reactor building base slab. Lateral supports are provided at the lower portion of each steam generator to transfer horizontal loads to the primary shield wall (or refueling canal walls) and the secondary shield walls. These lateral supports interface with embedded anchor bolt assemblies in the walls. The upper part of each steam generator is supported by a support ring which is restrained by means of shear keys and limit stops. These shear keys and limit stops transfer horizontal loads to the refueling canal walls and the secondary shield walls by interfacing with embedded anchor bolt assemblies in the walls. The steam generators are supported and restrained to resist normal operating loads, seismic loads, and loads induced by pipe rupture. The support system prevents the rupture of the primary coolant pipes due to a postulated rupture in the main steam and feedwater lines and vice versa. The system is designed to minimize resistance to the thermal movements expected during operation.

3.8.3.1.3 Reactor Coolant Pump Support System

The general arrangement and principal features of the reactor coolant pump support system are provided in Figures 3.8-56 and 3.8-57. Each of the four reactor coolant pumps is supported by three vertical columns and three tie rods which are designed, furnished, and fabricated by the NSSS manufacturer (refer to Section 5.4.14). The columns transfer vertical loads to the reactor building base slab. The tie rods transfer horizontal loads to the primary shield wall (or refueling canal walls) and the secondary shield walls by interfacing with structural steel built-up members which are embedded in the walls. The reactor coolant pumps are supported to prevent excessive deflections during normal operating, seismic, and pipe rupture conditions. Under LOCA loads, the pumps are prevented from becoming missiles or generating missiles that might damage

other safety-related components. The system is designed to minimize resistance to the thermal movements expected during operation.

3.8.3.1.4 Primary Shield Wall and Reactor Cavity

The general arrangement and principal features of the primary shield wall are provided in Figures 3.8-58 through 3.8-61. The primary shield wall is a heavily reinforced concrete cylindrical structure extending from the base slab to the seal ring level, with a minimum thickness of 7 feet. The primary shield wall forms the reactor cavity and houses the reactor vessel, provides shielding, and is designed to withstand the pressure of a LOCA. The wall provides support for the reactor vessel, the steam generators, reactor coolant pumps, cross-over legs, and the refueling canal walls above the reactor cavity. Uplift loads arising from lateral forces acting on the wall are transferred to the reactor building base slab by means of the anchorage system. The inside surface of the reactor cavity is lined with welded carbon steel plates. Large penetrations in the primary shield wall are provided for the primary loop piping and the cavity ventilation system.

A permanently installed cavity seal ring/neutron shield assembly rests on the embedment ring. This seal/shield is toroidal in shape, fabricated out of stainless steel and radiation shielding material, and bridges the annular gap between the reactor vessel and vessel cavity wall.

3.8.3.1.5 Secondary Shield Walls

The general arrangement and principal features of the secondary shield walls are provided in Figures 3.8-62 through 3.8-65. The reinforced concrete secondary shield walls are 3 feet 6 inches thick and are anchored to the reactor building base slab. The walls extend from the base slab to a level above the top of the steam generator tube bundle to provide shielding for the reactor coolant system.

Portions of the secondary shield walls above the operating floor are designed to be removable for steam generator removal. These reinforced concrete wall panels are bolted together at vertical joints to provide for structural continuity and integrity. They are keyed into the slab at the bottom of the panels and are prevented from becoming missiles during a seismic event.

The secondary shield walls, in conjunction with the primary shield wall and refueling canal walls, form the loop compartments and provide support for the steam generators, reactor coolant pumps, pressurizer, cross-over legs, piping, various equipment, platforms, and elevated floors.

3.8.3.1.6 Pressurizer Support System

The general arrangement and principal features of the pressurizer support system are provided in Figures 3.8-66 and 3.8-67. The pressurizer is located in a compartment formed by the secondary shield walls and the refueling canal walls, and is supported by

steel assemblies which are designed, furnished, and fabricated by the NSSS manufacturer (refer to Section 5.4.14). The pressurizer support skirt at the bottom of the pressurizer interfaces with heavy structural steel framing which transfers vertical and lateral loads to the secondary shield walls by means of embeds. The upper portion of the pressurizer is supported laterally by lugs which are restrained by means of structural steel assemblies which interface with the embedded anchor bolt assemblies in the secondary shield walls. Using this system, the pressurizer is supported and restrained to resist normal operating loads, seismic loads, and loads induced by postulated pipe rupture. The upper lateral support system is designed to minimize resistance to the thermal movements expected during operation.

3.8.3.1.7 Refueling Canal Walls

The general arrangement and principal features of the refueling canal (pool) walls are provided in Figures 3.8-68 and 3.8-69. The refueling canal is located above and to the south of the reactor cavity on the fuel building side of the reactor. The entire refueling canal is constructed of minimum 4-foot-thick reinforced concrete walls internally lined with a ¼-inch-thick stainless steel liner plate. The canal is flooded during the reactor refueling operation. The refueling canal walls, in conjuction with the secondary shield walls, form the loop compartments and provide support for the steam generators, reactor coolant pumps, piping, various equipment, platforms, and elevated floors.

3.8.3.1.8 Operating Floor

The general arrangement and principal features of the operating floor are provided in Figure 3.8-70. The operating floor level is divided between El. 2,047 feet 6 inches and El. 2,051 feet and is supported by the walls of the refueling pool, the secondary shield walls, and the reactor building shell. The floor supports at the shell consist of structural steel brackets welded to the shell liner and anchored into concrete. As described in Section 3.8.1.1 and shown in Figure 3.8-71, adequate separation is provided between the floor slab and the shell to allow for differential horizontal movement. The floor is constructed of reinforced concrete or steel grating, supported by structural steel framing.

Plugs and removable hatches are provided for equipment removal. They are keyed in to prevent their movement in the horizontal direction. During a seismic event, the vertical components of acceleration will not overcome gravity. Those plugs and removable hatches which are subject to loads during a LOCA are secured from becoming missiles.

3.8.3.1.9 Intermediate Floors and Platforms

The general arrangement and principal features of the intermediate floors are provided in Figures 3.8-72 and 3.8-73. The intermediate floor levels are at El. 2,026 and El. 2,068 feet 6 inches (partial floor). The floors, as well as miscellaneous platforms, are constructed and supported in a manner similar to the operating floor.

3.8.3.1.10 Reactor Missile Shield

The reactor missile shield is integral to the Integrated Head Assembly (IHA) as shown in Figure 3.8-74. The missile shield consists of a 2-inch thick steel plate above the reactor vessel to provide protection against postulated CRDM missiles. As an integral component of the IHA, the missile shield is removed with the removal of the IHA and replacement reactor vessel closure head. During power operation, the missile shield is designed to withstand seismic loads and is restrained laterally for seismic loads by the CRDM seismic support assembly.

3.8.3.1.11 Polar Crane Support System

The general arrangement and principal features of the polar crane support system are provided in Figure 3.8-75. The polar crane is supported by structural steel built-up crane girders mounted on crane brackets evenly spaced around the inside face of the reactor building wall. The crane brackets are welded from steel plates and embedded in the reactor building wall concrete. Further details of these brackets are discussed in Section 3.8.1.1.3.

3.8.3.1.12 Deleted

3.8.3.2 Applicable Codes, Standards, and Specifications

The following codes, regulations, standards, and specifications are utilized in the design of concrete and steel internal structures of the reactor building.

Applicable codes, standards, and specifications for the reactor coolant component supports are discussed in Section 5.4.14.

3.8.3.2.1 Regulations

a. 10 CFR 50, "Licensing of Production and Utilization Facilities"

3.8.3.2.2 Codes

a. American Concrete Institute, Building Code Requirements for Reinforced Concrete (ACI 318-71)

b. American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th Edition, adopted February 12, 1969, and Supplement Nos. 1, 2, and 3 (See FSAR Table 3.2-1, Note 19)

c. American Institute of Steel Construction (AISC), Structural Joints Using ASTM A325 or A490 Bolts, May 8, 1974

d. American Institute of Steel Construction (AISC), Code of Standard Practice for Steel Buildings and Bridges, October, 1972

e. American Welding Society, Structural Welding Code (AWS D1.1-75) (See FSAR Table 3.2-1, Note 19)

f. International Conference of Building Officials, Uniform Building Code, 1973

g. ASME Boiler and Pressure Vessel Code (1974 Edition, including Summer 1975 Addenda)

   Section II - Material Specifications

   Section III, Division 1 - Nuclear Power Plant Components

   Section V - Nondestructive Examination

   Section IX - Welding and Brazing Qualifications

h. Acceptable ASME Code cases per Regulatory Guides 1.84 and 1.85, as addressed in Appendix 3A

3.8.3.2.3 Standards and Specifications

Industry standards, such as those published by the ASTM, are used whenever possible to specify material properties, testing procedures, fabrication, and construction methods. The applicable standards used are discussed in Section 3.8.3.6.

Structural specifications are prepared to cover the areas related to the design and construction of the reactor building internal structures. These specifications are prepared specifically for the SNUPPS project. These specifications emphasize important points of the industry standards for these structures and reduce options such as would otherwise be permitted by the industry standards. These specifications cover the following areas:

a. Concrete material properties

b. Mixing, placing, and curing of concrete

c. Reinforcing steel and splices

d. Structural steel

e. Stainless steel and carbon steel liner plate and embeds

f. Miscellaneous and embedded steel

g. Anchor bolts

h. Grating

i. Deleted

j. RCS support embeds, pipe whip restraints, and embeds

3.8.3.2.4 Design Criteria

a. 10 CFR 50, Appendix A - GDC 2, 3, 4, and 16. (Compliance is discussed in Section 3.1)

b. Bechtel Power Corporation Topical Reports, as referenced in Section 1.6

3.8.3.2.5 NRC Regulatory Guides

NRC Regulatory Guides 1.10, 1.15, 1.55, 1.69, 1.84, 1.85, and 1.94 are applicable to the design and construction of the reactor building internal structures. Specific editions and the extent of compliance with these guides is discussed in Appendix 3A.

3.8.3.3 Loads and Loading Combinations

The loads and loading combinations used in the design of these structures are provided in the sections below.

Loading combinations and design stress limits for the reactor coolant system component supports are discussed in Sections 3.9(N).1.1 and 3.9(N).1.4.7.

3.8.3.3.1 Definitions

The following nomenclature and definition of terms apply to the design of seismic Category I structures. All the major loads to be encountered and/or to be postulated are listed. All the loads listed, however, are not necessarily applicable to all structures and their elements. Loads and the applicable load combinations for which each structure is designed are dependent upon the conditions to which that particular structure is subjected (see Section 3.8.3.3.2).

a. Normal Loads

Normal loads are those loads to be encountered during normal plant operation and shutdown. They include the following:

D = Dead loads or their related internal moments and forces, including any permanent equipment loads and hydrostatic loads

L = Live loads or their related internal moments and forces, including any moveable equipment loads and other loads which vary with intensity and occurrence, such as: Floor area loads, moveable equipment loads, lateral earth pressure, 100-year recurrence snowpack load (listed in Table 1.2-1), and all other live loads during plant operation

T_o = Thermal effects and loads during normal operating and shutdown conditions, based on the most critical transient or steady state condition

R_o = Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition

b. Severe Environmental Loads

Severe environmental loads are those loads that could infrequently be encountered during the plant life. They include the following:

E = Loads generated by the operating basis earthquake (OBE)

W = Loads generated by the design wind, as specified in Section 3.3.1

c. Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but are highly improbable. They include the following:

E' = Loads generated by the safe shutdown earthquake (SSE)

W_t = Loads generated by the design basis tornado, as specified in Section 3.3.2. They include loads due to tornado wind pressure, loads due to the tornado-created differential pressures, and loads due to tornado-generated missiles.

N = Probable maximum winter precipitation (PMWP) in the form of snow, applied to the roofs of safety-related structures, as specified in Table 1.2-1.

d. Abnormal Loads

Abnormal loads are those loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

P_a = Pressure equivalent static load within or across a compartment and/or building, generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load

T_a = Thermal loads under thermal conditions generated by the postulated break and including T_o

R_a = Pipe reactions under thermal conditions generated by the postulated break and including R_o

Y_r = Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load

Y_j = Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load

Y_m = Missile impact equivalent static load on a structure generated by or during the postulated break, such as pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load

In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios and as long as excessive deflections will not result in loss of function of any safety-related system.

e. Other Definitions

S = For concrete structures, S is the required section strength based on the working stress design methods and the allowable stresses defined in Section 8.10 of ACI 318.

For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

The 33 percent increase in allowable stresses for concrete and steel due to seismic or wind loadings is not permitted.

U = For concrete structures, U is the section strength required to resist design loads based on methods described in ACI 318.

Y = For structural steel, Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

3.8.3.3.2 Load Combinations

Structures and components are designed to resist the load combinations given below. Definitions of individual loads are given in Section 3.8.3.3.1.

a. Concrete structures and components

The load combinations and factors for each individual load are given in Table 3.8-4. Wind (W), tornado (W_t), and probable maximum winter precipitation (N) loadings are not applicable for the design of internal structures.

b. Steel structures and components

The load combinations are given in Table 3.8-5. Wind (W), tornado (W_t), and probable maximum winter precipitation (N) loadings are not applicable for the design of internal structures.

3.8.3.3.3 Explanation of Load Combination Cases

a. Loading cases (1) to (3), (1a) to (3a), (1b) to (3b)

These cases include all loads which are expected to be applied during the normal plant operation, including the loads from the design wind and the OBE, as well as loads from thermal effects and pipe reactions.

b. Loading cases (4), (5), and (9)

These cases include events and the resulting loads which are highly improbable, such as the safe shutdown earthquake, tornado, and the probable maximum winter precipitation in the form of snow.

c. Loading case (6)

This case includes the pressure loads and temperature effects resulting from a postulated accident together with pipe rupture loading and generated missiles, where applicable.

d. Loading cases (7) and (8)

These cases include a combination of postulated accident loading, together with loads generated by the operating basis earthquake (OBE) or the safe shutdown earthquake (SSE).

3.8.3.3.4 Specific Considerations

a. In cases (6) to (8), shown in Tables 3.8-4 and 3.8-5, the peak loading effects of pipe rupture and pressurization are considered as acting simultaneously unless time histories of the loading are developed to show the time relationship of the various loads.

b. The mass considered in developing earthquake loading shall be only the mass contributing to dead loads and identifiable live loads.

c. In all loading cases, the live load is considered to vary from zero to the maximum specified value in determining the most critical loading condition.

d. For load cases including either earthquake or tornado loads, the live load (L) shall be limited to only that live load expected to be present when the plant is operating.

3.8.3.3.5 Design Allowables

The section strengths given below are used to evaluate the capacity of the section under consideration.

a. Concrete structures and components.

1. Section strengths are determined in accordance with ACI 318.

2. When the effects of tornado missile impact or pipe rupture impulsive or impactive loading are combined in loading cases (5), (7), and (8) of Table 3.8-4, yield strain and displacement may be exceeded to the limits given in Section 4.3 of BC-TOP-9-A.

3. Yielding of reinforcement is permitted in loading cases (6) to (8) of Table 3.8-4 when T_a is combined with the other loadings, provided the following is satisfied:

(a) The effects of T_a are self-relieving.

(b) The ability of the structure to resist the other loadings is not jeopardized. The stress in concrete in compression is restricted to 0.85 f'_c.

b. Steel structures and components

1. Section strengths are determined in accordance with AISC Specification, Part I. The symbol S is defined as the AISC allowable stress. The permissible stress to be used for each loading case is given in Table 3.8-5.

2. When the effects of tornado missile impact or pipe rupture impulsive or impactive loading are combined in loading cases (5), (7), and (8) of Table 3.8-5, yield strain and displacement may be exceeded to the limits given in Section 4.3 of BC-TOP-9-A.

3. Yielding is permitted in loading cases (6) to (8) of Table 3.8-5 when T_a is combined with the other loadings, provided the following is satisfied:

(a) The effects of T_a are self-relieving.

(b) The ability of the structure to resist the other loadings is not jeopardized.

3.8.3.4 Design and Analysis Procedures

The basic techniques of analyzing the internal structures can be broadly classified into two groups: (1) conventional methods involving simplifying assumptions such as found in beam theory and (2) those based on plate and shell theories of different degrees of approximation. Analytical methods using computer programs, as described in Appendix 3.8A, are also used. Seismic analyses for the internal structures conform to the procedures outlined in Section 3.7(B).

Internal concrete structures are designed, using the strength methods defined in ACI-318. The proportioning of reinforcing steel in concrete structures is based upon accepted codes of practice and detailing methods.

Internal steel structures, except for the NSSS supports, are designed in accordance with AISC specifications. The selection of structural steel sections and the methods of fabrication and connection are in accordance with engineering codes and accepted industry practices. NSSS supports are designed in accordance with ASME Section III Division 1, Subsection NF.

The internal structures are designed to behave within the elastic range under design loads. However, the ability of the structures to perform beyond yield is considered for loads associated with a pipe break as it affects compartment pressurization, jet impingement and pipe whip, and structural loads associated with missile impact.

The loads and loading combinations used in the design of internal structures, as well as the design allowables, are presented in Section 3.8.3.3. As described in Section 3.8.3.1, the internal structures are designed to transfer loads to the foundation by means of anchorage systems. The applicable codes, standards, and specifications used are discussed in Section 3.8.3.2.

The following sections discuss, in greater detail, the procedures used for analyzing and designing the reactor coolant system supports, the primary shield wall and reactor cavity, the secondary shield walls, and the refueling canal walls.

3.8.3.4.1 Reactor Coolant System Supports

Models and methods of analysis for the reactor coolant system component supports are discussed in Section 3.9(N).1.4.4.

3.8.3.4.2 Primary Shield Wall and Reactor Cavity

The primary shield wall is designed to resist all of the applicable loads, including those due to differential pressure and temperature resulting from a LOCA, operating temperatures, OBE and SSE, and those loads transmitted through the reactor vessel supports. During normal plant operation, a thermal loading on the wall is generated by the attenuation heat of gamma and neutron radiation originating from the reactor core. An insulation and cooling system is provided on the inside face of the wall to reduce the severity of this loading by limiting the core concrete temperatures to 150°F except for the area directly below the seal ring support which is limited to 300°F.

Analysis of the primary shield wall, depending on the loading condition being considered, is performed using classical techniques and the SAP, ASHSD, and FINEL computer programs described in Appendix 3.8A. The boundary conditions simulate actual conditions at the reactor building base slab and intersections with the refueling canal walls. Analyses for LOCA loads applicable to the primary shield wall, such as those for differential pressure and pipe rupture reaction forces, are treated as time-dependent loads by performing a static analysis and utilizing the peak of the forcing function amplified by an appropriately chosen dynamic load factor.

The methods used for determining the effective dynamic load factors are in accordance with recognized dynamic analysis methods, such as those described by Reference 1. The analysis considers the nonaxisymmetric application of loads to the structure. The finite element model used for the analysis of the primary shield wall is shown in Figure 3.8-83.

Design of the primary shield wall is performed, using the strength design methods described in ACI-318.

3.8.3.4.3 Secondary Shield Walls

The secondary shield walls are designed to resist all of the applicable loads, including those due to differential pressure and temperature resulting from a LOCA, RCS component support forces, OBE and SSE, dead and live loads from the operating floor and intermediate platforms and walkways, and those loads resulting from a postulated pipe break. The design of the ‘C’ loop steam generator cubicle secondary shield wall does not have to consider the dynamic effects of postulated pipe breaks in the RCS primary loop and certain Class 1 branch lines (i.e., the 12-inch RHR hot leg suction lines and the 10-inch accumulator injection lines). Postulated pipe breaks in the pressurizer surge line and in piping with a diameter less than 10 inches must still be considered in the structural design basis.

Analysis of the secondary shield walls is performed, using classical techniques and the SAP and ANSYS computer programs described in Appendix 3.8A. Design for the effects of postulated pipe breaks is performed using BN-TOP-2.

The finite element model used for analyzing the secondary shield walls consists of a three-dimensional model of one-half of the structure in plan about an axis of symmetry. An additional finite element model is used for analyzing these secondary shield walls at the pressurizer. Appropriate boundary conditions are modeled to simulate actual conditions at the axis of symmetry and at the intersections with the base slab, refueling canal walls, floors, and RCS component supports. The analysis for time-dependent loads, such as those for differential pressure and pipe rupture reaction forces, is performed in a manner similar to that used for the primary shield wall. The finite element models used for the secondary shield walls are shown in Figures 3.8-79 through 3.8-82.

Design of the secondary shield walls is performed, using the strength design methods described in ACI-318.

3.8.3.4.4 Refueling Canal Walls

The refueling canal walls are designed to resist all of the applicable loads, including those due to differential pressure and temperature resulting from a LOCA, RCS component support forces, OBE and SSE, hydrostatic loading during the refueling operation, dead and live loads from the operating floor and intermediate platforms and walkways, and those loads resulting from a postulated pipe break.

Analysis of the refueling canal walls is performed, using classical techniques and the SAP computer program described in Appendix 3.8A. Design for the effects of postulated pipe breaks is performed using BN-TOP-2.

The finite element model used for analyzing the refueling canal walls consists of a three-dimensional model of the entire structure. Appropriate boundary conditions are modeled to simulate actual conditions at the intersections with the base slab, secondary shield walls, primary shield wall, floors, and RCS component supports. The analysis for time-dependent loads, such as those for differential pressure and pipe rupture reaction forces, is performed in a manner similar to that used for the primary shield wall. The finite element model used for the refueling canal walls is shown in Figures 3.8-77 and 3.8-78.

Design of the refueling canal walls is performed using the strength-design methods described in ACI-318.

3.8.3.5 Structural Acceptance Criteria

The structural acceptance criteria for the concrete and steel internal structures are defined in Section 3.8.3.3.

Stress criteria for the reactor coolant system component supports are discussed in Section 3.9(N).1.4.7.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

This section contains information relating to the materials, quality control programs, and special construction techniques used in the fabrication and construction of concrete and steel internal structures of the reactor building.

3.8.3.6.1 Concrete

Structural concrete used in the construction of the reactor building internal structures has a compressive strength, $f'_{c'}$ of 4,000 psi at 28 days. The concrete materials, mix design, examination, and placement are described in Section 3.8.1.6.1.

3.8.3.6.2 Reinforcing Steel and Splices

The reinforcing steel and splices used in the construction of the reactor building internal structures, including materials, examination, and erection tolerances, are described in Section 3.8.1.6.2.

3.8.3.6.3 Structural Steel

The following sections describe the basic materials, examination, and erection of structural steel items.

3.8.3.6.3.1 Materials

Structural steel shapes, plates, and bars conform to the requirements of the Specification for Structural Steel (ASTM A36).

High strength bolting materials conform to the requirements of the Specification for High Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers (ASTM A325) or the Specification for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints (ASTM A490). Other bolting materials conform to the requirements of the Standard Specification for Low-Carbon Steel Fasteners (ASTM A307).

Welding electrode materials are selected on the basis of the welding process used and the type of materials to be joined and in accordance with the requirements of AWS D1.1. Written welding material control procedures are required which define the measures used to control the use of the materials throughout all welding operations.

Certified material test reports are obtained for structural steel shapes, plates, and bars. All other structural steel materials are furnished with certificates of compliance.

3.8.3.6.3.2 Examination

Nondestructive examination of structural steel welds is performed in accordance with the requirements of AWS D1.1 and as augmented by design documents prepared for the SNUPPS project. Inspection of high strength bolted joints is performed in accordance with the requirements of the AISC Specification for Structural Joints Using ASTM A325 or A490 Bolts and as augmented by design documents prepared for the Callaway Plant (see FSAR Table 3.2-1, Note 19).

3.8.3.6.3.3 Erection

Structural steel is erected to the following codes, to the extent described:

a. AWS D1.1 Structural Welding Code is used with the following exceptions (see FSAR Table 3.2-1, Note 19):

   1. Undercut of welds shall not exceed 1/32 inch.

   2. Fillet welds need not satisfy the convexity limitations of Section 3.6.1 provided that all other parameters of acceptable weld profile are maintained.

   3. Fillet welds deposited on opposite sides of a common plane of contact between two parts need not be interrupted at the corner common to both welds as specified by Section 8.8.5. The

connecting weld shall be inspected for defects such as undercut and cracking, but need not be inspected for size.

b. AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, Sections 1.23 and 1.25, are used without exception (see FSAR Table 3.2-1, Note 19).

c. AISC Specification for Structural Joints Using ASTM A325 and A490 Bolts is used without exception.

d. Erection tolerances are in accordance with the AISC Code of Standard Practice for Steel Buildings and Bridges without exception (see FSAR Table 3.2-1, Note 19).

3.8.3.6.4 Restraints and Embedded Items

The following sections describe the basic materials, examination, and erection of pipe whip restraints, pipe whip restraint embeds, RCS component support embeds, and other miscellaneous embedded carbon and stainless steel items.

3.8.3.6.4.1 Materials

Structural steel plates, shapes, and bars conform to the requirements of the Specification for Structural Steel (ASTM A36) or the Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service (ASTM A516), Grade 70, or the Specification for Pressure Vessel Plates, Alloy Steel, Quenched and Tempered (ASTM A533), Class 2.

Materials for high strength steel bolts conform to the requirements of the AISC Specification for Structural Joints Using ASTM A325 or A490 bolts. Materials for other bolts and upset rods conform to the requirements of the Specification for Carbon Steel Externally and Internally Threaded Standard Fasteners (ASTM A307).

Materials for shear connector studs conform to the requirements of the Specification for Steel Bars, Carbon, Cold-Finished, Standard Quality (ASTM A108), Grades 1015 and 1020, cold drawn steel.

Materials for upset rods conform to the requirements of the Specification for Stainless and Heat-Resisting Steel Bars and Shapes for Use in Boilers and Other Pressure Vessels (ASTM A479) or to the requirements of the Specification for Carbon Steel Externally and Internally Threaded Standard Fasteners (ASTM A307).

Materials for structural pipe conform to the requirements of the Specification for Welded and Seamless Steel Pipe (ASTM A53), Grade B, or the Specification for Seamless Carbon Steel Pipe for High-Temperature Service (ASTM A106), Grade B, or the Specification for Blank and Hot Dipped Zinc Coated (Galvanized) Welded and Seamless

Steel Pipe for Ordinary Uses (ASTM A120) or the American Petroleum Institute Specification for High Test Line Pipe (API-5L), Grade B, or the Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes (ASTM A500), Grade B, or the Specification for Hot-Formed Welded and Seamless Carbon Steel Structural Tubing (ASTM A501).

Materials for shear pins conform to the requirements of the Specification for Alloy-Steel and Stainless Steel Bolting Materials for High-Temperature Services (ASTM A193) Grade B7 or to the requirements of the Specification for Alloy Steel Bolting Materials for Special Applications (ASTM A540), Grade B23.

Materials for stainless steel plates conform to the requirements of the Specification for Heat Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels (ASTM A240).

Embedded anchor bolt materials conform to the applicable requirements of ASTM A36 or ASTM A193 or the Specification for Carbon and Alloy Steel Nuts for Bolts for High Pressure and High Temperature Service (ASTM A194) or ASTM A307 or ASTM A325 or the Specification for Quenched and Tempered Alloy Steel Bolts and Studs With Suitable Nuts (ASTM A354) or the Specification for Quenched and Tempered Steel Bolts and Studs (ASTM A449) or ASTM A490 or the Specification for Alloy Steel Bolting Materials for Special Applications (ASTM A540).

Welding electrode materials are selected based on the welding process used and the type of material being joined and in accordance with the requirements of AWS D1.1 or the ASME Code. Written welding material control procedures are required which define the measures used to control the use of the materials throughout all welding operations.

All materials used for restraints and embedded items described above are furnished with certified material test reports or certificates of compliance.

3.8.3.6.4.2 Examination

One of the following nondestructive examinations is selectively performed prior to operation on pipe whip restraint, pipe whip restraint embed, and RCS component support embed welds:

a. Visual examination of all welds (see FSAR Table 3.2-1, Note 19).

b. Magnetic particle or liquid penetrant examination of welds, in accordance with AWS D1.1

c. Radiographic examination of welds in accordance with AWS D1.1

All other welds are examined in accordance with AWS D1.1 (see FSAR Table 3.2-1, Note 19).

High strength bolted joints are examined in accordance with the requirements of the AISC Specification for Structural Joints Using ASTM A325 or A490 Bolts.

Examination of embedded anchor bolt materials used for RCS component support embeds meets the requirements of Section NF-2580 of the ASME Code for Class 1 component supports.

3.8.3.6.4.3 Erection

Restraints and embedded items are erected in accordance with the following:

a. AWS D1.1 Structural Welding Code is used, except that the qualification of welders and welding operators may, alternatively, be in accordance with ASME Section IX. In addition, weld procedures for joining structural steel and sleeves used for mechanical splicing of reinforcing steel may be qualified in accordance with ASME Section IX. The following exceptions are allowed for welding between anchor studs and plates embedded in concrete (see FSAR Table 3.2-1, Note 19):

    1. Vertical leg of weld may be up to 1/16 inch smaller than that specified on drawings.

    2. Unequal legs are permitted.

    3. Weld profile and convexity requirements for these welds need not be imposed.

    4. An undercut of up to 1/16 inch for 10 percent of weld length may be permitted.

b. AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings Sections 1.23 and 1.25 are used without exception (see FSAR Table 3.2-1, Note 19).

c. AISC Specification for Structural Joints Using ASTM A325 or A490 Bolts is used without exception.

d. Erection tolerances for pipe whip restraints, pipe whip restraint embeds, and RCS component support embeds are in accordance with the following:

    1. AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, for rolled plates and shapes

    2. AWS D1.1 Structural Welding Code for welded assemblies

3. Additional tolerance requirements are specified in design documents prepared for the SNUPPS project for bearing or contact points, clearances, and transverse locations of restraints

e. Erection tolerances for other embedded items described above are the same as those for concrete forms. All embedded items are secured and protected during placement of concrete.

3.8.3.6.5 Reactor Coolant System Supports

Materials, quality control, and special construction techniques for the reactor coolant system supports are discussed in Section 5.4.14.

3.8.3.6.6 Quality Control

In addition to the quality control procedures discussed in Sections 3.8.3.6.1 through 3.8.3.6.5, the construction quality control program is discussed in Chapter 17.0.

3.8.3.6.7 Special Construction Techniques

The reactor building internal structures are constructed, using proven methods common to heavy industrial construction. No special, new, or unique construction techniques are used.

3.8.3.7 <u>Testing and Inservice Surveillance Requirements</u>

Tests and inspections for the reactor coolant system component supports are discussed in Section 5.4.14.

No formal testing or inservice surveillance is required of the internal structures.

3.8.4 OTHER CATEGORY I STRUCTURES

3.8.4.1 <u>Description of the Structures</u>

The general arrangement of all standard plant seismic Category I structures is shown in Figure 3.8-84. The standard plant seismic Category I structures other than the reactor building are:

a. Auxiliary building

b. Fuel building

c. Control building

d. Diesel generator building

e. Refueling water storage tank and valve house

f. Emergency fuel oil storage tanks and vault

g. Buried power block duct banks and piping

The site-related seismic Category I structures are described in each Site Addendum.

All standard plant seismic Category I structures are physically separated from adjacent structures by isolation joints, with the exception of the auxiliary and control buildings which share a common base slab and wall. The isolation joints at the roof, base slab, and exterior walls of all the buildings contain waterstops to provide environmental protection while allowing free rotation and translation between structures. Figure 3.8-85 shows typical isolation joint details.

3.8.4.1.1 Auxiliary Building

The auxiliary building is a multistory, structural steel and reinforced concrete structure which houses the safety injection system, residual heat removal system, CVCS monitoring system, auxiliary feedwater pumps, steam and feedwater isolation and relief valves, heat exchangers, other pumps, tanks, filters, and demineralizers, and heating and ventilating equipment. The arrangement of the auxiliary building is shown in Figures 3.8-86 through 3.8-93. The RAM storage building has been constructed upon the section of roof at elevation 2047'-2".

The auxiliary building shares a common base mat and wall with the control building. The building interior is enclosed on one side by the reactor building wall.

The foundation for the auxiliary building is a two-way mat foundation with a minimum thickness of 5.0 feet. The lowest floor elevation is 25.5 feet below plant grade, except for the RHR and containment spray pumps pit which is 33.5 feet below grade. The roof is 74.7 feet above plant grade, except for the southwest corner which is 48 feet above grade (serves as the floor for the RAM storage building), two penthouses which are 84 feet above grade, and the roof over the main steam tunnel, which is 103 feet above plant grade.

The intermediate floors and the roof are reinforced concrete slabs supported by structural steel beams and girders. The floor and roof framing are supported by exterior reinforced concrete bearing walls and interior steel columns. The roof slab and exterior walls are designed to prevent penetration by tornado-generated missiles.

Concrete plugs provided in the roof for equipment removal are designed to resist tornado missiles. These plugs and additional concrete plugs and removable hatches provided for servicing equipment within the building are adequately anchored or keyed into slabs to prevent displacement during a seismic event.

Blockouts are provided in the interior walls for equipment removal and servicing. These blockouts are closed with multiwythes of solid concrete blocks, laid such that the vertical and horizontal joints are not continuous. The blocks are seismically restrained on both faces.

Concrete block walls are reinforced to withstand seismic loadings.

3.8.4.1.2 Fuel Building

The fuel building is a rectangular, structural steel, reinforced concrete structure which houses the spent fuel pool, transfer canal, cask loading pool and cask washdown pit, spent fuel pool bridge crane, cask handling crane, and other miscellaneous equipment. The arrangement of the fuel building is shown in Figures 3.8-94 through 3.8-98.

The fuel building is supported on a two-way, reinforced concrete base mat which is founded 6 feet below plant grade. The minimum thickness of the mat is 6.5 feet, and the mat beneath the spent fuel pool is 12 feet thick. The top of the roof slab is 107 feet above plant grade.

The elevated floors and the roof are reinforced concrete slabs supported by structural steel beams and girders. The floor and roof framing are supported by reinforced concrete bearing walls. The exterior walls have integral reinforced concrete pilasters to stiffen the walls against lateral loads and to support the cask-handling crane girders. The roof and exterior walls are designed to prevent penetration by tornado-generated missiles.

The walls and base slab of the spent fuel pool, transfer canal, and cask loading pool are lined with stainless steel plates for ease of decontamination. A leak chase system is provided to check the leaktightness of the liners, although leaktightness is not the primary liner function.

The cask handling crane is single failure proof, and is capable of moving a loaded fuel cask. The crane travel is limited to prevent movement over the entire fuel storage pool.

3.8.4.1.3 Control Building

The control building is a rectangular structural steel and reinforced concrete structure which houses the access control areas, control room, upper and lower cable spreading rooms, electrical and mechanical equipment rooms, and locker rooms.

The arrangement of the control building is shown in Figures 3.8-99 through 3.8-104.

The control building shares a common base slab and wall with the auxiliary building. The bottom of the base mat is 31.5 feet below plant grade, and the mat thickness is 6 feet. The top of the roof is 81.7 feet above plant grade. The intermediate floors and roof are reinforced concrete slabs supported by structural steel beams and girders. The floor and

roof framing are supported by exterior reinforced concrete bearing walls and interior steel columns. The roof slab and exterior walls are designed to prevent penetration by tornado-generated missiles.

Concrete block walls are reinforced to withstand seismic loadings.

3.8.4.1.4 Diesel Generator Building

The diesel generator building is a single-story, rectangular, structural steel and reinforced concrete structure which houses the standby diesel generators, fuel oil day tank, exhaust silencers, and exhaust stacks. The diesel generator building arrangement is shown in Figures 3.8-105 through 3.8-109.

The foundation for the diesel generator building is a 10.5-foot-thick base mat founded 10 feet below plant grade. The highest portion of the roof is 66.5 feet above plant grade. The roof is a reinforced concrete slab supported by structural steel beams and girders. The roof framing is supported by reinforced concrete bearing walls and steel columns. The roof and exterior walls are designed to prevent penetration by tornado-generated missiles.

3.8.4.1.5 Refueling Water Storage Tank

The refueling water storage tank consists of an above-grade cylindrical steel tank founded on a 5-foot-6-inch-thick reinforced concrete base slab and an associated valve house. Although serving a safety-related function and designed as a seismic Category I structure, the refueling water storage tank is not required for safe shutdown of the plant following a tornado event and is, therefore, not designed to resist the effects of the design-basis tornado. The steel tank is described in Section 6.3. Details of the tank foundation and valve house are shown in Figures 3.8-110 and 3.8-111.

3.8.4.1.6 Emergency Fuel Oil Storage Tanks

The emergency fuel oil storage tanks consist of two buried cylindrical steel tanks and associated reinforced concrete access vaults. The steel tanks are described in Section 9.5.4.

Details of the access vaults are shown in Figures 3.8-112 and 3.8-113.

3.8.4.1.7 Buried Duct Banks and Piping

Buried, reinforced concrete electrical duct banks and steel piping that serve safety-related functions are classified as seismic Category I and are shown in Figures 3.8-114 and 3.8-115.

3.8.4.2 Applicable Codes, Standards, and Specifications

The codes, regulations, standards, and specifications utilized in the design of the standard plant seismic Category I structures other than the reactor building are the same as those listed in Section 3.8.3.2, with the following exceptions:

a. Structural Specification for Maintenance Truss

b. Structural Specification for RCS Support Embeds, Pipe Whip Restraints, and Embeds

c. The applicable standards used are discussed in Section 3.8.4.6.

In addition to the documents listed in Section 3.8.3.2, the following documents are also utilized:

a. NRC Regulatory Guide 1.59

b. NRC Regulatory Guide 1.76

c. Bechtel Power Corporation Topical Report BC-TOP-3A, Tornado and Extreme Wind Design Criteria for Nuclear Power Plants, Revision 3, August, 1974.

3.8.4.3 Loads and Load Combinations

The loads and load combinations used in the design of the standard plant seismic Category I structures other than the reactor building are the same as those described in Section 3.8.3.3 with the following exception. In accordance with the discussion in Section 3.8.4.1.1 and Appendix 3B.4 the terms Y_j, Y_r, and Y_m in Tables 3.8-4 and 3.8-5 do not apply to the main steam isolation valve room since no pipe breaks are postulated in that area.

3.8.4.4 Design and Analysis Procedures

The analysis of standard plant seismic Category I structures other than the reactor building is performed, using conventional analytical methods which are common to standard engineering practice and analytical methods using computer programs. Analytical methods using computer programs are described in Appendix 3.8A. Seismic analysis conforms to the procedures outlined in Section 3.7(B). Concrete structures are designed, using the strength methods defined in ACI-318. The reinforcing steel is proportioned in accordance with accepted engineering formulae and conforms to the applicable codes and standards. The effects of design variables are accounted for by the use of conservative loads and load combinations and the use of load factors and capacity reduction factors.

Steel structures and components, except for tanks and piping, are designed in accordance with AISC specifications. The selection of steel sections is in accordance with accepted engineering formulae and conforms to the applicable codes and standards. The effects of design variables are accounted for by the use of conservative loads, load combinations, and allowable stresses.

These structures are designed to behave within the elastic range, under normal operating loads. However, the ability of the structures to perform beyond the yield point is considered for loads associated with missile impact, jet impingement, and pipe whip.

The loads, load combinations, and design allowables used in the design of these structures are presented in Section 3.8.4.3. The applicable codes, regulations, standards, and specifications used are discussed in Section 3.8.4.2.

The following sections discuss, in greater detail, the procedures used for the analysis and design of the auxiliary and control buildings, fuel building, and diesel generator building.

3.8.4.4.1 Auxiliary and Control Building

The auxiliary and control buildings are supported on a common base slab. All vertical loads are transferred to the base slab through reinforced concrete bearing walls and structural steel columns. All lateral loads are resisted by diaphragm action of the roof and intermediate floor slabs which transfer these loads to shear walls, which, in turn, transfer the lateral loads to the base slab. All lateral loads are transferred to the subgrade by friction and passive earth pressure. Typical connection details between the walls and slabs are shown in Figures 3.8-116 through 3.8-118.

The reinforced concrete roof and intermediate floor slabs are analyzed and designed for vertical loads as one-way or two-way slabs supported by bearing walls and structural steel beams and girders. The reinforced concrete interior and exterior walls are analyzed and designed for lateral loads as one-way or two-way slabs supported by the base slab, intermediate floor slabs, roof slab, and perpendicular walls.

Structural steel beams and girders supporting reinforced concrete slabs are analyzed and designed as composite sections.

The reinforced concrete base slab is analyzed and designed as a rigid slab on an elastic foundation.

The main steam isolation valve room is located in the northwest corner of the auxiliary building as shown in Figure 3B-2. It is designed to withstand the environmental effects, by means of venting, of a main steam or main feedwater line break equivalent to the flow area of a single-ended pipe rupture. Although no specific pipe breaks are postulated in the main steam/main feedwater isolation valve compartment, this consideration provides

an additional level of assurance of operability to the building structure and the safety-related equipment in this compartment.

3.8.4.4.2 Fuel Building

The fuel building is supported on a base slab. All vertical loads are transferred to the base slab through the exterior walls, interior walls, and fuel storage pool walls. All lateral loads are transferred to the base slab by diaphragm action of the roof slab and intermediate floor slabs which transfer loads to shear walls. All hydrostatic and hydrodynamic loads due to the presence of water in the fuel storage pool are transferred to the base slab through the fuel storage pool walls. All lateral loads are transferred to the subgrade by friction and passive earth pressure. Typical connection details between exterior, interior, and fuel storage pool walls and the base slab are shown in Figures 3.8-116 and 3.8-118.

The reinforced concrete roof and intermediate floor slabs are analyzed and designed for vertical loads as one-way or two-way slabs supported by bearing walls and structural steel beams and girders. The fuel storage pool is analyzed and designed as an open top, reinforced concrete tank.

The reinforced concrete interior and exterior walls are analyzed and designed for lateral loads as one-way slabs supported by the base slab, intermediate floor slabs, and roof slab. Structural steel beams and girders supporting reinforced concrete slabs are analyzed and designed as composite sections.

The reinforced concrete base slab is analyzed and designed as a rigid slab on an elastic foundation.

3.8.4.4.3 Diesel Generator Building

The diesel generator building is supported on a base slab. All vertical loads are transferred to the base slab through exterior walls, interior walls, and columns. All lateral loads are transferred to the base slab by diaphragm action of roof slab and intermediate floor slab, which transfer loads to shear walls and bracing. All lateral loads are transferred to the subgrade by friction and passive earth pressure. Typical connection details between the exterior and interior walls and the base slab are shown in Figures 3.8-116 and 3.8-118.

The reinforced concrete roof and intermediate floor slabs are analyzed and designed for vertical loads as one-way or two-way slabs supported by the base slab, intermediate floor slab, roof slab, and intersection walls.

Structural steel beams and girders supporting reinforced concrete slabs are analyzed and designed as composite sections. The reinforced concrete base slab is analyzed and designed as a rigid slab resting on an elastic foundation.

3.8.4.5 Structural Acceptance Criteria

The structural acceptance criteria for the standard plant seismic Category I structures other than the reactor building are the same as those defined in Section 3.8.3.3.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

The materials, quality control programs, and special construction techniques used in the fabrication and construction of standard plant seismic Category I structures other than the reactor building are described in the following sections.

3.8.4.6.1 Concrete

Structural concrete used in the construction of these structures has a minimum compressive strength, $f'_{c'}$ of 4,000 psi at 28 days. The concrete materials, mix design, examination, and placement are described in Section 3.8.1.6.1.

3.8.4.6.2 Reinforcing Steel and Splices

The reinforcing steel and splices used in the construction of these structures, including materials, examination, and erection tolerances, are described in Section 3.8.1.6.2.

3.8.4.6.3 Structural Steel

The structural steel used in the construction of these structures, including materials, examination, and erection, are described in Section 3.8.3.6.3.

3.8.4.6.4 Embedded Items

The embedded carbon steel items used in the construction of these structures, including materials, examination, and erection, are described in Section 3.8.3.6.4.

3.8.4.6.5 Quality Control

The quality control measures are discussed in Sections 3.8.4.6.1 through 3.8.4.6.4. The construction quality control program is discussed in Chapter 17.0.

3.8.4.6.6 Special Construction Techniques

These structures are constructed of concrete and steel, using proven methods common to heavy, industrial construction. No special, new, or unique construction techniques are used.

3.8.4.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance are not required for seismic Category I structures other than the reactor building. Hence, no formal program of testing and inservice surveillance is planned.

3.8.5 FOUNDATIONS

3.8.5.1 Description of the Foundations

All standard plant seismic Category I structures have reinforced concrete mat foundations resting on existing rock, undisturbed soil, or engineered backfill. All vertical loads are transferred to the subgrade by direct bearing of the base mat on the foundation media. Horizontal shears, such as those produced by winds and earthquakes, are transferred to the subgrade by friction along the bottom of the base mat. There is no waterproofing membrane between the base mats and the subgrade.

The foundation for each structure is separated by isolation joints from adjacent foundations and structures, with the exception of the auxiliary and control buildings which share a common base mat. All the foundations are adequately designed to prevent overturning due to horizontal loads.

The following sections describe the standard plant Category I foundations. Figure 3.8-116 shows the general arrangement of these foundations. Refer to Section 3.8.5 of each Site Addendum for a description of the nonstandard (site-related) Category I structures.

3.8.5.1.1 Reactor Building

The reactor building foundation is a 10-foot-thick reinforced concrete mat, 154 feet in diameter, founded 11 feet below plant grade. The central reactor cavity and instrumentation tunnel extend below the reactor building foundation, with the bottom of the 5.5-foot-thick foundation slab located 36 feet below grade. The 8-foot-wide tendon access gallery, located beneath the perimeter of the reactor building mat, has a 4.25-foot-thick foundation slab, the bottom of which is 25.25 feet below grade. The plan and details of the reactor building foundation are shown in Figures 3.8-1 and 3.8-8 through 3.8-11.

Refer to Section 3.8.3.1 for a description of the anchorage of internal structures and equipment to the foundation.

3.8.5.1.2 Auxiliary and Control Buildings

The auxiliary and control buildings are supported by a common, reinforced concrete mat foundation, with a minimum thickness of 5 feet, founded 31.5 feet below plant grade. The foundation under the RHR and containment spray pumps pit in the auxiliary building

is a 6-foot-thick mat, the bottom of which is 38.5 feet below grade. The shape of the base mat in plan conforms to the arrangement of the building it supports, and the base mat is approximately 220 feet wide at its widest section. The plan and details of the foundation for the auxiliary and control buildings are shown in Figures 3.8-119 and 3.8-120.

The equipment in these buildings, such as tanks, heat exchangers, switchgear, and control panels, is rigidly attached to the base mat, intermediate floor slabs, or walls, by means of anchor bolts or welding to embedments in the concrete. All loads from equipment and internal structures not directly attached to the base mat are transferred to the base mat through structural steel columns, which are attached to the base mat by anchor bolts, or reinforced concrete bearing and shear walls, which are anchored to the base mat by reinforcing steel dowels.

3.8.5.1.3 Fuel Building

The fuel building foundation is a 6.5-foot-thick reinforced concrete mat extending 6 feet below plant grade. The mat is essentially rectangular with overall dimensions of 137 feet long and 91 feet wide. The thickness of the mat below the spent fuel pool is increased to 12 feet. Figures 3.8-121 and 3.8-122 show the general arrangement and details of the fuel building foundation.

The fuel storage racks are supported by the base slab of the fuel storage pool. Other equipment is rigidly attached to the base mat, intermediate floors, or walls by means of anchor bolts or welding to embedments in the concrete. All loads from equipment and internal structures not directly attached to the base mat are transferred to the base mat through reinforced concrete walls and pilasters, which are anchored to the base mat by reinforcing steel dowels.

3.8.5.1.4 Diesel Generator Building

The diesel generator building is supported by a 10.5-foot-thick reinforcing concrete mat, the bottom of which is 10 feet below plant grade. The mat is rectangular, and is 88.25 feet long and 66.25 feet wide. Figure 3.8-123 shows the general arrangement and details of the diesel generator building foundation.

The diesel generators are rigidly attached to the base mat by means of anchor bolts. Other equipment is rigidly attached to the base mat, intermediate platforms, walls, or roof by means of anchor bolts or welding to structural steel framing or embedments in the concrete. All loads from equipment and internal structures not directly attached to the base mat are transferred to the base mat through structural steel columns, which are attached to the foundation by anchor bolts, or reinforced concrete walls, which are anchored to the base mat by reinforcing steel dowels.

3.8.5.1.5 Refueling Water Storage Tank

The refueling water storage tank is supported by a 5.5-foot-thick reinforced concrete base mat which extends 4.5 feet below plant grade. The base mat is octagonal, with a distance of 43 feet between parallel edges. Figures 3.8-110 and 3.8-111 show the general arrangement and details of the refueling water storage tank foundation.

The refueling water storage tank is rigidly attached to the base mat by means of anchor bolts which transfer all loads, including seismic lateral forces, to the foundation.

3.8.5.2 Applicable Codes, Standards, and Specifications

Applicable codes, regulations, standards, and specifications are the same as those discussed in Section 3.8.1.2 for the reactor building and in Section 3.8.4.2 for the other standard plant seismic Category I structures.

3.8.5.3 Loads and Load Combinations

The reactor building foundation loads and load combinations are as discussed in Section 3.8.1.3.

Foundation loads and load combinations for the other standard plant seismic Category I structures are as discussed in Section 3.8.4.3.

3.8.5.4 Design and Analysis Procedures

The design and analysis procedures for the reactor building foundation are discussed in BC-TOP-5-A.

The foundations for other standard plant seismic Category I structures are analyzed as flat slabs on elastic supports. Loads are applied to the slab through structural steel columns and reinforced concrete walls, with the resulting foundation-bearing pressures being determined using well-established principles and methods of engineering mechanics.

The foundations for the standard plant seismic Category I structures are designed using the strength design methods defined in ACI 318. The reinforcing steel is proportioned in accordance with accepted engineering formulas and conforms to the applicable codes and standards. The effects of design variables are accounted for by the use of conservative loads and load combinations and the use of load factors and capacity reduction factors.

3.8.5.5 Structural Acceptance Criteria

The foundations for all standard plant seismic Category I structures are designed to meet the same structural acceptance criteria as the structures themselves. The criteria are discussed in Sections 3.8.1.5 and 3.8.4.3.

Minimum safety factors for seismic Category I foundations, for the load combinations given in Sections 3.8.1.3 and 3.8.4.3, are:

Overturning	1.50
Sliding	1.10
Buoyancy	1.25

The limiting conditions for the foundation media are given in Section 2.5.4.10.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The foundations for the standard plant seismic Category I structures are constructed of reinforced concrete, using proven methods common to heavy industrial construction. For further discussion, refer to Sections 3.8.1.6 and 3.8.4.6.

3.8.5.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance are not required, nor planned, for the foundations of the seismic Category I structures.

3.8.6 RADWASTE BUILDING AND TUNNEL

3.8.6.1 Description of the Structures

3.8.6.1.1 Radwaste Building

The radwaste building is a rectangular, multistory, structural steel and reinforced concrete structure which houses radioactive waste treatment facilities, tanks, filters, and other miscellaneous equipment. Figures 3.8-124 through 3.8-130 show the general arrangement of the building.

The radwaste building is supported on a reinforced concrete mat foundation with a minimum thickness of 4.5 feet. The building extends 33.5 feet below plant grade. Intermediate floors are reinforced concrete slabs with metal decking, supported by structural steel beams and girders, and reinforced concrete bearing walls. The building has a built-up roof, the top of which is 56 feet above grade, supported by structural steel beams and girders. The roof and intermediate floor framing are supported by structural steel columns and reinforced concrete bearing walls.

The drum storage area of the radwaste building is a single-story, structural steel building supported on a reinforced concrete mat foundation, the top of which is 6 inches above plant grade. This area is separated from the adjacent portion of the radwaste building by isolation joints and houses the radioactive waste drum handling facilities and storage areas. The drum storage areas are enclosed by reinforced concrete shield walls. The building has a built-up roof with a high point 31 feet above plant grade.

3.8.6.1.2 Radwaste Pipe Tunnel

The radwaste pipe tunnel is a below grade, reinforced concrete, two-cell box structure connecting the auxiliary building and the radwaste building. It is separated from both buildings by isolation joints. The bottom of the tunnel is 25.5 feet below plant grade, and the top is 8 feet below grade. The tunnel provides access and carries electrical cable trays and piping between the auxiliary building and the radwaste building.

3.8.6.2 Applicable Codes, Standards and Specifications

These structures are designed in accordance with the codes and standards listed in the following sections.

3.8.6.2.1 Codes

a. American Concrete Institute, Building Code Requirements for Reinforced Concrete (ACI 318-71).

b. American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th Edition, adopted February 12, 1969, and Supplement Nos. 1, 2, and 3 (see FSAR Table 3.2-1, Note 19).

c. American Institute of Steel Construction (AISC), Structural Joints Using ASTM A325 or A490 Bolts, May 8, 1974.

d. American Institute of Steel Construction (AISC), Code of Standard Practice for Steel Buildings and Bridges, October, 1972.

e. American Welding Society, Structural Welding Code (AWS D1.1-75) (see FSAR Table 3.2-1, Note 19).

f. International Conference of Building Officials, Uniform Building Code, 1973.

3.8.6.2.2 Standards and Specifications

Nationally recognized industry standards, such as those published by the ASTM, are used whenever possible to describe material properties, testing procedures, fabrication,

and construction methods. The applicable standards used are discussed in Section 3.8.3.6.

Structural specifications are prepared to cover the areas related to the design and construction of these structures. The specifications are prepared specifically for the SNUPPS project. They emphasize important points of the industry standards for these structures and reduce options such as would otherwise be permitted by the industry standards. The specifications cover the following areas:

a. Concrete material properties

b. Mixing, placing, and curing of concrete

c. Reinforcing steel and splices

d. Structural steel

e. Miscellaneous and embedded steel

f. Anchor bolts

g. Grating

3.8.6.3 Loads and Load Combinations

The radwaste building and tunnel are designed for the applicable loads and load combinations specified in the codes listed in Section 3.8.6.2.1.

3.8.6.4 Design and Analysis Procedures

3.8.6.4.1 Radwaste Building

The intermediate concrete floor slabs are designed for the combination of dead, live, and lateral loads, in accordance with ACI-318. The structural steel beams and girders are designed as composite sections, in accordance with the AISC manual.

The exterior reinforced concrete walls are designed as one-way or two-way slabs supported at the base slab, intermediate floors, roof, and transverse walls, as applicable. The loading combinations are given in ACI-318.

The base slab is designed as a slab on an elastic foundation for loads and load combinations given in ACI-318.

The seismic loads for the structure are obtained by the following procedures:

a. The input motion at the foundation of the radwaste building is defined by normalizing the Regulatory Guide 1.60 spectra to the OBE maximum ground acceleration of 0.12g, as outlined in Section 3.7(B).1.1. The damping values given in Table 3.7(B)-1 are used. These are consistent with the damping values recommended in Regulatory Guide 1.61.

   A simplified analysis is performed to determine appropriate seismic loads and floor response spectra pertinent to the location of the systems. The simplified analysis involves the modeling of the building by a several-degrees-of-freedom mathematical model and time-history analysis to generate the floor response spectra for radwaste systems and the seismic loads for the building. The design time-histories are defined in Section 3.7(B).1.2.

b. The simplified method for determination of seismic loads for the building consists of (1) calculation of modal frequencies and participation factors for the building, (2) determination of modal seismic loads by item a, input spectra, and (3) combination of modal seismic loads by the square-root-of-the-sum-of-the-squares (SRSS) rule. Only two orthogonal horizontal inputs need to be considered in two separate analyses, and the greater of the two results of the analyses is used for building design.

c. Time-history analysis is performed to generate floor response spectra. Item a, design time-histories, will be used as input.

d. The load factors and load combinations used for the building are those given in ACI-318. The allowable stresses for steel components are those given in the AISC Manual of Steel Construction.

e. The construction and inspection requirements for the building elements comply with those stipulated in the AISC or ACI Code, as appropriate (see FSAR Table 3.2-1, Note 19).

f. The foundation media of the radwaste building does not liquefy during the operating basis earthquake.

3.8.6.4.2 Radwaste Pipe Tunnel

The radwaste tunnel is analyzed as a rigid box in the transverse direction. Dynamic soil and hydro pressures are obtained in accordance with Section 2.5.4.10.3. Longitudinally it is designed as a beam on an elastic foundation. The tunnel is isolated from the radwaste and auxiliary buildings by isolation joints. The load factors and the loading combinations are given in ACI-318.

3.8.6.5 Structural Acceptance Criteria

These structures are designed for structural acceptance criteria defined in the codes listed in Section 3.8.6.2.1.

3.8.6.6 Materials, Quality Control, and Special Construction Techniques

The materials, quality control programs, and special construction techniques used in the fabrication and construction of these structures are described in the following sections.

3.8.6.6.1 Concrete

Structural concrete used in the construction of these structures has a minimum compressive strength, $f'_{c'}$ of 4,000 psi at 28 days. The concrete materials, mix design, examination, and placement are described in Section 3.8.1.6.1.

3.8.6.6.2 Reinforcing Steel and Splices

The reinforcing steel and splices used in the construction of these structures, including materials, examination, and erection tolerances, are described in Section 3.8.1.6.2.

3.8.6.6.3 Structural Steel

The structural steel used in the construction of these structures, including materials, examination, and erection, are described in Section 3.8.3.6.3.

3.8.6.6.4 Embedded Items

The embedded carbon steel items used in the construction of these structures, including materials, examination, and erection, are described in Section 3.8.3.6.4.

3.8.6.6.5 Quality Control

The quality control measures are discussed in Section 3.8.5.6.

3.8.6.6.6 Special Construction Techniques

These structures are constructed of concrete and steel, using proven methods common to heavy, industrial construction. No special, new, or unique construction techniques are used.

3.8.6.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance are not required for these structures. No formal program of testing and inservice surveillance is planned.

3.8.7 REFERENCES

1. Biggs, J.M, Introduction to Structural Dynamics, McGraw Hill, Inc., 1964.

TABLE 3.8-1 CONTROL TESTS FOR CONCRETE

Material	Requirements	Test Method	Minimum Frequency
Cement	Standard physical and chemical properties	ASTM C150	Each 1,200 tons
Fly ash and pozzolans	Chemical and physical properties in accordance with ASTM C618	ASTM C3111	Each 200 tons
Aggregate	Gradation	ASTM C136	Once per shift during production
	Moisture content	ASTM C566	Once per shift during production
	Material finer than #200 sieve	ASTM C117	Daily during production
	Organic impurities	ASTM C40	Once per shift during production
	Flat and elongated particles	CRD C-119*	Twice per month during production
	Friable particles	ASTM C142	Monthly during production
	Lightweight particles	ASTM C123	Monthly during production
	Soft fragments	ASTM C235	Monthly during production
	Specific gravity and absorption	ASTM C127 (coarse) ASTM C128 (fine)	Initially
	Los Angeles abrasion	ASTM C131	Every 6 months during production
	Potential reactivity	ASTM C289	Every 6 months during production
	Soundness	ASTM C88	Every 6 months during production
Water and Ice	Effect on compressive strength	AASHTO T-26	Every 6 months. If chemical data indicates that the water quality is unchanged, the tests may be waived by the owner.
	Setting time	AASHTO T-26	
	Soundness	AASHTO T-26	
	Total solids	AASHTO T-26	
	Chlorides	AASHTO T-26	
Admixtures	Chemical composition	Infrared spectrophotometry	Composite of each shipment
Concrete	Mixer uniformity	ASTM C94	Initially and every 6 months
	Sampling method	ASTM C172	
	Compression cylinders	ASTM C31	
	Compressive strength	ASTM C39	One set of 2 cylinders from each 100 cubic yards or a minimum of one set per day for each mix design, for each strength test.
	Slump	ASTM C143	First batch mixed each shift and every 50 cubic yards placed.
	Air content	ASTM C231	First batch mixed each shift and every 50 cubic yards placed.
	Temperature	-	First batch mixed each shift and every 50 cubic yards placed.
	Unit weight	ASTM C138	Every 100 cubic yards during production.

* Alternately, the project technical specifications provide for a procedure that may be used in lieu of the test method inidicated.

TABLE 3.8-2 MAXIMUM ALLOWABLE OFFSET IN FINAL WELDED JOINTS OF REACTOR BUILDING LINER PLATE

Section Thickness (in.)	Direction of Joints in Circumferential Shells Longitudinal	Circumferential
Up to 1/2 incl.	1/4 t	1/4 t
Over 1/2 to 3/4 incl.	1/8 in.	1/4 t
Over 3/4 to 1.5 incl.	1/8 in.	3/16 in.
Over 1.5	1/8 in.	1/8 t

"t" is the nominal thickness of the thinner section at the joint.

TABLE 3.8-3 STRESS LIMITS FOR STEEL PORTIONS OF CONCRETE CONTAINMENTS DESIGNED IN ACCORDANCE WITH SUBSECTION NE OF THE ASME CODE

Section 3.8.2.3.11 Combination No.		Primary Stresses Gen. Memb. P_m	Local Memb. P_L	Bend + Local Memb. $P_B + P_L$	Primary & Secondary Stresses	Peak Stresses	Buckling Note (3)
(1)		$.9S_y$	$1.25S_y$	$1.25S_y$	$3S_m$	Consider for fatigue analysis	125% of allow. given by NE-3133
(2) & (3)		S_m	$1.5S_m$	$1.5S_m$	$3S_m$	Consider for fatigue analysis	Allow. given by NE-3133
(4) & (5)		S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	Allow. given by NE-3133
(6) & (7)	Not integral and continuous	S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	Allow. given by NE-3133
	Integral and continuous	The greater of $1.2S_m$ or S_y	The greater of $1.8S_m$ or $1.5S_y$	The greater of $1.8S_m$ or $1.5S_y$	N/A	N/A	120% of allow. given by NE--3133
(8)	Not integral and continuous	The greater of $1.2S_m$ or S_y	The greater of $1.8S_m$ or $1.5S_y$	The greater of $1.8S_m$ of $1.5S_y$	N/A	N/A	120% of allow. given by NE-3133
	Integral and continuous	85% of stress intensity limits of Appendix F			N/A	N/A	85% of allow. given by F-1325 of App. F

NOTES: (1) Thermal stresses need not be considered in computing P_m, P_L, and P_B

(2) Thermal effects are considered in:

(a) Specifying stress intensity limits as a function of temperature.

(b) Analyzing effects of cyclic operation (NE-3222.4).

(3) If a detailed analysis considering ineleastic behavior is performed for checking instability (buckling), such an analysis should demonstrate that the applied stress is less than 50 percent of the critical buckling stress. Designs utilizing vertical stiffeners are permitted. The allowable axial compressive stress may be determined by considering the effects of circumferential stiffener spacing and the effects of water, if present.

TABLE 3.8-4 LOAD COMBINATIONS AND LOAD FACTORS FOR CATEGORY I CONCRETE STRUCTURES

A. Load Combinations For Service Load Conditions

a. Working Stress Design Method

(1) $S = D + L$

(2) $S = D + L + E$

(3) $S = D + L + W$

(1a) $1.3S = D + L + T_o + R_o$

(2a) $1.3S = D + L + T_o + R_o + E$

(3a) $1.3S = D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent are checked.

b. Strength Design Method

(1) $U = 1.4\ D + 1.7\ L$

(2) $U = 1.4\ D + 1.7\ L + 1.9\ E$

(3) $U = 1.4\ D + 1.7\ L + 1.7\ W$

(1b) $U = (0.75)\ (1.4\ D + 1.7\ L + 1.7\ T_o + 1.7\ R_o)$

(2b) $U = (0.75)\ (1.4\ D + 1.7\ L + 1.9\ E + 1.7\ T_o + 1.7\ R_o)$

(3b) $U = (0.75)\ (1.4\ D + 1.7\ L + 1.7\ W + 1.7\ T_o + 1.7\ R_o)$

Both cases of L having its full value or being completely absent are checked against the following combinations:

(2b') $U = 1.2\ D + 1.9\ E$

(3b') $U = 1.2\ D + 1.7\ W$

Where soil and/or hydrostatic pressures are present, in addition to all the above combinations where they have been included in L and D, respectively, the requirements of Sections 9.3.4 and 9.3.5 of ACI-318 are also satisfied.

TABLE 3.8-4 (Sheet 2)

B. Load Combinations For Factored Load Conditions

For extreme environmental, abnormal, abnormal/severe environmental and abnormal/extreme environmental conditions, respectively, the strength design method should be used, and the following load combinations are satisfied:

(4) $U = D + L + T_o + R_o + E'$

(5) $U = D + L + T_o + R_o + W_t$

(6) $U = D + L + T_a + R_a + 1.5\ P_a$

(7) $U = D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$

(8) $U = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

(9) $U = D + L + T_o + R_o + N$

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, are used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) are satisfied first without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these loads, however, local section strength capacities may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent are checked.

TABLE 3.8-5 LOAD COMBINATIONS AND LOAD FACTORS FOR CATEGORY I STEEL STRUCTURES

A. <u>Load Combinations For Service Load Conditions</u>

a. Working Stress Design Method

(1) $S = D + L$

(2) $S = D + L + E$

(3) $S = D + L + W$

(1a) $1.5S = D + L + T_o + R_o$

(2a) $1.5S = D + L + \ + T_o + R_o + E$

(3a) $1.5S = D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent are checked.

b. Plastic Design Method

(1) $Y = 1.7\ D + 1.7\ L$

(2) $Y = 1.7\ D + 1.7\ L + 1.7\ E$

(3) $Y = 1.7\ D + 1.7\ L + 1.7\ W$

(1b) $Y = 1.3\ (D + L + T_o + R_o)$

(2b) $Y = 1.3\ (D + L + E + T_o + R_o)$

(3b) $Y = 1.3\ (D + L + W + T_o + R_o)$

Both cases of L having its full value or being completely absent are checked.

B. <u>Load Combinations for Factored Load Conditions</u>

a. Working Stress Design Method

(4) $1.6\ S = D + L + T_o + R_o + E'$

(5) $1.6\ S + D + L + T_o + R_o + W_t$

TABLE 3.8-5 (Sheet 2)

(6) $1.6\ S = D + L + T_a + R_a + P_a$

(7) $1.6S^* = D + L + T_a + R_a + P_a + 1.0\ (Y_j+Y_r+Y_m) + E$

(8) $1.7s^* = D + L + T_a + R_a + P_a + 1.0\ (Y_j+Y_r+Y_m) + E'$

(9) $1.6S = D + L + T_o + R_o + N$

*For these two combinations, (7) and (8), in computing the required section strength, S, the plastic section modulus of steel shapes is used.

b. Plastic Design Method

(4) $.90\ Y = D + L + T_o + R_o + E'$

(5) $.90\ Y = D + L + T_o + R_o + W_t$

(6) $.90\ Y = D + L + T_a + R_a + 1.5\ P_a$

(7) $.90\ Y = D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_j+Y_r+Y_m) + 1.25\ E$

(8) $.90\ Y = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_j+Y_r+Y_m) + E'$

(9) $.90\ Y = D + L + T_o + R_o + N$

In combination B (a) and (b) above, thermal loads are neglected when it is shown that they are secondary and self-limiting in nature and where the material is ductile.

In combinations (6), (7) and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r and Y_m, including an appropriate dynamic load factor, are used unless a time-history analysis is performed to justify otherwise.

Combination (5), (7), and (8) are first satisfied without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these loads, however, local section strengths may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety-related system.

APPENDIX 3.8A - COMPUTER PROGRAMS USED FOR STRUCTURAL AND SEISMIC ANALYSES

3.8A.1 Computer Programs Used for Structural and Seismic Analyses by Bechtel Power Corporation

3.8A.1.1 Bechtel CE 201, Bechtel Structural Analysis Program - Post Processor (BSAP-POST)

3.8A.1.2 Bechtel CE 239, Hemispherical Dome Tendon Analysis (TENDON)

3.8A.1.3 Bechtel CE 309, Structural Engineering Systems Solver (STRESS)

3.8A.1.4 Bechtel CE 316, Finite Element Stress Analysis (FINEL)

3.8A.1.5 Bechtel CE 400, Concrete Column Design (PCACOL)

3.8A.1.6 Bechtel CE 639, Hemispherical Dome Tendon Analysis (STRESS)

3.8A.1.7 Bechtel CE 779, Structural Analysis Program (SAP)

3.8A.1.8 Bechtel CE 786, Ground Spectrum Raise

3.8A.1.9 Bechtel CE 798, Engineering Analysis System (ANSYS)

3.8A.1.10 Bechtel CE 800, Bechtel Structural Analysis Program (BSAP)

3.8A.1.11 Bechtel CE 801, Finite Element Stress Analysis (FINEL)

3.8A.1.12 Bechtel CE 802, Response Spectra Analysis (SPECTRA)

3.8A.1.13 Bechtel CE 803, Axisymmetric Shell and Solid Computer Program (ASHSD)

3.8A.1.14 Bechtel CE 901, The Structural Design Language (ICES STRUDL)

3.8A.1.15 Bechtel CE 915, A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites (SHAKE)

3.8A.1.16 Bechtel CE 917, Modal Dynamic Analysis

3.8A.1.17 Bechtel CE 918, Response Spectrum Analysis

3.8A.1.18 Bechtel CE 920, Time-History Analysis of Structures

3.8A.1.19 Bechtel CE 921, Response Spectrum Calculations

3.8A.1 COMPUTER PROGRAMS USED FOR STRUCTURAL AND SEISMIC ANALYSES BY BECHTEL POWER CORPORATION

Computer programs are continually updated under strict quality control procedures to enhance capabilities and to extend their applicability. As such, earlier versions of these programs, also verified, may have been used during earlier stages of the design effort.

3.8A.1.1 Bechtel CE 201 Bechtel Structural Analysis Program-Post Processor (BSAP-POST)

a. Description

BSAP-POST (CE 201) is a general-purpose, post-processor program for the BSAP (CE 800) finite-element analysis program. BSAP-POST can take the output from BSAP and display this data (graphically and/or on a line printer) or perform additional calculations. In addition, some of the capabilities of BSAP-POST can be used independently. For example, the concrete design module, OPTCON, can have design loads obtained from BSAP output or from punched cards.

BSAP-POST consists of a number of modules that can be used independently or sequentially to display or modify the contents of a data base under the control of an executive supervisor program. The data base consists of the contents of a file (TAPE 27) created by a BSAP analysis problem. The executive supervisor ensures that each module in BSAP-POST is compatible with every other module, and initiates the execution of each module when required by input data supplied by the user.

b. Validation

The BSAP-POST program has been prepared by Bechtel and has a complete set of documentation, including a users' manual, verification report, and theoretical manual. These documents are on file with Bechtel Data Processing.

c. Extent of Application

The program was used in the design of the reactor building and internals.

3.8A.1.2 Bechtel CE 239 Hemispherical Dome Tendon Analysis (TENDON)

a. Description

The dome tendon computer program calculates forces and pressures on a hemispherical dome of a prestressed, three-buttress concrete containment building, resulting from prestress by two orthogonal groups of vertical dome tendons and one group of horizontal hoop tendons. One group of vertical dome tendons is located in parallel, vertical planes normal to the x-axis*. The second group is located in vertical planes normal to the y-axis** . The third group is located in horizontal planes normal to the z-axis. Each of the vertical dome tendons (the first two groups) has equal areas and equal spacing measured along the springline***. The hoop dome tendons have equal areas, but the spacing may be either constant or may vary linearly with the latitude. The hoop tendons extend from the springline into the dome region up to 45 degrees latitude. Each hoop tendon is anchored at buttresses 240 degrees apart. Successive hoop tendons are anchored at alternate buttresses.

In the analysis, the dome is subdivided into a grid pattern specified by the user. The program calculates the total pressure due to tendon forces at each grid node in the radial direction, normal to the dome surface, and in the circumferential (hoop or azimuth) and meridional directions. Nodal forces in the hoop and meridional directions are calculated at each node point. The pressures and forces calculated by this program are intended for use as input to a finite element computer program to determine the stress distribution in the dome.

b. Validation

The TENDON program has a complete set of documentation, including a user's manual, verification report, and theoretical manual. These documents are on file with Bechtel Data Processing.

c. Extent of Application

The program was used as a verification of the program used in the design of the reactor building.

3.8A.1.3 Bechtel CE 309, Structural Engineering Systems Solver (STRESS)

a. Description

---

* Extending from 90 degrees to 180 degrees azimuth angle

** and extends from zero to 90 degree azimuth.

*** They are anchored at the base of the containment building.

STRESS is a programming system for the solution of structural engineering problems. The system is capable of executing the linear, elastic, and static analyses of 2- and 3-dimensional framed structures of the following types:

1. Plane truss
2. Plane frame
3. Plane grid
4. Space truss
5. Space frame

The programming system was originally developed at the Massachusetts Institute of Technology in 1964 and is now in the public domain.

b. Validation

The program has been verified by the ICES STRUDL II program. A sample problem of plane frame analysis was run, using the CE 309 program and the commercially available version (Version 2) of the ICES STRUDL II program. The results from these runs were found to be identical. Verification is on file with Bechtel Power Corporation.

c. Extent of Application

The program was used to perform structural analysis for steel structures.

d. Reference

Fenves, S. J., Logcher, R. D., and Mauch, S. P., Stress Reference Manual, M.I.T. Press, Cambridge, Mass., 1964.

3.8A.1.4 Bechtel CE 316, Finite Element Stress Analysis (FINEL)

a. Description

The program performs the static analyses of plane or axisymmetric structures, using the finite element method, in which a structure is idealized as an assemblage of finite elements. The finite elements are of either triangular or quadrilateral shape and connected at their corners (nodal points). The applied loads may be concentrated, uniformly distributed, or inertial, or may be temperature distributions. At boundaries, displacements may be forced.

The program develops the force-displacement relationship (element stiffness matrix) for each individual element from its geometry and material properties. The element relationships are then assembled into an overall structure force-displacement relationship (structure stiffness matrix). Equilibrium equations are developed for each degree of freedom at each nodal point in terms of the structure force-displacement relationship, the unknown nodal point displacement, and the externally applied nodal point forces. Finally, these equations are solved simultaneously for the unknown nodal point displacements by a modified Gaussian elimination scheme. Once the nodal point displacements are known, element stresses are calculated.

b. Assumptions

The stress and the strain are assumed to be constant within each element.

c. Validation

The program has been verified for use by comparison to the most recent version of FINEL, CE 801. The results from these runs were found to be essentially the same. Verification is on file with Bechtel Power Corporation.

d. Extent of Application

The program was used to compute stresses in the reactor building base slab, wall, and dome.

3.8A.1.5 Bechtel CE 400, Concrete Column Design (PCACOL)

a. Description

The program designs reinforced concrete compression members to resist a given combination of loadings and investigates the adequacy of a given cross section to resist a similar set of loadings. Each loading case consists of an axial compressive load combined with uniaxial or biaxial bending. The method of solution is based upon either ACI 318-71, Building Code Requirements for Reinforced Concrete, or AASHTO Standard Specifications for Highway Bridges.

b. Validation

The program was developed by the Portland Cement Association in 1974. The program is a recognized program and has had sufficient history of use to justify its applicability and validity without further demonstration. Program user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing.

c. Extent of Application

The program was used to investigate reinforced concrete compression members in the fuel building.

3.8A.1.6 Bechtel CE 639 Hemispherical Dome Tendon Analysis (STRESS)

a. Description

The dome tendon computer program calculates forces and pressures on a hemispherical dome of a prestressed, concrete containment building, resulting from prestress by two orthogonal groups of vertical dome tendons and one group of horizontal hoop tendons. One group of vertical dome tendons is located in parallel, vertical planes normal to the x-axis*. The second group is located in vertical planes normal to the y-axis**. The third group is located in horizontal planes normal to the z-axis. Each of the vertical dome tendons (the first two groups) has equal areas and equal spacing measured along the springline***. The hoop dome tendons have equal areas, but the spacing may be either constant or may vary linearly with the latitude. The hoop tendons extend from the springline into the dome region up to 45 degrees latitude.

In the analysis, the dome is subdivided into a grid pattern specified by the user. The program calculates the total pressure due to tendon forces at each grid node in the radial direction, normal to the dome surface, and in the circumferential (hoop or azimuth) and meridional directions. Nodal forces in the hoop and meridional directions are calculated at each node point. The pressures and forces calculated by this program are intended for use as input to a finite element computer program to determine the stress distribution in the dome.

b. Validation

The STRESS program was verified by comparison to CE 239 (TENDON). Verification is on file with Bechtel Power Corporation.

c. Extent of Application

The program was used in the design of the reactor building.

---

* Extending from 135 degrees to 225 degrees azimuth angle

** and extends from 45 to 135 degree azimuth.

*** They are anchored at the base of the containment building.

3.8A.1.7 Bechtel CE 779, Structural Analysis Program (SAP)

a. Description

The program performs the static and dynamic analyses of linear, elastic, three-dimensional structures, using the finite element method. The finite element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved for either applied loads or temperature distributions. Concentrated loads, pressures, or gravity loads may be applied. Temperature distributions are assigned as an appropriate uniform temperature change in each element. Prestressing may be simulated by using artificial temperature changes on rod elements.

Dynamic response routines are available for solving arbitrary dynamic loads or seismic excitations, using either modal superposition or direct integration. The program can also perform response spectrum and time-history analyses.

b. Validation

The solutions to test problems have been demonstrated to be essentially identical to the results obtained, using the BSAP program. Verification is on file with Bechtel Power Corporation.

c. Extent of Application

The program was used to perform structural analysis for concrete structures, such as the reactor cavity and secondary shield walls.

3.8A.1.8 Bechtel CE 786, Ground Spectrum Raise

a. Description

The program modifies a given ground, time-history accelerogram, such that its acceleration spectrum can be raised locally at any frequency by a desired amount. The principle is to superimpose to the original accelerogram a sinusoidal motion.

b. Validation

The program is verified by comparing the envelope of the modified accelerogram with an accepted ground response spectrum, such as that in NRC Regulatory Guide 1.60.

c. Extent of Application

The program was used to modify the Bechtel time-history accelerograms to comply with NRC Standard Review Plan Section 3.7.1.

3.8A.1.9 Bechtel CE 798, Engineering Analysis System (ANSYS)

a. Description

ANSYS is a large-scale, general purpose finite element computer program with applications to many classes of engineering problems. Structural analysis methods include static options for the solution of elastic, plastic, and nonlinear large and small deflection problems. Also, dynamic options are available to perform nonlinear transient, harmonic response and mode-frequency analysis. The finite element library is extensive and includes beam, spar, plate, shell, and nonlinear gap elements.

The matrix displacement method of finite element analysis is used in the formulation of the problem, and equations are solved by the wave front method.

b. Validation

The ANSYS program was licensed from Swanson Analysis Systems, Inc. (SASI), which has supplied a complete set of documentation including a user's manual, verification report, and theoretical manual. These documents are on file with Bechtel Data Processing.

c. Extent of Application

The program was used to perform a stress analysis of embedded base plates and for the access opening in the ‘C’ loop steam generator cubicle secondary shield wall.

3.8A.1.10 Bechtel CE 800, Bechtel Structural Analysis Program (BSAP)

a. Description

The program performs the static and dynamic analyses of linear, elastic, three-dimensional structures, using the finite element method. The finite element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved for either applied loads or temperature distributions. Concentrated loads, pressures, or gravity loads

may be applied. Temperature distributions are assigned as an appropriate uniform temperature change in each element. Prestressing may be simulated by using artificial temperature changes on rod elements.

Dynamic response routines are available for solving arbitrary dynamic loads or seismic excitations, using modal superposition. The program can also perform response spectrum and time-history analyses.

b. Validation

The solutions to test problems have been demonstrated to be essentially identical to the results obtained, using the following recognized public-domain computer programs:

- EASE - Elastic Analysis Corporation
- STARDYN - Mechanics Research Incorporated
- MARC/CDC - MARC Analysis Corporation
- ICES/STRUDL - McDonnell-Douglas Automation
- ASKA - Institut fur Statik and Dynamik, Stuttgart, Prof. A. J. Argyris

Agreement has also been established between BSAP program results and the results presented in the ASME Library of Benchmark Computer problems and solutions (Ref. 4) and in recognized technical journals. A complete set of documentation including a user's manual, verification report, and theoretical manual is on file with Bechtel Data Processing.

c. Extent of Application

The program was used to perform structural analysis for concrete structures and embedded plates.

d. References

1. Wilson, E. L., "SAP, A General Structural Analysis Program," University of California Structural Engineering Laboratory, Report No. UCSESM 70-20, September, 1970.

2. Wilson, E. L., "SOLID SAP - A Static Analysis Program for Three-Dimensional Solid Structures," University of California, Berkeley, Department of Civil Engineering, SESM Report No. 71-19, September, 1971.

3. Wilson, E. L., "SAP-IV-A Structural Analysis Program for Static and Dynamic Response of Linear Systems," University of California, Berkeley, EERC Report No. 73-11, June, 1973.

4. "Pressure Vessel and Piping - 1972 Computer Programs Verification," ASME Committee on Computer Technology, Pressure Vessel and Piping Division.

3.8A.1.11 Bechtel CE 801, Finite Element Stress Analysis (FINEL)

a. Description

The program performs the static analyses of plane or axisymmetric structures, using the finite element method, in which a structure is idealized as an assemblage of finite elements. The finite elements are of either triangular or quadrilateral shape and connected at their corners (nodal points). The applied loads may be concentrated, uniformly distributed, or inertial, or may be temperature distributions. At boundaries, displacements may be forced.

The program develops the force-displacement relationship (element stiffness matrix) for each individual element from its geometry and material properties. The element relationships are then assembled into an overall structure force-displacement relationship (structure stiffness matrix). Equilibrium equations are developed for each degree of freedom at each nodal point in terms of the structure force-displacement relationship, the unknown nodal point displacement, and the externally applied nodal point forces. Finally, these equations are solved simultaneously for the unknown nodal point displacements by a modified Gaussian elimination scheme. Once the nodal point displacements are known, element stresses are calculated.

b. Assumptions

The stress and the strain are assumed to be constant within each element.

c. Validation

The program has been verified by manual calculations. Document traceability is on file with Bechtel Data Processing.

d. Extent of Application

The program was used to compute stresses in the reactor building base slab, wall, and dome.

3.8A.1.12 Bechtel CE 802, Response Spectra Analysis (SPECTRA)

a. Description

The program computes the response spectra from an acceleration record digitized at equal time intervals. These spectra are plots of the maximum response of a simple oscillator over a range of values of its natural periods and dampings.

The numerical method for computing the spectral values is based on the exact analytical solution of the governing differential equation. It is assumed that the accelerogram varies linearly between the time-history points. The response spectra are constructed by monitoring of the maximum values of response parameters of each step of integration. The computed spectra are then widened to account for the effect of structural frequency variation.

b. Validation

The solutions of the program have been verified to be substantially identical with the closed formed analytical solutions of the three following test problems:

1. Undamped system with a triangular load pulse

2. Undamped system with a sinusoidal forcing function

3. Damped system with a sinusoidal forcing function

Program user's manual, verification report, and theoretical manual are on file with Bechtel Power Corporation.

c. Extent of Application

The program was used to develop floor response spectra curves for all seismic Category I structures.

3.8A.1.13 Bechtel CE 803, Axisymmetric Shell and Solid Computer Program (ASHSD)

a. Description

The program performs the static and dynamic analyses of linear, elastic, axisymmetric structures with axisymmetric or nonaxisymmetric loadings, utilizing the finite element technique. The program computes the element stresses and nodal displacements due to uniform, concentrated, or

pressure loads, or temperature distributions, either over the surface area or through the wall thickness. Prestress forces may be simulated by applying the forces as equivalent concentrated temperature gradients.

b. Validation

The solutions of the program for various loadings have been demonstrated to be essentially identical to the results obtained by manual calculations and to those obtained from accepted experimental tests of analytical results published in technical literature (Ref. 1 and 2). Program user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing.

c. Extent of Application

The program was used to analyze the reactor cavity.

d. References

1. Ghosh, S., Wilson, E. L., "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," Report No. EERC 69-10, University of California, Berkeley, September 1969, pp 69-81.

2. "Topical Report on Dynamic Analysis of Reactor Vessel Internals under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 Mwe Class Reactors," Combustion Engineering Report CENPD-42, Combustion Engineering, Inc., Nuclear Power Department, Combustion Division, Windsor, Conn. Appendix A.

3.8A.1.14 Bechtel CE 901, The Structural Design Language (ICES STRUDL)

a. Description

STRUDL is a structural analysis program with the capability to perform frame analysis and finite element analysis. A wide variety of loads may be accommodated by the program. The program also is capable of performing dynamic analysis as well as static analysis. The STRUDL program performs both steel and concrete design and checks the applicable code in each case.

b. Assumptions

The program assumes a linear, elastic, static, small displacement analysis, member properties are required, and the program treats the joint displacements as unknowns.

c. Validation

Program user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing.

d. Extent of Application

The program was used in the structural analysis of seismic cable tray and duct supports, miscellaneous frame structures, reactor cavity shielding platform, reactor vessel supports, and pressurizer compartment.

3.8A.1.15 Bechtel CE 915, A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites (SHAKE)

a. Description

The program computes the responses in a system of homogeneous, viscoelastic layers of infinite horizontal extent subjected to vertically traveling shear waves. The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent linear soil properties, using an iterative procedure to obtain values for modulus and damping compatible with the effective strains in each layer. The program handles systems with variation in both moduli and damping and takes into account the effect of the elastic base.

b. Validation

The program was developed as Report No. EERC 72-12 at the College of Engineering, University of California, Berkeley, California, by P. B. Schnabel, J. Lysmer, and H. B. Seed.

c. Extent of Application

The program was used to increase the time step of a Bechtel time-history accelerogram from 0.005 sec. to 0.01 sec.

3.8A.1.16 Bechtel CE 917, Modal Dynamic Analysis

a. Description

The program computes the reduced stiffness matrix from the basic geometry input for plane frame or truss models, or accepts the reduced

stiffness matrix for any structure as input. It calculates mode shapes, frequencies, participation factors, and modal damping values for a lumped mass model.

Special Features:

1. Can accept either diagonal or full mass matrices.

2. Generates output tape for input to Bechtel CE 920 and Bechtel CE 933.

3. Can be used for horizontal or vertical earthquakes with minimal input changes.

b. Validation

Current version of program user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing. Prior version verification is on file with Bechtel Power Corporation.

c. Extent of Application

The program was used to obtain the fixed-base mode shapes and natural frequencies of seismic Category I structures and cable tray supports.

3.8A.1.17 Bechtel CE 918, Response Spectrum Analysis

a. Description

This program is supplemental to the modal dynamic analysis program (Bechtel CE 917). It computes the modal response of general plane frame or truss models. Response spectrum technique is used, and output is expressed in terms of displacements, accelerations, support reactions, member forces and moments, and spring forces.

b. Validation

Current version of program user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing. Prior version verification is on file with Bechtel Power Corporation.

c. Extent of Application

The program was used to calculate fixed-base responses of structure acceleration, shear, moment, displacement, etc.

3.8A.1.18 Bechtel CE 920, Time-History Analysis of Structures

a. Description

The program performs the earthquake response time-history analysis of lumped mass models, using modal superposition. Program input consists of frequencies, mode shapes, modal damping, and the base acceleration time-history.

b. Validation

Program user's manual, verification report and theoretical manual are on file with Bechtel Data Processing.

c. Extent of Application

The program was used to generate the time-histories for the radwaste building.

3.8A.1.19 Bechtel CE 921, Response Spectrum Calculations

a. Description

The program calculates response acceleration, velocity, and displacement spectra for a specified acceleration time-history. It can produce printed plots of the calculated response spectra.

b. Validation

Program user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing.

c. Extent of Application

The program was used to generate acceleration, velocity, and displacement spectra at the radwaste building equipment locations and to print plots of these response spectra.

3.8A.1.20 Bechtel CE 933, Fourier Analysis of Soils (FASS)

a. Description

The program calculates the seismic time-history response of a soil-structure interaction system using (1) input from Bechtel CE 917, (2) the foundation impedance approach, and (3) the frequency domain analysis method. Both horizontal and vertical interaction analyses can be

performed, using this program. Because the foundation impedances are frequency dependent, a rigorous seismic response analysis of the soil structure interaction system cannot directly apply the standard time domain analysis procedure, such as the modal superposition method or the direct integration method. Consequently, the program adopts the frequency domain analysis procedure and uses the Fourier transform method for the response calculation.

b. Validation

The program's user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing.

c. Extent of Application

The program was used in computation of seismic deflections for seismic Category I structures.

3.8A.1.21 Bechtel CE 935, Earthquake Acceleration Time-Histories

a. Description

Refer to BC-TOP-4-A, Rev. 3.

b. Validation

Refer to BC-TOP-4-A, Rev. 3.

c. Extent of Application

The data file was used in seismic analysis of all seismic Category I structures.

3.8A.1.22 Bechtel CE 970, Impedance Functions for a Rigid Circular Foundation on a Layered Viscoelastic Medium (LUCON)

a. Description

LUCON is a program developed to evaluate the impedance functions for a rigid circular (or equivalent circular) foundation placed on a layered viscoelastic medium. The program computes the vertical, rocking, and horizontal impedance functions and their reciprocals, the compliance functions, for any given set of frequencies with site characteristics and the foundation geometry. The foundation medium may be layered or may be a uniform elastic half-space. The two types of material damping in the soil are constant hysteretic-type damping and Voigt-type damping. The type of

damping must be the same for all layers, but the values of the damping constants may differ from layer to layer.

b. Validation

The solutions of the program for various loadings have been demonstrated to be essentially identical to analytical results published in technical literature (Ref. 1 through 5). Program user's manual, verification report, and theoretical manual are on file with Bechtel Data Processing.

c. Extent of Application

The program was used to compute the impedance functions for all seismic Category I structures for use in seismic deflection analyses of the structures.

d. References

1. Veletsos, A. S., and Verbic, B., "Vibration of Viscoelastic Foundations," Report No. 18, Dept. of Civil Engineering, Rice University, Houston, Texas, April 1973.

2. Shah, P. M., "On the Dynamic Response of Foundation System," Ph.D. Thesis, Rice University, Houston, Texas, 1968.

3. Veletsos, A. S., and Wei, Y. T., "Lateral and Rocking Vibration of Footings," Journal of the Soil Mechanics and Foundations Division, ASCE, Vol. 97, 1971.

4. Luco, J. E., and Westmann, R. A., "Dynamic Response of Circular Footings," Journal ofthe Engineering Mechanics Division, ASCE, Vol. 97, 1971.

5. Luco, J. E., "Impedance Functions for a Rigid Foundation on a Layered Medium," Nuclear Engineering and Design, 1974.

3.8A.1.23 Computer Programs for Seismic Soil-Structure Interaction Analysis

3.8A.1.23.1 Bechtel CE 988 (FLUSH)

a. Description

The program uses finite element techniques to analyze soil-structure interaction effects during earthquakes, especially for embedded structures. The program provides consideration of variations of ground motion with

depth in the soil-structure response evaluations. Some of the special features of the program include:

1. Plain strain quadrilateral elements for modeling of soils and structures

2. Beam elements for modeling of structures

3. Multiple nonlinear soil properties for equivalent linear analysis

4. An approximate 3-dimensional ability, making it possible to perform meaningful structure-soil-structure interaction analyses

5. Generates output time-histories of acceleration and bending moments

6. Computation of maximum moments, shear forces, and axial forces in beam elements

7. Generates acceleration and velocity response spectra

b. Validation

The program was developed as Report No. EERC 75-30 at the College of Engineering, University of California, Berkeley, California, by J. Lysmer, T. Udaka, C. F. Tsai, and H. B. Seed.

c. Extent of Application

The program was used to seismically analyze all seismic Category I structures.

3.8A.1.23.2 FLUSH (Control Data Corp. Version)

a. Description

The description of the FLUSH program contained in Section 3.8A.1.23.1 applies to CDC's version of the program. Enhancements made by CDC to the original version of the program, which was developed at the University of California at Berkeley, have led to reduced execution costs and made the program more convenient to use.

b. Validation

Verification of CDC's version of FLUSH has been performed and appropriate documentation, as defined by Control Data Corp. policy, is maintained by CDC's Utilities Service Center.

c. Extent of Application

The program was used to seismically analyze seismic Category I structures.

3.8A.1.24 DISCOM, a FLUSH Postprocessor (Control Data Corp. Version)

a. Description

DISCOM postprocesses optional output files from the FLUSH program (Control Data Corp. version, see Section 3.8A.1.23.2) to provide relative displacements between points in a FLUSH model.

b. Validation

The program was developed by the Utilities Service Center of the Control Data Corp. Verification of the program was performed and appropriate documentation maintained by the Utilities Service Center under Control Data Corporate policy.

c. Extent of Application

The program was used to obtain the relative seismic displacements within and between seismic Category I structures.

3.8A.1.25 The Structural Design Language (ICES-STRUDL, McDonnell-Douglas Automation Version)

a. Description

The program performs structural analysis. Frame members can be used in conjunction with finite elements. Some special features include a built-in table for rolled steel wide flange shapes, a member selection procedure based upon the AISC Code, a reinforced concrete member design and checking capability, and a dynamic analysis capability.

b. Validation

The program has been verified, and document traceability is available at McDonnell-Douglas Automation.

c. Extent of Application

The program was used to perform structural analysis for the reactor building instrument tunnel and reactor cavity shielding platform.

3.8A.1.26 Other Computer Programs Used in Structural Analysis

In the course of structural design calculations, several programs of limited scope were developed to assist the designers in lengthy, repetitious calculations. The programs were validated by example problems or manual design checks. These validations are incorporated into the project design calculation books. These programs are not itemized here due to their simplicity and nature of use.

3.8A.2 COMPUTER PROGRAMS USED FOR STRUCTURAL ANALYSES BY SUPPLIERS

3.8A.2.1 INRYCO, Nuclear Force Computation (NUCFOR)

a. Description

The program computes post-tensioning force of tendons used in nuclear vessels and prepares the field stressing cards for individual tendons. Final effective forces along the tendon are computed at both ends and at points where the curve of the tendon changes. The program calculates the theoretical elongation at each stressing end. Only circular curve and straight lines are considered by the program. The program handles dome, hoop, and vertical tendons.

b. Validation

The program has been verified, and document traceability is available at INRYCO, Incorporated.

c. Extent of Application

The program was used to compute the post-tensioning force of tendons in the reactor building and to prepare the field stressing cards for individual tendons.

3.8A.2.2 CBI Program 7-81, Shells of Revolution

a. Description

The program calculates the stresses and displacements in thin-walled elastic shells of revolution when subjected to static edge, surface, and/or temperature loads with arbitrary distribution over the surface of the shell.

The geometry of the shell must be symmetric, but the shape of the median is arbitrary. It is possible to include up to three branch shells with the main shell in a single model. In addition, the shell wall may consist of four layers of different orthotropic materials, and the thickness of each layer and the elastic properties of each layer may vary along the median.

b. Validation

The program has been verified, and document traceability is available at Chicago Bridge & Iron Company.

c. Extent of Application

The program was used for design of ASME Class MC portions of the reactor building.

d. Reference

Kalnins, A., "Analysis of Shells of Revolution Subjected to Symmetrical and Nonsymmetrical Loads," Journal of Applied Mechanics, 1964.

3.8A.2.3 CBI Program 1027, Stress Intensities at Loaded Attachments for Spheres or Cylinders with Round or Square Attachment

a. Description

The program calculates the stress intensities in a sphere or cylinder at a maximum of 12 points around an externally loaded round or square attachment. Stresses resulting from external loads are superimposed on an initial pressure stress situation. The program computes stresses at three levels of plate thicknesses: outside, inside, and centerline of plate. The program determines the following three components for each stress intensity:

1. $\sigma\chi$ = a normal stress parallel to the vessel's longitudinal axis

2. $\sigma\phi$ = a normal stress in a circumferential direction

3. τ = a shear stress

The program has an option, whereby the penetration load will be considered reversible or nonreversible in a direction. Under the reversible option, only the data associated with the most severe loading situation is printed.

Most of the analysis and notation used in the program is taken directly from the "Welding Research Council (WRC) Bulletin #107" of December 1968, and the program contains extrapolations of the curves for cylinders in WRC 107 for γ up to 570.

b. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. Document traceability is available at Chicago Bridge & Iron Company.

c. Extent of Application

The program was used for design of ASME Class MC portions of the reactor building.

3.8A.2.4 CBI Program 1691

a. Description

The program analyzes two- or three-dimensional frames or trusses for member end forces, and moments, joint deflections, and rotations. An analysis can be made on structures with rigid, hinged, or free support conditions, rigid, hinged, or free support conditions, rigid or hinged member end conditions, and any number of loading conditions. Included in the program is a provision to use rectangular or cylindrical coordinates to describe the structure and a plotting option for a geometry check. The program can combine several loading conditions and can analyze the structure for member deadloads when the unit weight of the material deadloads when the unit weight of the material has been input.

b. Validation

The program has been verified, and document traceability is available at Chicago Bridge & Iron Company.

c. Extent of Application

The program was used for design of ASME Class MC portions of the reactor building.

3.8A.2.5 STAADIII/ISDS

a. Description

The program performs finite element static analysis of steel structures. The program can review any number of load cases and will calculate Natural Frequency values. The program is supplied by Research Engineers, Inc.

b. Validation

The program has been verified for safety related application by Union Electric Co. The documentation of the verification can be found in UE's file for software verification specifications.

c. Extent of Application

The program is used for structural analysis related to plant modifications at Callaway.

3.8A.2.6 ALGOR

a. Description

The program performs both dynamic and static analysis of structures as well as local finite element analysis. The program system is comprised of several sub programs which act together to generate a final analysis. The program has the ability to perform dynamic modal, response spectra as well as static analysis. The program is supplied by ALGOR Interactive Systems, Inc.

b. Validation - See 3.8A.2.5(b)

c. Extent of Application - See 3.8A.2.5(c)

3.9(B) MECHANICAL SYSTEMS AND COMPONENTS

3.9(B).1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9(B).1.1 Design Transients

Refer to Section 3.9(N).1.1 for a description of the operating conditions considered in the design of the RCS, RCS component supports, and reactor internals. Class 1 piping systems are designed and analyzed using design transients that are compatible with those described in Section 3.9(N).1.1.

Class 2 and 3 piping systems and components do not require thermal transient analysis. Class 2 and 3 piping systems and components are designed and analyzed for dynamic transients, as listed in Section 3.9(B).2.

3.9(B).1.2 Computer Programs Used in Analyses

For NSS systems, refer to Section 3.9(N).1.2.

3.9(B).1.2.1 Seismic Category I Items Other Than the NSSS

Table 3.9(B)-1 lists computer programs used in the balance-of-plant system components. The verification of programs is as follows:

3.9(B).1.2.1.1 ME-632 Program

The ME-632 program is used to determine stresses and loads due to thermal expansion, deadweight, earthquake, and transient force functions such as those created by fast relief valve opening and closing, pipe break, or fast activation of high-capacity pumps (water hammer effects).

The results obtained from pipe stress program ME-632 have been compared with a) ASME Benchmark problem results, b) Pipe Stress Program TPIPE, c) general purpose program ANSYS, and d) long-hand calculations. The comparison of the results are given in the verification report of the ME-632 program (Ref. 3).

A description of this computer code is included in Table 3.9(B)-1.

Appendix 3.9(B)A provides a verification report for the ME-632 program.

3.9(B).1.2.1.2 ME-101, SUPERPIPE, and TPIPE Programs

The ME-101, SUPERPIPE, and TPIPE computer programs are used to determine stresses and loads due to restrained thermal expansion, deadweight, dynamic, seismic anchor movement, and earthquake in the following piping:

a. Seismic Category I ASME Section III Class 1, 2, and 3 piping 2 1/2 inches and larger.

b. Seismic Category I ASME Section III Class 1, 2, and 3 piping 2 inches and smaller that cannot be analyzed per M-18.

c. ANSI B31.1 Power Piping Included in High Energy Piping Systems.

A description of these programs is included in Table 3.9(B)-1.

Computer Code ME-632 is a predecessor of ME-101 (Ref. 1) and incorporates compliance with NRC Regulatory Guide 1.92. The purpose of the programs is basically identical. ME-101 results have been compared against the results from ME-632, and the results of the hand calculations (13 test problems in all) and the values agree within 2 percent. The verification report is on file at Bechtel. TPIPE was developed by PMB Systems Engineering, San Francisco, Calif. for TVA. It has been verified using PIPSOL (EDS Nuclear, Inc.) and ME-632.

A synthesis of closely spaced modes is provided based on equation (4) of Regulatory Guide 1.92.

3.9(B).1.2.1.3 ANSYS Program

The ANSYS program is a general purpose computer program for the solution of several classes of engineering problems. It is used in the detailed analysis of the main steam and feedwater torsional restraints.

A description of this computer code is included in Table 3.9(B)-1.

The ANSYS has been developed and verified by Swanson Analysis Systems, Inc.

3.9(B).1.2.1.4 ME-602 Program

The ME-602 program performs the analysis of seismic Category I ASME Section III Class 2 and 3 piping 2 inches and smaller.

A description of this computer code is included in Table 3.9(B)-1.

ME-602 is based on the theory and equations of BP-TOP-1 (Ref. 2), a report on the seismic analysis of piping systems, written by the Bechtel Power Corporation, San Francisco, Calif. ME-602 programs the equations of BP-TOP-1. All NRC concerns relative to this approach to seismic analysis have been addressed and are noted in Appendices E and G of BP-TOP-1. Verification is presented in Appendix D of the report.

3.9(B).1.2.1.5 ME-210 Program

ME-210 computes the local stresses in cylindrical shells that result from external loadings. It is used in pipe support design to calculate the local stresses in piping produced by welded stanchions or lugs.

The program is based on Welding Research Council Bulletin 107, August 1965. The program has been verified based upon hand calculations.

3.9(B).1.2.1.6 CE901 ICES/STRUDL-II

The ICES/STRUDL-II code is used in the design of component supports. For ASME Section III Class 1 piping support design, the program is used to obtain stiffness properties of the support. The results of the analyses are incorporated into overall reactor vessel internal models which calculate the dynamic response due to seismic and LOCA conditions and yield dynamic stresses. In the design of ASME Section III Class 2 and 3 piping supports, models of certain indeterminate support designs are programmed in order to obtain support loads and stresses.

A description and validation of this program are included in Section 3.8A.1.14 of Appendix 3.8A.

3.9(B).1.2.1.7 CE800 (BSAP), CE802 (SPECTRA), and CE786

These programs were used to determine the seismic response spectra of the NSSS for reactor coolant loop branch piping analysis, stresses, and displacements of the main feedwater and main steam system in the reactor building, and to determine seismic anchor movements of the NSSS for incorporation into the piping analysis.

A description and validation of these programs are included in Sections 3.8A.1.10, 3.8A.1.12, and 3.8A.1.8 of Appendix 3.8A.

3.9(B).1.3 Experimental Stress Analysis

3.9(B).1.3.1 NSS System

Refer to Section 3.9(N).1.3.

3.9(B).1.3.2 Seismic Category I Items Other Than the NSSS

Experimental stress analysis methods are not used in the design of Code or non-Code components for the faulted condition. For code components, the stresses will not exceed the limits of the ASME B and PV Code, Section III.

3.9(B).1.4 Considerations for the Evaluation of the Faulted Condition

A listing of all seismic Category I safety-related mechanical systems and components is included in Table 3.2-1.

3.9(B).1.4.1 Seismic Category I Items in the NSSS

Refer to Section 3.9(N).1.4.

3.9(B).1.4.2 Seismic Category I Items Other Than the NSSS

For statically applied loads, the stress allowables of Appendix F of ASME Section III are used for Code components. For non-Code components, allowables are based on tests or accepted standards consistent with those in the 1974 edition of Appendix F of ASME III.

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-1322 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 material standards consistent with those in Appendix F for linear elastically analyzed components.

In those cases where component stresses exceed yield, an elastic-inelastic time history analysis is performed, using the ANSYS computer program, described in 3.9(B).1.2.1.3. This analysis is based on a bilinear stress-strain curve of a particular material type and the maximum allowable strain limit is maintained at a very low percentage of the material breaking strain.

Analysis concerning the rupture of high-energy piping is addressed in Section 3.6.

3.9(B).2 DYNAMIC TESTING AND ANALYSIS

3.9(B).2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

A vibration operational test program 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 will be performed.

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 are operated in their normal mode (emergency mode for auxiliary feedwater turbine pump), and measurements are recorded on the systems during and following the event that causes the transient induced vibrations. The systems and the associated transients to be included in the preoperational test program to verify the piping system are:

1. Main steam

   (a) Main steam turbine stop valve trip*

   (b) Main steam atmospheric dump valves opening

   (c) Main steam condenser dump valves opening

2. Pressurizer power-operated relief valve piping

   (a) Relief valve operation

3. Auxiliary turbine system

   (a) Auxiliary feedwater pump turbine stop valve trip

   Selected snubbers on pressurizer power-operated relief valve piping subjected to transients are instrumented during preoperational testing to assure proper snubber operation.

   All of the above are upset transients, and 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 Table 3.9(B)-2.

b. Steady state vibration

System vibration resulting from flow disturbances falls into this category. Positive displacement pumps may cause such flow variation and vibrations

* Main steam turbine stop valve trip transient test to be performed during power ascension.

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 visual steady state vibration inspection is made during system operation. Measurements are recorded where any one of the below listed conditions exist:

Frequency $\leq$10 Hz

For safety-related systems $\geq$0.125 inches (peak-to-peak)

For nonsafety-related systems $\geq$0.25 inches (peak-to-peak)

Safety-related systems, including associated instrumentation, and high-energy systems,* except the reactor coolant loop and pressurizer surge line, will be monitored for steady-state vibration for all modes of system operation encountered during the preoperational test program defined in FSAR Chapter 14.0.

The acceptance criterion is that the maximum measured amplitude shall not induce a stress in the piping system greater than one-half the endurance limit (which corresponds to 10^6 cycles), as defined in Section III of the ASME Boiler and Pressure Vessel Code, 1974.

When required, additional restraints are provided to reduce the stresses to below the acceptance criterion levels.

During the thermal expansion test, pipe deflections will be recorded at selected locations. The system will also be visually monitored for hanger and snubber performance and for piping interferences with structure or other piping. One complete thermal cycle, i.e., cold position to hot position to cold position, will be monitored.

Selected portions of the following systems will be monitored during their normal mode of operation.

Main steam system

Main feedwater system

Letdown/charging system

---

* High-energy systems as defined in Regulatory Guide 1.68 are high-energy piping systems inside Seismic Category I structures, and high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level.

Residual heat removal system

Containment spray system*

Emergency core cooling system

Auxiliary feedwater system

Auxiliary turbine system

Steam generator blowdown system

More specific information concerning the locations where visual inspection or measurements are to be taken are addressed in the applicable test procedures. Acceptable criteria for the thermal and dynamic tests are addressed in the applicable FSAR Chapter 14 test abstracts.

Corrective action for a major deficiency identified as a result of the test program will be reported to NRC. Retesting will be performed in accordance with administrative control identified in Chapter 14.

3.9(B).2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

3.9(B).2.2.1 Safety-Related Equipment in the NSSS

Refer to Section 3.9(N).2.2.

3.9(B).2.2.2 Safety-Related Mechanical Equipment Other Than the NSSS

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 will be used if structural integrity alone can assure the design-intended function. Examples of such equipment which falls into this category includes:

Piping

* Design characteristics of the containment spray system do not permit actual testing to monitor thermal expansion of the suction piping from the containment sumps, during the recirculation mode. Verification of this piping will be attained by its similarity to the RHR suction lines from the RCS hot leg which will be monitored.

Ductwork

Tanks and vessels

Heat exchangers

Filters

Inactive valves

The seismic analysis of piping is described in Section 3.7(B).

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 Section 3.9(B).3.2.2.1. The procedure discussed therein applied, with some variations, to other items in this category.

b. Dynamic testing

   Dynamic testing is used for components which contain mechanisms which must change position or maintain position in order to perform their design-intended function and which, because of their complexity, do not lend themselves to analysis. Such components include valve extended top works and similar appurtenances for other mechanical equipment.

c. Combinations of analysis with testing

   A combination of analysis, static testing, and dynamic testing is used for seismic qualification of families of active valves. Individual valves within these families may be qualified using past test data and analysis of the new attributes.

   The seismic qualification of active valves is discussed more fully in Section 3.9(B).3.2.2.2.

d. The acceptance criteria are as follows:

   1. Tests, when used, demonstrate that the component is not prevented from performing its design-intended function during and after the test.

2. Analysis, when used for qualification of vessels, pumps, piping, or valves, verifies that stresses do not exceed the allowables specified in Tables 3.9(B)-5 through 3.9(B)-9 for the seismic conditions shown in Table 3.9(B)-2 and that deformations do not exceed those which will permit the component to perform its design-intended function.

3.9(B).2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady State Conditions

Refer to Section 3.9(N).2.3.

3.9(B).2.4 Preoperational Flow Induced Vibration Testing of Reactor Internals

Refer to Section 3.9(N).2.4.

3.9(B).2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Condition

Refer to Section 3.9(N).2.5.

3.9(B).2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Refer to Section 3.9(N).2.6.

3.9(B).3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

3.9(B).3.1 Loading Combinations, Design Transients, and Stress Limits

3.9(B).3.1.1 ASME Section III Class 2 and 3 Constructed Items Furnished with the NSSS

Refer to Section 3.9(N).3.1.

3.9(B).3.1.2 ASME Section III Constructed Items Not Furnished with the NSSS

The combinations of design loadings categorized with respect to plant operating conditions identified as Normal, Upset, Emergency, and Faulted which are specified for the design of ASME Code constructed items are presented in Table 3.9(B)-2. The design stress limits of the ASME Code are selected to ensure the integrity of safety equipment. The ASME Code requirements are supplemented by additional requirements in Regulatory Guide 1.48. The corresponding stress limits for each category of plant operating condition which are specified for each type of ASME Code constructed item are presented in Tables 3.9(B)-5 through 3.9(B)-9. The specified

component operating condition is the same as the plant operating condition for each transient event, except where pump, system, or valve function must be assured during an emergency or faulted condition in which case appropriate stress limits are used to provide proof that functional capability has been maintained.

The system or subsystem analysis used to establish or confirm loads specified for the design of components and supports was performed on an elastic basis. There are no deformation criteria associated with the design loading combinations, and plastic instability allowable limits given in ASME Section III are not used when dynamic analysis is performed. The limit analysis methods have the limits established by ASME Section III for the normal, upset, and emergency conditions. For these cases, the limits are sufficiently low to assure that the elastic system analysis is not invalidated. Stress limits for faulted loading conditions are discussed in Section 3.9(B).1.4. These faulted condition limits are established in such a manner that there is equivalence with the adopted elastic limits and consequently will not invalidate the elastic system analysis. Elastic stress analysis methods were also used in the design calculations to evaluate the effects of the loads on the components and supports.

Dynamic analysis, as described in Section 3.9(B).2, is performed to verify that the stresses are within the limits specified by the applicable code requirements.

The recommendations of Regulatory Guide 1.48 applicable to the design limits and loading combinations for seismic Category I fluid system components are met as discussed in Table 3.9(B)-13.

3.9(B).3.2 Pump and Valve Operability Assurance

3.9(B).3.2.1 Active ASME Section III Class 1, 2, and 3 Pumps and Valves Furnished with the NSSS

Refer to Section 3.9(N).3.2.

3.9(B).3.2.2 Active ASME Section III Class 2 and 3 Pumps and Class 1, 2, and 3 Valves Not Furnished With the NSSS

3.9(B).3.2.2.1 Pumps

Active pumps not furnished with the NSSS are identified in Table 3.9(B)-15. These pumps are subjected to stringent tests both prior to and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 150 percent of the design pressure, and (2) performance tests which are conducted while the pump is operated with flow to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor properties. Where appropriate, bearing temperatures and vibration levels are also monitored during these operating tests. Refer to Table 3.9(B)-15. After the pump is installed at the plant, it undergoes startup tests and required inservice inspection and operation.

In addition to these tests, the active pumps are qualified for operation during and after a faulted condition. That is, safety-related active pumps are qualified for operability during an SSE condition by assuring that (1) the pump will not be damaged during the seismic event and (2) the pump will continue operating despite the SSE loads.

The pump manufacturer is required to show by analysis, correlated by tests, prototype tests, or existing documented data, that the pump will perform its safety function when subjected to loads imposed by the maximum seismic accelerations and the maximum faulted nozzle loads. It is required that test or dynamic analysis be used to determine the lowest natural frequency of the pump. The pump, when having a natural frequency above 33 Hz, is considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 3.0g horizontal and 2.0g vertical, acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances.

In order to avoid damage to the pumps during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the limits specified in Tables 3.9(B)-8 and 3.9(B)-9. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

If the lowest natural frequency is found to be below 33 Hertz, the equipment is considered flexible. If flexible, the equipment is analyzed using the response spectrum modal analysis technique. The frequencies and mode shapes are determined in the vertical and horizontal directions. The loads due to the excitation of each mode and the loads due to the accelerations in the three orthogonal directions are added, using the SRSS method. Coupling effects shall be included in the mathematical model. The stress limits stated in Tables 3.9(B)-8 and 3.9(B)-9 must be satisfied. Performance of these analyses, based upon conservative loads and restrictive stress limits, assures that the critical parts of the pump will not be damaged during the faulted condition and, therefore, that the reliability of the pump for post-faulted condition operation will not be impaired by the seismic events.

The second criterion necessary to assure operability is that the pump will function throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., with no rotation.

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. If the testing option is chosen, sine-beat testing for the electrical equipment is justified by satisfying one or more of the following requirements to demonstrate that multifrequency response is negligible or the sine-beat input is of sufficient magnitude to conservatively account for this effect.

a. The equipment response is basically due to one mode.

b. The sine-beat response spectra envelop the floor response spectra in the region of significant response.

c. The floor response spectra consist of one dominant mode and have a narrow peak at this frequency.

The degree of coupling in the equipment, in general, determines if a single or multiaxis test is required. Multiaxis testing is required if there is considerable cross-coupling. If coupling is very light, then single axis testing is justified. Or, if the degree of coupling can be determined, then single axis testing is used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

From this regimen, it is concluded that the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings and, therefore, will perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

The functional ability of active pumps after a faulted condition is assured, since only operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted operating loads will be limited to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted ability of the pumps to function under these applied loads is proved during the normal operating plant conditions for active pumps.

3.9(B).3.2.2.2 Valves

The active valves are tabulated in Table 3.9(B)-16. Refer to the specifications listed in Table 3.10(B)-1 for the tests and analyses used to ensure proper seismic qualification.

Safety-related active valves are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, and are subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test in accordance with ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and functional tests which verify that the valve will open and close within the specified time limits. The operability qualification of power operators for the environmental conditions over the installed life is in accordance with IEEE 323 and IEEE 382. After installation, cold hydrostatic qualification tests, hot functional qualification tests, and required periodic inservice operations are performed to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant.

For all active valves with extended top works, an analysis is also performed for static equivalent SSE loads applied at the center of gravity of the extended structure to demonstrate structural integrity. The stress limits allowed in the analyses demonstrate structural integrity and are equal to the limits recommended by the ASME for the particular ASME class of valve analyzed. These limits for each of the loading combinations given in Table 3.9(B)-2 are presented in Table 3.9(B)-6. Operating capabilities are demonstrated by one of the methods listed below.

Method A - Combination of Analysis, Static Load Test, and Dynamic Test

This method is permitted only when the valve assembly has a first natural frequency of vibration greater than 33 Hz.

a. An analysis or test is performed to show that the valve assembly has a first natural frequency of vibration greater than 33 Hz.

b. While in the shop and installed in a suitable test rig, the valve is subjected to a static equivalent seismic load applied at the center of gravity of the operator in the direction of the weakest axis of the yoke. The design pressure of the valve is applied to the valve during the static load tests.

   The valve is then operated with equivalent seismic static load applied. The valve must perform its safety-related function within the specified time limits.

   The static load for this test is the equivalent of 4.5g's horizontal and 4.5g's vertical. The plant piping is supported in such a manner that the power operator accelerations are maintained below test levels.

   Step (b) may be omitted if it can be proven through analysis that functional operability is satisfied with all applicable design loads present. To permit this analysis the valve must be amenable to the analysis performed.

c. Prior to installation, power operators and other appurtenances are qualified in accordance with IEEE 344, Seismic Qualification Standards.

Method B - Dynamic Testing of Complete Valve Assembly

a. The valve unit is mounted on a test fixture in a manner which is representative of typical valve installations. The valve unit includes the operator and all appurtenances normally attached to the valve in service.

b. The valve is subjected to a dynamic test per IEEE 344. Details of this testing are contained in Table 3.10(B)-1 of Section 3.10(B).

c. The valve is pressurized to its design pressure and cycles during the dynamic test. The valve unit must perform its safety-related function within the specified time limits.

Following testing by method A or B, the valve unit is tested for seat leakage. The leakage rate must be less than the allowable leakage rate specified by the valve design specification.

The above testing program applies only to power-operated valves. The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types, e.g., motor-operated gate valve and air-operated globe valve, are tested. Valve sizes which cover the range of sizes in service are qualified by the tests, and the results are used to qualify all valves within the intermediate range of sizes. Stress analyses are used to support the interpolation.

Due to the particularly simple characteristics of check valves and other compact valves, they are not affected by seismic acceleration. Check valves have no extended structures that would distort the valve and cause a malfunction. Check valve discs are designed to allow sufficient clearance around the disc to prevent distortions due to nozzle or other imposed loads. They are qualified by a combination of the following tests and analysis:

a. Stress analysis of critical areas and parts for SSE loads in accordance with the ASME Code Case 1635-1

b. In-shop hydrostatic test

c. In-shop seat leakage test

d. Periodic valve exercise and inspection to assure the functional ability of the valve

Using the methods described, all the safety-related active valves in the systems are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and assure that the active valves will perform their safety-related function when necessary.

3.9(B).3.3 Design and Installation Details for Mounting of Pressure Relief Devices

The design of pressure relieving devices can be generally grouped in two categories--open discharge and closed discharge.

3.9(B).3.3.1 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:

a. 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 will be considered. These stresses are calculated in accordance with the applicable subsections of Section III of the ASME Code. These stresses are combined as shown in Table 3.9(B)-2, and compared to appropriate allowable stresses.

b. Thrust forces will include both pressure and momentum effects.

c. Where more than one safety or relief valve is installed on the same run pipe, valve spacing is as specified in ASME Code Case 1569.

d. Where more than one safety or relief valve is installed on the same run pipe, the sequence of valve openings which induce the maximum stresses is considered as required by Regulatory Guide 1.67.

e. The minimum moments to be used in stress calculations are those specified in ASME Code Case 1569.

f. The effects of the valve discharge on piping connected to the valve header are considered.

g. 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 dynamic time-history analysis. A dynamic load factor of 2.0, as required by Regulatory Guide 1.67, is used when a system is analyzed by static methods.

3.9(B).3.3.2 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. If required, water slug effects are also included.

3.9(B).3.3.3 Operational Qualification for Active Safety-Relief Valves

Active safety-relief valves are subjected to the following shop tests, hydrostatic, seat leak tests, and a static loading equivalent to the SSE applied at the top of the bonnet and pressure at the valve inlet increased until the valve mechanism actuates. Periodic in situ valve inspection is performed to assure the functional ability of the valves.

During a seismic event, it is anticipated that the seismic accelerations imposed upon the valve may cause it to open momentarily and discharge under system conditions which otherwise would not result in valve opening. This is of no real safety or other consequence.

3.9(B).3.4 Component Supports

3.9(B).3.4.1 Supports Furnished with the NSSS

Refer to Section 3.9(N).3.4.

3.9(B).3.4.2 Supports Not Furnished with the NSSS

The loadings, as specified in the Design Specifications, are taken into account in designing component supports for ASME Code constructed items. These loadings include but are not limited to the following.

a. Weight of the component and normal contents under operating and test conditions

b. Weight of the component support

c. Superimposed loads and reactions induced by the adjacent system components

d. Dynamic loads, including loads caused by earthquake vibration

e. Restrained thermal expansion

f. Anchor and support movement effects

The combinations of loadings categorized with respect to plant operating conditions identified as Normal, Upset, Emergency, and Faulted which are specified for the design of supports for ASME Code constructed items are presented in Table 3.9(B)-10. The stress limits which are specified for each plant operating condition are specified in Tables 3.9(B)-11 and 3.9(B)-12.

All ASME Section III, Class 2 and 3, supports are designed as welded attachments to embedded or surface-mounted plates. Bolting for plates is designed according to AISC

allowables with increases allowed by the loading cases identfied in FSAR Table 3.8-5. In no case do the tensile stresses in bolts exceed the yield stress of the bolting material at temperature.

3.9(B).3.4.2.1 Snubbers Used as Component Supports

The location and size of the snubbers are determined by stress analysis. The stress analysis uses the computer program mentioned in Section 3.9(B).1 and the loading combination given in Table 3.9(B)-10. 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 B & PV Code, Section III, Subsection NF standards.

The design specification requires consideration of the following:

a. The mechanical snubber is considered a Class 1 linear support. Design is in accordance with Subarticle NF-3200 of Section III.

b. A Certified Stress Report is furnished, showing the load capabilities of the snubber. Verification of the load carrying capability of the snubber is in accordance with NF-3132 of Section III.

c. The service loading of the snubber is equal to or less than the design strength established under listing b. above for the particular loading condition.

d. The frictional resistance due to normal thermal movement does not exceed 1 percent of the design, normal, and upset load rating of the snubber, as defined in NF-3231.1 or NF-3262.3, or 5 pounds, whichever is greater.

e. The peak-to-peak displacement across the unit, excluding end attachments, does not exceed 0.12 inch when subjected to cyclic loading in the frequency range of 3 to 33 Hertz.

f. The snubber is designed for normal operation within a temperature range of -20 to +300°F, and is capable of providing normal performance when exposed to an abnormal environmental temperature of 350°F for a period not longer than 12 hours.

g. All lubricants and other nonmetallic component parts are capable of withstanding the effects of an integrated neutron and gamma ray radiation dose of 3×10^9 rads without detriment to their physical properties.

h. Suppressor span is adjustable over a range of $\pm$ 3-1/2 inches from the designed length without changing the operating position of the unit.

i. The design, procurement, manufacture, inspection, handling, testing, storage, and shipping of units and their component parts are performed in accordance with the Quality Assurance Program and the vendor's standard quality assurance procedures.

The design specification requires that an installation manual be provided by the manufacturer to ensure correct installation, including dimensional detailed drawings giving materials of construction with installation and adjustment instruction. Visual confirmation and inspection are required in the field.

Also, the hot and cold position of the snubbers will be measured during the preoperational testing stage.

There are no formal provisions for accessibility for inspection, testing, and repair or replacement of snubbers. Snubbers are located in order to most efficiently minimize stresses in the components and piping. However, access will be provided for inspection, testing, repair, or replacement by removing obstructions, if necessary.

All non-NSSS snubbers are of the mechanical type. The fabricator of the mechanical non-NSSS snubbers is the Pacific Scientific Company. The function of the mechanical snubber is for shock arrest.

Two types of tests are performed on the snubber.

a. Production tests are made on every unit.

    1. Check unit to confirm acceleration level is less than specified maximum.

    2. Check unit to confirm that it operates freely over the total stroke.

    3. Measure and record the force required to initiate motion over the stroke in tension and compression.

    4. Measure and record lost motion of the snubber mechanism.

b. Qualification tests are performed on randomly selected production models. These tests are used to demonstrate the required load performance (load rating). These tests include dynamic load cycling, low temperature, high temperature, humidity, salt spray, sand, dust, life test, and faulted load test.

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. As

only mechanical snubbers are used, there is no impact on the performance of the snubber by entrapped air or temperature on fluid properties.

The recommendations of Regulatory Guide 1.124 applicable to the service limits and loading combinations for Class I linear supports, are met, as discussed in Table 3.9(B)-14.

3.9(B).4 CONTROL ROD DRIVE SYSTEMS

Refer to Section 3.9(N).4.

3.9(B).5 REACTOR PRESSURE VESSEL INTERNALS

Refer to Section 3.9(N).5.

3.9(B).6 INSERVICE TESTING OF PUMPS AND VALVES

Inservice testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves is performed in accordance with ASME OM Code as required by 10 CFR 50, Section 50.55a(f), except where specific written relief has been granted by the NRC, by 10 CFR 50, Section 50.55 a(f)(6)(i).

3.9(B).6.1 Inservice Testing of Pumps

The pump test program lists all safety-related Class 1, 2, and 3 pumps that are provided with an emergency power source and are necessary to safely shut down the plant or mitigate the consequences of an accident. The pump test program is in accordance with ASME OM Code, Subsection ISTB, pursuant to 10 CFR 50.55a. The hydraulic and mechanical test parameters to be measured or observed will be discussed and defined in the Inservice Testing Program.

3.9(B).6.2 Inservice Testing of Valves

The valve test program will list all safety-related (i.e., those valves necessary to safely shut down the plant 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 OM Code, Subsection ISTC, pursuant to 10 CFR 50.55a. Test parameters to be measured or observed will be defined in the Inservice Testing Program.

3.9(B).7 REFERENCES

1. "Program ME-101 and ME-632 Seismic Analysis of Piping Systems, Users Manual," Pacific International Computing Corp., March, 1971.

2. BP-TOP-1, Seismic Analysis of Piping Systems, Bechtel Power Corporation, San Francisco, California, Rev. 3, January, 1976.

3. “Seismic Analysis of Piping Systems Program ME-632 Verification Report," Version B10, Bechtel Power Corporation.

TABLE 3.9(B)-1 COMPUTER PROGRAMS USED IN ANALYSIS

Program Name	Purpose	Description	Classification
ME-101 ME-632	Used to calculate the stresses and loads in piping systems due to restrained thermal expansion, deadweight, seismic anchor movements, and earthquake	ME-101 and ME-632 analyze piping systems in compliance with ANSI and ASME piping codes. Using the stiffness method of finite element analysis, the displacements of the joints of a given structure are considered basic unknowns. The dynamic analysis by the modal synthesis method utilizes known maximum accelerations produced in a single degree of freedom model of certain frequency. Principal program assumptions are: a. It is a linearly elastic structure. b. Simultaneous displacement of all supports is described by a single time-dependent function. c. Lumped mass model satisfactorily replaces the structure. d. Modal synthesis is applicable. e. Rotational inertias of the masses have negligible effect.	Bechtel Power Corp. Proprietary

TABLE 3.9(B)-1 (Sheet 2)

Program Name	Purpose	Description	Classification
PIPSYSW	Used in the analysis and stress evaluation of piping systems and the determination of support loads.	Linear, three-dimensional finite element procedures are used to perform static dynamic analyses of systems modeled by beam elements. It was validated for use by Sargent & Lundy, LLC. Used for the replacement of existing buried ESW carbon steel pipe with polyethylene and stainless steel pipe.	Sargent & Lundy, LLC
WATPRO	Used for piping stress analysis involving welded attachments.	The evaluations are performed using ASME Section III Code Cases N-122-2, N-391-2, N-318-5, and N-392-3. These Code Cases are accepted for use by the NRC in Regulatory Guide 1.84. It was validated for use by Sargent & Lundy, LLC. Used for the replacement of existing buried ESW carbon steel pipe with polyethylene and stainless steel pipe.	Sargent & Lundy, LLC
ANCHOR	Used for piping stress analysis involving anchor attachments.	The evaluations are performed using ASME Section III, Code Case N-392. This Code case has been superseded by Code Case N-392-3, which is acceptable for use by the NRC in Regulatory Guide 1.84. The two versions of the Code case are technically equivalent as applied in the ANCHOR program. ANCHOR was validated for use by Sargent & Lundy, LLC. Used for the replacement of existing buried ESW carbon steel pipe with polyethylene and stainless steel pipe.	Sargent & Lundy, LLC

TABLE 3.9(B)-1 (Sheet 3)

Program Name	Purpose	Description	Classification
ANSYS	General static, thermal, and dynamic analysis for linear elastic and plastic analysis	ANSYS is a general purpose program for solving a wide variety of engineering analysis problems more efficiently than most special purpose programs. ANSYS includes capabilities for transient heat transfer analyses, including conduction, convection, and radiation; structural analyses, including static elastic, plastic, creep, dynamic and dynamic plastic analyses, and large deflection and stability analyses; and one-dimensional fluid flow analyses. The output from the transient heat transfer analysis is in the form required for thermal analyses at selected time points in the transient with the same analytical model.	Public domain - Bechtel Vendor.
ME-602	Used to calculate seismic spans, support reactions, and stresses for small-diameter piping	Performs a conservative seismic analysis by dividing piping systems into a series of spans limited by guides (two mutually perpendicular restraints normal to the pipe) at all concentrated masses (e.g., valves) at all extended masses and at maximum spacing on straight runs of piping. The length of span is determined by dynamic calculations based on a modified spectrum curve. The spectrum curve is modified for a particular building elevation so that the flexible side of the peak of the curve will remain constant at the peak spectral acceleration for decreasing frequencies.	Bechtel Power Corp. Proprietary.
ME-210	Computes local stresses in piping due to external loads	Incorporates the theory and equations of Welding Research Council Bulletin 107, August, 1965	Bechtel Power Corp. Proprietary.

TABLE 3.9(B)-1 (Sheet 4)

Program Name	Purpose	Description	Classification
ICES/STRUDL	See Appendix 3.8A.		
CE-800 (BSAP)	See Appendix 3.8A.		
CE-802 (SPECTRA)	See Appendix 3.8A.		
CE-786	See Appendix 3.8A.		
TPIPE	Used to calculate the stresses and loads in piping systems due to earthquake	See ME-101 and ME-632	PMB Systems Engineering, Inc.
CAEPIPE	Used to calculate the stresses and loads in piping systems due to static loading conditions	Program is a finite element based program, verified for use by Union Electric Co. See the UE verification specifications for documentation.	SST Systems, Inc.
PS-CAEPIPE	See TPIPE	See CAEPIPE	SST Systems, Inc.
STAADIII/IDS	See Appendix 3.8A.		
ALGOR	See Appendix 3.8A.		
PIPESTRESS	See ME-101	See ME-101 and PS-CAEPIPE	DST Computer Services S.A..

TABLE 3.9(B)-2 DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3 COMPONENTS

Condition	Design Loading Combinations[1,2]
Design	PD
Normal	PO + DW + NL
Upset	(a) PO + DW + OBE + NL
	(b) PO + DW + RVC + NL
	(c) PO + DW + FV + NL
	(d) PO + DW + OBE + RVO + NL
	(e) PO + DW + DU + NL
Emergency[3]	(a) PO + DW + DE + NL
Faulted[3]	(a) PO + DW + SSE + RVO + NL
	(b) PO + DW + SSE + NL
	(c) PO + DW + DF + NL

LEGEND:
- PD - Design pressure
- PO - Operating pressure
- DW - Piping deadweight
- OBE - Operating basis earthquake (inertia portion)
- SSE - Safe shutdown earthquake (inertia portion)
- FV - Fast valve closure
- RVC - Relief valve - closed system (transient)
- RVO - Relief valve - open system (sustained)
- DU - Other transient dynamic events associated with the upset plant condition
- DE - Dynamic events defined as emergency condition
- DF - Dynamic events associated with a LOCA during which or following which the piping system being evaluated must remain intact

TABLE 3.9(B)-2 (Sheet 2)

NL - Equipment nozzle loads

NOTES:

1. As required by the appropriate subsection, i.e., NC, ND, or NF, of ASME Section III Division 1, other loads, such as thermal transient, thermal gradients, and anchor point displacement portion of the OBE, may require additional consideration in addition to those primary stress-producing loads listed.

2. For components other than piping, appropriate nozzle loads associated with the particular plant operating conditions are also included.

3. If active valve function must be assured during emergency/faulted conditions, this requirement is included in the design specification and the specified emergency/faulted condition for the plant is considered as the normal condition for the valve or the valve operability is demonstrated.

TABLE 3.9(B)-3 DELETED

TABLE 3.9(B)-4 DELETED

TABLE 3.9(B)-5 STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 VESSELS

Condition	Stress Limits
Design and normal	The vessel shall conform to the requirements of ASME Section VIII, Division 1.
Upset	$\sigma_m \leq 1.1S$
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$
Emergency	$\sigma_m \leq 1.5S$
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80S$
Faulted	$\sigma_m \leq 2.0S$
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4S$

LEGEND:

σ_m = General membrane stress. This stress is equal to the average stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

σ_L = Local membrane stress. This stress is the same as σ_m, except that it includes the effect of discontinuities.

σ_b = Bending stress. This stress is equal to the linear varying portion of the stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

S = Allowable stress value given in Tables I-7.1, I-7.2, and I-7.3 of Appendix I of the ASME Section III Code. The allowable stress shall correspond to the highest metal temperature at the section under consideration during the condition under consideration.

The term "stress" in the above definitions means the maximum normal stress.

TABLE 3.9(B)-6 STRESS CRITERIA FOR ASME CODE CLASS 1, 2 AND 3 VALVES (ACTIVE AND INACTIVE)

Condition	Stress Limits [(1-5)]	P_{max} [(6)]
Design and normal	Valve bodies shall conform to the requirements of ASME Section III, NC-3500 (or ND-3500)	
Upset	$\sigma_m \leq 1.1S$	1.1
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$	
Emergency	$\sigma_m \leq 1.5S$	1.2
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80S$	
Faulted	$\sigma_m \leq 2.0S$	1.5
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4S$	

NOTES:

1. Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation: (1) section modulus and area of every plane, normal to the flow, through the region of valve body crotch is at least 10 percent greater than those for the piping connected (or joined) to the valve body inlet and outlet nozzles; and, (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of connected piping, the valve section modulus and area as calculated in (1) above shall be multiplied by the ratio of S_{pipe}/S_{valve}. If unable to comply with this requirement, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

5. These rules do not apply to Class 2 and 3 safety and relief valves. Safety relief valves will be designed in accordance with ASME Section III requirements.

TABLE 3.9(B)-6 (Sheet 2)

6. The maximum pressure resulting from upset, emergency, or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.

7. Definition of symbols used in this table are given in Table 3.9(B)-5.

TABLE 3.9(B)-7 DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 PIPING

Condition	Stress Limits
Normal, upset, and emergency	The piping shall conform to the requirements of Section III, Paragraphs NC-3600 and ND-3600.
Faulted	The piping shall conform to the requirements of Section III, Paragraphs NC-3600 and ND-3600. The sum of stress due to internal pressure, live and dead loads, and those due to occasional loads identified in the Design Specification as acting during a faulted event will not exceed 2.4 times the allowable stress S_h.

TABLE 3.9(B)-8 STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS

Condition	Stress Limits	P_{max}*
Design and normal	The pump shall conform to the requirements of ASME Section III, NC-3400 (or ND-3400)	
Upset	$\sigma_m \leq 1.1S$	
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$	1.1
Emergency	$\sigma_m \leq 1.5S$	
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80S$	1.2
Faulted	$\sigma_m \leq 2.0S$	
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4S$	1.5

LEGEND:

σ_m = General membrane stress. This stress is equal to the average stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

σ_L = Local membrane stress. This stress is the same as σ_m, except that it includes the effect of discontinuities.

σ_b = Bending stress. This stress is equal to the linear varying portion of the stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

S = Allowable stress value given in Tables I-7.1, I-7.2, and I-7.3 of Appendix I of Section III of the Code. The allowable stress shall correspond to the highest metal temperature at the section under consideration during the condition under consideration.

The term "stress" in the above definitions means the maximum normal stress.

NOTE:

* The maximum pressure shall not exceed the tabulated factors listed under "P_{max}" times the design pressure.

TABLE 3.9(B)-9 STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 ACTIVE PUMPS

Condition	Design Criteria	P_{max}*
Normal	ASME Section III, Subsections NC-3400 and ND-3400	
Upset	$\sigma_m \leq 1.0S$	1.1
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5S$	
Emergency	$\sigma_m \leq 1.1S$	1.2
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$	
Faulted	$\sigma_m \leq 1.2S$	1.5
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.8S$	

LEGEND:

Definition of symbols used in this table are given in Table 3.9(B)-8.

NOTE:

* See Note 1, Table 3.9(B)-8.

TABLE 3.9(B)-10 DESIGN LOADING COMBINATIONS FOR SUPPORTS FOR ASME CODE CLASS 1, 2, AND 3 COMPONENTS

Condition		Design Loading Combinations
Normal		DW + TH
Upset	(a)	DW + OBE + SAM + TH
	(b)	DW + RVC + TH
	(c)	DW + FV + TH
	(d)	DW + OBE + RVO + SAM + TH
	(e)	DW + DU + TH
Emergency	(a)	DW + DE + TH
Faulted	(a)	DW + SSE + RVO + SAM + TH
	(b)	DW + SSE + SAM + TH
	(c)	DW + DF + TH

LEGEND:

TH = Thermal

DW = Piping deadweight

OBE = Operating basis earthquake (inertia portion)

SSE = Safe shutdown earthquake (inertia portion)

FV = Fast valve closure

RVC = Relief valve-closed system (transient)

RVO = Relief valve-open system (sustained)

DU = Other transient dynamic events associated with the upset plant condition

DE = Dynamic events defined as emergency condition

DF = Dynamic events defined as a faulted condition

SAM = Anchor displacement of OBE

TABLE 3.9(B)-11 ALLOWABLE STRESS LIMITS FOR CLASS 1 COMPONENT SUPPORTS

Support Type	Design	Normal	Conditions Upset	Emergency	Faulted
Plate and shell design by analysis	NF-3221	NF-3222	NF-3223	NF-3224	NF-3225
Linear type supports by analysis	NF-3231	NF-3231	NF-3231	NF-3231	NF-3231
Component standard supports design by analysis	NF-3240	NF-3240	NF-3240	NF-3240	NF-3240
Component supports design by load rating	NF-3260	NF-3260	NF-3260	NF-3260	NF-3260

NOTE:

Paragraph numbers refer to ASME Code, Section III 1974, Subsection NF, including Winter 1974.

TABLE 3.9(B)-12 ALLOWABLE STRESS LIMITS FOR CLASS 2 AND 3 COMPONENT SUPPORTS

Support Type	Design	Normal	Conditions Upset	Emergency	Faulted
Plate and shell design by analysis	NF-3321	NF-3321	NF-3321	$\sigma_1 \leq 1.2S$	$\sigma_1 \leq$ the lesser of 1.5S or $0.4S_u$
				$\sigma_1 + \sigma_2 \leq 1.8S$	$\sigma_1 + \sigma_2 \leq$ the lesser of 2.25S or $0.6S_u$
Linear	NF-3231	NF-3231	NF-3231	NF-3231	NF-3231
Component standard supports design by analysis	NF-3221 or NF-3231	NF-3222 or NF-3231	NF-3223 or NF-3231	NF-3224 or NF-3231	NF-3225 or NF-3231
Component supports design by load rating	NF-3260	NF-3260	NF-3260	NF-3260	NF-3260

LEGEND:

σ_1 and σ_2 are defined in NF-3321.1

S_u = Minimum ultimate tensile strength of material, from Table I-12.1

S = Minimum yield strength of material, from Table I-2.1

NOTES:

Paragraph numbers refer to ASME Code, Section III 1974, Subsection NF, including Winter 1974 addendum.

TABLE 3.9(B)-13 RESPONSE TO REGULATORY GUIDE 1.48 FOR COMPONENTS NOT FURNISHED WITH THE NSSS

Regulatory Guide 1.48 Position	Union Electric Position
Seismic Category I fluid system components should be designed to withstand the following loading combinations within the design limits specified.	
1. ASME Code[2] Class 1 vessels and piping:	N/A
a. The design limits specified in NB-3223 and NB-3654 of the ASME Code for vessels and piping, respectively, should not be exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition[3] and the vibratory motion of 50 percent of the Safe Shutdown Earthquake (SSE).	
b. The design limits specified in NB-3224 and NB-3655 of the ASME Code for vessels and piping, respectively, should not be exceeded when the component is subjected to loadings associated with the emergency plant condition.	N/A
c. The design limits specified in NB-3225 and NB-3656 of the ASME Code for vessels and piping, respectively, should not be exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	N/A
2. Non-active ASME Code Class 1 pumps and valves[4] that are designed by analysis:	

TABLE 3.9(B)-13 (Sheet 2)

Regulatory Guide 1.48 Position	Union Electric Position: Valves	Union Electric Position: Pumps
a. The design limits specified in NB-3223[5] of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.	Complies.	N/A
b. The design limits specified in NB-3224 of the ASME Code should not be exceeded when the component is subjected to loadings associated with the emergency plant condition.	Complies.	N/A
c. The design limits specified in NB-3225 of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Complies.	N/A
3. Non-active ASME Code Class 1 valves that are designed by standard or alternative design rules:		
a. The primary-pressure rating P_r should not be exceeded by more than 10 percent when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.	Complies.	

TABLE 3.9(B)-13 (Sheet 3)

Regulatory Guide 1.48 Position	Union Electric Position	
	Valves	Pumps
b. P_r should not be exceeded by more than 20 percent when the component is subjected to the loadings associated with the emergency plant condition.	Complies.	
c. P_r should not be exceeded by more than 50 percent when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Complies.	
4. Active ASME Code Class 1 pumps and valves[4] that are designed by analysis:		
a. The design limits[6] specified in NB-3222[5,7,8] of the ASME Code should not be exceeded when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE, or (2) loadings associated with the emergency plant condition, or (3) concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Complies.	N/A

TABLE 3.9(B)-13 (Sheet 4)

Regulatory Guide 1.48 Position	Union Electric Position
5. Active ASME Code Class 1 valves that are designed by standard or alternative design rules:	Complies with 5.a.(1) and 5.a.(2). Deviates from 5.a.(3) in that P_r should not be exceeded by more than 50 percent when the component is subjected to either concurrent loading associated with the normal plant conditions, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.
a. The primary-pressure rating P_r[6] should not be exceeded when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE, or (2) loadings associated with the emergency plant condition, or (3) concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	
6. ASME Code Class 2 and 3 vessels designed to Division 1 of Section VIII of the ASME Code:	
a. The allowable stress value S[9] should not be exceeded by more than 10 percent when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE, or (2) loadings associated with the emergency plant condition.	Complies.
b. S should not be exceeded by more than 50 percent when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Complies.
7. ASME Code Class 2 vessels designed to Division 2 of Section VIII of the ASME Code:	

TABLE 3.9(B)-13 (Sheet 5)

Regulatory Guide 1.48 Position	Union Electric Position	
a. The design limits specified in NB-3223 of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.	Complies.	
	Valves	Pumps
b. The design limits specified in NB-3224 of the ASME Code should not be exceeded when the component is subjected to loadings associated with the emergency plant condition.	Complies.	
c. The design limits specified in NB-3225 of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Complies.	
8. ASME Code Class 2 and 3 piping:		
a. The design limits specified in NC-3611.1(b)(4)(c)(b)(1) of the ASME Code should not be exceeded when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition and the vibratory motion of 50 percent of the SSE, or (2)[10] loadings associated with the emergency plant condition.	Complies.	
b. The design limits specified in NC-3611.1(b)(4)(c)(b)(2) of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Deviates as described. in FSAR Table 3.9(B)-7.	

TABLE 3.9(B)-13 (Sheet 6)

Regulatory Guide 1.48 Position	Union Electric Position
9. Non-active ASME Code Class 2 and 3 pumps: a. The primary membrane stress should not be exceeded by more than 10 percent of the allowable stress value S, and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 65 percent of S when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE, or (2) loadings associated with the emergency plant condition.	Complies with 9.a.(1). Deviates from 9.a.(2) in that primary membrane stress should not be exceeded by more than 50 percent of the allowable stress value and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 80 percent of S when subjected to emergency loads.
b. The primary membrane stress should not be exceeded by more than 20 percent of S, and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 80 percent of S when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	The primary membrane should not be exceeded by more than 100 percent of S, and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 140 percent of S when subjected to to these loads.

TABLE 3.9(B)-13 (Sheet 7)

Regulatory Guide 1.48 Position	Union Electric Position
10. Active ASME Code Class 2 and 3 pumps:	
a. The primary membrane stress[11] should not exceed the allowable stress value S, and the sum of the primary membrane and the primary bending stresses[11] should not be exceeded by more than 50 percent of S when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE, or (2) loadings associated with the emergency plant condition, or (3) concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Complies with 10.a.(1). Deviates from 10.a.(2) in that the primary membrane stress should not exceed the allowable stress value by more than 10 percent and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 65 percent of S when subjected to emergency loads. Deviates from 10.a.(3) in that the primary membrane stress should not exceed the allowable stress value by more than 20 percent and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 80 percent of S when subjected to faulted loads.
11. Non-active ASME Code Class 2 and 3 valves:	
a. The primary-pressure rating P_r should not be exceeded by more than 10 percent when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE, or (2) loadings associated with the emergency plant condition.	Complies with 11.a.(1). Deviates from 11.a.(2) in that P_r should not be exceeded by more than 20 percent when subjected to emergency loads.
b. P_r should not be exceeded by more than 20 percent when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Deviates in that P_r should not be exceeded by more than 50 percent when subjected to faulted loads.
12. Active ASME Code Class 2 and 3 valves:	

TABLE 3.9(B)-13 (Sheet 8)

Regulatory Guide 1.48 Position	Union Electric Position
a. The primary-pressure rating P_r[11] should not be exceeded when the component is subjected to either (1) concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE, or (2) loadings associated with the emergency plant condition, or (3) concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the faulted plant condition.	Same response as for 5.a.

NOTES:

[1] Applies to all components (vessels, piping, pumps, and valves) that are relied upon to cope with the effects of specified plant conditions.

[2] Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, including the 1972 Winter Addenda thereto.

[3] Identification of the specific transients or events to be considered under each plant condition will be addressed in a future regulatory guide.

[4] The requirements of the Case 1552 (Interpretations of ASME Boiler and Pressure Vessel Code) should be met for all sizes of Code Class 1 valves designed by analysis.

[5] The provisions of NB-3411 and NB-3413 may be applied for all sizes of Code Class 1 pumps designed by analysis.

[6] In addition to compliance with the design limits specified, assurance of operability under all design loading combinations should be provided by an appropriate combination of the following suggested measures:

TABLE 3.9(B)-13 (Sheet 9)

NOTES (Cont.):

a. In situ testing (e.g., preoperational testing after the component is installed in the plant).

b. Full-scale prototype testing.

c. Reduced-scale prototype testing.

d. Detailed stress and deformation analyses (includes experimental stress and deformation analyses).

In the performance of tests or analyses to demonstrate operability, the structural interaction of the entire assembly (e.g., valve-operator assembly and pump-motor assembly) should be considered. If superposition of test results for other than the combined loading condition is proposed, the applicability of such a procedure should be demonstrated. The design limits for nonactive pumps and valves designed by analysis may be used for the applicable loading combinations if assurance is provided by detailed stress and deformation analyses that operability is not impaired when designed to these limits. Similarly, the primary-pressure ratings P_r for nonactive valves designed by standard or alternative design rules may be used for the applicable loading combinations if appropriate testing demonstrates that operability is not impaired when the valve is so rated.

[7] Secondary effects (stresses and deformations) should be evaluated for the loading combinations designated by regulatory positions 4.a.(2) and 4.a.(3). Local effects (peak stresses) need not be considered for these loading combinations.

[8] Table I-3.0, "Permanent Strain Limiting Factors," of Appendix I of the ASME Boiler and Pressure Vessel Code, Section III, may be used as an aid in determining the relationship between design stress and deformation (see note 2 to Table I-1.2 of Section III of the ASME Code).

[9] Division 1 of Section VIII of the ASME Boiler and Pressure Vessel Code does not provide rules for design by analysis. If a detailed analysis is performed, Division 1 vessels should meet, as a minimum, equations a and b below, which are applicable to regulatory positions 6.a. and 6.b., respectively.

TABLE 3.9(B)-13 (Sheet 10)

NOTES (Cont.):

a. $\sigma_m \leq 1.1S < \frac{\sigma_m + \sigma_b}{1.5}$

b. $\sigma_m \leq 1.5S < \frac{\sigma_m + \sigma_b}{1.5}$

where:

σ_m = primary membrane stress;

σ_b = primary bending stress;

S = allowable stress value as specified in Appendix I of Section III of the ASME Boiler and Pressure Vessel Code.

[10] For the loadings designated in regulatory position 8.a.(2), only equation 9 of NC-3651 need be met.

[11] In addition to compliance with the design limits specified, assurance of operability under all design loading combinations should be provided by any appropriate combination of the following suggested measures:

a. In situ testing (e.g., preoperational testing after the component is installed in the plant).

b. Full-scale prototype testing.

c. Reduced-scale prototype testing.

d. Detailed stress and deformation analyses (includes experimental stress and deformation analyses).

In the performance of tests or analyses to demonstrate operability, the structural interaction of the entire assembly (e.g., valve-operator and pump-motor assembly) should be considered. If superposition of test results for other than

TABLE 3.9(B)-13 (Sheet 11)

NOTES (Cont.):

the combined loading condition is proposed, the applicability of such a procedure should be demonstrated. The design limits for nonactive pumps and valves may be used for the applicable loading combinations if appropriate analyses and/or testing confirms that operability is not impaired when designed to these limits.

TABLE 3.9(B)-14 RESPONSE TO REGULATORY GUIDE 1.124 FOR COMPONENTS NOT FURNISHED WITH THE NSSS

Regulatory Position	Response
ASME Code[1] Class 1 linear-type component supports excluding snubbers, which are not addressed herein, should be constructed to the rules of Subsection NF of Section III as supplemented by the following:[2]	
1. The Classification of component supports should, as a minimum, be the same as that of the supported components.	Complies.
2. Values of S_u at a temperature t should be estimated by one of three following methods on an interim basis until Section III includes such values:	
a. Method 1. This method applies to component support materials whose values of ultimate strength S_u at temperature have been tabulated by their manufacturers in catalogs or other publications.	Complies.

$$S_u = S_{ur} \frac{S'_u}{S'_{ur}} \text{ but not greater than } S_{ur}$$

where

S_u = ultimate tensile strength at temperature t to be used to determine the service limits

S_{ur} = ultimate tensile strength at room temperature tabulated in Section III, Appendix I, or the latest accepted version[3] of Code Case 1644

S'_u = ultimate tensile strength at temperature t tabulated by manufacturers in their catalogs or other publications

S'_{ur} = ultimate tensile strength at room temperature tabulated by manufacturers in the same publications.

TABLE 3.9(B)-14 (Sheet 2)

Regulatory Position	Response
b. Method 2. This method applies to component support materials whose values of ultimate tensile strength at temperature have not been tabulated by their manufacturers in any catalog or publication.	Complies.

$$S_u = S_{ur} \frac{S_y}{S_{yr}}$$

where

S_u = ultimate tensile strength at temperature t to be used to determine the service limits

S_{ur} = ultimate tensile strength at room temperature tabulated in Section III, Appendix I, or the latest accepted version[3] of Code Case 1644

S_y = minimum yield strength at temperature t tabulated in Section III, Appendix I, or the latest accepted version[3] of Code Case 1644

S_{yr} = minimum yield strength at room temperature, tabulated in Section III, Appendix I, or the latest accepted version[3] of Code Case 1644.

Regulatory Position	Response
c. Method 3. When the values of allowable stress or stress intensity at temperature for a material are listed in Section III, the ultimate tensile strength at temperature for that material may be approximated by the following expressions.	Complies.

S_u = 4S or

S_u = $3S_m$

where

S_u = ultimate tensile strength at temperature t to be used to determine the service limits

S = listed value of allowable stress at temperature t in Section III

TABLE 3.9(B)-14 (Sheet 3)

Regulatory Position	Response
S_m = listed value of allowable stress intensity at temperature t in Section III	
3. The Code levels A and B service limits for component supports designed by linear elastic analysis which are related to S_y, should meet the appropriate stress limits of Appendix XVII of Section III but should not exceed the limit specified when the value of 5/6 S_u is substituted for S_y. Examples are shown below in a and b.	Complies.
a. The tensile stress limit F_t for a net section as specified in XVII-2211(a) of Section III should be the smaller value of $0.6S_y$ or $0.5S_u$ at temperature. For net sections at pin-holes in eye-bars, pin-connected plates, or built-up structural members, F_t as specified in XVII-2211(b) should be the smaller value of $0.45S_y$ or $0.375S_y$ at temperature.	Complies.
b. The shear stress limit F_v for a gross section as specified in XVII-2212 of Section III should be the smaller value of $0.4S_y$ or $0.33S_u$ at temperature.	Complies.
Many limits and equations for compression strength specified in Sections XVII-2214, XVII-2224, XVII-2225, XVII-2240, and XVII-2260 have built-in constants based on Young's Modulus of 29,000 Ksi. For materials with Young's Modulus at working temperatures substantially different from 29,000 Ksi, these constants should be rederived with the appropriate Young's Modulus unless the conservatism of using these constants as specified can be demonstrated.	
4. Component supports designed by linear elastic analysis may increase their level A or B service limits according to the provisions of NF-3231.1(a), XVII-2110(a), and F-1370(a) of Section III. The increase of level A or B service limits provided by NF-3231.1(a) is for stress range. The increase of level A or B service limits provided by F-1370(a) for level D service limits should be the smaller factor of 2 or $1.167S_u/S_y$, if $S_u \geq 1.2S_y$ or 1.4 if $S_u \leq 1.2S_y$, where S_y and S_u are component-support material properties at temperature.	Complies.

TABLE 3.9(B)-14 (Sheet 4)

Regulatory Position	Response
However, all increases [i.e., those allowed by NF-3231.1(a), XVII-2110(a), and F-1370(a)] should always be limited by XVII-2110(b) of Section III. The critical buckling strengths defined by XVII-2110(b) of Section III should be calculated using material properties at temperature. This increase of level A or B service limits does not apply to limits for bolted connections. Any increase of limits for shear stresses above 1.5 times the Code level A service limits should be justified.	
If the increased service limit for stress range by NF-3231.1(a) is more than $2S_y$ or S_u, it should be limited to the smaller value of $2S_y$ or S_u unless it can be justified by a shake-down analysis.	
5. Component supports subjected to the combined loadings of system mechanical loadings associated with (1) either (a) the Code design condition or (b) the normal or upset plant conditions and (2) the vibratory motion of the OBE should be designed within the following limits:[4,5]	
a. The stress limits of XVII-2000 of Section III and Regulatory Position 3 of this guide should not be exceeded for component supports designed by the linear elastic analysis method. These stress limits may be increased according to the provisions of NF-3231.1(a) of Section III and Regulatory Position 4 of this guide when effects resulting from constraints of free-end displacements are added to the loading combination.	Complies.
b. The normal condition load rating or the upset condition load rating of NF-3262.3 of Section III should not be exceeded for component supports designed by the load-rating method.	Complies.
c. The lower bound collapse load determined by XVII-4200 adjusted according to the provision of XVII-4110(a) of Section III should not be exceeded for component supports designed by the limit analysis method.	N/A
d. The collapse load determined by II-1400 of Section III divided by 1.7 should not be exceeded for component supports designed by the experimental stress analysis method.	N/A

TABLE 3.9(B)-14 (Sheet 5)

Regulatory Position	Response
6. Component supports subjected to the system mechanical loadings associated with the emergency plant condition should be designed within the following design limits except when the normal function of the supported system is to prevent or mitigate the consequences of events associated with the emergency plant condition (at which time Regulatory Position 8 applies):[4,5]	
a. The stress limits of XVII-2000 of Section III and Regulatory Positions 3 and 4, increased according to the provisions of XVII-2110(a) of Section III and Regulatory Position 4 of this guide, should not be exceeded for component supports designed by the linear elastic analysis method.	
b. The emergency condition load rating of NF-3262.3 of Section III should not be exceeded for component supports designed by the load-rating method.	
c. The lower bound collapse load determined by XVII-4200 adjusted according to the provision of XVII-4110(a) of Section III should not be exceeded for component supports designed by the limit analysis method.	
d. The collapse load determined by II-1400 of Section III divided by 1.3 should not be exceeded for component supports designed by the experimental stress analysis method.	
7. Component supports subjected to the combined loadings of (1) the system mechanical loadings associated with the normal plant condition, (2) the vibratory motion of the SSE, and (3) the dynamic system loadings associated with the faulted plant condition should be designed within the following limits except when the normal function of the supported system is to prevent or mitigate the consequences of events associated with the faulted plant condition (at which time Regulatory Position 8 applies):	Complies.
a. The stress limits of XVII-2000 of Section III and Regulatory Position 3 of this guide, increased according to the provisions of F-1370(a) of Section III and Regulatory Position 4 of this guide, should not be exceeded for component supports designed by the linear elastic analysis method.	Complies.

TABLE 3.9(B)-14 (Sheet 6)

Regulatory Position	Response
b. The smaller value of T.L. x $2S/S_u$ or T.L. x 0.7 S'_u /S_u should not be exceeded, where T.L., S, and S_u are defined according to NF-3262.1 of Section III, and S'_u is the minimum ultimate tensile strength of the material at service temperature for component supports designed by the load-rating method.	N/A
c. The lower bound collapse load determined by XVII-4200 adjusted according to the provision of F-1370(b) of Section III should not be exceeded for component supports designed by the limit analysis method.	N/A
d. The collapse load determined by II-1400 adjusted according to the provision of F-1370(b) of Section III should not be exceeded for component supports designed by the experimental stress analysis method.	N/A
8. Component supports in systems whose normal function is to prevent or mitigate the consequences of events associated with an emergency or faulted plant condition should be designed within the limits described in Regulatory Position 5 or other justifiable limits provided by the Code. These limits should be defined by the Design Specification and stated in the PSAR, such that the function of the supported system will be maintained when they are subjected to the loading combinations described in Regulatory Positions 6 and 7.	Complies.

NOTES:

[1] American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, 1974 Edition, including the 1976 Winter Addenda thereto.

[2] If the function of a component support is not required during a plant condition, the design limits of the support for that plant condition need not be satisfied, provided excessive deflection or failure of the support will not result in the loss of function of any other safety-related system.

TABLE 3.9(B)-14 (Sheet 7)

3 Regulatory Guide 1.85, "Code Case Acceptability--ASME Section III Materials," provides guidance for the acceptability of ASME Section III Code Cases and their revisions, including Code Case 1644. Supplementary provisions for the use of specific code cases and their revisions may also be provided and should be considered when applicable.

4 Since component supports are deformation sensitive in the performance of their service requirements, satisfying these criteria does not ensure that their functional requirements will be fulfilled. Any deformation limits specified by the design specification may be controlling and should be satisfied.

5 Since the design of component supports is an integral part of the design of the system and the design of the component, the designer must make sure that methods used for the analysis of the system, component, and component support are compatible (see Table F-1322.2-1 in Appendix F of Section III). Large deformations in the system or components should be considered in the design of component supports.

TABLE 3.9(B)-15 ACTIVE PUMPS NOT FURNISHED WITH THE NSSS

Pump	Item Number	System	ANS Safety Class	Normal Mode	Post-LOCA Mode	Basis
Turbine-Driven Auxiliary Feedwater Pump	PAL02	AFWS	3	Off	On/Off	Provide makeup to the intact S/Gs following a MSLB. See Table 15.0-6
Electric Motor-Driven Auxiliary Feedwater	PAL01	AFWS	3	Off	On/Off	Provide makeup to the intact S/Gs following a MSLB. See Table 15.0-6
Pumps A and B						
Component Cooling Water Pumps A, B, C, and D	PEG01	CCWS	3	On/Off	On/Off	Circulates cooled water to the reactor coolant pumps, RHR, and spent fuel pool cooling heat exchangers
Spent Fuel Pool Cooling Pumps A and B (See Note 3)	PEC01	FPCS	3	On/Off	On/Off	Circulates cooled water to the spent fuel pool
Containment Spray Pumps A and B	PEN01	CSS	2	Off	On/Off	Depressurization of the containment following a LOCA and MSLB
Essential Service Water Pumps A and B (see Note 2)	PEF01	ESWS	3	Off	On	Provide cooling water to ECCS auxiliaries such as component cooling water heat exchangers
Emergency Fuel Oil Transfer Pumps A and B (see Notes 1 and 2)	PJE01	EF0S	3	Off	On	Transfers fuel oil from the storage tank to the day tank

NOTES: (1) Vibration measurements (shop and in-plant) were not obtained due to the fact that these pumps are immersed in the fuel oil in the emergency fuel oil storage tank.

(2) Bearing temperatures (shop and in-plant) were not measured due to the fact that these pumps are immersed in the pumped fluid.

(3) The fuel-pool cooling pumps are operated either continuously or intermittently during normal plant operation, thus ensuring their operability.

TABLE 3.9(B)-16 ACTIVE VALVES

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
AB-HV-005	Main Steam	Air Cylinder	4.0	Globe/2	Closed	3, 4
AB-HV-006	Main Steam	Air Cylinder	4.0	Globe/2	Closed	3, 4
AB-HV-011	Main Steam	System - Medium	28.0	Gate/2	Open	3
AB-HV-014	Main Steam	System - Medium	28.0	Gate/2	Open	3
AB-HV-017	Main Steam	System - Medium	28.0	Gate/2	Open	3
AB-HV-020	Main Steam	System - Medium	28.0	Gate/2	Open	3
AB-HV-048	Main Steam	Air Cylinder	1.0	Globe/2	Open	3, 4
AB-HV-049	Main Steam	Air Cylinder	1.0	Globe/2	Open	3, 4
AB-LV-007	Main Steam	Air Cylinder	2.0	Globe/2	Closed	3
AB-LV-008	Main Steam	Air Cylinder	2.0	Globe/2	Closed	3
AB-LV-009	Main Steam	Air Cylinder	2.0	Globe/2	Closed	3
AB-LV-010	Main Steam	Air Cylinder	2.0	Globe/2	Closed	3
AB-PV-001	Main Steam	Air Cylinder	8.0	Globe/2	Closed	2, 3, 5
AB-PV-002	Main Steam	Air Cylinder	8.0	Globe/2	Closed	2, 3, 5
AB-PV-003	Main Steam	Air Cylinder	8.0	Globe/2	Closed	2, 3, 5
AB-PV-004	Main Steam	Air Cylinder	8.0	Globe/2	Closed	2, 3, 5
AB-V-007	Main Steam	Manually	10.0	Gate/2	Open	3
AB-V-018	Main Steam	Manually	10.0	Gate/2	Open	3

TABLE 3.9(B)-16 (Sheet 2)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
AB-V-029	Main Steam	Manually	10.0	Gate/2	Open	3
AB-V-040	Main Steam	Manually	10.0	Gate/2	Open	3
AB-V-045	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-046	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-047	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-048	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-049	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-055	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-056	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-057	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-058	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-059	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-065	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-066	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-067	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-068	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-069	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-075	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-076	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-077	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AB-V-078	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5

TABLE 3.9(B)-16 (Sheet 3)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
AB-V-079	Main Steam	Self-Actuated	6.0	Pressure Safety/2	Closed	3, 5
AE-FV-039	Main Steam	System-Medium	14.0	Gate/2	Open	3
AE-FV-040	Feedwater	System-Medium	14.0	Gate/2	Open	3
AE-FV-041	Feedwater	System-Medium	14.0	Gate/2	Open	3
AE-FV-042	Feedwater	System-Medium	14.0	Gate/2	Open	3
AE-V-120	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-121	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-122	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-123	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-124	Feedwater	ΔP	4.0	Check/2	NA	4
AE-V-125	Feedwater	ΔP	4.0	Check/2	NA	4
AE-V-126	Feedwater	ΔP	4.0	Check/2	NA	4
AE-V-127	Feedwater	ΔP	4.0	Check/2	NA	4
AL-HV-005	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-006	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4
AL-HV-007	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-008	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4
AL-HV-009	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-010	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4
AL-HV-011	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-012	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4

TABLE 3.9(B)-16 (Sheet 4)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
AL-HV-030	Auxiliary Feedwater	Electric Motor	6.0	Butterfly/3	Closed	4
AL-HV-031	Auxiliary Feedwater	Electric Motor	6.0	Butterfly/3	Closed	4
AL-HV-032	Auxiliary Feedwater	Electric Motor	8.0	Butterfly/3	Closed	4
AL-HV-033	Auxiliary Feedwater	Electric Motor	8.0	Butterfly/3	Closed	4
AL-HV-034	Auxiliary Feedwater	Electric Motor	8.0	Gate/3	Open	6
AL-HV-035	Auxiliary Feedwater	Electric Motor	8.0	Gate/3	Open	6
AL-HV-036	Auxiliary Feedwater	Electric Motor	10.0	Gate/3	Open	6
AL-V-001	Auxiliary Feedwater	ΔP	10.0	Check/3	NA	6
AL-V-002	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	6
AL-V-003	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	6
AL-V-006	Auxiliary Feedwater	ΔP	6.0	Check/3	NA	4
AL-V-009	Auxiliary Feedwater	ΔP	6.0	Check/3	NA	4
AL-V-012	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	4, 6
AL-V-015	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	4, 6
AL-FV-030	Auxiliary Feedwater	ΔP	6.0	Check/recirc/3	NA	4
AL-V-033	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-036	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-FV-042	Auxiliary Feedwater	ΔP	6.0	Check/recirc/3	NA	4
AL-V-045	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-048	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-053	Auxiliary Feedwater	ΔP	3.0	Check/3	NA	4

TABLE 3.9(B)-16 (Sheet 5)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
AL-V-054	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	4
AL-V-057	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-062	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-067	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-072	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
BB-HV-013	Reactor Coolant	Electric Motor	3.0	Gate/3	Open	6
BB-HV-014	Reactor Coolant	Electric Motor	3.0	Gate/3	Open	6
BB-HV-015	Reactor Coolant	Electric Motor	3.0	Gate/3	Open	6
BB-HV-016	Reactor Coolant	Electric Motor	3.0	Gate/3	Open	6
BB-V-001	Reactor Coolant	ΔP	1.5	Check/1	NA	2, 7, 8
BB-V-022	Reactor Coolant	ΔP	1.5	Check/1	NA	2, 7, 8
BB-V-040	Reactor Coolant	ΔP	1.5	Check/1	NA	2, 7, 8
BB-V-059	Reactor Coolant	ΔP	1.5	Check/1	NA	2, 7, 8
BB-V-118	Reactor Coolant	ΔP	2.0	Check/2	NA	2
BB-V-120	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-121	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-122	Reactor Coolant	ΔP	1.5	Check/3	NA	6
BB-V-148	Reactor Coolant	ΔP	2.0	Check/2	NA	2
BB-V-150	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-151	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-152	Reactor Coolant	ΔP	1.5	Check/3	NA	6

TABLE 3.9(B)-16 (Sheet 6)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
BB-V-178	Reactor Coolant	ΔP	2.0	Check/2	NA	2
BB-V-180	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-181	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-182	Reactor Coolant	ΔP	1.5	Check/3	NA	6
BB-V-208	Reactor Coolant	ΔP	2.0	Check/2	NA	2
BB-V-210	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-211	Reactor Coolant	ΔP	2.0	Check/1	NA	2, 8
BB-V-212	Reactor Coolant	ΔP	1.5	Check/3	NA	6
BB-V-474	Reactor Coolant	ΔP	1.5	Check/3	NA	6
BB-V-476	Reactor Coolant	ΔP	1.5	Check/3	NA	6
BB-V-479	Reactor Coolant	ΔP	1.5	Check/3	NA	6
BB-V-480	Reactor Coolant	ΔP	1.5	Check/3	NA	6
BG-V-91	Chemical and Volume Control	ΔP	2.0	Check/2	NA	2, 7
BG-V-95	Chemical and Volume Control	ΔP	2.0	Check/2	NA	2, 7
BG-V-135	Chemical and Volume Control	ΔP	0.8	Check/2	NA	1
BG-V-147	Chemical and Volume Control	ΔP	3.0	Check/2	NA	4
BG-V-155	Chemical and Volume Control	ΔP	0.8	Check/3	NA	4

TABLE 3.9(B)-16 (Sheet 7)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
BG-V-165	Chemical and Volume Control	ΔP	3.0	Check/2	NA	4
BG-V-167	Chemical and Volume Control	ΔP	0.8	Check/3	NA	4
BG-V-174	Chemical and Volume Control	ΔP	2.0	Check/2	NA	4
BG-V-589	Chemical and Volume Control	ΔP	1.0	Check/2	NA	4
BG-V-590	Chemical and Volume Control	ΔP	1.0	Check/2	NA	4
BG-V-605	Chemical and Volume Control	ΔP	3.0	Check/2	NA	7
BG-V-606	Chemical and Volume Control	ΔP	3.0	Check/2	NA	7
BM-HV-001	Steam Generator Blowdown	Air Cylinder	4.0	Globe/2	Open	3
BM-HV-002	Steam Generator Blowdown	Air Cylinder	4.0	Globe/2	Open	3
BM-HV-003	Steam Generator Blowdown	Air Cylinder	4.0	Globe/2	Open	3
BM-HV-004	Steam Generator Blowdown	Air Cylinder	4.0	Globe/2	Open	3
BM-HV-019	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Closed	3

TABLE 3.9(B)-16 (Sheet 8)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
BM-HV-020	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Closed	3
BM-HV-021	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Closed	3
BM-HV-022	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Closed	3
BM-HV-035	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BM-HV-036	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BM-HV-037	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BM-HV-038	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BM-HV-065	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BM-HV-066	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BM-HV-067	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BM-HV-068	Steam Generator Blowdown	Solenoid	1.0	Globe/2	Open	3
BN-HV-003	Borated Refueling Water Storage	Electric Motor	12.0	Gate/2	Open	4, 6

TABLE 3.9(B)-16 (Sheet 9)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
BN-HV-004	Borated Refueling Water Storage	Electric Motor	12.0	Gate/2	Open	4, 6
EC-HV-011	Fuel Pool Cooling and Cleanup	Electric Motor	12.0	Butterfly/3	Throttled	4
EC-HV-012	Fuel Pool Cooling and Cleanup	Electric Motor	12.0	Butterfly/3	Throttled	4
EF-HV-023	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-024	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-025	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-026	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-031	Essential Service Water	Electric Motor	14.0	Butterfly/2	Open	1
EF-HV-032	Essential Service Water	Electric Motor	14.0	Butterfly/2	Open	1
EF-HV-033	Essential Service Water	Electric Motor	14.0	Butterfly/2	Open	1
EF-HV-034	Essential Service Water	Electric Motor	14.0	Butterfly/2	Open	1
EF-HV-037	Essential Service Water	Electric Motor	30.0	Butterfly/3	Closed	4
EF-HV-038	Essential Service Water	Electric Motor	30.0	Butterfly/3	Closed	4
EF-HV-039	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-040	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-041	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-042	Essential Service Water	Electric Motor	30.0	Butterfly/3	Open	6
EF-HV-045	Essential Service Water	Electric Motor	14.0	Butterfly/2	Open	1, 4
EF-HV-046	Essential Service Water	Electric Motor	14.0	Butterfly/2	Open	1, 4

TABLE 3.9(B)-16 (Sheet 10)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
EF-HV-047	Essential Service Water	Electric Motor	10.0	Butterfly/2	Open	1, 4
EF-HV-048	Essential Service Water	Electric Motor	10.0	Butterfly/2	Open	1, 4
EF-HV-049	Essential Service Water	Electric Motor	14.0	Butterfly/2	Closed	1, 4
EF-HV-050	Essential Service Water	Electric Motor	14.0	Butterfly/2	Closed	1, 4
EF-HV-051	Essential Service Water	Electric Motor	24.0	Butterfly/3	Closed	4
EF-HV-052	Essential Service Water	Electric Motor	24.0	Butterfly/3	Open	4
EF-HV-059	Essential Service Water	Electric Motor	24.0	Butterfly/3	Closed	4
EF-HV-060	Essential Service Water	Electric Motor	24.0	Butterfly/3	Open	4
EF-HV-097	Essential Service Water	Electric Motor	3.0	Gate/3	Open	4
EF-HV-098	Essential Service Water	Electric Motor	3.0	Gate/3	Open	4
EF-PDV-019	Essential Service Water	Electric Motor	3.0	Gate/3	Closed	4
EF-PDV-020	Essential Service Water	Electric Motor	3.0	Gate/3	Closed	4
EF-V-001	Essential Service Water	ΔP	30.0	Check/3	NA	4
EF-V-004	Essential Service Water	ΔP	30.0	Check/3	NA	4
EF-V-046	Essential Service Water	ΔP	2.5	Check/3	NA	6
EF-V-076	Essential Service Water	ΔP	2.5	Check/3	NA	6
EF-HV-043	Essential Service Water	Air Cylinder	2.0	Globe/3	Open	6
EF-HV-044	Essential Service Water	Air Cylinder	2.0	Globe/3	Open	6
EG-HV-011	Component Cooling Water	Electric Motor	1.5	Globe/3	Closed	4
EG-HV-012	Component Cooling Water	Electric Motor	1.5	Globe/3	Closed	4

TABLE 3.9(B)-16 (Sheet 11)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
EG-HV-013	Component Cooling Water	Electric Motor	1.5	Globe/3	Closed	4
EG-HV-014	Component Cooling Water	Electric Motor	1.5	Globe/3	Closed	4
EG-HV-015	Component Cooling Water	Electric Motor	18.0	Butterfly/3	Open	4, 6
EG-HV-016	Component Cooling Water	Electric Motor	18.0	Butterfly/3	Open	4, 6
EG-HV-053	Component Cooling Water	Electric Motor	18.0	Butterfly/3	Open	4, 6
EG-HV-054	Component Cooling Water	Electric Motor	18.0	Butterfly/3	Open	4, 6
EG-HV-058	Component Cooling Water	Electric Motor	12.0	Gate/2	Open	1
EG-HV-059	Component Cooling Water	Electric Motor	12.0	Gate/2	Open	1
EG-HV-060	Component Cooling Water	Electric Motor	12.0	Gate/2	Open	1
EG-HV-061	Component Cooling Water	Electric Motor	4.0	Gate/2	Open	1
EG-HV-062	Component Cooling Water	Electric Motor	4.0	Gate/2	Open	1
EG-HV-069A	Component Cooling Water	Air Cylinder	14.0	Butterfly/3	Open	6

TABLE 3.9(B)-16 (Sheet 12)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
EG-HV-069B	Component Cooling Water	Air Cylinder	14.0	Butterfly/3	Open	6
EG-HV-070A	Component Cooling Water	Air Cylinder	14.0	Butterfly/3	Open	6
EG-HV-070B	Component Cooling Water	Air Cylinder	14.0	Butterfly/3	Open	6
EG-HV-101	Component Cooling Water	Electric Motor	18.0	Butterfly/3	Open	4
EG-HV-102	Component Cooling Water	Electric Motor	18.0	Butterfly/3	Open	4
EG-TV-029	Component Cooling Water	Air Cylinder	20.0	Butterfly/3	Throttled	4
EG-TV-030	Component Cooling Water	Air Cylinder	20.0	Butterfly/3	Throttled	4
EG-V-003	Component Cooling Water	ΔP	20.0	Check/3	NA	4
EG-V-007	Component Cooling Water	ΔP	20.0	Check/3	NA	4
EG-V-012	Component Cooling Water	ΔP	20.0	Check/3	NA	4
EG-V-016	Component Cooling Water	ΔP	20.0	Check/3	NA	4
EG-V-130	Component Cooling Water	ΔP	18.0	Check/3	NA	4, 6

TABLE 3.9(B)-16 (Sheet 13)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
EG-V-131	Component Cooling Water	ΔP	18.0	Check/3	NA	4, 6
EG-V-204	Component Cooling Water	ΔP	12.0	Check/2	NA	1
EM-V-001	High Pressure Coolant Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EM-V-002	High Pressure Coolant Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EM-V-003	High Pressure Coolant Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EM-V-004	High Pressure Coolant Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EM-V-005	High Pressure Coolant Injection	ΔP	1.5	Check/2	NA	7
EM-V-006	High Pressure Coolant Injection	ΔP	1.0	Check/2	NA	1
EM-V-007	High Pressure Coolant Injection	ΔP	1.5	Check/2	NA	7
EN-HV-001	Containment Spray	Electric Motor	12.0	Gate/2	Closed	1
EN-HV-006	Containment Spray	Electric Motor	10.0	Gate/2	Closed	4
EN-HV-007	Containment Spray	Electric Motor	12.0	Gate/2	Closed	1
EN-HV-012	Containment Spray	Electric Motor	10.0	Gate/2	Closed	4
EN-V-002	Containment Spray	ΔP	12.0	Check/2	NA	4
EN-V-003	Containment Spray	ΔP	12.0	Check/2	NA	4

TABLE 3.9(B)-16 (Sheet 14)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
EN-V-004	Containment Spray	ΔP	10.0	Check/2	NA	4
EN-V-008	Containment Spray	ΔP	12.0	Check/2	NA	4
EN-V-009	Containment Spray	ΔP	12.0	Check/2	NA	4
EN-V-010	Containment Spray	ΔP	10.0	Check/2	NA	4
EN-V-013	Containment Spray	ΔP	10.0	Check/2	NA	1, 4
EN-V-017	Containment Spray	ΔP	10.0	Check/2	NA	1, 4
EP-V-010	Accum. Safety Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EP-V-020	Accum. Safety Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EP-V-030	Accum. Safety Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EP-V-040	Accum. Safety Injection	ΔP	2.0	Check/1	NA	1, 7, 8
EP-V-046	Accum. Safety Injection	ΔP	1.0	Check/2	NA	1
FC-FV-310	Auxiliary Turbines	Air Cylinder	1.0	Globe/3	Open	6
FC-FV-313	Auxiliary Turbines	Electric Motor	4.0	Globe	Open	4
FC-HV-312	Auxiliary Turbines	Electric Motor	4.0	Gate	Closed	4
FC-V-001	Auxiliary Turbines	ΔP	4.0	Check/2	NA	4, 6
FC-V-002	Auxiliary Turbines	ΔP	4.0	Check/2	NA	4, 6
FC-V-024	Auxiliary Turbines	ΔP	4.0	Check/2	NA	4, 6
FC-V-025	Auxiliary Turbines	ΔP	4.0	Check/2	NA	4, 6
GG-RV-027A	Fuel Building HVAC	Solenoid	1.0	Gate	Closed	4
GG-RV-027B	Fuel Building HVAC	Solenoid	1.0	Gate	Open	4
GG-RV-027C	Fuel Building HVAC	Solenoid	1.0	Gate	Closed	4

TABLE 3.9(B)-16 (Sheet 15)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
GG-RV-027D	Fuel Building HVAC	Solenoid	1.0	Gate	Open	4
GG-RV-028A	Fuel Building HVAC	Solenoid	1.0	Gate	Closed	4
GG-RV-028B	Fuel Building HVAC	Solenoid	1.0	Gate	Open	4
GG-RV-028C	Fuel Building HVAC	Solenoid	1.0	Gate	Closed	4
GG-RV-028D	Fuel Building HVAC	Solenoid	1.0	Gate	Open	4
GK-VO-765	Control Building HVAC	Electric Motor	3.0	Globe/3	Open	4
GK-VO-766	Control Building HVAC	Electric Motor	3.0	Globe/3	Open	4
GK-VO-767	Control Building HVAC	Electric Motor	3.0	Globe/3	Open	4
GK-VO-768	Control Building HVAC	Electric Motor	3.0	Globe/3	Open	4
GS-HV-003	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-004	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-005	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-008	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-009	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-012	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-013	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1

TABLE 3.9(B)-16 (Sheet 16)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
GS-HV-014	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-017	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-018	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Closed	1
GS-HV-020	Containment Hydrogen Control	Electric Motor	6.0	Butterfly/2	Closed	1
GS-HV-021	Containment Hydrogen Control	Electric Motor	6.0	Butterfly/2	Closed	1
GS-HV-031	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1
GS-HV-032	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1
GS-HV-033	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1
GS-HV-034	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1
GS-HV-036	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1
GS-HV-037	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1
GS-HV-038	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1

TABLE 3.9(B)-16 (Sheet 17)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
GS-HV-039	Containment Hydrogen Control	Solenoid	1.0	Gate/2	Open	1
GT-HZ-004	Containment Purge	Air Cylinder	18.0	Butterfly/2	Open	1
GT-HZ-005	Containment Purge	Air Cylinder	18.0	Butterfly/2	Open	1
GT-HZ-006	Containment Purge	Air Cylinder	36.0	Butterfly/2	Closed	1
GT-HZ-007	Containment Purge	Air Cylinder	36.0	Butterfly/2	Closed	1
GT-HZ-008	Containment Purge	Air Cylinder	36.0	Butterfly/2	Closed	1
GT-HZ-009	Containment Purge	Air Cylinder	36.0	Butterfly/2	Closed	1
GT-HZ-011	Containment Purge	Air Cylinder	18.0	Butterfly/2	Open	1
GT-HZ-012	Containment Purge	Air Cylinder	18.0	Butterfly/2	Open	1
JE-V-085	Emergency Fuel Oil	ΔP	2.0	Check/3	Closed	2, 7
JE-V-086	Emergency Fuel Oil	ΔP	2.0	Check/3	Closed	2, 7
KA-FV-029	Compressed Air	Air Cylinder	2.0	Globe/2	Open	1
KA-V-204	Compressed Air	ΔP	1.5	Check/2	NA	1
KJ-PV-001A	Standby Diesel Generator	Solenoid	0.4	Globe/3	Closed	4
KJ-PV-001B	Standby Diesel Generator	Solenoid	0.4	Globe/3	Closed	4
KJ-PV-101A	Standby Diesel Generator	Solenoid	0.4	Globe/3	Closed	4
KJ-PV-101B	Standby Diesel Generator	Solenoid	0.4	Globe/3	Closed	4

TABLE 3.9(B)-16 (Sheet 18)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
KJ-TCV-034	Standby Diesel Generator	Self-Actuated	5.0	3-Way/3	NA	4, 9
KJ-TCV-056	Standby Diesel Generator	Self-Actuated	6.0	3-Way/3	NA	4, 9
KJ-TCV-060	Standby Diesel Generator	Self-Actuated	6.0	3-Way/3	NA	4, 9
KJ-TCV-134	Standby Diesel Generator	Self-Actuated	5.0	3-Way/3	NA	4, 9
KJ-TCV-156	Standby Diesel Generator	Self-Actuated	6.0	3-Way/3	NA	4, 9
KJ-TCV-160	Standby Diesel Generator	Self-Actuated	6.0	3-Way/3	NA	4, 9
KJ-V-711A	Standby Diesel Generator	ΔP	0.8	Check/Mfr Std	NA	6
KJ-V-711B	Standby Diesel Generator	ΔP	0.8	Check/Mfr Std	NA	6
KJ-V-712A	Standby Diesel Generator	ΔP	0.8	Check/Mfr Std	NA	6
KJ-V-712B	Standby Diesel Generator	ΔP	0.8	Check/Mfr Std	NA	6
LF-FV-095	Floor and Equipment Drains	Electric Motor	6.0	Gate/2	Open	1
LF-FV-096	Floor and Equipment Drains	Air Cylinder	6.0	Globe/2	Closed	1

TABLE 3.9(B)-16 (Sheet 19)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
LF-HV-105	Floor and Equipment Drains	Electric Motor	6.0	Gate/3	Open	6
LF-HV-106	Floor and Equipment Drains	Electric Motor	6.0	Gate/3	Open	6
SJ-HV-005	Nuclear Sampling	Solenoid	1.0	Globe/2	Open	1
SJ-HV-006	Nuclear Sampling	Solenoid	1.0	Globe/2	Open	1
SJ-HV-012	Nuclear Sampling	Solenoid	1.0	Globe/2	Closed	1
SJ-HV-013	Nuclear Sampling	Solenoid	1.0	Globe/2	Closed	1
SJ-HV-018	Nuclear Sampling	Solenoid	1.0	Globe/2	Closed	1
SJ-HV-019	Nuclear Sampling	Solenoid	1.0	Globe/2	Closed	1
SJ-HV-127	Nuclear Sampling	Solenoid	1.0	Globe/2	Closed	1
SJ-HV-128	Nuclear Sampling	Solenoid	1.0	Globe/2	Open	1
SJ-HV-129	Nuclear Sampling	Solenoid	1.0	Globe/2	Open	1
SJ-HV-130	Nuclear Sampling	Solenoid	1.0	Globe/2	Closed	1
EF-HV-065	Essential Service Water	Electric Motor	30.0	Butterfly/3	Closed	4
EF-HV-066	Essential Service Water	Electric Motor	30.0	Butterfly/3	Closed	4

BASIS

1. Containment isolation

2. Safety grade cold shutdown operation

3. Secondary side pressure boundary isolation

TABLE 3.9(B)-16 (Sheet 20)

4. System operation

5. Pressure/relief

6. System pressure boundary isolation

7. ECCS safeguards operation

8. RCPB isolation

9. Not required to be in Inservice Testing Program per ASME OM Code

APPENDIX 3.9(B)A - ME-632 VERIFICATION REPORT

The following is a comparison of the ME-632 program results with the results of the Engineering Data System computer program.

The two piping systems chosen for stress checks were:

a. The Core Spray Piping System - Monticello Nuclear Generating Plant Unit 1

b. Lines 48223-18-HE, 50056-10-HE, and 50057-10-HE-SMUD Rancho Seco Unit 1

These two test cases were chosen because independent piping stress analyses performed by Engineering Data Systems (EDS) under contract to Bechtel were available for comparison purposes. The EDS (PISOL 3) analysis of the core spray piping system consisted of both deadweight and thermal loading while the SMUD Rancho Seco piping system was an earthquake response spectrum analysis.

The ME-632 piping stress analyses were performed in the September 18-20, 1972 period on PICC's Honeywell 635 computer. A relocatable binary deck of the program is stored on tape No. 8312 and will be retained indefinitely for documentation purposes.

A comparison of the ME-632 and EDS analyses is shown in Table 3.9(B)A-1. Due to differing sign conventions, the reactions have opposite signs. The EDS program prints the effects of the support on the piping system while ME-632 prints the effect of the piping system on the support. In some cases, the maximum values for the ME-632 analysis occurred at the middle of the bend. However, since the EDS program does not compute output quantities at the middle of a bend, these maximums are not shown in Table 1. The maximums shown in the table occurred at the same physical point on the piping system in both analyses.

In all cases, the maximum difference in output quantities was less than 5 percent, based upon the corresponding peak value for the particular load case.

It is, therefore, concluded that ME-632 correctly performs static and thermal analysis of piping systems, consistent with the assumptions of the elastic beam theory and applicable flexibility and stress intensification factors specified in ASME Section III.

TABLE 3.9(B)A-1 SUMMARY OF MAXIMUM DEFLECTIONS, STRESSES, AND REACTIONS CORE SPRAY PIPING SYSTEM MONTICELLO NUCLEAR GENERATING PLANT, UNIT 1

	Gravity		Thermal 1	
Max. deflections	ME632	EDS	ME632	EDS
X	-.0323	-.0327	- .236	- .244
Y inches	-.0714	-.0722	1.622	1.622
Z	-.0148	-.0151	- .625	-0.651
Max. stress				
σ_{eff} - psi	2133	2100	16099	15990
Max. reactions				
F_x	±72	±73	±441	±426
F_y lb	- 2949	2956	±2692	±2650
F_z	±34	±35	±296	±383
M_x	4110	4031	-31804	31584
M_y lb-feet	- 933	945	- 5913	5950
M_z	1110	1122	- 5929	5828

A comparison of maximum stresses, deflections, and reaction forces is shown in Table 3.9(B)-A2. Unless otherwise noted, the corresponding maximums occurred at identical locations. In all cases, the maximum difference between the two programs was less than 5 percent, based upon the peak deflection, stress, moment, or force for the particular load case.

The natural periods obtained from the two programs are shown in Table 3.9(B)-A3. Again there is excellent agreement.

It is, therefore, concluded that the ME-632 computer program correctly performs a dynamic analysis of piping systems consistent with the assumptions of the lumped mass, response spectrum approach for elastic systems and applicable stress intensification and flexibility factors per the ASME Section III Code.

TABLE 3.9(B)A-2 SUMMARY OF MAXIMUM DEFLECTIONS, STRESSES, AND REACTIONS SMUD RANCHO SECO, UNIT 1 PIPING SYSTEM

	X + Y Earthquake		Z + Y Earthquake	
Max. deflections	ME-632	EDS	ME-632	EDS
X	0.0505	.0496	.0080	.0117
Y inches	0.0086	.0084	.0033	.0036
Z	0.0040	.0054	.0460	.0437
Max. stress				
σ_{eff} psi	1396*	1377	1644	1564
Max. reactions				
F_x	871	881	892	963
F_y kips	377	372	118	149
F_z	664	663	3195	3128
M_x	272	268	119	122
M_y kip-feet	269	349	1964	1919
M_z	1668	1646	269	394

NOTE:

* The peak stress shown here occurred at the beginning of the bend defined by tangent intersection point 20. A higher stress occurred at the middle of this bend, but EDS output does not give stresses at the middle of the bends.

TABLE 3.9(B)A-3 COMPARISON TO NATURAL PERIODS SMUD RANCHO SECO, UNIT 1 PIPING SYSTEM

	Period-Seconds	
	ME-632	EDS
1	.1077	.1060
2	.1035	.1030
3	.0658	.0656
4	.0561	.0569
5	.0532	.0552
6	.0509	.0524
7	.0502	.0509

The following stress analyses were performed, using the ME-632 piping stress analysis computer program:

a.	Monticello Nuclear Power Plant Core Spray Piping System Unit 1	Deadweight: Thermal with anchor movements
b.	SMUD Rancho Seco, Unit 1 Lines 48223-18-HE, 50056-10-HE, and 50057-10-HE	Earthquake

The resulting forces, moments, deflections, and stresses were compared with independent analyses performed on the same piping systems, using the same loadings. A comparison of results showed that differences in the output quantities were less than 5 percent, based upon the corresponding maximum value.

Based upon these results, the ME-632 program may be used with confidence to analyze piping systems per the ASME Section III Nuclear Piping Code.

3.9(N) MECHANICAL SYSTEMS AND COMPONENTS

3.9(N).1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9(N).1.1 Design Transients

The following five operating conditions, as defined in Section III of the ASME Code, are considered in the design of the reactor coolant system (RCS), RCS component supports, and reactor internals.

a. Normal conditions

Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted, or testing conditions.

b. Upset conditions (incidents of moderate frequency)

Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition shall be included in the design specifications.

c. Emergency conditions (infrequent incidents)

Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences over the plant design lifetime 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. Faulted conditions (limiting faults)

Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to

the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

e. Testing conditions

Testing conditions are those pressure overload tests including hydrostatic tests and pneumatic tests specified. Other types of tests shall be classified under normal conditions.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The following design conditions are given in the equipment specifications for RCS components.

The analyzed design transients and the number of cycles of each that are normally used for fatigue evaluations are shown in Table 3.9(N)-1. The monitored design transients and the number of cycles of each per Technical Specification 5.5.5 are shown in Table 3.9(N)-1A. In accordance with the ASME Code, Section III, emergency and faulted conditions are not included in fatigue evaluations.

<u>Normal Conditions</u>

The following primary system transients are considered normal conditions:

a. Heatup and cooldown at 100°F per hour

b. Unit loading and unloading at 5 percent of full power per minute

c. Step load increase and decrease of 10 percent of full power

d. Large step load decrease with steam dump

e. Steady state fluctuations

1. Initial

2. Random

f. Feedwater cycling at hot shutdown

g. Loop out of service

h. Unit loading and unloading between 0 and 15 percent of full power

i. Boron concentration equalization

j. Reactor coolant pump startup and shutdown

k. Reduced temperature return to power

l. Refueling

m. Turbine roll test

n. Primary side leakage test

o. Secondary side leakage test

p. Feedwater heaters out of service

Heatup and Cooldown at 100°F per Hour

The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F per hour. (These operations can take place at lower rates, administratively controlled by procedure.)

For these cases, the heatup occurs from ambient (assumed to be 120°F*) to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

a. Material ductility considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

b. Slower initial heatup rates when using pump energy only.

---

* RCS temperature can be as low as 70°F if the system is depressurized.

c. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry, and gas adjustments.

The number of such complete heatup and cooldown operations is specified as 200 each, which corresponds to five such occurrences per year for the 40-year plant design life.

Unit Loading and Unloading at 5 Percent of Full Power per Minute

The following discussion conservatively establishes the number of anticipated cycles. The rod control system is normally operated under automatic control. Automatic rod withdrawal is no longer available. Automatic rod insertion accommodates a ramp load decrease of 5% per minute over the entire power range.

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 15-percent load and full load (the C-5 interlock at 15% load is no longer associated with the rod control system). This load swing is the maximum possible consistent with operation under automatic reactor control. The reactor temperature will vary with load, as prescribed by the reactor control system. The number of loading and unloading operations is defined as 13,200. One loading operation per day yields 14,600 such operations during the 40-year design life of the plant. By assuming a 90 percent availability factor, this number is reduced to 13,200.

Step Load Increase and Decrease of l0 Percent of Full Power

The following discussion conservatively establishes the number of anticipated cycles. The rod control system is normally operated under automatic control. Automatic rod withdrawal is no longer available. Automatic rod insertion accommodates a step load decrease of 10% over the entire power range.

The ±10 percent step change in load demand is a transient which is assumed to be a change in turbine control valve opening due to disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a ±10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent full load, the power range for automatic reactor control (the C-5 interlock at 15% load is no longer associated with the rod control system). In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average

temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control rods are inserted to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The operator manually withdraws the control rods to increase core power (automatic rod withdrawal is no longer available). 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 2,000 times or 50 per year for the 40-year plant design life.

<u>Large Step Load Decrease With Steam Dump</u>

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 safety valves. Thus, since the SNUPPS plants are designed to accept a step decrease of 50 percent from full power the steam dump system provides the heat sink to accept 40 percent of the turbine load. The remaining 10 percent of the total step change is compensated for by the reactor control system (control rods). If a steam dump system was not provided to cope with this transient, there would be such a strong mismatch between what the turbine is asking for and what the reactor is delivering that a reactor trip and lifting of steam generator safety valves would occur.

The number of occurrences of this transient is specified at 200 times or five per year for the 40-year plant design life.

<u>Steady State Fluctuations</u>

The reactor coolant temperature and pressure at any point in the system vary around the nominal (steady state) values. For design purposes, two cases are considered:

a. Initial fluctuations

These are due to control rod cycling during the first 20 full power months of reactor operation. Temperature is assumed to vary by ±3°F and pressure by ±25 psi, once during each 2-minute period. The total number of such occurrences considered is 1.5×10^5. These fluctuations are assumed to occur consecutively, and not simultaneously with the random fluctuations.

b. Random fluctuations

Temperature is assumed to vary by ±0.5°F and pressure by ±6 psi, once every 6 minutes. With a 6-minute period, the total number of occurrences during the plant design life does not exceed 3.0×10^6.

Feedwater Cycling at Hot Shutdown

These transients are assumed to occur when the plant is at no-load conditions, during which intermittent feeding of 32°F feedwater into the steam generators is assumed. Due to fluctuations arising from this mode of operation, the reactor coolant average temperature decreases to a lower value and then immediately begins to return to normal no-load temperature. This transient is assumed to occur 2,000 times over the life of the plant.

Loop Out of Service

The plant may be operated at a reduced power level with a single loop out of service for limited periods of time. This is accomplished by reducing power level and tripping a single reactor coolant pump.

It is assumed that this transient occurs twice per year or 80 times in the life of the plant. Conservatively, it is assumed that all 80 occurrences can occur in the same loop. In other words, it must be assumed that the whole RCS is subjected to 80 transients while each loop is also subjected to 80 inactive loop transients.

Unit Loading and Unloading Between 0 and 15 Percent of Full Power

The unit loading and unloading cases between 0 and 15 percent power are represented by continuous and uniform ramp power changes, requiring 30 minutes for loading and 5 minutes for unloading. During loading, reactor coolant temperatures are increased from the no-load value to the normal load program temperatures at the 15-percent power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot shutdown conditions, with 32°F feedwater cycling. During the 2-hour period following the beginning of loading, the feedwater temperature increases from 32°F to 300°F due to steam dump and turbine

startup heat input to the feedwater. Subsequent to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal requirements, and feedwater temperature decays from 300°F to 32°F.

The number of these loading and unloading transients is assumed to be 500 each during the 40-year plant design life, which is equivalent to about one occurrence per month.

Boron Concentration Equalization

Following any large change in boron concentration in the RCS, spray is initiated in order to equalize concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heaters, thus causing a pressure increase, which will initiate spray at a compensated pressurizer pressure of approximately 2,275 psia. The proportional sprays return the pressure to 2,250 psia and maintain this pressure by matching the heat input from the backup heater until the concentration is equalized. For design purposes, it is assumed that this operation is performed once after each load change in the design load follow cycle. With two load changes per day and a 90-percent plant availability factor over the 40-year design life, the total number of occurrences is 26,400.

Reactor Coolant Pumps Startup and Shutdown

The reactor coolant pumps are started and stopped during routine operations such as RCS venting, plant heatup and cooldown, and in connection with recovery from certain transients such as loop out of service and loss of power. Other (undefined) circumstances may also require pump starting and stopping.

Of the spectrum of RCS pressure and temperature conditions under which these operations may occur, three conditions have been selected for defining transients:

a. Cold condition (70°F and 400 psig)

b. Pump restart condition (100°F and 400 psig)

c. Hot condition (557°F and 2235 psig)

For reactor coolant pump 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.

Occurrences for the pump starting and stopping conditions are given in Table 3.9(N)-1.

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 to approximately 1 percent per minute by the ability of the boron system to dilute the reactor coolant. The reduction in primary coolant temperature is limited by the protection system to about 20°F below the programmed value.

The reduced temperature return-to-power operation is not intended for daily use. It is designed to supply additional plant capabilities when required because of network fault or upset conditions. Hence this mode of operation is not expected to be used more than once a week in practice (2,000 times in 40 years).

Refueling

At the end of plant cooldown, the temperature of the fluid in the RCS is ≤140°F. At this time, the vessel head is removed and the refueling canal is filled. This is done by pumping water from the refueling water storage tank, which is located outside and conservatively assumed to be at 32°F, into the loops by means of the low head safety injection pumps. The refueling water flows directly into the reactor vessel via the accumulator connections and two pathways (1) hot leg recirculation or alternatively (2) the accumulator cold leg loop.

This operation is assumed to occur twice per year or 80 times over the life of the plant.

Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power will be used to heat the reactor coolant to operating temperature (no-load conditions), and the steam generated will be used to perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F per hour design rate.

The number of such test cycles is specified at 20, to be performed at the beginning of plant operating life prior to fuel loading. This transient occurs before plant startup, and the number of cycles is therefore independent of other operating transients. Included in the total number of such test cycles is the full flow test for the turbine-driven auxiliary feedwater pump.

Primary Side Leakage Test

Subsequent to each time the primary system has been opened, a leakage test will be performed. During this test, the primary system pressure is assumed, for design purposes, to be raised to 2,500 psia, with the system temperature above the minimum

temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

In actual practice, the primary system will be pressurized to the normal operating pressure to prevent the pressurizer safety valves from lifting during the leak test.

During this leakage test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1,600 psi. This is accomplished with the steam, feedwater, and blowdown lines closed off. For design purposes, it is assumed that 200 cycles of this test will occur during the 40-year life of the plant.

Secondary Side Leakage Test

During the life of the plant, it may be necessary to check the secondary side of the steam generator (particularly, the manway closure) for leakage. For design purposes, it is assumed that the steam generator secondary side is pressurized to just below its design pressure, to prevent the safety valves from lifting. In order not to exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized. The primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements. It is assumed that this test is performed 80 times during the 40-year life of the plant.

Feedwater Heaters Out of Service

These transients occur when one or more feedwater heaters are taken out of service. During the period of time that the heater(s) is out of service, it is desirable to maintain the plant at full rated thermal load. To accomplish this, first the steam flow is reduced to the amount that will maintain the plant at full rated thermal load when the heater(s) is taken out of service. It takes approximately 10 minutes for plant conditions to reach a new steady-state. Then the heater(s) is taken out of service.

Upset Conditions

The following primary system transients are considered upset conditions:

a. Loss of load (without immediate reactor trip)

b. Loss of power

c. Partial loss of flow

d. Reactor trip from full power

e. Inadvertent RCS depressurization

f. Inadvertent startup of an inactive loop

g. Control rod drop

h. Inadvertent safety injection actuation

i. Operating Basis Earthquake (OBE)

j. Excessive feedwater flow

Loss of Load (Without Immediate Reactor Trip)

This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the reactor protection system. Since redundant means of tripping the reactor are provided as a part of the reactor protection system, transients of this nature are not expected, but are included to ensure a conservative design.

The number of occurrences of this transient is specified at 80 times or two times per year for the 40-year plant design life.

Loss of Power

This transient applies to a blackout situation involving the loss of outside electrical power to the station, assumed to be operating initially at 100-percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are de-energized and, following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater, assumed to be at 32°F, from the auxiliary feedwater system operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times or one per year for the 40-year plant design life.

Partial Loss of Flow

This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooler water then flows

through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 80 times or two times per year for the 40-year plant design life.

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 a result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the reactor protection system causes the control rods to move into the core.

Various moderator cooldown transients associated with reactor trips can occur as a result of excessive feed or steam dump after trip or large load increase. For design purposes, reactor trip is assumed to occur a total of 400 times or 10 times per year over the life of the plant. The various types of trips and the number of occurrences for each are as follows:

a. Reactor trip with no inadvertent cooldown - 230 occurrences.

b. Reactor trip with cooldown but no safety injection - 160 occurrences.

c. Reactor trip with cooldown actuating safety injection - 10 occurrences.

d. Reactor trip with no inadvertent cooldown overspeed.

   For design purposes, 20 occurrences of the reactor trip with no inadvertent cooldown (Case a - 230 occurrences total) are assumed to be accompanied by an emergency turbine overspeed. This situation could be caused by malfunction of the turbine control system following a large step load decrease with steam dump resulting in turbine speed increase past the turbine overspeed trip setpoint. It is assumed that the reactor trips and that the speed increases to 120 percent of nominal, with accompanying proportional increases in generator bus frequency, reactor coolant pump speed, and reactor coolant flow rate.

   Approximately 30 seconds after the reactor trip, the house load is transferred from the generator to the outside bus and a loss of outside

power (blackout) occurs. This is assumed to be covered by the 40 occurrences of the loss of power transient.

Inadvertent RCS 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 an "umbrella" case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

When a pressurizer safety valve opens, and remains open, the system rapidly depressurizes, the reactor trips, and the safety injection system is actuated. Also, the passive accumulators of the safety injection system are actuated when pressure decreases by approximately 1,600 psi, about 12 minutes after the depressurization begins. The depressurization and cooldown are eventually terminated by operator action. All of these effects are completed within approximately 18 minutes. It is conservatively assumed that none of the pressurizer heaters are energized.

With pressure constant and safety injection in operation, boiloff of hot leg liquid through the pressurizer and open safety valve will continue.

For design purposes, this transient is assumed to occur 20 times during the 40-year design life of the plant.

Inadvertent Startup of an Inactive Loop

This transient can occur when a loop is out of service. With the plant operating at maximum allowable power level, the reactor coolant pump in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. This transient is assumed to occur 10 times during the life of the plant.

Control Rod Drop

This transient occurs if a bank of control rods drops to the fully inserted position due to a single component failure. The reactor status depends on the time in core life and the magnitude of the reactivity insertion (see Section 15.4.3). It is assumed that this transient occurs 80 times over the life of the plant.

Inadvertent Safety Injection Actuation

A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head ECCS centrifugal charging pumps. These pumps deliver borated injection to the RCS cold legs. The initial portion of this transient is similar to the reactor trip from full power with no cooldown. Controlled steam dump and feedwater flow after the trip removes core residual heat. Reactor coolant temperature and pressure decrease as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase to the pressurizer power-operated relief valve setpoint and the primary and secondary temperatures to decrease gradually. The transient continues until the operator stops the ECCS charging pumps (including the normal charging pump if not tripped by the safety injection signal - see Section 15.5.1). It is assumed that the plant is then returned to no-load conditions, with pressure and temperature changes controlled within normal limits.

For design purposes, this transient is assumed to occur 60 times over the 40-year design life of the plant.

Operating Basis Earthquake

The mechanical stresses resulting from the OBE are considered on a component basis. Fatigue analysis, where required by the codes, is performed by the supplier as part of the stress analysis report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis.

The number of occurrences of this transient is specified in Table 3.9(N)-1.

Excessive Feedwater Flow

An excessive feedwater flow transient is conservatively defined as an umbrella case to cover occurrence of several events of the same general nature. The postulated transient results from inadvertent opening of a feedwater control valve while the plant is at the hot standby or no-load condition, with the feedwater, condensate, and heater drain systems in operation. The transient discussion below is intended to provide a general description of an event that may occur up to 30 times during the life of the plant. It is not intended to

represent any specific hazard or transient analysis. For specific analyses of this event refer to Sections 3B.4.2.3 and 15.1.2.

It is assumed that the stem of a feedwater control valve fails and the valve immediately reaches the full open position. In the steam generator directly affected by the malfunctioning valve ("failed loop"), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of all operating feedwater pumps. Steam flow is assumed to remain at zero, and the temperature of the feedwater entering the steam generator is conservatively assumed to be 32°F. Feedwater flow is isolated on a reactor coolant low T_{avg} signal (P-4 coincident with low T_{avg}) unless it is bypassed, in which case feedwater isolation occurs on SG high-high water level and the low pressurizer pressure signal actuates the safety injection system. Auxiliary feedwater flow, initiated by the safety injection signal, is assumed to continue with all pumps discharging into the affected steam generator. It is also assumed, for conservatism in the secondary side analysis, that auxiliary feedwater flows to the steam generators not affected by the malfunctioned valve, in the "unfailed loops." Plant conditions stabilize at the values reached in 600 seconds, at which time auxiliary feedwater flow is terminated. The plant is then either taken to cold shutdown, or returned to the no-load condition at a normal heatup rate with the auxiliary feedwater system under manual control.

For design purposes, this transient is assumed to occur 30 times during the life of the plant.

RCS Cold Overpressurization

RCS cold overpressurization occurs during startup and shutdown conditions at low temperature, with or without the existence of a steam bubble in the pressurizer, and is especially severe when the reactor coolant system is in a water-solid configuration. The event is inadvertent, and can potentially occur by any one of a variety of malfunctions or operator errors. All events which have occurred to date may be categorized as belonging to either events resulting in addition of mass (mass input transients) or events resulting in the addition of heat (heat input transients). All of these possible transients are represented by composite, “umbrella” design transients, referred to here as RCS cold overpressurization.

The number of occurrences of this transient is specified as 10 times for the 40-year plant design-operating period.

Emergency Conditions

The following primary system transients are considered emergency conditions:

a. Small LOCA

b. Small steam break

c. Complete loss of flow

Small Loss-of-Coolant Accident

For design transient purposes, the small LOCA is defined as a break equivalent to the severance of a 1-inch inside diameter branch connection. (Breaks smaller than 0.375-inch inside diameter can be handled by the normal makeup system and produce no significant fluid systems transients.) Breaks which are much larger than 1 inch will cause accumulator injection soon after the accident and are regarded as faulted conditions. For design purposes, it is assumed that this transient occurs five times during the life of the plant. It should be assumed that the emergency core cooling system (ECCS) is actuated immediately after the break occurs and subsequently delivers water at a minimum temperature of 32°F to the RCS.

Small Steam Break

For design transient purposes, a small steam break is defined as a break equivalent in effect to a steam safety valve opening and remaining open. This transient is assumed to occur five times during the life of the plant. The following conservative assumptions are used in defining the transients:

a. The reactor is initially in a hot, zero power condition.

b. The small steam break results in immediate reactor trip and ECCS actuation.

c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

d. The ECCS operates at a design capacity and repressurizes the RCS within a relatively short time.

Complete Loss of Flow

This accident involves a complete loss of flow from full power resulting from simultaneous loss of power to all reactor coolant pumps. The consequences of this incident are a reactor trip and turbine trip on undervoltage followed by automatic opening of the steam dump system. For design purposes, this transient is assumed to occur five times during the plant lifetime.

Faulted Conditions

The following primary system transients are considered faulted conditions. Each of the following accidents is evaluated for one occurrence:

a. Reactor coolant pipe break (large LOCA)

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 (SSE)

Reactor Coolant Pipe Break (Large LOCA)

Following rupture of a reactor coolant pipe resulting in a large loss of coolant, the primary system pressure decreases, causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. It is conservatively assumed that the safety injection system is actuated to introduce water at a minimum temperature of 32°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

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 safety injection actuation.

c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

d. The safety injection system operates at design capacity and repressurizes the RCS within a relatively short time.

The above conditions result in the most severe temperature and pressure variations which the primary system will encounter during a steam line break accident.

Feedwater Line Break

This accident involves a double-ended rupture of the main feedwater piping while operating at full power, resulting in the rapid blowdown of one steam generator and the

termination of main feedwater flow to the others. The blowdown is completed in approximately 27 seconds. Conditions were conservatively chosen to give the most severe primary side and secondary side transients. All auxiliary feedwater flow exits at the break.

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.

Control Rod Ejection

This accident is based on the single most reactive control rod being instantaneously ejected from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the RCS, such that the pressurizer safety valves will lift, and also causes a more severe temperature transient in the loop associated with the affected region than in the other loops. For conservatism, the analysis is based on the reactivity insertion and does not include the mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head vacated by the ejected rod.

Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube, resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to the resulting safety injection signal. In addition, safety injection actuation automatically isolates the feedwater lines, by tripping all feedwater pumps (this design feature is not credited in the accident analyses of Chapter 15) and closing the feedwater isolation valves. When this accident occurs, some of the reactor coolant blows down into the affected steam generator, causing the shell side level to rise. The primary system pressure is reduced below the secondary safety valve setting. Subsequent recovery procedures call for isolation of the steam line leading from the affected steam generator. This accident will result in a transient which is no more mechanically severe than that associated with a reactor trip from full power. Therefore, it requires no special treatment insofar as fatigue evaluation is concerned, and no specific number of occurrences is postulated.

Safe Shutdown Earthquake

The mechanical dynamic or static equivalent loads due to the vibratory motion of the SSE are considered on a component basis.

Test Conditions

The following primary system transients under test conditions are discussed:

a. Primary side hydrostatic test

b. Secondary side hydrostatic test

c. Tube leakage test

Primary Side Hydrostatic Test

The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydrostatic test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3,107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3,107 psig coincident with a steam generator secondary side pressure of 0 psig. The RCS is designed for 10 cycles of these hydrostatic tests, which are performed prior to plant startup.

Additional hydrostatic tests will be performed to meet the inservice inspection requirements of the ASME Code, Section XI, Articles IWB-2500 and IWB-5222. A total of four such tests is expected. The increase in the fatigue usage factor caused by these tests is easily covered by the conservative number (200) of primary side leakage tests that are considered for design.

Secondary Side Hydrostatic Test

The secondary side of the steam generator is pressurized to 1,481 psig with a minimum water temperature of 120°F coincident with the primary side at 0 psig.

For design purposes, it is assumed that the steam generator will experience 10 cycles of this test.

These tests may be performed either prior to plant startup, or subsequently following shutdown for major repairs or both.

Tube Leakage Test

During the life of the plant, it may be necessary to check the steam generator for tube leakage and tube-to-tube sheet leakage. This is done by visual inspection of the underside (channel head side) of the tube sheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdowns.

For these tests, the secondary side of the steam generator is pressurized with water (maximum secondary side test pressure is 840 psig), initially at a relatively low pressure, and the primary system remains depressurized. The underside of the tube sheet is examined visually for leaks. If any are observed, the secondary side is then depressurized and repairs made by tube plugging.

The total number of tube leakage test cycles is defined as 800 during the 40-year life of the plant. Following is a breakdown of the anticipated number of occurrences at each secondary side test pressure:

Test Pressure (psig)	Number of Occurrences
200	400
400	200
600	120
840	80

Both the primary and secondary sides of the steam generators will be at ambient temperature during these tests.

3.9(N).1.2 Computer Programs Used in Analysis

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of seismic Category I components and equipment. Computer programs (a) through (c) are described and verified in References 1 and 2. Computer program (d) is described and verified in Reference 15.

a. WESTDYN- static, dynamic, and fatigue analysis of redundant piping systems.

b. WESAN - reactor coolant loop equipment support structures analysis and evaluation.

c. WECAN - finite element structural analysis.

d. BWSPAN - finite element structural analysis of piping and structural systems.

e. EMDAC-FEA - general purpose finite element code. (Reference 10 and 11)

3.9(N).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 Section 3.9(N).2.

3.9(N).1.4 Considerations for the Evaluation of the Faulted Condition

3.9(N).1.4.1 Loading Conditions

The structural stress analyses performed on the RCS consider the loadings specified in Table 3.9(N)-2. These loads result from thermal expansion, pressure, weight, OBE, SSE, and design basis LOCA, and plant operational thermal and pressure transients.

3.9(N).1.4.2 Analysis of the Reactor Coolant Loop and Supports

The loads used in the analysis of the reactor coolant loop piping are described in detail below.

Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations, in accordance with the ASME Code.

The term operating pressure is used in connection with determination of the system deflections and support forces. The steady state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at change in direction or flow area.

Weight

A deadweight analysis is performed to meet Code requirements by applying a 1.0g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

Seismic

The forcing functions for the reactor coolant loop seismic piping analyses are derived from dynamic response analyses of the containment building subjected to seismic ground motion. Input is in the form of floor response spectrum curves at various elevations within the containment building.

For the OBE and SSE seismic analyses, 2- and 4-percent critical damping, respectively, are used in the reactor coolant loop supports system analysis.

In the response spectrum method of analysis, the total response loading obtained from the seismic analysis consists of two parts--the inertia response loading of the piping system and the differential anchor movements loading. Two sets of seismic moments

are required to perform an ASME Code analysis. The first set includes only the moments resulting from inertia effects, and these moments are used in the resultant moment (M_i) value for Equations 9 and 13 of NB-3650. The second set includes the moments resulting from seismic anchor motions and are used in Equations 10 and 11 of NB-3650. Differential anchor movement is discussed in Section 3.7(N).

Loss-of-Coolant Accident

Blowdown loads are developed in the broken and unbroken reactor coolant loops as a result of transient flow and pressure fluctuations following a postulated pipe break in one of the reactor coolant loops. 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.

Broken loop time history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time-dependent hydraulic forcing functions used in the analysis of the reactor coolant loop for each break case. For a further description of the hydraulic forcing functions, refer to Section 3.6.

Transients

The Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are tabulated in Section 3.9(N).1.1.

The vertical thermal growth of the reactor pressure vessel nozzle centerlines is considered in the thermal analysis to account for equipment nozzle displacement as an external movement.

The hot moduli of elasticity E, the coefficient of thermal expansion at the metal temperature α, the external movements transmitted to the piping due to vessel growth, and the temperature rise above the ambient temperature ΔT, define the required input data to perform the flexibility analysis for thermal expansion.

To provide the necessary high degree of integrity for the RCS, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions.

3.9(N).1.4.3 Reactor Coolant Loop Analytical Models and Methods

The analytical methods used in obtaining the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectra method for seismic dynamic analysis, and time history integration method for the LOCA dynamic analysis.

The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffnesses of auxiliary line piping which affect the system, and the stiffness of piping restraints. 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.

Static

The reactor coolant loop/supports system model, constructed for the WESTDYN computer program, is represented by an ordered set of data which numerically describe the physical system. Figure 3.9(N)-1 shows an isometric line schematic of this mathematical model. The steam generator and reactor coolant pump vertical and lateral support members are described in Section 5.4.14.

The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity E, the coefficient of thermal expansion 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. Due to the symmetry of the static loadings, the reactor pressure vessel centerline is represented by a fixed boundary in the system mathematical model. The vertical thermal growth of the reactor vessel nozzle centerline is considered in the construction of the model.

The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this matter, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multipled by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for weight, thermal, and general pressure loading conditions are obtained by using the WESTDYN-7 computer program. The derivation of the hydraulic loads for the LOCA analysis of the loop is covered in Section 3.6.2.

Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. All of the piping loops are included in the system model. The effect of the equipment motion on the reactor coolant loop/ supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The replacement steam generator is represented by nine discrete masses. The lower mass is located near the intersection of the centerlines of the inlet and outlet nozzles of the replacement steam generator. The other masses are located at various locations along the center line of the replacement steam generator.

The reactor coolant pump is 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 is represented by nine 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 when the plant is at power and under the rapid motions of the reactor coolant loop components that occur from the dynamic loadings and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The response spectra method employs the lumped mass technique, linear elastic properties, and the principle of modal superposition. The floor response spectra are applied along both horizontal axes and the vertical axis simultaneously.

From the mathematical description of the system, the overall stiffness matrix [K] is developed from the individual element stiffness matrices, using the transfer matrix method. After deleting the rows and columns representing rigid restraints, the stiffness matrix is revised to obtain a reduced stiffness matrix $[K_R]$ associated with mass degrees of freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are determined. The modal participation factor matrix is computed and combined with the appropriate response spectra value to give the modal amplitude for each mode.

The modal amplitudes are then converted to displacements in the global coordinate system and applied to the corresponding mass point. From these data, the forces, moments, deflections, rotations, support reactions, and piping stresses are calculated for all significant modes in each direction.

The total response (i.e., forces, moments, etc.) in each direction is obtained by combining the contributions of the significant modes, using the methods described in Section 3.7(N).2.7. The combined total seismic response is then calculated using the square-root-sum-of-the-squares method applied to the resultant unidirectional responses.

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 piping at the postulated break location. Modifications include addition of the mass characteristic of the piping and equipment. To obtain the proper dynamic solution, two masses, each containing six dynamic degrees of freedom and located on each side of the break, are included in the mathematical model. The natural frequencies and eigenvectors are determined from this broken loop model.

The time-history hydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural lumped-mass node points.

The dynamic structural solution for the full power LOCA and steam line break is obtained by using a modified-predictor-corrector-integration technique and normal mode theory.

When elements of the system can be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple nonlinear elements can be applied at the same node, if necessary.

The time-history solution is performed in a subprogram of WESTDYN. The input to this subprogram consists of the natural frequencies, normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified reactor coolant loop dynamic model are determined with the WESTDYN-7 program. To properly simulate the release of the strain energy in the pipe, the internal forces, due to the initial steady state hydraulic forces, thermal forces, and weight forces, in the system at the postulated break location, are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response of the system from the static equilibrium position). The time history displacement solution of all dynamic degrees of freedom is obtained using a subprogram of WESTDYN and employing 4-percent critical damping.

The LOCA displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the reactor coolant loop. The LOCA analysis of the reactor vessel includes all the forces acting on the vessel, including internal reactions, cavity

pressure loads, and loop mechanical loads. The reactor vessel analysis is described in Section 3.9(N).1.4.6.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

The support loads are computed by multiplying the support stiffness matrix and the displacement vector at the support point. The support loads are used in the evaluation of the supports.

The time-history displacements are used as input to WESTDYN to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation, the displacements are treated as imposed deflections on the reactor coolant loop masses. The results of this solution are used in the piping stress evaluation.

Transients

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into three parts--a uniform, a linear, and a nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall, and the nonlinear portion causes a skin stress.

The transients, as defined in Section 3.9(N).1.1, are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat conduction program is used to solve the thermal transient problem. The pipe is represented by at least 50 elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time-varying heat transfer due to free and forced convection.

The outer surface is assumed to be adiabatic while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. An arbitrary temperature distribution across the wall is shown in Figure 3.9(N)-2.

The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. This integration is performed for all time steps so that T_A is determined as a function of time.

$$T_A(t) = \frac{1}{H}\int_0^H T(X, t)dX$$

The range of temperature between the largest and smallest value of T_A is used in the flexibility analysis to generate the moment loadings caused by the associated temperature changes.

The thermal moment about the mid-thickness of the wall caused by the temperature distribution through the wall is equal to:

$$M = E\alpha \int_0^H \left(X - \frac{H}{2}\right) T(X, t) dX$$

The equivalent thermal moment produced by the linear thermal gradient as shown in Figure 3.9(N)-2 about the mid-wall thickness is equal to:

$$M_L = E\alpha \frac{\Delta T_1}{12} H^2$$

Equating M_L and M, the solution for ΔT_1 as a function of time is:

$$\Delta T_i(t) = \frac{12}{H^2} \int_0^H \left(X - \frac{H}{2}\right) T(X, t) dX$$

The maximum nonlinear thermal gradient, ΔT_2, will occur on the inside surface and can be determined as the difference between the actual metal temperature on this surface and half of the average linear thermal gradient plus the average temperature.

$$\Delta T_{2I}(t) = [T(0, t) - T_A(t)] - \left[\frac{\Delta T_1(t)}{2}\right]$$

<u>Load Set Generation</u>

A load set is defined as a set of pressure loads, moment loads, and through-wall thermal effects at a given location and time in each transient. The method of load set generation is based on Reference 3. The through-wall thermal effects are functions of time and can be subdivided into four parts:

a. Average temperature (T_A) is the average temperature through-wall of the pipe which contributes to general expansion loads.

b. Radial linear thermal gradient which contributes to the through-wall bending moment (ΔT_1).

c. Radial nonlinear thermal gradient (ΔT_2) which contributes to a peak stress associated with shearing of the surface.

d. Discontinuity temperature (T_A - T_B) represents the difference in average temperature at the cross-sections on each side of a discontinuity.

Each transient is described by at least two load sets, representing the maximum and minimum stress state during each transient. The construction of the load sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient.

a. ΔT_1

b. ΔT_2

c. $\alpha_A T_A - \alpha_B T_B$

d. Moment loads due to T_A

e. Pressure loads

This procedure produces at least twice as many load sets as transients for each point.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus ensuring the most conservative combination of seismic loads are used in the stress evaluation.

For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors, and cumulative usage factors are calculated. The WESTDYN program is used to perform this analysis in accordance with the ASME Code, Section III, Subsection NB-3650. Since it is impossible to predict the order of occurrence of the transients over a 40-year life, it is assumed that the transients can occur in any sequence. This is a very conservative assumption.

The combination of load sets yielding the highest alternating stress intensity range is used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles < 10^6 are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9(N).1.4.4 Primary Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models. The

equipment support structure models are dual-purpose since they are required: 1) to quantitatively represent the elastic restraints which the supports impose upon the loop and 2) to evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

Models for the STRUDL computer program are constructed for the steam generator lower, steam generator upper lateral, reactor coolant pump lower, and pressurizer supports. The reactor vessel supports are modeled, using the WECAN computer program. Structure geometry, topology, and member properties are used in the modeling.

A description of the supports is found in Section 5.4.14. Detailed models are developed, using beam elements and plate elements, where applicable.

The respective computer programs are used with these models to obtain support stiffness matrices and member influence coefficients for the steam generator, reactor coolant pump, pressurizer, and reactor vessel supports. Unit force along and unit moment about each coordinate axis are applied to the models at the equipment vertical centerline joint. Stiffness analyses are performed for each unit load for each model.

Joint displacements for applied unit loads are formulated into flexibility matrices. These are inverted to obtain support stiffness matrices which were included in the reactor coolant loop model.

Loads acting on the supports obtained from the reactor coolant loop analysis, support structure member properties, and influence coefficients at each end of each member are input into the WESAN program.

For each support case used, the following is performed:

a. Combine the various types of support plane loads to obtain operating condition loads (normal, upset, emergency, or faulted).

b. Multiply member influence coefficients by operating condition loads to obtain all member internal forces and moments.

c. Solve appropriate stress or interaction equations for the specified operating condition. Maximum normal stress, shear stress, and combined load interaction equation values are printed as a ratio of maximum actual values divided by limiting values. ASME Code, Section III, Subsection NF stress and interaction equations are used with limits for the operating condition specified.

The reactor vessel support structure is analyzed for all loading conditions, using a finite element model. Vertical and horizontal forces delivered to the support structures from

the reactor vessel shoe are applied to the structure, and element stresses and concrete forces obtained.

3.9(N).1.4.5 Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop includes the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is ANS Safety Class 1 and the pressure boundary meets the requirements of the ASME Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3.9(N)-2. The equipment is analyzed for: 1) the normal loads of deadweight, pressure and thermal, 2) mechanical transients of Operating Basis Earthquake, Safe Shutdown Earthquake, and pipe ruptures, and 3) pressure and temperature transients outlined in Section 3.9(N).1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant 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 deviations where the actual load is larger than the umbrella load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator and pressurizer are performed using 2-percent damping for the OBE and 4-percent damping for the SSE. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump internals is performed, using the damping for bolted steel structures, that is, 4 percent for the OBE and 7 percent for the SSE (2 percent for OBE and 4 percent for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump since the main flange, motor stand, and motor are all bolted assemblies (see Section 5.4). The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of the ASME Code, Section III.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the line connects to the primary system piping are orificed to a 3/8-inch hole. This hole restricts the flow such

that loss through a severance of one of these lines can be made up by normal charging flow.

3.9(N).1.4.6 Reactor Vessel Support LOCA Loads

The LOCA analysis which is performed for the reactor vessel support loads includes nonaxisymmetric pressure distributions on the internals and on the vessel exterior walls. A detailed dynamic model of the reactor vessel and internals is prepared which includes the stiffnesses of the reactor vessel support and the attached piping. Hydraulic forces are developed in the internals for the break at the reactor vessel nozzle; these forces are characterized by time-dependent forcing functions on the vessel and core barrel. In the derivation of these forcing functions, the fluid-structure (or hydroelastic) interaction in the downcomer region between the barrel and the vessel is taken into account. The break at the vessel nozzle also allows an asymmetric pressure distribution, and a subsequent force on the side of the vessel is calculated on a time-history basis for these asymmetric loads. As a result of the pipe break, loop mechanical loads are also applied to the vessel.

The loads from these three sources--the internals reactions, reactor cavity pressure loads, and the loop mechanical forces--are applied simultaneously in a nonlinear elastic dynamic time-history analysis on the model of the vessel, reactor vessel supports and internals. The results of this analysis are the dynamic loads on the reactor vessel supports and vessel time-history displacements. The maximum loads are combined with other applicable loads, such as seismic and deadweight, and applied statically to the vessel support structure. The maximum stresses in the support are calculated and compared to faulted condition stress allowables given in Section 3.9(N).1.4.7.

3.9(N).1.4.7 Stress Criteria for Class 1 Components and Component Supports

All Class 1 components and supports are designed and analyzed for the design, normal, upset, and emergency conditions to the rules and requirements of the ASME Code, Section III. The design analysis or test methods and associated stress or load allowable limits that will be used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code with supplementary option outlined below:

a. The test 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 reactor vessel support pads are qualified using the test option. The reactor pressure vessel support pads are designed to restrain unidirectional horizontal motion, in addition to supporting the vessel. The design of the supports allows radial growth of the vessel but restrains the vessel from horizontal displacements, since tangential displacement of the vessel is prevented at each vessel nozzle.

To duplicate the loads that act on the pads during faulted conditions, the tests, which utilized a one-eighth linear scale model, were performed by applying a unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted by a support shoe which was mounted to the test fixture.

The above modeling and application of load thus allows the maximum load capacity of the support pads to be accurately established. The test load, L_T, was then determined by multiplying the maximum collapse load by 64 (ratio of prototype area to model area) and including 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 the test load method insure that the experimentally obtained value for L_T is accurate and that the support pad design is adequate for its intended function.

b. In the design of component supports, member compressive axial loads shall be limited to 0.67 times the critical buckling strength, per F-1370(c) of the ASME Code, Section III.

Loading combinations and allowable stresses for ASME Code, Section III, Class 1 components and supports are given in Tables 3.9(N)-2 and 3.9(N)-3.

The methods of load combination for each operating condition are as follows:

Design: Loads are combined by algebraic sum.

Normal, Upset: These loads are used in the fatigue evaluation in accordance with the methods prescribed in the ASME Code. Loadsets are defined for each transient, including the OBE, and are combined such that the maximum stress ranges are obtained without regard to the order in which the transients occur. (This is discussed in more detail in Section 3.9(N).1.4.3).

Emergency: Loads are combined by algebraic sum.

Faulted: For primary equipment, primary equipment supports, and Class 1 branch lines, LOCA and SSE loads are combined using the square-root-of-the-sum-of-the-squares (SRSS) method on a load component basis (i.e., the LOCA F_x is combined with the SSE F_x by SRSS, the LOCA F_y is combined with the SSE F_y by SRSS, and likewise for F_z, M_x, M_y, M_z). The sustained loads, such as weight effects, are combined with the SRSS result by algebraic sum.

For reactor coolant loop piping, the deadweight moments were added to the LOCA moments prior to the SRSS combination of the LOCA and SSE loads.

Full structural weld overlays (FSWOLs) have been installed on Class I piping locations. FSWOL adds two loads to the piping, a new weight load, and a load caused by shrinkage which was modeled as a thermal stress. Weight load additions will be treated as any other weight loads. The shrinkage loads were required to be calculated by the ASME Code Cases used to apply the welds. ASME Section III does not require cold loads, such as shrinkage or pipe spring, to be included in the loading combinations. The shrinkage load acts in an opposite direction from the hot thermal load, therefore omission of this load from operational pipestress loads for components and component supports is conservative. In some cases there are components and component supports analyzed for non-operational (cold conditions) stresses, or the non-operational stresses are limiting. In these cases the loadings caused by shrinkage must be taken into account. Shrinkage from the FSWOLs will be noted in calculations and design drawings where shrinkage results in increased stresses. In these cases the loading from shrinkage should be treated as a thermal stress and be combined as an algebraic sum.

3.9(N).2 DYNAMIC TESTING AND ANALYSIS

3.9(N).2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

A preoperational piping vibration and dynamics effects testing program will be conducted for the reactor coolant loop/supports system during startup functional testing of the SNUPPS units. The purpose of these tests will be to confirm that the systems have been adequately designed and supported for vibration as required by Section III of the ASME Code, Paragraph NB-3622.3.

The preoperational piping vibration and dynamic effects test program for the primary coolant loop system (this includes the hot legs, cold legs, crossover legs, reactor coolant pumps, and steam generators) at the SNUPPS units is as follows:

a. The primary coolant loop system as defined previously will be instrumented with accelerometers to measure the dynamic response of the system during normal and transient operating conditions. In addition to normal steady-state operation, the test conditions will include steady-state operation with various combinations of reactor coolant pumps in operation and transient conditions due to the starting and tripping of the reactor coolant pumps.

b. The test data will be analyzed to determine the maximum alternating stress induced in the piping due to the measured vibration. This alternating stress will be compared to acceptance criteria based on one half the endurance limit at 10^6 cycles, defined in the ASME Code.

c. In the event that the measured vibration is found to be unacceptable based on the comparison with the acceptance criteria, appropriate corrective action will be implemented. This may consist of either:

1. Further testing or analysis to demonstrate that the observed levels do not cause ASME stress and fatigue limits to be exceeded.

2. System modification to eliminate the unacceptable vibration with subsequent test verification.

It should be noted that the layout, size, etc., of the reactor coolant loop piping used in the SNUPPS units are very similar to those employed in Westinghouse plants now in operation. The operating experience that has been obtained from these plants indicates that the reactor coolant loop piping is adequately designed and supported to minimize vibration. In addition, vibration levels of the reactor coolant pump, which is the only mechanical component that could cause vibration of the reactor coolant loop piping, are held to acceptable limits.

Thus, excessive vibration of the reactor coolant loop piping should not be present. However, as added assurance that excessive vibration is not present in the SNUPPS units, the reactor coolant loop system will be subjected to the test program as discussed previously. Visual inspections of the reactor coolant loop pressurizer surge line piping, performed prior to initial criticality, verify that there will not be excessive vibration of the surge line.

3.9(N).2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

The operability of Category I mechanical equipment must be demonstrated if the equipment is determined to be active, i.e., mechanical operation is relied on to perform a safety function. The operability of active Class 2 and 3 pumps, active Class 1, 2, or 3 valves, and their respective drives, operators, and vital auxiliary equipment is shown by satisfying the criteria given in Section 3.9(N).3.2. Other active mechanical equipment is shown operable by either testing, analysis, or a combination of testing and analysis. The operability programs implemented on the other active equipment are similar to the program described in Section 3.9(N).3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10(N) for electrical equipment are used to demonstrate operability if the component is mechanically or structurally complex such that its response cannot be adequately predicted analytically. Analysis may be used if the equipment is amenable to modeling and dynamic analysis.

Inactive seismic Category I equipment is shown to have structural integrity during all plant conditions in one of the following manners: 1) by analysis satisfying the stress criteria applicable to the particular piece of equipment or 2) by test showing that the equipment retains its structural integrity under the simulated test environment.

A list of seismic Category I equipment and the method of qualification used is provided in Table 3.2-1.

3.9(N).2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady State Conditions

The vibration characteristics and behavior due to flow-induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed-form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be practically performed independent of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitations depends on many factors, such as type and location of component and flow conditions. The effects of these forcing functions have been studied from tests performed on models and prototype plants as well as component tests (Ref. 6, 7, 8, and 14).

The Indian Point No. 2 plant (Docket No. 50-247) has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant (Docket No. 50-344) instrumentation program and the Sequoyah No. 1 plant (Docket No. 50-327) instrumentation program provides prototype data applicable to SNUPPS (Ref. 6, 8, and 14).

The SNUPPS plants are similar to Indian Point No. 2; the only significant differences are the modifications resulting from the use of 17 x 17 fuel, replacement of the annular thermal shield with neutron shielding pads, and the change to the UHI-style inverted top hat support structure configuration. These differences are addressed below.

a. 17 x 17 fuel

   The only structural change in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly is the guide tube. The new 17 x 17 guide tubes are stronger and more rigid, hence they are less susceptible to flow-induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation of internals vibration is expected from the vibration with the 15 x 15 fuel assemblies.

b. Neutron shielding pads lower internals

The primary cause of core barrel excitation is flow turbulence, generated in the downcomer annulus (Ref. 8). The vibration levels due to core barrel excitation for Trojan and SNUPPS, both having neutron shielding pads, are expected to be similar. The coolant inlet density of SNUPPS is slightly lower than Trojan, and the flow rate is slightly higher. Scale model tests show that the core barrel vibration varies as velocity is raised to a small power (Ref. 7). The difference in fluid density and flow rate results in approximately 4 percent higher core barrel vibration for SNUPPS than for Trojan. However, scale model test results (Ref. 7) and results from Trojan (Ref. 6) show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information and the fact that low core barrel stresses and large safety margins were measured at Indian Point No. 2 (thermal shield configuration) lead to the conclusion that stresses less than or equal to those of Indian Point No. 2 will result on the SNUPPS internals.

c. UHI-style inverted top hat upper support configuration

The components of the upper internals are excited by turbulent forces due to axial and crossflows in the upper plenum and by pump-related excitations (Ref. 6 and 8). Sequoyah and SNUPPS have the same basic upper internals configuration; therefore, the general vibration behavior is not changed. The SNUPPS upper internals adequacy has been determined from data from instrumented plant tests at Sequoyah No. 1, scale model tests, and numerous operating plants. The results of testing at Sequoyah No. 1 (Ref. 14) showed that the components are excited by flow-induced and pump-related excitations. Analyses of the data indicate that the instrumented components have adequate factors of safety, the random flow-induced responses are adequately predicted by scale models, and that the margins are higher with the core in place than during hot functional testing.

In addition, the SNUPPS upper internals configuration was tested in scale model tests, using the same modeling techniques as for the scale model tests of the UHI configuration. The responses of the SNUPPS upper internals have been calculated using the Sequoyah No. 1 and scale model information. The results show adequate factors of safety for all components.

The original test and analysis of the four-loop configuration is augmented by References 6, 7, 8, and 14 to cover the effects of successive hardware modifications.

3.9(N).2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the SNUPPS reactor internals design configuration is well characterized, as was discussed in Section 3.9(N).2.3, it is not considered necessary to conduct

instrumented tests of the SNUPPS hardware. The recommendations of Regulatory Guide 1.20 are satisfied by conducting the confirmatory pre- and post-hot functional examination for integrity. This examination will include in excess of 30 features illustrated in Figure 3.9(N)-3 with special emphasis on the following areas.

a. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

b. The lateral, vertical, and torsional restraints provided within the vessel.

c. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

d. Those other locations on the reactor internal components which are similar to those which were examined on the prototype Indian Point No. 2, and on Trojan and Sequoyah No. 1.

e. The inside of the vessel will be inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign material are in evidence.

A particularly close inspection will be made on the following items or areas, using a 5X or 10X magnifying glass, where applicable.

a. Lower internals

   1. Upper barrel to flange girth weld.

   2. Upper barrel to lower barrel girth weld.

   3. Upper core plate aligning pin. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.

   4. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

   5. Baffle assembly locking devices. Check for lockweld integrity.

   6. Lower barrel to core support girth weld.

   7. Neutron shielding pads screw locking devices and dowel pin lockwelds. Examine the interface surfaces for evidence of tightness. Check for lockweld integrity.

   8. Radial support key welds.

9. Insert screw locking devices. Examine soundness of lockwelds.

10. Core support columns and instrumentation guide tubes. Check the joints for tightness and soundness of the locking devices.

11. Secondary core support assembly weld integrity.

12. Lower radial support keys and inserts. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. Subsequent to the hot functional testing, the bearing surfaces of the key and keyway will show burnishing, buffing, or shadow marks which indicate pressure loading and relative motion between these parts. Minor scoring of engaging surfaces is also possible and acceptable.

13. Gaps at baffle joint. Check gaps between baffle-to-baffle joints.

b. Upper internals

1. Thermocouple conduits, clamps, and couplings.

2. Guide tube, support column, and thermocouple assembly locking devices.

3. Support column and thermocouple conduit assembly clamp welds.

4. Upper core plate alignment inserts. Examine bearing surface for shadow marks, burnishing, buffing, or scoring. Check the locking devices for integrity of lockwelds.

5. Thermocouple conduit fitting locktab and clamp welds.

6. Guide tube enclosure and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals will be subjected to a total operating time at greater than normal full-flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition, there will be some operating time with only one, two, and three pumps operating.

Pre- and post-hot functional inspection results serve to confirm that the internals are well behaved. When no signs of abnormal wear or harmful vibrations are detected and no apparent structural changes take place, the four-loop core support structures are considered to be structurally adequate and sound for operation.

3.9(N).2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The response of reactor internals components due to an excitation produced by complete severance of a branch line pipe is analyzed. Assuming a pipe break occurs in a very short period of time, e.g., 1 millisecond, the rapid drop of pressure at the break produces a disturbance that propagates along the primary loop and excites the internal structures.

Mathematical Model of the Reactor Pressure Vessel (RPV) System

The mathematical model of the RPV system is a three-dimensional nonlinear finite element model, which represents dynamic characteristics of the reactor vessel/internals/fuel in the six geometric degrees of freedom. The finite element 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 retrained by reactor vessel supports and by the attached primary coolant piping. The reactor vessel support system is represented by stiffness matrices.

The second submodel represents the reactor core barrel assembly, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first submodel, and is connected to it by a stiffness matrix at the internal support ledge. The core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radical support locations.

The third and innermost submodel represents the upper support plate, guide tubes, support columns, upper and lower core plates, and the fuel. This submodel is connected to the first and second submodels by stiffness matrices and nonlinear elements.

The computer code, which is used to determine the response of the reactor vessel and its internals, is a general-purpose finite element code. In the finite element approach, the structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing the appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for the dynamic solution of the differential equation of motion for the structure:

$$[M]\{\ddot{U}\} + [D]\{\dot{U}\} + [K]\{U\} = \{F\} \quad \text{(Equation 1)}$$

where: [M] = Global inertia matrix

[D] = Global damping matrix

[K] = Global stiffness matrix

{Ü} = Acceleration array

{Ú} = Velocity array

{U} = Displacement array

{F} = Force array, including impact, thrust and hydraulic forces, constraints, and weight.

The finite element code solves Equation (1) using a direct time integration solution. The first time step performs a static solution of Equation (1) to determine the initial displacements of the structure due to deadweight and normal operating hydraulic forces. After the initial time step, the dynamic solution of Equation (1) is calculated, Time-history nodal displacements and impact forces are stored for post-processing.

The following typical discrete elements are used to represent the reactor vessel and internals components:

- Three-dimentional elastic pipe
- Three-dimensional mass with rotary inertia
- Three-dimensional beam
- Three-dimensional linear spring
- Concentric impact element
- Linear impact element
- 6 X 6 stiffness matrix
- 18 Card stiffness matrix
- 18 Card mass matrix

- Three-dimensional friction element

Analytical Methods

The RPV system finite element model as described above was used to perform the loss-of-coolant accident (LOCA) analysis. Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant reaults in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wave front with low pressure on one side and high pressure on the other. The wave front translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The LOCA loads applied to the RPV system consist of: (1) reactor internal hydraulic loads (vertical and horizontal), and (2) reactor coolant mechanical loads. All loads are calculated individually and combined in a time-history manner.

RPV Internal Hydraulic Loads

Depressurization waves propogate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle.

After a postulated break in the cold leg, the depressurization path for waves enter the reactor vessel through the inlet nozzle into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, around, and down the downcomer annulus, then up through the region circumferentially enclosed by the core barrel, that is, the fuel region.

The region of the downcomer annulus close to the break depressurizes rapidly, but because of restricted flow areas and finite wave speed (approximately 3,000 feet per second) the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and RPV. As the depressurzation wave propagates around the downcomer annulus and up through the core, the barrel differential pressure reduces, and similarly, the resulting hydraulic forces drop.

In the case of the postulated break in the hot leg, the waves follow dissimilar depressurization path, passing through the outlet nozzle and directly into the upper internals region, depressurizing the core and entering the downcomer annulus from the bottom exit of the core barrel. Thus, after a break in the hot leg, the downcomer annulus would be depressurized with very little difference in pressure across the outside diameter of the core barrel.

A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the downcomer annulus is not directly involved) and internal differential pressures are not as large as for a cold leg

break. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for a hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves would continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX computer code (Reference 9) calculates the hydraulic transients within the entire primary coolant system. It considers the subcooled transition, and two-phase (saturated) blow-down regimes. The MULTIFLEX program employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture.

The MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator systems. A beam model of the core support barrel is developed from the structural properties of the core barrel. In this model, the cylindrical barrel is vertically divided into various segments and the pressure, as well as the wall motions, are projected onto the plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into segments; each segment consists of three separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces that act on the 10 mass points of the beam model. Each flexible wall is bounded on either side of the hydraulic flow path. The motion of flexible walls is determined by solving the global equations of motion for the masses representing the forced vibration of an undamped beam.

Reactor Coolant Loop Mechanical Loads

The reactor coolant loop mechanical loads are applied to the RPV nozzles by the primary coolant loop piping. The loop mechanical loads result from the release of normal operating forces present in the pipe prior to the separation as well as transient hydraulic forces in the reactor coolant system. The magnitudes of the loop release forces are determined by performing a reactor coolant loop analysis for normal operating loads (pressure, thermal, and deadweight). The loads existing in the pipe at the postulated break location are calculated and are “released” at the initiation of the LOCA transient by application of the loads to the broken piping ends. These forces are applied with the ramp time of 1 millisecond because of the assumed instantaneous break opening time. For breaks in the branch lines, the force applied at the reactor vessel would be insignificant. The restraints on the main coolant piping would eliminate any force to the reactor vessel caused by a break in the branch line.

Results of the Analysis

The severity of a postulated break in a reactor vessel is related to three factors: the distance from the reactor vessel to the break location, the break opening area, and the break opening time. The nature of the decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of

simialr area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reactor vessel. The LOCA hydraulic and mechanical loads described in the previous sections were applied to the model of the RPV system.

The results of LOCA analysis include time-history displacements and nonlinear impact forces for all major components. The time-history displacements of the upper core plate, lower core plate, and core barrel at the upper core plate elevation are provided as inputs for the reactor core evaluations. The impact forces calculated at the vessel / internals interfaces are used to evaluate the strutural integrity of the reactor vessel and its internals. Component linear forces are also calculated using the appropriate post-processors.

Seismic Evaluation

The nonlinear dynamic seismic analysis of the RPV system uses the RPV system model described above and the synthesized time-history accelerations. The only difference between the seismic and LOCA model is that in the seismic model fluid-solid interactions are represented by hydrodynamic mass matrices in the downcomer region (between the core barrel and reactor vessel). In LOCA analysis, the fluid-solid interactions are accounted for through the hydraulic forcing functions generated by the MULTIFLEX code.

Seismic Results

The results of system seismic analysis include time-history displacements and impact forces for all major components. The time-history displacements of the upper core plate, lower core plate, and core barrel at the upper core plate elevation are provided as input for the reactor core evaluations. The impact forces calculated at the vessel / internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals.

Components Subjected to Transverse Excitations

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

Core Barrel - For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

The barrel is then analyzed for dynamic buckling, using the following conservative assumptions:

a. The effect of the fluid environment is neglected.

b. The shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

a. The core barrel is analyzed as a shell with two variable sections to model the support flange and core barrel.

b. The barrel with the core and thermal shielding pads is analyzed as a beam elastically supported at the lower radial support and the dynamic response is obtained.

Guide Tubes - The guide tubes in closest proximity to the outlet nozzle of the ruptured loop are the most severely loaded during a blowdown. The transverse guide tube forces decrease with increasing distance from the ruptured nozzle location.

All of the guide tubes are designed to maintain the function of the control rods for a break size of 144 in.^{2} and smaller. No credit for the function of the control rods is assumed for break size areas above 144 in.^{2}. However, the design of the guide tube will permit control rod operation in all but four control rod positions, which is sufficient to maintain the core in a subcritical configuration, for break sizes up to a double-ended hot leg break. This double-ended hot leg break imposes the limiting lateral guide tube loading.

Upper Support Columns - Upper support columns located close to the broken nozzle during hot leg break will be subjected to transverse loads due to crossflow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes, i.e., by taking into consideration the increase in flow across the column during the accident. The columns are studied as beams with variable sections, and the resulting stresses are obtained, using the reduced section modulus and appropriate stress risers for the various sections.

The stresses due to the SSE (vertical and horizontal components) are combined with the blowdown stresses in order to obtain principal stresses and deflection.

All reactor internals components were found to be within acceptable stress and deflection limits for both hot leg and cold leg LOCAs occurring simultaneously with the SSE.

Both static and dynamic stress intensities are within acceptable limits. In addition, the cumulative fatigue usage factor is also within the allowable usage factor of unity.

The stresses due to the SSE (vertical and horizontal components) were combined with the blowdown stresses by the SRSS method in order to obtain the largest principal stress and deflection.

These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to ensure control rod insertion. For the guide tubes deflected above the no-loss-of-function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9(N).2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

As stated in Section 3.9(N).2.3, it is not considered necessary to conduct instrumented tests of the SNUPPS reactor vessel internals. Adequacy of these internals are verified by use of the Sequoyah and Trojan results, supported by scale model tests. References 7 and 8 describe predicted vibration behavior based on studies performed prior to the plant tests. These studies, which utilize analytical models, scale model test results, component tests, and results of previous plant tests, are used to characterize the forcing functions and establish component structural characteristics so that the flow-induced vibratory behavior and response levels for SNUPPS are estimated. These estimates are then compared to values deduced from plant test data obtained from the Sequoyah and Trojan internals vibration measurement programs.

3.9(N).3 ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES

The ASME Code Class components are constructed in accordance with the ASME Code, Section III.

A detailed discussion of ASME Code Class 1 components is provided in Section 3.9(N).1. For core support structures, design loading conditions are discussed in Section 3.9(N).5.

In general, for reactor internals components and for core support structures the criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be ensured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned

principally with the maximum allowable deflections and stability of the parts in addition to a stress criterion to ensure integrity of the components.

For the LOCA plus the SSE condition, deflections of critical internal structures are limited. In a hypothesized downward vertical displacement of the internals, energy-absorbing devices limit the displacement after contacting the vessel bottom head, ensuring that the geometry of the core remains intact.

The following mechanical functional performance criteria apply:

a. Following the design basis accident, the functional criterion to be met for the reactor internals is that the plant shall be shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This criterion implies that the deformation of critical components must be kept sufficiently small to allow core cooling.

b. For large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. The subsequent refilling of the core by the ECCS uses borated water to maintain the core in a subcritical state. Therefore, the main requirement is to ensure the effectiveness of the ECCS. Insertion of the control rods, although not needed, gives further ensurance of the ability to shut the plant down and keep it in a safe shutdown condition.

c. The inward upper barrel deflections are controlled to ensure no contacting of the nearest rod cluster control guide tube. The outward upper barrel deflections are controlled in order to maintain an adequate annulus for the coolant between the vessel inner diameter and core barrel outer diameter.

d. The rod cluster control guide tube deflections are limited to ensure operability of the control rods.

e. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited.

Method of analysis and testing for core support structures are discussed in Sections 3.9(N).2.3, 3.9(N).2.5, and 3.9(N).2.6. Stress limits and deformation criteria are given in Section 3.9(N).5.

3.9(N).3.1 <u>Loading Combinations Design Transients, and Stress Limits (For ASME Code Class 2 and 3 Components)</u>

Design pressure, temperature, and other loading conditions that provide the bases for the design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9(N).3.1.1 Design Loading Combinations

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9(N)-4. The design loading combinations are categorized with respect to normal, upset, emergency, and faulted conditions. Stress limits for each of the loading combinations are component oriented and are presented in Tables 3.9(N)-5 and 3.9(N)-6 for tanks, Table 3.9(N)-7 for inactive* pumps, Table 3.9(N)-8 for active pumps, and Table 3.9(N)-9 for valves. Active** pumps and valves are discussed in Section 3.9(N).3.2. Design of component supports is discussed in Section 3.9(N).3.4.

3.9(N).3.1.2 Design Stress Limits

The design stress limits established for the components are sufficiently low to ensure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading combinations, are component oriented and are presented in Tables 3.9(N)-5 through 3.9(N)-9.

3.9(N).3.2 Pump and Valve Operability Assurance

3.9(N).3.2.1 Pump and Valve Operability Program

Mechanical equipment classified as safety related must be capable of performing its function under postulated plant conditions. Equipment with faulted condition functional requirements includes active pumps and valves in fluid systems important to safety. Seismic analysis is presented in Section 3.7(N) and covers all safety-related mechanical equipment. A list of all active pumps supplied by Westinghouse is presented in Table 3.9(N)-10. Active valves supplied by Westinghouse or others are listed in Table 3.9(N)-11. (Although the Westinghouse nuclear steam supply system (NSSS) check valves are included in Table 3.9(N)-11, they are not considered to be active (powered) components in the Westinghouse design with respect to the Emergency Core Cooling System (ECCS) failure modes and effects analysis (FMEA) of active components or the single active failure analysis for ECCS components. The NSSS check valves are therefore not described or included as active components in Tables 6.3-5 and 6.3-6. Refer to Section 6.3.2.5.)

All active pumps are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include: 1) hydrostatic tests of pressure-retaining parts to 150 percent of the design

---

* Inactive components are those whose operability are not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.

** Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

pressure, 2) seal leakage tests at the same pressure used in the hydrostatic tests, and 3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head requirements, and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer based on the bearing material, clearances, oil type, and rotational speed. These limits are approved by Westinghouse. After the pump is installed in the plant, it undergoes the cold hydrostatic tests, hot functional tests, and the required periodic inservice inspection and operation. These tests demonstrate that the pump functions as required during all normal operating conditions of the plant.

In addition to these tests, the safety-related active pumps are qualified for operability during SSE conditions by ensuring that the pump will continue operating and not be damaged during the seismic event.

The pump manufacturer is required to show that the pump operates normally when subjected to the maximum seismic accelerations and maximum faulted nozzle loads. It is required that test or analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 2.1 g in two orthogonal horizontal directions and 2.1 g vertical acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the material elastic limit, as indicated in Table 3.9(N)-8. In addition, the pump casing stresses caused by the maximum seismic nozzle loads are limited to stresses outlined in Table 3.9(N)-8. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to ensure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9(N)-8 as allowables, ensures that critical parts of the pump would not be damaged during the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation would not be impaired by the seismic event.

Where 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 adjusted accelerations are determined, using the same conservatisms contained in the 2.1 g horizontal and 2.1 g vertical accelerations used for "rigid" structures. The static analysis is performed, using the adjusted accelerations; the stress limits stated in Table 3.9(N)-8 are still satisfied.

The second criterion necessary to ensure operability is that the pump continues to function throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., with no rotation. Typically, the rotor can be seized 5 full seconds before a circuit breaker, to prevent damage to the motor, shuts down the pump. However, the high rotary inertia in the operating pump rotor and the nature of the random, short duration loading characteristics of the seismic event prevent the rotor from losing its function. In actuality, the seismic loadings cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump would not shut down during the SSE and would operate at the design speed despite the SSE loads.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. The pump motor is qualified by meeting the requirements of IEEE Standard 344-1975 with the additional requirements and justifications outlined in Section 3.9(N).3.2.2. Any auxiliary equipment identified to be vital to the operation of the pump or pump motor that is not qualified for operation, along with the pump analysis or motor qualification, is separately qualified for operation at the accelerations it experiences at its mounting.

The operability program above gives the required ensurance that the safety-related pump and motor assemblies will not be damaged and will continue operating and performing their intended functions under SSE loadings. Program requirements take into account the complex characteristics of the pump and its motor drive.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is ensured since only normal operating loads and steady state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads are identical to the normal plant operating loads. This is ensured by requiring that the imposed nozzle loads (steady state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

Analysis was used to show that active pumps meet the operability criteria set forth herein. Testing was used in selected cases to determine natural frequencies of the equipment.

The safety-related valves are subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Code, Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and operational tests to verify that the valve will open and close. Qualification of motor operators for environmental conditions is discussed in Section 3.11(N) and Appendix 3A, Regulatory Guide 1.73. Cold hydrostatic qualification tests, hot functional qualification tests, required periodic inservice inspections, and

required periodic inservice operation are performed in-situ to verify and ensure the functional ability of the valve. These tests guarantee the reliability of the valve for the design life of the plant. The valves are constructed in accordance with the ASME Code, Section III. The maximum stress limits used for active Class 2 and 3 valves are shown in Table 3.9(N)-9. On active valves, an analysis of the extended structure is also performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure.

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 conservatively represents typical valve installations. The valve includes the operator, pilot solenoid valves, and limit switches when such are normally attached to the valve in service. The faulted condition nozzle loads are shown, by analysis, to not affect the operability of the valve. The operability of the valve during a faulted condition is demonstrated by satisfying the following criteria:

a. All the active valves are designed to have a first natural frequency which is greater than 33 Hz.

b. The actuator and yoke of the valve system is statically deflected an amount equal to the deflection caused by the faulted condition accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

c. The valve is cycled while in the deflected position. The time required to open or close the valve in the deflected position is compared to similar data taken in the undeflected condition to evaluate the significance of any change.

d. Motor operators, external limit switches, and pilot solenoid valves necessary for operation are qualified by IEEE Standard 344-1975 with the additional requirements and justifications as supplied in Section 3.9(N).3.2.2.

The accelerations that are used for the static valve qualification shall be equivalent, as justified by analysis, to 4.0 g acting in two orthogonal horizontal directions and 4.0 g vertical. The piping designer must maintain the operator accelerations to these levels, unless the valves have been qualified for higher acceleration levels.

If the natural frequency of the valve is less than 33 Hz, amplified accelerations are derived from the valve location response spectra and the valve dynamic characteristics.

The adjusted accelerations are then used in the static analysis and the valve operability testing described above.

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 qualified by the tests, and the results are used to qualify all valves within the intermediate range of sizes.

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.

The 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 nozzle loads due to maximum seismic excitation do not affect the functional ability of the valve since the valve disc is typically designed to be isolated from the body wall. The clearance supplied by the design around the disc prevents the disc from becoming bound or restricted due to any body distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is ensured using standard methods, the ability of the valve to operate is ensured by the design features.

For these reasons, the Westinghouse NSSS check valves are treated differently than other safety-related valves in the NSSS scope with respect to the above described testing program for valves with extended structures. (For these same reasons, and notwithstanding the fact that the NSSS check valves are subject to certain testing requirements described below, the NSSS check valves are not considered to be active (powered) components in Tables 6.3-5 and 6.3-6 with respect to the Emergency Core Cooling System (ECCS) failure modes and effects analysis (FMEA) or the single active failure analysis for ECCS components.)

Although considered separately with respect to the above valve operability program, the NSSS check valves are subject to the following: 1) in shop hydrostatic tests, 2) in shop seat leakage tests, and 3) periodic in-situ valve exercising and inspection to ensure the functional ability of the valves.

The pressurizer safety valves are qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves): stress and deformation analyses of critical items that may affect operability for faulted condition loads, in shop hydrostatic and seat leakage tests, and periodic in-situ valve inspection. In addition to these tests, a static load equivalent to that applied by the faulted condition is applied at the top of the bonnet, and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve ensures its overpressurization safety capabilities during a seismic event.

Using these methods, all the safety-related valves in the systems are qualified for operability during a faulted event. These methods outlined above conservatively simulate the seismic event and ensure that the active valves perform their safety-related function when necessary.

3.9(N).3.2.2 Pump Motor and Valve Operator Qualification

Active pump motors and active valve motor operators (and limit switches and solenoid valves) are seismically qualified in accordance with IEEE Standard 344-1975. Where the testing option is chosen, sine-beat testing is justified. This justification is provided by satisfying one or more of the following requirements to demonstrate that multifrequency response is negligible or the sine-beat input is of sufficient magnitude to conservatively account for this effect.

a. The equipment response is basically due to one mode.

b. The sine-beat response spectra envelopes the floor response spectra in the region of significant response.

c. The floor response spectra consists of one dominant mode and has a peak at this frequency.

If the degree of coupling in the equipment is small, then single axis testing is justified. Multiaxis testing is required if there is considerable cross coupling; however, if the degree of coupling can be determined, then single axis testing can be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

Seismic qualification by analysis alone, or by a combination of analysis and testing, is used when justified. The analysis program is justified by: 1) demonstrating that equipment being qualified is amenable to analysis, and 2) that the analysis either correlates with test results or is performed using standard analysis techniques.

3.9(N).3.3 Design and Installation Details in Mounting of Pressure Relief Devices

Refer to Section 3.9(B).3.3.

3.9(N).3.4 Component Supports (ASME Code Class 2 and 3)

Refer to Section 3.9(N).1 for a discussion of ASME Code Class 1 component supports.

Class 2 and 3 component supports are designed and analyzed for design, normal, upset, and emergency conditions to the rules and requirements of Subsection NF of Section III of the ASME Code. The design analyses or test methods and associated stress or load allowable limits 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. Supplementary requirements are outlined below.

a. For linear type supports designed by analysis for ASME Code Class 2 and 3 components, the following applies. The increased design limit for stress range identified in NF-3231.1(a) is limited to the smaller of 2 S_y or S_u, unless otherwise justified by shakedown analysis.

b. Supports for active Class 2 and 3 pumps are designed so that stresses do not exceed S_y. Additionally, the requirements presented in Section 3.9(N).3.2 that include stress analysis and evaluation of pump/motor support alignment are met. Thus the operability of active pumps is not compromised by the supports during faulted conditions.

c. Active valves are, in general, supported only by the attached piping. Exterior supports on the valve are not used.

3.9(N).4 CONTROL ROD DRIVE SYSTEM (CRDS)

3.9(N).4.1 Descriptive Information of CRDS

Control Rod Drive Mechanism

Control rod drive mechanisms (CRDMs) are located on the dome of the reactor vessel. They are coupled to rod control clusters which have absorber material over the entire length of the control rods. The CRDM is shown in Figures 3.9(N)-4 and 3.9(N)-5.

The primary function of the CRDM is to insert or withdraw rod cluster control assemblies (RCCAs) within the core to control average core temperature and to shut down the reactor.

The CRDM is a magnetically operated jack. A magnetic jack is an arrangement of three electro-magnets which are energized in a controlled sequence by a power cycler to insert or withdraw rod cluster control assemblies in the reactor core in discrete steps. Rapid insertion of the rod cluster control assemblies occurs when electrical power is interrupted.

The CRDM consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

a. The pressure vessel includes a latch housing and a rod travel housing which are connected by a threaded, seal welded, maintenance joint which facilitates replacement of the latch assembly. The CRDM housing plug is an integral part of the rod travel housing.

The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

b. The coil stack assembly includes the coil housings, electrical conduit and connector, and three operating coils: 1) the stationary gripper coil, 2) the movable gripper coil, and 3) the lift coil.

The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment. Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

c. The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches: 1) the movable gripper latches and 2) the stationary gripper latches.

The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8-inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8-inch step.

d. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

The drive rod has 5/8-inch grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and provides the means for coupling to the rod cluster control assembly.

The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the rod cluster control assembly and permits remote disconnection of the drive rod.

The CRDM is a trip design. Tripping can occur during any part of the power cycler sequencing if electrical power to the coils is interrupted.

The CRDM is butt welded to a penetration nozzle on top of the reactor vessel and is coupled to the rod cluster control assembly directly below.

The mechanism is capable of raising or lowering a 360-pound load (which includes the drive rod weight) at a rate of 45 inches/ minute. Withdrawal of the RCCA is accomplished by magnetic forces while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2,500 psia. The three operating coils are designed to operate at 392°F with forced air cooling required to maintain the coils below or at 392°F.

The CRDM shown schematically in Figure 3.9(N)-5 withdraws and inserts a RCCA as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation the stationary gripper coil of the drive mechanism holds the RCCA in a static position until a stepping sequence is initiated at which time the movable gripper coil and lift coil is energized sequentially.

Rod Cluster Control Assembly Withdrawal

The RCCA is withdrawn by repetition of the following sequence of events (refer to Figure 3.9(N)-5).

a. Movable gripper coil (B) - ON

   The latch locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 1/16-inch axial clearance exists between the latch teeth and the drive rod.

b. Stationary gripper coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

c. Lift coil (C) - ON

   The 5/8-inch gap between the movable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 inch).

d. Stationary gripper coil (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing and the stationary gripper

latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The 1/16-inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

e. Movable gripper coil (B) - OFF

The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

f. Lift coil (C) - OFF

The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

g. Repeat Step a

The sequence described above (Items a through f) is termed as one step or one cycle. The rod cluster control assembly moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute and the drive rod assembly (which has a 5/8 inch groove pitch) is raised 72 grooves per minute. The RCCA is thus withdrawn at a rate up to 45 inches per minute.

Rod Cluster Control Assembly Insertion

The sequence for RCCA insertion is similar to that for control rod withdrawal, except that the timing of lift coil (C) ON and OFF is changed to permit the lowering of the control assembly.

a. Lift coil (C) - ON

The 5/8-inch gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

b. Movable gripper coil (B) - ON

The latch locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A 1/16-inch axial clearance exists between the latch teeth and the drive rod assembly.

c. Stationary gripper coil (A) - OFF

The force of gravity, acting upon the drive rod assembly and attached RCCA, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached RCCA is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

d. Lift coil (C) - OFF

The force of gravity and spring force separates the movable gripper pole from the lift pole and the drive rod assembly and attached RCCA drop down 5/8 inch.

e. Stationary gripper (A) - ON

The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The 1/16-inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

f. Movable gripper coil (B) - OFF

The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

g. Repeat Step a

The sequence is repeated, as for RCCA withdrawal, up to 72 times per minute which gives an insertion rate of 45 inches per minute.

<u>Holding and Tripping of the Control Rods</u>

During most of the plant operating time, the CRDMs hold the RCCAs withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached RCCAs hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the RCCA plus the stationary gripper return spring are sufficient to move the latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half

against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight stationary gripper return spring and weight acting upon the latches. After the RCCA is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9(N).4.2 Applicable CRDS Design Specifications

For those components in the CRDS comprising portions of the reactor coolant pressure boundary, conformance with the General Design Criteria and 10 CFR 50, Section 50.55a is discussed in Sections 3.1 and 5.2. Conformance with Regulatory Guides pertaining are discussed in Sections 4.5 and 5.2.3.

Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

Design Stresses

The CRDS is designed to withstand stresses originating from various operating conditions as summarized in Table 3.9(N)-1.

Allowable Stresses: For normal operating conditions Section III of the ASME Code is used. All pressure boundary components are analyzed as Class 1 components.

Dynamic Analysis: The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the CRDS.

Control Rod Drive Mechanisms

The CRDM pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the CRDMs when a seismic disturbance has been postulated to confirm the ability of the pressure housing to meet ASME Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the CRDMs are:

a. 5/8-inch step

b. 147-inch travel

c. 360-pound maximum load

d. Step in or out at 45 inches/minute (72 steps/minute)

e. Electrical power interruption shall initiate release of drive rod assembly

f. Trip delay time of less than 150 milliseconds - Free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F.

g. 45-year design life with normal refurbishment

3.9(N).4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9(N).4.3.1 Pressure Vessel

The pressure retaining components are analyzed for loads corresponding to normal, upset, emergency, and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

a. Control rod trip (equivalent static load)

b. Differential pressure

c. Spring preloads

d. Coolant flow forces (static)

e. Temperature gradients

f. Differences in thermal expansion

1. Due to temperature differences

2. Due to expansion of different materials

g. Interference between components

h. Vibration (mechanically or hydraulically induced)

i. All operational transients listed in Table 3.9(N)-1

j. Pump overspeed

k. Seismic loads (OBE and SSE)

l. Blowdown forces (due to cold and hot leg break)

The main objective of the analysis is to satisfy allowable stress limits, to ensure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to ensure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied, using experimental test data and experience from operating reactors.

3.9(N).4.3.2 Drive Rod Assembly

All postulated failures of the drive rod assemblies either by fracture or uncoupling lead to a reduction in reactivity. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with and is guided by the RCCA. This always results in reactivity decrease.

3.9(N).4.3.3 Latch Assembly and Coil Stack Assembly

<u>Results of Dimensional and Tolerance Analysis</u>

With respect to the CRDM system as a whole, critical clearances are present in the following areas:

a. Latch assembly - thermal clearances

b. Latch arm - drive rod clearances

c. Coil stack assembly - thermal clearances

d. Coil fit in coil housing

The following discussion defines clearances that are designed to provide reliable operation in the CRDM in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of Type 410 stainless steel fit over parts made from Type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearances of these parts at 68°F is 0.011 inch. At the maximum design temperature of 650°F, minimum clearance is 0.0045 inch and at the maximum expected operating temperatures of 550°F is 0.0057 inch.

Latch Arm - Drive Rod Clearances

The CRDM incorporates a load transfer action. The movable or stationary gripper latch are not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9(N)-6 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9(N)-7 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearances of the coil stack assembly over the latch housing was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F, the inside diameter of the coil stack is 7.308/7.298 inches. The outside diameter of the latch housing is 7.260/7.270 inches.

Thermal expansion of the mechanism due to operating temperature of the CRDM results in minimum inside diameter of the coil stack being 7.310 inches at 222°F and the maximum latch housing diameter being 7.302 inches at 532°F.

Under the extreme tolerance conditions listed above, it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

Four coil stack assemblies were removed from four hot CRDM mounted on 11.035-inch centers on a 550°F test loop, allowed to cool, and then placed without incident as a test to prove the preceding.

Coil Fit in Core Housing

CRDM and coil housing clearances are selected so that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot CRDM.

3.9(N).4.4 CRDS Performance Assurance Program

The ability of the pressure housing components to perform throughout the design lifetime as defined in the equipment specification is confirmed by the stress analysis report required by the ASME Code, Section III.

Internal components subjected to wear will withstand a minimum of 3,000,000 steps without refurbishment as confirmed by life tests (Ref. 12). Latch assembly inspection is recommended after 2.5×10^6 steps have been accumulated on a single CRDM.

To confirm the mechanical adequacy of the fuel assembly, the CRDM, and RCCA, functional test programs have been conducted on a full-scale 12-foot control rod. The 12-foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1,000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test, the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow-excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive-line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the CRDMs meet the design requirement of 2.7 seconds from start of RCCA motion to dashpot entry. This trip time requirement will be confirmed for each CRDM prior to initial reactor operation and at periodic intervals after initial reactor operation, as required by the Technical Specifications.

There are no significant differences between the prototype CRDMs and the production units. Design materials, tolerances, and fabrication techniques are the same.

These tests have been reported in Reference 12.

It is expected that all control rod drive mechanisms will meet specified operating requirements for the duration of plant life with normal refurbishment. However, a Technical Specification pertaining to an inoperable rod cluster control assembly has been set. Latch assembly inspection is recommended after 2.5×10^6 steps have been accumulated on a single control rod drive mechanism.

If a rod cluster control assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip.

Thus, inability to move one RCCA can be tolerated. More than one inoperable RCCA could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable RCCAs has been limited to one, as discussed in the Technical Specifications.

In order to demonstrate proper operation of the control rod drive mechanism and to ensure acceptable core power distributions during RCCA partial-movement, checks are performed on the RCCA (refer to the Technical Specifications). In addition, periodic drop tests of the rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical rod cluster control assembly ejection. During these tests, the acceptable drop time of each assembly is not greater than 2.7 seconds, at full flow and operating temperature, from the beginning of motion to dashpot entry.

Actual experience in operating Westinghouse plants indicates excellent performance of control rod drive mechanisms.

All units are production tested prior to shipment to confirm ability of the control rod drive mechanism to meet design specification-operation requirements.

Each production control rod drive mechanism undergoes a production test as listed below:

Test	Acceptance Criteria
Cold (ambient) hydrostatic	ASME Code, Section III
Confirm step length and load transfer (stationary gripper to movable gripper or movable gripper to stationary gripper)	Step Length 5/8 + 0.015 inch axial movement Load Transfer 0.047 inch nominal axial movement
Cold (ambient) performance test at design load - 5 full travel excursions	Operating Speed 45 inches/minute Trip Delay Free fall of drive rod to begin within 150 milliseconds

3.9(N).5 REACTOR PRESSURE VESSEL INTERNALS

3.9(N).5.1 Design Arrangements

The SNUPPS reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure, and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDMs, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provides guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.9(N)-8. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel head flange, and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are machined.

Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to

provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel and are approximately 48 inches wide by 148 inches long by 2.8 inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guide by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in Reference 13.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an inconel clevis block is welded to the vessel inner diameter. Another inconel insert block is bolted to each of these blocks and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within ASME Code, Section III, limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy-absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy-absorbing devices of the internals to the vessel.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

Upper Core Support Assembly

The SNUPPS upper core support assembly, shown in Figures 3.9(N)-9 and 3.9(N)-10, consists of the top support plate assembly and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the top support plate assembly and the upper core plate and are fastened at top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies sheath and guide the control rod drive shafts and control rods. They are fastened to the top support plate and are restrained by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support plate and guide tube.

The upper core support assembly is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which, in turn, engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90 degrees from each other. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies, and control rods are thereby ensured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns to the top support plate assembly and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7.7-9 shows the basic flux-mapping system).

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper

support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches, and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. These are the only conditions which affect the incore instrumentation support structure.

3.9(N).5.2 Design Loading Conditions

Normal and Upset Conditions

The normal and upset loading conditions that provide the basis for the design of the reactor internals are:

a. Fuel and reactor internals weight

b. Fuel and core component spring forces, including spring preloading forces

c. Differential pressure and coolant flow forces

d. Temperature gradients

e. Vibratory loads including OBE seismic loads

f. Normal and upset operational thermal transients listed in Table 3.9(N)-1

g. Control rod trip (equivalent static load)

h. Loads due to loop(s) out of service

i. Loss of load/pump overspeed

Emergency Conditions

The emergency loading conditions that provide the basis for the design of the reactor internals are:

a. Small LOCA

b. Small steam break

c. Complete loss of flow

Faulted Conditions

The faulted loading conditions that provide the basis for the design of the reactor internals are:

a. Large LOCA

b. SSE

3.9(N).5.3 Design Loading Categories

The combination of design loadings fit into either the normal, upset, emergency, or faulted conditions as defined in the ASME Code, Section III, as indicated by Figures NG-3221-1, NG-3224-1, and by Appendix F, "Rules for Evaluating Faulted Conditions."

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions, including thermal shock of the ECCS following a LOCA, as summarized in Table 3.9(N)-1.

The scope of the stress analysis problem is very large, requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For LOCA plus the SSE condition, the deflection criteria of critical internal structures are the limiting values given in Table 3.9(N)-12. The corresponding no-loss-of-function limits are included in Table 3.9(N)-12 for comparison purposes with the allowed criteria.

The criteria for the core drop accident is based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately 1/2 inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1-1/4 inches, which is insufficient to permit the tips of the RCCA to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident involving the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This structure limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated, assuming a complete and instantaneous failure of the primary core support, and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15 percent, after which a positive stop is provided to ensure support.

3.9(N).5.4 Design Bases

The design bases for the mechanical design of the SNUPPS reactor vessel internals components are as follows:

a. The reactor internals in conjunction with the fuel assemblies directs reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head is provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

b. In addition to neutron shielding provided by the reactor coolant, a separate neutron pad assembly is provided to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.

c. Provisions are made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

d. The core internals are designed to withstand mechanical loads arising from the OBE, SSE, and pipe ruptures and meet the requirements of Item e below.

e. The reactor has mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

f. Following the design basis accident, the plant is capable of being shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.9(N)-12. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in Table 3.9(N)-12.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9(N).2.

The basis for the design stress and deflection criteria is identified below:

Allowable Stresses

For normal operating conditions, Section III of the ASME Nuclear Power Plant Components Code is used as a basis for evaluating the acceptability of calculated stresses. Both static and alternating stress intensities are considered.

It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the DBA used for the SNUPPS reactor internals are based on the 1974 Edition of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

Internal structures are analyzed to meet the intent of the ASME Code in accordance with Subsection NG, paragraph NG-3311(c). Stresses in the core support structure induced

by interactions with internal structures are analyzed and shown to be in conformance with core support code limits. Design and construction for core support structures meet Subsection NG in full.

3.9(N).6 INSERVICE TESTING OF PUMPS AND VALVES

Refer to Section 3.9(B).6.

3.9(N).7 REFERENCES

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

2. "Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program," WCAP-8929, April, 1977.

3. "Sample Analysis of a Class 1 Nuclear Piping System," prepared by ASME Working Group on Piping, ASME Publication, 1972.

4. Witt, F. J., Bamford, W. H. and Esselman, T. C., "Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," WCAP-9283, March, 1978.

5. Bogard, W. T. and Esselman, T. C., "Combination of Safe Shutdown Earthquake and Loss-of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants," WCAP-9279, March, 1978.

6. Bloyd, C. N., Ciaramitaro, W. and Singleton, N. R., "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8766 (Proprietary) and WCAP-8780, (Non-Proprietary), May, 1976.

7. Lee, H., "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," WCAP-8303-P-A (Proprietary) and WCAP-8317-A (Non-Proprietary), July, 1975.

8. Bloyd, C. N. and Singleton, N. R., "UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations, "WCAP-8516-P (Proprietary) and WCAP-8517 (Non-Proprietary), March, 1975.

9. Takeuchi, K., et al., "MULTIFLEX - A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708-P-A, Volumes I and II (Proprietary) and WCAP-8709-A, Volumes I and II (Non-Proprietary), September, 1977.

10. Electro-Mechanical Design Analysis Code - Finite Element Analysis (EMDAL-FEA) User Manual Version 11.4 October 2010 Revision C, Curtiss-Wright Electro-Mechanical Corporation (Proprietary Class 2).

11. Beraquista, E. M. and Neubert, K. B., LTR-RIDA-12-52, Rev. 0 “Benchmarking of EMDAL-FEA to WECAN/PLUS 99,” March 7, 2012.

12. Cooper, F. W. Jr., "17 x 17 Driveline Components- Tests Phase IB, II, III D-Loop Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), December, 1974.

13. Kraus, S., "Neutron Shielding Pads," WCAP-7870, June, 1972.

14. Altman, D. A., et al., "Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Tests on the Sequoyah 1 Power Plant," WCAP-9944 (Proprietary) and WCAP-9945 (Non-Proprietary), July 1981.

15. “Program Verification for BWSPAN PC Version,” AREVA Calculation 32-5007216-016, Revision 16, February 2013.

TABLE 3.9(N)-1 SUMMARY OF REACTOR COOLANT SYSTEM ANALYZED DESIGN TRANSIENTS

Normal Conditions	Occurrences
1. Heatup and cooldown at 100°F/hr (pressurizer cooldown 200/hr)	200 (each)
2. Unit loading and unloading at 5 percent of full power per minute	13,200 (each)
3. Step load increase and decrease of 10 percent of full power	2,000 (each)
4. Large step load decrease with steam dump	200
5. Steady state fluctuations	
a. Initial fluctuations	1.5×10^5
b. Random fluctuations	3.0×10^6
6. Feedwater cycling at hot shutdown	2,000
7. Loop out of service	
a. Normal loop shutdown	80
b. Normal loop startup	70
8. Unit loading and unloading between 0 and 15 percent of full power	500 (each)
9. Boron concentration equalization	26,400
10. Reactor coolant pump startup and shutdown	
a. Cold condition	
(1) RCS venting	800
(2) RCS heatup, cooldown	200
b. Pump restart condition	
(1) Hot functionals, reactor coolant pump stops, starts	500
c. Hot condition	
(1) Transients and miscellaneous	2,500

TABLE 3.9(N)-1 (Sheet 2)

11.	Reduced temperature return to power	2,000
12.	Refueling	80
13.	Turbine roll test	20
14.	Primary side leakage test	200
15.	Secondary side leakage test	80
16.	Feedwater heaters out of service	
	a. One heater out of service	120
	b. One bank of heaters out of service	120
Upset Conditions		Occurrences
1.	Loss of load (without immediate reactor trip)	80
2.	Loss of offsite power (with natural circulation in the RCS)	40
3.	Partial loss of flow (loss of one pump)	80
4.	Reactor trip from full power	
	a. Without cooldown	230
	b. With cooldown, without safety injection	160
	c. With cooldown and safety injection	10
	d. With no inadvertent cooldown - emergency overspeed	20
5.	Inadvertent RCS depressurization	20
6.	Inadvertent startup of an inactive loop	10
7.	Control rod drop	80
8.	Inadvertent safety injection actuation	60
9.	Operating Basis Earthquake (20 earthquakes of 10 cycles each)	200
10.	Excessive Feedwater Flow	30
11.	RCS Cold Overpressurization	10

TABLE 3.9(N)-1 (Sheet 3)

Emergency Conditions*		Occurrences
1.	Small loss-of-coolant accident	5
2.	Small steam break	5
3.	Complete loss of flow	5

Faulted Conditions*		Occurrences
1.	Reactor coolant pipe break (large loss-of-coolant accident)	1
2.	Large steam line break	1
3.	Feedwater line break	1
4.	Reactor coolant pump locked rotor	1
5.	Control rod ejection	1
6.	Steam generator tube rupture	(included under upset conditions, reactor trip from full power with safety injection)
7.	Safe Shutdown Earthquake	1

Test Conditions		Occurrences
1.	Primary side hydrostatic test	10
2.	Secondary side hydrostatic test	10
3.	Tube leakage test	800

---

* In accordance with the ASME Nuclear Power Plant Components Code, emergency and faulted conditions are not included in fatigue evaluation.

TABLE 3.9(N)-1A MONITORED COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at < 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	50 leak tests.	Pressurized to ≥ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to ≥ 3106 psig.
Secondary Coolant System	1 large steam line break.	Break in a > 6-inch steam line
	5 hydrostatic pressure tests.	Pressurized to ≥ 1350 psig.

TABLE 3.9(N)-2 LOADING COMBINATIONS FOR ASME CLASS 1 COMPONENTS AND SUPPORTS (EXCLUDING PIPE SUPPORTS)

Condition Classification	Loading Combination
Design	Design pressure, design temperature, deadweight, Operating Basis Earthquake
Normal	Normal condition transients, deadweight
Upset	Upset condition transients, deadweight, Operating Basis Earthquake
Emergency	Emergency condition transients, deadweight
Faulted	Faulted condition transients, deadweight, Safe Shutdown Earthquake or Safe Shutdown Earthquake and Pipe Rupture Loads

TABLE 3.9(N)-3 ALLOWABLE STRESSES FOR ASME CODE, SECTION III, CLASS 1 COMPONENTS (A)(C)

Operating Condition Classification	Vessels/Tanks	Piping	Pumps	Valves	Component Supports (d)
Normal	NB-3222 (Level A)	NB-3653 (Level A)	NB-3222 (Level A)	NB-3525 (Level A)	NF-3222 NF-3231.1(a) (Level A)
Upset	NB-3223 (Level B)	NB-3654 (Level B)	NB-3223 (Level B)	NB-3525 (Level B)	NF-3223 NF-3231.1(a) (Level B)
Emergency	NB-3224 (Level C)	NB-3655 (Level C)	NB-3224 (Level C)	NB-3526 (Level C)	NF-3224 NF-3231.1(b) (Level C)
Faulted	NB-3225 (Level D)	NB-3656 (Level D)	NB-3225 (Level D)	(b)	NF-3225 NF-3231.1(c) (Level D)

(a) A test of the components may be performed in lieu of analysis.

(b) CLASS 1 VALVE FAULTED CONDITION CRITERIA

Active

a) Calculate P_m from para. NB3545.1 with Internal Pressure P_s = 1.25P_s
$P_m \leq 1.5S_m$

b) Calculate S_n from para. NB3545.2 with
C_p = 1.5
P_s = 1.25P_s
Q_{t^2} = 0
P_{ed} = 1.3X value of P_{ed} from equations of 3545.2(b) (1)
$S_n \leq 3S_m$

Inactive

a) Calculate P_m from para. NB3545.1 with Internal Pressure P_s = 1.50 P_s
$P_m \leq 2.4S_m$ or 0.7 s_u

b) Calculate S_n from para. NB3545.2 with
C_p = 1.5
P_s = 1.50 P_s
Q_{t^2} = 0
P_{ed} = 1.3X value of P_{ed} from equations of 3545.2(b) (1)
$S_n \leq 3S_m$

P_s, P_e, P_m, P_b, Q_t, C_p, S_n & S_m, as defined by Section III of the ASME Code

(c) Limits identified refer to subsections of the ASME Code, Section III.

(d) Also see Appendix 3A, Regulatory Guides 1.124 and 1.130.

TABLE 3.9(N)-4 DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3 COMPONENTS AND SUPPORTS (A) (EXCLUDING PIPE SUPPORTS)

Loading Combination(b,c)		Design/Service Level Requirements
1.	Design pressure, design temperature, deadweight	Design
2.	Normal condition pressure, normal condition metal temperature, deadweight, nozzle loads	Service Level A
3.	Upset condition pressure, upset condition metal temperature, deadweight, nozzle loads, Operating Basis Earthquake	Service Level B
4.	Emergency condition pressure, emergency condition metal temperature, deadweight, nozzle loads	Service Level C
5.	Faulted condition pressure, faulted condition metal temperature, deadweight, nozzle loads, Safe Shutdown Earthquake	Service Level D

NOTES:

(a) The responses for each loading combination are combined using the absolute sum method. On a case-by-case basis, algebraic summation may be used when signs are known for final design evaluations.

(b) Temperature is used to determine allowable stress only.

(c) Nozzle loads, pressures, and temperatures are those associated with the respective plant operating conditions (i.e., normal, upset, emergency, and faulted), as noted for the component under consideration.

TABLE 3.9(N)-5 STRESS CRITERIA FOR SAFETY-RELATED ASME CLASS 2* AND CLASS 3 VESSELS

Design/Service Level	Stress Limits**
Design and Service Level A	$\sigma_m \leq 1.0S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5S$
Service Level B	$\sigma_m \leq 1.1S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65S$
Service Level C	$\sigma_m \leq 1.5S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.80S$
Service Level D	$\sigma_m \leq 2.0S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4S$

* Applies for vessels designed in accordance with the ASME Code, Section III, NC-3300.

** Stress limits are taken from ASME III, Subsections NC and ND, or, for vessels procured prior to the incorporation of these limits into ASME III, from Code Case 1607.

TABLE 3.9(N)-6 STRESS CRITERIA FOR SAFETY-RELATED CLASS 2 VESSELS*

Design/Service Level	Stress Limits**
Design and Service Level A	$P_m \leq 1.0S_m$ $P_L \leq 1.5S_m$ $(P_m \text{ or } P_L) + P_b \leq 1.5S_m$
Service Level B	$P_m \leq 1.1S_m$ $P_L \leq 1.65S_m$ $(P_m \text{ or } P_L) + P_b \leq 1.65S_m$
Service Level C	$P_m \leq 1.25S_m$ $P_L \leq 1.8S_m$ $(P_m \text{ or } P_L) + P_b \leq 1.8S_m$
Service Level D	$P_m \leq 2.0S_m$ $P_L \leq 3.0S_m$ $(P_m \text{ or } P_L) + P_b \leq 3.0S_m$

* Applies for vessels designed in accordance with the ASME Code, Section III, NC-3200

** Stress limits are from ASME III, Subsection NC.

TABLE 3.9(N)-7 STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS AND PUMP SUPPORTS

Design/Service Level	Stress Limits*
Design and Service Level A	$\sigma_m \leq 1.0S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5S$
Service Level B	$\sigma_m \leq 1.1S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65S$
Service Level C	$\sigma_m \leq 1.5S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.80S$
Service Level D	$\sigma_m \leq 2.0S$ $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4S$

* Stress limits are taken from ASME III, Subsections NC and ND, or, for pumps procured prior to the incorporation of these limits into ASME III, from Code Case 1636.

TABLE 3.9(N)-8 DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS

Design/Service Level	Design Criteria*
Design, Service Level A and Service Level B	$\sigma_m \leq 1.0S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5S$
Service Level C	$\sigma_m \leq 1.1S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$
Service Level D	$\sigma_m \leq 1.2S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.8S$

* The stress limits specified for active pumps are more restrictive than the ASME III limits to provide assurance that operability will not be impaired for any operating condition.

TABLE 3.9(N)-9 STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VALVES

Design/Service Level	Stress Limits[a, b, c, d, and f]	P_{max}[e]
Design and Service Level A	Valve bodies shall conform to ASME Code, Section III	1.0
Service Level B	$\sigma_m \leq 1.1S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65S$	1.1
Service Level C	$\sigma_m \leq 1.5S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.8S$	1.2
Service Level D	$\sigma_m \leq 2.0S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4S$	1.5

NOTES:

(a) Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied: 1) the section modulus and area of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110 percent of those for the piping connected (or joined) to the valve body inlet and outlet nozzles; and 2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, or connected piping material. If the valve body material allowable stress is less than that of the connected piping, the required acceptance criteria ratio shall be 110 percent multiplied by the ratio of the pipe allowable stress to the valve allowable stress. If unable to comply with this requirement, an analysis in accordance with the design procedure for Class 1 valves is an acceptable alternate method.

(b) Casting quality factor of 1.0 shall be used.

(c) These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

(d) Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

(e) The maximum pressure resulting from upset, emergency, or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure. If these pressure limits are met, the stress limits in this table are considered to be satisfied.

(f) Stress limits are taken from ASME III, Subsections NC and ND, or, for valves procured prior to the incorporation of these limits into ASME III, from Code Case 1635.

TABLE 3.9(N)-10 ACTIVE PUMPS

Pump	Item Number	System	ANS Safety Class	Normal Mode	Post-LOCA Mode	Basis
ECCS centrifugal charging pumps 1 and 2	APCH	CVCS	2	On/Off	On	ECCS safeguards operation and safety grade cold shutdown
Boric acid transfer pumps 1 and 2	APBA	CVCS	2	On/Off	Off	Boration and cold shutdown if RWST is rendered unavailable
Residual heat removal pumps 1 and 2	APRH	RHRS	2	Off	On	ECCS safeguards operation and safety grade cold shutdown
Safety injection pumps 1 and 2	APSI	SIS	2	Off	On	ECCS safeguards operation

TABLE 3.9(N)-11 ACTIVE VALVES

VALVE LOCATION NUMBER	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/ANS SAFETY CLASS	NORMAL POSITION	BASIS
BB-HV-8000A/B	Reactor Coolant	Motor	3	Gate/1	Open	2, 6
BB-HV-8001A/B	Reactor Coolant	Solenoid	1	Globe/2	Closed	2
BB-HV-8002A/B	Reactor Coolant	Solenoid	1	Globe/2	Closed	2
BB-V-8010A/B/C	Reactor Coolant	Self-actuated	6	Relief/1	Closed	5
BB-HV-8026	Reactor Coolant	Air	1	Diaphragm/2	Closed	1
BB-HV-8027	Reactor Coolant	Air	1	Diaphragm/2	Closed	1
BL-V-8046	Reactor Makeup	ΔP	3	Check/2	N/A	1, 7
BL-HV-8047	Reactor Makeup	Air	3	Diaphragm/2	Open	1
BB-PCV-455A	Reactor Coolant	Solenoid	3	Globe/1	Closed	2, 6
BB-PCV-456A	Reactor Coolant	Solenoid	3	Globe/1	Closed	2, 6
BG-HV-8100	Chemical Volume Control	Motor	2	Globe/2	Open	1
BG-HV-8104	Chemical Volume Control	Motor	2	Globe/2	Closed	4
BG-HV-8105	Chemical Volume Control	Motor	3	Gate/2	Open	1, 2, 3
BG-HV-8106	Chemical Volume Control	Motor	3	Gate/2	Open	2, 3
BG-HV-8110	Chemical Volume Control	Motor	2	Globe/2	Open	2, 3
BG-HV-8111	Chemical Volume Control	Motor	2	Globe/2	Open	2, 3
BG-HV-8112	Chemical Volume Control	Motor	2	Globe/2	Open	1
BG-HV-8152	Chemical Volume Control	Air	3	Globe/2	Open	1
BG-HV-8153A/B	Chemical Volume Control	Solenoid	1	Globe/1	Closed	2
BG-HV-8154A/B	Chemical Volume Control	Solenoid	1	Globe/1	Closed	2
BB-HV-8157A/B	Reactor Coolant	Solenoid	1	Globe/2	Closed	2
BG-HV-8160	Chemical Volume Control	Air	3	Globe/2	Open	1
BG-HV-8357A/B	Chemical Volume Control	Motor	1	Globe/2	Closed	2
BB-V-8378A/B	Reactor Coolant	ΔP	3	Check/1	N/A	6, 7
BG-V-0645	Chemical Volume Control	ΔP	3	Check/2	N/A	3, 7
BG-V-8381	Chemical Volume Control	ΔP	3	Check/2	N/A	1, 7
BG-V-8481A/B	Chemical Volume Control	ΔP	4	Check/2	N/A	2, 3, 7

TABLE 3.9(N)-11 (Sheet 2)

VALVE LOCATION NUMBER	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/ANS SAFETY CLASS	NORMAL POSITION	BASIS
BG-V-8497	Chemical Volume Control	ΔP	3	Check/2	N/A	3, 7
BG-V-8546A/B	Chemical Volume Control	ΔP	8	Check/2	N/A	2, 3, 7
BG-LCV-112B/C	Chemical Volume Control	Motor	4	Gate/2	Open	2, 3
BN-LCV-112D/E	Borated Refueling Water Storage	Motor	8	Gate/2	Closed	2, 3
EJ-HV-8701A/B	Residual Heat Removal	Motor	12	Gate/1	Closed	2, 6
BB-PV-8702A/B	Reactor Coolant	Motor	12	Gate/1	Closed	2, 6
EJ-V-8708A/B	Residual Heat Removal	Self-Actuated	3	Relief/2	Closed	2, 5
EJ-HV-8716A/B	Residual Heat Removal	Motor	10	Gate/2	Open	2, 3
EJ-V-8730A/B	Residual Heat Removal	ΔP	10	Check/2	N/A	2, 3, 7
EJ-FCV-610	Residual Heat Removal	Motor	3	Gate/2	Open	2, 3
EJ-FCV-611	Residual Heat Removal	Motor	3	Gate/2	Open	2, 3
BN-HCV-8800A/B	Borated Refueling Water Storage	Air	3	Globe/2	Closed	3
EM-HV-8801A/B	High Pressure Coolant Injection	Motor	4	Gate/2	Closed	2, 3
EM-HV-8802A/B	High Pressure Coolant Injection	Motor	4	Gate/2	Closed	3
EM-HV-8803A/B	High Pressure Coolant Injection	Motor	4	Gate/2	Closed	2, 3
EJ-HV-8804A/B	Residual Heat Removal	Motor	8	Gate/2	Closed	3
BN-HV-8806A/B	Borated Refueling Water Storage	Motor	8	Gate/2	Open	3
EM-HV-8807A/B	High Pressure Coolant Injection	Motor	6	Gate/2	Closed	3
EP-HV-8808A/D	Accumulator Safety Injection	Motor	10	Gate/2	Open	2, 3
EJ-HV-8809A/B	Residual Heat Removal	Motor	10	Gate/2	Open	2, 3
EJ-HV-8811A/B	Residual Heat Removal	Motor	14	Gate/2	Closed	3
BN-HV-8812A/B	Borated Refueling Water Storage	Motor	14	Gate/2	Open	2, 3
BN-HV-8813	Borated Refueling Water Storage	Motor	2	Globe/2	Open	3
EM-HV-8814A/B	High Pressure Coolant Injection	Motor	1-1/2	Globe/2	Open	3
EM-V-8815	High Pressure Coolant Injection	ΔP	3	Check/1	N/A	2, 3, 6, 7
EP-V-8818A/B/C/D	Accumulator Safety Injection	ΔP	6	Check/1	N/A	3, 6, 7
EM-HV-8821A/B	High Pressure Coolant Injection	Motor	4	Gate/2	Open	3

TABLE 3.9(N)-11 (Sheet 3)

VALVE LOCATION NUMBER	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/ANS SAFETY CLASS	NORMAL POSITION	BASIS
EM-HV-8823	High Pressure Coolant Injection	Air	3/4	Globe/2	Closed	1
EM-HV-8824	High Pressure Coolant Injection	Air	3/4	Globe/2	Closed	1
EJ-HCV-8825	Residual Heat Removal	Air	3/4	Globe/2	Closed	1
EM-HV-8835	High Pressure Coolant Injection	Motor	4	Gate/2	Open	3
EJ-HV-8840	Residual Heat Removal	Motor	10	Gate/2	Closed	3
EJ-V-8841A/B	Residual Heat Removal	ΔP	6	Check/1	N/A	3, 6, 7
EM-HV-8843	High Pressure Coolant Injection	Air	3/4	Globe/2	Closed	1
EM-HV-8871	High Pressure Coolant Injection	Air	3/4	Globe/2	Closed	1
EP-HV-8880	Accumulator Safety Injection	Air	1	Globe/2	Closed	1
EM-HV-8881	High Pressure Coolant Injection	Air	3/4	Globe/2	Closed	1
EM-HV-8888	High Pressure Coolant Injection	Air	1	Globe/2	Closed	1
EJ-HCV-8890A/B	Residual Heat Removal	Air	3/4	Globe/2	Closed	1
EM-V-8922A/B	High Pressure Coolant Injection	ΔP	4	Check/2	N/A	3, 7
EM-HV-8923A/B	High Pressure Coolant Injection	Motor	6	Gate/2	Open	3
EM-V-8926A/B	High Pressure Coolant Injection	ΔP	8	Check/2	N/A	3, 7
BB-V-8948A/B/C/D	Reactor Coolant	ΔP	10	Check/1	N/A	3, 6, 7
BG-V-8440	Chemical Volume Control	ΔP	4	Check/2	N/A	2, 3, 7
BB-V-8949/B/C/D/E	Reactor Coolant	ΔP	6	Check/1	N/A	3, 6, 7
EP-HV-8950A/B/C/D/E/F	Accumulator Safety Injection	Solenoid	1	Globe/2	Closed	2
EP-V-8956A/B/C/D	Accumulator Safety Injection	ΔP	10	Check/1	N/A	3, 6, 7
EJ-V-8958A/B	Residual Heat Removal	ΔP	14	Check/2	N/A	3,7
EM-HV-8964	High Pressure Pressure Coolant Injection	Air	3/4	Globe/2	Closed	1
EJ-V-8969A/B	Residual Heat Removal	ΔP	8	Check/2	N/A	3,7
HB-HV-7126	Liquid Radwaste	Air	3/4	Diaphragm/2	Open	1
HB-HV-7136	Liquid Radwaste	Air	3	Diaphragm/2	Open	1
HB-HV-7150	Liquid Radwaste	Air	3/4	Diaphragm/2	Open	1
HB-HV-7176	Liquid Radwaste	Air	3	Diaphragm/2	Open	1

TABLE 3.9(N)-11 (Sheet 4)

VALVE LOCATION NUMBER	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/ANS SAFETY CLASS	NORMAL POSITION	BASIS
BB-HV-8037A/B	Reactor Coolant	Motor	4.0	Gate/3	Closed	2
BB-8379A/B	Reactor Coolant	ΔP	3.0	Check/1	N/A	6,7
BGLCV0459	Chemical Volume Control	Air	3	Globe/1	Open	6
BGLCV0460	Chemical Volume Control	Air	3	Globe/1	Open	6

BASIS
1. Containment isolation
2. Safety grade cold shutdown operation
3. ECCS safeguards operation
4. Active component in the path from the boric acid tanks. The boric acid transfer pumps are Class 1E pumps powered from Class 1E sources; however, the pump controls are non-Class 1E.
5. Pressure/relief
6. RCPB isolation
7. The definition of an active component for the purpose of supporting the pump and valve operability includes NSSS check valves. These check valves, although not powered components, meet the definition of having mechanical motion and are therefore included in Table 3.9(N)-11. However, NSSS check valves are not considered to be active (powered) components in the Westinghouse design with respect to the Emergency Core Cooling System (ECCS) Failure Modes and Effects Analysis (FMEA) of active components or the single active failure analysis for ECCS components. Refer to Section 6.3.2.5.

TABLE 3.9(N)-12 MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (in.)	No-Loss-of Function Deflections (in.)
Upper Barrel		
Radial inward	4.1	8.2
Radial outward	1.0	1.0
Upper Package	0.10	0.15
Rod Cluster Guide Tubes	1.00	1.75

3.10(B) SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Refer to Table 3.10(B)-1 for a listing of non-NSSS seismic Category I instrumentation and electrical equipment requiring seismic qualifications.

3.10(B).1 SEISMIC QUALIFICATION CRITERIA

The seismic Category I instrumentation and electrical equipment are qualified to withstand the effects of the safe shutdown earthquake (SSE) and remain functional during normal and accident conditions.

The seismic Category I instrumentation and electrical equipment is divided into three further classifications--equipment which is designed to maintain its functional capability during and after an SSE, equipment which is designed to maintain its functional capability after, but not during an SSE, and equipment which, although not required to function actively, 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 of the seismic Category I electrical items and their respective supports are structural as well as functional. Where applicable, the structural requirements are in accordance with AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," adopted February 12, 1969, AISI, "Specification for the Design of Cold-Formed Steel Structural Members", 1968 Edition, or similar codes applicable for other construction materials (refer to Section 3.8.4). Field welding of supports is in accordance with AWS D1.1, "Structural Welding Code," except for Section I, Part C, Paragraphs 2.7.1.5 and 2.7.1.1, where adequate weld is provided for the structural requirements, and as noted in FSAR Section 3.8.3.6.3.3 (see FSAR Table 3.2-1, Note 19).

The structural requirements for instrumentation equipment and systems which are required to maintain pressure boundary integrity are in accordance with ASME Section III, 1977.

In addition to the above, the standby power system and seismic Category I instrumentation and electrical equipment associated with engineered safety features are qualified to withstand seismic disturbances of the intensity of the SSE during post-accident operation.

The engineered safety features actuation system is designed with the capability to initiate protective actions during the SSE.

3.10(B).2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Qualification and documentation procedures used for seismic Category I equipment and/or systems within Bechtel's scope of supply meet the provisions of IEEE 344, 1975 and Regulatory Guide 1.100.

3.10(B).2.1 Analysis

Analysis without testing is acceptable when it can be demonstrated that the analytical technique ensures the design-intended function. The procedures described in IEEE 344, Paragraph 5.0, are followed.

3.10(B).2.2 Testing

Seismic tests are performed by subjecting equipment to vibratory motion which simulates the required response spectrum (RRS) at the equipment mounting. A combination or one of the following techniques, as defined in IEEE 344, is applicable.

1. Fragility testing

2. Proof testing

3. Device testing

4. Assembly testing

3.10(B).2.3 Combined Analysis and Testing

Equipment that cannot be qualified from a practical standpoint by analysis or testing because of its size and/or complexity is qualified by combined analysis and testing. The procedures described in IEEE 344, Paragraph 7.0, are followed.

3.10(B).2.4 Generic Qualification

In addition to the foregoing methods for qualification of Class 1E equipment, generic programs are used which qualify electrical or instrument equipment by testing and/or analyzing representative types which are similar with respect to type, load level, and size.

3.10(B).3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF ELECTRICAL EQUIPMENT

3.10(B).3.1 Analysis

The analysis method and much of the design criteria pertaining to electrical raceway systems was verified by the "Cable Tray and Conduit Raceway Test Program," as noted in Section 1.5.3.2. During that program, some 2,000 dynamic tests were performed on several hundred varied cable tray and conduit support systems. The effects of numerous parameters which could possibly influence system dynamics were investigated. Also, several different types of tray, conduit, and supports from various manufacturers were tested. As a result of this extensive test program and related activities, a conservative design basis for Class 1E cable tray and conduit systems has been developed.

The following bases were used in the seismic design and analysis of Class 1E electrical raceways:

a. All electrical raceway supports are designed by dynamic analysis, using the response spectrum method or the equivalent static load method described in Section 3.7(B).2.7.1.

b. Analysis and seismic restraint measures for raceways are based on combined limiting values for static load, span length, and computed seismic response.

c. Maximum allowable stresses during the SSE for all components are limited to 90 percent of maximum yield.

d. An analysis was performed for the angle fittings used at the connections of strut hangers to overhead supports, or at interhanger locations. A cumulative usage factor was calculated and compared to a fatigue curve. The usage factor was developed based upon the IEEE Standard 344-1975, which states that the maximum number of OBE and SSE events plausible during the power plant's lifetime is five and one, respectively. A factor of safety of 1.5 was applied against the number of fatigue cycles to failure in order to establish an allowable number of design fatigue cycles.

e. An analysis is performed to satisfy the requirements of the SSE and OBE load criteria. Stress levels are analyzed for the SSE load cases, and fatigue resistance is analyzed for critical raceway components under the OBE. The test program results demonstrated that the repetition of many earthquakes up to an equivalent SSE ground motion of 2/3g will not result in a loss of function in the support system or electrical circuitry. In addition, the design procedure considers low cycle fatigue phenomena for connections.

3.10(B).3.2 Testing

The scope of the "Cable Tray and Conduit Raceway Test Program" included the evaluation of a large number of variables in the design of cable trays. Included in the test report are discussions of the following variables:

- Type of tray
- Type and length of hanger
- Location of splices
- Number of tiers
- Trapeze and cantilever support
- Connection details, such as
    - Single clip angle
    - Double clip angle
    - Gusseted clip
    - Tray to strut type hanger
- Type and location of bracing
- Amount of cable fill
- Size and distribution of cables
- Cable ties
- Combined conduit and tray systems
- Sprayed fire protection material

In order to evaluate the effects of these and other variables, over 2,000 individual dynamic vibration tests were performed over a period of 11 months of testing. As a result of these tests, over 50 volumes of raw data were generated and evaluated. The results of the evaluation of these data form the basis for the conclusion contained in the test report and the design recommendations implemented in the Callaway design.

In addition to the wide range of variables that were evaluated, tests were performed on tray and strut systems similar to the SNUPPS design.

As a result of the evaluation of the variables described above and the testing of hardware and support configurations similar to the SNUPPS design, a set of design recommendations was formulated. These recommendations were developed to be generally applicable to a wide variety of hardware and specifically applicable to the support configurations used by this project and the other test program participants. For example, the recommended damping in intermittently braced strut supported trapeze hanger systems was determined from the data of over 100 dynamic tests on these types of systems. Figure 3.10(B)-1 shows the recommended damping as a function of floor acceleration in the form of a bilinear curve. As can be seen from this curve, the recommended damping, for the most part, represents a lower bound of all the data obtained from the test program. Similar conservative recommendations were formulated from the results of the test program for other aspects of design. Consequently, it is concluded that the design recommendations formulated as a result of the cable tray and conduit raceway test program are broadly applicable to the design of strut supported raceway systems and were conservatively applied in the design of the Callaway raceway supports.

3.10(B).3.2.1 Test Conditions

The test fixture used to test cable trays was specifically designed for this test program. Its inverted pendulum design permitted seismic input to suspended tray support systems. Additionally, the fixture was designed to accommodate a 40-foot-long tray system segment of up to 5 tiers and a hanger of up to 13 feet in length. Sufficient width was provided in the test bay to accommodate two parallel runs, including cross connections and attached conduit. This facility allowed for testing of long, multitiered tray systems with various bracing arrangements.

The test program included tests of a large number of varied tray types and support types in various configurations. These test configurations were used during the testing program in order to simulate the actual field installed conditions. Supports with or without bracing and with multitier cable trays were tested. In addition, a combined system configuration comprised of various tray fittings such as tees, elbow, vertical bend, and multitiers of straight cable tray runs was tested. The cable tray and conduit raceway test input loading was applied at 45 degrees (vector biaxial) because the shake table used was limited to vector biaxial motion. In choosing the 45-degree relationship (i.e., horizontal equals vertical), the floor response spectra of many containments and auxiliary buildings were reviewed, and this equality of horizontal and vertical motion was deemed most appropriate.

IEEE-344 and NRC regulatory guides recommend, but do not require, independent biaxial input. In the case of raceways, the modes of vibration are symmetrical and are dominantly either horizontal or vertical and so would be adequately excited by vector biaxial motion. As the different modes of a given raceway generally have quite distinct resonant frequencies, there is no problem introduced by the zero phase between horizontal and vertical loading (i.e., vertical and horizontal responses will be randomly varying in and out of phase even though the vertical and horizontal inputs are in phase).

Independent biaxial input is preferred in nonsymmetrical cases and in the possible but unusual case of testing a structure with a mode whose axis of sensitivity would be at 90 degrees to the vector biaxial input, and hence not excited. The raceways are simple structure systems with distinct vertical, transverse, and longitudinal modes; this was confirmed during testing. Therefore, the test results are not affected by the use of vector biaxial input.

As described above, widely spaced modes of vibration with little cross coupling were observed during the testing. For example, longitudinal swaying modes were quite low (1.8 Hz), transverse modes followed (3.2 Hz) with tray modes following at 6.1 and 15 Hz for a typical 4'6" single tier unbraced raceway. This data is illustrated in Figures 7.8 and 7.13 of Volume 1 (of test report "Cable Tray and Conduit Raceway Seismic Test Program") for a 100-percent cable loaded raceway of 0.10 g peak response. Similar frequency ratios for longer strut hung raceways are illustrated in relevant data.

The purpose of the cable tray test program was essentially to verify the mathematical model used in the analysis, not to seismically qualify the raceway systems by testing only. In view of the scope of testing and the various test setups, it was concluded that these tests simulate conditions encountered in the field and, therefore, the results of the testing would be applicable to the design of cable trays for Callaway.

3.10(B).3.2.2 Modal Damping - Cable Trays

In a linear dynamic analysis, velocity dependant forces (i.e., viscous damping) are introduced to account for various mechanisms of energy dissipation. These mechanisms include such things as: friction and slip-in bolted connections, hysterisis, radiation of energy away from a foundation, the effects of fluids, and no doubt, other mechanisms as well. Since these various mechanisms cannot be accounted for explicitly in a linear analysis, their effect is lumped in a single viscous damping. Dynamic testing is used to determine an effective viscous damping, appropriate for seismic response. This procedure is common to all structural dynamic analysis.

During the cable tray and conduit raceway test program, the random vibration of cables was identified as one of the significant energy dissipating mechanisms. This occurred because the cables represent most of the mass of the system, are able to move relative to each other, and were not rigidly attached to the supporting tray. During the tests, this phenomenon manifested itself as a noticeable relative movement and impact of the cables within the tray. As is the case with other energy dissipating mechanisms, this effect was quantified in terms of an equivalent viscous damping based upon the relationship between the recorded response and the applied input to each test specimen. The test report entitled "Cable Tray and Conduit Raceway Seismic Test Program" provides a detailed discussion of the methods used to compute an equivalent viscous damping from the recorded results of the dynamic tests. This discussion can be found in Section 5 with supplementary information in Appendices G, H, and I.

The computed damping values from the various tests are tabulated in Appendix K of the test report. Data was taken from these tables and plotted as shown in Figure 3.10(B)-1. On this figure, the data points of computed equivalent viscous damping are plotted as a function of input acceleration (floor spectrum ZPA) for over 100 tests of various braced strut hanger tray systems. These results represent all the data from simulated earthquake inputs. Low level sinusoidal and snap back test data are not included, since they are not directly applicable. Since these tests represented a wide variety of tray types, connection details, struts, and cable configurations, there is a broad scatter in the data. These data, however, do clearly show that the recorded responses of the tested tray systems are best described by a dynamic system with an equivalent viscous damping. It should be noted that the data realistically can be utilized with accepted curve fitting techniques to obtain a "best-fit" curve which reflects the statistical average of the test data. Such an approach would result in a maximum damping value far in excess of the conservative 20 percent value. However, in the interest of conservatism, a bilinear curve, which effectively bounds the lower end of nearly all the points, was utilized. This curve is given in Figure 3.10(B)-1. This curve represents the recommended design values of equivalent viscous damping.

Damping of the cable tray system is dependent on the amount of cable in the trays and the input amplitude of vibration. Figure 3.10(B)-2 presents the lower bound values of equivalent viscous damping as a function of input floor response spectrum ZPA and amount of cable in the tray. To be able to use the maximum value of damping, 20 percent, the instructure response spectra must have at least a ZPA value of 0.35 g and the tray must be at least 50 percent full by weight of cable.

In addition to the determination of equivalent viscous damping, as described in the test report, linear analysis was performed on finite element models of several of the tray system test setups. These analyses confirmed that a very high viscous damping was required in order to predict responses similar to those recorded during the dynamic testing. These analyses confirmed that the application of the damping values recommended for design in a linear analysis was consistent with the results of the test program and, therefore, would result in a conservative design of support systems.

3.10(B).3.2.3 Modal Damping - Conduit

During the cable tray and conduit raceway seismic test program, various tests were performed on conduit runs on a trapeze raceway to determine their dynamic characteristics. A large number of variables were considered in this test program. The description and results of conduit raceway testing can be found in Section 8 of the test report.

The critical damping value computed from test data is 7 percent at 0.1 g input acceleration. Higher damping value trend was observed for input acceleration higher than 0.1 g. But at the present time, for design of conduit raceway system it is recommended to use 7 percent critical damping for all levels of input acceleration at and

above 0.1 g. For lower input acceleration, it is recommended to use linear interpolation from 7 percent to 0 percent damping for 0.1 g input to zero input acceleration.

3.10(B).4 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF INSTRUMENTATION PANELS, MOUNTING STRUCTURES FOR FIELD MOUNTED INSTRUMENTS, AND SUPPORTS FOR INSTRUMENT TUBING

3.10(B).4.1 Instrumentation Panels

Table 3.10(B)-1 lists the instrumentation panels required to be qualified as seismic Category I. The methods used to qualify the panels are discussed in the seismic procedures and final test reports maintained in the subject equipment specification files.

3.10(B).4.2 Mounting Structures for Field Mounted Instruments

Mounting structures comply with the following:

a. The mounting structure for Category I instruments has a fundamental frequency of 33 Hz or greater.

b. The stress level in the mounting structure does not exceed the material allowable stress when subjected to the maximum acceleration level of the mounting location. The weight of the instrument and instrument accessories is included.

   Material allowable stress is determined from ASME Section III, Subsection NF or Code Case 1644-6.

3.10(B).4.3 Supports for Instrument Tubing

The Category I instrument tubing systems are supported so that the allowable stresses permitted by ASME Section III are not exceeded when the tubing is subjected to the loads specified in Section 3.9(B).3 for Class 2 and 3 piping.

3.10(B).5 OPERATING LICENSE REVIEW

Results of tests and analyses to demonstrate adequate seismic qualification and implementation for equipment specifications listed in Table 3.10(B)-1 are contained in the seismic procedures and final test reports maintained in the subject equipment specification files.

TABLE 3.10(B)-1 SEISMIC CATEGORY 1 INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN THE BALANCE-OF-PLANT SCOPE OF SUPPLY

Equipment	Specification
Electric Metal Clad Switchgear	E-009
Large Induction Motors	E-012
Load Center Switchgear and Transformers	E-017
Motor Control Centers and Contactors	E-018,E-018A
AC/DC Switchboards	E-020 E-053
Local Control Stations	E-028
Control Switches	E-028A E-028B
Electrical Penetration Assemblies	E-035
Electrical Penetration Modules	E-035B
Batteries and Battery Racks	E-050
Battery Chargers	E-051
Essential Service Water System Dry Transformers	E-075
AC Regulating Transformers	E-077
Load Shedder and Emergency Load Sequencer	E-092
Auxiliary Relay Racks	E-093
Status Indicating System	E-094
Engineered Safety Features Actuation System	J-104
Main Steam and Feedwater Isolation Actuation System	J-105
BOP Computer Termination Cabinets	J-106
Annunciator Termination Isolation Cabinets	J-108
Control Stations	J-110
Electrical Indicators (panel-mounted, vertical scale)	J-110
Instrumentation and Control Cabinets	J-110

TABLE 3.10(B)-1 (Sheet 2)

Equipment	Specification
Recorders	J-110
Main Control Boards	J-200
Main Control Board Components	J-200
Auxiliary Control Panels	J-201
Auxiliary Control Panel Components	J-201
Electronic Pressure and Differential Pressure Transmitters	J-301
Containment Post-LOCA Hydrogen Monitoring Equipment	J-359
Safety-Related Airborne Radioactivity Monitors	J-361
Post-Accident Containment Radiation Monitors	J-361A
Neutron Flux Monitoring Equipment	J-364
Safety-Related Level Transmitters	J-481
Safety-Related Level Switches	J-481
Resistance Temperature Devices	J-558B
Nuclear Service Control Valves - Air-Operated	J-601A J-601B
Nuclear Service Control Valves - Electric Motor-Operated with Modulating Controller	J-601A
Nuclear Service Solenoid Valves	J-603A
Nuclear Service Butterfly Valves - Air Operated	J-605A
Transmitters (mechanical only)	J-1011
Transmitters	J-1030
Transmitters	J-1032
UHS Cooling Tower Fan Motors	M-015
Emergency Diesel Generators	M-018
Auxiliary Feedwater Pump Turbine	M-021
Spent Fuel Pool Cooling Pump Motors	M-084

TABLE 3.10(B)-1 (Sheet 3)

Equipment	Specification
Emergency Fuel Oil Transfer Pump Motors	M-087
Containment Spray Pump Motors	M-088
ESW Self-Cleaning Strainer Motors	M-154
Limit Switches	M-221
Motor-Actuated Gate Valves	M-223A M-223C M-224B M-225 M-231B M-231C
Motor-Actuated Globe Valves	M-231B
30-inch Butterfly Valves with Limitorque Operators	M-235
Motor-Actuated Butterfly Valves	M-236,M-237
Containment Purge and Mini-Purge Butterfly Valves with Bettis Actuators	M-237
Room Coolers	M-612, M-1089
Safety-Related Fans	M-619.2
Hydrogen Mixing Fans	M-619.3
Containment Cooler Motors	M-620
Air Cleaning Devices	M-621
Packaged Air Conditioning Units	M-622.1
HVAC Dampers	M-627A
Main Steam and Main Feedwater Isolation Valves	M-628 M-630

3.10(N) SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that instrumentation and electrical equipment classified as Seismic Category I are capable of performing designated safety-related functions in the event of an earthquake. The information presented includes identification of the Category I instrumentation and electrical equipment that are within the scope of the Westinghouse nuclear steam supply system (NSSS), the qualification criteria employed for each item of equipment, the designated safety-related functional requirements, the definition of the applicable seismic environment, and documentation of the qualification process employed to demonstrate the required seismic capability.

3.10(N).1 SEISMIC QUALIFICATION CRITERIA

3.10(N).1.1 Qualification Standards

NRC recommendations concerning the methods to be employed for seismic qualification of electrical equipment are contained in Regulatory Guide 1.100, which endorses IEEE-344-1975. The qualification of NSSS-supplied equipment meets this standard, as modified by Regulatory Guide 1.100, by either type test, analysis, or an appropriate combination of these methods. Westinghouse-supplied equipment is qualified by employing the methodology described in Reference 1.

According to Regulatory Guide 1.89, qualification of equipment for plants in the stage of construction permit application and having the issue date of the Safety Evaluation Report after July 1, 1974 must take into account aging and environmental effects prior to seismic qualification, as specified in the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974. SNUPPS has committed to meet IEEE Standard 323-1974. Required seismic tests conform to the procedures specified in IEEE Standard 344-1975 which account for multiaxis and multifrequency effects of seismic excitation and fatigue effects caused by a number of OBE events. This commitment will be satisfied by implementation of the final NRC-approved version of Reference 1. Reference 2 presents the Westinghouse testing procedures used to qualify equipment by type testing. Seismic qualification testing of this equipment to IEEE Standard 344-1971 is documented in References 3 through 8. Reference 9 presents the theory and practice, as well as justification, for the use of single axis sine beat test inputs used in the seismic qualification of electrical equipment. In addition, it is noted that Westinghouse has conducted a seismic qualification "Demonstration Test Program" (Ref. 10) to confirm equipment operability during a seismic event.

For the seismic qualification of Westinghouse electrical equipment outside of the containment, the above-noted demonstration test program, in conjunction with the justification for the use of single axis sine beat tests (presented in Ref. 14) and the original tests (documented in Ref. 3 through 8 and 13), meets the requirements of IEEE Standard 344-1975.

Thus, since the "Demonstration Test Program" was successfully completed, the equipment's operability has been demonstrated to meet the requirements of IEEE Standard 344-1975.

The acceptability criteria for the SSE notes that there may be permanent deformation of the equipment provided that the capability to perform its function is maintained.

3.10(N).1.2 Performance Requirements for Seismic Qualification

Reference 11 contains an equipment qualification data package (EQDP) for every item of instrumentation and electrical equipment classified as Seismic Category I within the Westinghouse NSSS scope of supply. Table 3.10(N)-1 identifies the Category I equipment supplied by Westinghouse for this application and references the applicable EQDP contained in Supplement 1 to Reference 1. Each EQDP in Supplement 1 contains a section entitled "Performance Specifications." This specification establishes the safety-related functional requirements of the equipment to be demonstrated during and after a seismic event. The required response spectrum (RRS) employed by Westinghouse for generic seismic qualification is also identified in the specification, as applicable.

3.10(N).1.3 Acceptance Criteria

Seismic qualification must demonstrate that Category I instrumentation and electrical equipment are capable of performing designated safety-related functions during and after an earthquake of magnitude up to and including the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) without the initiation of undesired spurious actuation which might result in consequences adverse to safety. The qualification must also demonstrate the structural integrity of mechanical supports and structures at the OBE level. Some permanent mechanical deformation of supports and structures is acceptable at the SSE level providing that the ability to perform the designated safety-related functions is not impaired.

3.10(N).2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

In accordance with IEEE 344-1975, seismic qualification of safety-related electrical equipment is demonstrated by either type testing, analysis, or a combination of these methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including practicality, complexity of equipment, availability of previous seismic qualification to earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the individual Equipment Qualification Data Packages (EQDPs) of Reference 11.

3.10(N).2.1 Seismic Qualification by Type Test

From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE 344-71 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 2. In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Demonstration Test Program (Reference 10). This retesting was performed at the request of the NRC staff on selected items of equipment employing multi-frequency, multi-axis test inputs (Reference 12) to demonstrate the conservatism of the original sine-beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE 344-1975.

The original single axis sine beat testing and the additional retesting completed under the Demonstration Test Program has been the subject of generic review by the staff. For equipment which has been previously qualified by the single axis sine beat method and included in the NRC seismic audit and, where required by the staff, the Demonstration Test Program (Reference 10), no additional qualification testing is required to demonstrate acceptability to IEEE 344-1975 provided that:

a. The Westinghouse aging evaluation program for aging effects on complex electronic equipment located outside containment demonstrates there are no deleterious aging phenomena. In the event that the aging evaluation program identifies materials that are marginal, either the materials will be replaced or the projected qualified life will be adjusted.

b. Any changes made to the equipment due to a. above or due to design modifications do not significantly affect the seismic characteristics of the equipment.

c. The previously employed test inputs can be shown to be conservative with respect to applicable plant-specific response spectra.

This equipment is identified in Reference 1, Table 7.1 and the test results in the applicable EQDPs of Reference 11.

For equipment tests after July 1974 (i.e., new designs or equipment not previously qualified or previously qualified that does not meet a., b., and c. above), seismic qualification by test is performed in accordance with IEEE 344-1975. Where testing is utilized, multi-frequency multi-axis inputs are developed by the general procedures outlined in Reference 14. The test results contained in the individual EQDPs of Reference 11 demonstrate that the measured test response spectrum envelops the applicable required response spectrum (RRS) defined for generic testing as specified in Section 1 of the EQDP (Reference 11). Qualification for plant use is established by verification that the generic RRS specified by Westinghouse envelops the SNUPPS

response spectra. Alternative test methods, such as single frequency, single axis inputs, are used in selected cases as permitted by IEEE 344-1975 and Regulatory Guide 1.100.

3.10(N).2.2 Seismic Qualification by Analysis

Employing motors as an example, the structural integrity of safety-related motors is demonstrated by a static seismic analysis, in accordance with IEEE 344-1975, with justification. Should analysis fail to show the resonant frequencies to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, and 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 1) maximum rotor deflection, 2) maximum shaft stresses, 3) maximum bearing load and shaft slope at the bearings, 4) maximum stresses in the stator core welds, 5) maximum stresses in the stator core to frame welds, 6) maximum stresses in the motor mounting bolts and, 7) maximum stresses in the motor feet.

The analytical models employed and the results of the analysis are described in Section 4 of the applicable EQDPs (Reference 11).

3.10(N).3 METHODS AND PROCEDURES FOR QUALIFYING SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Where supports for the electrical equipment and instrumentation are within the Westinghouse NSSS scope of supply, the seismic qualification tests and/or analysis are conducted including the supplied supports. The EQDPs contained in Reference 11 identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure subsequent SNUPPS installation does not prejudice the generic qualification established by Westinghouse.

3.10(N).4 OPERATING LICENSE REVIEW

The results of tests and analyses that ensure that the criteria established in Section 3.10(N).1 have been satisfied employing the qualification methods described in Section 3.10(N).2 and 3.10(N).3 are included in the individual EQDPs contained in Reference 11.

3.10(N).5 REFERENCES

1. Butterworth, G. and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587, Revision 6A, November, 1983.

2. Morrone, A., "Seismic Vibration Testing with Sine Beats," WCAP-7558, October, 1971.

3. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment," WCAP-7397-L (Proprietary) January, 1970 and WCAP-7817 (Non-Proprietary), December, 1971.

4. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)," WCAP-7397-L, Supplement 1 (Proprietary) January, 1971 and WCAP-7871, Supplement 1 (Non-Proprietary), December, 1971.

5. Potochnik, L. M., "Seismic Testing of Electric and Control Equipment (Low Seismic Plants)," WCAP-7817, Supplement 2, December, 1971.

6. Vogeding, E. L., "Seismic Testing of Electric and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)," WCAP-7817, Supplement 3, December, 1971.

7. Reid, J. B., "Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants)," WCAP-7817, Supplement 4, November, 1972.

8. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel) (Low Seismic Plants)," WCAP-7817, Supplement 5, March, 1974.

9. Fischer, E. G. and Jarecki, S. J., "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974," WCAP-8373, August, 1974.

10. Letter NS-CE-692, dated July 10, 1975, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC).

11. EQDP "Equipment Qualification Data Packages," Supplement 1 to WCAP-8587.

12. Jarecki, S. J., "General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary).

13. Figenbaum, E. K. and Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (Type DB Reactor Trip Switchgear)," WCAP-7817; Supplement 6, August, 1974.

14. Kelly, R. E. and McInerey, J. J., "Methodology for the Seismic Qualification of Westinghouse WRD Supplied Equipment," WCAP-9714-PA (Proprietary), WCAP-9750-A (Non-Proprietary).

TABLE 3.10(N)-1 SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

Equipment	EQDP
Pressure Transmitters	ESE-1A, 1B and 2
Differential Pressure Transmitters	ESE-3 and 4
Resistance Temperature Detectors (Well-Mounted)	ESE-6
Resistance Temperature Detectors (Strap-On)	ESE-42A
Solid State Protection System and Safeguards Test Cabinets (2 Train)	ESE-16
Nuclear Instrumentation System Cabinets	ESE-10
Reactor Trip Switchgear	ESE-20
Excore Neutron Detectors (Power Range)	ESE-8
7300 Process Protection System Cabinets	ESE-13A, B, C, D
Remote Digital Display and Printer	ESE-46A, B
High Volume Pressure Sensors	ESE-48A
Core Cooling Monitor Microprocessor	ESE-51
Incore Thermocouples and Connectors	ESE-43A, J1064
Incore Thermocouple Reference Junction Box	ESE-44A, Z
Hydraulic Isolators	ESE-49A
Pressure Sensors	ESE-21
Differential Pressure Indicating Switches Group B	ESE-40A
Boron Dilution Mitigating Equipment	ESE-47A, B, C
Indicators (Post-Accident Monitoring)	ESE-14
Safety-Related Valve Electric Motor Operators	HE-1 and 4
Safety-Related Solenoid Valves	HE-2/5
Safety-Related Externally Mounted Limit Switches	HE-3/6
Pressurizer Safety Valve Position Switches	HE-7, 7Z

TABLE 3.10(N)-1 (Sheet 2)

Equipment	EQDP
Electrical Connectors for Solenoid Valves and Limit Switches	HE-8
PORV Solenoid-Operated Pilot Valves and Position Indicators	HE-9
Head Vent System	HE-10A, B, C
Hydrogen Recombiners	SP-1
Large Pump Motors	AE-2
Canned Pump Motors	AE-3
Operator Interface Modules	ESE-12A
DS-416 STA and Auto Shunt Trip Panel	ESE-62A

(See also the J-1011, J-1030 and J-1032 specifications of Table 3.10(B)-1, and specification M-2012(Q), "Technical Specification for the Replacement Reactor Vessel Closure Head (RRVCH) Project. This equipment was purchased by U.E. for use in NSSS equipment applications.)

3.11(B) ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section provides information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features, the reactor protection systems, and other safety-related systems are designed to ensure acceptable performance during normal and design basis accident (DBA) environmental conditions. This section includes that information related to balance of plant (BOP) systems, whereas Section 3.11(N) includes that information related to the NSS system.

Tables 3.11(B)-1 and 3.11(B)-2 provide the normal and DBA environmental conditions for all primary mechanical and electrical equipment within the plant, including BOP and NSSS scope of supply. Table 3.11(B)-3 provides a listing of safety-related equipment and identifies specific equipment and components required for a DBA and/or safe shutdown of the plant.

A review of equipment environmental qualification programs against NUREG-0588 positions was performed. The scope of the review was limited to plant areas exposed to harsh environments following a loss of coolant accident (LOCA), a main steam line break (MSLB), or a high energy line break (HELB). Table 3.11(B)-7 lists the equipment specifications reviewed under the NUREG-0588 program.

The Callaway design is based on utilizing only Class 1E powered electrical equipment to meet the criteria specified under Safety-Related System Listing in Section 3.11(B).1.1. Table 3.11(B)-3 includes all safety-related electrical equipment, regardless of the accident that required the equipment to be categorized as Class 1E. No Class 1E equipment is excluded from the list due to location or any other reason. Section 7.1.1, Identification of Safety-Related Systems, identifies the criteria for the selection of instrumentation and controls (I&C) equipment as being safety related.

Plant hazards, seismic/nonseismic interaction, control room fire hazards analysis, and other integrated design reviews have been conducted to ensure the validity of this design concept. Appendix 3B discusses the Callaway hazards review program. Additionally, Section 3.11(B).7 discusses a review of the safety-related and nonsafety-related control system interfaces. Accordingly, there is no nonsafety-related equipment needed to support, or whose failure could prevent, a safety function of the safety-related equipment.

Appendix 7A identifies the Union Electric position on Regulatory Guide 1.97. A categorized list of equipment is included in Appendix 7A. All Regulatory Guide 1.97 Category I instruments are included in Table 3.11(B)-3. Additionally, all Category II electrical components powered by a Class 1E power source (as shown in Appendix 7A) are also included in Table 3.11(B)-3.

Section 3.7(B) describes the seismic design bases for the plant. Sections 3.9(B) and 3.10(B) describe the seismic qualification programs for seismic Category I instrumentation, mechanical, and electrical equipment.

Environmental design criteria for the facilities conform to 10 CFR 50, Appendix A, General Design Criteria 1, 2, 4, 23, and 50, as discussed in Section 3.1.

3.11(B).1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

3.11(B).1.1 Equipment and System Lists

Table 3.11(B)-3 identifies the safety-related equipment and components required to mitigate the consequences of a DBA and to ensure the safe shutdown of the plant. This table also gives the room number in which the equipment is located and the equipment category as defined in NUREG-0588, Appendix E. Refer to Figure 12.3-2 for room locations.

3.11(B).1.1.1 Equipment List Development

To develop the safety-related equipment list in Table 3.11(B)-3, for the NUREG-0588 review, the Project Q-List and FSAR were used to identify systems and major components. This information was used to enter design documents for a more detailed equipment listing. Examples of design documents utilized include piping and instrument drawings, the instrument index, the "Q" instrument list, equipment listings, equipment specifications, and the Westinghouse Project Information Package. To ensure completeness, several measures were taken. The list was prepared and independently checked. The list was then compared to other master equipment listings, valve logs, and a special sort of the project electrical circuit schedule. This sort of the circuit schedule provided a listing of all pieces of equipment to which Class 1E cables were connected. When the list was completed, Union Electric checked the list by verifying its completeness for various systems.

Upon completing the list, the location of each piece of equipment, by room number, was then identified. Once the list was developed and equipment location identified, each piece of equipment was categorized according to NUREG-0588, Appendix E, for each of three accident groups. The accident groupings were LOCA, MSLB/MFLB inside containment and steam/feedwater tunnel, and HELB outside containment (except MSLB and MFLB). The equipment located in a harsh environment, for any of the three accident groups, was reviewed under the NUREG-0588 program.

Equipment required as a result of NUREG-0737 was incorporated into the design by the time Table 3.11(B)-3 was being developed. Accordingly, new equipment added to the design was included in the equipment list and reviewed to the same criteria as all other safety-related equipment. FSAR Section 18.0 identifies the Callaway design relative to the NUREG requirements.

3.11(B).1.1.2 Safety-Related System Listing

Safety-related systems are those plant systems necessary to ensure:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor and maintain it in a safely shutdown condition.

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

Systems that perform these type functions are those systems required to achieve or support emergency reactor shutdown, containment isolation, reactor core cooling, containment heat removal, core residual heat removal, and prevention of significant release of radioactive material to the environment. The listing of systems that perform or support these functions is contained in the Callaway Equipment List (CEL). The specific safety function of each system is described in FSAR system description sections and in the CEL database.

Class 1E powered I&C devices are included in the system that they serve (e.g., EG-FT-0108 is a flow transmitter in the component cooling water system [EG]). The I&C devices can be divided into two categories, NSSS and BOP supplied. Each type can be identified in the fourth column of Table 3.11(B)-3. The BOP supplied devices that are purchased by the Bechtel I&C Group have a specification number that begins with the letter J (e.g., J-301 for EG-FT-0108). The NSSS-supplied devices are identified in the fourth column by the respective Westinghouse EQDP number (e.g., ESE-4).

3.11(B).1.2 Plant Environments

3.11(B).1.2.1 Normal Environments

Pressure, Temperature, Humidity, and Radiation

Normal operating environmental conditions are defined as conditions existing during routine plant operations. These environmental conditions, as listed in Table 3.11(B)-1, represent the normal maximum and minimum conditions expected during routine plant operations.

Dust

In the NUREG-0588 review, dust was considered and was determined to be an insignificant factor in equipment qualification because outside air sources and ventilation units are typically equipped with filters which remove airborne dust. Also concrete

coating, plant housekeeping, dust seals, and equipment maintenance requirements provide assurance that dust will not degrade equipment performance.

3.11(B).1.2.2 Accident Environments - Inside Containment

Accident environmental conditions are defined as those deviating from the normal operating environmental conditions. These conditions are specified in Table 3.11(B)-2.

In the NUREG-0588 review, Callaway LOCA/HELB/MSLB pressure, temperature, humidity, radiation, chemical spray, and submergence environmental conditions were evaluated. Where required, plant-unique environmental conditions were developed using the Category I criteria of NUREG-0588. The development of these conditions is described below. The post-accident parameters used in the equipment review are provided in summary form in Table 3.11(B)-2 and as used in the review, in Figures 3.11(B)-1 through 84. HELB P/T curves are also located in Reference 24.

Radiation

Using the guidance of NUREG-0588, post-LOCA radiation environments were determined in all areas of the containment. The original fission product release data used in this analysis were obtained from Westinghouse. The isotopic inventory provided by Westinghouse was for an equilibrium cycle Callaway core. The data were calculated at the end of cycle life and, therefore, represent maximums suitable for post-accident evaluations. This source term is referred to as the licensing basis EQ source term, applicable to the initial core load. Subsequent cycles have seen changes in fuel type (from STD/LOPAR to OFA to VANTAGE 5 to VANTAGE+), power level (from 3425 MWt to 3579 MWt), and burnup (up to 60,000 MWd/MTU as discussed in Section 4.2.1). The doses reported in Table 3.11(B)-4 have been increased by 5% to account for these effects. In addition, the airborne gamma doses were increased by another 3% to account for the replacement of the active spray additive system with a passive system of baskets adjacent to the containment recirculation sumps containing trisodium phosphate. The following discussion refers to the initial calculations performed with the licensing basis EQ source term and a 50% cesium release fraction.

The accident scenario assumed that a LOCA event occurred causing core damage. The entire source of 100 percent noble gas inventory, 50 percent of the core halogen inventory, 50 percent of the cesium, and 1 percent of the other solids was released to the containment. This release was conservatively assumed to occur at time zero. For the liquid source, 50 percent of the halogens, 50 percent of the cesium, and 1 percent of the remaining fission product solids were assumed to go directly to the sump and were diluted by the volume of the refueling water storage tank (RWST) and the liquid volume of the reactor coolant system. For the airborne source, 100 percent of the noble gases and 50 percent of core halogens were assumed to be released to the free volume of the containment. The simultaneous release of 50 percent of the halogens to the atmosphere and to the sump introduced additional conservatism.

Credit was taken for mechanistic removal of the airborne iodine via containment spray and plateout. The spray removal lambdas for elemental and particulate iodine ($25.7\ hr^{-1}$ + $0.73\ hr^{-1}$) were taken from the calculated values listed in Table 6.5-2. The plate-out removal lambda ($15.8\ hr^{-1}$) was calculated using methodology outlined in NUREG/CR-0009. The surface area available for plateout was assumed to be equivalent to the heat sink area used in the containment pressure analysis given in Table 6.2.1-4. In addition, two of the four hydrogen mixing fans were assumed to be operating, at 42,500 cfm each, to provide mixing between the sprayed (86 percent) and unsprayed (14 percent) regions of the containment. These removal processes were assumed to persist until the elemental and particulate iodine in the sprayed region were reduced by factors of 200 and 10,000, respectively.

These decontamination factors (DFs) were taken from Reference 22. The spray removal rate for elemental iodine was calculated in Section 6.5A.2 to be $25.7\ hr^{-1}$. This spray removal rate plus the plateout removal rate ($25.7\ hr^{-1}$ + $1.58\ hr^{-1}$) were assumed to be effective in the sprayed region until an elemental iodine decontamination factor (DF) of 200 was reached in the EQ dose calculations. Only the plateout removal rate was assumed to be effective in the unsprayed region until an elemental iodine DF of 2 was reached in the EQ dose calculations. The spray removal rate for particulate iodine was calculated to be $0.73\ hr^{-1}$ in Section 6.5A.1 and was assumed to be effective in the sprayed region until a particulate iodine DF of 10,000 was reached in the EQ dose calculations.

It is noted that the offsite and control room doses discussed in Section 15.6.5 were calculated using an elemental iodine spray removal rate of $10\ hr^{-1}$ and a particulate iodine spray removal rate of $0.45\ hr^{-1}$, until a DF of 28.7 was reached for elemental species and a DF of 50 was reached for particulate species. No plateout removal lambda was used in the Section 15.6.5 dose calculations since credit was taken for the instantaneous plateout of half of the iodines released to the containment atmosphere (i.e. 25% of the core iodines).

With the replacement of the spray additive system with trisodium phosphate baskets, the minimum equilibrium sump fluid pH is reduced to 7.1. This reduced pH results in a reduced spray partition coefficient (H, from Equation 6.5A-15 on page 6.5A-7) of 1100 per Reference 22. Using Equation 6.5A-15, the resulting elemental iodine DF was calculated to be 28.7 for the analysis of offsite and control room doses discussed in Section 15.6.5. Per Reference 23, the particulate iodine spray removal rate, calculated using Equation 6.5A-1 on page 6.5A-2, can conservatively be based on an assumed E/D of 10 per meter initially, changing to 1 per meter after a DF of 50. After the particulate iodine spray removal rate is reduced, there is no DF limit. However, for simplicity and conservatism, removal was assumed to stop after a DF of 50 was reached in the analysis of offsite and control room doses. With consideration given to these reduced DF values for elemental and particulate iodines, airborne gamma doses listed in Table

3.11(B)-4 have been estimated to increase by 3% as a result of the use of the trisodium phosphate baskets.

To determine the gamma dose rate inside the containment, the multigroup, three-dimensional, point kernal code QAD-CG was used to take credit for all major internal structures. The containment was divided into regions, and the maximum dose rate within each region as a function of time was determined. These dose rates were assumed to apply to all equipment within that region. Each dose rate was numerically integrated to obtain the 180-day integrated dose for each region. The beta dose rate as a function of time was obtained assuming a semi-infinite cloud model. These dose rate values were also numerically integrated to obtain the 180-day beta doses for each region. The gamma plate-out was modeled using a cylinder with a height and radius equal to that of the containment. The dose rate was obtained at the center of the cylinder without taking credit for air attenuation. Beta dose rate contributions due to plate-out were obtained assuming a contact dose rate.

The resulting containment integrated dose curves are provided as Figures 3.11(B)-50 through 3.11(B)-84.

Per the commitments to Regulatory Guides 1.7 and 1.89 in Appendix 3A, a 1% cesium source term is sufficient for Callaway. However, the radiation levels reported in Table 3.11(B)-4, obtained using a 50% cesium source term, were utilized during the NUREG-0588 review. Due to the extreme conservatism in the equipment specifications, most components were qualified to this radiation level. For the isolated cases where the 50% cesium source term radiation proved too severe (i.e. electrical specifications E-018, J-301, J-481, J-1030, ESE-3A and mechanical specifications ESE-21, ESE-48A), the equipment was evaluated against a 1% cesium source term.

Pressure, Temperature, and Humidity

Callaway unique containment pressure-temperature profiles were utilized for the current equipment evaluation to NUREG-0588. The temperature and pressure conditions were evaluated for both LOCA and MSLB accidents. The resulting containment temperature and pressure profiles are provided in Figures 3.11(B)-1 through 6. The maximum containment temperatures and pressures for LOCA and MSLB conditions are provided in Table 6.2.1-2.

For the evaluation of equipment located inside containment, pressure-temperature enveloping profiles for Callaway have been generated. These environments were generated for a spectrum of MSLBs and LOCAs. For LOCAs, double-ended breaks in the pump suction line were evaluated. A double-ended hot leg break was also analyzed. For the main steam lines, a spectrum of break sizes (split and double-ended) at various power levels with minimum entrainment were evaluated. For these evaluations, loss of offsite power and a worst single failure were assumed. Pressure and temperature mitigation from the operation of safety-related containment sprays, air coolers, and heat transfer to structures was considered.

All methods applied in the determination of environments are in accordance with Sections 1.1 and 1.2 of NUREG-0588, Revision 1 for Category I plants. The evaluation of mass and energy release rates is as described in Section 6.2.1. The evaluation of the containment environmental response is as described in Section 6.2.1.

For MSLB environments, credit was sometimes taken for specific equipment surface temperature response. The methods used to calculate equipment surface temperatures are described in Bechtel Topical Report BN-TOP-3, Revision 4, Section 3.4. The Bechtel standard computer program COPATTA (NE100) was used to model the containment. Westinghouse-supplied blowdown data based on the old steam generators, the performance of the various engineered safety features, and heat sink data were input to the program and the resulting containment pressure and temperature as well as the heat sink temperatures were calculated. The equipment of interest was modeled as a heat sink in the containment model and its temperature was calculated as part of the COPATTA calculation. The heat transfer methods used to model the equipment heat sinks were taken from NUREG-0588, Revision 1, Appendix B. The heat transfer rate equations and the convective and condensing heat transfer coefficients used in the COPATTA analysis of equipment surface temperature were taken directly from NUREG-0588, Appendix B. An example of the heat transfer model of a typical motor-operated valve is shown in Figure 3.11(B)-49.

A typical selection of representative equipment and components designed to accomplish protective actions in response to a design basis event and thus requiring environmental qualification includes motor-operated valves, containment penetrations, electronic differential pressure transmitters, and cables.

To conduct a transient temperature analysis with the COPATTA code, equipment modeling was required. The technique adopted, as outlined below, was primarily based on equipment size (heat transfer area) and material properties (thermal conductivity):

a. Motor-operated valves were modeled as a slab. The air gap was reduced to maximize heat transfer to the inside and the wall thickness utilized was smaller (conservative) than any Callaway motor-operated valves.

b. Containment penetrations were modeled as a slab, with a steel cover, air gap, and cable consisting of insulation and a copper core. Again, the air gap was reduced to maximize heat transfer to the inside.

c. Electronic differential pressure transmitters were modeled as a slab consisting of a cast aluminum cover and an air gap to a copper wire.

d. Power, control, and instrument cables were modeled as a cylinder consisting of jacketing, insulation, and a copper core in the most conservative configuration relative to the cable installed at Callaway.

Further Callaway specific information regarding the determination of mass and energy releases and the containment environmental response is provided in detail in Section 6.2. Surface temperature curves in Figures 3.11(B)-7, 3.11(B)-7A, and 6.2.1-85 are based on the old steam generator design, but remain conservative for the replacement steam generators.

Table 6.2.1-56 lists the 24 cases that were analyzed to determine the worst case containment pressures and temperatures following a main steam line break.

There was no impact on equipment qualification as a result of the plant uprating to 3579 MWT.

Containment Spray

The Callaway design utilizes two redundant trains to supply containment spray for temperature and pressure reduction and fission product removal from the containment atmosphere. Table 3.11(B)-5 identifies the containment spray requirements. The Standard Review Plan indicates that single failures should be evaluated to determine the worst case chemical concentrations. The worst case concentrations are pH = 4.0 and pH = 11.0, as discussed in Section 6.5.2.3.

A caustic spray with an upper limit of pH = 11.0 will be used in EQ reviews. A boron concentration of 2050 ppm was used in the EQ reviews. The Cycle 4 change to an RWST boron concentration of 2350-2500 ppm has a negligible effect on peak pH, therefore the corrosive effects of the containment spray are not increased. As such, there is no adverse EQ impact arising from this change in RWST boron concentration.

3.11(B).1.2.3 Accident Environments - Outside Containment

Radiation

Using the guidance of NUREG-0588 and NUREG-0737, post-LOCA dose rates and doses were determined in those areas of the auxiliary building where safety-related equipment qualification would be reviewed. The fission product release data used in this analysis were the same as discussed in Section 3.11(B).1.2.2. The analysis for the auxiliary building yielded a conservative upper bound estimate for the doses to all safety-related electrical equipment as required by NUREG-0588. See Section 3.11(B).1.2.2 regarding source term changes since the initial core load. With the replacement of the spray additive system with trisodium phosphate baskets in the containment recirculation sumps, the doses in penetration rooms 1409-1412 and 1506-1509 in Table 3.11(B)-2 have been estimated to increase by 8% due to the harder spectrum of gamma energies associated with the iodines. The following discussion refers to the initial calculations performed with the licensing basis EQ source term and a 50% cesium release fraction.

For those systems containing pressurized reactor coolant, 100 percent of the noble gases, 50 percent of the iodines, 50 percent of the cesium, and 1 percent of the other particulates were assumed to be present with a dilution volume equal to the reactor coolant system liquid volume. Systems containing recirculating liquid were assumed to have 50 percent of the halogens, 50 percent of the cesium, and 1 percent of the other particulates diluted by the RWST liquid volume and the liquid volume of the reactor coolant system. The contained airborne sources were assumed to have 100 percent of the noble gases and 25 percent of the iodines. The dilution volume for the contained airborne source was the entire volume of the containment.

The resulting accident total integrated doses for the rooms with safety-related electrical equipment in the auxiliary building are provided in Table 3.11(B)-2. The values provided are the doses that result from both radiation penetrating the containment and radiation from recirculating sump fluids. The values provided are the worst case for the identified room. When the worst case values exceed the qualified dose, additional analyses have been performed to provide total integrated doses for specific equipment. This was accomplished by performing a location-specific calculation of the dose to the component to more accurately define the actual environment in which the component would be expected to operate following an accident. This approach often provided a substantially lower dose than the worst case dose if the component of interest was not extremely close to a major cluster of pipes (the worst case dose point always is). The location-specific dose calculations used the following techniques to reduce the dose:

a. Direct doses from contained sources were calculated for each dose point.

b. Geometry reduction factors were used for reducing doses from penetration/duct streaming.

c. Finite cloud beta dose was calculated for small enclosures.

To determine the location-specific gamma dose from contained sources, major piping and components were modeled accounting for the geometry of each case and any intervening shielding structures. The dose rate from these sources was calculated using a point kernel computer code that utilizes the semiempirical methods developed by T. Rockwell (Reference 6) for calculating the direct gamma dose from a homogeneous cylindrical volumetric source through slab shields. Individual buildup factors for source materials and shield materials were taken from the work of Capo (Reference 7). Broder's method (Reference 8) was used in the code to accommodate multilayer shield buildup. The dose rates determined using this code were then numerically integrated to determine the 6-month integrated dose.

The second major technique used was to reduce the penetration streaming component of the dose. The basic radiation source in this case is the post-accident containment airborne source (noble gases and halogens) assumed to be distributed uniformly within the containment free volume. The effective source is the radiation that shines or streams through the containment penetrations. This component had been incorporated into the

worst case doses in a conservative manner. As an example, the streaming dose contribution stated for the electrical penetration room is the sum of the dose at each of the penetration exits. This is conservative for two reasons:

a. No one point in space will receive the entire sum of the exit doses from all the penetrations.

b. Very little equipment is located at the containment wall directly in front of the penetration exit.

The dose at the penetration exit had been calculated by first determining the dose rate just inside the containment wall at the proper elevation and azimuth. This was accomplished by using the multigroup, three-dimensional point kernel computer code QAD-CG (Reference 9). Knowing this, the dose rate at the penetration exit is determined by calculating an annular reduction factor for the penetration. This annular reduction factor is strictly a function of geometry--dependent on the penetration length, radius, and configuration (circular or annular). The annular reduction factor was calculated using the "ray-analysis" technique. A computer code was written to assess this reduction by numerically solving the integral equation documented in Reference 10. The product of the dose rate just inside the containment wall and the annular reduction factor yields the dose rate at the outer surface of the containment on the centerline of the penetration. The dose rate calculated at the penetration exit was then numerically integrated to obtain the 6-month integrated dose.

The dose contribution at a specific location is further determined by assessing the geometrical attenuation incurred going from the actual location of each penetration to the dose point of interest. This geometrical attenuation, or more properly solid angle attenuation, factor is determined using the work by J. H. Hubbell, et al (Reference 11) to describe the detector response to a finite plane circular source. This geometrical reduction factor is rigorously the fraction of the solid angle subtended at a point in space. A computer code was developed to evaluate the solid angle by numerically evaluating the elliptic integrals. This was accomplished using the work by A. V. Masket (Reference 12). The geometrical reduction factor DW was evaluated assuming the angular distribution of the $\frac{\Delta\Omega}{\Omega}$ source emerging from the penetration is cosine shaped.

The third dose reduction method used was for beta radiation rather than gamma. The calculated airborne beta dose is based on a semi-infinite cloud model using the methodology discussed in Section 3.11(B).1.2.2 under Radiation. For small volumes such as NEMA enclosures (electrical boxes), the semi-infinite cloud model is very conservative. To be more rigorous, a finite cloud correction was applied to the semi-infinite cloud dose. The technique developed to perform this correction is based on empirical relationships developed by R. Loevinger (Reference 13). This correction is based on the geometry of the enclosure (size and shape) and the end-point energy of the contributing beta particles. Because of this dependence, the finite cloud correction

factor had to be evaluated for each individual beta for each of the isotopes considered for each enclosure size.

It was conservatively assumed in the analyses that the airborne activity concentration in the enclosures was identical to the containment atmosphere concentration.

No credit was taken for the actual time delay it would take the atmosphere of the box to reach equilibrium with the containment atmosphere. Since this correction was applied to the dose, an integral quantity, the dynamics of the source, i.e., decay, containment spray, and plateout removal of the iodine, and mixing caused by the containment mixing fans have already been accounted for in the semi-infinite cloud calculation.

The equipment specific analyses still provide conservative results.

Pressure, Temperature, and Humidity

In the NUREG-0588 review, the equipment qualification temperature and pressure environments for postulated MSLBs and HELBs outside the containment were determined based on a conservative model as summarized below:

a. Room pressure and temperature profiles were generated to determine the worst local environments.

b. No credit was taken for cooling by non-Class 1E HVAC.

c. The only mechanism considered for temperature dissipation was a conservative model of heat transfer to passive heat sinks.

d. Conservative break isolation times were used.

These items are discussed below in greater detail.

Room pressure and temperature profiles were generated to determine the worst local environments. Maximum humidity values were also established for the analyzed pressure/ temperature profile cases using the assumptions which maximize the pressure/temperature conditions. Environments were determined based on compartmental analyses and, hence, environments for rooms downstream of a break volume were also determined (i.e., the adjoining rooms were analyzed to determine the effects of breaks). Rooms or volumes selected as compartments were sufficiently defined such that the calculated compartment average temperature appropriately describes the local temperature.

Venting out of the compartment was conservatively modeled so that pressurization was adequately determined. Superheat was modeled in calculation of the compartment temperatures.

No cooling credit was taken for non-Class 1E HVAC for breaks outside of the containment. Operation of these systems would reduce the severity of the temperature and pressure qualification environments. The HVAC system exhaust ducting was used as a transfer mechanism of the break energy.

The modeled mechanism for heat removal was heat transfer to passive heat sinks. Passive heat sinks were conservatively calculated for each qualification environment. Treatment of passive heat sinks is described in Bechtel Topical BN-TOP-3, Revision 4. The lower bounding Uchida condensing heat transfer correlation was modeled for condensing heat transfer. Additional HELB analyses, performed with the GOTHIC 7.2b computer code (Reference 25), use the diffusion layer model (DLM) for condensing heat transfer. Predicted temperatures using either of these correlations are significantly higher than experimentally measured temperature.

Credit was taken for action of automatic break isolation. For breaks outside of the main steam tunnel area which did not have automatic isolation systems, a 1/2-hour manual isolation time was used.

The GOTHIC 7.2b computer code or methods described in Bechtel Topical Report BN-TOP-4, Revision 1, and presented in Section 3.6 were used to determine local pressures and temperatures. Both GOTHIC 7.2b and Bechtel Topical Report BN-TOP-4 have been reviewed and approved by the NRC for use in subcompartment pressure and temperature analysis. Section 3.6 provides a detailed description of the methodology utilized in identifying, analyzing, and evaluating high-energy line breaks and moderate-energy cracks. Table 3.11(B)-2, Figures 3.11(B)-1 through 84, and Reference 24 identify the auxiliary building temperature, pressure, and humidity conditions. The auxiliary building pressure and temperature environments were developed for an MSLB in the main steam/main feed tunnel and for HELBs (Auxiliary Steam System and CVCS) in the rest of the building. The MSLB in room 1331, Turbine Building Auxiliary Feedwater Pump Room, is included in this evaluation to determine impact on adjacent areas. However, none of the equipment located in that room is required to function following the break. Accordingly, none of the equipment is qualified for the accident environment. Failure of the equipment in the room will not cause a safety concern or mislead operators. Refer to Section 3B.4.2 for details of the pressure and temperature conditions in the main steam tunnel area.

The temperature of many of the rooms does not reach the saturation temperature at the calculated pressure because of the presence of large quantities of air. The saturation condition is always in reference to the steam partial pressure. Unless the room has all of its air purged out, the total pressure of the room has a large component due to the partial pressure of air. The room could very well be saturated at the steam partial pressure, but since the steam partial pressure is small compared to the room total pressure, the saturation temperature is very low. Thus, a room with a total pressure of 14.7 psia will not have a temperature of 212°F or greater unless all the air has been purged out of the room.

The auxiliary building, except for the main steam tunnel area, does not have dedicated blowout panels for venting steam to the outside atmopshere following a HELB. The normal HVAC exhaust ducts were utilized as an exhaust path in the pressure-temperature model. Fire damper closure at the specified set point was also included in the model. Even though no makeup air was assumed to enter the auxiliary building, the results of the various calculations indicate that a significant portion of the original air remains inside the building.

Flooding

The effects of flooding were considered in the NUREG-0588 review. The flood levels are identified in Table 3.11(B)-6. The identified flood levels for the auxiliary building were not developed solely for the purpose of the NUREG-0588 review. As a result, some flood levels are generated by breaks that are not assumed to happen concurrent with an MSLB or LOCA. However, each piece of equipment that was identified as being submerged was evaluated individually to determine if submerged operation for the particular accident was required for plant safety.

3.11(B).1.3 Voltage and Frequency

The normal (and post-accident) voltage and frequency limits for Class 1E equipment are:

NOMINAL SYSTEM/RATED VOLTAGE	ACCEPTABLE OPERATING RANGE
4.16/4.0 kV	3600-4400 V
480/460 V	414-506 V
125/- V dc	90-140 V
120/115 V ac	108-132 V
Frequency: 60 Hz	58.8 - 61.2 Hz

The voltage variations for the ac system are either operational variations which are to be expected from the offsite power sources or variations from the diesel generator upon loss of offsite power. The variations have been accounted for in the qualification of safety-related equipment.

The dc voltages at the battery can vary between 105 and 140 volts. This range was established by determining the minimum discharged voltage of the station batteries (105 V dc) and the maximum output voltage of the battery chargers (140 V dc). Due to cable voltage drop, the minimum voltage at each device may be a minimum of 90 volts. Since fully discharged battery output at the component and maximum battery charger output are the two bounding conditions for the dc system, the established voltage range is the maximum dc variation to be experienced by safety-related equipment.

Transient conditions are not included in the above described parameters. Where transient conditions apply (e.g., LOCA sequencing of large loads) these criteria are addressed in the design specifications. These transients are discussed in Section 8.3; however, they are considered outside the scope of the NUREG-0588 review.

The specified frequency band is 60 Hz plus 2 percent minus 2 percent. This band is compatible with the design of the components which are powered by the diesel generator. Since the diesel generator frequencies are controlled and the offsite power grid has small, even frequency fluctuations, the safety-related equipment will not experience any higher frequency excursions.

3.11(B).1.4 Environmental Design Criteria

Compatibility of equipment with the specified environmental conditions is provided to fulfill the following design criteria:

a. For normal operation, systems and components required to mitigate the consequences of a DBA or to provide for hot or cold shutdown from the control room are designed to remain functional after exposure to the environmental conditions in Table 3.11(B)-1.

   Where possible, all safety-related systems and components are designed to withstand the maximum expected 40-year integrated radiation dose at their respective locations within the plant. If it cannot be assured that equipment is designed for the 40-year dose, a replacement program for that equipment is established. The replacement program ensures operational integrity of the equipment throughout the life of the plant.

b. In addition to the normal operation environmental requirements given in a. above, systems and components required to mitigate the consequences of a DBA or to provide for hot or cold shutdown of the reactor are designed to remain functional after exposure to the following environmental conditions. Qualification time is based on the operating duration following a DBA and any potential consequences of component failure after its function has been completed.

   1. Such components inside the containment are designed for the temperature, pressure, submergence, humidity, and chemical spray environment inside the containment after a design basis LOCA or main steam line break accident.

   2. Such components inside the containment which are required after a LOCA are designed for the post-LOCA radiation dose to which the equipment is exposed.

3. Such components outside the containment which are required to mitigate the consequences of a design basis LOCA are designed for the expected integrated accident radiation dose at the equipment location.

4. Such components outside the containment are designed for the temperature, pressure, submergence, and humidity environmental conditions summarized in Tables 3.11(B)-2 and Table 3.11(B)-6. These conditions consider high-energy line breaks outside of the containment where such breaks affect systems or equipment necessary to mitigate the consequences of the break or are required for safe shutdown of the plant following that break.

The engineered safety features and other safety-related equipment which must remain operable during and after a DBA are further discussed in the following FSAR chapters:

a. Mechanical equipment in Chapters 6.0, 9.0, and 10.0.

b. Class 1E electrical equipment in Chapter 8.0.

c. Instrumentation and controls in Chapter 7.0.

The quality assurance program for this equipment is outlined in Chapter 17.0 of the FSAR.

3.11(B).2 QUALIFICATION TESTS AND ANALYSES

Qualification is generally based on environmental testing. Qualification consists of a simulation of actual physical conditions on an actual component or prototype, analyses, or a combination of tests and analyses, as applicable. The testing period is sufficient to ensure the capability to function during and for the required interval after a DBA. For example, the containment coolers are qualified to operate for 6 months in a post-LOCA environment, which is 179.5 days greater than the expected service requirement following a LOCA. Qualification tests are performed by recognized testing agencies which use recognized standards, as applicable.

Seismic qualification is discussed in Sections 3.10(B) and (N). Additionally, assurance that damaging vibration effects do not occur in service is provided by the preoperational tests and inspections as well as by the periodic on-line testing performed in accordance with the Technical Specifications.

3.11(B).2.1 Equipment Inside Containment

As stated in Section 3.11(B).1, the equipment listed in Table 3.11(B)-3 is designed for 40 years of operation in the environment that exists at the equipment location during normal operation. In cases where a 40-year life under such conditions is not within the

state-of-the-art, a replacement program is established to ensure continuous, reliable operation. Furthermore, the equipment is designed to remain functional in the environment that exists at the equipment location at the time it is required to perform after a design basis loss-of-coolant or main steam line break accident.

Other IEEE standards and qualification criteria are used in conjunction with IEEE 323-74 to qualify certain equipment. These are discussed below:

a. Continuous-duty motors used inside the containment are type tested under simulated LOCA conditions. IEEE 334-1974, "Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations," is used. Insofar as practicable, auxiliary equipment which is part of the installed motor assembly is likewise qualified in accordance with IEEE 334, under simulated design basis event conditions.

b. Motor-operated valves used inside the containment are type tested in accordance with IEEE 382-1972 (ANSI N41.6), "Trial-Use Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations." (Also see Regulatory Guide 1.73.)

c. Type tests for each type of cable to assure acceptability for use in the containment post-accident environment are performed in accordance with IEEE 383-1974, "Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations."

d. Electrical containment penetrations are tested in accordance with IEEE 317-1972, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

e. Pressure boundary components inside the containment are designed for the temperature, pressure, and humidity environments in accordance with the applicable code to which the component is constructed. Appropriate pressure boundary components are included in the mechanical equipment qualification program discussed in 3.11(B).6; qualification testing is not necessary for such components.

f. A total (normal plus accident) integrated dose of less than 10^4 rads will not hamper the strength or properties of most materials used (Ref. 2). Hence, further environmental qualification analyses and tests for such components which will be exposed to less than 10^4 rads are not necessary. For higher integrated doses, components are qualified either by qualification testing or by evaluating the materials used for the dose involved, using reliable accumulated data on radiation effects, as contained in References 2 and 4. The effects of accident doses greater than 10^3 rads were evaluated, as appropriate (e.g., for solid-state devices).

g. Pressure boundary and structural components inside the containment are qualified for chemical spray by using components of known compatibility with the containment spray solution. Aluminum and zinc are not used as pressure boundary or structural materials. Cupronickel is used as a pressure boundary material only in the containment fan cooler coils, and its corrosion rate in the spray solution is acceptably low (Ref. 3). Gaskets, when used in piping systems, are flexitallic or equivalent with metal windings and filler material compatible with the spray solution. Gasket materials on the fuel transfer tube and on containment equipment and personnel hatches are selected to be compatible with the spray solution. Other pressure boundary and structural materials used are stainless and carbon steel and concrete, which do not suffer significant degradation in the spray environment (Ref. 3).

Equipment environmental qualification tests and analyses are responsive to Regulatory Guides 1.30, 1.40, 1.63, 1.73, 1.89, and 1.131, as described in Appendix 3A.

3.11(B).2.2 Auxiliary and Fuel Building Equipment

Safety-related equipment located in the auxiliary and fuel buildings are normally exposed to ambient temperatures up to 104°F during the summer and down to 60°F during the winter months (except for the extremes between 50°F and 123.8°F identified in Table 3.11(B)-1). Notes 4, 6, and 7 of Table 3.11(B)-1 discuss the effects of pump operation on room temperatures. Normal operating radiation environments are provided in Table 3.11(B)-1.

The design environmental conditions for DBAs, including cumulative radiation exposure, are given in Table 3.11(B)-2.

The temperature for the auxiliary building is maintained by Class 1E and non-Class 1E ventilation systems. On a loss of normal ventilation affecting rooms serviced by non-Class 1E ventilation (except as discussed in Notes 2 and 7 of Table 3.11(B)-1), the ambient temperature of the auxiliary building rooms and corridors will not normally exceed 120°F. This is primarily due to the lack of heat sources in these areas without power available. As a result, a temperature of 120°F for these areas is considered the "anticipated abnormal" condition. This temperature may be exceeded when the Class 1E coolers are out of service. The temperatures of the RHR heat exchanger rooms, turbine-driven auxiliary feedwater pump room, and the main steam / main feedwater isolation valve rooms rise to the values listed in Table 3.11(B)-1 after a loss of normal ventilation. The duration of a loss of normal ventilation is considered short and, accordingly, the temperatures generated by this condition were not utilized in aging calculations in the NUREG-0588 review.

In the event of a fuel-handling accident, equipment in the fuel building, such as the ventilation system, would not be exposed to radiation levels higher than 1 x 10^4 rads. These levels are well below the damage threshold of the ventilation equipment.

All safety-related equipment is designed to withstand the previously stated environmental conditions as required to perform its safety function. Qualification documentation based on equipment type testing and/or analyses demonstrate that this equipment operates satisfactorily under the specified environmental conditions as required to perform its safety function.

3.11(B).2.3 Control Building Equipment

3.11(B).2.3.1 Control Room

Normally, the temperature and humidity in the control room are maintained at less than 80°F and approximately 50 percent, respectively. In the event of a failure of the control building normal heating, ventilating, and air-conditioning system, the control room air-conditioning system provides the cooling, filtration, and ventilation required to maintain habitability of the control room and the integrity of the control room equipment.

The safety-related control room equipment supplied by Westinghouse is qualified to operate in an environment up to 120°F with no degradation in performance. The remainder of the safety-related (i.e., safety-related protection, not control systems) equipment in the control room is qualified to operate in an environment up to 104°F with no degradation of performance. The margin between the maximum temperature which will be experienced in the control room and the qualification limit assures that degradation of performance will not occur.

All safety-related equipment in the control room is designed to operate satisfactorily under these environmental conditions. Documentation of tests verify that this equipment operates satisfactorily under these environmental conditions.

3.11(B).2.3.2 Class 1E Electrical Equipment Rooms

The air-conditioning systems installed for these areas are designed to maintain the room temperature at or below 90°F under all operating conditions when the outdoor air is at summer design conditions.

All safety-related equipment in the Class 1E electrical equipment rooms is designed to sustain the specified environment conditions. Documentation of tests and/or analyses confirm that this equipment operates satisfactorily under the specified environmental conditions.

3.11(B).2.4 Essential Service Water Pump House

The area inside the pump house is weather protected. It is normally heated to maintain 50°F to protect against freezing during winter, and is limited to a maximum temperature of 122°F during summer. Documentation verifies that the safety-related equipment operates satisfactorily over this temperature range.

3.11(B).2.5 Equipment Located Outside of Buildings

The design summer outside air conditions used for ventilation is 97°F db and 79°F wb. This is based on applicable area weather data. These temperatures are equaled or exceeded only 2-1/2 percent of the time during the summer months (June through September).

The design winter outside air conditions used for ventilation are a temperature of -25°F and a wind velocity of 15 mph.

Engineered safety features systems, components, and structures which are exposed to the outside environment will be capable of sustaining extreme temperature conditions, precipitation, and other weather variations, including icing, without a loss of function.

3.11(B).3 QUALIFICATION TEST RESULTS

The summaries and results of the qualification tests for electrical equipment and components in the harsh environment areas listed in Table 3.11(B)-3 are maintained in an auditable form.

3.11(B).4 LOSS OF VENTILATION

Category I cooling and/or ventilation and/or filtration systems, described in Section 9.4, are powered from the preferred and the standby Class 1E electrical power supplies and are provided for the following equipment and locations:

a. Control room

b. Class 1E battery and DC switchboard rooms

c. Class 1E switchgear rooms

d. Safety injection pump rooms

e. Residual heat removal pump rooms

f. Containment spray pump rooms

g. ECCS centrifugal charging pump rooms

h. Component cooling water pump rooms

i. Essential service water pump rooms

j. Diesel generator rooms

k. Motor-driven auxiliary feedwater pump rooms

l. Containment

m. Fuel storage pool pump rooms

n. Electrical penetration rooms

Except for the control room and battery rooms temperature surveillance capability for the above locations housing safety-related equipment is available to the control room operators from the plant computer. The temperature inputs originate from temperature instrumentation within the rooms and provide high room temperature alarms via the computer.

The control room HVAC system is designed to maintain the control room at 78°F ± 6°F during the summer and greater than or equal to 60°F during the winter for all modes of plant operation (See Table 3.11(B)-1 for normal operating temperature). Redundant, seismic Category I A/C systems are provided so that a single failure cannot impair the ability of the system to cool the control room; therefore, it is not considered a credible event to lose all control room cooling. In any event, appropriate action would be taken in accordance with the plant Technical Specification 3.7.11 and FSAR Section 16.7.4 should the design temperature in the control room be exceeded.

The other seismic Category I cooling and/or ventilation systems are also designed so that the single failure of an active component after a DBA cannot impair the ability of the systems cooled by the cooling/ventilation systems to fulfill their safety functions. Should a train in a seismic Category I ventilation system become inoperative during normal operation, sufficient ventilation equipment will still be available to mitigate the consequences of a DBA.

Safety-related and reactor protection system instrumentation and cables located outside the containment and not cooled by a seismic Category I ventilation system are designed for continued operation in the event of the failure of the normal ventilation system concurrent with a loss of the preferred electrical power source.

3.11(B).5 NUREG-0588 PROGRAM REQUIREMENTS

3.11(B).5.1 Display Instrumentation

Callaway safety-related display instrumentation is listed on Table 7.5-1. Safety-related instrument sensors located in harsh environments were included in the NUREG-0588 review. Instrument sensors and readout devices not in harsh environments were excluded from the NUREG-0588 review, but are included in Table 3.11(B)-3.

Union Electric has responded to Regulatory Guide 1.97 "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions

During and Following an Accident." The response has been included in Appendix 7A. All Category I instruments are included in the NUREG-0588 program.

3.11(B).5.2 Equipment Operability

For the NUREG-0588 review, a post-DBA maximum operability requirement of 6 months (180 days) was utilized. Equipment was evaluated against this period for operability unless a shorter operability duration was justified. This value was selected as a conservative bounding time for termination of accident effects within the containment. The containment pressure-temperature analysis, as reflected in Figures 3.11(B)-3 and 6, indicates that containment conditions return to normal or below normal operating conditions within 30 days. It should also be noted that Regulatory Guide 1.4 provides criteria for evaluating the offsite radiological consequences of a LOCA event for a maximum of 30 days following the accident.

Margins of 1 hour or more for equipment with required operability times of less than 10 hours have generally been used for the Callaway equipment qualification review. However, margins of less than 1 hour have been used when adequate technical justification could be provided. Union Electric concurs with the AIF position on the 1-hour time margin, as stated in a letter to Mr. Harold Denton dated January 4, 1982, in that an arbitrary time margin of 1 hour appears inappropriate and should not be required when adequate technical justification for a shorter period exists.

3.11(B).5.3 Margins

The discussions in Section 3.11(B).1 show that post-accident environmental parameters were conservatively and uniquely determined using plant-specific data. Hence, the guideline generic techniques discussed in NUREG-0588 are not applicable.

The values for margin identified in Section 6.3.1.5 of IEEE-323-1974 were used as acceptance criteria during the NUREG-0588 review. The only regular exception to the IEEE-323-1974 margins was for radiation. As identified in Item 1.4 of NUREG-0588, additional margin need not be added to the radiation parameters if the methods identified in Appendix D of NUREG-0588 are utilized. The methods used to determine the Callaway radiation parameters are consistent with the Appendix D methodology. Hence, the radiation margins required by Section 6.3.1.5 of IEEE-323-1974 were not necessary.

3.11(B).5.4 Aging

During the NUREG-0588 review, two general observations were made concerning equipment aging:

1. Some IEEE-323-1974 equipment underwent accelerated thermal aging based on the Arrhenius method. This approach was considered acceptable.

2. Some IEEE-323-1974 equipment underwent accelerated thermal aging based on the 10 C rule. The review of this approach consisted of a check for vendor comparison to the Arrhenius method or performance of a confirmatory calculation.

3.11(B).5.5 Exemption From Qualification

The equipment identified in Table 3.11(B)-3 is provided with a category for each of three accidents as described in Appendix E of NUREG-0588.

Equipment was reviewed on a specification basis. If all of the equipment associated with a given specification was located in a mild environment (Category D) for all three accidents, then the package was classified as mild and processed as identified in Section 3.11(B).5.7. It should be noted that since the equipment was reviewed by specification some equipment located in mild environments (but part of a specification with equipment located in a harsh environment) was reviewed to the harsh environment criteria. However, the qualification contingencies identified in the harsh environment review are not applicable to Category D equipment.

As defined in NUREG-0588, Category C equipment need not be qualified for any accident environment. Category C equipment need only be qualified to its non-accident environment. Therefore, qualification contingencies identified in the harsh environment review are not applicable to Category C equipment. This equipment can be treated in the same way as mild environment equipment, as discussed in Section 3.11(B).5.7.

If the only components in a harsh environment for a given package are Category C, the entire specification is then treated as a mild package and processed as identified in Section 3.11(B).5.7. The justification for the C categorization for these pieces of equipment is provided in an auditable form in the equipment files and is summarized in Table 3.11(B)-8.

Equipment that performs its function before its exposure to the harsh environment may also be exempted. This exemption is only utilized if the adequacy of the associated time margin is justified. Before exempting this category of equipment, a review was performed to verify that subsequent failure of the equipment as a result of the harsh environment does not degrade other safety functions. If specific equipment is deleted for the above reason, it is identified in the individual qualification package. If an entire specification was deleted as a result of the above, the specification is listed in Table 3.11(B)-8. For further discussion, see the position on Regulatory Guide 1.89 in Appendix 3A.

3.11(B).5.6 Maintenance and Surveillance Activities

The Callaway maintenance program provides for control, testing, failure evaluation, trending, and programmed replacement of environmentally qualified safety-related electrical equipment.

The procurement and material control program provides for the controlled procurement of Class 1E parts and components to ensure that appropriate qualification and technical requirements are identified and reviewed by engineering disciplines. Material Engineering is involved in specifying the technical and quality requirements on procurement documents. Nuclear Oversight (NOS) Supplier Quality is involved on the front-end of the procurement process by assessing and evaluating suppliers' capabilities to provide the desired items or services and through maintenance of the listing of qualified suppliers. Qualification of suppliers is assured through independent NOS audits performed to verify that part or component procurement requirements are met and documented. The program assumes controlled storage, handling, and issuing of parts or components and identification of shelf life and maintenance requirements while the parts or components are in storage.

Inspection, testing, and replacement requirements identified as a result of the qualification review are incorporated in the preventive maintenance and calibration procedures. Vendor technical manual recommendations are reviewed; and if additional testing, inspection, or replacement recommendations are identified, they are incorporated, as appropriate.

Results of these tests, inspections, or replacement activities are routed for engineering review when they do not conform to defined acceptance criteria. As new requirements are identified through the engineering evaluation, procurement, equipment operational history, or changes to regulatory requirements, they are factored into the program.

Maintenance performed as a result of part or component failure will be reviewed by maintenance groups and engineering to categorize the cause of failure. Failures which occur as a result of environmental application, including aging, will be evaluated to determine what, if any, preventive maintenance action may be taken to protect from further failures. Examples of the evaluation methods to be used are:

- An onsite program of review to categorize cause of failures and establish a data base for trending purposes.

- Participation in industry-wide data gathering programs such as NPRDS for purposes of identifying generic or common mode failures.

- Utilization of the LER program to provide additional information relating to reoccurring failures throughout the industry.

Results of those evaluations will be factored into the preventive maintenance program. Equipment upgrade requirements resulting from these evaluations shall be factored into procurement documents through the design change process.

Maintenance personnel will receive training to assure their awareness of specific requirements relating to inspection, cleaning, testing, and replacement of Class 1E environmentally qualified equipment. Training will include requirements for verifying

equivalency of replacement parts or components through part number comparison and physical comparison. These requirements will ensure that replacement parts and components are installed in the correct physical configuration in the system or parent component, and that appropriate supervisory and engineering personnel are notified where initial investigation shows the cause of failure to be environmentally induced or where inspection or test results are not within acceptance limits.

It should be noted that the maintenance and surveillance activities identified in this section apply to all Class 1E equipment (i.e., equipment in a harsh or mild environment).

Specific maintenance surveillance requirements are provided in the response to Question 270.14 for the following:

a. Cables located inside containment

b. Limitorque valve operators

c. Amphenol electrical penetrations

d. Motor control center relays and breakers

e. Barton pressure transmitters

3.11(B).5.7 Equipment Located In Mild Environments

Each room of the auxiliary building was evaluated to determine if it had a mild environment for each of the three accidents.

An environment was considered mild if it did not exceed its anticipated abnormal condition or, as a result of the accident, the room environment remained below all of the following parameters:

Temperature	$\leq$ 110°F	
Pressure	$\leq$ 16.1 psia	
Radiation	$\leq 10^3$ rads	(Unless an engineering evaluation is performed to support 10^4 rads)
Humidity	$\leq$ 90%	

Equipment located in mild environments (as defined above) were not included in the NUREG-0588 review program.

The qualified lives established during the NUREG-0588 review program for equipment located in a harsh environment are not applicable to equipment located in a mild

environment. For further discussion, see the position on Regulatory Guide 1.89 in Appendix 3A.

3.11(B).5.8 Synergistic Effects

Present synergistic effect information is minimal and not conclusive. The Callaway equipment qualification effort did consider synergisms as identified below.

a. If the vendor identified a synergistic effect, it was evaluated.

b. If the reviewer was aware of a synergistic effect, it was evaluated. As additional synergistic effect data became available it was evaluated and factored into the program.

c. If neither a. nor b. existed, then no further actions were taken to determine if any synergistic effects were known (e.g., a literature search).

It should be noted that NUREG/CR-2157, Occurrence and Implications of Radiation Dose - Rate Effects for Material Aging Studies, and NUREG/CR-2156, Radiation - Thermal Degradation of PE and PVC: Mechanism of Synergism and Dose Rate Effects, were considered along with other information on synergisms. It was concluded that NUREG/CR-2157 was applicable to the Callaway cable, but considering the margins applied to the Callaway cable relative to radiation (typically >200 percent) that the margin compensated for potential synergistic effects of radiation application rate. Also, NUREG/CR-0276 indicates that, for PE and cross-linked polyolefin, dose rate effects are negligible. NUREG/CR-2156 addresses PE and PVC cable. Callaway PE cable is cross-linked and, accordingly, the relevance of the study is questionable. However, Callaway did evaluate this synergistic effect. The Callaway cable was tested in accordance with IEEE-383-74, in that the thermal pre-aging was performed prior to the radiation dose application. The DBA test was then performed. This sequence is consistent with the actual events that will occur during the plant life (assuming a LOCA at the end of life). The thermal aging independent of radiation is consistent with the actual plant condition due to the low radiation exposure that the cable receives. Prior to the elevated temperatures of the DBA, the radiation dose is applied. This sequence is consistent with the Sandia report which states, "The joint effect of gamma radiation and elevated temperature was also found to occur when the two environments were applied in a sequential fashion, but only when the experiments were performed in the order: radiation at room temperature followed by elevated temperature." It should also be noted that the radiation applied is >200% of that required to simulate the accident conditions. This sequence and margin indicate that the synergistic concerns are adequately addressed. Additionally, the Sandia report uses percent elongation as the criterion to failure. This criterion appears to be inappropriate relative to actual plant requirements. The cable is securely placed in cable trays or conduit. The real concern is the insulating capability. This is not addressed by the report. The Callaway cable was typically meggered through the test sequence and at the end of the test the cable was wound around a mandrel, submerged in water, and a voltage withstand test was

performed. This method of test evaluation is more severe and relevant than the Sandia evaluation method (i.e., percent elongation).

Additionally, as Section 3.11(B).5.6 indicates, maintenance performed as a result of component failure will be reviewed by maintenance and engineering to categorize the cause of failure. This program can provide early indication of premature deterioration which could be the result of unexpected synergistic effects.

In addition to the failure evaluation program, the surveillance and maintenance program for the safety-related motors includes a provision for the megger of insulation in accordance with manufacturer's recommendations, typically every 18 to 24 months. This testing is planned to be performed for the motors from the associated motor control center or switchgear. Thus, the cables and electrical penetration assemblies will be meggered with the motor windings. This surveillance is capable of detecting insulation degradation and the location of degradation can be traced to determine which component is at fault. Callaway has established a periodic inspection program to monitor in-service aging of electrical cable insulation on selected cables inside containment.

It is also the intent of Union Electric to stay abreast of information on synergistic effects as it becomes available. Union Electric is a member of EPRI and receives information from this source as well as other industry sources and the NRC.

3.11(B).6 MECHANICAL EQUIPMENT QUALIFICATION

The mechanical equipment qualification effort began at the start of the Callaway engineering effort. The seismic and environmental requirements for the various safety-related mechanical components were identified in each purchase specification. Each vendor was requested to supply equipment that could withstand the specified environments. The vendor submittals were reviewed to ensure conformance.

To provide additional verification of mechanical equipment qualification an additional review program was implemented.

The Callaway program for the review of environmental qualification of safety-related mechanical equipment involved a four-step process:

1. Identification of all safety-related mechanical equipment.

2. Categorization of equipment in accordance with NUREG-0588, Appendix E, based on equipment location and function.

3. Verification of qualification for active mechanical equipment in harsh environment areas.

4. Identification of aging concerns and establishment of replacement intervals as required.

Table 3.11(B)-3 identifies safety-related mechanical equipment and provides the room number in which the equipment is located and the equipment category based on the definitions in NUREG-0588, Appendix E.

The next step in the review process was the verification of mechanical equipment environmental qualification for the subset of active mechanical equipment located in harsh environment areas. This step involved a review to determine if the equipment had been previously qualified because, in some cases, mechanical equipment was aged and tested together with associated electrical equipment for IEEE-323-1974. If the equipment had not been qualified by test or analysis under a previous qualification program, then a detailed review of the equipment was performed to identify components which could be adversely affected by post-accident environmental conditions or could be subject to significant aging mechanisms. The review concentrated on the components that are subject to deterioration in these environmental conditions (normal and/or post-accident) because they are "soft," non-metallic components such as seals, gaskets, diaphragms, packing, etc. Identified components which could adversely affect the safety function of the equipment were then evaluated on the basis of material performance data or failure modes and effects analysis to verify that the equipment is qualified for its intended use.

As part of the qualification review, replacement intervals were identified either on the basis of aging performed during an IEEE-323-1974 qualification program or on the basis of published material aging data. It should be noted that, because all harsh environment Callaway Class 1E equipment has been reviewed under the NUREG-0588 program, and all harsh environment safety-related mechanical equipment has been evaluated under the program described above, concerns regarding the effect of aging on seismic performance of all safety-related equipment located in harsh environment areas have been adequately addressed for Callaway.

3.11(B).7 CONTROL SYSTEMS QUALIFICATION (IE INFORMATION NOTICE 79-22)

Callaway reported on the matters addressed in IE information Notice 79-22 to the NRC in References 14 and 15. These reports were submitted to NRC Inspection and Enforcement pursuant to 10 CFR 50.55(e). The latter report stated that final resolution would be provided in revisions to the Callaway FSAR. The resolution and/or current status is provided below.

Westinghouse identified four control systems for generic consideration of nonsafety grade/safety grade interface interactions.

a. Steam generator power-operated relief valve control system - A piping failure in the vicinity of the steam generator relief valves could be assumed to cause the valves to stick open. The combination of the pipe failure, an

assumed single failure, and the stuck open valve(s) may result in inadequate auxiliary feedwater flow.

Westinghouse performed generic analyses for this type of event during development of the Emergency Response Guidelines (ERGs). ERG ECA 2.1, "Uncontrolled Depressurization of All Steam Generators" considers a worst-case multi-steam generator depressurization. The Westinghouse analysis, which bounds the relief valve opening event, concluded that a stabilized plant and safe cooldown can be achieved with a flow equivalent to one motor-driven auxiliary feedwater pump. Therefore, this scenario does not present a safety problem for the Callaway design.

b. Pressurizer power-operated relief valves control system - The Callaway pressurizer, PORV associated pressure transmitters, and automatic actuation circuitry meet Class 1E requirements and are qualified to the postulated accident environments inside the containment. Therefore, this scenario does not present a safety problem for the Callaway design.

c. Main feedwater control system - A small feedwater line break could affect normal feedwater flow control, causing low steam generator levels prior to protective actions for the break.

   The Callaway feedwater line break accident has been reanalyzed, assuming the control and protection grade system interaction. The analysis shows that this scenario can be accommodated without violating design conditions and acceptance criteria. A summary of the analysis may be found in Section 15.2.8. The summary includes an identification of the analysis assumptions that are different from those used in Reference 18.

d. Automatic rod control system - This control/protection system interaction is no longer applicable to Callaway. An intermediate size high-energy line break is assumed to affect the rod control system, such that the initial conditions previously assumed for the break may not be valid.

   During the NRC review of the Callaway design, the commitment was made to perform a Callaway specific evaluation of the effects of a steam line break in the vicinity of the main turbine impulse pressure transmitters. A steam line rupture outside the containment is assumed to cause an adverse environment for the turbine impulse pressure transmitters, causing the control rods to begin withdrawal prior to receipt of a reactor trip signal. This evaluation was submitted to the NRC by Reference 16, and revised by Reference 17. The evaluation concluded that the consequences of the postulated event are bounded by previous accident analyses described in the Callaway FSAR. The results of this evaluation were used in the review of the effects of postulated high-energy line breaks on the power range

excore detectors and associated in-containment equipment. Based on the review, the following was determined:

1. The steamline rupture is the limiting event to be considered with respect to a consequential control rod withdrawal.

2. The results of the above mentioned evaluation of the steamline break outside containment apply to the postulated inside containment steamline break with coincident control rod withdrawal.

As noted above, the consequences of the postulated event are bounded by previous accident analyses described in the Callaway FSAR. As a result of this review, it is concluded that the power range ex-core detectors and associated equipment are not required to be qualified for postulated high energy line break environments inside containment. The automatic rod withdrawal function has been removed, thus eliminating the possibility of this transient.

For additional evaluation of control grade system failures, refer to the response to Question 420.4.

3.11(B).8 REFERENCES

1. DiNunno, J. J., Baker, R. E., Anderson, F. D., and Waterfield, R. L., "Calculation of Distance Factors for Power and Test Reactor Sites," Technical Information Document 14844, Division of Licensing and Regulation, AEC, Washington, D.C., 1962.

2. Kircher, J. F. and Bowman, R. E., Effects of Radiation on Materials and Components, Van Nostrand Reinhold, New York, 1964.

3. Griess, J. C. and Bacarella, A. L., "Design Considerations of Reactor Containment Spray Systems-Part III, The Corrosion of Materials in Spray Solutions," ORNL-TM-2412, Part III, Oak Ridge National Laboratory, Oak Ridge, Tennessee, December, 1969.

4. EPRI Report NP-2129, "Radiation Effects on Organic Materials in Nuclear Plants," November 1981.

5. SLNRC 83-0015, "Report of Independent Review of Environmental Qualification Programs to NUREG-0588," March 10, 1983 and revisions per SLNRC 83-0030, May 27, 1983; SLNRC 84-0013, February 1, 1984; and SLNRC 86-02, January 17, 1986.

6. T. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., New York (1956).

7. M. A. Capo, Polynominal Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source, APEX 510.

8. R. G. Jaeger, et al, Engineering Compendium on Radiation Shielding, Shielding Fundamentals and Methods, I (1968).

9. QAD-CG, A Combinatorial Geometry Version of QAD-P5A (Bechtel Code).

10. J. Nilsson and R. Sandlin, "Straight Ducts," Engineering Compendium on Radiation Shielding, Shielding Fundamentals and Methods, I (1968).

11. J. H. Hubbell, R. L. Bach, and R. J. Herbold, "Radiation Field from a Circular Disc Source," Journal of Research of the National Bureau of Standards--C Engineering and Instrumentation, Vol. 65C, No. 4, October-December 1961.

12. A. V. Masket, "Solid Angle Contour Integrals, Series and Tables," Review of Scientific Instruments, Vol. 28, No. 3, 191-197, March 1957.

13. R. Loevinger, "Discrete Radioisotope Sources," Radiation Dosimetry, Academic Press.

14. SLNRC 79-15, "Qualification of Control Systems," September 28, 1979.

15. SLNRC 80-6, "Qualification of Control Systems," February 5, 1980.

16. SLNRC 83-005, "Qualification of Control Systems," February 2, 1983.

17. SLNRC 83-054, "Instrumentation and Control Systems Branch Review," October 27, 1983.

18. WCAP-9230, Rev. 0, "Report on the Consequences of a Postulated Main Feedline Rupture, Proprietary."

19. ULNRC-1471 "Callaway Plant Uprating Submittal", March 31, 1987.

20. ULNRC-1618 "Responses to Questions on Callaway Uprating", September 18, 1987.

21. NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Part II, Appendix D, page IID-4, July 1981.

22. E. C. Beahm, W. E. Shockley, C. F. Weber, S. J. Wisbey, and Y.M. Wang, "Chemistry and Transport of Iodine in Containment," NUREG/CR-4697, October 1986.

23. NUREG-0800, Standard Review Plan Section 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System," December 1988.

24. Calculation M-YY-49, Pressure/Temperature (P/T) Profiles for each room in the Aux. Building resulting from HELB, and its associated addenda.

25. Geroge, TL, et al., GOTHIC Containment Analysis Package User Manual, Version 7.2b (QA), NAI 8907-02, Rev. 18, Numerical Applications, Inc., Richland, WA, March 2009.

TABLE 3.11(B)-1 PLANT ENVIRONMENTAL NORMAL CONDITIONS[1]

Room No.	Area	Normal Operating Pressure (psig)	Normal Oper. Temp. Max/Min	Relative Humidity Max/Min (%)	Oper. Dose Rate (R/hr)	Norm. Oper. Integ. Dose (Rad)	Environmental pH Normal
Reactor Building							
2000 Series	Operating floor (above)	± 2	120/50	100/50	0.01-0.4	3.5×10^3	7
	Steam generator loop compartment	± 2	120/50	100/50	1.4-20	6×10^6	7
	Reactor cavity	± 2	150/100	100/50	10^5	2.8×10^{10}	7
	Reactor Cavity Seal Ring Support*	± 2	300/100	100/50	10^5	2.8×10^{10}	7
	Outside S/G loop compartment	± 2	120/50	100/50	0.005-0.4	3.5×10^3	7
Auxiliary Building[2,3]							
1101	General floor area No. 1	Atmospheric	104/60	70/5	<0.0005	<200	7
1102	Chiller and surge tank area	Atmospheric	104/60	70/5	<0.0005	<200	7
1107	ECCS charging pump room	Atmospheric	104/60[4]	70/5	0.005-1	3.5×10^5	7
1108	Safety inj. pump room	Atmospheric	104/60[4]	70/5	0.005-0.015	3.5×10^3	7
1109	RHR pump room	Atmospheric	104/60[4]	70/5	0.015-1.2	3.5×10^5	7
1110	Ctmt. spray pump room	Atmospheric	104/60[4]	70/5	0.005-0.015	3.5×10^3	7
1111	RHR pump room	Atmospheric	104/60[4]	70/5	0.015-1.2	3.5×10^5	7
1112	Ctmt. spray pump room	Atmospheric	104/60[4]	70/5	0.005-0.015	3.5×10^3	7
1113	Safety inj. pump room	Atmospheric	104/65[4]	70/5	0.005-0.015	3.5×10^3	7
1114	ECCS charging pump room	Atmospheric	104/60[4]	70/5	0.005-1	3.5×10^5	7
1115	Normal charging pump room	Atmospheric	104/60[4]	70/5	0.005-1	3.5×10^5	7
1116	Boric acid tank room (B)	Atmospheric	104/75	70/5	0.005-0.015	3.5×10^3	7
1117	Boric acid tank room (A)	Atmospheric	104/75	70/5	0.005-0.015	3.5×10^3	7
1119	Stairwell	Atmospheric	104/60	70/5	<0.0008	<350	7
1120	General floor area	Atmospheric	104/60	70/5	<0.001	<350	7

TABLE 3.11(B)-1 (Sheet 2)

Room No.	Area	Normal Operating Pressure (psig)	Normal Oper. Temp. Max/Min	Relative Humidity Max/Min (%)	Oper. Dose Rate (R/hr)	Norm. Oper. Integ. Dose (Rad)	Environmental pH Normal
Auxiliary Building (2,3)							
1121	Access pit	Atmospheric	104/60	70/5	<0.0005	<200	7
1122	General floor area No. 3 El. 1974'	Atmospheric	104/60	70/5	<0.0005	<200	7
1126	Boron inj. room	Atmospheric	104/60	70/5	<0.0005	<200	7
1127	Stair	Atmospheric	104/60	70/5	<0.005	<250	7
1128	General area No. 5 Elev. 1974'	Atmospheric	104/60	70/5	<0.0005	<200	7
1129	Aux. steam condenser recovery & storage tank room	Atmospheric	104/60	70/5	<0.0005	<200	7
1202	Access area and chiller surge tank area	Atmospheric	96/70	70/5	<0.015	<600	7
1203	Pipe space (B)	Atmospheric	104/75	70/5	0.015-1.2	3.5×10^5	7
1204	Pipe space (A)	Atmospheric	104/75	70/5	0.015-1.2	3.5×10^5	7
1207	Pipe chase	Atmospheric	104/60	70/5	<0.0005	<200	7
1301	Corridor No. 1	Atmospheric	104/60	70/5	<0.0005	<200	7
1302	Filter compartments	Atmospheric	104/75	70/5	0.015-1.2	3.5×10^5	7
1304	Aux. feedwater pipe chase	Atmospheric	104/50	70/5	<0.0005	<200	7
1305	Aux. feedwater pipe chase	Atmospheric	104/50	70/5	<0.0005	<200	7
1306	Valve compartment	Atmospheric	104/75	70/5	0.015-1.2	3.5×10^5	7
1307	Corridor	Atmospheric	104/75	70/5	0.015-1.2	3.5×10^5	7
1308	Valve compartment, El. 2000'	Atmospheric	104/75	70/5	0.015-1.2	3.5×10^5	7
1309	RHR heat exch. room	Atmospheric	104/60	70/5	0.015-1.2	3.5×10^5	7
1310	RHR heat exch. room	Atmospheric	104/60	70/5	0.015-1.2	3.5×10^5	7
1314	Load center area, El. 2000'	Atmospheric	104/60	70/5	<0.0005	<200	7
1315	Ctmt. spray additive tank area	Atmospheric	104/60	70/5	0.005-0.015	3.5×10^3	7
1316	Valve compartment	Atmospheric	104/60	70/5	<0.005	<250	7

TABLE 3.11(B)-1 (Sheet 3)

Room No.	Area	Normal Operating Pressure (psig)	Normal Oper. Temp. Max/Min	Relative Humidity Max/Min (%)	Oper. Dose Rate (R/hr)	Norm. Oper. Integ. Dose (Rad)	Environmental pH Normal
Auxiliary Building (2,3)							
1317	Seal water heat exch. room	Atmospheric	104/60	70/5	0.9-7.9	3.0×10^6	7
1318	Valve compartment	Atmospheric	104/65	70/5	0.015-1.2	3.5×10^5	7
1320	Corridor No. 4, El. 2000'	Atmospheric	104/60	70/5	<0.0005	<200	7
1321	Exit vestibule	Atmospheric	104/60	70/5	<0.0005	<200	7
1322	Piping penetration room	Atmospheric	104/60	70/5	0.01-4	$<2.0 \times 10^6$	7
1323	Piping penetration room	Atmospheric	104/60	70/5	0.01-4	$<2.0 \times 10^6$	7
1324	Feedwater pump valve compartment No. 1	Atmospheric	104/50	70/5	<0.0005	<200	7
1325	Auxiliary feed pump (motor) room	Atmospheric	104/50(4)	70/5	<0.0005	<200	7
1326	Auxiliary feed pump (motor) room	Atmospheric	104/50(4)	70/5	<0.0005	<200	7
1327	Feedwater pump valve compartment No. 2	Atmospheric	104/50	70/5	<0.0005	<200	7
1328	Feedwater pump valve compartment No. 3	Atmospheric	104/50	70/5	<0.0005	<200	7
1330	Feedwater pump valve compartment No. 4	Atmospheric	104/50	70/5	<0.0005	<200	7
1331	Auxiliary feed pump (turbine) room	Atmospheric	121/50(6)	70/5	<0.0005	<200	7
1401	CCW pump room	Atmospheric	104/60(7)	70/5	0.002-0.004	<1000	7
1402	Corridor No. 1, El. 2026	Atmospheric	104/60	70/5	<0.0005	<200	7
1406	CCW pump room	Atmospheric	104/60(7)	70/5	0.002-0.004	<1000	7
1408	Corridor	Atmospheric	104/60	70/5	0.002-0.004	<1000	7
1409	Electrical penetration room	Atmospheric	104/60	70/5	<0.001	<350	7
1410	Electrical penetration room	Atmospheric	104/60	70/5	<0.001	<350	7
1411	Main feedwater room No. 1	Atmospheric	120/50	70/5	<0.001	<350	7
1412	Main feedwater room No. 2	Atmospheric	120/50	70/5	<0.001	<350	7
1413	Auxiliary shutdown panel room	Atmospheric	104/60	70/5	<0.001	<350	7

TABLE 3.11(B)-1 (Sheet 4)

Room No.	Area	Normal Operating Pressure (psig)	Normal Oper. Temp. Max/Min	Relative Humidity Max/Min (%)	Oper. Dose Rate (R/hr)	Norm. Oper. Integ. Dose (Rad)	Environmental pH Normal
Auxiliary Building (2,3)							
1501	Control room a/c equip. room	Atmospheric	104/60	70/5	<0.001	<350	7
1502	CCW surge tank area (B)	Atmospheric	104/60	70/5	0.005-0.015	3.5×10^3	7
1503	CCW surge tank area (A)	Atmospheric	104/60	70/5	0.005-0.015	3.5×10^3	7
1504	Ctmt. purge exhaust & mech equip. room (B)	Atmospheric	104/60	70/5	0.005-0.015	3.5×10^3	7
1506	Ctmt. purge supply air handling unit room (A)	Atmospheric	104/60	70/5	0.005-0.015	3.5×10^3	7
1507	Personnel hatch area El. 2047'-6"	Atmospheric	104/60	70/5	<0.0005	<200	7
1508	Main steam/main feedwater isolation valve room	Atmospheric	120/50	70/5	<0.001	<350	7
1509	Main steam/main feedwater isolation valve room	Atmospheric	120/50	70/5	<0.001	<350	7
1512	Control room a/c equip. room	Atmospheric	104/60	70/5	<0.001	<350	7
1513	Control building a/c equip. room	Atmospheric	104/60	70/5	<0.001	<350	7
Control Building (2)							
3101	Pipe space and tank Area, El. 1974	Atmospheric	104/60	70/30	<0.0005	<200	7
3105	Control building cable chase	Atmospheric	104/60	70/30	<0.0005	<200	7
3106	Control building cable chase	Atmospheric	104/60	70/30	<0.0005	<200	7
3202	Controlled HP Instrument & Tool storage room	Atmospheric	104/60	70/5	<0.0005	<200	7
3211	Hall	Atmospheric	85/60	70/5	<0.0005	<200	7
3218	RWP sign-out area	Atmospheric	85/60	70/5	<0.0005	<200	7
3222	ALARA Office/Dosimetry Issue, El. 1984	Atmospheric	78/60	70/30	<0.0005	<200	7
3223	HP Work Space	Atmospheric	78/60	70/30	<0.0005	<200	7
3224	Vestibule No. 2, El. 1984	Atmospheric	85/60	70/30	<0.0005	<200	7

TABLE 3.11(B)-1 (Sheet 5)

Room No.	Area	Normal Operating Pressure (psig)	Normal Oper. Temp. Max/Min	Relative Humidity Max/Min (%)	Oper. Dose Rate (R/hr)	Norm. Oper. Integ. Dose (Rad)	Environmental pH Normal
Control Building [2]							
3229	Control building cable chase	Atmospheric	104/60	70/30	<0.0005	<200	7
3230	Control building cable chase	Atmospheric	104/60	70/30	<0.0005	<200	7
3301	ESF switchgear room	Atmospheric	90/60	70/10	<0.0005	<200	7
3302	ESF switchgear room	Atmospheric	90/60	70/10	<0.0005	<200	7
3403	Non-vital switchgear and transformer room (No. 1)	Atmospheric	90/60	70/30	<0.0005	<200	7
3404	Switchboard room (No. 4)	Atmospheric	90/60	70/30	<0.0005	<200	7
3405	Battery room	Atmospheric	90/60	70/10	<0.0005	<200	7
3407	Battery room	Atmospheric	90/60	70/10	<0.0005	<200	7
3408	Switchboard room (No. 1)	Atmospheric	90/60	70/30	<0.0005	<200	7
3409	Non-vital switchgear and transformer room (No. 2)	Atmospheric	90/60	70/30	<0.0005	<200	7
3410	Switchboard room (No. 2)	Atmospheric	90/60	70/30	<0.0005	<200	7
3411	Battery room	Atmospheric	90/60	70/10	<0.0005	<200	7
3413	Battery room	Atmospheric	90/60	70/10	<0.0005	<200	7
3414	Switchboard room (No. 3)	Atmospheric	90/60	70/30	<0.0005	<200	7
3415	HVAC equipment room	Atmospheric	90/60	70/30	<0.0005	<200	7
3416	HVAC equipment room	Atmospheric	90/60	70/30	<0.0005	<200	7
3501	Lower cable spreading room	Atmospheric	104/60	70/10	<0.0005	<200	7
3601	Main control room	Atmospheric	84/60	70/30	<0.0005	<200	7
3605	Control rm. equip. cabinet area	Atmospheric	84/60	70/30	<0.0005	<200	7
3609	SAS room	Atmospheric	84/72	70/30	<0.0005	<200	7
3613	Computer room	Atmospheric	80/72	60/40	<0.0005	<200	7
3613A	Conference Room	Atmospheric	80/72	60/40	<0.0005	<200	7

TABLE 3.11(B)-1 (Sheet 6)

Room No.	Area	Normal Operating Pressure (psig)	Normal Oper. Temp. Max/Min	Relative Humidity Max/Min (%)	Oper. Dose Rate (R/hr)	Norm. Oper. Integ. Dose (Rad)	Environmental pH Normal
Control Building (2)							
3613B	E.O. Ready room	Atmospheric	80/72	60/40	<0.0005	<200	7
3801	Upper cable spreading room	Atmospheric	104/60	70/10	<0.0005	<200	7
Turbine Building							
4401	General floor area	Atmospheric	110/60	95/5	<0.0005	<200	7
Diesel Generators (2) Building							
5000 Series	Diesel generator rooms	Atmospheric	122/60	95/5	<0.0005	<200	7
Fuel Building (2)							
6104	Fuel pool cooling pump and heat exch. room	Atmospheric	122/60	95/5	0.0025-0.4	5×10^4	7
6105	Fuel pool cooling pump and heat exch. room	Atmospheric	122/60	95/5	0.0025-0.4	5×10^4	7
6000 Series	General areas remaining	Atmospheric	110/60	95/5	<0.0025	<1000	7
ESW Pump House							
U104, U105	General Areas	Atmospheric	122/50	95/0	<0.0005	<200	7
UHS Electrical Room							
U301 U302 U304-U307	General Areas	Atmospheric	122/50	95/0	<0.0005	<200	7
Radwaste Building							
7133	Non-radioactive Tunnel Accesses	Atmospheric	104/60	95/5	<0.0005	<200	7
Other							
9101	CST valve compartment	Atmospheric	120/50	95/5	<0.0005	<200	7
9102	RWST valve compartment	Atmospheric	120/50	95/5	<0.0005	<200	7

TABLE 3.11(B)-1 (Sheet 7)

* The Reactor Cavity Seal Ring Support is defined as that portion of the reactor cavity concrete directly below the seal ring down a distance of 36 inches.

NOTES:

(1) Environmental effects of localized hazards such as pipe breaks are reviewed on a case-by-case basis for equipment qualification.

(2) With the exception of the RHR heat exchanger rooms, the auxiliary feedwater turbine-driven pump room, and the main steam/main feedwater isolation valve rooms, the ambient temperature outside of the containment in rooms and corridors which do not have ESF coolers will not exceed 120°F during loss of normal ventilation conditions, because of the lack of heat sources. The RHR heat exchanger rooms will not exceed 175°F with the heat exchangers in operation, the auxiliary feedwater turbine-driven pump room will not exceed 147.7°F with the pump idle, and the main steam/main feedwater isolation valve rooms will not exceed 166°F during a loss of normal ventilation.

3) Those areas of the system which contain 7000 to 7700 ppm boron solution are maintained at a minimum of 75°F.

(4) When the pumps operate, the room temperature will be limited to 122°F.

(5) Deleted

(6) When the pump operates, the room temperature will be limited to 123.8°F.

(7) With both the 1E room cooler and the non-1E fan coil unit operating, the temperature for rooms 1401 and 1406 will be limited to 104°F with one pump running and 122°F with both pumps running.

TABLE 3.11(B)-2 ENVIRONMENTAL QUALIFICATION PARAMETERS FOR SNUPPS NUREG-0588 REVIEW (LOCA, MSLB AND HELB)(11)(13)

Room No.	Area	DBA Pressure Max. (psig)(9)	DBA Temp. Max. F(8)(9)	DBA RH % Max. (8)(9)	DBA Dose (Rad)(14)	Environmental ph DBA
Reactor Building						
2000 Series	Operating floor	See Table 6.2.1-2	See Table 6.2.1-2	100	See Table 3.11(B)-4	11.0 max(4) (See Table 3.11(B)-5)
	Steam generator loop compartment	Same as operating floor conditions(1)				
	Outside S/G loop compartment	Same as operating floor conditions				
	Reactor cavity	Same as operating floor conditions(2)				
Auxiliary Building						
1101	General floor area No. 1	1.0	141	100	5.46×10^6	
1102	Chiller and surge tank area	1.0	141	100	7.08	
1103	Letdown chiller hx. room	2.0	216	100	1.38×10^4	
1104	Letdown reheat hx. room	2.0	216	100	1.38×10^4	
1105	Valve compartment	2.0	216	100	1.38×10^4	
1106	Moderating hx. room	2.0	216	100	1.38×10^4	
1107(16)	ECCS charging pump room	1.0	109	72	4.38×10^6	
1108(16)	Safety inj. pump room	1.0	109	72	9.46×10^6	
1109(16)	RHR pump room	1.0	109	72	1.25×10^7	
1110(16)	Ctmt. spray pump room	1.0	109	72	1.60×10^7	
1111(16)	RHR pump room	1.0	109	72	1.28×10^7	
1112(16)	Ctmt. spray pump room	1.0	109	72	1.41 x 107	
1113(16)	Safety inj. pump room	1.0	109	72	9.38×10^6	
1114(16)	ECCS charging pump room	1.0	109	72	4.38×10^6	
1115(16)	Normal charging pump room	1.0	109	73	1.38×10^7	

TABLE 3.11(B)-2 (Sheet 2)

Room No.	Area	DBA Pressure Max. (psig)(9)	DBA Temp. Max. F(8)(9)	DBA RH % Max. (8)(9)	DBA Dose (Rad)(14)	Environmental ph DBA
Auxiliary Building						
1116	Boric acid tank room (B)	1.0	109	73	2.88×10^4	
1117	Boric acid tank room (A)	1.0	109	73	2.88×10^4	
1119	Stairwell	1.0	109	70	4.39×10^3	
1120	Gen. floor area	1.0	141	100	5.05×10^5	
1121	Access pit	1.0	141	100	1.74×10^7	
1122	General floor area No. 3, El. 1974'	1.0	133	100	5.13×10^5	
1123	Passage	1.2	214	100	1.95×10^3	
1124	Valve compartment	1.2	214	100	1.65×10^3	
1125	Letdown hx. room	1.2	214	100	1.65×10^3	
1126	Boron inj. room	1.0	110	80	2.46×10^6	
1127	Stairwell	1.0	109	70	1.42×10^5	
1128	General area No. 5 El. 1974'	1.0	133	100	2.69×10^4	
1129	Aux. steam cond. recovery and storage tank room	1.0	133	100	2.69×10^4	
1130	Corridor	1.0	133	100	1.62×10^2	
1201	Vestibule	1.0	143	100	2.77×10^3	
1202	Access area and chiller surge tank area	1.0	143	100	2.77×10^3	
1203	Pipe space (B)	2.0	183	100	1.54×10^7	
1204	Pipe space (A)	2.0	117	100	1.84×10^7	
1205	Access area	1.0	133	100	1.90×10^4	
1206	Pipe chase	1.0	110	90	2.65×10^4	
1207	Pipe chase	1.0	110	90	2.65×10^4	

TABLE 3.11(B)-2 (Sheet 3)

Room No.	Area	DBA Pressure Max. (psig)(9)	DBA Temp. Max. F(8)(9)	DBA RH % Max. (8)(9)	DBA Dose (Rad)(14)	Environmental ph DBA
Auxiliary Building						
1301	Corridor No. 1(North)	1.0	180	83	3.41×10^3	
	(West)	1.0	110	83	3.41×10^3	
1302	Filter compartments	1.0	107	71	1.00×10^6	
1304	Aux. feedwater pipe chase	Atmospheric	104	70	1.10×10^3	
1305	Aux. feedwater pipe chase	Atmospheric	104	70	2.63×10^1	
1306	Valve compartment	1.0	107	71	1.31×10^6	
1307	Corridor	1.0	107	71	8.98×10^4	
1308	Valve compartment, El. 2000'	1.0	107	71	7.23×10^3	
1309(15)	RHR heat exch. room	1.0	107	71	2.24×10^7	
1310(15)	RHR heat exch. room	1.0	107	71	2.11×10^7	
1311	Sampling room	1.0	107	71	4.42×10^4	
1312	Boron meter & R.C.	1.0	107	71	3.05×10^3	
1313	VCT room	1.0	119	100	3.83×10^4	
1314	Load center area El. 2000'	1.0	110	77	2.15×10^6	
1315	Containment spray additive tank area	1.0	110	77	6.45×10^4	
1316	Valve compartment	1.0	107	71	1.52×10^4	
1317	Seal water heat exch. room	1.0	107	71	1.52×10^4	
1318	Valve compartment	1.0	119	100	3.23×10^6	
1320	Corridor No. 4 El. 2000'	1.0	180	100	1.85×10^4	
1321	Vestibule	1.0	104	71	4.12×10^2	
1322	Piping penetration room	1.0	107	75	5.10×10^6	

TABLE 3.11(B)-2 (Sheet 4)

Room No.	Area	DBA Pressure Max. (psig)(9)	DBA Temp. Max. F(8)(9)	DBA RH % Max. (8)(9)	DBA Dose (Rad)(14)	Environmental ph DBA
Auxiliary Building						
1323	Piping penetration room	1.0	107	75	6.83×10^6	
1324	Feedwater pump valve compartment No. 1	Atmospheric	104	70	8.79×10^2	
1325(16)	Auxiliary feedpump (motor) room	Atmospheric	104	70	7.26×10^2	
1326(16)	Auxiliary feedpump (motor) room	Atmospheric	104	70	6.66×10^1	
1327	Feedwater pump valve compartment No. 2	Atmospheric	104	70	8.79×10^2	
1328	Feedwater pump valve compartment No. 3	Atmospheric	104	70	8.79×10^2	
1329	Vestibule	1.0	110	73	8.79×10^2	
1330	Feedwater pump valve compartment No. 4	Atmospheric	104	70	8.79×10^2	
1331(19)	Auxiliary feedpump (turbine) room	0.5	148.6	100	8.85×10^1	
1401(16)	CCW pump room	1.0	106	71	4.48×10^1	
1402	Corridor No. 1, El. 2026'	1.0	106	71	1.55×10^2	
1406(16)	CCW pump room	1.0	106	71	4.85×10^2	
1407	Boric Acid Batching Tank	1.0	109	73	-(20)	
1408	Corridor	1.0	106	71	7.88×10^2	
1409	Electrical penetration room	1.0	106	71	1.27×10^6	
1410	Electrical penetration room	1.0	106	71	1.74×10^6	
1411	Main feedwater room No. 1	6.7	324(18)	100	1.16×10^6	
1412	Main feedwater room No. 2	6.7	324(18)	100	1.18×10^6	
1413	Auxiliary shutdown panel room	1.0	106	71	1.10×10^2	
1501	Control room a/c equip. room	Atmospheric	104	71	7.14×10^1	
1502	CCW surge tank area (B)	1.0	106	71	8.92×10^2	

TABLE 3.11(B)-2 (Sheet 5)

Room No.	Area	DBA Pressure Max. (psig)(9)	DBA Temp. Max. F(8)(9)	DBA RH % Max. (8)(9)	DBA Dose (Rad)(14)	Environmental ph DBA
Auxiliary Building						
1503	CCW surge tank area (A)	1.0	106	71	9.58×10^2	
1504	Ctmt. purge exhaust and mech. equip. room (B)	1.0	106	71	3.97×10^2	
1506	Ctmt. purge supply air handling unit room (A)	Same as room 1504 conditions			7.80×10^5	
1507	Personnel hatch area El. 2047'-6"	1.0	106	71	1.09×10^6	
1508	Main steam/main feedwater isolation valve room(9)	6.7	324(18)	100	1.16×10^6	
1509	Main steam/main feedwater isolation valve room(9)	6.7	324(18)	100	1.18×10^6	
1512	Control room a/c equip. room	Atmospheric	104	71	3.13×10^2	
1513	Control bldg a/c equip. room	1.0	106	71	3.13×10^2	
Control Building						
3101	Pipe space tank area El. 1974'	Atmospheric	120	95	<2.5	
3105	Control building cable chase	Atmospheric	120	95	<2.5	
3106	Control building cable chase	Atmospheric	120	95	<2.5	
3222	ALARA Office/Dosimetry Issue, El. 1984'	Atmospheric	120	95	<2.5	
3223	HP Work Space, E1. 1984'	Atmospheric	120	95	<2.5	
3224	Vestibule No. 2 El. 1984'	Atmospheric	120	95	<2.5	
3229	Control building cable chase	Atmospheric	120	95	<2.5	
3230	Control building cable chase	Atmospheric	120	95	<2.5	
3301	ESF switchgear room	Atmospheric	104	70	<2.5	
3302	ESF switchgear room	Atmospheric	104	70	<2.5	
3404	Switchboard room (No. 4)	Atmospheric	104	70	<0.0005	
3405	Battery room	Atmospheric	104	70	<2.5	
3407	Battery room	Atmospheric	104	70	<2.5	

TABLE 3.11(B)-2 (Sheet 6)

Room No.	Area	DBA Pressure Max. (psig)(9)	DBA Temp. Max. F(8)(9)	DBA RH % Max. (8)(9)	DBA Dose (Rad)(14)	Environmental ph DBA
Control Building						
3408	Switchboard room (No. 1)	Atmospheric	104	70	<0.0005	
3410	Switchboard room (No. 2)	Atmospheric	104	70	<0.0005	
3411	Battery room	Atmospheric	104	70	<2.5	
3413	Battery room	Atmospheric	104	70	<2.5	
3414	Switchboard room (No. 3)	Atmospheric	104	70	<0.0005	
3415	HVAC equipment room	Atmospheric	104	95	<2.5	
3416	HVAC equipment room	Atmospheric	104	95	<2.5	
3501	Lower cable spreading room	Atmospheric	120	95	<2.5	
3601	Main control room	1/4 in. w.g. above atmospheric	84(10)	70	<2.5	
3605	Control room equip. cabinet area	Atmospheric	84(10)	70	<2.5	
3801	Upper cable spreading room	Atmospheric	120	95	<2.5	
Diesel Generator Building						
5000 Series	Diesel generator rooms(16)	Atmospheric	122	95	<500	
Fuel Building						
6000 Series	General areas (16)	Atmospheric	122	95	<1000	
ESW Pump House(16)						
U104	General areas	Atmospheric	122	95	<500	
U105						
UHS Electrical Room(16)						
U301	General areas	Atmospheric	122	95	<500	
U302						
U304-U307						

TABLE 3.11(B)-2 (Sheet 7)

Room No.	Area	DBA Pressure Max. (psig)(9)	DBA Temp. Max. F(8)(9)	DBA RH % Max. (8)(9)	DBA Dose (Rad)(14)	Environmental ph DBA
Radwaste Building						
7133	Non-radioactive tunnel and personnel access	Atmospheric	120	95	<2.5	
Other Buildings						
9101	CST Valve Compartment	Atmospheric	120	95	<500	
9102	RWST Valve Compartment	Atmospheric	120	95	<500	

NOTES:

(1) Short-term pressure differential across the steam generator loop compartment walls is <20 psi, and the short-term temperature differential across the steam generator loop compartment walls is <475°F.

(2) Short-term pressure differential across the reactor cavity wall is <150 psi, and the short-term temperature differential across the reactor cavity wall is <500°F.

(3) Deleted

(4) A pH of 4.0 could be experienced following a single failure in the containment spray system. The 11.0 pH could occur for approximately 30 minutes.

(5) The reactor building integrity will be tested at a maximum pressure of 69 psig, at up to 100 percent relative humidity, and up to 120°F, simultaneously. In addition, the containment has a negative design pressure of -3.0 psig.

(6) Following a postulated main steam line break, the containment vapor could become superheated, and the temperature of the vapor could exceed the containment design value of 320°F for a short period of time. Equipment design considers the following containment vapor condition:

Superheated vapor temperature	384.9°F (See Table 6.2.1-2)
Saturated (condensing) vapor temperature	250°F
Duration of superheated conditions	120 seconds

Sections 6.2.1.1.3 and Section 6.2.1.4.3.3 provide the results of conservative containment pressure/temperature analyses. As shown in Figure 6.2.1-82, these analyses show that the containment temperature exceeds the containment design temperature for a brief period of time. The old steam generator equipment qualification envelope is conservative, since the superheated vapor temperature is assumed to exist for 120 seconds, and as shown on Figure 6.2.1-85, the equipment surface temperatures remain significantly below this. Comparable surface temperatures for cables under the old steam generator analyses are shown on Figure 3.11(B)-7A. These surface temperature curves are conservative for replacement steam generator conditions.

(7) Deleted

(8) Except as detailed in Notes 9, 15, 16, and 19 of Table 3.11(B)-2, and Note 2 of Table 3.11(B)-1, the ambient temperature outside of the containment in rooms and corridors which do not have ESF coolers will not exceed 120°F because of lack of heat sources. Also, rooms and corridors which are not served by ESF coolers may experience relative humidities up to 95 percent following a loss of normal ventilation.

TABLE 3.11(B)-2 (Sheet 8)

(9) Maximum pressure, temperature, and humidity are based upon postulated pipe breaks for all rooms in the auxiliary building assuming initial conditions of atmospheric pressure, 104°F and 70 percent relative humidity. Note that the maximum temperature and relative humidity in these rooms may be higher for the loss of normal ventilation case as detailed in Notes 8, 15, and 16.

(10) Qualification temperature is 104°F.

(11) The conditions given are for the general plant areas and rooms which contain safety-related equipment. Supplemental analyses, protection, or shielding may be provided to meet qualification requirements as documented in Reference 5.

(12) Deleted

(13) The conditions provided are for the limiting DBA (LOCA, except for MSLB in rooms 1411, 1412, 1508, and 1509 and as detailed in Note 9). Specific LOCA, MSLB, and HELB profiles are documented in Figs. 3.11(B)-1 thru 3.11(B)-48 and calculation M-YY-49, including associated addenda.

(14) DBA dose values shown are derived by integrating calculated dose rates for 6 months following an accident; time dependant doses are documented in Figures 3.11(B)-50 through 3.11(B)-84.

(15) The temperature in rooms 1309 and 1310 will not exceed 175°F following a loss of normal ventilation with the RHR heat exchangers in operation.

(16) For rooms served by ESF coolers or fans outside containment, except for rooms 1409 and 1410 and as detailed in Note 17, the temperature and relative humidity will not exceed 122°F and 95 percent, repectively, following a loss of normal ventilation with the major components in the room operating (e.g., pumps) for extended periods.

(17) Deleted

(18) Following a postulated MSLB, room temperature could exceed the design value of 324°F. Appendix 3B, Section 3B.4.2 provides the results of temperature/pressure analyses for the rooms (1411, 1412, 1508, and 1509).

(19) The SBO steady state temperature for room 1331 is 144.5°F. The temperature following a loss of normal ventilation with the pump idle is 147.7°F. The DBA temperature for room 1331 (148.6°F) is for loss of offsite power (pump operating).

(20) DBA dose rates are not calculated for Room 1407.

TABLE 3.11(B)-3 IDENTIFICATION OF SAFETY-RELATED EQUIPMENT AND COMPONENTS: EQUIPMENT QUALIFICATION

Information previously contained

in

FSAR TABLE 3.11(B)-3

is now

controlled and maintained

in the

CALLAWAY EQUIPMENT LISTS (CEL).

TABLE 3.11(B)-4 CONTAINMENT WORST CASE RADIATION LEVELS (MRADS)

SOURCE	UPPER CTMT	ABOVE SUMP	SUBMERGED IN SUMP
Gamma			
Airborne Source	8.80 + 0	3.10 + 0	Negl.
Liquid Source	1.52 + 1	6.32 + 1	1.26 + 2
Plateout Source	9.24 - 2	1.39 - 1	Negl.
Total	2.41 + 1	6.65 + 1	1.26 + 2
Beta			
Airborne Source	1.46 + 2	1.46 + 2	0
Liquid Source	0	0	1.55 + 1
Plateout Source	1.40 + 1	2.08 + 1	0
Total	1.60 + 2	1.67 + 2	1.55 + 1
Total	1.84 + 2	2.33 + 2	1.42 + 2

TABLE 3.11(B)-5 CONTAINMENT SPRAY REQUIREMENTS

Sprayed Fluid	Injection Phase	
	Aqueous Solution, pH	4.0-7.0
	Boric Acid, ppm boron (max./min.)	2,500/2,350
Sprayed Fluid	Recirculation Phase	
	Aqueous Solution, pH	7.1 (at equilibrium)-11.0
	Boric Acid, ppm boron (max./min.)	2,500/1,971
Final Equilibrium Sump Fluid	Aqueous Solution, pH	7.1-8.1
	Boric Acid, ppm boron (max./min.)	2,500/1,971

TABLE 3.11(B)-6 FLOOD LEVELS IN THE AUXILIARY BUILDING AND CONTAINMENT

Room No.	Flood Level Elevation	Room No.	Flood Level Elevation
1101	1976' 9.72"	1301	2000' 7.87"
1102	1976' 9.72"	1302	2000' 0"
1103	1976' 9.72"	1304	2013' 6"
1104	1976' 9.72"	1305	2013' 6"
1105	1976' 9.72"	1306	2000' 0"
1106	1976' 9.72"	1307	2000' 7.87"
1107	1974' 0"	1308	2000' 0"
1108	1974' 0"	1309	2000' 0"
1109	1973' 4"	1310	2000' 0"
1110	1973' 4"	1311	2000' 0"
1111	1973' 4"	1312	2000' 0"
1112	1973' 4"	1313	2000' 7.87"
1113	1974' 0"	1314	2000' 7.87"
1114	1974' 0"	1315	2000' 7.87"
1115	1976' 9.72"	1316	2000' 0"
1116	1976' 9.72"	1317	2000' 0"
1117	1976' 9.72"	1318	2000' 7.87"
1119	1976' 9.72"	1320	2000' 7.87"
1120	1976' 9.72"	1322	2000' 0"
1121	1976' 9.72"	1323	2000' 0"
1122	1976' 9.72"	1324	2000' 0"
1123	1976' 9.72"	1325	2000' 0"
1124	1976' 9.72"	1326	2000' 0"
1125	1976' 9.72"	1327	2000' 0"
1126	1976' 9.72"	1328	2000' 0"
1127	1976' 9.72"	1329	2000' 0.07"
1128	1976' 9.72"	1330	2000' 0"
1129	1976' 9.72"	1331	2000' 0"
1130	1976' 9.72"	1401	2026' 6"
1201	1988' 0"	1402	2026' 6"
1202	1988' 0"	1403	2026' 0"
1203	1988' 7.32"	1406	2026' 0"
1203A	1988' 7.32"	1408	2026' 6"
1204	1988' 0"	1409	2026' 0"
1206	1990' 6.07"	1410	2026' 0"
1207	1990’ 6.07"	1411	2028' 2"

TABLE 3.11(B)-6 (Sheet 2)

Room No.	Flood Level Elevation
1412	2028' 2"
1413	2026' 0"
1501	2047' 6.2"
1502	2047' 6"
1503	2047' 6"
1504	2047' 6"
1505	2047' 6"
1506	2047' 6"
1507	2047' 6"
1508	2028' 2"
1509	2028' 2"
1512	2047' 6.2"
1513	2047' 6"
2000	
LOCA	2004’ 8”
MSLB	2004’ 4”

Note 1: If this table is revised, review FSAR SP Table 3.6-6 for potential impact.

TABLE 3.11(B)-7 SPECIFICATIONS REVIEWED UNDER THE NUREG-0588 PROGRAM

SPECIFICATION	DESCRIPTION
E01013-1	Termination Material (stub conn. kits)
E01013-2	Termination Material (ht. shrink fld spl.)
E01013-3	Termination Material (motor conn. kits)
E01013-4	Termination Material (end sealing kits)
E009	Switchgear Potential Transformer Cubicles *
E018	Motor Control Centers
E028	Local Control Stations/Terminal Boxes
E028A	Switches
E029	5 kV Power Cable
E035	Electrical Penetrations
E035B	Electrical Penetration Modules
E057	600 V Control Cable
E057A	600 V Control Cable
E057B	600 V Control and Power Cable
E058	600 V Power Cable
E060-1	Triaxial and Coaxial Cable
E060-2	Triaxial Cable Assembly (nuclear detectors) (*)
E061	Thermocouple Cable
E062	600 V Instrumentation Cable
E093	Auxiliary Relay Racks
J301-1	Pressure Transmitters (IC)
J301-2	Pressure Transmitters (OC)
J301-3	Pressure Transmitters Conduit Seals
J301-4	Pressure Transmitters - "R" Electronics

TABLE 3.11(B)-7 (Sheet 2)

SPECIFICATION	DESCRIPTION
J359	Hydrogen Monitoring System
J361A-1	Radiation Monitors
J361A-2	Radiation Monitor Cable
J364	Neutron Flux Monitoring System
J481	Level Transmitters
J558B	RTDs
J601A	Control Valves
J601B	Atmospheric Relief Valves
J603A-1	Solenoid Valves
J603A-2	Solenoid Valve Connector
J605A	Butterfly Valves
J1030	Pressure Transmitter (TOBARS)
J1032	Pressure Transmitter (Rosemounts)
J1064	Core Exit Thermocouple Connector and Extension Cable Upgrade
M021	Turbine Driven Auxiliary Feedwater Pump (*)
M088	Containment Spray Pumps
M221	Valve Limit Switch (*)
M223A-1	Motor-Operated Gate and Globe Valves (IC)
M223A-2	Motor-Operated Gate and Globe Valves (OC)
M223C	Motor-Operated Gate and Globe Valves
M224B	Motor-Operated Gate and Globe Valves
M225-1	Motor-Operated Gate and Globe Valves (IC)
M225-2	Motor-Operated Gate and Globe Valves (OC)
M231C	Motor-Operated Gate and Globe Valves (*)
M236	Motor-Operated Butterfly Valves (*)

TABLE 3.11(B)-7 (Sheet 3)

SPECIFICATION	DESCRIPTION
M237-1	Butterfly Valves (Limitorque) (OC)
M237-2	Butterfly Valves (Limitorque) (IC)
M237-3	Butterfly Valves (Bettis)
M612, M-1089	Room Coolers
M619.3	Hydrogen Mixing Fans
M620	Containment Cooling Fans
M627A	Dampers
M628	Steam Isolation Valves
M630	Feedwater Isolation Valves
M1142	Replacement 1" Stainless Steel Throttle Valves
S1027	Narrow Range RCS RTDs
W(AE2)	Large Pump Motors
W(AE3)	Canned Safety-Related Pump Motors (*)
W(ESE-01A)-1	Pressure Transmitters (A) (Barton-IC)
W(ESE-01A)-2	Pressure Transmitters (A) (Barton-OC)
W(ESE-01B)	Pressure Transmitters (A) (Veritrak)
W(ESE-03)	D.P. Transmitters (A) (Barton)
W(ESE-04)	D.P. Transmitters (B) (Barton)
W(ESE-06)	RTDs
W(ESE-08)	Excore Neutron Detectors (power range) (*)
W(ESE-40A)	Differential Pressure Indicating Switch (B)(*)
W(ESE-42A)	RVLIS - RTDs
W(ESE-43A)	CCMS - IC Thermocouples
W(ESE-44A)	CCMS - Reference Junction Box
W(ESE-44Z)	CCMS - Reference Junction Box

TABLE 3.11(B)-7 (Sheet 4)

SPECIFICATION	DESCRIPTION
W(ESE-47A,B,C)	Boron Dilution Mitigation System (*)
W(ESE-49A)	Differential Pressure Indicating Switch (A)
W(HE-01)	Motor-Operated Valves (A)
W(HE-02)	Solenoid Operated Valves (A)
W(HE-03)	Limit Switches (A)
W(HE-04)	Motor-Operated Valves (B)
W(HE-05)	Solenoid-Operated Valves (B)
W(HE-06)	Limit Switches (B)
W(HE-07)	Safety Valve Lift Indicating Switch Assembly
W(HE-07Z)	Safety Valve Lift Indicating Switch Assembly
W(HE-08)	Conax Connectors
W(HE-09)	Power-Operated Relief Valves
W(HE-10A)	Head Vent System - Isolation Valves
W(HE-10B)	Electronic Control Modules (*)
W(HE-10C)	Modulating Valves (*)
W(SP-1)	Hydrogen Recombiner

NOTE:

---

* Exempted from qualification. See Table 3.11(B)-8.

TABLE 3.11(B)-8 EXEMPTIONS FROM NUREG-0588 QUALIFICATION

SPECIFICATION	DESCRIPTION	EXPLANATION FOR EXCLUSION
E009	Switchgear Potential Transformer Cubicles	These devices provide anticipatory RCP trip functions only. They sense RCP bus voltage and frequency and provide RCP trips to prevent flow coast-down accidents. These trips are redundant to the reactor trip. However, no credit is taken for the RCP trip in the LOCA, MSLB, or HELB analyses. If a DBA occurs, these devices provide no additional function. Additionally, failure of these devices during a LOCA will not provide any adverse effects since the RCPs are not required during a LOCA.
E060	Triaxial Cable Assembly (Nuclear Detectors)	Refer to Specification W(ESE-8) for an explanation of exemption.
M021	Turbine Driven Auxiliary Feedwater Pump	This component and its associated auxiliaries are located in a room that is isolated from the rest of the Auxiliary Building. The room has a blow-out panel to the Turbine Building to prevent a HELB in that room from overpressurizing the room walls and pressurizing the adjacent Auxiliary Building rooms. The environment in this room, as a result of the HELB, would preclude equipment operation. However, the HELB would not affect the remaining two trains of auxiliary feedwater. Therefore, the turbine-driven auxiliary feedwater pump need not function during or following this HELB.

TABLE 3.11(B)-8 (Sheet 2)

SPECIFICATION	DESCRIPTION	EXPLANATION FOR EXCLUSION
M221	Valve Limit Switch	The limit switch for valve EN-V-97 is on the discharge line from the containment spray additive tank. This valve is a locked open manual gate valve. The failure of the limit switch post-LOCA will not adversely affect this valve or any other part of the containment spray system. The limit switch provides indication to the ESF status panel. The limit switch is used to verify the valve position following maintenance on the valve. Therefore, this limit switch is not required for a LOCA.
M-231C	Hydrogen Makeup Air Supply Valve	This valve is normally closed and fails closed, which is the desired position. This valve does not serve a containment isolation function nor does it receive any automatic signals (CIS-A, -B, or SIS). This valve is a part of the non-safety related hydrogen purge system, used only if both hydrogen recombiners fail. As described in Sections 6.2.5.1.1 and 6.2.5.2.2.4, by the time purging would be necessary, a number of containment penetrations could be used for the addition of air to the containment if this valve could not be opened after a LOCA.
M-236	Auxiliary Feed Pumps Suction Valve from ESW	These valves are not required post LOCA, as recovery is accomplished utilizing the ECCS systems and containment spray.
M-1142	Charging Pumps to RCP Seals Valves	Valves BG-HV-8357A and B are provided in the supplemental safety grade seal injection path. Qualification of these valves for a harsh environments not required, as stated below.

TABLE 3.11(B)-8 (Sheet 3)

SPECIFICATION	DESCRIPTION	EXPLANATION FOR EXCLUSION
		These valves are redundant motor operated valves on the discharge of the ECCS centrifugal charging pumps. The valves provide an alternate seal injection path for the reactor coolant pumps. Failure of these valves is acceptable since the normal seal injection path should be available and, if it is not, redundant seal cooling via component cooling water to the RCP thermal barrier cooling coil is available.
W(AE-3)	Canned Safety-Related Pump Motors	One pair of pumps is covered by this package. The boric acid transfer pumps are not utilized as a source of boron during a LOCA. The source of borated water is the refueling water storage tank (RWST). The boric acid transfer pumps are utilized as a source of boron in the event of a failure of the RWST during a tornado. A LOCA and tornado are not postulated to occur simultaneously. Therefore, these pumps are not required to operate during a LOCA.
W(ESE-8)	Two Section Power Range Excore Neutron Detectors	Power range high neutron flux trips are not assumed in the mitigation of a LOCA or main steam line breaks. These detectors may fail in any manner after an LOCA or MSLB, because reactor trip will occur as a result of a low pressurizer pressure or safety injection signal, with overtemperature delta-T as a backup. Therefore, the power range detectors are not required to be qualified to a harsh environment.

TABLE 3.11(B)-8 (Sheet 4)

SPECIFICATION	DESCRIPTION	EXPLANATION FOR EXCLUSION
W(ESE-40A)	Differential Pressure Switches for RHR and CCP Mini-flow Isolation	As discussed in Section 6.3.2.2 and Figure 6.3-1,EM-FIS -0917C and EM-FIS-0917D are interlocked with SIS to close BG-HV-8110 and BG-HV-8111 on an SIS coincident with ECCS centrifugal charging pump flow greater than 258.9 gpm. Following a LOCA, an SIS will open the Boron Injection path to the RCS so that safety injection can proceed. During this phase, the switches are not exposed to accident dose radiation. They will operate normally, protecting the ECCS CCPs against dead heading and providing the required flow to the RCS. Upon initiation of cold leg recirculation, the RCS pressure will have dropped enough that the ECCS CCPs can't deadhead themselves. Therefore, should the switches' failure cause the mini-flow valves (BG-HV-8110 and BG-HV-8111) to close, no hazard exists.

TABLE 3.11(B)-8 (Sheet 5)

SPECIFICATION	DESCRIPTION	EXPLANATION FOR EXCLUSION
		During cold leg recirculation from the containment sump, ECCS CCP flow delivered to the RCS will exceed the required flow. During this mode, the RHR pumps supply the ECCS CCPs and SI pumps. Should the switches' failure cause the mini-flow valves to fail open, approximately 60 gpm per CCP will be recirculated through the mini-flow path back to the ECCS CCP suction. However, the required flow will still be delivered to the RCS (see M-01BG03-E, FSAR Fig. 6.3-2, and FSAR Table 15.6-10). If necessary, these switches can be isolated during the recirculation phase by closing EM-V-041, 042, 244, and 245 in room 1126. These switches' safety function is for pump protection in the event of a feedwater line rupture (high RCS pressure), which does not cause a harsh environment at these switches' location (room 1126). The RHR mini-flow switches are located in a mild environment.

TABLE 3.11(B)-8 (Sheet 6)

SPECIFICATION	DESCRIPTION	EXPLANATION FOR EXCLUSION
W (ESE-47)	Boron Dilution Mitigation System Components	These components are not required following a LOCA or MSLB because the boron dilution mitigation system is not required to mitigate a LOCA or an MSLB and a boron dilution event is not postulated to occur concurrent with these DBAs. In addition, automatic valve realignments following the receipt of an SIS isolates the path of unborated water from the VCT and opens the RWST valves to the suction of the ECCS centrifugal charging pumps (valves BGLCV0112B and C close, and valves BNLCV0112D and E open). The boron dilution mitigation system provides the identical valve realignments to terminate the dilution event.
W(HE-10B and 10C)	Modulating Valves and Electronic Control Modules	One pair of redundant modulating valves and their electronic control modules is provided in the supplemental safety grade letdown path. Qualification of this equipment for a harsh environment is not required, as stated below. Valves BB-HV-8157A and B are redundant modulating valves on the excess letdown path to the pressurizer relief tank. Failure of these valves in any position is acceptable for the following accidents:

TABLE 3.11(B)-8 (Sheet 7)

SPECIFICATION	DESCRIPTION	EXPLANATION FOR EXCLUSION
		1. For a LOCA, excess letdown is not required and the flow path is isolated by redundant, qualified valves upstream of the excess letdown heat exchanger.
	.	2. For an MSLB, if safety grade letdown is required, the path to the PRT through the pressurizer power operated relief valves is available.

TABLE 3.11(B)-9 DELETED

The information provided in Table 3.11(B)-9 has been deleted. This table provided a safety related system listing. The listing of systems that perform or support these safety related functions is contained in the Callaway Equipment List (CEL). The specific safety function of each system is described in FSAR system description sections and the CEL database.

TABLE 3.11(B)-10 DELETED

3.11(N) ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The mechanical and electrical portions of the engineered safety features and the reactor protection system are designed to ensure acceptable performance in all environments anticipated under normal, test, and design basis accident conditions. This section presents information on the design basis and qualification verifications for mechanical and electrical equipment in the engineered safety features and the reactor protection system that are within the scope of the Westinghouse nuclear steam supply system (NSSS). Section 3.7(N) presents the seismic design requirements, and Section 3.10(N) presents the seismic qualification of electrical equipment.

3.11(N).1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Refer to Section 3.11(B).1.

3.11(N).2 QUALIFICATION TESTS AND ANALYSES

For Westinghouse NSSS Class 1E equipment, Westinghouse will meet the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974, "Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," including IEEE 323a-1975, the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975, by an appropriate combination of any or all of the following: type testing, operating experience, qualification by analysis, and ongoing qualification.

Reference 1 provides the general qualification methods Westinghouse uses in meeting the requirements of IEEE Standard 323-1974. The NRC Safety Evaluation Report dated November 10, 1983 accepted the Westinghouse methodology for equipment qualification. Reference 2 provides additional information concerning performance specifications and requirements and test plans for each safety-related equipment type. Table 3.11(B)-3 provides the equipment qualification data package reference for each piece of Westinghouse-supplied, safety-related equipment.

In the overall Class 1E Westinghouse equipment qualification program, generic environmental conditions (e.g., temperature, pressure, humidity, chemistry, and radiation) were established for the various pieces of Westinghouse-supplied Class 1E equipment. These conditions vary according to the location of the equipment. The environmental conditions for which the equipment is qualified are reported in the specific equipment qualification data package.

The requirements of GDC-1, 4, 23, and 50 are addressed in Section 3.1. Specific information concerning GDC-1 and 4 is reported in the applicable equipment qualification data packages (Ref. 2). Specific information concerning GDC-23 may be found in Section 7.2.2.2, and information regarding GDC-50 is provided in Section 6.2.

Information concerning how Appendix B of 10 CFR 50 is met is located in Section 17.2 of the Site Addendum. Regulatory Guides 1.30, 1.40, 1.73, and 1.89 are addressed in Appendix 3A.

3.11(N).3 QUALIFICATION TEST RESULTS

Table 3.11(B)-3 provides a cross-reference to the qualification results for each piece of Westinghouse-supplied equipment.

The results of qualification tests are reported in Reference 2. As the qualification program progresses, the qualification data package in Reference 2 will be updated accordingly.

3.11(N).4 LOSS OF VENTILATION

Refer to Section 3.11(B).4.

3.11(N).5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

Tables 3.11(B)-1 and 3.11(B)-2 provide the design source term for the chemical and radiation environment for normal operation and design accident environments, respectively. Source terms and chemical environments for which the NSSS scope equipment is qualified are provided in the appropriate equipment qualification data package (Ref. 2).

3.11(N).6 REFERENCES

1. Butterworth, G., and Miller, R. B., "Methodology For Qualifying Westinghouse WRD-Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587, Revision 6A, November 1983.

2. "Equipment Qualification Data Packages," WCAP-8587, Revision 1, Supplement 1, November 1978.

APPENDIX 3A - CONFORMANCE TO NRC REGULATORY GUIDES

This appendix briefly discusses the extent to which the standard plant conforms to NRC published regulatory guides, Division 1. The Standard Plant FSAR Appendix 3A may refer to the Addendum Appendix 3A or the Union Electric Company Operational Quality Assurance Manual (OQAM) for the specific regulatory commitment for certain regulatory guides. However, in cases where a reference is not made to the Addendum Appendix 3A or the OQAM, the commitment is as stated in the Standard Plant Appendix 3A and the same regulatory position is not repeated in the Addendum Appendix 3A or the OQAM. The statement of specific regulatory commitment for the following regulatory guides is located as indicated:

Callaway FSAR, Standard Plant - Regulatory Guides 1.1, 1.2, 1.3, 1.4, 1.5, 1.6, 1.7, 1.9, 1.10, 1.11, 1.12, 1.13, 1.14, 1.15, 1.18, 1.20, 1.22, 1.24, 1.25, 1.26, 1.29, 1.31, 1.32, 1.34, 1.35, 1.36, 1.40, 1.41, 1.42, 1.43, 1.44, 1.45, 1.46, 1.47, 1.48, 1.49, 1.50, 1.51, 1.52, 1.53, 1.54, 1.55, 1.56, 1.57, 1.59, 1.60, 1.61, 1.62, 1.63, 1.65, 1.66, 1.67, 1.68, 1.68.1, 1.68.2, 1.69, 1.70, 1.71, 1.72, 1.73, 1.75, 1.76, 1.77, 1.78, 1.79, 1.80, 1.81, 1.82, 1.83, 1.84, 1.85, 1.87, 1.89, 1.90, 1.92, 1.93, 1.95, 1.96, 1.97, 1.98, 1.99, 1.100, 1.101, 1.102*, 1.103, 1.104, 1.105, 1.106, 1.107, 1.108, 1.110, 1.112, 1.115, 1.117, 1.118, 1.119, 1.120, 1.121, 1.122, 1.124, 1.126, 1.128, 1.129, 1.130, 1.131, 1.133, 1.136, 1.137, 1.139, 1.140, 1.141, 1.142, 1.143, 1.147, 1.150, 1.152, 1.155, 1.158, 1.160, 1.163, 1.181, 1.182, 1.187, 1.195, and 1.205.

Callaway FSAR, Site Addendum - Regulatory Guides 1.17, 1.21, 1.23, 1.27, 1.59, 1.86, 1.91, 1.102*, 1.109, 1.111, 1.113, 1.114, 1.125, 1.127, 1.132, 1.134, 1.138, and 1.145.

Union Electric Operational Quality Assurance Manual - Regulatory Guides 1.8, 1.28, 1.30, 1.33, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88, 1.94, 1.116, 1.123, 1.144, and 1.146.

Exceptions to the guides are identified, and justification is presented or referenced. In the discussion of each guide, the sections or tables of the FSAR, where more detailed information is presented, are referenced. The referenced tables provide a position-by-position comparison to each regulatory position of section C of the regulatory guides. All statements within the Regulatory Position Section (C) of the Regulatory Guides are considered requirements unless a specific exception or clarification has been committed to by Union Electric. This is true regardless of the qualifier (i.e., "shall" or "should") which prefaces the statement. As regards to standards endorsed by the Regulatory Guide, unless further qualified within the Regulatory Guide, "shall" statements denote requirements while "should" statements denote recommendations.

---

* Refer to both the Callaway FSAR Standard Plant and the Callaway FSAR Site Addendum for the complete statement of regulatory commitment.

REGULATORY GUIDE 1.1 REVISION 0 DATED 11/2/70

Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 6.2.2.1.3, Safety Evaluation Eleven for the containment heat removal system pumps and Section 6.3.2.2 for the ECCS pumps.

REGULATORY GUIDE 1.2 REVISION 0 WITHDRAWN (Historical)

Thermal Shock to Reactor Pressure Vessels (Safety Guide 2)

DISCUSSION:

All recommendations of this regulatory guide have been followed. Regulatory Position C.1 is followed by Westinghouse's own analytical and experimental programs as well as by participation in the Heavy Section Steel Technology (HSST) program at Oak Ridge National Laboratory.

Analytical techniques have been developed by Westinghouse to perform fracture evaluations of reactor vessels under thermal shock loadings.

Under the HSST program, a number of 6-inch-thick, 39-inch-outside-diameter steel pressure vessels containing carefully prepared and sharpened surface cracks are being tested. Test conditions include both hydraulic internal pressure loadings and thermal shock loadings. The objective of this program is to validate analytical fracture mechanics techniques and demonstrate quantitatively the margin of safety inherent in reactor pressure vessels.

A number of vessels have been tested under hydraulic pressure loadings, and results have confirmed the validity of fracture analysis techniques. The results and implications of the hydraulic pressure tests are summarized in Oak Ridge National Laboratory report ORNL-TM-5090.

Four thermal shock experiments have been completed and are being evaluated. For representative conditions, flaws are shown to initiate and arrest in a predictable manner.

Westinghouse is continuing to obtain fracture toughness data for reactor pressure vessel steels through internally funded programs as well as HSST-sponsored work.

Fracture toughness testing of irradiated compact tension fracture toughness specimens has been completed. The complete post-irradiation data on 0.394-, 2-, and 4-inch-thick specimens are available from the HSST program. Both static and dynamic

post-irradiation fracture toughness data have been obtained. Evaluation of the data obtained to date on material irradiated to fluences between 2.2 and 4.5 x 10^{19} n/cm^2 indicates that the reference toughness curve, as contained in the American Society of Mechanical Engineers (ASME) Code, Section III, remains a conservative lower bound for toughness values for pressure vessel steels.

Details of progress and results obtained in the HSST program are available in the HSST program progress reports issued by Oak Ridge National Laboratory.

Regulatory Position C.2 is followed, inasmuch as no significant changes have been made in approved core or reactor designs.

Regulatory Position C.3 is followed, since the vessel design does not preclude the use of an engineering solution to assure adequate recovery of the fracture toughness properties of the vessel material. If additional margin is needed, the reactor vessel can be annealed at any point in its service life. This solution is already feasible, in principle, and could be performed with the vessel in place.

NOTE: Regulatory Guide 1.2 (Safety Guide 2) has been withdrawn by the NRC Staff letter to Regulatory Guide Distribution List, June 17, 1991. The guide has been superseded by 10 CFR 50, Section 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events. 10 CFR 50, Section 50.61 establishes screening criteria to effectively limit the extent of irradiation embrittlement permitted for reactor pressure vessel materials. The pressurized thermal shock requirements are sufficient to address thermal shock concerns. The withdrawal of Regulatory Guide 1.2 (Safety Guide 2) does not alter prior or existing licensing commitments based on its use.

REGULATORY GUIDE 1.3 REVISION 2 DATED 6/74

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.4 REVISION 2 DATED 6/74

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 15.6-7.

REGULATORY GUIDE 1.5 REVISION 0 DATED 3/71

Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5)

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.6 REVISION 0 DATED 3/71

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6)

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 8.1.4.3.

REGULATORY GUIDE 1.7 REVISION 3 DATED 3/07

Control of Combustible Gas Concentrations in Containment

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.2.5-6.

REGULATORY GUIDE 1.8

Personnel Selection and Training

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual..

REGULATORY GUIDE 1.9 REVISION 3 DATED 7/93

Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used As Class 1E Onsite Electric Power Systems at Nuclear Power Plants

DISCUSSION:

With respect to the original selection, design and qualification of emergency diesel generators (per Revision 1 of the regulatory guide), the recommendations of this regulatory guide were met. With regard to periodic, in-service testing of the diesel generators per Revision 3 of this regulatory guide, testing is performed in accordance with the plant Technical Specifications. The testing requirements in the Technical

Specifications are based on Regulatory Guide 1.9, Revision 3. Differences between the test requirements of the Technical Specifications and the recommendations of this regulatory guide are due to the Standard Technical Specifications and/or approved changes to the Technical Specifications.

REGULATORY GUIDE 1.10 REVISION 1 DATED 1/73

Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures.

DISCUSSION:

The recommendations of this regulatory guide are met. The temperature at which visual inspection may proceed is taken as the temperature for which the splice has cooled sufficiently so that inspection operations are not hampered.

REGULATORY GUIDE 1.11 REVISION 0 DATED 3/71

Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11).

DISCUSSION:

The instrument lines that penetrate the containment are the containment pressure sensing lines and the reactor vessel level indication system (RVLIS) lines. The containment pressure sensing lines are part of the protection system and meet the recommendations of Regulatory Position C.1, except for separation of the sensing lines for GNPT0935 and GNPT0937 as described in Section 7.3.8.1.1. The design of the RVLIS is discussed in Table 6.2.4-2, Figure 6.2.4-1, and Section 18.2.13.

REGULATORY GUIDE 1.12 REVISION 1 DATED 4/74

Instrumentation for Earthquakes

DISCUSSION:

The recommendations of this regulatory guide are met with the exceptions noted in Section 3.7(B).4, Seismic Instrumentation Program.

REGULATORY GUIDE 1.13 REVISION 1 DATED 12/75

Spent Fuel Storage Facility Design Basis

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 9.1-3.

REGULATORY GUIDE 1.14 REVISION 1 DATED 8/75

Reactor Coolant Pump Flywheel Integrity

DISCUSSION:

The Westinghouse design follows the recommendations of Regulatory Guide 1.14, Revision 1, except for the following:

a. Postspin inspection

Westinghouse has shown in WCAP-8163, September 1973, "Reactor Coolant Pump Integrity in LOCA," that the flywheel would not fail at 290 percent of normal speed, for a flywheel flaw of 1.15 inches or less in length. Results for a double-ended guillotine break at the pump discharge, with full separation of pipe ends assumed, show the maximum overspeed to be less than 110 percent of normal speed. The maximum overspeed was calculated in WCAP-8163 to be about 280 percent of normal speed for the same postulated break and an assumed instantaneous loss of power to the reactor coolant pump. In comparison with the overspeed presented above, the flywheel is tested at 125 percent of normal speed. The flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125 percent, provided that no flaws greater than 1.15 inches are present. If the maximum speed were 125 percent of normal speed or less, the critical flaw size for failure would exceed 6 inches in length. Nondestructive tests and critical dimension examinations are all performed before the spin tests. The inspection methods employed (described in WCAP-8163) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 inches for 290 percent of normal speed would be detected. Flaws in the flywheel will be recorded in the prespin inspection program. Flaw growth attributable to the spin test (i.e., from a single reversal of stress, up to speed and back), under the most adverse conditions, is about three orders of magnitude smaller than that which nondestructive inspection techniques are capable of detecting. For these reasons, Westinghouse performs no postspin inspections and concludes that prespin test inspections are adequate.

b. Interference fit stresses and excessive deformation

Much of Revision 1 to Regulatory Guide 1.14 deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because the Callaway Plant design has a light interference fit between the flywheel and the shaft, at zero speed, the hoop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and, at normal speed, the flywheel

is not in contact with the shaft in the sense intended by Revision 1. Hence, the definition of "Excessive Deformation," as defined in Revision 1 of Regulatory Guide 1.14, is not applicable to the Callaway Plant design, since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft do not cause unbalance of the flywheel. Extensive Westinghouse experience with reactor coolant pump flywheels installed in this fashion has verified the adequacy of the design.

The combined primary stress levels, as defined in Revision 0 of Safety Guide 14 (Regulatory Positions C.2.a and C.2.c), are both conservative and proven and, therefore, no changes to these stress levels are considered to be necessary. Westinghouse designs to these stress limits and thus does not have permanent distortion of the flywheel bore at normal or spin test conditions.

c. Discussion B, cross-rolling ratio of 1 to 3

Specification of a cross-rolling ratio is considered to be unnecessary, since past evaluations have shown that ASME SA-533, Grade B, Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotropy in the flywheel material by means of cross rolling is unnecessary since adequate margins of safety are provided by both flywheel material selection (ASME SA-533, Grade B, Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material.

d. Regulatory Position C.1.a, relative to vacuum-melting and degassing process or the electroslag process

The requirements for vacuum-melting and the degassing process or the electroslag process are not essential in meeting the balance of the regulatory position nor do they, in themselves, ensure compliance with the overall regulatory position. The initial Safety Guide 14 (10/27/71) stated that the "flywheel material should be produced by a process that minimized flaws in the material and improves its fracture toughness properties." This is accomplished by using ASME SA-533 material, including vacuum treatment.

e. Regulatory Position C.2.b

The pumps are designed to the following criteria: "Design speed is 125 percent of normal speed, which is greater than the speed which is anticipated during a turbine generator overspeed."

f. Regulatory Position C.4.b, flywheel inservice inspection

In lieu of Position C.4.b, a qualified in-place UT examination over the volume from the inner bore of the flywheel to the inner circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals. (See Technical Specification 5.5.7)

REGULATORY GUIDE 1.15 REVISION 1 DATED 12/72

Testing of Reinforcing Bars for Category I Concrete Structures

DISCUSSION:

The recommendations of this regulatory guide are met. Revisions of ASTM A615 and A370 current with industry practice are utilized following appropriate engineering review.

REGULATORY GUIDE 1.17

Protection of Nuclear Power Plants Against Industrial Sabotage

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.18 REVISION 1 WITHDRAWN (Historical)

Structural Acceptance Test for Concrete Primary Reactor Containments

DISCUSSION:

Compliance with this guide is planned, insofar as practicable. The following exceptions are considered to be within the intent of this Regulatory Guide:

a. Paragraph C.1: A continuous increase in containment pressure, rather than incremental pressure increases, is considered acceptable, provided that data observations are made rapidly at each pressure datum. Rapidly is defined as requiring a time interval for the data point sample sufficiently short so that the change in pressure during the observation would cause a change in structural response of less than 5 percent of the total anticipated change. For example, assume a total expected strain of 200 microstrain (micro-inches per inch). The period of a data observation, therefore, would be required to be equal to or less than the time during which pressurization would create a 10 microstrain change.

b. Paragraph C.1: It is intended that a hold period for at least 1 hour be provided at maximum test pressure or for such time as is necessary for recording crack patterns.

c. Paragraph C.2: It is intended that the number and distribution of measuring points for monitoring radial deflection be selected so that the as-built condition can be considered in the assessment of roundup, buttress-shell interaction, and general shell response. Measurements are made at points similar to those shown in Section 9.0 of BC-TOP-5-A. However, to obtain the most significant data, the measuring point locations may be changed to those where the as-built containment is at the limit of tolerance, if such points exist. Accordingly, an arbitrary selection of measurement points is not intended.

d. Paragraph C.3: Measurement of tangential deflections is not planned.

e. Paragraph C.5 is not applied for non-prototype containments.

f. Paragraph C.6: Shear strain measurements under end anchor bearing plates are not planned at the present state of the art. Experimental evidence contained in BC-TOP-7 and BC-TOP-8 is submitted in lieu of measurement of the vertical and horizontal strains under a vertical tendon end anchor bearing plate. For measurements of vertical and horizontal strains under vertical tendon end anchor bearing plates, this experimental evidence indicates that a gage location within approximately one quarter of the bearing plate width from the exposed face of the bearing plate must be used.

g. Paragraph C.9: It is intended to schedule structural integrity testing for periods when extremely inclement weather is not forecast. Should, despite the forecast, snow, heavy rain, or strong wind occur during the test, the test results will be considered valid unless there is evidence to indicate otherwise.

h. Paragraph C.10: Should, due to an unexpected condition, the test pressure drop to or below the next pressure level, it is intended to continue the test, without a restart at atmospheric pressure, unless the structural response deviates significantly from that expected.

i. Appendix A of the Regulatory Guide: The reactor building has no prototypal features; therefore, this appendix is not applicable.

j. The Callaway containment structural integrity test and the integrated leak rate test were performed in January 1984. The results of the tests are detailed in the formal report submittal to the NRC transmitted by ULNRC-794, dated April 9, 1984, from D.F. Schnell to H.R. Denton.

REGULATORY GUIDE 1.20 REVISION 2 DATED 5/76

Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 3.9(N).2.4.

REGULATORY GUIDE 1.21 REVISION 1 DATED 6/74

Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.22 REVISION 0 DATED 2/72

Periodic Testing of Protection System Actuation Functions (Safety Guide 22)

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 7.1.2.5 and Table 7.1-3.

REGULATORY GUIDE 1.23 REVISION 1 DATED 3/07

Meteorological Monitoring Programs for Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A in the Site Addendum.

REGULATORY GUIDE 1.24 REVISION 0 DATED 3/72

Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 15.7-1.

REGULATORY GUIDE 1.25 REVISION 0 DATED 3/72

Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 15.7-2.

REGULATORY GUIDE 1.26 REVISION 3 DATED 2/76

Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 3.2-4. As described in Section 3.2, Westinghouse utilizes the safety classes defined in ANSI N18.2a-1975.

REGULATORY GUIDE 1.27

Ultimate Heat Sink for Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A in the Site Addendum.

REGULATORY GUIDE 1.28

Quality Assurance Program Requirements (Design and Construction)

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.29 REVISION 3 DATED 9/78

Seismic Design Classification

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 3.2-3. As described in Section 3.2, Westinghouse utilizes the safety classes as defined in ANSI N18.2a-1975.

REGULATORY GUIDE 1.30

Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.31 REVISION 3 DATED 4/78

Control of Ferrite Content in Stainless Steel Weld Metal

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.1-9.

REGULATORY GUIDE 1.32 REVISION 2 DATED 2/77

Criteria for Safety-Related Electic Power Systems for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide (which is conditionally based on IEEE Standard 308-1974) are met, except that the requirements described in the regulatory guide and IEEE Standard 308-1974 pertaining to the maintenance, testing and replacement of lead-acid storage batteries are taken from IEEE Standard 450-1995 instead of IEEE Standard 450-1975. Refer to Sections 8.1.4.3 and 8.3.2.2.1.

REGULATORY GUIDE 1.33

Quality Assurance Program Requirements (Operation)

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.34 REVISION 0 DATED 12/72

Control of Electroslag Weld Properties

DISCUSSION:

Electroslag welding is not used for items within the Bechtel scope of supply.

Where electroslag welding is used in fabricating nuclear plant components, the Westinghouse procurement practice requires vendors to follow the recommendations of Regulatory Guide 1.34.

REGULATORY GUIDE 1.35	PROPOSED REVISION 3	DATED 4/79

Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures

DISCUSSION:

The post-tensioning system is described in Section 3.8.1.1.2. As described in Technical Specification 5.5.6, surveillance of the containment tendons is performed in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR 50.55a, in lieu of this Regulatory Guide. All tendons are accessible with the exception of two tendons at El. 2073, azimuth 281 degrees, which are blocked by the auxiliary building roof. Four additional tendons (two at approximately El. 2026 and two at approximately El. 2047) at azimuth 281 degrees are accessible for visual inspection only.

REGULATORY GUIDE 1.36	REVISION 0	DATED 2/73

Nonmetallic Thermal Insulation for Austenitic Stainless Steel

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.1-6.

REGULATORY GUIDE 1.37

Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.38

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.39

Housekeeping Requirements for Water-Cooled Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.40 REVISION 0 DATED 3/73

Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 3.11(B).

REGULATORY GUIDE 1.41 REVISION 0 DATED 3/73

Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Sections 8.1.4.3 and 8.3.2.2.1.

REGULATORY GUIDE 1.42 REVISION NA DATED NA

DISCUSSION:

This regulatory guide has been withdrawn by the NRC.

REGULATORY GUIDE 1.43 REVISION 0 DATED 5/73

Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

DISCUSSION:

Westinghouse practices achieve the same purpose as Regulatory Guide 1.43 by requiring qualification of any "high heat input" processes, such as the submerged-arc wide-strip welding process and the submerged-arc 6-wire process used on ASME SA-508, Class 2, material, with a performance test as described in Regulatory Position C.2 of the guide. No qualifications are required by the regulatory guide for ASME SA-533 material and equivalent chemistry for forging grade ASME SA-508, Class 3, material.

The fabricator monitors and records the weld parameters to verify agreement with the parameters established by the procedure qualification as stated in Regulatory Position C.3.

Stainless steel weld cladding of low-alloy steel components is not employed on components outside the NSSS.

REGULATORY GUIDE 1.44 REVISION 0 DATED 5/73

Control of the Use of Sensitized Stainless Steel

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.1-4.

REGULATORY GUIDE 1.45 REVISION 0 DATED 5/73

Reactor Coolant Pressure Boundary Leakage Detection Systems

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 5.2-6.

REGULATORY GUIDE 1.46 REVISION 0 DATED 5/73

Protection Against Pipe Whip Inside Containment

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 3.6-2 for the balance of plant and Section 3.6.1 for the NSSS.

REGULATORY GUIDE 1.47 REVISION 0 DATED 5/73

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 7.5-3. In addition, the bypassed and inoperable indicating system meets Branch Technical Position ICSB 21 titled Guidance for Application of Regulatory Guide 1.47.

REGULATORY GUIDE 1.48 REVISION 0 DATED 5/73

Design Limits and Loading Combinations for Seismic Category I Fluid System Components

DISCUSSION:

Westinghouse-supplied components are designed, using the stress limits and loading combinations presented in Sections 3.9(N).1 and 5.2 for Code Class 1 components and in Section 3.9(N).3 for Code Class 2 and 3 components. The conservatism in these limits and the associated ASME design requirements preclude any component structural failure.

The operability of active Code Class 1, 2, and 3 valves and active Code Class 2 and 3 pumps (there are no active Class 1 pumps) will be verified by methods detailed in Sections 3.9(N).1 and 5.2 for Code Class 1 components and in Section 3.9(N).3 for Code Class 2 and 3 components.

The use of the foregoing methods provides an acceptable alternate method to meeting the guidance of this regulatory guide.

For seismic Category I fluid system components not furnished with the NSSS, the recommendations of this regulatory guide are met as discussed in Section 3.9(B).3.1 and Table 3.9(B)-13.

REGULATORY GUIDE 1.49 REVISION 1 DATED 12/73

Power Levels of Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met, since the reactor core thermal power level is 3,565 MWt, compared with the limits of 3,800 MWt of this regulatory guide.

REGULATORY GUIDE 1.50 REVISION 0 DATED 5/73

Control of Preheat Temperature for Welding of Low-Alloy Steel

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.1-7.

REGULATORY GUIDE 1.51 REVISION NA DATED NA

DISCUSSION:

This regulatory guide has been withdrawn by the NRC.

REGULATORY GUIDE 1.52 REVISION 2 DATED 3/78

Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 9.4-2.

Prefilter design, construction and testing referenced in Table 9.4-2 paragraph 3.c is in accordance with Regulatory Guide 1.52 Revision 3 dated June 2001. The duration of maintenance runs referenced in Table 9.4-2 paragraph 4.d is also in accordance with Regulatory Guide 1.52 Revision 3. The frequency of maintenance runs referenced in Table 9.4-2 paragraph 4.d is controlled in accordance with the Surveillance Frequency Control Program (SFCP) described in the Administrative Controls section of the Technical Specifications.

REGULATORY GUIDE 1.53 REVISION 0 DATED 6/73

Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 7.1-4 for the portions of plant protection systems provided with the balance of plant. The Westinghouse-furnished systems meet the recommendations of this regulatory guide as described in Section 7.1.2.6.1.

REGULATORY GUIDE 1.54 REVISION 0 DATED 6/73

Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.1-2.

REGULATORY GUIDE 1.55 REVISION 0 DATED 6/73

Concrete Placement in Category I Structures

DISCUSSION:

The recommendations of this regulatory guide are met, except as described below.

BC-TOP-5-A is used as a design code in lieu of ACI/ASME Proposed Standard-Code for Concrete Reactor Vessels and Containments. ANSI N45.2.5-1974 (Rev. 1), Supplementary Q.A. Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants, is used in lieu of ANSI N45.2.5-1972 (proposed).

Creep tests are normally performed on prestressed structures only. Loss of prestress through creep is not applicable to nonprestressed structures.

Regulatory Position 2 of the regulatory guide lists the responsibilities of the "Designer." Under the designer's role are listed the responsibilities for checking the design and shop drawings for placement of reinforcing bars, location of embedded items, as well as locations of construction joints.

On the project, Bechtel engineering has the responsibility to check the design and shop drawings and locate the construction joints. Changes to design drawings by the "Constructor" require the "Engineer's" approval.

<u>REGULATORY GUIDE 1.56</u> <u>REVISION 1</u> <u>DATED 7/78</u>

Maintenance of Water Purity in Boiling Water Reactors

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

<u>REGULATORY GUIDE 1.57</u> <u>REVISION 0</u> <u>DATED 6/73</u>

Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

DISCUSSION:

The recommendations of this regulatory guide are met to the extent that they apply to ASME Code Class MC Mechanical and Electrical Penetration Assemblies described in Section 3.8.2.5.

<u>REGULATORY GUIDE 1.58</u>

Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.59 REVISION 2 DATED 8/77

Design Basis Floods for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met for the design of safety-related structures, systems and components. Refer to Section 3.4. Also, refer to Section 3.4 in each Site Addendum.

REGULATORY GUIDE 1.60 REVISION 1 DATED 12/73

Design Response Spectra for Seismic Design of Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are used for the non-NSSS design as the basis for the ground design response spectra. Refer to Section 3.7(B).1.1. Westinghouse utilizes the design response spectra of this regulatory guide in conjunction with the damping values approved by the NRC in WCAP-7921-AR, dated May 1974.

REGULATORY GUIDE 1.61 Rev. 0 Dated 10/73

Damping Values for Seismic Design of Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 3.7(B).1.3 for those items not supplied by Westinghouse, with the following exceptions. Supports for Class 1E cable tray are designed for the SSE considering up to 20-percent damping. Likewise, Class 1E conduit supports are designed for the SSE based on 7-percent damping.

In accordance with Regulatory Position C.2, these damping values were established as the result of a test program. Further discussion is included in Section 3.10(B).3.

The Westinghouse-supplied equipment satisfies the damping values suggested by the regulatory guide with the exception of the damping value (3 percent critical) for the faulted condition of large piping systems. Higher damping values, when justified by documented data, are allowed by Regulatory Position C.2. A conservative value of 4 percent critical has therefore been justified by testing for the Westinghouse reactor coolant loop configuration in WCAP-7921-AR and has been approved by the NRC. See Section 3.7(N).1.3 for further discussion.

Code Case N-411-1, Alternative Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1, may also be applied subject to the conditions imposed by the NRC staff in Regulatory Guide 1.84.

REGULATORY GUIDE 1.61 REVISION 1 DATED 3/07

Damping Values for Seismic Design of Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide were used in the analysis of the AREVA-supplied integrated head assembly (IHA). The Regulatory Guide 1.61 Revision 1, Table 1 note allowing use of a “weighted average” for the design-basis Safe Shutdown Earthquake (SSE) damping value applicable to steel structures of different connection types is also applied to determine the IHA design-basis Operating Basis Earthquake (OBE) damping value, as approved by the NRC via Amendment to the Callaway Operating License. Damping values more conservative (i.e. lower) than the calculated “weighted average” damping values have been used in conjunction with the response spectrum analysis of the IHA to qualify various structural components in the IHA and in developing the reaction loads from the IHA on the replacement reactor vessel closure head (RRVCH) and on the containment cavity wall seismic embedments. The current licensing basis use of Regulatory Guide 1.61 Revision 0 is retained for all structural analyses that do not address the structural qualification of the IHA.

REGULATORY GUIDE 1.62 REVISION 0 DATED 10/73

Manual Initiation of Protective Actions

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 7.1-5 for those safety-related systems provided with the balance of plant. The Westinghouse-furnished systems meet the recommendations of this regulatory guide as described in Section 7.3.8.2.

REGULATORY GUIDE 1.63 REVISION 2 DATED 7/78

Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 8.1.4.3.

REGULATORY GUIDE 1.64

Quality Assurance Requirements for the Design of Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.65 REVISION 0 DATED 10/73

Materials and Inspections for Reactor Vessel Closure Studs

DISCUSSION:

Westinghouse follows the recommendations of this regulatory guide, with the following exceptions:

a. The use of modified SA-540, Grade B-24, as specified in the ASME Code (Code Case 1605) is permitted by Westinghouse, but is not listed in this regulatory guide.

b. A maximum ultimate tensile strength of 170,000 psi is not specified by Westinghouse, as recommended by this regulatory guide.

Exception a. above is not an issue since Code Case 1605 has been found acceptable to the NRC for application in the construction of components for water-cooled nuclear power plants within the limitations discussed in Regulatory Guide 1.85. The use of Code Case 1605 for reactor vessel closure stud materials is not precluded by this regulatory guide.

Exception b. is not considered by Westinghouse to be a safety issue for the following reasons:

The ASME Code requirement for toughness for reactor vessel bolting has precluded the regulatory guide's additional recommendation for tensile strength limitation, since to obtain the required toughness levels the tensile levels are reduced.

Westinghouse has specified both 45 ft-lb and 25 mils lateral expansion for control of fracture toughness determined by Charpy-V testing, required by the ASME Code, Section III, Summer 1973 Addenda and 10 CFR 50, Appendix G (Paragraph IV.A.4). These toughness requirements ensure optimization of the stud bolt material tempering operation with the accompanying reduction of the tensile strength level when compared with previous ASME Code requirements.

Prior to 1972, the ASME Code required a 35 ft-lb toughness level which provided maximum tensile strength levels ranging from approximately 155 to 178 kpsi (Westinghouse review of limited data - 25 heats).

After publication of the Summer 1973 Addenda to the ASME Code and 10 CFR 50, Appendix G, wherein the toughness requirements were modified to 45 ft-lb with 25 mils lateral expansion, all bolt material data reviewed on Westinghouse plants showed tensile strengths of less than 170 kpsi.

The specification of both impact and maximum tensile strength as stated in the regulatory guide results in unnecessary hardship in procurement of material without any additional improvement in quality.

The closure-stud bolting material is procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. This strength level is compatible with the fracture toughness requirements of 10 CFR 50, Appendix G (Paragraph I.C), although higher-strength-level bolting materials are permitted by the ASME Code.

The primary concern of the regulatory position concerning a maximum tensile strength is to minimize the susceptibility of the bolting material to stress corrosion cracking.

Stress corrosion has not been observed in reactor vessel closure-stud bolting manufactured from material of this strength level. Accelerated stress corrosion test data do exist for materials of 170,000 psi minimum yield strength exposed to marine water environments stressed to 75 percent of the yield strength (given in Ref. 2 of the regulatory guide). These data are not considered applicable to Westinghouse reactor vessel closure-stud bolting because of the specified yield strength differences and a less severe environment; this has been demonstrated by years of satisfactory service experience.

Additional protection against the possibility of incurring corrosion effects is ensured by:

a. Decrease in level of tensile strength compatible with the requirement of fracture toughness as described above.

b. Design of the reactor vessel studs, nuts, and washers, allowing them to be completely removed during each refueling, permitting visual and/or nondestructive inspection in parallel with refueling operations to assess protection against corrosion, as part of the inservice inspection described in Section 5.2.4.

c. Design of the reactor vessel studs, nuts, and washers, providing protection against corrosion by allowing them to be completely removed during each refueling and placed in storage racks as required by refueling procedures. The stud holes in the reactor vessel flange are sealed with special plugs before removing the reactor closure. Thus, the bolting materials and stud

holes are never exposed to the borated refueling cavity water. When studs cannot be removed, engineered systems are employed (e.g., encapsulation during refueling) to protect the stud and its respective stud hole.

REGULATORY GUIDE 1.66 REVISION NA DATED NA

DISCUSSION:

This regulatory guide has been withdrawn by the NRC.

REGULATORY GUIDE 1.67 REVISION 0 DATED 10/73

Installation of Overpressure Protection Devices

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 3.9(B).3.3.

REGULATORY GUIDE 1.68 REVISION 2 DATED 8/78

Initial Test Programs for Water-Cooled Nuclear Power Plants

DISCUSSION:

The section describing commitments to Regulatory Guide 1.68 for the Callaway Initial Test Program has been deleted. The deleted material is contained in the FSAR on record as of the receipt of the Callaway Operating License No. NPF-30 on October 18, 1984. This information will not be reproduced in later revisions to the FSAR due to the historical status of the content. This information may be provided, however, upon request from the Union Electric Nuclear Licensing Department.

REGULATORY GUIDE 1.68.1 REVISION 1 DATED 1/77

Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.68.2 REVISION 1 DATED 7/78

Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Callaway has performed a test program to demonstrate the remote shutdown capability of the unit.

REGULATORY GUIDE 1.69 REVISION 0 DATED 12/73

Concrete Radiation Shields for Nuclear Power Plants

DISCUSSION:

The requirements of the regulatory guide as they apply to the construction of shielding structures are met, with the following clarification:

a. Accident condition analysis procedures and load combinations (Reference ANSI N101.6, Section 4.3.5) are in accordance with Section 3.8.

b. Condition of aggregate: (Reference ANSI N101.6, Section 5.1.6) When aggregates contain montmorillonite clays, top soil and claystone, fine aggregate shall have a minimum sand equivalent of 75 when tested in accordance with Test Method Calif. No. 217, and coarse aggregate shall have a minimum cleanness value of 75 when tested in accordance with Test Method No. 227, as specified in the California Division of Highways Test Methods. In addition, the aggregate must pass ASTM Test C-117 (Material finer than 200 Sieve) which provides a measure of cleanness.

c. Recommendations for forms: (Reference ANSI N101.6, Section 4.7, 6.0, 8.16) Forms are made of wood, metal, structural hardwood, or other suitable material that will produce the required surface finish. Forms are constructed in accordance with ACI 347, Recommended Practice for Concrete Framework, and are made to conform to the shape, form, line, and grade to prevent deformation under load, and are designed to be readily removable. Lumber to be reused is thoroughly cleaned before reuse. Form-release agents are compatible with protective coatings that will be subsequently used.

d. Tendons and anchors for prestressed concrete (Reference ANSI N101.6, Section 6.3.1) are in accordance with BC-TOP-5-A.

e. Aggregate samples are submitted for testing prior to acceptance for job use. However, no requirement exists to retain these samples for permanent job records. (Reference ANSI N101.6, Section 8.1.8)

f. Mixing time of concrete (Reference ANSI N101.6, Section 8.2.2) is in accordance with ASTM C-94:

"Where mixer performance tests have been made on given concrete mixtures in accordance with the testing program set forth in the following paragraphs and the mixers have been charged to their rated capacity, the acceptable mixing time may be reduced for those particular circumstance to a point at which satisfactory mixing, defined in 10.3.3, shall have been accomplished. When the mixing time is so reduced, the maximum time of mixing shall not exceed this reduced time by more than 60 seconds for air entrained concrete."

Additional details in regard to concrete mixing are included in Section 3.8.1.6.1.2.

g. Specific requirements for pressurized grouting (Reference ANSI N101.6, Section 8.6.2) are determined on a case-by-case basis. The resulting procedure must assume that all fillings are bonded tightly to the surface of the concrete and be sound and free from shrinkage, cracks, and hollow-sounding areas.

h. Curing of ordinary concrete (Reference ANSI N101.6, Section 8.7.2) is by one or a combination of the methods described in ACI 308, Recommended Practice for Curing Concrete, and occurs for a minimum of 7 days. Concrete is protected from freezing by adequate covering and heating or by insulated forms and covering. The concrete members are completely enclosed during cold weather in accordance with Chapter 1 of ACI 306, Recommended Practice for Cold Weather Concreting, but in no case are exposed to a temperature lower than 35°F.

i. Acceptance of concrete compressive strength (Reference ANSI N101.6, Section 11.4.1) is in accordance with ACI 318, Building Code Requirements for Reinforced Concrete, Chapter 4.3.

j. Instead of tests on the completed shield (Reference ANSI N101.6, Section 11.5) routine surveys in accordance with the ALARA program (Chapter 12.0) are used to establish that the general area radiation levels are within the limits of specified radiation zones. An exhaust system of sufficient capacity to maintain a negative pressure is also provided to prevent uncontrolled release of airborne radioactive material.

k. As an alternate to CRD C-119, Method of Test for Flat and Elongated Particles in Coarse Aggregate (Reference ANSI N101.6, Table 2), a method which identifies all particles having a maximum dimension in excess of four times the minimum dimension from a 5-pound sample may be used.

REGULATORY GUIDE 1.70 REVISION 3 DATED 11/78

Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants

DISCUSSION:

The Callaway FSAR is written to the format of Revision 2 of this regulatory guide and the information requested by Revision 3 has been incorporated. When FSAR revisions are required due to regulatory changes the information may be presented in a format consistent with the new regulations, and not exactly as prescribed in this regulatory guide. See commitment to Regulatory Guide 1.181.

REGULATORY GUIDE 1.71 REVISION 0 DATED 12/73

Welder Qualification for Areas of Limited Accessiblity

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.1-8.

REGULATORY GUIDE 1.72 REVISION 2 DATED 11/78

Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin

DISCUSSION:

The recommendations of this regulatory guide are not applicable to the SNUPPS applications.

REGULATORY GUIDE 1.73 REVISION 0 DATED 1/74

Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

DISCUSSION:

For the Westinghouse-supplied, safety-related, motor-operated valves located inside the containment, environmental qualification is discussed in Section 3.11(N). Auxiliary safety-related equipment (e.g., stem-mounted limit switches) is qualified separately. The conditions to which the equipment must be qualified (temperature, pressure, radiation, and chemistry) are those specified in Section 3.11(B). (Also see IEEE 382-1972.)

The balance-of-plant implementation of this regulatory guide is discussed in Section 3.11(B).2.1.

REGULATORY GUIDE 1.74

Quality Assurance Terms and Definitions

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.75 REVISION 2 DATED 9/78

Physical Independence of Electric Systems

DISCUSSION:

Westinghouse-furnished systems meet the recommendations of this regulatory guide in accordance with the comments of Section 7.1.2.2.1.

The balance of plant meets the recommendations of this regulatory guide. Refer to Section 8.1.4.3.

REGULATORY GUIDE 1.76 REVISION 0 DATED 4/74

Design Basis Tornado for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 3.3.2.

REGULATORY GUIDE 1.77 REVISION 0 DATED 5/74

Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors

DISCUSSION:

Westinghouse methods and criteria are documented in WCAP-7588, Revision 1A, which has been reviewed and accepted by the NRC.

The results of the Westinghouse analyses show agreement with Regulatory Positions C.1 and C.3. In addition, Westinghouse utilizes the assumptions given in Appendices A and B of the Regulatory Guide. However, Westinghouse takes exception to Regulatory Position C.2 which implies that the rod ejection accident should be considered as an emergency condition. Westinghouse considers this a faulted condition as stated in ANSI N18.2. Faulted condition stress limits are applied for this accident.

REGULATORY GUIDE 1.78 REVISION 0 DATED 6/74

Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.4-1.

REGULATORY GUIDE 1.79 REVISION 1 DATED 9/75

Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors

DISCUSSION:

The preoperational testing procedures comply with the positions in the guide with the following exceptions and clarifications:

C.1.b(2) Low-Pressure Safety Injection (LPSI) Recirculation Test (Cold Conditions)

The objective of this test is to demonstrate the capability to realign the valves and injection pumps to recirculate coolant from the containment floor or sump into the reactor coolant system. The testing should verify that the available net positive suction head is greater than that required at accident temperatures, as discussed in Regulatory Guide 1.1. The testing should include taking suction from the sump to verify (1) vortex control and (2) acceptable pressure drops across screening and suction lines and valves.

The test program meets the objective of this section, with the following clarification.

a. The ability to realign system valves is verified in preoperation test S-03EJ01, Residual Heat Removal System Cold Pre-Operational Test.

b. Verification of vortex control and acceptable pressure drops across the screening has been determined by hydraulic model testing. A geometric replica of the 90° sector of the reactor containment floor centered on the two sumps has been built to a scale of about 1:2.9. Testing included a variety of approach flow conditions, screen blockages, water levels, and pump operation combinations. The testing has been conducted by Alden Research Laboratory, which has previously demonstrated the validity of such testing. Verification of pressure drops across suction lines and valves has been accomplished using standard engineering calculations.

C.1.c(1) Core Flooding Flow Test (Cold Conditions)

The test program meets the intent of Regulatory Guide 1.79 by demonstrating proper system actuation and by verifying that the flow rate is as expected for the test conditions. To perform this test, the accumulators are filled to their normal level and pressurized, then discharged one at a time into an open reactor vessel by opening the motor-operated isolation valve. The discharge flow rate is calculated from measurements of the changes in accumulator water level as a function of time. Accumulator pressure and level are continuously recorded throughout the test. In the analysis of the data from this test, the accumulator valve opening time and valve characteristics are accounted for, ensuring that the valve operation does not influence the final results. This test has been conducted at similar plants with acceptable results, demonstrating that the current test program accurately provides verification of proper system actuation and required flow rates.

REGULATORY GUIDE 1.80 REVISION 0 DATED 6/74

Preoperational Testing of Instrument Air Systems

DISCUSSION:

The section describing commitments to Regulatory Guide 1.80 for the Callaway Instrument Air System Preoperational Test has been deleted. The deleted material is contained in the FSAR on record as of the receipt of the Callaway Operating License No. NPF-30 on October 18, 1984. This information will not be reproduced in later revisions to the FSAR due to the historical status of the content. This information may be provided, however, upon request from the Union Electric Nuclear Licensing Department.

REGULATORY GUIDE 1.81 REVISION 1 DATED 1/75

Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants

DISCUSSION:

This regulatory guide is not applicable since the Callaway Plant is a single-unit site.

REGULATORY GUIDE 1.82 REVISION 0 DATED 6/74

Sumps for Emergency Core Cooling and Containment Spray Systems

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.2.2-1.

REGULATORY GUIDE 1.83 | REVISION N/A | DATED N/A

Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes

DISCUSSION:

This regulatory guide has been withdrawn by the NRC. The S/G tube inservice inspection program is discussed in Section 5.4.2.4 and in the Callaway Technical Specifications.

REGULATORY GUIDE 1.84 | REVISION 26 | DATED 7/89

Design and Fabrication Code Case Acceptability-ASME Section III Division 1

DISCUSSION:

Regulatory Guides 1.84 and 1.85 are periodically revised to incorporate new code cases and revisions to existing code cases. Union Electric will review the revisions to these regulatory guides and comply with the most current revisions of these regulatory guides, as described below.

For components supplied with the NSSS, the following discussion applies:

a. Union Electric controls its suppliers to:

    1. Limit the use of code cases to those listed in Regulatory Position C.1 of the Regulatory Guides 1.84 and 1.85 revisions in effect at the time the equipment is ordered, except as allowed in item b. below.

    2. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the Regulatory Guides 1.84 and 1.85 revisions in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.

    3. Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

b. Union Electric seeks NRC permission for the use of Class 1 code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revisions in effect at the time the equipment is ordered and permits supplier use if NRC permission is obtained or is otherwise assured (e.g., a later version of the regulatory guide includes endorsement).

For components not supplied with the NSSS, the requirements of Regulatory Guides 1.84 and 1.85 are met with the following clarifications:

a. Components ordered to a specific version of a code case need not be changed because a subsequent revision to the code case is listed as the approved version in the current revision of the Regulatory Guide.

b. Components ordered to a code case that was previously approved for use need not be changed because the code case is listed as annulled in the current revision of the Regulatory Guide.

REGULATORY GUIDE 1.85 REVISION 26 DATED 7/89

Materials Code Case Acceptability-ASME Section III Division 1

DISCUSSION:

Refer to the discussion of Regulatory Guide 1.84.

REGULATORY GUIDE 1.86

Termination of Operating Licenses for Nuclear Reactors

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.87 REVISION 1 DATED 6/75

Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)

DISCUSSION:

The recommendations of this regulatory guide are not applicable to the SNUPPS application.

REGULATORY GUIDE 1.88

Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.89 REVISION 0 DATED 11/74

Qualification of Class 1E Equipment for Nuclear Power Plants

DISCUSSION:

For Westinghouse nuclear steam supply system Class 1E equipment, Westinghouse meets IEEE Standard 323-1974 (including the IEEE Standard 323a-1975 position statement of July 24, 1975) and this regulatory guide by an appropriate combination of any or all of the following: type testing, operating experience, and qualification by analysis. This commitment will be satisfied by implementation of the final NRC-approved version of WCAP-8587, Revision 6A, as discussed in Section 3.11(N).

The recommendations of this regulatory guide are met for the balance-of-plant systems and components. However, as supported by the statement of consideration for 10 CFR 50.49 (Federal Register, Volume 48, P2731, January 21, 1983), the recommendations of this regulatory guide need not be applied, in total, for Class 1E NSSS and BOP equipment located in a mild environment.

Safety-related mechanical and electrical equipment that is exempt from EQ (Category C) or is located in mild environment areas of Callaway (Category D) will be qualified as follows:

A. ENVIRONMENTAL QUALIFICATION

1. Safety-related mechanical and electrical equipment that is exempt from EQ or is located in mild environment areas shall be qualified by either:

   a. The guidelines of IEEE-323-1974, except that the equipment need not be replaced at the end of its qualified life established by artificial aging techniques (thermal, irradiation, cyclic, and vibrational), provided that the program elements of (d) below are followed, or

   b. An equipment specification, supported by a certificate of compliance, which documents that the equipment is designed for the range of normal and expected extremes (i.e., abnormal range) of environmental conditions postulated to occur at the equipment location, provided that the program elements of (d) below are followed, or

   c. For equipment procured as commercial grade and upgraded for safety-related use, an evaluation of equipment design which documents that the equipment is capable of operating in the range of normal and expected extremes of environmental conditions postulated to occur at the equipment location, and

d. Implementation of the following program elements:

(1). Periodic maintenance, inspection, and/or replacement requirements for equipment based on sound engineering practice and recommendations of the equipment manufacturer, which are updated as required by the results of equipment surveillance requirements;

(2). Periodic testing requirements to verify operability of safety-related equipment within its performance requirements. Testing requirements will be based on sound engineering practice and recommendations of the equipment manufacturer or, where applicable, in accordance with plant Technical Specifications;

(3). Equipment surveillance requirements, which include analysis of equipment and component failures and a review of preventive maintenance and periodic testing results; and

(4). The effects of aging must be addressed on an equipment-specific basis, especially if any of the following equipment or devices are involved which have demonstrated a possible link between aging and seismic/EQ performance:

i) lead storage batteries

ii) relays (including time delay relays)

iii) rotary, pressure, limit, and snap-acting switches

iv) contactors (motor starters)

v) motors with elastomeric mounting bushings

vi) electrolytic capacitors

vii) NIS detectors

2. The above equipment-specific qualification methods should take into consideration, especially during commercial grade dedications, the complexity of the equipment in question and the requisite critical characteristics (i.e., a resistor would not be subject to the same scrutiny as an assembly such as a motor). Engineering must approve the commercial grade dedication of the above equipment types (i - vii), including the requirements for aging (thermal, irradiation, cyclic, and vibrational) and seismic testing of equipment. In this regard, Union Electric assumes the

responsibility of compliance with IEEE-323-1974 (especially Sections 6.3.3 and 6.8) and IEEE-344-1975, within the above clarifications, for exempt and mild environment Class 1E equipment not subjected to the entire sequence of testing outlined in IEEE-323-1974.

B. SEISMIC QUALIFICATION

1. Safety-related mechanical and electrical equipment that is exempt from EQ or is located in mild environment areas (Category C or D) shall be qualified by:

   a. The guidelines of IEEE-344-1975, including an equipment-specific consideration of the effects of aging, provided that the program elements of A.1.d above are followed, or

   b. For equipment procured as commercial grade and upgraded for safety-related use, an evaluation of equipment design in accordance with utility procedures for upgrading commercial equipment, provided that the program elements of A.1.d above are followed.

<u>REGULATORY GUIDE 1.90</u> <u>REVISION 1</u> <u>DATED 8/77</u>

Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons

DISCUSSION:

The recommendations of this regulatory guide are not applicable to the Callaway application, since the containment design does not utilize grouted tendons.

<u>REGULATORY GUIDE 1.91</u>

Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

<u>REGULATORY GUIDE 1.92</u> <u>REVISION 1</u> <u>DATED 2/76</u>

Combining Modal Responses and Spatial Components in Seismic Response Analysis

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Sections 3.7(B).2.6, 3.7(B).2.7, 3.7(B).3.6, 3.7(B).3.7, 3.7(N).2.6, 3.7(N).2.7, 3.7(N).3.6, and 3.7(N).3.7.

REGULATORY GUIDE 1.93 REVISION 0 DATED 12/74

Availability of Electric Power Sources

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Technical Specifications.

REGULATORY GUIDE 1.94

Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.95 REVISION 1 DATED 1/77

Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.4-2.

REGULATORY GUIDE 1.96 REVISION 1 DATED 6/76

Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.97 REVISION 2 DATED 12/80

Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

DISCUSSION:

The recommendations of this regulatory guide are discussed in Appendix 7A.

REGULATORY GUIDE 1.98 REVISION 0 DATED 3/76

Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor

DISCUSSION:

The recommendations of this regulatory guide are not applicable to a PWR.

REGULATORY GUIDE 1.99 REVISION 2 DATED 5/88

Radiation Embrittlement of Reactor Vessel Materials.

DISCUSSION:

The reactor vessel material meets the end-of-life reference criterion of this regulatory guide.

Recent surveillance capsule data from the Callaway reactor vessel indicates a steady state condition of radiation damage well below that predicted by this regulatory guide. Therefore, the recommendations of this regulatory guide are met as discussed in Section 5.3.2.

REGULATORY GUIDE 1.100 REVISION 1 DATED 8/77

Seismic Qualification of Electric Equipment for Nuclear Power Plants

DISCUSSION:

The Westinghouse program for seismic qualification of safety-related electric equipment is discussed in Section 3.10(N). The balance-of-plant implementation of this regulatory guide is discussed in Section 3.10(B). Certain provisions of IEEE 344-1987, as approved by Revision 2 of this regulatory guide, may be used if approved on a case by case basis (e.g., the acceptance of non-enveloped Required Response Spectra). See also the position on Regulatory Guide 1.89.

REGULATORY GUIDE 1.101 REVISION NA DATED NA

Emergency Planning for Nuclear Power Plants

DISCUSSION:

This regulatory guide has been withdrawn by the NRC.

REGULATORY GUIDE 1.102 REVISION 1 DATED 9/76

Flood Protection for Nuclear Power Plants

DISCUSSION:

In regard to Position C.3, the roofs of the Standard Plant seismic Category I structures have no parapets or any other similar features that would induce loading in excess of the design basis in the event that the roof drains could not discharge the maximum precipitation intensities of the PMP.

The Callaway Plant is above the PMF level as discussed in Section 3.4.1.

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.103 REVISION 1 DATED 10/76

Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments

DISCUSSION:

The recommendations of this regulatory guide are met. The post-tensioned prestressing system described in Section 3.8.1 has been reviewed and approved by the NRC in previous plant applications.

REGULATORY GUIDE 1.104 REVISION 0 DATED 2/76

Overhead Crane Handling Systems for Nuclear Power Plants

DISCUSSION:

This regulatory guide has been withdrawn by the NRC.

REGULATORY GUIDE 1.105 REVISION 1 DATED 11/76

Instrument Setpoints

DISCUSSION:

For instrumentation not provided with the NSSS, the recommendations of this regulatory guide are met as described in Table 7.1-6.

For the instrumentation provided with the NSSS, the recommendations of this regulatory guide are met as described below.

Westinghouse setpoint studies performed with the original steam generators provide an allowance from the nominal trip setpoint to the technical specification allowable value (AV) to account for drift when measured at the rack during periodic testing, rack calibration accuracy, and rack comparator setting accuracy. The difference between the nominal trip setpoint (NTS) and the safety analysis limit includes the following items: a) the inaccuracy of the instrument (sensor temperature and pressure effects, sensor reference accuracy, sensor drift), b) process measurement accuracy, c) uncertainties in the sensor calibration, d) the potential transient overshoot determined in the accident analyses (this primary element accuracy may include compensation for the dynamic effect), e) environmental effects on equipment accuracy caused by postulated or limiting postulated events (only for those systems required to mitigate consequences of an accident), f) rack temperature error, g) rack and sensor M&TE errors, and h) the rack error terms discussed above (between the NTS and AV). Designers choose setpoints such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis.

Westinghouse setpoint studies performed for the replacement steam generators (RSGs) provide an allowance from the nominal trip setpoint to the technical specification allowable value to account only for rack calibration accuracy. The difference between the nominal trip setpoints for reactor trips and ESF actuations started by SG water level low-low, SG water level high-high, and low steamline pressure and their safety analysis limits includes the same error terms discussed above. The “Nominal Trip Setpoints and Allowable Values” section in the Background Bases for Technical Specifications 3.3.1 and 3.3.2 discuss some differences between the pre-RSG and post-RSG setpoint methodologies, but the major difference is the tightening of the band between the NTS and the AV for the above RTS and ESFAS functions. Designers choose setpoints, such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis.

The range of instruments is chosen, based on the span necessary for the instrument's function. Narrow range instruments will be used where necessary. Instruments will be selected, based on expected environmental and accident conditions. The need for qualification testing will be evaluated and justified on a case basis.

Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism, so that no integral setpoint securing device is required. Integral setpoint locking devices will not be supplied.

The assumptions used in selecting the setpoint values in Regulatory Position C.1, and the minimum margin with respect to the safety analysis limit and calibration uncertainty will be documented by Westinghouse. Drift rates and their relationship to testing intervals will not be documented by Westinghouse.

<u>REGULATORY GUIDE 1.106</u> <u>REVISION 1</u> <u>DATED 3/77</u>

Thermal Overload Protection for Electric Motors on Motor-Operated Valves

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 8.3.1.1.2. This regulatory guide does not apply to the auxiliary feedwater control valves (ALHV0005, -7, -9, and -11) each of which has its thermal overload protection device(s) located in the valve assembly.

REGULATORY GUIDE 1.107 REVISION 1 DATED 2/77

Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures

DISCUSSION:

The recommendations of this regulatory guide are not applicable to the SNUPPS application, since a prestressing system using ungrouted tendons is used.

REGULATORY GUIDE 1.108 REVISION 1 DATED 8/77

Periodic Testing of Diesel Generator Units Used As Onsite Electric Power Systems at Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide were met with regard to the periodic testing of standby diesel generators at Callaway Plant following initial licensing of the facility. However, since the issuance of Revision 3 of Regulatory Guide 1.9 and adoption of the Improved (NUREG-1431 based) Technical Specifications at Callaway Plant per License Amendment 133 (in 1999), periodic testing of the standby diesel generators has been based on the requirements and/or recommendations of those documents in lieu of Regulatory Guide 1.108. Further, Regulatory Guide 1.108 was withdrawn by the NRC in 1993 (58 FR 41813, 8/5/93) in light of the guidance provided in Regulatory 1.9, Revision 3 wich largely incorporated and superseded the guidance of Regulatory Guide 1.108.

REGULATORY GUIDE 1.109

Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.110 REVISION 0 DATED 3/76

Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors

DISCUSSION:

During the construction permit stage, the radwaste systems and equipment were demonstrated to have satisfied the Guides on Design Objectives (RM-50-2), hence no cost-benefit analysis is required.

REGULATORY GUIDE 1.111

Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.112 REVISION 0-R DATED 5/77

Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors

DISCUSSION:

The recommendations of this regulatory guide were met as described in the historical data contained in Table 11.1-3. Current methodology for calculating releases is maintained in the ODCM.

REGULATORY GUIDE 1.113

Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.114

Guidance on Being Operator at the Controls of a Nuclear Power Plant

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.115 REVISION 1 DATED 7/77

Protection Against Low-Trajectory Turbine Missiles

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 3.5.

REGULATORY GUIDE 1.116

Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.117 REVISION 1 DATED 4/78

Tornado Design Classification

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 3.3. Administrative controls for opening and closing missile shields/doors may be utilized for some locations/ applications to assure missile protection is provided when required.

REGULATORY GUIDE 1.118 REVISION 2 DATED 6/78

Periodic Testing of Electric Power and Protection Systems

DISCUSSION:

For the systems not provided with the NSSS, the recommendations of this regulatory guide are met as described in Table 7.1-7.

For systems provided with the NSSS, Westinghouse follows the recommendations of the regulatory guide with the following exceptions:

Westinghouse defines "Protective Action Systems" to mean the electric instrumentation and controls portions of those protection systems and equipment actuated and controlled by the protection system.

Equipment performing control functions, but actuated from protection system sensors, is not part of the safety system and will not be tested for time response. Status, annunciating, display, and monitoring functions, except those related to the post-accident monitoring system (PAMS), are considered by Westinghouse to be control functions. Reasonability checks, i.e., comparison between or among similar such display functions, will be made.

Response time testing for control functions operated from system sensors will not be performed. Moreover, NIS detectors will not be tested for time response, since their worst case response time is not a significant fraction of the total overall system response (i.e., less than 5 percent). Despite the fact that this exemption is no longer permitted by IEEE-338 (1977 version), Westinghouse believes that it is valid.

Refer to Section 7.1.2.6.2 for additional discussions on response time testing of protection sensors.

REGULATORY GUIDE 1.119 REVISION NA DATED NA

Surveillance Programs for New Fuel Assembly Designs

DISCUSSION:

This regulatory guide has been withdrawn by the NRC.

REGULATORY GUIDE 1.120 REVISION 1 DATED 11/77

Fire Protection Guidelines for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are not applicable due to the adoption of NFPA 805 Fire Protection Standard.

REGULATORY GUIDE 1.121 REVISION 0 DATED 8/76

Bases for Plugging Degraded PWR Steam Generator Tubes

DISCUSSION:

Position C.1: The term "unacceptable defects" is interpreted to apply to those imperfections resulting from service-induced mechanical or chemical degradation of the tube walls which have penetrated to a depth in excess of the plugging limit.

Position C.2a(2) and C.2.a(4): Ameren has incorporated the structural integrity performance criteria as specified in the Callaway Technical Specifications. The structural integrity performance criteria incorporates and supplements the recommendations of this regulatory guide. Callaway Plant has committed to ensuring a margin of 3 against tube failure for normal operation.

Position C.2.b: In cases where sufficient inspection data exist to establish a degradation allowance, the rate used will be an average time-rate determined from the mean of the test data.

Positions C.3.d(1) and C.3.d(3): The combined effect of these requirements would be to establish a maximum permissible primary-to-secondary leak rate which may be below the threshold of detection with current methods of measurement. Areva has determined the maximum acceptable length of a through-wall-crack, based on 3 times the normal operating pressure loads. A leak rate associated with the crack size determined on the basis of accident loadings will be used.

Position C.3.e(6): Computer code names and references will be supplied rather than the actual codes.

Position C.3.f: A minimum acceptable tube wall thickness (plugging/sleeving limit), based on structural requirements and consideration of loadings, measurement accuracy, and, where applicable, a degradation allowance, has been established as discussed in this position and in accordance with the general intent of this guide. The analysis used to determine this value is presented in NFEMG DC30

REGULATORY GUIDE 1.122 REVISION 1 DATED 2/78

Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

DISCUSSION:

Regulatory Guide 1.122 states that peaks in floor response spectra associated with structural frequencies should be broadened. The amount of broadening required is equal to ±15 percent of the peak frequencies, unless a smaller amount (greater than or equal to ±10 percent) is justified. The floor response spectra generated for the SNUPPS Project were broadened ±10 percent of all frequencies. Paragraph II.2.b of Section 3.7.1 of the Standard Review Plan permits broadening by only ±10 percent as long as the time history analyses, from which the spectra are generated, explicitly account for the effect of soil property variation.

Since this project, by its multiple site criteria (three or four site enveloping), has accounted for variation in soil properties in its analysis, use of floor response spectra broadened by ±10 percent is acceptable.

REGULATORY GUIDE 1.123

Quality Assurance Requirements for Control of Procurement ofItems and Services for Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.124 REVISION 1 DATED 1/78

Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports

DISCUSSION:

According to the NRC implementation guidance for this regulatory guide, it is not applicable to the SNUPPS units. However, the following discussion is provided for information purposes.

For ASME Section III components not supplied with the NSSS, the recommendations of this regulatory guide are met as discussed in Table 3.9(B)-14.

The Westinghouse position with respect to this regulatory guide is as follows.

a. The Regulatory Guide states in Paragraph B.1(b): "Allowable service limits for bolted connections are derived from tensile and shear stress limits and their nonlinear interaction; they also change with the size of the bolt. For this reason, the increases permitted by NF-3231.1, XVII-2110(a), and F-1370(a) of Section III are not directly applicable to allowable shear stresses and allowable stresses for bolts and bolted connections," and in Paragraph C.4: "This increase of level A or B service limits does not apply to limits for bolted connections."

   As noted above, the increase in bolt allowable stress under emergency and faulted conditions is not permitted. Westinghouse believes that the present ASME Code rules are adequate for bolted connections. This position is based on the following:

   It is recognized after extensive experimental work by several researchers that the interaction curve between the shear and tension stress in bolts is more closely represented by an ellipse and not a line. This has been clearly recognized by the ASME. Code Case 1644-6 specifies stress limits for bolts and represents this tension/shear relationship as a nonlinear interaction equation (incorporated into ASME III, Appendix XVII via the Winter 77 Addenda) and has a built-in safety factor that ranges between 2 and 3 (depending on whether the bolt load is predominantly tension or shear) based on the actual strength of the bolt as determined by test (Ref: "Guide to Design Criteria of Bolted and Riveted Joints," Fisher and Struik, copyright 1974, John Wiley and Sons, Page 54).

   Study of three interaction curves of allowable tension and shear stress based on the ASME Code (emergency condition allowables per XVII-2110 and faulted condition allowables per F-1370) and the ultimate tensile and shear strength of bolts (obtained from experimental work published by E. Chesson, Jr., N. L. Faustino, and W. H. Munse, "High Strength Bolts

Subjected To Tension and Shear," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, October 1965, Pages 155-180) indicates that there is adequate safety margin between the emergency and faulted condition allowables and failure of the bolts.

During their tests to determine the strength and behavior characteristics of single high strength bolts subjected to various combinations of tension and shear (T-S), Chesson, et. al. used a total of 115 bolts to ASTM Specificaions A 325-61T and A 354-Grade BC. The A 325-61T, which is a medium carbon steel, had a yield point of 77,000 psi to 88,000 psi and ultimate strength of 105,000 psi to 120,000 psi, depending upon the bolt diameter. The A 354-Grade BC, which is a heat treated carbon steel, had a yield point of 99,000 psig to 109,000 psi and ultimate strength from 115,000 psi to 125,000 psi, again depending upon the bolt diameter.

Figure 3A-1 shows the interaction curves for T-S loads on SA-325 bolts. Curve (1) represents the interaction relation (ellipse) permitted by Code Case 1644 (ASME III, Appendix XVII Winter 77 Addenda) for service levels A, B, and design condition. Curve (2) represents the interaction curve which considers the Code Case 1644 allowables and the increase permitted by XVII-2110(a) for service level C. Curve (3) represents the interaction curve which considers the Code Case 1644 allowables and the increase permitted by F-1370(a) for service level D. Curve (3) is the upper limit of the allowable stresses.

The design stress limits represented by Curves 1, 2, and 3 for A 325 bolts are then compared against the ultimate strength of the bolts represented by Curve 4, which is based on Chesson's test results. The area between Curve 3 and Curve 4 is the safety margin between the maximum bolt stress under service level D and minimum ultimate strength of the bolt.

Factor of safety against failure for A 325 bolts for various T-S ratios is shown in Figure 3A-2. The safety factor varies between a minimum of 1.36 and a maximum of 2.29, depending upon the value of T-S ratio. This is based upon the ultimate strength of the bolts from Chesson's test and the allowables obtained from Code Case 1644 and the increase permitted by F-1370(a) for service level D. Figure 3A-2 demonstrates that there exists an adequate factor of safety for the complete range of T-S loadings.

From this study it is observed that:

1. For the emergency condition, the safety factor (ratio of ultimate strength to allowable stress) varies between a minimum of 1.63 and a maximum of 2.73, depending upon the actual tensile stress/shear stress (T/S) ratio on the bolt.

2. For the faulted condition, the safety factor varies between a minimum of 1.36 to a maximum of 2.29, again depending upon actual T/S ratio on the bolt.

It is thus reasonable to allow an increase in these limits for the emergency and faulted conditions.

The Westinghouse design of component supports restricts the use of bolting material to the following applications:

1. Westinghouse design uses bolting predominantly in tension. Oversized holes are generally provided, and a mechanism other than the bolts is provided to take any shear loads. Shear or shear and tension interaction occur only in isolated locations.

2. Westinghouse bolts are limited to the following materials: A490, SA-354, SA-325, SA-540.

3. The diameters used range between 1/2 inches and 3 inches.

For the emergency condition, Westinghouse will use allowable bolt stresses specified in Code Case 1644, as increased according to the provisions of XVII-2110(a).

For the faulted condition, tensile loads in the bolts shall be limited to 0.7 Su, but not to exceed in any case 0.9 Sy. The allowables are taken at temperature. In those few cases where bolts are used in shear or tension and shear, ASME Code Appendix XVII - 2460 Requirements will apply with an increase factor that is defined in Regulatory Guide 1.124 or in Appendix F-1370, whichever is more restrictive. This provides an adequate margin of safety for the Westinghouse design.

b. In Paragraphs B.5 and C.8 of the Regulatory Guide, Westinghouse takes exception to the requirement that systems whose safety-related function occurs during emergency or faulted plant conditions, must meet level B limits. The reduction of allowable stresses to no greater than level B limits (which in reality are design limits since design, level A, and level B limits are the same for linear supports) for support structures in those systems with safety-related functions occurring during emergency or faulted plant conditions is overly conservative. The primary concern is that the system remains capable of performing its safety function. For active components, this is accomplished through the operability program, as discussed in Section 3.9(N).3.2.

c. Paragraph C.6(a) of the Regulatory Guide appears confusing as to what stress limits may be increased for the emergency condition. Westinghouse

will interpret this paragraph as follows: "The stress limits of XVII-2000 of Section III and Regulatory Position 3, increased according to the provisions of XVII-2110(a) of Section III and Regulatory Position 4, should not be exceeded for component supports designed by the linear elastic analysis method."

d. The method described in Paragraph C.7(b) of the Regulatory Guide is overly conservative and inconsistent with the stress limits presented in Appendix F. Westinghouse will use the provisions of F-1370(d) to determine service level D allowable loads for supports designed by the load rating method.

REGULATORY GUIDE 1.125

Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.126 REVISION 1 DATED 3/78

An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification

DISCUSSION:

The fuel densification model used for the SNUPPS units is presented in Section 2-2 of “Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations,” WCAP-10851-P-A, August 1988, which has been approved by NRC.

REGULATORY GUIDE 1.127

Inspection of Water-Control Structures Associated with Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.128 REVISION 1 DATED 10/78

Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to Section 8.3.2.2.1.

REGULATORY GUIDE 1.129 REVISION 1 DATED 2/78

Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

DISCUSSION:

This regulatory guide is based on IEEE Standard 450-1975. Amendment 133 to the Callaway Operating License converted the plant Technical Specifications to the improved Technical Specifications (ITS) based on NUREG-1431, "Standard Technical Specifications [STS], Westinghouse Plants, " Revision 1. As part of that amendment, the Bases for the Technical Specifications pertaining to the maintenance and testing of the station batteries were revised to refer to IEEE Standard 450-1995, consistent with NUREG-1431. Accordingly, the guidance of the 1995 version of IEEE Standard 450 is followed in lieu of the 1975 version. Refer to FSAR Sections 8.3.2.1.2 and 8.3.2.2.1.

REGULATORY GUIDE 1.130 REVISION 1 DATED 10/78

Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports

DISCUSSION:

According to the NRC implementation guidance for this regulatory guide, it is not applicable to the SNUPPS units. However, the following discussion is provided for information purposes.

For ASME Section III components not furnished with the NSSS, the Class 1 supports are of the linear type and not the plate and shell type. Therefore, this regulatory guide does not apply.

The Westinghouse position with respect to this regulatory guide is as follows.

a. Paragraph B.1 states that increases are not allowed for bolted connections for emergency and faulted conditions. The Westinghouse position is that it is reasonable to allow an increase in the limits for bolted connections for these conditions. Further justification concerning this position can be found in Item 1 of the discussion on Regulatory Guide 1.124.

b. The method described in Paragraph C.7(b) of the Regulatory Guide is overly conservative and inconsistent with the stress limits presented in Appendix F. Westinghouse will use the provisions of F-1370(d) to determine service level D allowable loads for supports designed by the load rating method.

REGULATORY GUIDE 1.131 REVISION 0 DATED 8/77

Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met with the exceptions noted in Section 8.1.4.3.

REGULATORY GUIDE 1.132

Site Investigations for Foundations of Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.133 REVISION 1 DATED 5/81

Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors

DISCUSSION:

The recommendations of this regulatory guide are met as described in FSAR Section 4.4.6.4 with the following exceptions:

Regulatory Position C.5, "Technical Specification for the Loose-Part Detection System" recommended that Technical Specifications (T/S) should be provided. Since issue of the Regulatory Guide, the NRC has provided additional guidance and clarification in 10 CFR 50.36 for information that is required to be placed in a plant's T/S. Using this criteria, the existing T/S for the Loose Parts Detection System (LPDS) was eliminated. However guidance was maintained and placed in FSAR Section 16.3.3.5.

Regulatory Position C.1.f, "Capability for sensor Channel Operability Tests" and associated footnote 2 discuss the types of periodic testing performed and provides definitions for those tests. The terminology and test descriptions were originally those used in the Standard Technical Specifications. Callaway has implemented the Improved Technical Specifications (ITS), which modified the terminology used for these tests. Terminology similar to ITS has been implemented in FSAR Section 16 Specifications. This change eliminated Channel Functional Tests and replaced them with Channel Operational Tests, which have a slightly different scope.

Regulatory Position C.3, "Using Data Acquisition Modes" Section a.2 and a.3 describe periodic testing to be performed on the LPDS. Callaway has adopted a modified

methodology for performing the 18-month channel calibration recommended in C.3.a.3. FSAR Section 16.3.3.5 describes Callaway's program for performing this testing.

REGULATORY GUIDE 1.134

Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.136 REVISION 1 DATED 10/78

Material for Concrete Containments

DISCUSSION:

The recommendations of Section 3.8.1.6 are used in lieu of the recommendations of this regulatory guide, which generally endorses ACI Standard 359-74.

REGULATORY GUIDE 1.137 REVISION 0 DATED 1/78

Fuel-Oil Systems for Standby Diesel Generators

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 9.5.4-3.

REGULATORY GUIDE 1.138

Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A of the Site Addendum.

REGULATORY GUIDE 1.139 REVISION 1, Draft 2 DATED 2/80

Guidance for Residual Heat Removal

DISCUSSION:

The recommendations of this regulatory guide are met as described in Appendix 5.4A.

REGULATORY GUIDE 1.140 REVISION 1 DATED 10/79

Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 9.4-3.

Prefilter design, construction and testing referenced in Table 9.4-3 paragraph 3.m is in accordance with Regulatory Guide 1.140 Revision 2 dated June 2001.

REGULATORY GUIDE 1.141 REVISION 0 DATED 4/78

Containment Isolation Provisions for Fluid Systems

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 6.2.4-2.

REGULATORY GUIDE 1.142 REVISION 0 DATED 4/78

Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)

DISCUSSION:

The recommendations of this regulatory guide, which generally endorses ACI-349-76, have not been applied to the design of safety-related concrete structures of the power block. The procedures and requirements described in ACI 318-71, Building Code Requirements for Reinforced Concrete, along with the exceptions, clarifications, and additions described in Sections 3.8.3, 3.8.4, and 3.8.5, have been used instead.

REGULATORY GUIDE 1.143 REVISION 0 DATED 7/78

Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met as described in Table 3.2-5.

REGULATORY GUIDE 1.144

Auditing of Quality Assurance Programs for Nuclear Power Plants

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.145

Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

DISCUSSION:

Refer to Appendix 3A in the Site Addendum.

REGULATORY GUIDE 1.146

Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants.

DISCUSSION:

Refer to the Union Electric Company Operational Quality Assurance Manual.

REGULATORY GUIDE 1.147

Inservice Inspection Code Case Acceptability, ASME Section XI Division I

DISCUSSION:

Regulatory Guide 1.147 is periodically revised to incorporate new code cases and revisions to existing code cases. Union Electric will review the revisions to this Regulatory Guide and may utilize those code cases which have been approved in revisions of the Regulatory Guide up to and including the most recent revision. Utilized code cases will be identified in the ISI Program Plan.

For inspection performed by suppliers, the following discussion applies:

a. Union Electric controls its suppliers to:

   1. Limit the use of code cases to those listed in Regulatory Position C.1 of the Regulatory Guide 1.147 revision in effect at the time the inspection is planned or performed, except as allowed in item b below.

   2. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the Regulatory Guide 1.147 revision in effect at the time the inspection is planned or performed, where use of such code cases is needed.

b. Union Electric will seek NRC permission for the use of code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the Regulatory Guide 1.147 revision in effect at the time the inspection is planned or performed, and permits supplier use if NRC permission is obtained or otherwise assured (e.g., a later version of the regulatory guide includes endorsement).

REGULATORY GUIDE 1.150 REVISION 1 DATED 2/83

Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations

DISCUSSION:

The recommendations of this regulatory guide are met as described in Section 5.2.4.

REGULATORY GUIDE 1.152 REVISION 1 DATED 1/96

Criteria for Digital Computers in Safety Systems of Nuclear Power Plants

DISCUSSION:

UE complies with the recommendations of this regulatory guide for new and/or replacement safety related digital equipment and for changes to existing safety-related digital equipment after Refuel 8 (Fall 1996).

REGULATORY GUIDE 1.155 INITIAL ISSUE (REISSUED)DATED 8/88

Station Blackout (Endorses NUMARC 87-00, dated 11/87)

DISCUSSION:

UE complies with the recommendations of this Regulatory Guide with the following clarifications:

In lieu of the guidance given in sections C.1.0 through C.3.4, UE shall follow the guidance contained in NUMARC 87-00 as endorsed by this Regulatory Guide and as described in Callaway's Final Safety Analysis Report (Section 8.3A).

Regulatory Position C.3.5 titled "Quality Assurance and Specification Guidance for Station Blackout Equipment That is Not Safety Related" is met as described in Callaway's Final Safety Analysis Report, Technical Specifications, and Operating Quality Assurance Manual.

REGULATORY GUIDE 1.158 REVISION 0 DATED 2/89

Qualification of safety-related lead storage batteries for nuclear power plants.

DISCUSSION:

The recommendations of this regulatory guide will be met if the safety-related lead storage batteries are replaced.

REGULATORY GUIDE 1.160 REVISION 2 DATED 3/97

Monitoring the effectiveness of maintenance at Nuclear Power Plants (Endorses NUMARC 93-01)

DISCUSSION:

UE complies with the requirements of this Regulatory Guide.

REGULATORY GUIDE 1.163 REVISION 0 DATED 9/95

Performance-Based Containment Leak-Test Program

DISCUSSION:

UE complies with the recommendations of this Regulatory Guide as discussed in the Leak Rate Test Program (ESP-SM-1001).

REGULATORY GUIDE 1.181 Initial Issue DATED 9/99

Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) (Endorses NEI 98-03)

DISCUSSION:

AmerenUE will comply with the recommendations of this Regulatory Guide as needed to improve the FSAR. Major changes, (for example removal/archiving of information and relocation of information) will be identified as exceptions to Regulatory Guide 1.70 as applicable. See Table 3A-1.

REGULATORY GUIDE 1.182 Initial Issue DATED 5/00

Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants

DISCUSSION:

AmerenUE complies with the requirements of this Regulatory Guide.

REGULATORY GUIDE 1.187 Initial Issue DATED 11/00

Guidance for Implementation of 10 CFR 50.59, Changes Tests and Experiments.

DISCUSSION:

AmerenUE complies with Regulatory Guide 1.187 with the following clarifications to NEI 96-07 Guidelines for 10 CFR 50.59 Implementation, dated November 2000:

1. With regard to Regulatory Position C.1 of Regulatory Guide 1.187, AmerenUE substitutes the word, "Implementation" for "Evaluations" to reflect title of NEI 96-07, dated November 2000.

2. NEI 96-07 Sections 1.2.4, 3.5: Compliance with RG 1.186 Guidance and Examples for Identifying 10 CFR 50.2 Design Bases, (NEI 97-04):

   AmerenUE has not established a position on Regulatory Guide 1.186

3. Regarding NEI 96-07, Section 4.1.2, Maintenance Activities:

   a. AmerenUE complies with the intent of this section with the understanding that Temporary Alterations Supporting Maintenance are those Alterations that meet the thresholds of the Temporary Procedure Change Program, the Temporary Modification Program or are under the scope of the Freeze Seal Control Program.

   b. Scaffold/Shielding: Prior to NEI 96-07, AmerenUE had in place programs to control the installation of scaffolding and shielding. These programs were developed per 10 CFR 50.59 so that erection of Scaffolds and Shielding in accordance with these programs would not require prior NRC Approval.

      Going forward to the guidance of NEI 96-07, these programs, when revised, will include a 10 CFR 50.59 Screening to ensure continued compliance with 10 CFR 50.59. Deviations from these established programs can be evaluated using 10 CFR 50.65 a(4) as allowed by the 50.59 Review Program. Hence, scaffolds and shielding meeting these program requirements are not required to undergo a 50.59 Review to remain in place over 90 days at power.

   c. AmerenUE's Procedurally Controlled Temporary Modification Program is a functionally equivalent program for controlling Temporary Alterations Supporting Maintenance that are imbedded in plant procedures.

4. Regarding NEI 96-07, Section 4.1.2, Maintenance Activities (Temporary Alterations Supporting Maintenance) and Section 4.4, Applying 10 CFR 50.59 to Compensatory Actions to Address Nonconforming or Degraded Conditions:

   a. Compensatory Actions are those considered to enter the 50.59 Review process at the threshold of the Temporary Procedure Change program or the Temporary Modification Program.

   b. Activities controlled by AmerenUE's Workman's Protection Program are not considered to be Temporary Alterations Supporting Maintenance or Compensatory Actions to Address Degraded or Nonconforming Conditions. This program meets the condition of 10 CFF 50.59 c(4), the regulation controlling these activities is 10 CFR 50 Appendix B Criteria XIV, Inspection Test and Operating Status.

   c. Prior to NEI 96-07, AmerenUE had in place programs to control the defeating of annunciators and computer points. These programs were developed per 10 CFR 50.59.

      Going forward to the guidance of NEI 96-07, these programs (stated in 4c) will continue to include a tie to the 10 CFR 50.59 Review Program through the Temporary Modification Program.

5. NEI 96-07 Section 4.3.8.2 Guidance for Changing from One Method of Evaluation to Another:

   AmerenUE compliance with this section is with the following clarification noting Regulatory Guide 1.187, Position C.2)

   Regarding the use of Generic Letter 83-11, Supplement 1 dated June 24, 1999, the following clarifies the guidance provided in Attachment 1 to GL 83-11 Supplement 1:

   5.1 1.0 INTRODUCTION

      No Clarifications.

   5.2 2.0 GUIDELINES

      AmerenUE will not backfit any NRC notifications regarding in-house calculations performed prior to implementation of NEI 96-07 Rev. 1. Additionally, since Generic Letter 83-11, Supplement 1 is discussed by a document endorsed by Regulatory Guide 1.187, the NRC has been implicitly notified of AmerenUE's application of the guidelines.

      1. 2.1 Eligibility

No Clarifications.

2. 2.2 Application Procedures

AmerenUE calculations are performed in accordance with approved procedural controls. Separate procedures will not be developed for each analysis topic, code, or method. Computer codes used for safety analysis will be controlled in accordance with AmerenUE's software control procedures.

3. 2.3 Training and Qualification of Licensee Personnel

Training and qualification of personnel performing Safety Analysis calculations will be accomplished in accordance with AmerenUE's Engineering Support Personnel training program.

4. 2.4 Comparison Calculations

Comparison and benchmark calculations will be performed in accordance with approved procedural controls. Computer codes used for safety analysis will be controlled in accordance with AmerenUE's software control procedures.

5. 2.5 Quality Assurance and Change Control

Safety Analysis calculations will be performed in accordance with the AmerenUE OQAP, which implements 10CFR50, Appendix B Criterion III. Computer codes used for safety analysis will be controlled in accordance with AmerenUE's software control procedures.

<u>REGULATORY GUIDE 1.</u>195 <u>REVISION 0</u> <u>DATED 5/03</u>

Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors

DISCUSSION:

The recommendations of this regulatory guide are met as described in the analysis of FSAR design basis accidents and their radiological consequences.

<u>REGULATORY GUIDE 1.205</u> <u>REVISION 01</u> <u>DATED 12/2009</u>

Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants

DISCUSSION:

The recommendations of this regulatory guide are met. Refer to the FSAR Section 9.5.1.

TABLE 3A-1 LISTING OF MAJOR USES OF NEI 98-03 AND EXCEPTIONS TO REGULATORY GUIDE 1.70

FSAR/Section/Table/Figure	Description of Change
Section 1.7.2	Adds a statement to the FSAR providing clarification to Callaway's commitment to Regulatory Guide 1.70 stating that the designation of valve and damper positions on P&ID's in the FSAR is beyond the level of detail needed in the FSAR.
Section 11.1	Adds a statement to the FSAR providing clarification to Callaway's commitment to Regulatory Guide 1.70 stating that Section 11.1 information will be designated as "historical" per NEI 98-03, Rev. 1.
Table 3B-1	Table 3B-1 is considered an historical example per Section 3B.1.
Section 1.9 Table 1.9-1 Table 1.9-2 Table 1.9-3 Table 1.9-4	The information contained in Section 1.9 and Tables 1.9-1 through 1.9-4 was determined to be historical per NEI 98-03, Rev. 1. The affected Section and Tables are identified as “Historical” in the revision identifier at the bottom of each page.
Figure 2.3-119	This Figure shows the construction excavation plan for the plant site. The information shown on this Figure was accurate at the time the plant was constructed, but is not intended or expected to be updated for the life of the plant. This information meets the NEI 98-03, Rev. 1 criteria for historical information and the Figure will be indentified as historical.
Appendix 3A	Replaced the revision date for Regulatory Guide (RG) 1.2 and RG 1.18 with “WITHDRAWN (Historical).”
Section 13.1.1.4 Section 13.1.3.2	The personal resumes contained in Section 13.1.1.4 and Section 13.1.3.2 were removed after being determined to be “excessive detail” per NEI 98-03, Rev. 1. The affected Sections state that personal resumes are maintained on file for review.

APPENDIX 3B - HAZARDS ANALYSIS

3B.1 INTRODUCTION

The SNUPPS powerblock has been designed to provide protection for safety-related equipment from hazards and events which could reasonably be expected to occur. This protection is provided to ensure that recovery from the event is possible, to ensure the integrity of the reactor coolant pressure boundary, to minimize the release of radioactivity, and to enable the plant to be placed in a safe condition.

This appendix provides the results of integrated hazards analyses for selected areas of the plant to demonstrate the type of analyses conducted for each safety-related area of the plant to ensure that the SNUPPS units can withstand the postulated events. Analyses are also provided for the effects of a pipe rupture in the main steam line compartment, the effects of pipe ruptures in the auxiliary feedwater pump rooms, and the effects of a circulating water pipe expansion joint rupture.

Table 3B-1 provides the details of a typical integrated hazards analysis using the 1974 elevation of the auxiliary building as an example. Since this table is intended only to show a typical hazards analysis, it will not be updated to reflect the as-built plant.

The items considered in the evaluation of each plant area include wind and tornadoes, floods, missiles, pipe breaks, fires, and seismic events. (Refer to Sections 3.3 through 3.7, 9.5, and Section 9.5.1.) Even though each area of the plant and each system is designed individually to properly consider the above events, an integrated analysis of rooms, systems, and events is performed to ensure that the above objectives are realized for each postulated event.

The hazards analyses are conducted on a room by room basis. All components within the room are reviewed for the effects of earthquake-induced failures, effects of high and moderate energy piping breaks (flooding, sprays, and jet impingement), and the effects of missiles.

The effects of the high energy breaks on equipment are reported in Section 3.6.2.5. Fire protection and the effects of fires in the various fire areas are discussed in Section 9.5.1.

3B.2 ANALYSIS ASSUMPTIONS

In the analysis of an event or hazard, it is assumed that the plant will be operated in accordance with the requirements of the Technical Specifications. Should the event result in a turbine or reactor trip, the plant will be placed in a hot standby condition. If required by a Limiting Condition of Operation or if recovery from the event will cause the plant to be shut down for an extended period of time, the plant will be taken to a cold shutdown condition. Safe shutdown is discussed in Appendix 5.4A.

During the hot standby condition, an adequate heat sink is provided to remove reactor core residual heat. Boration capability is provided to compensate for xenon decay and to maintain the required core shutdown margin. Boration is required within 25 hours after reactor shutdown to maintain the reactor in a hot standby condition.

Redundancy or diversity of systems and components is provided to enable continued operation at hot standby or to cool the reactor to a cold shutdown condition. If required, it is assumed that temporary repairs can be made to circumvent damages resulting from the hazard. Loss of offsite power is not assumed, unless a trip of the turbine generator system or the reactor protection system is a direct consequence of the hazard. All available systems, including nonsafety-related systems and those systems requiring operator action, may be employed to mitigate the consequences of the hazard.

In determining the availability of the systems required to mitigate the consequences of a hazard and those required to place the reactor in a safe condition, the direct consequences of the hazard are considered. The feasibility of carrying out operator actions are based on ample time and adequate access to the controls, motor control center, switchgear, etc., associated with the component required to accomplish the proposed action.

When the postulated hazard occurs in and results in damage to one of two or more redundant or diverse trains, single failures of components in other trains (and associated supporting trains) are not assumed. The postulated hazard is precluded, by design, from affecting the opposite train or from resulting in a design basis accident. For the situation in which a hazard affects a safety-related component, the event and subsequent activities are governed by Technical Specification requirements in effect when that component is not functional.

3B.2.1 EARTHQUAKE ANALYSIS ASSUMPTIONS

When evaluating the effects of any earthquake, no other major hazard or event is assumed, and no seismic Category I equipment is assumed to fail as a result of the earthquake. Certain nonseismic Category I components are designed and constructed to ensure that their failure could not reduce the functioning of a safe shutdown component to an unacceptable safety level. This criterion meets the intent of Regulatory Guide 1.29, Position C.2. Evaluation of component failure includes drop impact forces and secondary effects, such as spray and flooding from piping failure.

Loss of offsite power is assumed following an SSE. An earthquake, as a single event, will affect the entire plant; hence, all the rooms dedicated to items associated with either safety-related trains are considered in total.

3B.2.2 PIPE BREAK ANALYSIS ASSUMPTIONS

All high- and moderate-energy lines whose failure could reduce the functioning of a safe shutdown component to an unacceptable safety level are evaluated for pipe breaks or

cracks. Thrust forces, jet impingement forces, and environmental effects are considered. Section 3.6 provides a description of the location and types of breaks and the forcing functions that are considered for analyzing pipe breaks.

Evaluation of environmental effects of moderate energy pipe cracks has been made based on the characteristics of the flow from the postulated cracks. The locations of the cracks are discussed in Section 3.6.2.1. The evaluations include the effects of spraying or wetting safe shutdown equipment and the effect of flooding from the worst-case pipe crack in each room or general area. Flooding volumes are based on assuming automatic isolation or operator termination of flow to the pipe failure within a reasonable period after indication of the hazard. An interval of 30 minutes plus operator travel time to any station outside of the main control room is assumed.

3B.2.3 MISSILES ANALYSIS ASSUMPTIONS

There are two general sources of postulated internally-generated missiles within the plant:

a. Rotating component failure

b. Pressurized component failure

Section 3.5 provides a description of the design bases for the selection of missiles. Table 3B-6 provides a listing of major missiles resulting from pressurized component failure generated within the plant.

Analysis of impact from missiles that could be generated by rotating equipment or by the severance of a circumferential weld, causing the ejection of an unrestrained pipe section or dead end flange, is done for all rotating equipment and high-energy piping.

3B.2.4 FLOODING ANALYSIS ASSUMPTIONS

In the event of a pipe failure, sufficient flooding might result and jeopardize the function of safety-related equipment required to mitigate the consequences of the pipe break or to maintain the plant in a safe shutdown condition.

Flooding rates are based on the worst-case pipe failure in each safety-related room. The level of the flood water is based on automatic isolation or operator action after a reasonable delay time following indication of flow from the breaks or crack. The delay time is 30 minutes plus any time required for the operator to travel to a location outside of the control room.

Since all sites are dry (PMF below site grade), flood water evaluations are not required. Section 3.4 demonstrates that ground water infiltration is not credible and need not be addressed in the analyses of the safety-related rooms. Roof drain failures due to seismic events and their failure as moderate-energy pipe failures are postulated where required.

The equipment and floor drainage system is discussed in Section 9.3.3. All water released because of pipe breaks in the auxiliary building drains to the corridor at elevation 1974. Refer to Section 9.3.3.2.1.1 for a discussion of this design.

3B.3 PROTECTION MECHANISMS

The plant layout arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, in the event an accident occurs within the plant, there is minimal effect on other systems or components which are required for safe shutdown of the plant or to mitigate the consequence of the hazard.

Since it is not always feasible to provide separation in every hazard situation, other protection features are employed. These protection features include the following:

a. Structural enclosures

b. Structural barriers

c. Pipe whip restraints

d. Seismic restraints

e. Seismically designed components

f. Low stress levels

3B.4 HAZARDS EVALUATIONS

As stated above, Table 3B-1 provides a hazards evaluation of El. 1974 of the auxiliary building. Each room on that elevation is shown on Figure 3B-1 and has been reviewed to ensure that the integrated design of the plant acceptably addressed all postulated hazards. Since the evaluations for all safety-related areas are documented and available for audit, they are not provided in the FSAR.

Specific evaluations of certain areas of the plant have been of licensing concern in the past. These evaluations are provided in the following paragraphs.

3B.4.1 AUXILIARY FEEDWATER PUMP ROOMS

The effects of moderate energy cracks in the motor driven auxiliary feedwater pump rooms and of high-energy line breaks in the turbine steam supply line in the turbine driven auxiliary feedwater pump room have been evaluated. There are no pipes classified as high energy in the motor-driven auxiliary feedwater pump rooms. The effects of moderate-energy cracks include room pressurization (turbine-driven pump room only), flooding, and operability of the auxiliary feedwater system.

There are three separate auxiliary feedwater pump rooms, each housing one pump. Each of two motor-driven pumps is sized to deliver the feedwater flow required for decay heat removal. The single turbine-driven pump supplies twice the capacity of a motor-driven pump and is sufficient to remove decay heat and, additionally, to cool down the reactor at 50°F/hr. The turbine-driven pump provides system diversity to both motor-driven pumps.

The results of the pressurization analysis for the turbine-driven auxiliary feedwater pump room indicate that adequate vent area is provided to limit the room pressure to its design value of 3 psig.

Analysis of auxiliary feedwater piping failures shows that loss of a redundant train does not prevent decay heat removal. The capability to provide adequate feedwater flow to remove decay heat is assured by operation of either:

a. One of two motor-driven pumps or

b. The turbine-driven pump.

Similarly, flooding caused by piping failures will not cause a loss of function of auxiliary feedwater because separation is provided between all three auxiliary feedwater pump rooms. Based on analysis of the worst-case internal flooding event, the curbs between these rooms prevent propagation of flooding and ensure that the required capability of the auxiliary feedwater system is maintained.

Analysis of the other hazards shows that adequate redundancy and separation are provided to ensure the operability of at least one train of the auxiliary feedwater system.

3B.4.2 MAIN STEAM/MAIN FEEDWATER ISOLATION VALVE COMPARTMENT

The main steam/main feedwater isolation valve compartment is located in the northeast portion of the auxiliary building between the reactor building and the turbine building. Figure 3B-2 provides plan and elevation views of this area. The main steam, main feedwater, and steam generator blowdown piping in this area consist of straight piping runs approximately 40-feet long, extending from the containment penetrations to torsional restraints mounted in the auxiliary building wall through which these lines enter the turbine building. The main steam line isolation valves, main steam safety valves, atmospheric relief valves, main feedwater isolation valves, and steam generator blowdown isolation valves are in this compartment. Also in the compartment are various pressure transmitters and branch piping lines of the auxiliary feedwater system, chemical addition system, steam supply to the turbine-driven auxiliary feedwater pump, bypass loops of the main steam isolation valves, pressure instrumentation, and drains.

3B.4.2.1 Break Size and Location

All of the piping in this compartment is designed to the criteria stated in Section 3.6.2.1 for those portions of the piping passing through the primary containment and extending to the first pipe whip restraint past the first outside isolation valve. In accordance with these criteria, no specific pipe breaks are postulated in the main steam/main feedwater isolation valve compartment. However, to provide an additional level of assurance of operability of safety-related equipment in this compartment, the building structure and safety-related equipment have been evaluated for the environmental conditions (pressure, temperature, and flooding) that would result from a break, equal in area to one cross-sectional pipe area, of either a main steam line or main feedwater line or from a spectrum of main steam line breaks.

Pressurization of the main steam/main feedwater isolation valve compartment due to such a rupture is limited by providing adequate venting of the compartment and designing the compartment to withstand the maximum resultant pressure. Venting is accomplished by including adequate passageways between compartments, designing doorways to provide a path of least resistance to adjacent compartments, or other acceptable venting schemes. Engineered safety features required to bring the reactor to safe shutdown, which are located within these compartments, have been evaluated for the associated temperature, pressure, and humidity conditions.

The following cases are analyzed to determine the worst environmental conditions for the main steam/main feedwater isolation valve compartment.

Case 1a: Blowdown from a main steam line break equivalent to the flow area of a single ended rupture of a 28-inch line with a 1.5 inch minimum wall thickness (3.41 ft^2). This case results in the maximum compartment pressure.

Case 1b: Blowdown from a spectrum of main steam line breaks of 0.05, 0.1, 0.2, 0.3, 0.4, 0.5, 0.6, 0.7, 0.8, 0.9, 1.0, 1.2, 1.4, 2.0, and 4.6 ft^2 in area, with backflow. These 15 cases consider superheated steam and result in the maximum compartment temperature.

Case 2: Blowdown from a main feedwater line break equivalent to the flow area of a single ended rupture of a 14-inch schedule 80 line (0.86 ft^2). This case results in the maximum valve compartment flood level.

3B.4.2.2 Method of Analysis

The Case 1a analysis was performed using the COPDA computer code, which is described in Reference 2. The Case 1b analysis was performed using the GOTHIC computer code. For Case 1b, the diffusion layer model (DLM) was used to account for the condensation heat transfer. This was used in accordance with the conditions of NRC

approval for the GOTHIC computer code (see Reference 15). Case 2 analysis was performed using the fluid flow equations identified in Reference 1 for cold water flow.

3B.4.2.3 Mass and Energy Release

The mass and energy release data for Case 1a and the mass release for Case 2 are provided in Tables 3B-3 and 3B-4. The mass and energy release data for Case 1b are provided in References 16 and 18.

Case 1a

Case 1a mass and energy release data were developed, using the mass release rate of a 1.4 ft^2 steam line break for one steam generator (Figure 3B-3) raised by a factor of 2.44 to account for a 3.41 ft^2 break. This procedure overestimates the break flow, because it ignores the 1.4 ft^2 restriction provided both at the steam generator outlet nozzle (see Section 5.4.4) and by the 18-inch schedule 80 connection between the main steam header and each main steam line, as well as pressure drops in the steam lines (see Section 10.3.2.2). Thus the break flow at time zero could be no more than that for a 2.8 ft^2 break, or 6,480 lb/sec. The difference in break flows between 3.41 ft^2 and 2.8 ft^2 is considered to be available margin. Note that the steam generator pressure of 1,106 psia used in the analysis is based on the no-load condition and requires that the four main turbine control valves be closed; therefore, back flow from the high pressure turbine is not possible. Postulated breaks at other steam generator pressures would result in less severe transients.

Case 1b

The method, model, and blowdown data used for Case 1a maximize compartment pressure but not temperature. Therefore, Case 1b was performed to determine maximum compartment temperature. The Case 1b analyses are based on the blowdown data provided in References 16 and 18.

These blowdown data consider the effects of steam superheating caused by heat transfer from the uncovered portion of the SG tubes.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectra of both variables were evaluated by Westinghouse in References 4 and 14. At plant power levels of 102% and 70%, various break sizes have been defined in the main steam system from 4.6 ft^2 down to 0.05 ft^2.

The cases examined in the Reference 14 study for the RSG design were based on the results of the analyses in Reference 4 for Category-1 plants, which represents Callaway's configuration. A subset of the cases noted in Table III.B-4 of Reference 4, specifically, Cases 59 through 63, was originally identified as representing the Callaway licensing-basis superheated steam mass and energy releases outside containment.

These 5 cases were subsequently included in the analysis supporting a reduction in the analysis value for the minimum plant shutdown margin and revision to the analysis assumptions for the auxiliary feedwater (AFW) flow. Plant-specific confirmation of the critical inputs used in the RSG steamline break analyses outside containment for RSG conditions is documented in Reference 14. Fuel-related analysis inputs are Callaway-specific based on Vantage 5H fuel parameters. The important plant conditions and features that were assumed are discussed in Reference 14.

During startup or shutdown evolutions when safety injection on low pressurizer pressure or low steamline pressure is blocked and steamline isolation on low steamline pressure is blocked below P-11 (pressurizer pressure less than 1970 psig), the high negative steamline pressure rate (HNPR) signal is enabled by P-11 to provide steamline isolation. With RCS T_{avg} greater than 450°F, steamline isolation will be provided by the HNPR signal for all break sizes greater than or equal to 0.02 ft^2. Between 400°F and 450°F, steamline isolation will be provided dependent on the assumed break size. Below 400°F, steamline isolation will not be provided by the HNPR signal for any break size and manual steamline isolation will be performed in accordance with established procedures. Section II.C.1.a of Reference 4 states that steamline breaks initiated from lower power levels result in lower levels of steam superheating than cases initiated from full power. For that reason, the mass and energy release calculations presented in Reference 4 were initiated from either full power, plus 2% uncertainty, or from 70% power. Based on the lower levels of steam superheating, the mass and energy releases for Mode 3 (hot standby) steamline breaks outside containment not accompanied by an automatic steamline isolation signal will continue to be bounded by the current (RSG) results of References 12 and 14. Steamline breaks in the lower Modes 4-6 would be even less severe for environmental qualification purposes due to lower RCS temperatures and an effective decoupling of the steam generators from the core as the reactor coolant pumps are removed from service and the RHR system is used for decay heat removal.

Case 2

Case 2 mass release rates are based upon the condensate and feedwater systems' responses to the postulated break of a feedwater line while operating at a limiting power level (40% power). A FATHOM model has been created to analyze the break flow rate due to the Case 2 hazard scenario. The FATHOM model assumes a line break to occur on the ‘A’ train feedwater line immediately after the point where the feedwater line enters the main steam tunnel (also known as Area 5). At the time of the break:

- The ‘A’ main feedwater regulating valve (MFRV) goes 100% open
- The ‘B,’ ‘C,’ and ‘D’ MFRVs begin repositioning to 100% open to meet SG demand
- The ‘A’ and ‘B’ main feedwater pumps (MFPs) begin to spin up to 100% capacity to meet demand

- All four main feedwater regulating valve bypass valves (MFRBVs) remain closed for the duration of the event.

<u>Case 2 Analysis Assumptions</u>

Analysis assumptions include the following:

1. Flooding occurs from the feedwater line break for 67 seconds. This bounds the approximate time for actual isolation of the leak which is 66.3 seconds.

2. Submergence of the main feedwater flow transmitters causes on MFRV associated with auxiliary building room 1411 (or 1412) in Area 5 to fully open and the other MFRV to fully close. This provides the maximum flow rate from the pipe break.

3. The MFRV which fails closed due to transmitter submergence fails at the time of submergence rather than the normal valve stroke time. This provides the maximum flow rate from the pipe break.

4. The flooding flow is calculated using the FATHOM software in Archon calculation ARC-967 Revision 0 addendum 2.

5. This calculation takes the form of a simple heat balance. Each SG serves as the heat sink for one fourth of the thermal power of the reactor. The heat being transferred to a SG is used to boil off the initial inventory of liquid water in the SG. If there is no feedwater flow, the water level in the SG will begin to drop rapidly and will eventually reach the SG water level low-low trip setpoint.

6. The calculated water level is determined using incompressible fluid flow equations from Crane Technical Paper 410 © 1988 Equation 3-21 which determines the height the water reaches in the room over time.

7. Room 1411 is served by two 4” drain lines, LE-120-HCD-4” and LE-519-HCD-4”, and one 20” drain line, LE-706-HBD-20”. Room 1412 is served by two 4” drain lines, LE-121-HCD-4” and LE-324-HCD-4”, and one 20” drain line LE-707-HBD-20”. There are twenty-two wall sleeves that will allow cross flooding between the rooms. Rooms 1411 and 1412 have identical drainage systems and have the same square footage.

8. The feedwater line break is assumed to impact one of the two rooms (either 1411 or 1412). Therefore, two of the four main steamline low point drain isolation valves (MSLPDIVs) are adversely impacted and fail to close.

9. Assuming a pump controller malfunction, in conjunction with a MFRV failing open, conservatively increases the flooding level in the room, assuring the 67-second level is not exceeded.

Case 2 Sequence of Events

Archon calculation ARC-967 Addendum 2 presents the following sequence of events in Scenario 2.

The ‘A’, MFRV (AEFCV0510) fails to 100% open simultaneously with the occurrence of feedwater line break hazard. The ‘B,’ ‘C,’ and ‘D’ MFRVs (AEFCV0520, AEFCV0530, and AEFCV0540) are initially at 80% open and linearly go to 100% open over three seconds. The ‘A’ and ‘B’ MFPs (PAE01A and PAE01B) have an initial speed of 4950 rpm (93.4% of full speed). Over 25 seconds the speed will ramp to a full speed of 5347 rpm (100.8%). The maximum pump output for PAE01B was set at 100% demand. Pump PAE01A was allowed to speed up to its cavitation limit, which is 100.9% from 26 seconds to 52 seconds into the flooding event. At 52 seconds, and for the remainder of the event, PAE01A speeds up to 103.6% before being cavitation. All four MFRBVs (AEFCV0550, AEFCV0560, AEFCV0570, and AEFCV0580) are assumed to remain closed for the duration of the event.

When the break occurs, the feedwater header pressure upstream of the lines branching to each steam generator will drop to 1,059 psia (from 1,248 psia). No reverse flow from the affected steam generator through the break occurs since the isolation (check) valve in the feedwater line closes.

At 51.3 seconds, the low-low steam generator level signal (17% narrow range span) is received which trips the reactor, isolates main feedwater, and starts auxiliary feedwater. From 67 seconds (allowing 15 seconds for MFRV closure) through 9.7 minutes following the break, the condensate pumps continue to feed the break at a rate of 19,500 gpm until the condenser inventory of 159,000 gallons is exhausted. At 9.7 minutes, the condensate pumps trip on low-low condenser level.

The above flow rates are based on the system's piping resistance characteristics for water flow and the head-flow curves for the condensate and feed pumps. All fluid discharged from the break is assumed to remain as water to maximize the water which has to be drained from the area.

The flood level is limited by a floor drain system which discharges into the turbine building. Each of the two areas containing the main steam and main feedwater pipes is provided with one floor drain opening which is located under a grating platform in the north end of the room. There are drain passages between the two main steam line compartments which help to control the flood level in each compartment. Penetrations through the tunnel floor are waterproof.

Several floor drains in the steam tunnel floor at El. 2026' are interconnected with other drain lines serving rooms at El. 2000' and 1988', and all discharge into the sump located at El. 1974', which is the basement level of the auxiliary building. Any water drainage to this elevation will not affect any safety-related equipment required for mitigation of the break due to the 7-foot design flood depth of the auxiliary building basement.

The back flow of steam through the interconnecting drain lines has not been modeled into the pressure/temperature analysis of the steam tunnel due to the minimal flows expected and the commitment to qualify only the safety-related components in the steam tunnel for the effects of the nonmechanistic breaks in the steam tunnel. However, an evaluation of the rooms at the lower elevations indicates that steam escape is not likely to affect safety-related equipment due to the small driving force (steam tunnel pressure) and because fire dampers in the ventilation ducts close when the room temperature exceeds that normally anticipated. When the dampers close, the driving force equalizes and passive heat sinks take effect to reduce room temperature.

3B.4.2.4 Compartment Volumes and Vent Areas

For Case 1a, a flow schematic showing the subcompartment volumes and vent areas used in the nodalization models is provided in Figure 3B-4. For Case 1a, the main steam/main feedwater isolation valve compartment is divided into ten subcompartments, based on the physical structures which exist in this area. For Case 1b, the compartment is divided into two subcompartments, the east and west bays. Table 3B-5 identifies the volumes, vent paths, vent area, and flow coefficients for both Cases 1a and 1b.

Case 1a

Compartment 1, the one in which the break is assumed to occur, is bounded below by a concrete floor at El. 2026'-0", above by grating at El. 2042'-0" and 2037'-7 1/4", and on the sides by structural walls. Compartment 1 houses two main steam lines, two main feedwater lines, and two steam generator blowdown lines, and is located on the west side of the structural dividing wall in the middle of the valve compartment. Compartment 2 has the same boundaries as Compartment 1, except that it is located on the east side of the dividing wall. Compartments 1 and 2 are connected by a 3'-4" x 7' opening.

Compartments 3 and 4 house the torsional restraints for the piping from Compartments 2 and 1, respectively. Thirty-inch diameter crawlways provide access between Compartments 2 and 3 and between 3 and 4. Compartment 5 includes the volume between the grating elevations above Compartment 2 and the roof at El. 2088'-0". Compartment 6 is identical to Compartment 5, except that it is located directly above Compartment 1. A 23' x 27' opening exists between Compartments 5 and 6. Compartments 7 and 8 extend from elevation 2088’ to the roof deck enclosure. Node 7 is on the East side of the dividing wall above node 5 and node 8 is on the West side above node 6. Compartments 9 and 10 extend from elevation 2090’ to the roof deck enclosure at elevation 2102’-6". Node 9 is on the East side and node 10 is on the West side. Node 11 is the atmosphere.

Case 1b

Compartment 1 is the west bay of the steam tunnel at El. 2026'-0" up to the roof. Compartment 2 is the east bay of the steam tunnel. The break is assumed to occur in Compartment 1. Heat sink properties used in the analyses are provided in Table 3B-8.

Node 3 models the outside environment and interfaces with a constant pressure boundary node and a constant temperature infinite heat sink to maintain the environment pressure and temperature (outlined in Table 3B-2) throughout the analysis.

3B.4.2.5 Initial Conditions

Table 3B-2 provides the initial conditions for Cases 1a, 1b, and 2 analyses.

3B.4.2.6 Results

Case 1a

A plot of the time-history of compartment pressure (Case 1a) is given in Figure 3B-6. The Case 1a plot was truncated at 0.5 second after maximum compartment pressure (6.7 psig) had been reached. The truncated plot was merged with the long-term compartment pressure profile for a main steam line break equivalent in flow area to a single ended rupture (1.4 ft^2) with backflow. The methodology used for this 1.4 ft^2 case, which was the break originally used to establish the compartment temperature response prior to the inclusion of superheated steam effects, did not maximize the pressure response. The pressure profiles were merged on Figure 3B-6 to provide both peak and long-term pressure on one curve.

Case 1b

Case 1b results are presented in Figures 3B-7 through 3B-21. The higher temperature curves for the west (break) compartment were used in the evaluation of temperature effects on equipment in both compartments. The Case 1b peak temperatures exceed the qualification requirements originally established for equipment in these rooms based on a 1.4 ft^2 break without superheat; therefore, the surface temperature response of the equipment was evaluated to demonstrate the proper operation of equipment before it was calculated to be heated above its qualified temperature. Failure modes and effects analyses were also employed, when required, to evaluate certain electrical circuits and determine equipment performance.

The surface temperature response was calculated for various representative pieces of equipment and components which may be required following an MSLB in the steam tunnel. The room temperature and equipment surface temperature responses were calculated within the GOTHIC main steam tunnel model.

At any given time, the greater of four times the Uchida condensing heat transfer rate (based on the compartment air to steam mass ratio) or the convective heat transfer rate was used to evaluate the transient surface temperature response of the selected equipment. The McAdams correlation for forced convection was used to calculate the convective heat transfer coefficient. In the evaluation of the heat transfer coefficient for a component, the characteristic velocity was taken as the time dependent average velocity of the flow between the east and west rooms of the steam tunnel. The film properties used in the evaluation of the McAdams equation were based on the state of the air and steam in the stream.

As only the outside casing of equipment was modeled, the lumped-capacity method was used to calculate the surface temperature response of the equipment. This approach is justified by the thinness and the high thermal conductivity of the modeled external casings. For the steamline pressure transmitters, more detailed, multi-layer one-dimensional thermal lag analyses were performed to determine equipment temperatures. Details of the equipment modeling are provided in Reference12.

The surface temperature analysis showed that, with certain exceptions, the equipment surface temperature did not exceed the qualified temperature limits prior to the time when a steam line isolation signal (SLIS) or feedwater isolation signal (FWIS) was initiated. Surface temperature values at the time of SLIS are provided in Reference 12. Surface temperatures at the time of a FWIS are lower since feedwater isolation was conservatively assumed to occur coincident with reactor trip or on low RCS T_{-avg} following reactor trip, but in all cases much sooner than SLIS (see References 12 and 14). The exceptions are the main steam pressure transmitter instrument cable, and air-operated valve control cable. A failure modes and effects analysis showed that failure of these cables will not affect the ability to safely shut down the plant following a main steam line break as the affected equipment either fails safe or alternative capability is provided. Further analysis showed that the failure of equipment subsequent to its actuation will not result in equipment repositioning or in misleading the plant operators. Reference 9 provides additional details concerning failure modes and effects analysis. Discussions of qualification margins may be found in References 7 and 9. The Case 1b analyses demonstrate that the equipment located in the main steam tunnel credited to mitigate a main steam line break will not exceed their qualification temperatures prior to performing and completing their safety function.

The effect of superheated steam temperatures on the tunnel structures was considered. It was concluded that the higher temperatures resulting from superheated blowdown will have no detrimental effects on the steam tunnel structural steel and reinforced concrete due to the relatively short duration of these events.

The qualification temperatures of safety-related mechanical equipment were evaluated and were found to exceed the calculated steam tunnel compartment temperatures for all break sizes.

Case 2

The water height in the affected valve compartment reaches 22” at 52 seconds into the hazard scenario. At this time the feedwater flow transmitters are submerged and are assumed to fail. The ‘A’ transmitter continues to send a full open signal to the ‘A’ MFRV The ‘D’ transmitter sends a full-closed signal to the ‘D’ MFRV. The ‘D’ MFRV is assumed to be fully closed 52 seconds into the event (maximum water addition to the room) and thus forces additional flow through the broken line. The FATHOM model event is assumed to last 67 seconds and accumulates 25.9 inches of water in the room. The time necessary to reach a SG water level low-low trip signal at 40% RTP has been calculated to be 51.3 seconds. With an additional 15-second MFRV closure time, the hazard is over in approximately 66.3 seconds, which is bounded by the 67-second Case 2 analysis. At this point, the condensate pumps continue to feed the break for approximately 10 minutes though the failed open MFRV at a rate approximately 45% of the total MFP flow. The floor drains in the room are sized to manage this flow rate, and thus, the 25.9 inches water level is bounding.

3B.4.2.7 Design Provisions

The Case 1b analyses with consideration of superheated blowdowns resulted in the peak environmental conditions presented in Table 3B-9:

The pressure values are well below the Case 1a limit of 21.4 psia.

Table 3B-2 provides the design values of compartment pressure and temperature. Although the design temperature of 324°F is exceeded by the peak compartment temperatures of Case 1b, the analysis of equipment surface temperature response showed that equipment that must function to bring the plant to a safe shutdown condition following a steam line break in the main steam/main feedwater isolation valve compartment will perform their design functions in the environmental conditions following a steam line break including superheated steam effects. Reference 11 provides the NRC safety evaluation of the licensing basis OSG Case 1b analysis.

The floor drain system is designed so that the feedwater isolation valve actuator and limit switches are above the design floodwater level. All safety-related instrumentation, such as steam line pressure transmitters, are located above the design floodwater level. All other safety-related valves, except for the main steam line low point drain isolation valves, are located so that this water level does not affect their respective safety-related functions. Although the MSLPDIVs are submerged there is no challenge to the ability to achieve and maintain a safe shutdown condition.

3B.4.3 TURBINE BUILDING FLOODING EVALUATION

3B.4.3.1 Introduction

An analysis was performed on the circulating water system, which postulated a complete rupture of a single expansion joint.

The SNUPPS power block design eliminates the flooding potential by providing physical barriers to prevent circulating water from entering the safety-related areas.

The condenser pit and ground floor of the turbine building are shown in Figures 1.2-29 and 1.2-30.

The turbine building is designed as a non-Category I structure. The exterior walls are concrete block to 3 feet above grade and sheet metal siding above this level. There are several access points to the outside at ground level, as follows: two stairwells on the east side; two stairwells on the west side; and roll-up doors with adjacent personnel doors in the southeast and the northeast corners of the building.

There are four sumps located in the condenser pit of approximately 1,500-gallon capacity each. These sumps have remote level alarms in the control room.

Level switches are incorporated in the condenser pit to stop the circulating water pumps and close the pump discharge valves automatically upon a high pit water level indication, and thus reduce the water flow rate to the pit. The level switch is set to initiate circulating water pump stop at a maximum water level of 5 feet above the bottom of the condenser pit.

The flooding analysis evaluates the potential for floodwater to affect safety-related equipment in the auxiliary and control buildings. The areas which contain this equipment are separated from the turbine building by concrete walls with penetrations which may be potential flood paths. Stairway T-1 at the south end of the turbine building at El. 2000 provides a direct path down to the auxiliary building at El. 1974. Floodwater also could enter the control building through the communications corridor via a series of two doors at El. 2000. This latter route also represents an indirect path to the auxiliary building basement via a stairwell to El. 1984 in the control building and down a second stairwell to Room 1101. Another indirect path to the auxiliary building basement is provided by door 33044 at El. 2000 into Room 1301.

3B.4.3.2 CWS Rupture Analysis

The condenser pit is an area within the turbine building, below grade, which houses the main condensers and turbine auxiliary equipment. It is a large rectangular area 152 feet by 136 feet with a smaller northern extension of 62 feet by 48 feet. The pit is 17 feet deep, extending from El. 1983 to 2000, encompassing a net free volume of 310,000 ft^3

(equivalent to 2.4 million gallons of floodwater). The equivalent water volume per foot of depth is 142,000 gal/ft.

The analysis postulates a failure of a single expansion joint which could result in a flow rate as great as 680,000 gpm, considering the runout of the pumps. The volume of the turbine building sumps and the operation of the sump pumps are not credited. The corresponding rate at which the condenser pit is filled is about 5 feet per minute. After 1 minute, the level switch automatically stops the pumps and closes the isolation valves. Indication of the rupture is provided in the control room by the four sump level alarms, the high pit level alarm, turbine trip along with main feedwater and condensate pump trips, and CWS pump isolation valve indications.

Because the isolation valves are assumed not to close completely, the flow rate through the break is not completely terminated. It is assumed that flow into the condenser pit continues at the 10 percent of full break flow rate. Therefore, the condenser pit water level continues to rise at 0.5 feet per minute. In 25 minutes the circulating water will overflow the condenser pit onto the turbine building floor at El. 2000 if the leak continues at this rate.

3B.4.3.3 CWS Rupture Evaluation

Advance warning of an impending flooding situation in the turbine building resulting from CWS expansion joint failure is provided by sump level alarms. At least 1 minute later, a condenser pit high level alarm is received. Concurrently, the CW pumps are tripped and isolated automatically, terminating the flow when the water level is approximately 12 feet below the top of the condenser pit.

The most likely failure condition is to assume the isolation valves to not close completely, the flow rate through the break is reduced but not completely terminated. When the circulating pumps spin down, flow from the cooling tower will terminate. Since the piping expansion joints in the turbine building are located above the Cooling Tower Basin water level elevation, the cooling tower will not siphon through the break into the condenser pit.

Back flow from the condenser water boxes and cross over piping will gravity drain into the condenser pit. That amount of water, 852,000 gallons, is equivalent to six feet of level in the condenser pit. The pit should not over fill with water from this event.

If it cannot be verified that the flow has been automatically terminated, backup action may be taken by the operators to secure the CWS or to provide flow paths through access routes to the outside. The curbs along the west side of the condenser pit and around the T-1 stairwell are provided to prevent water from spilling over into the safety-related spaces.

3B.4.4 EVALUATION OF RCS LOOP BRANCH LINE BREAKS

The evaluation of effects on safety-related equipment resulting from branch line breaks in the reactor coolant system is presented in Table 3B-7. The evaluation shows that breaks in the RCS will not compromise the capability to safely shut down the plant.

3B.5 REFERENCES

1. FATHOM, Version 7.0 (release 2012.08.08)

2. Subcompartment Pressure and Temperature Transient Analysis, BN-TOP-4, Revision 1, Bechtel Corporation, San Francisco, California, July 1976.

3. Bechtel Power Corporation, FLUD, Theoretical and Users Manual (NE017), Revision 1 (Bechtel Computer Code), April 1981.

4. "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," WCAP-10961, Revision 1, October 1985.

5. Deleted.

6. Deleted.

7. Letter SLNRC 86-06, dated April 4, 1986, N. A. Petrick (SNUPPS) to H. R. Denton (NRC).

8. Letter ULNRC-1473, dated March 24, 1987, D. F. Schnell (UE) to NRC.

9. Letter ULNRC-1640, dated October 5, 1987, D. F. Schnell (UE) to NRC.

10. Letter ULNRC-1724, dated February 10, 1988, D. F. Schnell (UE) to NRC.

11. Letter dated February 18, 1988, T. W. Alexion (NRC) to D. F. Schnell (UE).

12. Calculation ZZ-524.

13. Deleted.

14. “Callaway Replacement Steam Generator Program NSSS Engineering Report,” WCAP-16140 (Proprietary), July 2004.

15. ULNRC-05117 dated February 11, 2005, Response to NRC Request for Additional Information #1.

16. Callaway Plant Accident Analysis Basis Document (AABD) Module 20B.0, Steamline Break Mass & Energy Releases Outside Containment.

17. RFR 018764A, “Turbine Building Flooding Analysis”

18. Letter SCP-08-36, dated July 18, 2008, Jim Roberts (Westinghouse) to Phil Myers (AmerenUE).

TABLE 3B-1 (See Note 5)
HAZARDS ANALYSIS OF AUXILIARY BUILDING - ELEVATION 1974 ' 0"

Room Number 1101 Title General Floor Area #1

Design Approach

	Only safety-related equipment (SRE) is in the room.
	Only nonsafety-related equipment (NSRE) is in the room.
X	Minimize SRE in the room and segregate from NSRE.
	Minimize NSRE in the room and segregate from SRE.
	Other, see remarks.

Flooding Analysis

	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
	Flooding from sources external to the room is not credible even with a single active failure.
X	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
X	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
	Other, see remarks.

Missile Analysis

X	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
	Other, see remarks.

Pipe Break Analysis

	There are no high-energy lines in the room.
X	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 1 with break locations shown in Figure 3.6-1.
X	Moderate energy cracks within the room do not adversely affect SRE in the room.
	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Flooding from any source does not adversely affect SRE because all SRE is located above the maximum design flood depth of 7 feet (el. 1981).

TABLE 3.B-1 (Sheet 1A) (See Note 5)

Listing of items in room 1101

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipe	EF-129-HBC-24"	X	Y	
S2	Conduit[4]	4U1A3A,B,C	X	Y	
	J-box	4UJ001	X	Y	
S3	Pipes	EG-036-HBC-3"	X	Y	
		EG-047-HBC-3"	X	Y	
S4	Pipes	EF-128-HBC-18"	X	Y	
		EF-150-HBC-18"	X	Y	
S5	Pipe	EF-031-HBC-18"	X	Y	
S6	Conduit power & control to Valve HV-8111	4J1A3D	X	Y	Redundant valve HV-8110 available and manual override as backup
S7	Conduit	1UF2A,B,C	X	Y	
	J-box	1UJ001	X	Y	
S8	Pipes	EF-137-HBC-24"	X	Y	
		EF-138-HBC-30"	X	Y	
S9	Conduit	4U1014, 1015	X	Y	
S10	Conduit	1U1J3A,B,C	X	Y	
	J-box	1UJ002	X	Y	
S11	Pipes	EF-76-HBC-24"	X	Y	
		EF-80-HBC-24"	X	Y	
		EF-20-HBC-30"	X	Y	
		EF-81-HBC-30"	X	Y	
S12	Pipes	EF-73-HBC-16"	X	Y	
		EF-35-HBC-18"	X	Y	
S13	Conduit	4J1031		Y	BAT level indication;
		4J1A2B		Y	not required for SSD
S14	Pipe and valves	LF-132-HCC-6"		Y	No LOCA assumed;
		HV105 & 106		Y	isolation not required
N1	Racked trays	5U4C, 5U5L,		S	
		5J1V		S	
N2	Pipes	GB-062-HBD-1"		N	Does not adversely
		GB-063-HBD-1"		S	affect SRE
N3	Racked pipes (13) at El. 1985'			S	
N4	Pipe	FB-032-HBD-8"		S	
N5	Racked trays	6U1C,6U3K,6U5M		S	
		6J1B, 6J1C		S	
N6	Racked trays	5U1C,5U5K,5J1C		S	
N7	Pipe	KA-218-JBD-6"		S	
N8	Pipes	LF-185-HCD-4"		S	
		LF-174-HCD-4"		S	
		LF-10-HCD-6"		S	
		LF-255-HCD-4"		S	
N9	Pipes	KD-3",KD-1 1/2"		N	Does not adversely affect SRE
N10	Pipe	KA-029-JBD-1"		S	
N11	Conduit	5U1C1E		S	
		5U1C1J		S	
		5U1C1G		S	
N12	Pipes	LF-797-HCD-2 1/2"		N	Does not adversely
		LD-027-HCD-3"		N	affect SRE
		LD-042-HCD-2"		N	Does not adversely affect SRE
N13	Pipe	KD-2"		N	Does not adversely affect SRE
N14	Conduit	6U5M2D		S	
N15	Pipe	LB-021-YNG-12"		S	
N16	Pipe	HB-227-HBD-2"		N	Does not adversely affect SRE
N17	Pipe	HB-050-HCD-3"		S	

TABLE 3.B-1 (Sheet 2) (See Note 5)

Room Number 1102 Title Chiller and Surge Tank Area

Design Approach

	Only safety-related equipment (SRE) is in the room.
	Only nonsafety-related equipment (NSRE) is in the room.
	Minimize SRE in the room and segregate from NSRE.
	Minimize NSRE in the room and segregate from SRE.
X	Other, see remarks.

Flooding Analysis

	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
	Flooding from sources external to the room is not credible even with a single active failure.
X	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
X	Other, see remarks.

Missile Analysis

	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
	External missiles cannot enter the room due to missile protection.
X	Other, see remarks.

Pipe Break Analysis

	There are no high-energy lines in the room.
	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 2 with break locations shown in Figure 3.6-1.
	Moderate energy cracks within the room do not adversely affect SRE in the room.
X	Other, see remarks.

Remarks:

1) The only safety-related equipment in this area are two HVAC ducts; each has a tornado damper and two isolation dampers which close on an SIS. No LOCA is assumed concurrent with an SSE, thus these ducts are not required.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 3) (See Note 5)

Room Number 1103 | Title Letdown Chiller Heat Exchanger Room

Design Approach

- [ ] Only safety-related equipment (SRE) is in the room.
- [x] Only nonsafety-related equipment (NSRE) is in the room.
- [ ] Minimize SRE in the room and segregate from NSRE.
- [ ] Minimize NSRE in the room and segregate from SRE.
- [ ] Other, see remarks.

Flooding Analysis

- [ ] Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
- [ ] Flooding from sources external to the room is not credible even with a single active failure.
- [x] Other, see remarks.

Seismic Design Analysis

- [ ] Only SRE is in the room; therefore, there are no seismically induced failures.
- [ ] The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- [ ] The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- [ ] The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- [x] Other, see remarks.

Missile Analysis

- [ ] No credible missile sources exist in the room.
- [ ] Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- [ ] External missiles cannot enter the room due to missile protection.
- [x] Other, see remarks.

Pipe Break Analysis

- [x] There are no high-energy lines in the room.
- [ ] The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
- [ ] Moderate energy cracks within the room do not adversely affect SRE in the room.
- [x] Other, see remarks.

Remarks:

1) There is no safety-related equipment in the room.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 4) (See Note 5)

Room Number 1104 Title Letdown Reheat Heat Exchanger Room

Design Approach

- ___ Only safety-related equipment (SRE) is in the room.
- ___ Only nonsafety-related equipment (NSRE) is in the room.
- ___ Minimize SRE in the room and segregate from NSRE.
- X Minimize NSRE in the room and segregate from SRE.
- ___ Other, see remarks.

Flooding Analysis

- ___ Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
- ___ Flooding from sources external to the room is not credible even with a single active failure.
- X Other, see remarks.

Seismic Design Analysis

- ___ Only SRE is in the room; therefore, there are no seismically induced failures.
- ___ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- ___ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- X Other, see remarks.

Missile Analysis

- ___ No credible missile sources exist in the room.
- ___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- ___ External missiles cannot enter the room due to missile protection.
- X Other, see remarks.

Pipe Break Analysis

- ___ There are no high-energy lines in the room.
- ___ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 3 with break locations shown in Figure 3.6-1.
- ___ Moderate energy cracks within the room do not adversely affect SRE in the room.
- X Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) The only safety-related equipment in the room is piping and the heat exchanger associated with the letdown flowpath, which is not required for SSD.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 4A) (See Note 5)

Listing of items in room 1104

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipes	BG-028-ECB-3"		Y	Letdown path not
		BG-029-ECB-3"		Y	required for SSD
S2	Heat exchanger	EBG05		Y	
N1	Monorail	HKF26		Y	
N2	Pipe	BG-063-GCD-3"		N	
N3	Pipe	BG-060-GCD-3"		N	
N4	Pipes	LF-113-HCD-4"		N	
		LG-065-HCD-6"		N	

TABLE 3.B-1 (Sheet 5) (See Note 5)

Room Number 1105 — Title Auxiliary Heat Exchanger Valve Compartment

Design Approach

- ____ Only safety-related equipment (SRE) is in the room.
- ____ Only nonsafety-related equipment (NSRE) is in the room.
- ____ Minimize SRE in the room and segregate from NSRE.
- X Minimize NSRE in the room and segregate from SRE.
- ____ Other, see remarks.

Flooding Analysis

- ____ Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
- ____ Flooding from sources external to the room is not credible even with a single active failure.
- X Other, see remarks.

Seismic Design Analysis

- ____ Only SRE is in the room; therefore, there are no seismically induced failures.
- ____ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ____ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- ____ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- X Other, see remarks.

Missile Analysis

- ____ No credible missile sources exist in the room.
- ____ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- ____ External missiles cannot enter the room due to missile protection.
- X Other, see remarks.

Pipe Break Analysis

- ____ There are no high-energy lines in the room.
- ____ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 4 with break locations shown in Figure 3.6-1.
- ____ Moderate energy cracks within the room do not adversely affect SRE in the room.
- X Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) The only safety-related equipment in the room is associated with the letdown flowpath, which is not required for SSD.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 5A) (See Note 5)

Listing of items in room 1105

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipes	BG-028-ECB-3"		Y	Letdown flowpath
		BG-029-ECB-3"		Y	not required for SSE
S2	Pipe	BG-017-GCB-3"		Y	
S3	Pipe	BG-062-HCB-1"		Y	
	Relief valve	BG V-7006		Y	
N1	All NSRE			N	

TABLE 3.B-1 (Sheet 6) (See Note 5)

Room Number 1106 | Title Moderating Heat Exchanger Room

Design Approach

- ___ Only safety-related equipment (SRE) is in the room.
- X Only nonsafety-related equipment (NSRE) is in the room.
- ___ Minimize SRE in the room and segregate from NSRE.
- ___ Minimize NSRE in the room and segregate from SRE.
- ___ Other, see remarks.

Flooding Analysis

- ___ Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
- ___ Flooding from sources external to the room is not credible even with a single active failure.
- X Other, see remarks.

Seismic Design Analysis

- ___ Only SRE is in the room; therefore, there are no seismically induced failures.
- ___ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- ___ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- X Other, see remarks.

Missile Analysis

- ___ No credible missile sources exist in the room.
- ___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- ___ External missiles cannot enter the room due to missile protection.
- X Other, see remarks.

Pipe Break Analysis

- X There are no high-energy lines in the room.
- ___ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
- ___ Moderate energy cracks within the room do not adversely affect SRE in the room.
- X Other, see remarks.

Remarks:

1) There is no safety-related equipment in the room.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 7) (See Note 5)

Room Number 1107 Title Centrifugal Charging Pump Room B

Design Approach

____ Only safety-related equipment (SRE) is in the room.
____ Only nonsafety-related equipment (NSRE) is in the room.
____ Minimize SRE in the room and segregate from NSRE.
X Minimize NSRE in the room and segregate from SRE.
____ Other, see remarks.

Flooding Analysis

X Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X Flooding from sources external to the room is not credible even with a single active failure.
____ Other, see remarks.

Seismic Design Analysis

____ Only SRE is in the room; therefore, there are no seismically induced failures.
X The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
____ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
____ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
____ Other, see remarks.

Missile Analysis

____ No credible missile sources exist in the room.
X Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X External missiles cannot enter the room due to missile protection.
____ Other, see remarks.

Pipe Break Analysis

____ There are no high-energy lines in the room.
X The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 5 with break locations shown in Figure 3.6-1.
X Moderate energy cracks within the room do not adversely affect SRE in the room.
____ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 7A) (See Note 5)

Listing of items in room 1107

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Conduit	4U3F1H	X	Y	
		4U3F1J	X	Y	
S2	Conduit	4U3F5F	X	Y	
		4U1010,1011	X	Y	
S3	Switch	GL-HIS-26	X	Y	
S4	Conduit tray	4B2E1P	X	Y	
		4B2E	X	Y	
S5	Pipes	BN-08-HCB-8"	X	Y	
		BG-146-HCB-8"	X	Y	
		BG-151-HCB-6"	X	Y	
S6	Pipes	EG-036-HBC-3"	X	Y	
		EG-038-HBC-3"	X	Y	
		BG-277-HBC-2 1/2"	X	Y	
		BG-278-HBC-2"	X	Y	
S7	Pipes	BG-152-BCB-4"	X	Y	
		BG-157-BCB-4"	X	Y	
S8	Pipe	BG-155-BCB-2"	X	Y	
S9	Pipe	BG-153-BCB-2"	X	Y	
S10	Instrument	BG-PI-188	X	Y	
S11	Instrument	BG-PI-119	X	Y	For pressure boundary only
S12	Pipes	EF-094-HBC-4"	X	Y	
		EF-095-HBC-4"	X	Y	
		GL-112-HBC-4"	X	Y	
		GL-114-HBC-4"	X	Y	
S13	Valves	LCV-112E	X	Y	
		V-8546B	X	Y	
S14	Valves	V-8481B	X	Y	
		V-095	X	Y	
S15	Pipes	BG-037-GCB-3"		Y	
		BG-196-HCB-2"		Y	
		BG-163-BCB-2"		Y	
S16	Centrifugal charging pump	PBG05B	X	Y	
S17	Room cooler	SGL12B	X	Y	
N1	Pipe	LF-042-HCD-4"		S	
N2	(Deleted)				
N3	Conduit switch	6U1C1M		S	
		GL-TSH-43		S	
N4	Monorail	HKF19B		Y	
N5	Tubing from pipe in Rm. 1105	FE-BG-365		N	Does not adversely affect SRE
N6	Relief valve pipe	HE-041-HCD-1"		S	

TABLE 3.B-1 (Sheet 8) (See Note 5)

Room Number 1108

Title Safety Injection Pump Room B

Design Approach

____	Only safety-related equipment (SRE) is in the room.
____	Only nonsafety-related equipment (NSRE) is in the room.
____	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
____	Other, see remarks.

Flooding Analysis

X	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X	Flooding from sources external to the room is not credible even with a single active failure.
____	Other, see remarks.

Seismic Design Analysis

____	Only SRE is in the room; therefore, there are no seismically induced failures.
X	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
____	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
____	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
____	Other, see remarks.

Missile Analysis

X	No credible missile sources exist in the room.
____	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
____	Other, see remarks.

Pipe Break Analysis

X	There are no high-energy lines in the room.
____	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
X	Moderate energy cracks within the room do not adversely affect SRE in the room.
____	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 8A) (See Note 5)

Listing of items in room 1108

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Tray	4U3F	X	Y	
S2	Conduit	4U3F1K,1L,1M	X	Y	
		1N,1P,1Q,1T,1V	X	Y	
		4U3F5E	X	Y	
S3	Conduit	4U3F1D,1E,1F,1G	-	Y	
		4U3F1X,1Y	-	Y	
		4U3F5G	-	Y	
S4	Conduit	4U1006,07	-	Y	
		4U3F5D		Y	
S5	Switch	GL-HIS-27		Y	
S6	Conduit	4U3F1B,1C,		Y	
		1H,1J		Y	
		4U3F5F		Y	
S7	Conduit	4U3F1R,1S		Y	
		4U3F1V,1W		Y	
S8	Conduit tray	4B2D1N		Y	
		4B2D		Y	
S9	(Deleted)				
S10	Instruments	EM-FT-922		Y	For pressure boundary only
		EM-PT-923		Y	
S11	Instruments	EM-PI-978		Y	For pressure boundary only
S12	Pipe	EG-044-HBC-1"	X	Y	
		EG-045-HBC-1"	X	Y	
S13	Pipe	EF-090-HBC-4"		Y	
		EF-091-HBC-4"		Y	
		EF-108-HBC-3"		Y	
		EF-110-HBC-3"		Y	
S14	Pipe	BG-401-HBC-6"	X	Y	
		EM-22-HBC-6"	X	Y	
		EM-23-HBC-6"	X	Y	
S15	Pipe	EM-47-CCB-4"		Y	
		EM-14-CCB-4"		Y	
		EM-44-CCB-1 1/2"		Y	
S16	Pipe	BN-09-HCB-8"		Y	
		EM-003-HCB-8"		Y	
		EM-004-HCB-8"		Y	
S17	Pipe	EJ-058-ECB-8"	X	Y	
S18	Pipe	EM-045-CCB-3"		Y	
S19	Safety inj. pump	PEM01B		Y	
S20	Room cooler	SGL09B		Y	
S21	Valves	HV-8924,8807B	X	Y	
S22	Valves	HV-8821B,8922B		Y	
S23	Valve	HV-8813		Y	
S24	Valves	HV-8804B,8806B	X	Y	
		HV-8923B	X	Y	
S25	Valves	HV-8926B,8969B	X	Y	
S26	Pipe	EJ-40-ECB-10"	X	Y	
S27	Pipe	EG-039-HBC-2"	X	Y	
		EG-040-HBC-2"	X	Y	
		EG-041-HBC-2"	X	Y	
		EM-032-HCC-2"	X	Y	
		EM-034-HCC-2"	X	Y	
N1	Conduit	6J1008,09,10,20		S	
		6U1021		S	
N2	Pipe	HE-042-HCD-1"		S	Structural integrity only
		HE-043-HCD-1"		S	
N3	Pipe	BN-01-HCD-3"		S	Structural integrity only
N4	Pipe	BN-02-HCD-4"			See Room 1110, Item N2
N5	Monorail	HKF16B		Y	
N6	Instrument	BN-FI-968		S	Structural integrity only

TABLE 3.B-1 (Sheet 9) (See Note 5)

Room Number 1109 Title Residual Heat Removal Pump Room B

Design Approach

____	Only safety-related equipment (SRE) is in the room.
____	Only nonsafety-related equipment (NSRE) is in the room.
____	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
____	Other, see remarks.

Flooding Analysis

X	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X	Flooding from sources external to the room is not credible even with a single active failure.
____	Other, see remarks.

Seismic Design Analysis

____	Only SRE is in the room; therefore, there are no seismically induced failures.
X	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
____	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
____	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
____	Other, see remarks.

Missile Analysis

____	No credible missile sources exist in the room.
X	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
____	Other, see remarks.

Pipe Break Analysis

X	There are no high-energy lines in the room.
____	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
X	Moderate energy cracks within the room do not adversely affect SRE in the room.
____	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 9A) (See Note 5)

Listing of items in room 1109

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Conduit tray	4B2C1M	X	Y	
		4B2C	X	Y	
S2	Conduit	1U1050,51,52	X	Y	
S3	Conduit	4U3F1M,1N	X	Y	
S4	Conduit	4U3F1P,1Q	X	Y	
S5	Conduit	4U1008,38,39	X	Y	
		4U3F5E	X	Y	
S6	Switch	GL-HIS-28	X	Y	
S7	Instrument	EJ-PT-615	X	Y	
S8	Pipe	EJ-27-ECB-3"	X	Y	
S9	Pipes	BN-12-HCB-14"	X	Y	
		EJ-10-HCB-14"	X	Y	
		EJ-11-HCB-14"	X	Y	
S10	Pipe	EJ-016-ECB-10"	X	Y	
S11	Pipes	EF-085-HBC-4"	X	Y	
		EF-086-HBC-4"	X	Y	
S12	Pipes	EG-52-HBC-1"	X	Y	
		EG-53-HBC-1"	X	Y	
S13	Valves	EJ-HV-8812B	X	Y	
		EJ-HV-8958B	X	Y	
		EJ-FCV-611	X	Y	
S14	Valve	EJ-8724B	X	Y	
S15	RHR pump	PEJ01B	X	Y	
S16	Room cooler	SGL10B	X	Y	
N1	Conduit switch	6U1C1J		S	
		GL-TSH-48		S	
N2	Conduit	6U1C1E		S	
		5U1C1G		S	
N3	Pipes	LF-118-HCD-4"		S	
		LF-119-HCD-4"		S	
		LD-441-HCD-4"		S	
N4	(Deleted)				
N5	Pipe	LF-364-HCD-2 1/2"		S	
N6	Sump pumps	PLF01C		N	Rotating part,
		PLF01D		N	totally contained
N7	Sump duct			S	
N8	Monorail	HKF17B		Y	

TABLE 3.B-1 (Sheet 10) (See Note 5)

Room Number 1110 Title Containment Spray Pump Room B

Design Approach

___ Only safety-related equipment (SRE) is in the room.
___ Only nonsafety-related equipment (NSRE) is in the room.
___ Minimize SRE in the room and segregate from NSRE.
X Minimize NSRE in the room and segregate from SRE.
___ Other, see remarks.

Flooding Analysis

X Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X Flooding from sources external to the room is not credible even with a single active failure.
___ Other, see remarks.

Seismic Design Analysis

___ Only SRE is in the room; therefore, there are no seismically induced failures.
X The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
___ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
___ Other, see remarks.

Missile Analysis

X No credible missile sources exist in the room.
___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X External missiles cannot enter the room due to missile protection.
___ Other, see remarks.

Pipe Break Analysis

X There are no high-energy lines in the room.
___ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
X Moderate energy cracks within the room do not adversely affect SRE in the room.
___ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 10A) (See Note 5)

Listing of items in room 1110

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Conduit	4U1012,13		Y	
		4U3F5G		Y	
S2	Conduit	4U3F1D,1E		Y	
S3	Conduit	1U1052	X	Y	
S4	Conduit tray	4B2B1L		Y	
		4B2B		Y	
S5	Pipes	EF-088-HBC-4"		Y	
		EF-089-HBC-4"		Y	
S6	Pipes	EN-06-HCB-12"		Y	
		EN-05-HCB-12"		Y	
		EN-12-HCB-3"		Y	
S7	Pipes	EN-07-GCB-10"		Y	
		EN-08-GCB-3"		Y	
		EN-57-GCB-3"		Y	
		EN-59-GCB-3"		Y	
S8	Pipe	EN-016-GCB-4"		Y	
S9	Valves	EN-HV-03		Y	
		EN-V009,010		Y	
S10	Instrument	EJ-PT-615	X	Y	
S11	Instrument	EN-PT-10		Y	For pressure boundary only
S12	Instrument	EN-PI-8		Y	For pressure boundary only
S13	Instrument	EN-FT-14		Y	For pressure boundary only
S14	Instrument	EN-FT-11		Y	For pressure boundary only
S15	Eductor	SEN01B		Y	
S16	CS pump	PEN01B		Y	
S17	Room cooler	SGL13B		Y	
S18	Switch	GL-HIS-29		Y	
N1	Conduit	6J1012,13		S	
		6J1036,37,38		S	
		6U1021		S	
N2	Pipes	EN-15-GCD-4"		S	Structural integrity only
		BN-02-HCD-4"		S	
N3	Conduit	6J1039		S	
	Instrument	EN-FI-14B		S	
N4	Pipe	LF-118-HCD-4"		S	
N5	Duct from RHR sump			S	
N6	Monorail		HFK18B	Y	

TABLE 3.B-1 (Sheet 11) (See Note 5)

Room Number 1111 | Title Residual Heat Removal Pump Room A

Design Approach

- ___ Only safety-related equipment (SRE) is in the room.
- ___ Only nonsafety-related equipment (NSRE) is in the room.
- ___ Minimize SRE in the room and segregate from NSRE.
- X Minimize NSRE in the room and segregate from SRE.
- ___ Other, see remarks.

Flooding Analysis

- X Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
- X Flooding from sources external to the room is not credible even with a single active failure.
- ___ Other, see remarks.

Seismic Design Analysis

- ___ Only SRE is in the room; therefore, there are no seismically induced failures.
- ___ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- X The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- ___ Other, see remarks.

Missile Analysis

- ___ No credible missile sources exist in the room.
- X Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- X External missiles cannot enter the room due to missile protection.
- ___ Other, see remarks.

Pipe Break Analysis

- X There are no high-energy lines in the room.
- ___ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
- X Moderate energy cracks within the room do not adversely affect SRE in the room.
- ___ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 11A) (See Note 5)

Listing of items in room 1111

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipe	EJ-21-ECB-3"	X	Y	
S2	Pipes	EF-47-HBC-4"	X	Y	
		EF-48-HBC-4"	X	Y	
S3	Pipes	EG-23-HBC-1"	X	Y	
		EG-24-HBC-1"	X	Y	
S4	Conduit	1U3C1S	X	Y	
		1U1023,1048	X	Y	
S5	Conduit	1U3C1J,1K	X	Y	
S6	Conduit	1U3C1L,1M	X	Y	
S7	Conduit	4U1041	X	Y	
S8	Pipe	EJ-15-ECB-10"	X	Y	
S9	Pipes	EJ-3-ECB-14"	X	Y	
		EJ-4-HCB-14"	X	Y	
		EJ-5-ECB-14"	X	Y	
S10	Conduit	1B2B1N	X	Y	
	tray	1B2B	X	Y	
S11	Pipe	EJ-69-ECB-3/4"	X	Y	
S12	Valves	EF-V010,037	X	Y	
		EF-V038	X	Y	
S13	M.O. valve	EJ-HV-8812A	X	Y	
S14	M.O. valve	EJ-FCV-610	X	Y	
S15	Valve	EJ-8724A	X	Y	
S16	RHR pump	PEJ01A	X	Y	
S17	Room cooler	SGL10A	X	Y	
S18	Instrument	EJ-PI-601	X	Y	
S19	Conduit	1U1022,1049	X	Y	
	switch	GL-HIS-9	X	Y	
S20	Valve	EJ-8958A	X	Y	
S21	Instrument	EJ-PT-614	X	Y	
N1	Pipe	LF-116-HCD-4"		S	
		LF-124-HCD-4"		S	
		LF-432-HCD-4"		S	
N2	Monorail			Y	
N3	Conduit	5U1C1C,1L		S	
		6U1C1B		S	
N4	HVAC duct	6"		S	
N5	Sump	PLF01A		N	Rotating part
	pumps	PLF01B		N	totally contained
N6	Pipes	LF-175-HCD-2"		S	
		LF-176-HCD-2"		S	
		LF-177-HCD-2½"		S	
N7	Exit register			N	Does not adversely affect SRE
N8	Switch	GL-TSH-53		N	Does not adversely affect SRE

TABLE 3.B-1 (Sheet 12) (See Note 5)

Room Number 1112 Title Containment Spray Pump Room A

Design Approach

- ___ Only safety-related equipment (SRE) is in the room.
- ___ Only nonsafety-related equipment (NSRE) is in the room.
- ___ Minimize SRE in the room and segregate from NSRE.
- X Minimize NSRE in the room and segregate from SRE.
- ___ Other, see remarks.

Flooding Analysis

- X Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
- X Flooding from sources external to the room is not credible even with a single active failure.
- ___ Other, see remarks.

Seismic Design Analysis

- ___ Only SRE is in the room; therefore, there are no seismically induced failures.
- ___ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- X The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- ___ Other, see remarks.

Missile Analysis

- X No credible missile sources exist in the room.
- ___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- X External missiles cannot enter the room due to missile protection.
- ___ Other, see remarks.

Pipe Break Analysis

- X There are no high-energy lines in the room.
- ___ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
- X Moderate energy cracks within the room do not adversely affect SRE in the room.
- ___ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 12A) (See Note 5)

Listing of items in room 1112

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Conduit	1U3C1A,1B		Y	
S2	Conduit	1U1020,1021		Y	
		1U3C1R		Y	
	J-box	GLHISB		Y	
S3	Pipes	EF-045-HBC-4"		Y	
		EF-046-HBC-4"		Y	
S4	Pipes	EN-10-HCB-3"		Y	
		EN-11-HCB-3"		Y	
		EN-04-GCB-3"		Y	
		EN-56-GCB-4"		Y	
S5	Pipe	EN-14-GCB-4"		Y	
S6	Pipe	EN-15-HCB-12"		Y	
S7	Pipes	EN-01-HCB-12"		Y	
		EN-03-GCB-10"		Y	
S8	Conduit	1B2C1P		Y	
	tray	1B2C		Y	
S9	Valves	EN-V-003,004		Y	
		EN-HV-04		Y	
S10	Valves	EN-V-005		Y	
		EN-V-024		Y	
		EN-V-098		Y	
S11	Valve	EN-V-014		Y	
S12	Instrument	EJ-PT-614	X	Y	
S13	Instruments	EN-PI-2,4		Y	
S14	Instrument	EN-FT-13		Y	
S15	CS pump	PEN01A		Y	
S16	Room cooler	SGL13A		Y	
S17	Instrument	EN-FT-5		Y	
S18	Eductor	SEN01A		Y	
N1	(Deleted)				
N2	Pipe	EN-15-GCD-4"		S	
N3	Conduit	5J1002,1003		S	
N4	Conduit	5J1004,1005		S	
N5	Conduit	5J1006,1007		S	
N6	Pipe	LF-116-HCD-4"		S	
N7	Ducts			S	
	10x6, 6x6			S	
N8	Monorail	HKF18B		Y	
N9	Conduit	5U1C1K		S	
N10	Instruments	GL-TSH-54		N	Does not adversely
		EN-FI-13B		N	affect SRE

TABLE 3.B-1 (Sheet 13) (See Note 5)

Room Number 1113

Title Safety Injection Pump Room A

Design Approach

____	Only safety-related equipment (SRE) is in the room.
____	Only nonsafety-related equipment (NSRE) is in the room.
____	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
____	Other, see remarks.

Flooding Analysis

X	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X	Flooding from sources external to the room is not credible even with a single active failure.
____	Other, see remarks.

Seismic Design Analysis

____	Only SRE is in the room; therefore, there are no seismically induced failures.
X	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
____	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
____	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
____	Other, see remarks.

Missile Analysis

X	No credible missile sources exist in the room.
____	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
____	Other, see remarks.

Pipe Break Analysis

X	There are no high-energy lines in the room.
____	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
X	Moderate energy cracks within the room do not adversely affect SRE in the room.
____	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 13A) (See Note 5)

Listing of items in room 1113

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Tray	1U3C	X	Y	
S2	Conduit	1U3C1C,1D,1E		Y	
		1U3C1F,1V,1W		Y	
S3	Conduit	1U3C1N,1P		Y	
		1U3C1T,1U		Y	
S4	Conduit	1U3C1J,1K,1L		Y	
		1U3C1M,1S		Y	
S5	Conduit	1B2D1Q		Y	
	tray	1B2D		Y	
S6	Conduit	1U3C1Q,1U1010		Y	
	box	GLHIS10		Y	
S7	Pipes	EG-023-HBC-1"	X	Y	
		EG-024-HBC-1"	X	Y	
S8	Pipes	BG-217-HCC-3"	X	Y	
		BG-451-HCC-2"	X	Y	
	Instrument	BG-FT-110	X	Y	
S9	Valve	BG-HV-8104	X	Y	
S10	Pipes	BG-240-HCB-3"	X	Y	
		BG-248-HCB-2"	X	Y	
S11	Pipes	EM-2-HCB-6"		Y	
		BN-33-HCB-8"		Y	
		EM-1-HCB-8"		Y	
S12	Pipes	EM-23-HCB-6"		Y	
		EM-22-HCB-6"		Y	
		EM-4-HCB-6"		Y	
S13	Pipes	EG-018-HBC-2"		Y	
		EG-019-HBC-2"		Y	
S14	Pipes	EG-015-HBC-2"	X	Y	
		EG-026-HBC-3"	X	Y	
S15	Pipes	EF-041-HBC-4"		Y	
		EF-042-HBC-4"		Y	
S16	Pipes	EM-6-CCB-4"		Y	
		EM-47-CCB-4"		Y	
S17	Pipes	EM-36-CCB-1 1/2"		Y	
		EM-44-CCB-1 1/2"		Y	
		EM-45-CCB-3"		Y	
S18	Valves	HV-8807A		Y	
		HV-8923A		Y	
		HV-8926		Y	
S19	Valve	EN-HV-8806A		Y	
S20	Valve	HV-8814B		Y	
S21	Valves	EM-8922A		Y	
		EM-HV-8814A		Y	
S22	Valve	EM-HV-8821A		Y	
S23	Valve	8921A		Y	
S24	A.O. valve	BG-FCV-110A	X	Y	
S25	SI pump			Y	
S26	Room cooler	SGL09A		Y	
S27	Instrument	EM-PT-919		Y	
S28	Instrument	EM-FT-918		Y	
S29	Instrument	BG-FT-183	X	Y	
S30	Instrument	EM-PI-977		Y	
N1	Conduit	5J1012,13,14,25		S	
N2	Conduit	6J1007,6J1C1A		S	
		6U5M2B,2C		S	
		5U5K1H,5J1C1E		S	
N3	Conduit	5U1C1K		S	
N4	(Deleted)				
N5	Pipes	BG-247-HCD-2"		S	Structural integrity only
		HE-040-HCD-1"		S	
		HE-039-HCD-1"		S	Structural integrity only
N6	Monorail			Y	
N7	Pipe	LF-422-HCD-1"		S	Structural integrity only

TABLE 3.B-1 (Sheet 14) (See Note 5)

Room Number 1114 Title Centrifugal Charging Pump Room A

Design Approach

____	Only safety-related equipment (SRE) is in the room.
____	Only nonsafety-related equipment (NSRE) is in the room.
____	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
____	Other, see remarks.

Flooding Analysis

X	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X	Flooding from sources external to the room is not credible even with a single active failure.
____	Other, see remarks.

Seismic Design Analysis

____	Only SRE is in the room; therefore, there are no seismically induced failures.
X	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
____	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
____	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
____	Other, see remarks.

Missile Analysis

____	No credible missile sources exist in the room.
X	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
____	Other, see remarks.

Pipe Break Analysis

____	There are no high-energy lines in the room.
X	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 6 with break locations shown in Figure 3.6-1.
X	Moderate energy cracks within the room do not adversely affect SRE in the room.
____	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

TABLE 3.B-1 (Sheet 14A) (See Note 5)

Listing of items in room 1114

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Tray	1U3D	X	Y	
S2	Conduit	1B2E1R	X	Y	
	tray	1B2E	X	Y	
S3	Conduit	1U3D1C,1D	X	Y	
S4	Conduit	4U1014,1015	X	Y	
S5	Conduit	1U1024,1025	X	Y	
		1U3D1E	X	Y	
	J-box	GL-HIS-11	X	Y	
S6	Conduit	1U3D1A,1B	X	Y	
S7	Instrument	BG-PI-118	X	Y	
S8	Instrument	BG-PI-187	X	Y	
S9	Pipes	BG-148-HCB-6"	X	Y	
		BG-146-HCB-8"	X	Y	
S10	Pipe	BN-17-HCB-8"	X	Y	
S11	Pipe	EG-16-HBC-2 1/2"	X	Y	
		EG-17-HBC-2 1/2"	X	Y	
S12	Pipe	EF-037-HBC-4"	X	Y	
		EF-038-HBC-4"	X	Y	
S13	Pipes	BG-149-BCB-4"	X	Y	
		BG-150-BCB-2"	X	Y	
		BG-153-BCB-2"	X	Y	
		BG-154-BCB-2"	X	Y	
		BG-155-BCB-2"	X	Y	
		BG-157-BCB-4"	X	Y	
S14	Valves	BG-8546	X	Y	
		BN-LCV-112D	X	Y	
S15	Valve	BG-HV-8111	X	Y	
S16	Valve	BG-HV-8110	X	Y	
S17	Valve	BG-8481A	X	Y	
S18	C.C. pump	PBG05A	X	Y	
S19	Room cooler		X	Y	
N1	Conduit	5U1C1M		S	
	box	GL-TSH-56		S	
N2	Monorail	SGL12A		Y	
N3	Pipes	LF-125-HCD-4"		S	
		LF-126-HCD-4"		S	
		LF-127-HCD-4"		S	

TABLE 3.B-1 (Sheet 15) (See Note 5)

Room Number 1115 | Title Normal Charging Pump Room

Design Approach

- ____ Only safety-related equipment (SRE) is in the room.
- ____ Only nonsafety-related equipment (NSRE) is in the room.
- ____ Minimize SRE in the room and segregate from NSRE.
- _X_ Minimize NSRE in the room and segregate from SRE.
- ____ Other, see remarks.

Flooding Analysis

- ____ Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
- ____ Flooding from sources external to the room is not credible even with a single active failure.
- _X_ Other, see remarks.

Seismic Design Analysis

- ____ Only SRE is in the room; therefore, there are no seismically induced failures.
- ____ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ____ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- _X_ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- ____ Other, see remarks.

Missile Analysis

- ____ No credible missile sources exist in the room.
- _X_ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- _X_ External missiles cannot enter the room due to missile protection.
- ____ Other, see remarks.

Pipe Break Analysis

- ____ There are no high-energy lines in the room.
- _X_ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 7 with break locations shown in Figure 3.6-1.
- _X_ Moderate energy cracks within the room do not adversely affect SRE in the room.
- ____ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Safety-related equipment in this room has passive function. Flooding will not compromise this function.

TABLE 3.B-1 (Sheet 15A) (See Note 5)

Listing of items in room 1115

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipes	BG-269-HBC-3"	X	Y	
		BG-270-HBC-3"	X	Y	
S2	N.C. pump	PBG04		Y	
S3	Pipe	BG-020-HCB-4"	X	Y	
S4	Pipe	BG-021-BCB-3"	X	Y	
N1	Coil unit	SGL07		S	
N2	Valves	HE-V144		S	
		HE-V195		S	
N3	Floor drains			S	
N4	Pump drains			N	Does not adversely affect SRE
N5	Conduit	6U5M2E		S	
	J-box	GL-TS-12		S	
N6	Conduit	5U5K1A,1B		S	
N7	Conduit	5GED1A		S	
N8	Pipe	KA-341-JDD-1"		S	
N9	HVAC duct	MHO-1029		S	
N10	Conduit	BG-FT-121		S	
N11	Monorail			Y	

TABLE 3.B-1 (Sheet 16) (See Note 5)

Room Number 1116 Title Boric Acid Tank Room B

Design Approach

	Only safety-related equipment (SRE) is in the room.
	Only nonsafety-related equipment (NSRE) is in the room.
	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
	Other, see remarks.

Flooding Analysis

X	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X	Flooding from sources external to the room is not credible even with a single active failure.
	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
X	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
	Other, see remarks.

Missile Analysis

X	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
	Other, see remarks.

Pipe Break Analysis

X	There are no high-energy lines in the room.
	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
	Moderate energy cracks within the room do not adversely affect SRE in the room.
X	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) The only SRE in this room to be protected is the essential service water piping.

2) The boric acid storage and transfer equipment is not qualified for service post-LOCA or post-SSE. For these cases the Boron Injection Header (BIH) is used. The only accident for which the BA storage and transfer equipment is required to function is a high or moderate energy pipe failure in the BIH room. The accident scenario does not postulate turbinetrip or loss of offsite power, therefore, the BA transfer pumps are available.

Additionally, an SSE is not postulated so that no missiles are generated which could cause failure of the BA storage and transfer equipment. Following the BIH room pipe break, redundant BA storage and transfer systems are available to circumvent the consequences of a single active failure.

TABLE 3.B-1 (Sheet 16A) (See Note 5)

Listing of items in room 1116

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipe	EF-076-HBC-24"	X	Y	
		EF-080-HBC-24"	X	Y	
S2	Pipe	BG-221-HCC-2"		Y	
		BG-289-HCC-1"		Y	
		BG-388-HCC-3"		Y	
		BG-232-HCC-2"		Y	
S3	BA tank	TBG03B		Y	
S4	BA transfer pump	PBG02B		N	Not required post-SSE. See pipe break Analysis
S5	Instruments	LT-105,106		Y	
S6	Conduit	4J1003		Y	
S7	Pipe	BG-229-HCC-3"		Y	
		BG-231-HCC-3/4"		Y	
		BG-232-HCC-2"		Y	
		BG-230-HCC-3"		Y	
		BG-216-HCC-3"		Y	
		BG-238-HCC-2"		Y	
N1	Tray	563A		S	
N2	Pipes	LF-223-HCD-4"		S	
		LF-225-HCD-4"		S	
		HE-025-HCD-2"		S	
N3	Stairway and			S	
	platform			S	
N4	All other NSRE			N	Does not adversely affect SRE

TABLE 3.B-1 (Sheet 17) (See Note 5)

Room Number 1117 | Title Boric Acid Tank Room A

Design Approach

____ Only safety-related equipment (SRE) is in the room.
____ Only nonsafety-related equipment (NSRE) is in the room.
____ Minimize SRE in the room and segregate from NSRE.
X Minimize NSRE in the room and segregate from SRE.
____ Other, see remarks.

Flooding Analysis

X Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
X Flooding from sources external to the room is not credible even with a single active failure.
____ Other, see remarks.

Seismic Design Analysis

____ Only SRE is in the room; therefore, there are no seismically induced failures.
X The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
____ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
____ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
____ Other, see remarks.

Missile Analysis

X No credible missile sources exist in the room.
____ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X External missiles cannot enter the room due to missile protection.
____ Other, see remarks.

Pipe Break Analysis

____ There are no high-energy lines in the room.
____ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet with break locations shown in Figure 3.6-1.
____ Moderate energy cracks within the room do not adversely affect SRE in the room.
X Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) The only SRE in this room to be protected is the essential service water piping.

2) See the analysis for Room 1116.

TABLE 3.B-1 (Sheet 17A) (See Note 5)

Listing of items in room 1117

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Conduit	4J1004,1J1004		Y	
S2	BA tank	TBG03A		Y	
S3	BA transfer pump	PBG02A		N	Not required post-SSE. See pipe break analysis
S4	Pipe	EF-076-HBC-24"	X	Y	
S5	Pipes	BG-237-HCC-2"		Y	
		BG-238-HCC-2"		Y	
		BG-219-HCC-2"		Y	
		BG-216-HCC-2"		Y	
		BG-218-HCC-3/4"		Y	
S6	Pipe	BG-215-HCC-3"		Y	
S7	Instrument	LT-104,102		Y	
S8	Pipes	BG-384-HCC-3"		Y	
		BG-223-HCC-2"		Y	
		BG-385-HCC-1"		Y	
S9	Pipe	BG-252-HCB-1"		Y	
N1	Pipes	FB-081-HBD-2"		N	Does not adversely affect SRE
		FB-082-HBD-2"		N	
N2	Stairway and platform			S	
N3	All other NSRE			N	Does not adversely affect SRE

TABLE 3.B-1 (Sheet 18) (See Note 5)

Room Number 1119

Title Stairway A-1

Design Approach

	Only safety-related equipment (SRE) is in the room.
X	Only nonsafety-related equipment (NSRE) is in the room.
	Minimize SRE in the room and segregate from NSRE.
	Minimize NSRE in the room and segregate from SRE.
	Other, see remarks.

Flooding Analysis

	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
	Flooding from sources external to the room is not credible even with a single active failure.
X	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
X	Other, see remarks.

Missile Analysis

	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
	External missiles cannot enter the room due to missile protection.
X	Other, see remarks.

Pipe Break Analysis

X	There are no high-energy lines in the room.
	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
	Moderate energy cracks within the room do not adversely affect SRE in the room.
X	Other, see remarks.

Remarks:

1) There is no SRE in the room.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 19) (See Note 5)

Room Number 1120 Title General Floor Area No. 2

Design Approach

	Only safety-related equipment (SRE) is in the room.
	Only nonsafety-related equipment (NSRE) is in the room.
	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
	Other, see remarks.

Flooding Analysis

	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
	Flooding from sources external to the room is not credible even with a single active failure.
X	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
X	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
	Other, see remarks.

Missile Analysis

X	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
	Other, see remarks.

Pipe Break Analysis

X	There are no high-energy lines in the room.
	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
X	Moderate energy cracks within the room do not adversely affect SRE in the room.
	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Flooding from any source does not adversely affect SRE because all SRE is routed above the maximum design flood depth of 7 feet (El. 1981).

TABLE 3.B-1 (Sheet 19A) (See Note 5)

Listing of items in room 1120

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipes	EG-051-HBC-6"	X	Y	
		BG-032-HBC-6"	X	Y	
		BG-34-HBC-6"	X	Y	
		BG-35-HBC-6"	X	Y	
		EG-052-HBC-6"	X	Y	
S2	Valves	V-071,V-021	X	Y	
		TV-130,V-309	X	Y	
		V-205	X	Y	
S3	Flow element	EF-FE-58	X	Y	
	from pipe	EF-117-HBC-14"	X	Y	
N1	Conduit	6J4A1C		S	
		6U5A1E		S	
		5U1C1H		S	
N2	Tray	6U5A,5B		S	
N3	Pipes	LF-052-HCD-6"		S	
		LF-115-HCD-4"		S	
N4	Monorail			Y	
N5	All other NSRE			N	Does not adversely affect SRE

TABLE 3.B-1 (Sheet 20) (See Note 5)

Room Number 1121 Title Access Pit (To Containment Spray Pump Rooms)

Design Approach

___ Only safety-related equipment (SRE) is in the room.
___ Only nonsafety-related equipment (NSRE) is in the room.
___ Minimize SRE in the room and segregate from NSRE.
_X_ Minimize NSRE in the room and segregate from SRE.
___ Other, see remarks.

Flooding Analysis

___ Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
___ Flooding from sources external to the room is not credible even with a single active failure.
_X_ Other, see remarks.

Seismic Design Analysis

___ Only SRE is in the room; therefore, there are no seismically induced failures.
___ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
_X_ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
___ Other, see remarks.

Missile Analysis

_X_ No credible missile sources exist in the room.
___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
_X_ External missiles cannot enter the room due to missile protection.
___ Other, see remarks.

Pipe Break Analysis

_X_ There are no high-energy lines in the room.
___ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
_X_ Moderate energy cracks within the room do not adversely affect SRE in the room.
___ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Room contains no safety-related equipment which would be adversely affected by flooding up to the maximum design flood depth (El. 1981).

TABLE 3.B-1 (Sheet 20A) (See Note 5)

Listing of items in room 1121

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipe	EF-117-HBC-1"	X	Y	
S2	Marine doors		X	Y	
N1	All NSRE			N	Does not adversely affect SRE

TABLE 3.B-1 (Sheet 21) (See Note 5)

Room Number 1122 Title General Floor Area No. 3

Design Approach

_____ Only safety-related equipment (SRE) is in the room.
_____ Only nonsafety-related equipment (NSRE) is in the room.
_____ Minimize SRE in the room and segregate from NSRE.
X Minimize NSRE in the room and segregate from SRE.
_____ Other, see remarks.

Flooding Analysis

_____ Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
_____ Flooding from sources external to the room is not credible even with a single active failure.
X Other, see remarks.

Seismic Design Analysis

_____ Only SRE is in the room; therefore, there are no seismically induced failures.
X The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
_____ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
_____ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
_____ Other, see remarks.

Missile Analysis

X No credible missile sources exist in the room.
_____ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X External missiles cannot enter the room due to missile protection.
_____ Other, see remarks.

Pipe Break Analysis

_____ There are no high-energy lines in the room.
X The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 9 with break locations shown in Figure 3.6-1.
X Moderate energy cracks within the room do not adversely affect SRE in the room.
_____ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Flooding from any source does not adversely affect SRE because all SRE is located above the maximum flood depth of 7 feet (El. 1981).

TABLE 3.B-1 (Sheet 21A) (See Note 5)

Listing of items in room 1122

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipes	EF-099-HBC-8"	X	Y	
		EF-111-HBC-6"	X	Y	
		EF-108-HBC-4"	X	Y	
S2	Pipes	EF-100-HBC-4"	X	Y	
		EF-101-HBC-4"	X	Y	
S3	Conduit	1J3B1A		Y	BAT level indication;
		1J1004			not required for SSD
S4	Conduit	1U3E1A	X	Y	
S5	HVAC emer exhaust			Y	
S6	Pipes	EM-76-BCB-1"	X	Y	
		EM-74-BCB-4"	X	Y	
		EM-79-BCB-4"	X	Y	
S7	Pipes*	EM-107-HCC-1"*			
		EM-103-BCB-1"*			
S8	Pipes	EF-062-HBC-4"	X	Y	
		EF-055-HBC-4"	X	Y	
		EF-056-HBC-4"	X	Y	
		EF-054-HBC-8"	X	Y	
S9	(Deleted)				
S10	(Deleted)				
S11	Door to turb bldg. El. 1974' column A1/AK			Y	
N1	Trays	5U1C,5J1C		S	
		5U5M		S	
		5U5M01,02,03		S	
N2	Pipes	HB-291-HCD-3"		S	
		HB-211-HCD-3"		S	
		AN-042-HCD-3"		S	
N3	Pipe	KA-003-JBD-8"		S	
N4	(Deleted)				
N5	(Deleted)				
N6	Pipes	FB-050-HBD-3"		S	
		FB-095-HBD-3"		S	
N7	Pipe	KA-023-JBD-6"		S	
N8	Trays (North/South)	5U1C,5J1B		S	
		5U5J,5K		S	
		5J1C		S	
N9	Trays	5U1C,5J1B		S	
		5U5J,5K		S	
		5J1C			
N10	HVAC duct	(South East)		S	
N11	Pipe	BM-305-GBD-3"		S	
		BL-25-HCD-1"		S	
N12	Pipe	LF-022-HCD-4"		S	
N13	Pipes	KA-024-JBD-2 1/2"		S	
		KA-148-JBD-1"		S	
N14	HVAC duct	(South End)		S	
N15	Conduit	6U1C1F,1G		S	
N16	Pipe domestic water	1"		N	Does not adversely affect SRE
N17	Trays	5A3C		S	
		5G3D		S	

* These pipes have been permanently removed from service.

TABLE 3.B-1 (Sheet 22) (See Note 5)

Room Number 1123

Title Letdown Heat Exchanger Passageway

Design Approach

	Only safety-related equipment (SRE) is in the room.
	Only nonsafety-related equipment (NSRE) is in the room.
	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
	Other, see remarks.

Flooding Analysis

	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
	Flooding from sources external to the room is not credible even with a single active failure.
X	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
X	Other, see remarks.

Missile Analysis

	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
	External missiles cannot enter the room due to missile protection.
X	Other, see remarks.

Pipe Break Analysis

X	There are no high-energy lines in the room.
	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
	Moderate energy cracks within the room do not adversely affect SRE in the room.
X	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) SRE in room associated with letdown flow path which is not required for SSD.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 22A) (See Note 5)

Listing of items in room 1123

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Instruments	BG-FT-132		Y	Letdown path not required
		BG-PT-131		Y	
		BG-TIS-129		Y	Letdown path not required
S2	Pipe	EF-163-HBC-1"	X	Y	
N1	Conduit	5U1001		N	Letdown path not required
		5J1016,17,18		N	
		6J1001		N	Letdown path not required
N2	Pipe	KC-447-KBF-8"		N	Letdown path not required
N3	HVAC duct			N	Letdown path not required

TABLE 3.B-1 (Sheet 23) (See Note 5)

Room Number 1124 | Title Letdown Heat Exchanger Valve Compartment

Design Approach

- ___ Only safety-related equipment (SRE) is in the room.
- ___ Only nonsafety-related equipment (NSRE) is in the room.
- ___ Minimize SRE in the room and segregate from NSRE.
- X Minimize NSRE in the room and segregate from SRE.
- ___ Other, see remarks.

Flooding Analysis

- ___ Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
- ___ Flooding from sources external to the room is not credible even with a single active failure.
- X Other, see remarks.

Seismic Design Analysis

- ___ Only SRE is in the room; therefore, there are no seismically induced failures.
- ___ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- ___ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- X Other, see remarks.

Missile Analysis

- ___ No credible missile sources exist in the room.
- ___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- ___ External missiles cannot enter the room due to missile protection.
- X Other, see remarks.

Pipe Break Analysis

- ___ There are no high-energy lines in the room.
- ___ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4. With break locations shown in Figure 3.6-1.
- ___ Moderate energy cracks within the room do not adversely affect SRE in the room.
- X Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) SRE in room associated with letdown flowpath which is not required for SSD.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 23A) (See Note 5)

Listing of items in room 1124

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipes	BG-11-ECB-3" BG-12-ECB-3" BG-29-ECB-3" BG-30-ECB-3" BG-31-ECB-2"		Y Y Y Y Y	Letdown path not required Letdown path not required Letdown path not required
S2	Pipes	BG-13-GCB-3" BG-37-GCB-3" BG-308-GCB-2" BG-040-GCB-3" BG-036-ECB-2"		Y Y Y Y Y	Letdown path not required Letdown path not required Letdown path not required
S3	Valve	TCV-129		Y	Letdown path not required
S4	Instruments	PT-131 TIS-129 FE-132		Y Y Y	Letdown path not required Letdown path not required
N1	Pipe	BG-015-GCD-3"		N	Letdown path not required
N2	Pipe	LF-105-HCD-4"		N	Letdown path not required
N3	Pipe	KA-356-JDD-1"		N	Letdown path not required
N4	HVAC duct			N	Letdown path not required
N5	Conduit	5U1001,02 6J1001,02,03 6U5A1E 6J4A1C 5U1C1H 5J1017		N N N N N N	Letdown path not required Letdown path not required Letdown path not required

TABLE 3.B-1 (Sheet 24) (See Note 5)

Room Number 1125 Title Letdown Heat Exchanger Room

Design Approach

	Only safety-related equipment (SRE) is in the room.
	Only nonsafety-related equipment (NSRE) is in the room.
	Minimize SRE in the room and segregate from NSRE.
X	Minimize NSRE in the room and segregate from SRE.
	Other, see remarks.

Flooding Analysis

	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
	Flooding from sources external to the room is not credible even with a single active failure.
X	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
X	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
	Other, see remarks.

Missile Analysis

X	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
X	External missiles cannot enter the room due to missile protection.
	Other, see remarks.

Pipe Break Analysis

	There are no high-energy lines in the room.
X	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 11 with break locations shown in Figure 3.6-1.
X	Moderate energy cracks within the room do not adversely affect SRE in the room.
	Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Flooding from any source does not adversely affect SRE because no SRE within this room is susceptible to external fluid initiated failure.

TABLE 3.B-1 (Sheet 24A) (See Note 5)

Listing of items in room 1125

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Letdown H.X.	EBG01	X	Y	
S2	Pipes	BG-011-ECB-3"		Y	
		BG-012-ECB-3"		Y	
		BG-029-ECB-3"		Y	
		EJ-035-ECB-3"		Y	
		BG-040-GCB-3"		Y	
S3	Pipes	BG-032-HBC-6"	X	Y	
		BG-040-GCB-3"	X	Y	
S4	Pipes	BG-164-BCB-2"	X	Y	
		BG-195-HCB-2"	X	Y	
N1	Pipe	BG-015-GCO-3"		S	
N2	HVAC duct			S	
N3	Monorail			Y	
N4	Conduit	6U5A1E		S	
		6J4A1C		S	
		5U1C1H		S	

TABLE 3.B-1 (Sheet 25) (See Note 5)

Room Number 1126 | Title Boron Injection Room

Design Approach

- [ ] Only safety-related equipment (SRE) is in the room.
- [ ] Only nonsafety-related equipment (NSRE) is in the room.
- [ ] Minimize SRE in the room and segregate from NSRE.
- [X] Minimize NSRE in the room and segregate from SRE.
- [ ] Other, see remarks.

Flooding Analysis

- [ ] Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
- [X] Flooding from sources external to the room is not credible even with a single active failure.
- [X] Other, see remarks.

Seismic Design Analysis

- [ ] Only SRE is in the room; therefore, there are no seismically induced failures.
- [X] The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- [ ] The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- [ ] The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- [ ] Other, see remarks.

Missile Analysis

- [X] No credible missile sources exist in the room.
- [ ] Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- [X] External missiles cannot enter the room due to missile protection.
- [ ] Other, see remarks.

Pipe Break Analysis

- [ ] There are no high-energy lines in the room.
- [ ] The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 13 with break locations shown in Figure 3.6-1.
- [ ] Moderate energy cracks within the room do not adversely affect SRE in the room.
- [X] Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Flooding from pipe failure within this room does not affect safety shutdown because the BATs are available in this accident scenario.

2) See remarks for Room 1116.

TABLE 3.B-1 (Sheet 25A) (See Note 5)

Listing of items in room 1126

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Conduit	1U1026,27,28	X	Y	
S2	Conduit	4U1016,17,18	X	Y	
	J-box	4U3D5H	X	Y	
S3	(Deleted)				
S4	Conduit	4U1019	X	Y	
S5	BIT*	TEM01*			
S6	Boron injection surge tank*	TEM02*			
S7	BI recirc pump*	PEM02A*			
S8	BI recirc pump*	PEM02B*			
S9	(Deleted)				
S10	(Deleted)				
S11	Instruments*	EM-TIS-944,945*			
S12	Instrument	EM-FT-917A,B	X	Y	For PAM indication & pressure boundary
		EM-PT-947			
		EM-FIS-949*			
S13	Valve	HV-8803A	X	Y	
S14	Valve	HV-8803B	X	Y	
S15	Valve*	HV-8870B*			
S16	Valve*	HV-8883*			
S17	Pipe (up to 8803A and B)	EM-074-BCB-4"	X	Y	
		EM-075-BCB-4"	X	Y	
S18	Pipe (downstream of 8803A and B)	EM-79-BCB-4"	X	Y	
S19	Pipes*	EM-107-HCC-1"*			
		EM-105-HCC-1"*			
		EM-111-BCC-1"*			
		EM-112-BCC-1"*			
S20	Pipes*	EM-101-BCB-1"*			
		EM-103-BCB-1"*			
		EM-95-BCB-1"*			
		EM-100-BCB-1"*			
S21	Pipe	EM-76-BCB-1"	X	Y	
S22	Pipe	EM-79-BCB-3/4"	X	Y	
S23	Pipe*	EM-118-HCC-3/4"*			
S24	Pipes*	EM-91-HCC-2"*			
		EM-96-HCC-2"*			
N1	HVAC duct			S	
N2	Heat trace*	3A and 3B*			
N3	Heat trace*	6A and 6B*			
N4	Heat trace*	4A and 4B*			
N5	Heat trace*	5A and 5B*			
N6	Heat trace*	7A and 7B*			
N7	Instrument	EM-FI-3		S	Structural integrity only
N8	Conduit	5U5K1C,5U1006		S	
N9	Conduit	5J1C1B,5J1030		S	
		6U5D1D		S	
N10	Pipe	LF-238-HCD-4"		S	
N11	Pipe	KA-349-JDD-3/4"		S	
N12	(Deleted)				
N13	Pipes	HE-028-HCD-1"		S	Structural integrity only
		EM-113-HCD-1"		S	
N14	Pipe	EM-114-HCD-1"		S	Structural integrity only
N15	Pipe	EM-123-HCD-2"		S	
N16	Ladder			N	Does not adversely affect SRE
N17	Conduit for item N18*				
N18	Instruments*	EM-LIS-946*			
		EM-LIS-948*			

* These components have been permanently removed from service.

TABLE 3.B-1 (Sheet 26) (See Note 5)

Room Number 1127 | Title Stairwell A-2

Design Approach

- ___ Only safety-related equipment (SRE) is in the room.
- ___ Only nonsafety-related equipment (NSRE) is in the room.
- X Minimize SRE in the room and segregate from NSRE.
- ___ Minimize NSRE in the room and segregate from SRE.
- ___ Other, see remarks.

Flooding Analysis

- ___ Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
- ___ Flooding from sources external to the room is not credible even with a single active failure.
- X Other, see remarks.

Seismic Design Analysis

- ___ Only SRE is in the room; therefore, there are no seismically induced failures.
- X The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- ___ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- ___ Other, see remarks.

Missile Analysis

- X No credible missile sources exist in the room.
- ___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- X External missiles cannot enter the room due to missile protection.
- ___ Other, see remarks.

Pipe Break Analysis

- ___ There are no high-energy lines in the room.
- X The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 13 with break locations shown in Figure 3.6-1.
- X Moderate energy cracks within the room do not adversely affect SRE in the room.
- ___ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Flooding from any source does not adversely affect SRE because all SRE is located above the maximum design flood depth of 7 feet (El. 1981).

TABLE 3.B-1 (Sheet 26A) (See Note 5)

Listing of items in room 1127

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Pipes	EF-099-HBC-8"	X	Y	
		EF-108-HBC-4"	X	Y	
S2	Trays	1U1K,1J1L	X	Y	
S3	Instrument	EG-FT-62	X	Y	
S4	Conduit	4U3D5Z	X	Y	
		4U3D5C	X	Y	
S5	Trays	4U3D,3E	X	Y	
		4J3C	X	Y	
S6	Conduit	4J3C1C		Y	
N1	Conduit	6U53DZ		S	
N2	Pipes	FB-050-HBD-3"		N	
		FB-095-HBD-3"		N	
N3	Pipe	HF-107-HBD-3"		N	
N4	Stairs and platforms			Y	
N5	Pipe	BM-305-GBD-3"		S	
N6	Pipes	KC-300-KBF-2 1/2"		S	
		KC-468-KBF-2 1/2"		S	
N7	Pipes	KC-300-KBF-4"		S	
		KC-110-KBF-6"		S	
		KC-510-KBF-2 1/2"		S	
N8	Pipe	KA-351-JDD-1 1/2"		S	
N9	Trays	6J5B		S	
		6U5D,5E		S	

TABLE 3.B-1 (Sheet 27) (See Note 5)

Room Number 1128

Title General Area No. 5 Elev. 1974'

Design Approach

- ___ Only safety-related equipment (SRE) is in the room.
- ___ Only nonsafety-related equipment (NSRE) is in the room.
- X Minimize SRE in the room and segregate from NSRE.
- ___ Minimize NSRE in the room and segregate from SRE.
- ___ Other, see remarks.

Flooding Analysis

- ___ Flooding from sources within the room will affect only equipment within the same train/subsystem; therefore, safe shutdown is not compromised.
- ___ Flooding from sources external to the room is not credible even with a single active failure.
- X Other, see remarks.

Seismic Design Analysis

- ___ Only SRE is in the room; therefore, there are no seismically induced failures.
- X The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
- ___ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
- ___ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
- ___ Other, see remarks.

Missile Analysis

- X No credible missile sources exist in the room.
- ___ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
- X External missiles cannot enter the room due to missile protection.
- ___ Other, see remarks.

Pipe Break Analysis

- ___ There are no high-energy lines in the room.
- X The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 14 with break locations shown in Figure 3.6-1.
- X Moderate energy cracks within the room do not adversely affect SRE in the room.
- ___ Other, see remarks.

Remarks:

See the reverse side for a listing of items located in the room.

1) Flooding from any source does not adversely affect SRE because all SRE is located above the maximum design flood depth of 7 feet (El. 1981).

TABLE 3.B-1 (Sheet 27A) (See Note 5)

Listing of items in room 1128

Item No. (1)	Description	Equipment Designation	Reqd for SSD (2)	Seismic Cat. (3)	Discussion
S1	Tray	1B2F	X	Y	
S2	Tray	4B2F	X	Y	
N1	Pipe	FB-050-HBD-3"		S	
N2	Pipe	HF-107-HBD-3"		S	
N3	Pipe	LE-025-HCD-4"		S	
N4	HVAC duct	6" x 12"		S	
N5	HVAC duct	8" x 8"		S	
N6	Conduit	5U5K1P		S	
N7	Pipe	LE-034-HCD-4"		S	
N8	All other NSRE			N	Does not adversely affect SRE

TABLE 3.B-1 (Sheet 28) (See Note 5)

Room Number 1129 Title Auxiliary Steam Condenser Recovery and Storage Tank Room

Design Approach

	Only safety-related equipment (SRE) is in the room.
X	Only nonsafety-related equipment (NSRE) is in the room.
	Minimize SRE in the room and segregate from NSRE.
	Minimize NSRE in the room and segregate from SRE.
	Other, see remarks.

Flooding Analysis

	Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
	Flooding from sources external to the room is not credible even with a single active failure.
X	Other, see remarks.

Seismic Design Analysis

	Only SRE is in the room; therefore, there are no seismically induced failures.
	The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
	The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
	The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
X	Other, see remarks.

Missile Analysis

	No credible missile sources exist in the room.
	Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
	External missiles cannot enter the room due to missile protection.
X	Other, see remarks.

Pipe Break Analysis

	There are no high-energy lines in the room.
	The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 15 with break locations shown in Figure 3.6-1.
	Moderate energy cracks within the room do not adversely affect SRE in the room.
X	Other, see remarks.

Remarks:

1) There is no SRE in the room.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 29) (See Note 5)

Room Number 1130 | Title North Corridor

Design Approach

____ Only safety-related equipment (SRE) is in the room.
X Only nonsafety-related equipment (NSRE) is in the room.
____ Minimize SRE in the room and segregate from NSRE.
____ Minimize NSRE in the room and segregate from SRE.
____ Other, see remarks.

Flooding Analysis

____ Flooding from sources within the room will affect only equipment within the same train/ subsystem; therefore, safe shutdown is not compromised.
____ Flooding from sources external to the room is not credible even with a single active failure.
X Other, see remarks.

Seismic Design Analysis

____ Only SRE is in the room; therefore, there are no seismically induced failures.
____ The NSRE in the room is seismically restrained, therefore, no seismically induced failures are postulated.
____ The NSRE, which is postulated to fail as a result of an SSE, does not impact SRE.
____ The NSRE, which is postulated to fail as a result of an SSE and impact SRE, does not adversely affect the SRE.
X Other, see remarks.

Missile Analysis

____ No credible missile sources exist in the room.
____ Missiles from rotating components generated within the room contain insufficient energy to escape their equipment housing(s).
____ External missiles cannot enter the room due to missile protection.
X Other, see remarks.

Pipe Break Analysis

____ There are no high-energy lines in the room.
____ The high-energy line breaks have been evaluated and do not adversely affect SRE. The effects of the breaks are discussed in Table 3.6-4, Sheet 16 with break locations shown in Figure 3.6-1.
____ Moderate energy cracks within the room do not adversely affect SRE in the room.
X Other, see remarks.

Remarks:

1) There is no SRE in the room.

2) Postulated hazards have no effect on SSD.

TABLE 3.B-1 (Sheet 30)

Notes:

(1) Item prefix S - Safety-related equipment (SRE)
N - Nonsafety-related equipment (NSRE)

(2) An X denotes that equipment is required for safe shutdown (SSD) of the reactor.

(3) Y - Component is functionally and structurally designed and constructed to meet seismic Category I requirements, as defined in Regulatory Guide 1.29.
N - Component is nonseismic Category I.
S - Component is seismically designed per requirements of position C.2 of R.G. 1.29.

(4) All conduit in the auxiliary building El. 1974', except in Room 1128, is seismically supported.

(5) Per FSAR Section 3B.1 this table is intended to show an example hazards analysis, it will not be updated to reflect the as-built plant. (Ref. RFR 18587A)

TABLE 3B-2 MAIN STEAM/MAIN FEEDWATER ISOLATION VALVE COMPARTMENT DESIGN PARAMETERS

I. Initial Conditions for Analysis*

Temperature	120°F
Pressure	14.7 psia
Relative Humidity	100% (Cases 1a and 2) 70% (Case 1b)
Water Level	0 ft

II. Design Conditions

Temperature	324°F**
Pressure	6.7 psig
Floodwater Level	2.16 ft

* See References 12 and 14 for additional Case 1b assumptions.

** This temperature represents the licensing basis MSLB, equivalent in flow area to a single-ended steam line rupture (1.4 ft^2), with backflow but without superheat effects.

TABLE 3B-3 MASS AND ENERGY RELEASE DATA FOR MAIN STEAM LINE BREAK IN MAIN STEAM/MAIN FEEDWATER ISOLATION VALVE COMPARTMENT

PRESSURE ANALYSIS (Case 1a)

Time (Sec)	Mass Rate (Lbs/Sec)	Enthalpy (Btu/Lb)
0	0	0
0	7886	1200
2.5	6710	1200
5.0	5612	1200
7.5	4880	1200
10.0*	4441	1200

TEMPERATURE ANALYSIS** (Case 1b)

* Peak pressure reached prior to this point.

** Refer to the mass and energy release data of Reference 16.

TABLE 3B-4 MASS RELEASE DATA FOR MAIN FEEDWATER LINE BREAK IN MAIN STEAM/MAIN FEEDWATER ISOLATION VALVE COMPARTMENT

		Header Pressure		Pump Speed	
Time	Break Flow(gpm)	Initial (psia)	Post Break (psia)	PAE01A	PAE01B
0	39,414	1248	1,058.64	93.4	93.4
1	39,402	1248	1,057.89	93.7	93.7
2	39,396	1248	1,057.39	93.99	93.99
3	39,393	1248	1,057.11	94.29	94.29
4	39,415	1248	1,058.16	94.58	94.58
5	39,347	1248	1,059.22	94.88	94.88
6	39,459	1248	1,060.29	95.17	95.18
7	39,481	1248	1,061.38	95.47	95.47
8	39,504	1248	1,062.47	95.76	95.77
9	39,527	1248	1,063.58	96.06	96.06
10	39,550	1248	1,067.70	96.35	96.36
11	39,574	1248	1,065.83	96.65	96.66
12	39,598	1248	1,066.97	96.94	96.95
13	39,621	1248	1,068.12	97.24	97.25
14	39,646	1248	1,069.28	97.53	97.54
15	39,670	1248	1,070.46	97.83	97.84
16	39,695	1248	1,071.64	98.12	98.14
17	39,720	1248	1,072.83	98.42	98.43
18	39,745	1248	1,074.04	98.71	98.73
19	39,770	1248	1,075.25	99.01	99.02
20	39,796	1248	1,076.47	99.3	99.32
21	39,821	1248	1,077.71	99.6	99.62

TABLE 3B-4 (Sheet 2)

22	39,847	1248	1,078.95	99.89	99.91
23	39,874	1248	1,080.21	100.2	100.2
24	39,900	1248	1,081.47	100.5	100.5
25	39,927	1248	1,081.21	100.8	100.8
26	39,927	1248	1,081.21	100.9	100.0
52	40,795	1248	1,132.11	103.6	100.0
67	40,795	1248	1,132.11	103.6	100.0

TABLE 3B-5 SUMMARY OF NODALIZATION MODEL

PRESSURE ANALYSIS (Case 1a)

Compartment	Compartment Volume (Ft^3)	Vent Path (X-Y)	Vent Area (Ft^2)	Flow Coefficient (C)
1	11,2911	1-2	23.31	Variable orifice2
		1-6	587.3	Variable orifice[(2)]
2	11,432[(1)]	2-3	4.91	Variable orifice[(2)]
		2-5	610.4	Variable orifice[(2)]
3	2113	3-4	4.91	Variable orifice[(2)]
4	2113			
5	37,873	5-6	550.0	0.86
		5-7	187.0	0.85
6	37,873	6-8	187.0	0.85
7	3726.32	7-9	198	0.94
8	3726.32	8-10	198	0.94
9	6208.6	9-10	203.14	0.94
		9-ATM.	203.14	0.95
10	6208.6	10-ATM.	203.14	0.95

TEMPERATURE ANALYSIS (Case 1b)

Compartment	Compartment Volume (Ft^3)	Vent Path (X-Y)	Vent Area (Ft^2)	Flow Coefficient (C)
		1-3	203.14	0.82
1(West)	59,098.92[(1)]	1-2	666.81	0.82
2(East)	59,239.92[(1)]	2-3	203.14	0.82
3	Outside Atmosphere			

1 The difference in the volumes between compartments 1 and 2 is due to different volumes of HVAC ductwork.

2 The variable orifice flow coefficient (described in Ref. 2) is calculated internally by the COPDA Code. The calculated coefficient is a function of the ratio of the upstream and downstream pressures and the isentropic exponent (averaged) of the flowing gases (air and steam).

TABLE 3B-6 MISSILES

SUMMARY OF CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

Postulated Missile	Weight (lb)	Thrust Area (in.^{2})	Impact Area (in.^{2})	Impact Velocity (ft/sec)	Kinetic Energy (ft-lb)	Penetration (in.)
Drive shaft*	135	2.40	2.41	171	55,000	13.88
Drive shaft latched to mechanism	1200	12.57	11.04	NA	NA	NA

* The critical missile is the drive shaft alone. It is the limiting case and envelopes the other cases listed.

TABLE 3B-6 (Sheet 2)

PIPING TEMPERATURE ELEMENT ASSEMBLY - MISSILE CHARACTERISTICS

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

Characteristics	"without well"	"with well"
Flow discharge area	0.11 in.^{2}	0.60 in.^{2}
Thrust area	7.1 in.^{2}	9.6 in.^{2}
Missile weight	11.0 lbs	15.2 lbs
Area of impact	3.14 in.^{2}	3.14 in.^{2}
$\frac{\text{Missile Weight}}{\text{Impact Area}}$	3.5 psi	4.84 psi
Velocity	20 ft/sec	120 ft/sec

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

Characteristics	"without well"	"with well"
Flow discharge area	0.11 in.^{2}	0.60 in.^{2}
Thrust area	3.14 in.^{2}	3.14 in.^{2}
Missile weight	4.0 lbs	6.1 lbs
Area of impact	3.14 in.^{2}	3.14 in.^{2}
$\frac{\text{Missile Weight}}{\text{Impact Area}}$	1.27 psi	1.94 in.^{2}
Velocity	75 ft/sec	120 ft/sec

TABLE 3B-6 (Sheet 3)

CHARACTERISTICS OF OTHER MISSILES
POSTULATED WITHIN REACTOR CONTAINMENT

	Reactor Coolant Pump Temperature Element	Instrument Well of Pressurizer	Pressurizer Heaters
Weight	1.86 lbs	5.5 lbs	15 lbs.
Discharge area	0.37 in.^{2}	0.442 in.^{2}	0.61 in.^{2}
Thrust area	0.79 in.^{2}	1.35 in.^{2}	2.4 in.^{2}
Impact area	0.1 in.^{2}	1.35 in.^{2}	2.4 in.^{2}
Missile Weight / Impact Area	18.6 psi	4.1 psi	6.25 psi
Velocity	110 ft/sec	100 ft/sec	55 ft/sec

TABLE 3B-7 EVALUATION OF RCS LOOP BRANCH LINE BREAKS

(See Figure 3.6-3)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
Loop 1 Cold Leg			
Nozzle 5	SIS from Boron Injection (BI) (small break criteria)	BB-05-BCA-1 1/2"	No piping available for whip. Jet impingement from a break in this line will not affect any other lines.
		EM-083-BCA-1 1/2"	No motive force for break in this portion of piping.
Nozzle 10	Normal charging (small break criteria)	BB-004-BCA-3" to BB-8378B (2nd valve) (Pipe whip from both ends to break)	A break in this line will not propagate* to cause a break in any of the following: lines attached to any other loop, the hot leg and crossover leg of Loop 1, the BI line (Nozzle 5), RTD cold leg manifold (Nozzle 7), line EM-87-BCA-1 1/2" to Loop 4, and EM-83-BCA-1 /12" to Loop 1.
	Normal charging upstream of valve BB-8378B (2nd valve) (MEB 3-1 breaks on lines) (small break no LOCA criteria)	BG-24-BCB-3"	A break in this line will not propagate to break: any line directly connected to Loops 1 & 2 which could result in a loss-of-coolant accident or the Loop 2 seal injection lines.
Nozzle 12	Pressurizer spray line (small break criteria)	BB-003-BCA-4" (max propagation = 12.5 in.^{2})	A break in this line will not propagate to cause a break in any of the following: the hot leg, crossover leg, lines connected to other loops, RTD cold leg manifold (Nozzle 7), BI line (Nozzle 5), charging line (Nozzle 10), line EM-087-BCA-1 1/2" to Loop 4, and all of Loop 4.
Nozzle 7	Cold leg RTD thermowell	BB-16-BCA-2"	Ejection of this thermowell will not propagate to cause a break in any other part of Nozzle 1 or Nozzle 2.

* In this context, propagation is defined as the failure of other pipes caused by the initial pipe break.

TABLE 3B-7 (Sheet 2)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
Nozzle 14	Accumulator line (60.132 in^2) (large break criteria)	BB-002-BCA-10 (Pipe whip from both sides of break)	A break in this line will not propagate to cause a break in any line attached to any other loop. The following provision assures that propagation of this break will not cause further breaks whose combined area exceeds 12.03 $in.^2$: Nozzle 10 is designed for jet impingement loads, and Nozzle 12 is protected from pipe whip and designed for jet impingement loads, or All Loop 1 branch piping, except Nozzle 7 and 12, is protected from pipe whip and designed for jet impingement loads.
	Upstream of valve 8948A	EP-03-BCA-10" and EP-26-BCA-6"	A break in this portion of the line will not result in a LOCA. Whip of this pipe will not impact other branch lines which might cause LOCA rupture of them.
Loop 1 - Crossover Leg			
Nozzle 6	Loop drain line (small break criteria)	BB-20-BCA-2"	A break of this line will not affect any other lines.
Nozzle 8	Crossover leg RTD manifold line (small break criteria)	BB-15-BCA-3"	A break in this line will not propagate to any other loop. This line was capped when the RTD Bypass was removed.
Nozzle 2	Flow instrument lines (3/4") (.375 inch hole) (small break criteria)		A break of these lines will not affect any other line.
Loop 1 Hot Leg			
Nozzle 1	Sample connection (2.34-inch hole) (small break criteria)	BB-18-BCB-3/4"	A break in this line will not affect any other line.
Nozzle 4	Hot leg RTD thermowell (small break criteria)	BB-09-BCA-1" BB-10-BCA-1" BB-11-BCA-1"	Ejection of this thermowell will not propagate to the other manifold connections.

TABLE 3B-7 (Sheet 3)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
Nozzle 16	RHR shutdown suction line (large break criteria)	BB-007-BCA-12"	A break of this line will not affect lines to any other loops. The following provision assures that propagation of this break will not cause further breaks whose combined area exceeds 17.32 in.^{2}: protection from pipe whip and design for jet impingement loads is provided for a and (b or c). a. Accumulator line b. All Loop 1 branch lines except the RTD manifold and pressurizer spray line. c. Pressurizer spray line
<u>Loop 2 - Cold Leg</u>			
Nozzle 5	SIS from Boron Injection (small break criteria)	BB-24-BCA- 1 1/2"	No piping is available for whip. Jet impingement from a break in this line will not affect any other lines.
Nozzle 12	Pressurizer spray line (9.283 in.^{2})	BB-023-BCA-4"	A break in this line will not propagate to cause a break in any line attached to any other loop, the hot leg or crossover leg of Loop 2, the RTD cold leg manifold (Nozzle 7), or Nozzle 5. In the specific area of this pipe routing to the pressurizer, propagation of this break to the normal charging line to Loop 1, all of Loop 1, all of Loop 4, or the Boron Injection line to Loop 3 will not occur.
Nozzle 7	Cold leg RTD thermowell (small break criteria)	BB-34-BCA-2"	Ejection of this thermowell will not propagate to cause a break in any of the flow taps on the crossover leg. <u>No other pipes are affected.</u>
Nozzle 14	Accumulator line (60.132 in.^{2}) (large break criteria)	BB-022-BCA-10" (Pipe whip from both ends of break)	A break in this line will not propagate to cause a break in any line attached to any other loop. The following provision assures that propagation of this break will not cause further breaks whose combined area exceeds 12.03 in.^{2}: protection from pipe whip and design for jet impingement loads is provided for a and (b or c). a. RHR and SI hot leg recirculation

TABLE 3B-7 (Sheet 4)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
			b. All Loop 2 branch lines except the pressurizer spray line and the RTD manifold
			c. Pressurizer spray line
	Upstream of valve 8948B (large break, non-LOCA)	EP-06-BCA-10" & EP-28-BCA-6"	A break in this portion of the line will not result in a LOCA.
			Propagation of this break to the crossover leg RTD manifold connection, the charging line, the seal injection line, or the Boron Injection line to Loop 3 will not occur.
<u>Loop 2 Crossover Leg</u>			
Nozzle 6	Loop drain line (small break criteria)	BB-037-BCA-2"	A break of this line will not affect any other lines.
Nozzle 8	Crossover leg RTD manifold line (12.5 in.^{2} total) (small break criteria)	BB-33-BCA-3"	A break of this line will not propagate to any other loop. This line was capped when the RTD Bypass was removed.
Nozzle 2	Flow instrument lines (3/4") (.375 inch hole) (small break criteria)		A break of these lines will not affect any other lines.
<u>Loop 2 Hot Leg</u>			
Nozzle 4	Hot leg RTD thermowell	BB-27-BCA-1" BB-28-BCA-1" BB-39-BCA-1"	Ejection of this thermowell will not propagate to other connections.
Nozzle 13	RHR and SI hot leg recirculation (21.16 in.^{2})(large break criteria)	BB-026-BCA-6"	A break of this line will not propagate to the RTD manifold with the exception of connections (Nozzle 4).
<u>Loop 3 Cold Leg</u>			
Nozzle 5	SIS from Boron Injection (small break criteria)	BB-40-BCA-1 1/2"	No piping is available for whip. Jet impingement from a break of this line will not affect any other lines.
Nozzle 7	Cold leg RTD thermowell	BB-50-BCA-2"	Ejection of this thermowell will not propagate to cause a break in any of the flow taps on the crossover leg. No other pipes are affected.

TABLE 3B-7 (Sheet 5)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
Nozzle 14	Accumulator line (60.132 in.^{2}) (large break criteria)	BB-39-BCA-10" (pipe whip from both ends of break)	A break in this line will not propagate to cause a break in any line attached to any other loop. To assure that propagation of this break will not cause additional breaks whose combined area exceeds 12.03 in.^{2}, protection for pipes on this loop is provided as follows: 1. The RHR and SI hot leg recirculation line Nozzle 13 and the letdown line Nozzle 9 are designed for jet impingement loads. 2. Protection of the remainder of the pipes attached to this loop is not required.
	Upstream of valve 8948C result (large break, non-LOCA)	EP-09-BCA-10" and EP-30-BCA-6"	A break in this portion of the line will not in a LOCA, or propagate to the crossover RTD, letdown line, or seal injection lines.
Loop 3 Crossover leg			
Nozzle 9	Letdown line (small break criteria)	BB-054-BCA-3"	A break of this line will not propagate to any other loop, the hot leg or cold leg of Loop 3, or the RTD mainfold connection on the crossover leg of Loop 3. No protection of the flow taps is required.
Nozzle 8	Crossover leg RTD manifold line (small break criteria)	BB-49-BCA-3"	A break of this line will not propagate to any other loop. This line was capped when the RTD Bypass was removed.
Nozzle 2	Flow instrument lines (3/4") (.375 inch holes) (small break criteria)		A break of these lines will not affect any other line.
Loop 3 Hot Leg			
Nozzle 1	Sample connection (3/4" pipe) (.234 inch hole) (small break criteria)	BB-52-BCB-3/4"	A break of this line will not affect any other line.
Nozzle 4	Hot leg RTD thermowell	BB-43-BCA-1" BB-44-BCA-1" BB-45-BCA-1"	Ejection of this thermowell will not propagate to other connections.
Nozzle 13	RHR and SI hot leg recirculation (21.16 in.^{2}) (large break criteria)	BB-042-BCA-6"	A break of this line will not propagate to the RTD manifold except the hot leg RTD manifold connections (Nozzle 4) or to other lines except the sample line connection (Nozzle 1).

TABLE 3B-7 (Sheet 6)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
Loop 4 Cold Leg			
Nozzle 5	SIS from Boron Injection (small break criteria)	BB-59-BCA-1 1/2"	Jet impingement from a break in this line will not affect any other line. No piping is available for whip.
Nozzle 11	Alternate charging (small break criteria)	BB-57-BCA-3" (pipe whip from both ends of break)	A break in this line will not propagate to cause a break in any line attached to any other loop, the hot leg or crossover leg of Loop 4, the Boron Injection Line (Nozzle 5), or the RTD cold leg manifold (Nozzle 7).
	Upstream of 8379B	BG-25-BCB-3"	A break in this line will not propagate to the normal charging line, the seal injection lines, or any line on the other loops whose rupture could cause a LOCA.
Nozzle 7	Cold leg RTD thermowell	BB-67-BCA-2"	Ejection of this thermowell will not propagate to cause a break in any of the flow taps on the crossover leg. No other pipes are affected.
Nozzle 14	Accumulator line (60.132 in.^{2}) (large break criteria)	BB-58-BCA-10" (pipe whip from both ends of break)	A break in this line will not propagate to cause a break in any line attached to any other loop. A break in this line will not propagate to an additional break area greater than 12.03 in.^{2}. Therefore: 1. The RHR shutdown suction line (Nozzle 16) and pressurizer surge line (Nozzle 15) are designed for jet impingement and are protected from pipe whip. 2. With respect to the small pipes attached to this loop, one of the following combinations of additional line losses may be tolerated: a The entire RTD manifold, boron injection line, and the flow taps or, b. The cold leg RTD manifold connection, alternate charging line, Boron Injection and the flow taps, or c. None of the manifold and all other lines, or d. The entire RTD manifold, excess letdown, and the flow taps.

TABLE 3B-7 (Sheet 7)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
	Upstream of valve 8948D	EP-12-BCA-10"	A break in this portion of the line will not result in a LOCA. In addition, propagation of this break to the RTD manifold, the SI hot leg recirculation line, the RHR shutdown suction line, pressurizer surge line, alternate charging, excess letdown, or seal injection lines will not cause a LOCA.
Loop 4 Crossover Leg			
Nozzle 3	Loop drain with excess letdown (small break criteria)	BB-074-BCA-2"	A break in this line will not propagate to any other loop, the hot or cold leg of Loop 4, the RTD manifold connnection on the crossover leg of Loop 4, or the alternate charging line (non-LOCA portion). The air-operated valves on this line will not whip and impact larger pipes. No protection of the flow taps is required.
Nozzle 8	Crossover leg RTD manifold (small break criteria)	BB-66-BCA-3"	A break of this line will not propagate to any other loop. This Line was capped when the RTD Bypass was removed.
Nozzle 2	Flow instrument lines (3/4") (.375 inch holes) (small break criteria)		A break in these lines will not affect any other line.
Loop 4 Hot Leg			
Nozzle 4	Hot leg RTD Thermowell	BB-61-BCA-1" BB-62-BCA-1" BB-63-BCA-1"	Ejection of this thermowell will not propagate to the remainder of the manifold or the Boron Injection line to Loop 4.
Nozzle 16	RHR shutdown suction line (large break criteria)	BB-070-BCA-12"	A break of this line will not affect the linesto any other loops.
			In order that propagation of this break will not cause further breaks whose combined area exceeds 17.32 $in.^2$, the accumulator line and pressurizer surge line are protected from pipe whip and designed for jet impingement loads.
Nozzle 15	Pressurizer surge line (large break criteria)	BB-069-BCA-14"	A break of this line will not affect lines to any other loops, including pressurizer spray lines.
			The following provision assures that propagation of this break will not cause further breaks whose combined area exceeds 20.77 $in.^2$: protection from pipe whip and design for jet impingement loads is provided for a and (b or c).

TABLE 3B-7 (Sheet 8)

RCS Loop Nozzle Number	Branch Line Description	Branch Line Identification	Pipe Break Evaluation
			a. Accumulator line and the RHR shutdown suction line
			b. All lines other than the 6" SI recirculation line
			c. SI recirculation line connected to the RHR shutdown suction line.

Assuming a non-LOCA break, the RHR and SI cold leg injection lines to Loop 4 are restrained to preclude the loss of the seal injection lines in that area.

TABLE 3B-8 CALLAWAY STEAM TUNNEL HEAT SINKS FOR CASE 1B

Heat Sink	Materia	Node	Surface Area (ft^2)	Thickness (ft)	Boundary Condition Left	Boundary Condition Right
1	Structural Steel	1	4287.13	0.042	Uchida	Adiabatic
2	Structural Steel	2	4300.96	0.042	Uchida	Adiabatic
3	Concrete Floor	1	945.	2.	Uchida	Adiabatic
4	Concrete Floor	2	945.	2.	Uchida	Adiabatic
5	Concrete Column	1	3200.	1.	Uchida	Adiabatic
6	Concrete Column	2	3200.	1.	Uchida	Adiabatic
7	Concrete Column	1	3352.5	2.	Uchida	Adiabatic
8	Concrete Column	2	3352.5	2.	Uchida	Convective
9	Interior Concrete	1	1665.	1.	Uchida	Adiabatic
10	Interior Concrete	2	1665.	1.	Uchida	Adiabatic
11	Concrete Column	1	1639.	2.	Uchida	Adiabatic
12	Concrete Column	2	1639.	2.	Uchida	Adiabatic
13	Concrete Wall	1	1865.	4.	Uchida	Adiabatic
14	Concrete Wall	2	1865.	4.	Uchida	Adiabatic
15	Concrete Roof	1	365.	2.	Uchida	Convective
16	Concrete Roof	2	365.	2.	Uchida	Convective
17	Environmental Heat Sink	3	-	-	Infinite HTC	95°F

TABLE 3B-9 CALLAWAY MAIN STEAM TUNNEL STEAM LINE BREAK ANALYSIS
PEAK TEMPERATURE AND PRESSURE RESULTS FOR CASE 1B

Break Size	Peak Temperature		Peak Pressure	
(ft^2)	Temp (°F)	Time (sec)	Press(psia)	Time (sec)
0.05	199.2	25	14.79	0.32
0.1	238.8	23	14.78	0.32
0.2	409.9	2066	14.77	0.32
0.3	421.2	1846	14.79	0.11
0.4	436.0	1081	14.81	0.11
0.5	443.3	786	14.83	0.11
0.6	448.6	616	14.85	0.11
0.7	452.1	506	14.87	0.11
0.8	456.4	431	14.89	0.11
0.9	459.2	374	14.91	0.11
1.0	460.5	333	14.94	0.11
1.2	448.5	270	15.01	0.22
1.4	453.7	89	15.09	0.22
2.0	453.9	86	15.32	0.22
4.6	454.8	78	16.56	0.32
0.4*	445.0	645		
0.5*	451.2	566		
0.6*	454.4	455		
0.7*	457.3	382		
0.8*	459.5	331		
0.9*	461.2	295		
1.0*	462.7	266		
1.2*	456.9	101		

* Values for Tavg Coastdown. The Tavg Coastdown impacts the available protection function provided by a reactor trip on OPΔT. Therefore only the intermediate break sizes (0.4 through 1.2 ft^2) which credit a trip on OPΔT are analyzed for the Tavg Coastdown. The peak pressures for the Tavg Coastdown cases are not provided since they are bounded by the peak pressures for the larger break sizes of Cases 1a and 1b. (Refer to Section 15.0.2.2 and References 12 and 16.)