3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

The following sections discuss conformance with the NRC "General Design Criteria for Nuclear Power Plants" as specified in Appendix A to 10 CFR 50 effective May 21, 1971, and subsequently amended July 7, 1971, and October 23, 1978. Based on the content herein, the Carolina Power and Light Company believes that the Shearon Harris Nuclear Power Plant fully satisfies and is in compliance with the General Design Criteria.

3.1.1 CRITERION 1 – QUALITY STANDARDS AND RECORDS

CRITERION

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

Structures, systems, and components as listed in Table 3.2.1-1, which are important to safety were designed, fabricated, erected, and tested in accordance with the Engineering and Construction Quality Assurance Program which was approved by the NRC during the Construction Permit review. The structures, systems, and components important to safety are listed in Table 3.2.1-1. The intent of the quality assurance program was to assure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program assured adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. Section 17.3 describes the Quality Assurance Program that will be applied during the operating phase. The total quality assurance program of the applicant and its principal contractors is responsive to and satisfies the quality-related requirements of Title 10 CFR 50, including Appendix B.

Structures, systems, and components are classified, as indicated in Section 3.2, with respect to their safety function to be performed. This classification system was developed by the American Nuclear Society and is in accordance with the "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2, January, 1973, as revised and addended by ANSI N18.2a-1975 and Regulatory Guide 1.26 (See Section 1.8). Recognized codes and standards are applied to the equipment in these classifications as necessary to assure a quality in keeping with the required safety function. In cases where codes are not available or the existing code must be modified, an explanation is provided.

Documents are maintained which demonstrate that the requirements of the quality assurance program are being satisfied. This documentation shows that appropriate codes, standards and regulatory requirements are observed, specified materials are used, correct procedures are utilized, qualified personnel are provided and that the finished parts and components meet the applicable specifications for safe and reliable operation. These documents are maintained in accordance with ANSI N45.2.9, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants," and Regulatory Guide 1.88, "Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records," as described in FSAR Section 1.8.

For further discussion, see the following sections:

1.	Seismic Classification	3.2.1
2.	System Quality Group Classifications (Safety Class)	3.2.2
3.	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	3.10
4.	Environmental Design of Mechanical and Electrical Equipment	3.11
5.	Quality Assurance Program During the Operations Phase	17.3
6.	Instrumentation and Control	7.2.2, 7.4.2

3.1.2 CRITERION 2 – DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

CRITERION

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

DISCUSSION

The structures, systems and components important to safety are protected from or designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. Natural phenomena factored into the design of plant structures, systems and components important to safety were determined from recorded data for the site vicinity with appropriate margin to account for uncertainties in historical data.

The most severe natural phenomena considered in the design in terms of induced stresses are the safe shutdown earthquake (SSE) and the design basis tornado. Those structures, systems and components essential for the mitigation and control of postulated accident conditions are designed to withstand the effects of a loss-of-coolant accident (LOCA) coincident with the effects of the SSE. Structures, systems and components essential to the safe shutdown of the plant are designed to withstand the effects of the most severe natural phenomena, including floods, hurricanes, tornadoes or the SSE, as appropriate.

For further discussion, see the following sections:

a)	Meteorology	2.3
b)	Hydrological Engineering	2.4
c)	Geology, Seismology, and Geotechnical Engineering	2.5
d)	Design of Structures, Components, Equipment, and Systems	3.2 through 3.11
e)	Instrumentation and Control	7.2

3.1.3 CRITERION 3 - FIRE PROTECTION

CRITERION

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 fire 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 design of fire protection of the plant has been guided through consultation with experienced fire protection engineers qualified for membership in the Society of Fire Protection Engineers. They have addressed the criteria and guidance given in NRC regulatory guidelines, National Fire Protection Association Standards, and Nuclear Mutual Limited Property Loss Prevention Standards for Nuclear Generating Stations. Per NRC instructions, the Fire Protection System is designed to Appendix A of SRP 9.5-1 in lieu of Regulatory Guide 1.120 pending resolution of industry comments on Regulatory Guide 1.120.

Noncombustible and fire-resistant materials have been used to the extent practical throughout the plant, particularly in areas containing critical portions of the plant such as the Containment structure, Control Room, cable spreading rooms, and components of systems important to safety. Consistent with other safety requirements, structures, systems and components important to safety are designed and located to minimize the effects of fires on their redundant components. Facilities for the storage of combustible materials such as fuel oil are located, designed, and protected to minimize both the probability and the effects of a fire.

Fire detection and extinguishing systems of appropriate capacity and capability are provided to protect both plant and personnel from fire. Fire Protection System and Fire Detection System reliability is ensured by periodic tests and inspections. Administrative controls are used where applicable throughout the plant to minimize the probability and consequences of fires or explosions.

The Fire Protection System is designed to assure that a failure of any component, or inadvertent operation of the system:

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

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

For further discussion, see the following sections:

a)	Design of Seismic Category I Structures	3.8
b)	Identification of Safety Criteria	7.1.2
c)	Fire Protection for Cable Systems	8.3.3
d)	Fire Protection System	9.5.1
e)	Instrumentation and Control	7.4.2

3.1.4 CRITERION 4 - ENVIRONMENTAL AND DYNAMIC EFFECTS DESIGN BASES

CRITERION

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

DISCUSSION

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the pressure, temperature, humidity, chemical spray and radiation conditions associated with normal operation, maintenance, testing, and postulated accidents including a LOCA, in the area in which they are located.

Structures, systems, and components are designed, arranged or protected such that the external missiles will not cause an accident which could result in the release of significant amounts of radioactivity or prevent safe plant shutdown. Dynamic effects and missiles generated as a result of equipment failures will not cause or increase the severity of an accident which could result in the release of significant amounts of radioactivity or prevent safe plant shutdown. Dynamic effects and missiles resulting from a LOCA will not damage containment integrity or prevent engineered safety features from mitigating the effects of the LOCA.

Failure of high pressure lines external to the Containment will not cause a LOCA, or prevent safe shutdown of the Unit.

The Containment is designed to sustain dynamic loads (such as jet thrust, jet impingement and local pressure transients) which could result from failure of reactor coolant pressure boundary equipment and piping.

For further discussion, see the following sections:

a)	Missile Protection	3.5
b)	Protection Against Dynamic Effects Associated with Postulated Rupture of Piping	3.6
c)	Design of Category I Structures	3.8
d)	Environmental Design of Mechanical and Electrical Equipment	3.11
e)	Reactor Coolant Pump Flywheels	5.4.1
f)	Turbine Disk Integrity	10.2.3
g)	Instrumentation and Control	7.2, 7.4

3.1.6 CRITERION 10 - REACTOR DESIGN

CRITERION

The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specific 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 with adequate margins to:

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

b) Ensure return of the reactor to a safe state following an ANS Condition III event with only a fraction of fuel rods damaged although sufficient fuel damage might occur to preclude immediate resumption of operation.

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

Chapter 4 discusses the design bases and design evaluation of reactor components including the fuel and reactivity control materials. Section 3.9 discusses the design bases and design evaluation of the reactor vessel internals and the control rod drive mechanisms. Details of the control and protection systems instrumentation design and logic are discussed in Chapter 7.

This information supports the accident analyses of Chapter 15 which show that the acceptable fuel design limits are not exceeded for ANS Condition I and II occurrences.

3.1.7 CRITERION 11 - REACTOR INHERENT PROTECTION

CRITERION

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

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the nonpositive full power limit on moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design using low-enrichment fuel; the nonpositive moderator temperature coefficient of reactivity is assured by administratively controlling the dissolved absorber concentration or by burnable poison.

These reactivity coefficients are discussed in Section 4.3.

3.1.8 CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

CRITERION

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 the negative Doppler and nonpositive moderator temperature coefficient of reactivity as described in Section 4.3.

3.1.9 CRITERION 13 - INSTRUMENTATION AND CONTROL

CRITERION

Instrumentation shall be provided to monitor variables and systems over their anticipated range 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, temperatures, pressures, flows, and levels as necessary to assure that adequate plant safety can be maintained. Instrumentation is provided in the Reactor Coolant System (RCS), Steam and Power Conversion System, the Containment, Engineered Safety Features (ESF) Systems, radiological waste systems and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the Control Room in proximity with the controls for maintaining the indicated parameter in the proper range.

The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. For further discussion, see the following sections:

a) Engineered Safety Feature Systems	7.3
b) Systems Required for Safe Shutdown	7.4
c) Safety Related Display Instrumentation	7.5
d) All Other Instrumentation Systems Required for Safety	7.6

3.1.10 CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

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

<u>DISCUSSION</u>

The reactor coolant pressure boundary (RCPB) is designed to accommodate the system pressures and temperatures attained under all expected modes of Unit operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. See Sections 3.9 and 5.2 for details.

RCPB materials selection and fabrication techniques ensure a low probability of gross rupture or significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, as discussed in Section 3.6 and 3.7.

The system is protected from overpressure by means of pressure relieving devices, as required by applicable codes.

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

3.1.11 CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

CRITERION

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

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

Additionally, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components. Chapter 5 discusses the reactor coolant system design. Also, see Section 7.2.2.

3.1.12 CRITERION 16 - CONTAINMENT DESIGN

CRITERION

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

The Containment in conjunction with the Containment Isolation System provides an essentially leak tight barrier, designed to protect the public from the consequences of a LOCA, based on a postulated break of reactor coolant piping up to and including a double ended break of the largest reactor coolant pipe.

The Containment, and associated engineered safety features systems, are designed to safely withstand all internal and external environmental conditions that may be postulated to occur during the life of the plant, including both short and long term effects following a LOCA.

For further discussion, see the following sections:

a) General Plant Description 1.2

b) Concrete Containment 3.8.1

c) Containment Systems 6.2

d) Accident Analyses 15.0

3.1.13 CRITERION 17 - ELECTRIC POWER SYSTEMS

CRITERION

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

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

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two, physically independent circuits (not necessarily on separate rights of way) designed and located 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 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 from the transmission network, or the loss of power from the onsite electric power supplies.

DISCUSSION

Onsite and offsite electrical power systems are provided, and each is designed with adequate independence, capacity, redundancy, and testability to assure the functioning of safety related systems.

Alternate power systems are provided as follows:

a) Two 230 kV sources either of which is capable of supplying power for the engineered safety features in the event of loss of auxiliary transformer power;

b) Two auxiliary transformers;

c) Two start-up transformers; and

d) Two independent diesel generator sources which are each capable of supplying power for the Engineered Safety Feature Systems in the event of a loss of auxiliary transformer power and start-up transformer power.

The Offsite Power System extends from the utility grid to the auxiliary and startup standby transformers.

All emergency and vital equipment, as required to meet the safety function defined above, is redundant, with each division fed from separate and independent emergency buses. Each of these buses receives AC power in the following order of preference:

a) From the onsite source (main generator), via the Unit auxiliary transformer and the plant AC distribution system.

b) If a) is not available, then from the startup/standby transformer and the plant AC distribution system.

c) If neither a) nor b) is available, then from the onsite standby diesel generators. A separate full capacity diesel generator is provided for each emergency bus.

In compliance with Criterion 17:

a) The onsite electrical power systems and those offsite power circuits have the capacity and capability to perform required safety functions.

b) The onsite system has the requisite independence, redundancy, and testability.

c) Each offsite power circuit is available within a few seconds.

d) Design and arrangement is such as to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, loss of one of the three following supplies:

    1) Main generator,

    2) Transmission networks or,

    3) The onsite standby power supply (diesel generators).

For further discussion, see the following sections:

a)	General Plant Description	1.2
b)	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	3.10
c)	Environmental Design of Mechanical and Electric Equipment	3.11
d)	Electric Power	8.0
e)	Instrumentation and Control	7.3.2

3.1.14 CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

CRITERION

Electrical power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features such as wiring, insulation, connections, and switchboards to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems such as onsite power sources, relays, switches, 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

The engineered safety features power supply buses and associated diesel generators are arranged for independent periodic testing of each system. The testing procedure will simulate an accident signal or a loss of emergency bus voltage to start the diesel generator, bring it to operating condition and automatically connect it to the bus. Full load testing of the diesel generator can be performed by manually synchronizing to the normal supply and by connecting essential loads and assuming a portion of the station auxiliary load up to its nameplate rating. These tests, performed periodically in accordance with the Technical Specifications, will prove the operability of the onsite electric power system under conditions as close to design as practical to assess the continuity of the system and condition of the components.

The design of the onsite electrical power system provides testability in accordance with the requirements of Regulatory Guide 1.22 to the extent described in Section 1.8.

For further discussion, see the following sections:

a)	AC Power Systems	8.3.1
b)	DC Power Systems	8.3.2
c)	Initial Test Program	14.0
d)	Technical Specifications	16.0

3.1.15 CRITERION 19 - CONTROL ROOM

CRITERION

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 TEDE, 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 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

Following proven plant design philosophy, the Control Room contains the following equipment: control panels which contain those instruments and controls necessary for operation and surveillance of the plant functions, such as the reactor and its auxiliary systems, engineered safety features, turbine generator, steam and power conversion systems and station electrical distribution boards.

The design of the Control Room permits safe access and occupancy during abnormal conditions including loss-of-coolant accidents. The Control Room will be isolated from the outside atmosphere during the initial period following the occurrence of an accident. Food, water and other habitability systems will be provided for control room personnel during this period. The Control Room Air Conditioning System is designed to maintain the control room air temperature and relative humidity within comfort ranges. Fresh air makeup and air cleanup capabilities are provided to control airborne radioactivity. The air conditioning system includes two redundant emergency air cleanup units, each of which is designed to process the control room air and the makeup fresh air through high efficiency particulate filtration and charcoal adsorption units. Control room shielding and the air conditioning system are designed to limit radiation exposure experienced by an operator to within 5 rem whole body or its equivalent to any part of the body, for the duration of the design basis accident. Design provisions to allow access to the Control Room, without exceeding 5 rem TEDE, has been provided.

The Control Room has low leakage construction features to minimize leakage of hazardous chemicals when the Control Room is isolated following an external release of those chemicals. Means of fresh air intake isolation are provided.

Radiation detectors, alarms and emergency lighting are provided.

Centralized controls and instruments external to the Control Room are provided for equipment required to establish and maintain the plant in a hot shutdown condition.

It will also be possible to attain a cold shutdown condition outside of the Control Room through the use of local controls and suitable procedures.

For further discussions, see the following sections:

a) General Plant Description 1.2

b) Habitability Systems 6.4

c) Engineered Safety Feature Systems 7.3

d) Systems Required for Safe Shutdown 7.4

e) Control Room Area Ventilation System 9.4.1

f)	Fire Protection	9.5.1
g)	Lighting Systems	9.5.3
h)	Shielding	12.3.2
i)	Area Radiation and Airborne Radioactivity Monitoring Instrumentation	12.3.4
j)	Accident Analysis	15.0

3.1.16 CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

CRITERION

The protection systems 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 transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971, IEEE Standard 379-1972 (NSSS), and IEEE Standard 379-1977 (BOP). The Reactor Protection System automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

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

The ESF Actuation System automatically initiates emergency core cooling and other safeguard functions by sensing accident conditions using redundant analog channels measuring diverse variables. Manual actuation of safeguards may be performed where ample time is available for operator action but is not relied upon to satisfy this criterion. The ESF Actuation System automatically trips the reactor on manual or automatic safety injection signal generation.

3.1.17 CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

CRITERION

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 loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing to 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

Protection system inputs and actuating devices which interface with the NSSS logic devices are designed for high functional reliability and inservice testability of the safety function to be performed. Component redundancy and physical and electrical isolation satisfy the single failure criterion with respect to channel independence. The protection system is designed to comply with the requirements of IEEE 279-1971.

Periodic testing of protection system input sensors, logic trains, and actuation devices determines any redundancy losses or functional failures that may have occurred.

Compliance with appropriate codes and standards, reliability, and testing of the protection system logic is discussed in Sections 7.2 and 7.3.

For further discussion see the following chapters:

a) Instrumentation and Control 7.0

b) Technical Specifications 16.0

3.1.18 CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

CRITERION

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

DISCUSSION

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

Protective functions which result in a reactor trip are given on Table 7.2.1-1.

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 condition using conservatively large accelerations and applicable frequencies.

For further discussion see the following chapter:

a) Instrumentation and Control 7.0

3.1.19 CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

CRITERION

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 (i.e., electric power, instrument air), or postulated adverse environments (i.e., extreme heat or cold, fire, pressure, steam, water and radiation) are experienced.

DISCUSSION

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

For further discussion see the following chapters:

a) Engineered Safety Features 6.0

b) Instrumentation and Control 7.0

3.1.20 CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

CRITERION

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

DISCUSSION

The protection system is separate and distinct from the control systems. Control systems may be dependent on the protection system in that control signals are derived from protection system measurements where applicable. These signals are transferred to the control system by isolation devices which are classified as protection components. The 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.

For further discussion see the following chapter:

a) Instrumentation and Control 7.0

3.1.21 CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

CRITERION

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

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

For further discussion, see Instrumentation and Control, Sections 7.3.2 and 7.7.2

3.1.22 CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

CRITERION

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

DISCUSSION

Two reactivity control systems are provided. These are rod cluster control assemblies (RCCA's) and Chemical Shim Control System (boric acid). The RCCA's are inserted into the core by the force of gravity.

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

The Chemical Shim Control System (boric acid) will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

For further discussion see the following chapters:

a) Reactor 4.0

b) Instrumentation and Control 7.0

c) Auxiliary Systems 9.0

d) Accident Analysis 15.0

3.1.23 CRITERION 27 – COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

CRITERION

The reactivity control systems shall be designed to have a combined capability, in conjunction with soluable 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 for making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Chapters 4 and 9. Combined use of the Rod Control System and the Chemical and Volume 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.

For further discussion, see Sections 7.3 and 7.4.

3.1.24 CRITERION 28 – REACTIVITY LIMITS

CRITERION

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

DISCUSSION

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

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

Assurance of core-cooling capability following ANS Condition IV accidents, such as rod ejections and steam line break, is given by keeping the RCPB stresses within faulted condition limits as specified by applicable ASME Codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety features.

3.1.25 CRITERION 29 – PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

CRITERION

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 are designed to assure extremely high probability of performing their required safety functions in any anticipated operational occurrences. Likely failure modes of system components are designed to be safe modes. Equipment used in these systems is designed, constructed, operated, and maintained with a high level of reliability. Loss of power to the protection system results in a reactor trip.

For further discussion see the following chapter:

a) Instrumentation and Control 7.0

3.1.26 CRITERION 30 – QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

CRITERION

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

By using conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Components for the RCPB are designed, fabricated, inspected and tested in conformance with ASME B&PV Code, Section III. All components are classified according to ANSI N18.2-1973 and are accorded the quality measures appropriate to the classification. The design bases and evaluations of RCPB components are discussed in Chapter 5. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Criterion 14, "Reactor Coolant Pressure Boundary."

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

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

For further discussion, see the following sections:

a) Design of Structures, Components, Equipment, and Systems 3.0

b) Integrity of Reactor Coolant Pressure Boundary 5.2

c) RCPB Leakage Detection System 5.2.5

d) Reactor Vessel and Appurtenances 5.4

e)	Reactor Coolant Piping	5.4.3
f)	Pressurizer Pressure, Level Control	7.7.1

3.1.27 CRITERION 31 – FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

CRITERION

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

DISCUSSION

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

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

1. Ultrasonic Testing - Requirements for additional ultrasonic testing are discussed in Section 5.3.1.3.1.

2. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch and tensile 1/2 T (thickness) impact/tension fracture mechanics specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with the American Society for Testing and Materials E 185-82, "Standard Practice for Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels E-706 (IF)", and the requirements of 10 CFR 50, Appendix H.

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

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

Allowable pressure/temperature relationships for plant heatup and cooldown rates are calculated using methods presented in the ASME Code, Section III Appendix G, "Protection Against Non-Ductile Failure." The approach specifies that allowed stress intensity factors for vessel level A and B service limits and hydrostatic tests 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 temperature (RT_{NDT}) due to irradiation.

3.1.28 CRITERION 32 – INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

CRITERION

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

DISCUSSION

The design of the RCPB provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel including 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 Leak Detection System in assessing the RCPB components' integrity. The RCPB, as defined by 10 CFR 50.2(v) and 10 CFR 50.55a footnote 2, will be periodically inspected under the provisions of the ASME Code, Section XI for Operations Quality Group A requirements.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, forgings, weldments and associated heat-affected zones are performed in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and E 185-82, "Standard Practice for Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels E-706(IF)." These provisions are discussed in detail in Chapter 5.3.1. Samples of reactor vessel plate materials are retained and catalogued in the event future engineering development shows the need for further testing.

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

See the appropriate sections in Chapter 5 for further details on inspection and surveillance requirements.

3.1.29 CRITERION 33 – REACTOR COOLANT MAKEUP

CRITERION

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 CVCS provides a means of reactor coolant makeup and adjustment of the boric acid concentration. When in the automatic mode, makeup is added automatically if the level in the volume control tank falls below the preset level. Manual makeup may be necessary due to valve controllability limitations at very low or very high boron concentrations. Manual makeup may be performed at any boron concentration as deemed necessary. The high-pressure centrifugal charging pumps provided are capable of supplying the required makeup and reactor coolant pump 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. In the event of a loss of coolant larger than the capacity of the reactor coolant makeup path as described in Section 6.3.3.2, a safety injection actuation signal will occur as the result of low pressurizer pressure or high containment pressure. The charging pumps are realigned from normal makeup to the safety injection lines upon receipt of the safety injection actuation signal. A high degree of functional reliability is assured by provision of standby components assuring a safe response to probable modes of failure. Details of system design are included in Section 9.3 with details of the electric power system included in Chapter 8 and Instrumentation and Control in Chapter 7.

3.1.30 CRITERION 34 – RESIDUAL HEAT REMOVAL

CRITERION

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 (RHRS) is designed to transfer the fission product decay heat and other residual heat from the reactor core within acceptable limits.

Suitable redundancy for operation assuming a single failure is accomplished with two residual heat removal pumps (located in separate compartments with means available for draining and monitoring leakage), the two heat exchangers and the associated piping, cabling, and electric power sources. The RHRS is able to operate on either the onsite or offsite electrical power system.

Details of the system designs are given in Section 5.4.7 and 6.3.

3.1.31 CRITERION 35 – EMERGENCY CORE COOLING

CRITERION

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

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

DISCUSSION

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

Details of the capability of the systems are included in Section 6.3. An evaluation of the adequacy of the system functions is included in Chapter 15. Performance evaluations are conducted in accordance with 10 CFR 50.46, and 10 CFR 50, Appendix K. Discussion about Instrumentation and Control is in Section 7.3.2.

3.1.32 CRITERION 36 – INSPECTION OF EMERGENCY CORE COOLING SYSTEM

CRITERION

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

Design provisions facilities access to the critical parts of the injection nozzles, pipes, and valves for visual inspection for erosion, corrosion, and vibration wear evidence and for nondestructive inspection where such techniques are desirable and appropriate. The design is in accordance with ASME Code, Section XI requirements.

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

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

3.1.33 CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

CRITERION

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

DISCUSSION

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

For further discussion see the following sections:

a) Performance Evaluation (ECCS) 6.3

b) Engineered Safety Feature Systems 7.3

c) Onsite Power Systems 8.3

d) Technical Specifications 16.0

3.1.34 CRITERION 38 - CONTAINMENT HEAT REMOVAL

CRITERION

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

DISCUSSION

Containment heat removal is provided by two systems, the Containment Cooling System and the Containment Spray System.

The Containment Spray System consists of two completely independent subsystems, each of which is designed for 100 percent of the heat removal capability. Specifically, the heat removal capacity of the borated water flow from either containment spray pump is adequate to keep the containment pressure and temperature below design conditions for any size break up to and including a double-ended break of the largest reactor coolant pipe or secondary system pipe.

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

The Containment Cooling System (fan coolers and fan coil units) is designed to remove the heat released from all equipment and piping in the reactor Containment during normal operation or during the loss of offsite power, and post-LOCA. These fan coolers are each supplied by the diesel generator buses and will operate from either onsite or offsite power sources. The safety related containment fan coolers will operate during normal operation, a LOCA and post LOCA, or a secondary system pipe break. The fan coil units will operate during normal operation only.

For further discussion refer to the following sections:

a) Loss of Ventilation 3.11.4

b) Containment Heat Removal Systems 6.2.2

c) Instrumentation and Control 7.3

3.1.35 CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

CRITERION

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

All essential equipment of the Containment Spray System is located outside the Containment, except for spray headers, nozzles, containment sump, and associated piping. These components include the refueling water tank, containment spray pumps and containment spray headers.

Some portions of the containment spray suction piping between the sump and the refueling water tank are either embedded in concrete or buried in the ground and are not accessible for inspection. All other piping, pumps, and valves external to the containment structure are readily accessible for periodic inspection to check system leak tight integrity. Inservice inspection of the Containment Spray System will be performed as indicated in Section 6.2.2.4.

Portions of the Containment Heat Removal System entirely within the Containment can be inspected at appropriate intervals during refueling shutdowns. Cooling water systems external to the Containment which service the containment fan coolers are accessible for inspection at any time during plant operation or shutdown for refueling and maintenance.

For further discussion, see the following sections:

a)	Residual Heat Removal System	5.4.7
b)	Containment Systems	6.2
c)	Engineered Safety Feature Systems	7.3

3.1.36 CRITERION 40 – TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

CRITERION

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

DISCUSSION

System piping, valves, pumps, heat exchangers, and other components of the containment heat removal systems are designed to permit appropriate periodic testing to ensure their structural and leak tight integrity. The components are arranged so that each component can be tested periodically for operability and required functional performance.

The performance testing of containment spray pumps is conducted at some time other than refueling. The pumps are lined up to take suction from and return flow to the refueling water tank. Pump flow is measured and pump head can be calculated by utilizing the installed instrumentation.

Actuator operated valves can be cycled from the Control Room and operation verified by observing Control Room indication of operation. Check valves will be tested to ensure that the valves open properly. Check valves directly associated with the pumps will be tested by running the pumps.

These valves include the refueling water tank check valves and the valves on the inlets and outlets of the containment spray pumps. Spray nozzles will be tested utilizing air or another approved gas method.

Operability of the containment fan coolers is verified by their use during normal operation.

For further discussion, see the following sections:

a)	Inservice Testing of Pumps and Valves	3.9.6
b)	Containment Heat Removal Systems	6.2
c)	Containment Systems	6.6
d)	Engineered Safety Features Actuation System	7.3
e)	Onsite Power System	8.3
f)	Technical Specifications	(Appendix A to Operating License)
g)	All Other Instrumentation Systems Required for Safety	7.6

3.1.37 CRITERION 41 – CONTAINMENT ATMOSPHERE CLEANUP

CRITERION

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 (CSS) is provided to reduce the concentration and quantity of fission products released to the environment following a LOCA. The Combustible Gas Control System is provided to control the hydrogen concentration in the Containment following a beyond design-basis accident.

The CSS includes a chemical additive subsystem to enhance post-accident fission product removal. A basic sodium hydroxide-borate solution is utilized. Each containment spray train is supplied power from a separate bus. Each bus is connected to both the offsite and the standby power supply systems. This assures that for onsite or for offsite electrical power system failure, their safety function can be accomplished, assuming a single failure.

Hydrogen levels within the containment atmosphere following a beyond design-basis accident will be controlled by the Combustible Gas Control System. The Combustible Gas Control system consists of a hydrogen purge system and a hydrogen monitoring system.

Although not a containment atmosphere cleanup system, the Reactor Auxiliary Building Emergency Exhaust System is designed, consistent with the functioning of other engineered safety features systems, to reduce the concentration and quantity of fission products released from the Containment to the environment following a LOCA by establishing and maintaining negative pressure in specific areas of the Reactor Auxiliary Building to ensure that post-accident activity leakage from the Containment is routed through the high efficiency particulate filtration and charcoal adsorption system.

All of the above systems will be supplied power from the diesel generator buses under loss of preferred power conditions, thus assuring system operation.

For further discussion, see the following sections:

a) Containment Heat Removal Systems 6.2.2

b) Combustible Gas Control in Containment 6.2.5

c) Containment Spray Systems 6.5.2

d) Fission Product Control Systems 6.5.3

e) Accident Analyses 15.0

f) Engineered Safety Features Actuation Systems 7.3

3.1.38 CRITERION 42 – INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

CRITERION

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

DISCUSSION

The power supply, instrumentation and controls of the Containment Spray System, and all major components of the Hydrogen Purge System are located outside Containment and are readily accessible for periodic inspection. The hydrogen purge piping and valves located inside Containment may be inspected during plant shutdown.

For further discussion, see the following sections:

a) Containment Heat Removal Systems 6.2.2

b) Combustible Gas Control in Containment 6.2.5

c) Containment Spray Systems 6.5.2

d) Fission Product Control Systems 6.5.3

3.1.39 CRITERION 43 – TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

CRITERION

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under condition 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, and Hydrogen Control and Sampling Systems are designed and constructed to permit periodic pressure and/or functional testing.

Active components of the Combustible Gas Control and Hydrogen Sampling System can be tested periodically for operability and required functional performance.

The full operational sequence that would bring the systems into action, including the transfer to alternate power sources, can be tested since the Containment Atmosphere Purge Exhaust System is not safety related and is not required to operate under accident conditions. Emergency power is not provided (refer to Section 9.4.7.3). Offsite power failure will be simulated to verify proper transfer to onsite power for normal and single failure criterion.

For further discussion, see the following sections:

a)	Inservice Testing of Pumps and Valves	3.9.6
b)	Containment Heat Removal Systems	6.2.2
c)	Combustible Gas Control in Containment	6.2.5
d)	Fission Product Control Systems	6.5.3
e)	Engineered Safety Features Actuation Systems	7.3
f)	Initial Test Program	14.0
g)	Technical Specifications	16.0

3.1.40 CRITERION 44 – COOLING WATER

CRITERION

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 cooling water systems important to safety are: 1) Component Cooling Water System (CCW), which is a closed loop system which removes heat from the residual heat removal (RHR) heat exchanger and other essential components, 2) the Service Water System (SWS), that is an open system which removes heat from the Component Cooling Water System, containment cooling units and other essential components, and 3) the Essential Services Chilled Water System (ESCW).

The Essential Services Chilled Water System transfers heat from the local safety related room area coolers and rejects it to the Service Water System. The Service Water System transfers its heat to the Main or Auxiliary Reservoir. Either reservoir is capable of providing adequate ultimate heat sink capability assuming a design basis LOCA.

The Essential Services Chilled Water System consists of two large refrigeration units and two safety related pumps to cool and circulate chilled water through the safety related room area coolers located throughout the plant. The Component Cooling Water System consists of three pumps and two heat exchangers and associated piping, valves, and instrumentation. The Service Water System consists of six pumps (four safety related and two non-safety related) and associated piping, valves, and instrumentation. The piping for the CCWS and SWS are independently arranged into essential and non-essential heat loads. The non-essential heat load of each system will be isolated from the essential heat loads during emergency mode of operation. For each system there are two headers, serving redundant safety related components. The piping, valves, pumps, and heat exchangers in each system are arranged so that the system safety functions can be performed assuming a single system failure.

The combination of one CCW pump and one CCW heat exchanger has sufficient capacity to remove heat from one RHRS heat exchanger and various essential auxiliary equipment under postulated accident conditions.

The CCWS is normally pressurized permitting leakage detection by routine surveillance or monitoring instrumentation.

Electrical power for the operation of each system may be supplied from offsite or onsite emergency power sources, with distribution arranged such that a single failure will not prevent the system from performing its safety function.

For further discussion, see the following sections:

a) Service Water System 9.2.1

b) Component Cooling Water System 9.2.2

c) Essential Services Chilled Water System 9.2.8

d) Instrumentation and Control 7.3

3.1.41 CRITERION 45 – INSPECTION OF COOLING WATER SYSTEM

CRITERION

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

DISCUSSION

The Essential Services Chilled Water System, the Component Cooling Water System and the Service Water System are designed to permit periodic inspection, to the extent practical, of important components including pumps, strainers, heat exchangers, isolation valves, and piping to assure the integrity and capability of the systems. The CCWS is normally pressurized, permitting leakage detection by routine surveillance or monitoring instrumentation. The ultimate heat sink (UHS) is two open reservoirs within the site boundary and is thus accessible to inspection.

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

For further information, see the following sections:

a) Water Systems 9.2

b) Initial Test Program 14.0

3.1.42 CRITERION 46 – TESTING OF COOLING WATER SYSTEM

CRITERION

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

DISCUSSION

The ESCWS, CCWS and SWS are in operation during normal plant operation. The structural and leak tight integrity of the system components and the operability and performance of their active components are demonstrated in this way. The operation of pumps and heat exchangers will be rotated on a scheduled basis to monitor operational capability of redundant components. Data will be taken periodically during normal plant operation to confirm heat transfer capabilities.

The systems are designed to permit testing of system operability encompassing simulation of emergency reactor shutdown or LOCA conditions, including the transfer between normal and emergency power sources.

For further discussion, see the following sections:

a)	Inservice Testing of Pumps and Valves	3.9.6
b)	Water Systems	9.2
c)	Initial Test Program	14.0
d)	Technical Specifications	16.0
e)	Instrumentation and Control	7.3

3.1.43 CRITERION 50 – CONTAINMENT DESIGN BASIS

<u>CRITERION</u>

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 10 CFR 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.

<u>DISCUSSION</u>

The reactor containment, including access openings and penetrations, is designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a LOCA up to and including a double ended rupture of the largest reactor coolant pipe or secondary system piping rupture.

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

The containment and engineered safety feature systems have been evaluated for various combinations of energy release. The analysis accounts for system thermal and chemical energy, and for nuclear decay heat. The ECCS is designed such that no single failure could result in significant metal water reaction. The cooling capacity of the Containment Heat Removal System is adequate to prevent overpressurization of the structure, and to return the Containment to near-atmospheric pressure.

The maximum temperature and pressure reached in the containment during the worst case accident are below the design temperature and pressure of this structure.

For further discussions, see the following sections:

a) Concrete Containment 3.8.1

b) Containment Heat Removal Systems 6.2.2

3.1.44 CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

CRITERION

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

DISCUSSION

Compliance with GDC-51 is discussed in CP&L letter to the NRC (Serial: LAP-83-448), dated September 30, 1983.

3.1.45 CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

CRITERION

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 is designed so that initial integrated leakage rate testing can be performed at design pressure after completion and installation of penetrations and equipment.

Provisions have been made in the containment design to permit periodic leakage rate tests, at reduced pressure, to verify the continued leak tight integrity of the Containment

Periodic integrated leakage rate testing will be carried out in accordance with the requirements of Appendix J to 10 CFR 50 as discussed in Section 6.2.6.

For further discussion, see the following sections:

a) Containment Leakage Testing 6.2.6

b) Technical Specifications 16.0

3.1.46 CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

CRITERION

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

DISCUSSION

The absence of insulation on the containment liner will permit periodic inspection of the exposed interior surfaces of the Containment.

There are special provisions for conducting leakage rate tests on liner seam welds and containment penetrations. Penetrations may be visually inspected and pressure tested at periodic intervals. Other inspections and tests are conducted as required by Appendix J of 10 CFR 50 as discussed in Section 6.2.6.

For further discussions, see the following sections:

a) Testing and In-Service Surveillance Requirements 3.8.1.7

b) Containment Integrated Leakage Rate test (Type A Test) 6.2.6.1

3.1.47 CRITERION 54 – PIPING SYSTEMS PENETRATING CONTAINMENT

CRITERION

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

DISCUSSION

Piping penetrating the Containment is designed to withstand at least a pressure equal to the containment design pressure. This isolation system design requires a double barrier on all of the above piping so that no single active failure can result in loss of isolation or excessive leakage.

Valves isolating penetrations serving engineered safety features systems may be closed by remote manual operation from the Control Room to isolate any engineered safety feature line when required.

Proper valve closing time is achieved by appropriate selection of valve, operator type, and operator size.

To ensure continued integrity of the Containment Isolation System, periodic closure and leakage may be performed in accordance with 10 CFR 50 Appendix J.

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

For further discussion, see the following sections:

a) Containment Isolation System 6.2.4

b) Containment Leakage Testing 6.2.6

3.1.48 CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

<u>CRITERION</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:

a) One locked closed isolation valve inside and one locked closed isolation valve outside Containment; or

b) One automatic valve inside and one locked closed isolation valve outside Containment; or

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

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

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

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

SHNPP conforms to the criterion with exceptions as stated in Section 6.2.4.

For further discussions, see the following sections:

a) Integrity of Reactor Coolant Pressure Boundary 5.2

b) Containment Isolation Systems 6.2.4

c) Instrument and Controls 7.0

d) Accident Analysis 15.0

3.1.49 CRITERION 56 – PRIMARY CONTAINMENT ISOLATION

CRITERION

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

a) One locked closed isolation valve inside and one locked closed isolation valve outside Containment, or

b) One automatic isolation valve inside and one locked closed isolation valve outside Containment, or

c) One locked closed isolation valve inside and one automatic isolation valve outside Containment. A simple check valve may not be used as the automatic isolation valve outside Containment, or

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

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

DISCUSSION

Lines which connect directly to the containment atmosphere and are not used to mitigate the effects of a LOCA are provided with two valves in series, one inside and one outside of the Containment. These valves are either locked closed or capable of automatic closure.

Lines which penetrate the Containment, connect directly to the containment atmosphere, and are used for mitigating the effects of a LOCA are designed to withstand at least a pressure equal to the containment design pressure. This isolation system design requires a double barrier on all the above piping so that no single active failure can result in loss of isolation or excessive leakage. The only exception to the above is the RHR pump suction piping which connects to the containment sumps. Although this piping has no valve inside the Containment, the intent of GDC 56 is met as discussed in Section 6.2.4.

Automatic isolation valves, upon loss of power, are selected to fail-close, fail-as-is, or fail-open, whichever position provides greater safety. Isolation valves are located as close to the Containment as practical.

For further discussions, see the following sections:

a) Containment Isolation Systems 6.2.4

b) Instrumentation and Controls 7.0

c) Accident Analysis 15.0

3.1.50 CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

CRITERION

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

DISCUSSION

Each line that penetrates the Containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere, has at least one containment isolation valve located outside the Containment as close to the Containment as practical.

For further discussions, see the following sections:

a) Containment Isolation Systems 6.2.4

b) Process Auxiliaries 9.3

c) Accident Analyses 15.0

3.1.51 CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

CRITERION

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

Waste Handling Systems are incorporated in the facility design for processing and/or retention of normal operation radioactive wastes. Controls and monitors capable of closing discharge isolation valves are provided to assure that release to the environment is in accordance with NRC regulations as set forth in 10 CFR 20 and 10 CFR 50.

All releases to the environment will be monitored and controlled and the system has been designed to prevent accidental discharges. The source of gaseous effluents from the plant during normal operation is mainly from the hydrogen vented from the volume control tank. This gas is processed by a Gaseous Waste Processing System. The GWPS has sufficient holdup capacity for retention of gaseous effluents containing radioactive material. However, radiological considerations may make the accumulation of this gas inventory undesirable, and planned releases to the environment may occur. All releases to the environment will be done in a controlled manner, and will be sampled and analyzed prior to release. These releases will be made in accordance with the ODCM, 10 CFR 20 and 10 CFR 50 limits.

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

For further discussions, see the following sections:

1. Liquid Waste Management Systems 11.2
2. Gaseous Waste Management Systems 11.3
3. Solid Waste Management System 11.4
4. Process and Effluent Radiological Monitoring and Sampling Systems 11.5

3.1.52 CRITERION 61 – FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

CRITERION

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to ensure 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 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

Routine visual inspections and preventive maintenance of radioactive system components and instrumentation will be conducted periodically to detect and rectify system abnormalities before failure occurs. Redundancy is provided as required to provide protection against single failure and allow isolation for testing and/or maintenance. All instrumentation will be re-calibrated periodically, consistent with current regulations and good engineering practices. Automatically actuated equipment shall be functionally tested periodically as required. Where dual equipment is provided, its use will be alternated to maintain operability.

The spent fuel storage racks are located to provide sufficient shielding water over stored fuel assemblies to limit radiation at the surface of the water during the storage period. The exposure time during refueling will be limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20.

The spent fuel pools are located in restricted areas of the Fuel Handling Building. These areas provide confinement capability in the event of an accidental release of radioactive materials. All components important to safety in these functions are designed and located to permit periodic inspection as required and are tested in accordance with accepted codes and standards.

Those areas where off-normal or accident leakage of radioactive material can reasonably be expected are designed to confine such releases to the minimum practical area and are constructed of such materials and in such a manner as to enhance the ensuing clean-up operation. Plant areas wherein significant airborne contamination can reasonably be expected during normal operation are equipped with high efficiency ventilation exhaust filtration system to reduce the discharge of radioactive nuclides to within applicable guidelines. These areas will normally be maintained at a slightly negative pressure to reduce the possibility of unfiltered air escaping to the environment.

Following a fuel handling accident, most of the radioiodines released from damaged fuel rods are retained in the spent fuel pool water. All access and normal ventilation ducts to and from the Fuel Handling Building are sealed off automatically and the enclosure is maintained at sub-atmospheric pressure by an emergency exhaust system.

Radionuclides that escape from the pool water are passed through high efficiency filters and charcoal adsorbers. Radioactive auxiliary system fluid streams will be filtered to the highest degree practical to remove radioactive particulates before the fluid is distributed to its appropriate auxiliary systems and tanks.

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

The spent fuel pools are protected from postulated missiles and designed to withstand the seismic events without loss of the pool water or damage to stored fuel.

For further discussions, see the following sections:

1.	Other Seismic Category I Structures	3.8.4
2.	Spent Fuel Storage	9.1.2
3.	Spent Fuel Pool Cooling and Cleanup System	9.1.3
4.	Process Auxiliaries	9.3

3.1.53 CRITERION 62 – PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

CRITERION

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

DISCUSSION

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality of fuel assemblies in a dry fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is accomplished by fixing the minimum separation between assemblies. Fuel elements are limited by rack design to only top-loading and to specific fuel assembly positions.

New and spent fuel are stored under water in fuel pools. The racks in which fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Each rack is composed of individual vertical cells which are fastened together in any number to form a module.

The design and administrative controls ensure criticality safety in the fuel pools. Each rack design maintains a minimum separation between assemblies which is a required feature. A neutron absorber integral to the rack design is credited for the PWR racks in pools C and D and credited for the BWR racks in pools A, B and C. In these racks K_{eff} will remain less than or equal to 0.95 if the respective pool is flooded with unborated water. For PWR racks in pools A and B, the neutron absorber fabricated with the rack is no longer credited. The racks will maintain K_{eff} less than 1.0 when flooded with unborated water and will maintain K_{eff} less than or equal to 0.95 when credit is taken for soluble boron. Additional administrative controls apply to the PWR racks in pools A and B. These include a limit on burnup-versus enrichment for unrestricted (4-of-4) storage region. PWR fuel that does not meet the criteria for the unrestricted storage region must be stored in a 2-of-4 checkerboard pattern (restricted storage region).

The fuel racks are designed to withstand shipping, handling, normal operating loads (impact and dead loads) of fuel assemblies as well as SSE and OBE seismic loads meeting ANS Safety Class 3 and AISC requirements. The fuel racks are also designed to meet Seismic Category I requirements of Regulatory Guide 1.13.

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

For further discussion, see the following sections:

a) All Other Instrumentation Systems Required for Safety 7.6

b) Fuel Storage and Handling 9.1

3.1.54 CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

CRITERION

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 in the fuel pool cooling and purification system which will detect a loss of residual heat removal capability. Appropriate safety actions are initiated by operator responses. The Waste Management System does not require the capacity for residual or decay heat removal.

Process ventilation and area radiation monitors are provided to detect and alarm excessive radiation levels in the Fuel Handling Building and Waste Management Systems areas. High radiation level in the fuel pool area will automatically shutoff the normal ventilation system and will automatically start the emergency exhaust system.

For further discussion, see the following sections:

a) Process and Effluent Radiological Monitoring and Sampling System 11.5

b) Fuel Storage and Handling 9.1

c) Air Conditioning, Heating, Cooling, and Ventilation Systems 9.4

3.1.55 CRITERION 64 – MONITORING RADIOACTIVITY RELEASES

CRITERION

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 monitored for radioactivity due to particulate and noble gas nuclides. Continuous data is supplied to the plant operator in the Control Room. Alarms indicate excess radiation.

A detection system is installed so that the radioactive status of selected plant areas are monitored under normal operating conditions, including anticipated operational occurrences and from postulated accidents. These include, but are not limited to, rooms housing the decay heat removal and core spray pumps, the Control Room, radwaste processing and fuel handling areas and the containment air exhaust plenum. In addition, offsite monitors (TLDs) are provided.

The functional objective of the monitoring system is to prevent the release of radioactivity to the environment. Liquid, gas, and solid wastes will be collected, treated and made suitable, as appropriate, for reuse, concentration, or disposal in accordance with the requirements of 10 CFR 50, and the limits of 10 CFR 20. Decisions to continue the process, to recycle, to store, or to dispose will be made on the basis of the degree of radioactivity monitored. Monitors at strategic positions will alarm abnormal radioactivity levels. Solid wastes will be packaged in shipping containers and transported offsite for disposal (or stored onsite). Radioactivity of the contents of containers shall be monitored to satisfy the requirements of 10 CFR 20, 10 CFR 71, and 49 CFR .

For further discussion, see the following sections:

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

3.2.1 SEISMIC CLASSIFICATION

3.2.1.1 Balance of Plant Scope

Plant structures, systems and components important to safety are designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional if they are necessary to assure:

a) The integrity of the reactor coolant pressure boundary (RCPB),

b) The capability to safely shutdown the reactor and maintain it in a safe condition, or

c) 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 50.67.

Plant structures, systems and components, including their foundations and supports, that are designed to remain functional in the event of a safe shutdown earthquake are designated Seismic Category I and are listed in Table 3.2.1-1. These seismic classifications are consistent with the requirements of Regulatory Guide 1.29 (see Section 1.8).

For systems which are partially Seismic Category I, the Seismic Category I portion includes all components within the seismic boundary and extends to the first seismic restraint beyond the boundary.

The seismic design of Seismic Category I structures, systems and components is described in the following Sections:

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

All Seismic Category I structures, systems and components are analyzed under the loading conditions discussed in Section 3.7 which include safe shutdown earthquake (SSE) and operating basis earthquake (OBE) loads.

Non-seismic structures, systems and components are those whose failure would not result in the release of significant amounts of radioactivity and would not prevent reactor shutdown or degrade the operation of Engineered Safety Features System. Their failure may, however, interrupt power generation.

The occurrence of adverse interaction between safety and non-safety related components during SSE events has been eliminated by adherence to the following:

a) Whenever practical, the safety related components are separated from the non-safety related components to ensure that failure of the non-safety related component due to the SSE will not result in loss of function to the safety related components.

b) In those areas where adequate separation is not possible, the non-safety related components are provided with seismic supports, or barriers are provided between the safety related and non-safety related components.

3.2.1.2 Nuclear Steam Supply System Scope

Fluid system components are classified as Safety Class 1, 2, or 3 in accordance with Section 2.2 of ANSI N18.2a-1975 , "Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." Based on evaluation of seismic requirements, Westinghouse applies a rule that each component classified as Safety Class 1, 2, or 3 shall be qualified to remain functional in the event of the safe shutdown earthquake except where exempted by meeting all of the below conditions. Systems or portions of systems required to perform the same safety function as required of a safety class component shall be likewise qualified or granted exemption. Conditions to be met for exemption are:

a) Failure of the safety class system, portion of the system, or component would not directly cause a ANS Condition III or IV event (as defined in "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plant," ANSI N18.2-1973),

b) The safety class system, portion of the system, or component has no safety function to mitigate, nor could its failure prevent mitigation of, the consequences of an ANS Condition III or IV event,

c) Failure of the safety class system, portion of the system, or component during or following any ANS Condition II event would result in consequences no more severe than allowed for an ANS Condition III event, and

d) Routine post-seismic procedures would disclose loss of the safety function. These procedures include any which a prudent plant operator would undertake to ensure safe operation before startup following a seismic event.

A further explanation of the above exemption follows:

1) All Safety Class 1 systems, portions of systems, and components must be seismically qualified because a failure of any one can directly cause an ANS Condition III or IV event.

2) Safety Class 2 systems, portions of systems, and components that are a part of the reactor coolant pressure boundary must be seismically qualified because of the rule that, "Portions of a system required to perform the same safety function as required of a safety class component which is a part of that system shall be likewise qualified or granted exemption."

3) All other Safety Class 2 systems, portions of systems, and components must also be seismically qualified because they are required to mitigate, or their failure could prevent mitigation of, the consequences of an ANS Condition III or IV event.

4) Systems, portions of systems, and components placed in Safety Class 3 by reason of item (1) of Criterion 2.2.3, the Safety Class 3 definition of ANSI N18.2a, should always meet conditions a, b, and d above for granting the seismic design exemption, except for the gas decay tanks and interconnections to suitable interface (see Criterion 2.1.3.2 of ANSI N18.2). Thus, they need not be seismically qualified if they meet the third condition for granting the seismic design exception. The basis for judgment of this third condition is the rule of Criterion 2.1.3.3 of ANSI N18.2, "The release of radioactive material due to ANS Condition III incidents may exceed guidelines of 10 CFR Part 20, Standards for Protection Against Radiation, but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius."

5) Systems, portions of systems, and components placed in Safety Class 3 by reason of items (3) or (4) of Criterion 2.2.3, the Safety Class 3 definition of ANSI N18.2, should always meet conditions a, b and d above for granting the seismic exemptions. Thus, they need not be seismically qualified if they meet the third condition for granting the seismic design exemption. The same reasoning as given in item (4) above applies.

6) Systems, portions of systems, and components placed in Safety Class 3 by reason of item (2) of Criterion 2.2.3 of ANSI N18.2 must be seismically qualified for the same reason as given in item (3) above.

Table 3.2.1-1 shows the seismic classification for the listed components. Seismically qualified components are qualified to remain functional in the event of the safe shutdown earthquake.

Q-LIST

ATTACHMENT 1

(QUALITY CLASS "A")

Safety Designator - An "S" with or without a suffix (The suffix designates train or channel). For example, SA, SB, S, SN, SA/B, SAB, SR1, SR2, SR3, SR4, S1, S2, S3, or S4.

"Nuclear Safety-Related" Designation

Safety Class (ANSI N18.2) Designator 1, 2, or 3

Pipe Category 1, 2, or 3

Seismic Class 1

Instrument Class N1, N2

Class 1E (Electrical)

Westinghouse PIC Cabinet Drawings - "Safety" Designation

Westinghouse Safety Designation - Protection I, II, III, IV

(QUALITY CLASS "B")

Instrument Class N4

Seismic Designation for NNS Components in Equipment Specifications

Station Blackout

(QUALITY CLASS "C")

Radwaste-Q Boundary Drawings (CPL-2165-S-2120 through 2153)

(QUALITY CLASS "D")

(CAR-2165-A-002)

(QUALITY CLASS "E")

By Elimination When Applying the Above Criteria

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS

3.2.2.1 Design and Construction

During the design and construction phase, systems and components important to safety are classified in accordance with ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," 1973, and ANSI N18.2a, "Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," 1975. Safety Class 1, 2, 3 and Non-Nuclear Safety (NNS) components for the SHNPP are shown on Table 3.2.1-1.

This classification system is compatible with the requirements of Regulatory Guide 1.26 (see Section 1.8). System safety classifications and interfaces from one safety class to another are shown on the system flow diagrams in the appropriate sections of this document.

Components are classified as Safety Class 1, Safety Class 2, Safety Class 3, and Non-Nuclear Safety (NNS) in accordance with their importance to nuclear safety. This importance, as established by the assigned safety class, is applied in the design, materials, manufacture and fabrication, assembly, erection and construction. A single system may have components in more than one safety class. A safety system is any system that is necessary to shutdown the reactor, cool the core or another safety system or the Containment after an accident, or is any system that contains, controls, or reduces radioactivity released in an accident. Only those portions of a system that are designed primarily to accomplish one of the above functions, or the failure of which could prevent accomplishing one of the above functions, are included.

The definitions of safety classes listed below apply to fluid pressure boundary components and the Containment. Supports that have a nuclear safety function shall be the same safety class as the components that they support. Selection of loading combinations and design methods for supports is contained in Chapter 3.

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

b) Safety Class 2 - Safety Class 2 applies to Containment and to those components which are:

    1) Part of the reactor coolant pressure boundary and not Safety Class 1, or

    2) Part of safety systems necessary to remove heat directly from the reactor or Containment, circulate reactor coolant for any safety system purpose, control radioactivity released within the Containment, or control hydrogen in the Containment.

c) Safety Class 3 - Safety Class 3 applies to those components not in Safety Class 1 or Safety Class 2 and

    1) The failure of which would result in release to the environment of radioactive gases normally required to be held for decay,

    2) Provide or support any safety system function,

    3) Control the containment airborne radioactivity released to the environment in an accident, or

    4) Remove decay heat from spent fuel.

d) Non-Nuclear Safety - Non-Nuclear Safety (NNS) includes portions of the nuclear power plant not covered by Safety Classes 1, 2 or 3 that can influence safe normal operation or that may contain radioactive fluids. Design of NNS components is in accordance with applicable industry codes and standards.

3.2.2.2 Operations

During the initial test and operations phase, systems and components important to safety are classified in accordance with ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," 1973, and ANSI N18.2a, "Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," 1975. Safety Class 1, 2, 3 and Non-Nuclear Safety (NNS) components for the SHNPP are shown on Table 3.2.1-1.

This classification system is compatible with the requirements of Regulatory Guide 1.26 (see Section 1.8). System safety classifications and interfaces from one Safety Class to another are shown on system flow diagrams in the appropriate sections of this document.

3.2.2.3 Structures, Systems, and Component Classification List

Table 3.2.1-1 tabulates structures, systems, and components by design and construction (D&C) Safety Class, D&C Code and Code Class, D&C quality assurance requirements, Operations Quality Class, and operations quality assurance requirements.

Refer to Section 5.2.1 for a discussion of 10 CFR 50.55a minimum ASME III Code Requirements as applied to components forming part of the reactor coolant pressure boundary.

3.3 WIND AND TORNADO LOADINGS

As outlined in General Design Criterion (GDC) 2, "Design Basis for Protection Against Natural Phenomena", of Appendix A to 10 CFR 50, structures, systems, and components important to safety are designed to withstand the effects of natural phenomena, such as tornadoes or hurricanes without loss of capability to perform their safety functions.

The design basis reflects:

a) Appropriate consideration of the most severe wind historically reported for the site and its environs, with sufficient margin of safety.

b) Appropriate combinations of the effects of normal and accident conditions with the effect of wind.

c) Importance of safety functions to be performed.

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

a) the structure or component is designed to withstand design wind, tornado wind and tornado generated missiles, or

b) the system or components are housed within a structure which is designed to withstand the design wind, tornado wind and tornado generated missiles.

Table 3.3.0-1 lists all safety related structures and the method of wind/tornado protection as applicable. The Ta or Tb designation in the table refers to item a or b above.

3.3.1 WIND LOADING

This section describes provisions for calculating wind forces on structures and parts and portions thereof. The outlined procedures are based on "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures", ANSI A58.1-1972, hereinafter referred to as the ANSI Code, and the recommendation of ASCE Paper No. 3269, "Wind Forces on Structures", for cases not covered by ANSI A58.1, and ASCE Paper No. 4933, "Wind Loads on Dome-Cylinder and Dome-Cone Shapes," for detailed Containment Building dome pressure coefficients.

3.3.1.1 Design Wind Velocity

The plant Seismic Category I structures are designed to withstand the effects of the design wind, a maximum wind of 179 mph at 30 feet above plant grade. The design wind is based on a 1000-year return period "fastest mile of wind." Standard Review Plan 3.3.1 requires that the design wind velocity be based on a 100-year return period "fastest mile of wind," which for SHNPP is 117 mph at 30 ft. above plant grade. Since the design wind is based on the 1000-year return period, SHNPP design is conservative. The basis for the selection of the wind velocity for design is presented in Section 2.3.

3.3.1.2 Determination of Applied Forces

The wind loads which were applied to structures as static forces were derived from the recommendations of References 3.3.1-1 and 3.3.1-2.

3.3.1.2.1 Vertical Velocity Distribution and Gust Factors

The design wind, as discussed in Section 3.3.1.1, is defined by its basic design velocity, i.e., as a perfectly smooth, laminar motion of air at a constant speed.

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

a) Variation of wind velocity with height was compensated for by the introduction of velocity distribution coefficients, as indicated by the following expressions:

$$V_Z = V_{30}\left(\frac{Z}{30}\right)^{y} \quad (1)$$

where:

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

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

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

y = 0.143 = Velocity distribution factor

The value of the velocity distribution factor, y, is the same as that used by the ANSI Code for the exposure 3 (flat, open country, open flat coastal belts, and grasslands).

b) Sudden, brief fluctuations in the wind speed (gusts) and the dynamic nature of load were accounted for through application of the gust factors. The gust factors, G_f (for buildings and structures) and G_p (for parts and portions), were assigned the same values as those suggested by the ANSI Code. They provide conservatively for the dynamic response of ordinary buildings. In cases where the ratio of building height to the least horizontal dimension exceeds 5, a detailed analysis of building dynamic response was performed by using the method described in Section A6.3.4.1 of the ANSI Code.

3.3.1.2.2 Effective Velocity Pressure

The dynamic wind pressure (q30) in pounds per square foot was calculated from the wind speed using the formula:

$$q_{30} = 0.00256\, V_{30}^2 \quad (2)$$

where V_{30} is the wind speed in miles per hour.

The effective velocity pressures of winds for buildings and structures, q_F, and for parts and portions, q_p, at various heights above the ground, were computed in accordance with the following formula:

$$q_F = K_z\, G_f\, q_{30} = 0.00256\, G_f \left(\frac{Z}{30}\right)^{y} V_{30}^2 \quad (3)$$

$$q_p = K_z G_p q_{30} = 0.00256 G_p \left(\frac{z}{30}\right)^y V_{30}^2 \quad (4)$$

where K_z is a velocity pressure coefficient which depends upon the type of exposure and height z above ground, and G_f and G_p are gust factors which depend upon the type of exposure and dynamic response characteristics of the structure, or parts and portions thereof.

The values of q_F and q_p for various heights above ground are shown in Table 3.3.1-1.

3.3.1.2.3 Design Wind Pressure

Wind forces on a structure, or any element thereof, result from a differential pressure caused by the obstruction of the free flow of the wind. Therefore, in addition to being proportional to wind velocity, the design wind pressures are a function of the orientation, shape, and size of the object obstructing the free flow of wind. Pressures were obtained by multiplying the effective velocity pressure by the pressure coefficients given in Table 3.3.1-2 as indicated by the following expressions:

$$P_{jF} = C_p q_F - C_{pi} q_M \quad (5)$$

$$P_{jp} = (C_p \text{ or } C_{pl}) q_p - C_{pi} q_M \quad (6)$$

where:

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

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

C_p = External pressure coefficient

C_{pl} = External local pressure coefficient

C_{pi} = Internal pressure coefficient

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

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

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

A negative value for P_{jF} or P_{jp} in equations (5) and (6) indicates that the resultant pressure acts outward.

The coefficient C_{pi} is dependent on the number and location of openings in the portion of the structure exposed to the wind. Normally, air leakage due to small openings around windows, doors, skylights, and eaves will give rise to a net internal pressure or suction depending on whether the openings are chiefly in the windward or leeward surfaces of the structure. The Seismic Category I structures are closed low leakage rate structures of reinforced concrete construction, and the coefficient C_{pi} was not applicable, since they were considered to be airtight for the purpose of calculating internal pressures, unless there were actual openings provided, such as louvers. Therefore, the coefficient C_{pi} for structures with no openings in the portion of the structure exposed to the wind was taken as zero, and C_{pi} for buildings with openings was taken from Table 11 of Reference 3.3.1-1.

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

Coefficients listed as C_{pe} for h/d=1 (height to diameter ratio) in Table 4(f) of Reference 3.3.1-2 or Figure 6 of Reference 3.3.1-3 were used for the coefficient C_p for the cylindrical portion of the Concrete Containment Structure.

The pressure distribution coefficient for the dome of the Concrete Containment Structure is based on Figure 6 of Reference 3.3.1-3.

Figure 3.3.1-1 shows the wind pressure distribution around the containment cylindrical wall and Figure 3.3.1-2 shows the wind pressure distribution on the containment dome.

3.3.1.2.4 Design Wind Loads

The design wind loads on a structure or a portion thereof, W_F and W_p, were obtained by calculating the vector sum of the resultant forces acting on the structure or parts thereof:

$$W_F = \sum_{j=l}^{n} P_{jF} A_{jF} \quad (7)$$

$$W_p = \sum_{j=l}^{n} P_{jp} A_{jp} \quad (8)$$

where:

W_F	= Design wind load (lb) for buildings and structures
W_p	=Design wind load (lb) for parts and portions of buildings or structures
A_{jp} & A_{jF}	= Exposed areas (ft^2)
P_{jp} & P_{jF}	= As defined in Section 3.3.1.2.3
j & n	=Summation indices specifying that summation takes place over all exposed areas.

All structures were designed to withstand the sliding and overturning effects of the wind considering the vector sum of the forces acting on the windward and leeward sides. In certain cases when net pressure coefficients Cf listed in Reference 3.3.1-1 or drag coefficients CD listed in Reference 3.3.1-2 were available, the total design wind load was calculated directly by the following formula:

$$W = q_F (C_f \text{ or } C_D) A \quad (9)$$

where A is the projected area of the structure on a vertical plan normal to the wind directions.

3.3.2 TORNADO LOADING

This section describes provisions for calculating tornado generated forces on structures and parts or portions thereof.

3.3.2.1 Applicable Design Parameters

Parameters applicable to the design basis tornado are in accordance with Regulatory Guide 1.76 and are as follows:

a) External wind forces resulting from a tornado funnel with a horizontal rotational velocity of 290 mph and a horizontal translational velocity of 70 mph.

b) A radius of maximum rotational velocity of 150 feet.

c) A decrease in atmospheric pressure of 3 psi at a rate of 2 psi/sec.

d) Tornado-generated missile impact loads resulting from the postulated tornado driven missiles indicated in Section 3.5.

Regulatory Guide 1.76 Revision 1 was issued for use in March 2007. This regulatory guide provides licensees and applicants with new guidance that the staff of the NRC considers acceptable for use in selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear plant should be designed to withstand. This guidance divides the United States into three regions: the Harris Nuclear Plant is located in Region 1. The NRC staff accepts the methods described in Regulatory Guide 1.76 Revision 1 to evaluate submittals from operating reactor licensees after March 2007 who voluntarily propose to initiate system modifications that have a clear nexus with the guidance provided. No backfitting is intended or approved in conjunction with its issuance. The Harris Nuclear Plant adopts the guidance provided in Regulatory Guide 1.76 Revision 1 as an optional design basis for new system modifications occurring after March 2007.

3.3.2.2 Determination of Forces on Structures

Tornado wind speed was converted into equivalent velocity pressure using the standard equation for the kinetic energy of moving air expressed in terms of pressure, as specified in References 3.3.1-1 and 3.3.1-2.

3.3.2.2.1 Velocity Pressures

The velocity pressure calculations were based on procedures outlined in Section 3.3.1.2, taking velocity distribution coefficients, K_Z, and gust factors, G_f or G_p, equal to unity as follows:

$$q_T = 0.00256\ K_Z\,(G_f \text{ or } G_p)\ V_T^2 = 0.00256\ V_T^2 \qquad (10)$$

The design tornado wind speed, V_T, for conservatism, was taken as 360 mph, which is the summation of design horizontal rotational and translational vector tornado velocities.

For application of velocity pressure on the containment structure References 3.3.1-2 and 3.3.1-3 were used in addition to ANSI A58.1-1972.

For large structures or parts of structures whose horizontal dimensions perpendicular to the wind force were comparable to the radius of the tornado vortex at which the maximum tangential wind speed occurs, a more realistic, effective velocity pressure may be determined, as described below.

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

$$V_T(r) = C\frac{r}{R_c} \text{ for } r \leq R_C \qquad (11)$$

$$V_T(r) = C\frac{r}{R_c} \text{ for } r > R_C \qquad (12)$$

where:

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

r = Radial distance from COV (ft)

R_C = Radius of the maximum wind speed (ft)

C = Constant = 290 mph

The values of effective velocity pressures are shown on Figure 3.3.2-3.

3.3.2.2.2 Atmospheric Pressure Drop

The circular pattern of air motion in a tornado produces an atmospheric pressure drop within the vortex. The pressure drop is a function of tangential wind velocity and distance from the center of vortex, and can be determined by making use of the cyclostropic wind equation, given in Reference 3.3.2-1:

$$\frac{1}{\rho}\frac{d[\Delta p(r)]}{dr} = \frac{[V_T\,(r)]^2}{r} \qquad (13)$$

where:

$\Delta p(r)$ represents atmospheric pressure drop, ρ is mass density of air.

$V_T(r)$ and r are defined above.

The above equation was solved and the values of pressure differential were plotted on Figure 3.3.2-4.

3.3.2.2.3 Tornado-Generated Missiles

Tornado-generated missile parameters are presented in Section 3.5.1.

3.3.2.2.4 Combination of Applied Loads

In determining the total tornado load, W, the effects of the uniform tornado wind load, W_w, the tornado differential pressure load, W_p, and the tornado missile load, W_m, have been considered using the following combinations:

$$W = W_w \quad (14a)$$

$$W = W_p \quad (14b)$$

$$W = W_m \quad (14c)$$

$$W = W_w + 0.5\ W_p \quad (14d)$$

$$W = W_w + W_m \quad (14e)$$

$$W = W_w + 0.5\ W_p + W_m \quad (14f)$$

For each particular structure or portion thereof, the most adverse of the above combinations has been used as appropriate. These total tornado loads have then been combined with the other loads to design structures as specified in Sections 3.8.1, 3.8.4 and 3.8.5. No venting of structures is assumed for tornado load determination.

Total design tornado loads, due to the effects of the uniform tornado wind loads, W_w, are defined by:

$$W_w = \sum_{j=l}^{n} P_j A_j \quad (15)$$

where:

W_w =Total design tornado load due to the effects of the uniform tornado wind loads

A_j = Exposed area

P_j = Design wind pressure (as defined below) acting on A_j

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

Design tornado wind pressures, P_j, have been found by using the following expression:

$$P_j = C_p q_F \quad (16)$$

where:

P_j = Design wind pressure acting on area A_j

C_p = Velocity pressure coefficient

q_T = Effective velocity pressure as defined in Section 3.3.2.2.1

The numerical value of C_p for buildings with flat walls and for the cylindrical and dome portion of the Concrete Containment Structure are those specified in Section 3.3.1.2.3.

Figure 3.3.2-1 shows the tornado wind pressure distribution around the containment cylindrical wall and Figure 3.3.2-2 shows the tornado wind pressure distribution on the containment dome.

The structures under consideration were placed at various locations in the tornado field (at various distances from the center of vortex) to determine the maximum local and overall effects on the structure resulting from the design pressure, P_j, by making use of Figures 3.3.2-3 through 3.3.2-5.

For overall structural effects such as overturning, sliding, and torsion, the design pressure applied to the exterior surface of a structure was the actual pressure calculated on a plane normal to the direction of translation of the tornado that passes through the center of the structure as shown on Figure 3.3.2-5.

The total load on a building, due to design tornado wind, for the determination of safety against sliding and overturning was the vector sum of the forces acting on its elements. This design tornado wind load, for an enclosed building in the direction of the wind was determined by replacing the external pressure coefficient, C_p, by the drag coefficient C_D in Equation (16). The numerical values for the coefficient C_D for buildings with flat walls, and for the cylindrical and domed shape Concrete Containment Structure are discussed in Section 3.3.1.2.4.

The tornado pressure differential was considered in calculating tornado pressure loading for closed buildings. The maximum pressure drop of 3 psi occurs at the center of the vortex and diminishes with distance from the vortex center, as described in Section 3.3.2.2.2. However, for conservatism, a 1.5 psi pressure drop was combined with the maximum translational and rotational tornado wind velocity. The pressure drop in the tornado funnel was assumed not to change the total directional wind pressure on the entire building in the direction of the wind due to the maximum tornado wind speed of 360 mph.

The procedures for evaluation of impact loads and structural response due to the postulated tornado-generated missiles are described in Section 3.5.

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

To ensure the ability of Seismic Category I structures to perform despite failure of structures not designed for tornado loads, the following criteria are met:

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

b) Where the above criterion is not met, the affected non-Seismic Category I structure has been designed either to withstand tornado loads or not to collapse against Seismic Category I structures under tornado loadings.

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

d) The failure of any structural member or component in non-seismic structures caused by being hit by a tornado generated missile would be local in nature, would cause no damage to Seismic Category I structures or components, and would not prevent the safe shutdown of the reactor or result in uncontrolled release of radioactivity to the environment.

e) The Turbine Building has been evaluated for tornado loading to the following extent:

1) Structural framing has been designed for tornado wind on the exposed steel surfaces.

2) The Turbine Building gantry crane has been designed for tornado wind on the exposed steel surfaces, when located in the parked position on the unloading bay support structure.

3) Tornado-generated missiles are not considered in the design.

REFERENCES: SECTION 3.3

3.3.1-1 ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures".

3.3.1-2 ASCE 3269, "Wind Forces on Structures", American Society of Civil Engineers, Transactions, Vol 126, Part II, 1961

3.3.1-3 "Wind Loads on Dome-Cylinder and Dome-Cone Shapes, "F J Maher, Journal of the Structural Division, ASCE Vol 92, No. S T 5 Proc Paper 4933, October 1966

3.3.2-1 Stevenson, J. D. "Tornado Design of Class I Structures for NPPs", Westinghouse Nuclear Energy Systems, Pitts, Pa.

3.3.2-2 A Amirikian, "Design of Protective Structures," Bureau of Yards and Docks, Publication No. NAVDOCKS P-51, Department of the Navy, Washington, D. C. (August 1950).

3.3.2-2 R F Recht and T W Ipson, "Ballistic Perforation Dynamics," Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, September 1963.

3.3.2-4 R A Williamson and R R Alvy, "Impact Effect of Fragments Striking Structural Elements," Holmes and Narver, Inc., Revised November 1973.

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

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

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

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

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

Seismic Category I structures, systems, and components that fall under "a" and "b" above are listed under Conditions "A" and "B" in Table 3.4.1-1. The safety related systems, components and equipment that are housed in structures satisfying conditions "a" and "b" above are protected against floods by the structure.

The Main Dam and Spillway, Auxiliary Dam and Spillway, and Auxiliary Separating Dike are discussed in Section 2.5.6, and the Auxiliary Reservoir Channel, Emergency Service Water Intake Channel, and Emergency Service Water Discharge Channel are discussed in Section 2.4.8.

All the structures listed under Condition "B" in Table 3.4.1-1 are shown on Figure 3.4.1-1. The plant grade has been established at a minimum elevation of 260 ft. which is 21.1 ft. above the maximum main reservoir still water level of 238.9 ft. and 4.0 ft. above the maximum auxiliary reservoir still water level of 256.0 ft. Maximum wave run-up and wind setup level along the plant site in the Main and Auxiliary Reservoirs are expected to be at elevation 240.2 ft. and 257.7 ft. respectively (see Table 2.4.5-2). All structures shown on Figure 3.4.1-1 are thus protected against floods in the Main or Auxiliary Reservoirs.

The design basis of the plant site drainage is a storm of five in. per hour rainfall intensity. The maximum net accumulated water on the plant grade due to the probable maximum precipitation (PMP) for the project drainage area of 71.0 sq. mi. is approximately 6 in. (see Table 2.4.3-1). The maximum net accumulated water on the plant island due to a more severe PMP computed for a drainage area of one sq. mi. is approximately 15 in. (see Table 2.4.2-4).

All structures on the plant site are protected to at least Elevation 261 ft. and no structure has any access openings below Elevation 261 ft.

The maximum elevation to which water will pond on the plant site during a PMP event assuming the entire drainage system became blocked would be 261.27 ft. The storm runoff will flow freely into the Main and Auxiliary Reservoirs through the open channels and flow over the plant roads (crown elevation 261.0 ft.). However, ponding to elevation 261.27 ft. will not impact on the plant ability to safely shutdown, if necessary.

All safety-related structures which have entrances at elevation 261 ft. are protected against any ponding during a Probably Maximum Precipitation (PMP) event by the following features:

a) Artificial barriers such as watertight or airtight doors, or

b) Low structural barriers, i.e., curbs. The minimum curb elevation is 262.0 ft.

The only exceptions to the above are two entrances to the Waste Processing Building which are not protected against any ponding above EL 261.06 ft. However, these entrances provide access to areas which house locker room, shower stalls and do not house any safety-related equipment.

The elevation of the railroad in the plant area is 261.0 ft. (top of the rails) which is equivalent to the crown elevation of the plant roads. Therefore, the above analysis includes the effects of ponding caused by the railroad tracks.

The elevation of the plant security fencing is the same as plant grade. Concrete footing for the fence posts should be at or slightly below grade.

The rain storm water collected in the area between the Retaining Wall and the Fuel Handling Building (see Figure 3.4.1-2) will be pumped out to the storm drainage system by using sumps and pumps. In addition to the direct rainfall and groundwater infiltration through the retaining wall, this area will collect storm water as overflow from the Waste Processing Building and the Fuel Handling Building if the drains are assumed to be plugged during the PMP occurrence. If the failure of pumps is postulated, the water will accumulate to a level below elevation 236 ft. in this area. All openings in the Fuel Handling Building and the Waste Processing Building below elevation 236 ft. have been closed and other penetrations sealed to preclude access of storm water to safety related areas inside the buildings. The storm water from the cancelled Unit No. 2 Reactor Auxiliary Building and the Containment Building drains in to the centrally located sump and is pumped into the plant drainage system (see Figure 3.8.4-45). The sump and pump are sized for the design basis rain fall intensity. However, the wall heights are adequate to accommodate the PMP considering that the pump has failed. All openings below EL 243.00 ft. have been closed and waterproofed to minimize water seepage from this area into Unit No. 1 structures.

The west Fuel Handling Building retaining wall is depicted on Figures 3.8.4-42 through 3.8.4-44. The northeast Fuel Handling Retaining wall is depicted on Figure 3.8.4-45.

The structures that house safety related equipment are identified in Table 3.4.1-1. As discussed in Section 2.4.13, the design basis groundwater level for the structure at the plant site was established to be at Elevation 251 ft. Openings and penetrations below grade (Elevation 260) in the exterior walls of the structures housing safety related equipment are discussed below.

The Containment Building, Fuel Handling Building, Waste Processing Building, Reactor Auxiliary Building, Tank Building, Turbine Building, and Fuel Handling Unloading Area, as shown on Figure 3.4.1-2, are separated by seismic gaps between adjacent structures. All of these buildings are Seismic Category I reinforced concrete structures, except for the Turbine Building which are seismic Category I structures only in the portion where Diesel Generator leads and service water pipes pass through. The base mats of all these buildings are reinforced concrete slabs supported on sound rock. The seismic gap, which is in general one inch between the mats and two inches between the walls of the adjacent buildings, is cut off from groundwater by horizontal waterstops between the mats and vertical waterstops at the locations shown on Figure 3.4.1-2. The waterstops minimize groundwater seepage into the seismic gaps. The exterior walls of the Fuel Handling Building, Waste Processing Building, Reactor Auxiliary Building, Tank Building, Turbine Building and Fuel Unloading Area which are in direct contact with soil, are exposed to groundwater.

The elevations of the walls which have penetrations below plant grade are shown on Figures 3.4.1-3 through 3.4.1-8.

Other structures housing safety related equipment are the Diesel Generator Building, the Diesel Fuel Oil Storage Tank Building, the Emergency Service Water Screening Structure, and the Emergency Service Water and Cooling Tower Make-up Water Intake Structure. All of these structures are also Seismic Category I reinforced concrete structures with reinforced concrete mats resting on sound rock, and the elevations of their exterior walls which have penetrations below grade level are shown on Figures 3.4.1-9 through 3.4.1-12.

The electrical manholes for auxiliary and emergency power system cables are also Seismic Category I reinforced concrete structures, but are founded on soil. A typical detail of cable penetrations in the walls of electrical manholes is shown on Figure 3.4.1-13. The auxiliary and emergency power system cables are buried in soil and are shown in Figures 3.4.1-14.

Electrical manholes and duct runs for auxiliary and emergency power system cables are capable of normal function while completely or partially flooded. The duct runs are sloped towards the electrical manholes, shown on Figure 3.4.1-14, and groundwater finding its way into the PVC conduit will be drained to the electrical manhole. The electrical manholes, shown on Figure 3.4.1-13, have been provided with collection sumps for any water coming through PVC conduits or cracks in the reinforced concrete walls or slabs of the manholes. When necessary, the water in the sumps will be removed by pumps.

The Containment Building, as shown on Figure 3.4.1-15, is a steel-lined reinforced concrete structure. To preclude external water pressure on the steel liner, a continuous impervious PVC waterproofing membrane has been placed between the containment foundation mat and the foundation rock. The waterproofing membrane is continuous under the mat and terminates in the waterstops at the joint with adjacent structures. Although not expected, any leakage through the waterproofing membrane will be drained through porous concrete drains placed between the membrane and the mat. The porous concrete drains lead to two sumps in the reactor auxiliary building mat as shown on Figure 3.4.1-16. Each sump has been provided with two full capacity pumps for redundancy. The porous concrete drains are interconnected so that water at any place has two paths for egress. The pumps discharge water to the HVAC Condensate Drainage System. In case of failure of the sump pumps, water will overflow the pump casing pipe at Elevation 194 ft. and will be drained by the Floor Drain System. Since the top of the casing pipe is at Elevation 194 ft. and that of containment steel liner at the reactor cavity is at Elevation 210 ft., no water pressure will be exerted on the liner.

As shown on Figures 3.4.1-3 through 3.4.1-14, none of the structures housing safety related equipment have personnel access openings below Elevation 261 ft. The penetrations for pipes and electrical conduits have been sealed with waterstops and boots, as shown on Figures 3.4.1-3 through 3.4.1-14.

Exterior walls of the buildings which are exposed to groundwater have been provided with impervious bithuthene waterproofing membrane up to Elevation 259 ft. and all of the vertical and horizontal construction joints in the walls below grade and in the mats except for the construction joints in the north-west corner walls of the Waste Processing Building have been provided with waterstops. Any inleakage through the waterproofing membrane, construction joints or cracks in the reinforced concrete walls or base mats will be handled by floor drains routed to associated sumps and pumps. Any water finding its way into the seismic gaps will be drained into the lowest building through weepholes at the lowest level of the gap, as shown on Figure 3.4.1-2, and will be drained by the Floor Drain System. Any groundwater seeping through the vertical joints in the retaining wall or coming out of the retaining wall drainage system will be collected into drainage sumps and pumped out to the storm drainage system.

All safety related equipment is housed inside the Seismic Category I Structures listed in Table 3.4.1-1 Condition B and thus is protected against flooding, except the RWST level transmitters, which are located in the RWST pit area approximately 1.5 feet above grade. The RWST level transmitters are protected from flooding conditions by providing a completely submersible transmitter installation, fully capable of providing their design basis operation during and after the maximum PMP flooding event. All safety related structures will not be jeopardized as a result of the maximum still water level or wave run-up resulting from a PMF, or storm water accumulated at the plant site due to a PMP, and therefore, it will not be necessary to bring the reactor to a cold shutdown for flood conditions.

3.4.1.2 Permanent Dewatering System

No permanent dewatering system is provided for the plant.

3.4.2 ANALYSIS PROCEDURE

Since the plant grade elevation of 260 ft. is 19.8 ft. above the maximum wave run up and wind setup water level of 240.2 ft. in the Main Reservoir and 2.3 ft. above the maximum wave run-up and wind setup water level of 257.7 ft. in the Auxiliary Reservoir, all structures on the plant site are protected against floods occurring in the Main or Auxiliary Reservoirs. Dynamic effect of wind on storm water accumulated on the plant grade will be insignificant and has not been considered.

All Seismic Category I structures and non-Seismic Category I structures located adjacent to Seismic Category I structures are designed against full hydrostatic load and buoyancy due to design basis groundwater.

As discussed in Section 2.4.13.5, the design basis groundwater level for the structures on the plant site is at Elevation 251 ft.

The lateral hydrostatic loads and the buoyant forces are included as part of the dead load in all of the load combinations specified in Section 3.8.1 for the containment structures, in Section 3.8.4 for all other Seismic Category I structures and for load combinations 1 through 4 in Section 3.8.5 for the stability of foundations of all Seismic Category I structures. For conservatism, the buoyant force in load combination 5, specified in Section 3.8.5.5 is considered for water at plant grade Elevation 260.0 ft. The walls of Fuel Handling Building and Waste Processing Building exposed to accumulated storm water in the area between the

Retaining Wall and the Fuel Handling Building are capable of withstanding the corresponding lateral hydrostatic loads.

The Retaining Wall west of the Fuel Handling Building is designed for saturated backfill up to plant grade EL 260 ft. Hydrostatic pressure due to groundwater table is relieved through the transition filter provided at the back of the wall and open joints in the wall. (See Figures 3.8.4-42, 3.8.4-43, and 3.8.4-44)

3.5 MISSILE PROTECTION

3.5.0 MISSILE SELECTION AND DESCRIPTION

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

To reduce the probability of unacceptable consequences related to missile impact, key backup and/or redundant components and systems have been physically separated and shielded so that a single missile is incapable of negating the redundant functions. In addition, most Seismic Category I components are housed in Seismic Category I structures.

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

a) No perforation of the Containment Building (i.e., no loss of leak tightness)

b) Assurance that the plant can be maintained in a safe shutdown condition

c) Offsite exposure within 10 CFR 100 guidelines for missile damage resulting in radioactivity release.

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

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

Barriers are provided for missiles which cannot be oriented to take advantage of existing structures and which could cause failure of safety-related structures or components. Generally, these barriers are designed to contain or deflect the missiles from the safety related feature; but, as a minimum requirement, penetration of the missile through the barrier reduces the missile energy to levels which cannot cause failure of the safety feature function. Wherever possible, advantage is taken of walls and structures arising from functional requirements, other than missile considerations, by judicious arrangement of equipment.

The above design practices preclude missile caused failure of the Containment Building; capability to achieve safe shutdown is maintained.

Sources considered capable of generating potential missiles are as follows:

a) Tornadoes

All Seismic Category I structures are designed to withstand the tornado generated missiles specified in the "Missile Spectrum" of the Standard Review Plan 3.5.1.4 (Rev. 0) and listed in Table 3.5.1-3. All Seismic Category I structures are designed with f'c=4000 psi concrete and a minimum thickness of 24 inches in roofs and walls.

b) Main Turbine Failure

A discussion of turbine missiles is found in Section 3.5.1.3.

c) Structures and overhead cranes which may damage safety related equipment.

d) Dynamic equipment failures encompassing pumps, diesel engines, turbine drives, HVAC fans and compressors.

e) Valve stems and bonnets of significant size having the potential to violate any of the missile protection criteria.

f) Control rod drive mechanisms or parts thereof.

g) Miscellaneous

    1) Sand plugs

    2) Instrument wells and thimbles with mounted components.

3.5.1 MISSILE PROTECTION METHODS

Protection of safety related systems and equipment, including the containment liner, from missiles is accomplished by one or more of the following methods:

a) Compartmentalization

Enclosing equipment in missile protected compartments.

b) Barriers

Erecting barriers to stop potential missiles either at the source or at the equipment location. No composite sections are relied upon to provide protection from missile protection.

Layout and structural design protect ECCS injection paths, leading to unruptured reactor coolant loops, against damage from the maximum reactor coolant pipe rupture.

The injection lines that are located in the area between the secondary shield wall and the containment wall are protected from missiles generated within the secondary shield wall compartments. Those portions of the safety injection piping which enter the steam generator compartments, penetrate the shield wall at locations which are as close to the injection point on the RCS main loops as is practicable to reduce the length of piping within the secondary shield.

The containment spray system piping, and the containment cooling system fans, coolers and associated service water system lines within the Containment are located completely outside the secondary shield wall. The containment internal structure outside the secondary shield wall provides protection to the Containment Spray System, Hydrogen Recombiners System and Containment Cooling System from the effects of pipe whip resulting from the rupture of the reactor coolant pressure boundary.

c) Separation

Sufficient separation of redundant systems is provided so that a potential missile cannot impair both systems. Physical separation is provided between safety related equipment and high pressure piping such that failure of the pipe will not jeopardize the equipment.

Two mechanical and two electrical penetration areas are provided for redundant safety related systems entering the Containment Building.

Separation to the maximum extent practical is maintained for all piping and equipment of engineered safety features within the Containment.

Each safety injection line associated with a redundant ECCS train is located within separate steam generator compartments after penetrating the secondary shield wall. Therefore, no single missile generated within a steam generator compartment could damage redundant portions of ECCS piping. No safety injection piping is located within the reactor vessel cavity.

d) Restraints

Fastening down potential missiles by means of restraints.

e) Equipment Design

Designing structures or components to withstand missiles without loss of function.

f) Strategic Orientation

Facing equipment or parts of equipment in a direction that will point the potential missile paths away from other equipment. Pressurized piping is routed adjacent to walls, floors, ceilings, columns, abutments, and foundations, to limit movement due to pipe rupture and provide protection to safety related systems.

g) Distance

Locating equipment beyond range of potential missiles.

In cases where concrete or steel is used as missile protection, the calculation of the missile shield thickness required is based on the National Defense Research Committee (NDRC) formula or the standard steel penetration formula presented in Nuclear Engineering and Design, "The Design of Barricades of Hazardous Pressure Systems," C. V. Moore, 1967.

3.5.1.1 Internally Generated Missiles (Outside the Containment)

Seismic Category I structures, systems, and components outside the Containment whose failure could result in radiological consequences in excess of 10 CFR 100 guidelines or which are required for attaining and maintaining a safe shutdown during normal or accident conditions are listed in Table 3.5.1-1. Missile protection provisions and references to applicable system descriptions and drawings that demonstrate separation and independence are also listed in Table 3.5.1-1.

Potential sources of missiles are:

a) High-pressure systems

b) Rotating machinery

c) Gravitational missiles

d) Secondary missiles (those generated from the impact of primary missiles)

3.5.1.1.1 Nuclear Steam Supply System Vendor Scope

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

Equipment within NSSS scope outside Containment (see Section 3.5.2) has been evaluated for potential missile sources. As a result of this review, the following information concerning potential missile sources and systems within NSSS scope which require protection from internally generated missiles outside Containment is provided.

Systems outside Containment which are required for safe shutdown of the reactor must be protected regardless of the missile source. Potential missile sources from components within NSSS scope which are in these systems have been evaluated as discussed below.

Valves in high pressure systems have been reviewed. As a result of this review it is determined that no single failure associated with any potential valve part can result in the generation of a missile. Therefore, there are no credible sources of missiles postulated for valves outside the Containment within NSSS scope.

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

The motor-generator, which is used to provide power to the control rod drive mechanisms, is located outside Containment. The flywheel on the component has been evaluated as a potential missile. However, based upon fabrication techniques employed for the flywheel there are no credible missiles generated from the motor-generator flywheel.

3.5.1.1.2 Balance-of-Plant Scope

The SHNPP is designed with adequate separation and barriers so that missiles from external or internal sources:

a) Will not cause or increase the severity of a loss-of-coolant accident.

b) Will not damage engineered safety features such that the minimum required safety functions are jeopardized.

c) Will not cause a break in the main steam supply and feedwater piping up to and including the isolation valves.

d) Will not prevent safe shutdown and isolation of the reactor unit, nor prevent maintaining the plant in a safe shutdown condition.

e) Will not damage fuel stored in the fuel pool.

f) Will not perforate Containment (i.e., no loss of leak tightness).

For structures, systems, and components protected against postulated tornado missiles, see Table 3.5.1-2 and Figure 3.5.1-1.

Safety related accumulators are provided as part of the pneumatically operated butterfly valves for the two (2) Vacuum Relief Systems located in the RAB at Elev. 286.0 near column 36/G and 22 Fx and one (1) Hydrogen Purge Exhaust System located in the Containment at Elev. 236.0 near column C-4. Each valve is equipped with one accumulator charged with plant compressed air to a pressure of 80-100 psi. The accumulators are designed to withstand pressures greatly in excess of that experienced during normal operation. Additionally, the accumulators are part of the valve operators, and as such, undergo full environmental, seismic, and operability qualification testing. In the unlikely event that an accumulator or a part thereof should become a missile, it is of negligible concern because of the small amount of impact energy it can exert on the safety related equipment in the vicinity to cause any significant damage.

In HVAC systems (ducts) some wells for thermocouples are screwed type but since the thermocouple head is connected to the conduit it should not be considered as missile source. In HVAC systems (ducts) some wells for dual thermometers are screwed but since the pressure in the duct is low, these wells should not become missiles.

Specifications for safety-related in-line, axial, and centrifugal fans for use at SHNPP explicitly require that material gage and fan housing design be shown to be sufficient to withstand equipment-generated missile penetration at the maximum operation condition to which it can be field adjusted. As described in the fan specifications, the various vendors perform analyses and furnish calculations to demonstrate that the fan housing will preclude expulsion of postulated fan-generated missiles. Missiles generated by failure of safety-related fan components, therefore, need not be further evaluated for their effects.

Specifications for non-safety fans do not require a missile penetration analysis. A separate analysis will be performed for non-safety fans if they pose a risk to nearby safety-related equipment.

Credible missiles generated by high energy system instrument wells located outside containment are evaluated in Table 3.5.1-17.

Instrument wells located in the Turbine Building will not have a detrimental impact on safety related components located in the Turbine Building. Adjacent structures housing safety related equipment are protected by exterior building walls.

There are no credible missiles from any pressurized component that can enter the spent fuel pools from sources within the Fuel Handling Building for the following reasons:

a) All pressurized systems and components in the Fuel Handling Building are located below the pool floor level. The pool floor and walls will prevent any missiles from entering the pool.

b) Low energy fluid systems and HVAC systems located in the Fuel Handling Building above the Fuel Pools will not generate any missiles.

The auxiliary feedwater pump turbine is a rotating type equipment. No missiles generated by the AFW Turbine have been postulated based on the following. The turbine has been designed and manufactured in accordance with Safety Class 3 Standards and has been designed to prevent overspeed (125%). In addition, the equipment will not normally be operating (except during testing).

3.5.1.2 Internally Generated Missiles (Inside the Containment)

Systems and components inside the Containment whose failure could result in radiological consequences in excess of 10 CFR 100 guidelines or which are required for attaining and maintaining a safe shutdown during normal or accident conditions are listed in Table 3.5.1-1.

In addition to the discussion of design bases at the beginning of Sections 3.5.0 and 3.5.1, the principal design basis inside the Containment is that missiles in coincidence with a loss-of-coolant accident (LOCA) do not cause loss of function of any redundant engineered safety features. Potential sources of missiles are:

a) High-pressure systems

b) Rotating machinery

c) Gravitational missiles

d) Secondary missiles

3.5.1.2.1 Nuclear steam supply system vendor scope

The principal design bases are that missiles generated within the Containment, coincident with a LOCA, shall not cause loss of function of any redundant engineered safety feature. Equipment inside Containment (see Section 3.5.2) has been evaluated for potential missile sources. As a result of this review, the following information concerning potential missile sources and the systems which require protection from internally generated missiles inside Containment is provided.

3.5.1.2.1.1 Missile selection

Catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings, valves and piping leading to missile generation is not credible because of the material characteristics, inspections, quality control during fabrication, erection, and operation, conservative design, and prudent operation as applied to the particular component.

The reactor coolant pump flywheel is not considered a source of missiles for the reasons discussed in Section 5.4.1. The reactor coolant pump and motor components are not considered a source of potential missiles in the event of rupture in the pump suction or discharge sections of the reactor coolant system piping. A further discussion of the reactor coolant pump including flywheel integrity is provided in Section 5.4.1.

Nuts and bolts are of negligible concern because of their small amount of stored elastic energy.

The pressurizer relief tank rupture disks are designed such that their failure should not result in the generation of missiles. Failure of the rupture disk results in splitting of the disk into quadrants. The quadrants are held in place at the circumference of the disk. To ensure that rupture disk failure and/or fluid discharge will not impose a personnel or equipment hazard, the pressurizer relief tank is located low in the Containment, outside the missile barrier provided by the secondary shield wall. Further discussion is provided in Section 5.4.11.3.

Nuclear steam supply system components, which nevertheless, are considered to have a potential for missile generation inside the Containment are the following:

a) Control rod drive mechanism housing plug, drive shaft, and the drive shaft and drive mechanism latched together.

b) Certain valves.

c) Temperature and pressure sensor assemblies.

d) Pressurizer heaters.

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

1) The control rod drive mechanism latch housings are hydrotested after they are attached to the head adaptors and checked during the hydrotest of the completed closure head at 3107 psig.

2) The control rod drive mechanism rod travel housings are a non-vented one-piece design with an integrated top plug.

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

Valves within the reactor coolant pressure boundary have been examined to identify potential missiles. As a result of this review, there are no credible failures that could result in missile generation. Therefore, valves are not considered as credible sources of missiles. Motor-operated and air-operated valves contain design features which effectively preclude the ejection of valve stems. On motor-operated valves, these features typically include a backseat integral to the valve stem which allows the valve packing to be replaced while the system is under pressure, the inherent protection provided by a threaded valve stem, and the valve operator. Valve stem ejection would not occur even if the threads were assumed to be stripped because the valve stem backseat end is machined to a large diameter and could not pass through the bore in the valve bonnet. Secondly, the nonthreaded portion of the stem would not pass through the bore of the yoke even though the threads were stripped. In a similar manner air operated valves have been evaluated and valve stem ejection is not considered credible.

Valves are designed against bonnet-to-body connection failure and subsequent bonnet ejection by means of the following:

a) Compliance with the ASME Section III.

b) Control of load during tightening of bonnet-to-body studs.

Reactor coolant pressure containing parts are designed in accordance with the requirements of the ASME Section III, Code Class 1. The valves are hydrostatically tested in accordance with the ASME Section III.

In the special case of those valves located on the top of the pressurizer, which extends above the operating deck, certain missiles, although not credible, are postulated and protection is provided via the pressurizer compartment and missile shield due to the greater potential for damage to the containment liner, engineered safeguards piping, and components located outside the secondary compartments.

The only credible source of jet-propelled missiles from the reactor coolant piping and piping systems connected to the Reactor Coolant System are the temperature sensor assemblies. The resistance temperature sensor assemblies can be of two types, "with well" or "without well." Two rupture locations have been postulated. The first location is around the welding between the boss and the pipe wall. For the "without well" element the second location is at the welding (or thread) between the temperature element assembly and the boss. The second location for the "with well" element is at the welding (or thread) between the well and the boss.

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

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

Finally, it is assumed that the pressurizer heaters could become loose and become jet-propelled missiles.

3.5.1.2.1.2 Missile Description

The control rod drive mechanism missiles are summarized in Table 3.5.1-4. The velocity of the missiles have been calculated by balancing the forces due to the water jet. No spreading of the water jet has been assumed.

The missile characteristics of the bonnets of the valves in the region where the pressurizer extends above the operating deck are given in Table 3.5.1-5.

The missile characteristics of the piping temperature sensor assemblies are given in Table 3.5.1-6. A 10-degree expansion half angle water jet has been assumed. The missile characteristics of the piping pressure element assemblies are less severe than those given in Table 3.5.1-6 since the pressure element assembly is lighter than the temperature sensor assembly.

The missile characteristics of the reactor coolant pump temperature sensor and the instrumentation well of the pressurizer are given in Table 3.5.1-7. A 10 degree expansion half angle water jet has been assumed.

The missile characteristics of the pressurizer heaters are given in Table 3.5.1-7. A 10-degree-expansion, half angle water jet was assumed. Pressurizer heaters are located on EL 261.00'. If one or more of these heaters become loose they will fall down to EL 221.00'. This area between EL 261.00' and 221.00' is open, and there is no safety related equipment to be effected by missiles.

The missiles which would be generated from temperature-sensing elements and pressure sensing elements that are part of the Reactor Coolant System are surrounded by the 4 ft thick secondary shield wall and their components arranged so that a missile generated from one component will not damage adjacent safety related components.

Internal missiles which would be generated from pressure containing components (pressure, temperature-sensing elements, pressurizer heaters, valves, etc.) that are part of the Reactor Coolant System or other high energy systems are considered in the design of the Containment Building. FSAR Section 3.5.1 lists the design basis of missile protection methods.

All non-ASME III valves inside containment are in low energy systems which are not sources of potential missiles.

3.5.1.2.2 Balance-Of-Plant Scope

The balance-of-plant scope for internally generated missiles inside the Containment and methods for protection against these missiles are discussed in Section 3.5.1.1.2.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

The potential for damage to safety-related structures, systems, and components due to turbine failure was evaluated to determine whether additional protection, beyond that inherently provided by plant building orientation and existing structural shielding, need be provided to further reduce the probability of damage.

Major turbine failure of the type which could lead to ejection of massive external missiles is sufficiently unlikely as to be an insignificant risk to public health and safety.

A review of the literature on large, > 75MW(e), turbine-generator failures during the past decade reveals that the cause of the rare event of a catastrophic failure can be placed in one of the two following categories:

a) Failure occurring at speeds below the overspeed trip due to defects in the material,

b) Failure due to overstressing arising from excessive overspeed.

Turbine discs can fail at normal speed via either a high temperature rupture or brittle fracture failure mode. The steam environment in a nuclear turbine generator is below the lower limit for high temperature rupture; therefore, failure at normal speed is limited to the brittle fracture mode. Brittle fracture type failures require not only a brittle material, but also a severe stress concentration. Siemens Energy, Inc., (the SHNPP turbine generator supplier) has minimized the sources of severe stress concentration by minimizing geometric discontinuities through prudent disc and rotor design. Improved mill techniques such as vacuum pouring have resulted in much more uniform (defect free) forgings, while better understanding of alloy chemistry and metallurgical effects of forging heat cycles has reduced the brittle ductile transition temperature to well below ambient operating conditions.

The turbine-generator manufacturer, Siemens Energy INC., takes steps to assure that their turbine generator LP rotors have the highest reliability possible:

a) Requiring conformity to written specifications covering manufacturing processes, and the chemical and mechanical properties of the rotor components,

b) ultrasonic inspection of rotor shafts and disks, and

c) inspection for surface indications using magnetic particle inspection techniques.

The above steps together with the improvement in design and inspection techniques indicate that the likelihood of a massive turbine generator failure under expected operating conditions is extremely remote.

For the new HP rotor, the following tests and inspections have been defined by the applicable Siemens QST document (Quality Assurance Specification Turbine Plants).

1. Tests of rotor shaft at Supplier:

    1.1. Dimensional and visual Inspection

    1.2. Identification inspection

    1.3. Chemical composition

    1.4. Heat treatment check of data

    1.5. Mechanical properties

    1.6. Ultrasonic test after quality heat treatment

    1.7. Surface crack test by magnetic particle inspection (MT)

    1.8. Additional dimensional inspections including outer diameter and length; roughness depth of the journal area; and axial bore length, diameter and roughness

    1.9. Check of documentation

2. Tests before, during and after turning of the turbine rotor shaft:

    2.1 Radial and axial runout check after turning

    2.2 Surface crack test by magnetic particle inspection (MT) after turning

3. Tests prior to, during and after balancing of turbine bladed rotor:

    3.1 Overspeed test of the turbine rotor

    3.2 Balance test of the turbine rotor including vibration analyses and resolution of any remaining imbalance

4. Tests of turbine bladed rotor before shipment from factory:

    4.1 Radial and axial runout check before dispatch

    4.2 Final check for completeness of all inspection steps and documentation

The occurrence of an uncontrolled overspeed is extremely remote due to the inherent reliability of the Seimens hydraulic system interfaced with the Invensys TCS controller with redundant means of preventing such an overspeed. The probability of uncontrolled overspeed at SHNPP is very remote because:

a) Control fluid is not used for lubrication oil,

b) The valves are exercised on a frequent periodic basis

c) The control fluid is continuously filtered.

d) The control fluid is periodically monitored for purity and pH.

3.5.1.3.2 Probability of Turbine Missile Generation

The US Nuclear Regulatory Commission (NRC) has defined criteria governing nuclear steam turbine start-up, continued operation and shut down requirements (reference NUREG-0800 covering sections 3.5.1.3). Two power plant layouts, namely unfavorable and favorable orientations, have been identified Harris unit 1 has what is considered an unfavorable turbine orientation.

Current NRC guidance states that maintaining an acceptably low missile generation probability, P1, by means of a suitable program of periodic testing and inspection is a reliable method for ensuring that the objective of precluding generation of turbine missiles (and hence the possibility of damage to safety-related structures, systems, and components by those missiles) can be met. For unfavorably oriented turbines, such as Harris unit 1, an acceptable frequency (P1) is less than 1x10^5 event per year.

Siemens Energy Inc. (the turbine manufacturer) has conducted a turbine missile analysis of the applicable turbine configuration for Harris unit 1. The analysis is contained in Siemens Missile Analysis Report CT-27553 (Ref 3.5.1-1). The analysis and methodology used are in compliance with NRC approved Safety Evaluation Report for HNP (NUREG-1038), along with the most recent NRC Acceptance Letter and Safety Evaluation Report for Siemens design of LP rotors (Topical Report TP-01424 dated June 7, 2004 Ref 3.5.1-2). The conclusion from this analysis is that Harris unit 1 has a P1 value below the NRC required value of 1x10^5 and that the unit can be operated for 100,000 hours between inspections.

3.5.1.3.3 Turbine Overspeed Protection

Run-away overspeed events (>120% of rated speed) are due to failure of the overspeed protection system which consists of speed monitoring devices, trip and fast closure of throttle and governor valves. The Westinghouse Owners Group (WOG) issued to its participating members conditional probabilities of destructive overspeed for valve test frequencies based on the latest analysis and valve failure data. This analysis is contained in report WCAP-16501-P (Ref 3.5.1-3)

Harris unit 1 currently operates with a 6-month valve test interval. Therefore, a probability of overspeed of 4.19E^7 per year was utilized in the Siemens missile analysis calculation discussed in section 3.5.1.3.2 which corresponds to a 6-month valve test interval for a unit with an 18-month refueling cycle. As stated previously, the results is an acceptably low probability of turbine missile generation, P1.

3.5.1.3.4 Probability Analysis

Deleted in Amendment 62.

3.5.1.3.5 Potential Turbine Missile Targets Description

Deleted in Amendment 62.

3.5.1.3.6 Turbine Generator and Turbine Missile Identification & Characteristics

Deleted in Amendment 62.

3.5.1.3.7 Turbine Valve Testing and Turbine Characteristics

A description of the turbine valve testing and the turbine characteristics is provided in Section 10.2.

3.5.1.3.8 Compliance with Regulatory Guide 1.115

Shearon Harris Nuclear Power Plant complies with Regulatory Guide 1.115, Revision 1, (Protection Against Low-Trajectory Turbine Missiles) with the exception of Position C.2 as shown in Figure 3.5.1-3.

3.5.1.4 Missiles Generated by Natural Phenomena

The worst credible missiles generated by natural phenomena to be considered in SHNPP are those generated by the design basis tornado. All structures that house systems and components to be protected against tornado generated missiles and the types of protection have been presented in Table 3.5.1-2.

The postulated tornado missiles include representative objects in the plant area which could be picked up or injected into the tornado wind field. The characteristics of the tornado generated missiles considered in the plant design are given in Table 3.5.1-3. The missiles listed in this table are considered as striking in all directions.

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

a) Structure or component is designed to withstand tornado loading or tornado missile.

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

The NRC approved a license amendment in a letter dated March 29, 2019 [License Amendment No. 169, ADAMS Accession No. ML18347A385], for SHNPP that authorized use of the Tornado Missile Risk Evaluator (TMRE) methodology. TMRE is an NRC approved methodology for determining whether protection from tornado-generated missiles is required. The methodology can only be applied to discovered conditions where tornado missile protection was not provided, and cannot be used to avoid providing tornado missile protection in the plant modification process.

Except for SSCs listed in Table 3.5.1-2a, which were demonstrated to be acceptable using TMRE methodology, the design of SHNPP is such that the structures, systems, and components specified in the appendix to the guide are protected against tornadoes and tornado missiles.

All Seismic Category I structures are capable of resisting penetration of tornado driven missiles as explained in Section 3.5.1.1.2. The tornado generated missile spectrum used in the design of Seismic Category I structures are listed in Table 3.5.1-3.

Table 3.5.1-2 identifies the structures used for protection. All Seismic Category I structures are designed with a minimum f'c of 4000 psi concrete and a minimum thickness of 24 inches in roofs and walls. Design requirements are specified in Section 3.8.

3.5.1.5 Missiles Generated by Events Near the Site

The missiles generated by events near the site are discussed in Section 2.2.3.

3.5.1.6 Aircraft Hazards

The SHNPP is remote from federal airways, airports, airport approaches, military installation or airspace usage and, therefore, an aircraft hazard analysis is not required. Specific reasons are detailed below:

1. No federal airways or airport approaches pass within two miles of SHNPP. See Section 2.2.2.5 and Figure 2.2.2-1.

2. No airports are located within five miles of SHNPP. See Section 2.2.2.5 and Figure 2.2.2-1.

3. Raleigh Executive Airport (located at Holly Springs, six miles east of SHNPP) and Deck Airport (located at Apex, eight miles north-northeast of SHNPP) and Cox Airport (located at Apex, ten miles north-northeast of SHNPP) are small general aviation airports. The movements per year from each airport is well below the aircraft analysis limits of 18,000/yr and 32,000/yr and 50,000/ yr respectively. The Raleigh-Durham Airport (located between Raleigh and Durham, 19 miles north-northeast of SHNPP) is the only major airport in the region. The movement per year from this airport is also below the aircraft analysis limit of 361,000/yr. See Section 2.2.2.5, Table 2.2.2 2, and Figure 2.2.2-1.

4. The closest military airport is Pope Air Force Base, 32 miles south of SHNPP, and next to Fort Bragg. See Section 2.2.2.5.

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

Structures, systems, and components whose failure could prevent safe shutdown of the reactor or result in significant uncontrolled release of radioactivity from the Unit are protected against externally generated missiles; they are listed in Table 3.5.1-2. The only externally generated missiles for which protection is required are tornado missiles, as discussed in Section 3.5.1.4. Most safety-related systems are located within structures that are specifically designed and constructed to withstand external missiles; therefore they are adequately protected. The penetrations, access openings, and HVAC air intake and exhaust openings in these safety-related structures are protected by steel doors and/or concrete barriers designed to withstand external missiles.

Barriers used for missile protection are listed in Table 3.5.2-1.

3.5.3 BARRIER DESIGN PROCEDURE

Barriers are designed to withstand the effects of missile impact. The local effect of missile impact is evaluated for depth of penetration into the barriers, perforating potential, and spalling effects caused by missile impact. The overall effects are evaluated for response of the structure or target, and portions thereof, to missile impact. Missiles are assumed to strike the barriers normal to the surface, and the axis of each missile is assumed to be parallel to the line of flight. These assumptions result in a conservative estimate of missile effects on barriers.

3.5.3.1 Local Damage Prediction

The evaluation of local effects include the consideration of penetration depth, thickness to prevent perforation and spalling, and punching shear effects. The local damage predictions are determined from empirical formulae based upon experimental results, as described below.

3.5.3.1.1 Concrete Barriers

a) Perforation and spalling

1) Solid Steel Missiles and Pipe Missiles

The perforation and the spalling thickness for pipe missiles and solid steel cross section missiles, such as a one-in. steel rod, are established by using the modified National Defense Research Committee (NDRC) formula presented in References 3.5.3-29 and 3.5.3-31.

$$x = \sqrt{4KNWd\left(\frac{v_o}{1000d}\right)^{1.80}} \qquad for\ \frac{x}{d} \leq 2.0 \tag{1a}$$

or

$$x = KNW\left(\frac{v_o}{1000d}\right)^{1.80} + d \qquad for\ \frac{x}{d} \geq 2.0 \tag{1b}$$

where:

N = missile shape factor

= 0.72 flat nosed bodies

= 0.84 blunt nosed bodies

= 1.00 average bullet nose (spherical end)

= 1.14 very sharp nose

K = concrete penetrability factor = $\frac{180}{\sqrt{f'c}}$

d = missile diameter (in.). All of the experimental and theoretical work concerned with local impact effects has been developed for cylindrical projectiles. For missile with noncircular cross-sections, "d" is the diameter of an equivalent solid cylindrical shaped missile with the same contact surface area as the contact surface area of the actual missile.

v_o = striking velocity of missile (fps).

W = missile weight (lbs).

x = total penetration depth (in). The depth which a missile penetrates into an infinitely thick target.

f'c = specified compressive strength of concrete, psi.

As defined herein, the penetration depth neglects all rear boundary effects and is applicable only when the target thickness is sufficiently great to prevent rear face spalling.

To prevent perforation and spalling of the concrete walls or slabs, the thickness of the concrete is determined by using the following equations:

$$\frac{e}{d} = 3.19\frac{x}{d} - 0.718\frac{x^2}{d} \qquad for\ \frac{x}{d} \leq 1.35 \tag{2a}$$

$$\frac{s}{d} = 7.91\frac{x}{d} - 5.06\frac{x^2}{d} \qquad for\ \frac{x}{d} \leq 0.65 \tag{2b}$$

$$\frac{e}{d} = 1.32 + 1.24\frac{x}{d} \qquad for\ 1.35 \leq \frac{x}{d} \leq 13.5 \tag{2c}$$

$$\frac{s}{d} = 2.12 + 1.36\frac{x}{d} \qquad for\ 0.65 \leq \frac{x}{d} \leq 11.75 \tag{2d}$$

where:

e = perforation thickness (in.), which is the maximum thickness of a target which a missile with a given impact velocity will completely penetrate. Theoretically, the exit velocity of the missile is equal to zero. For concrete, the perforation thickness is considerably greater than the penetration depth "x" due to spalling of concrete from the rear face of the target.

s = spalling thickness (in.). The thickness of the target required to prevent spalling of material from the rear face of the target for a missile with a given impact velocity.

2) Wood Missiles

Tests conducted by Sandia Laboratories for Electric Power Research Institute (Reference 3.5.3-33) and by Calspan Corporation (Reference

3.5.3-32) have indicated that postulated wood missiles splinter into pieces without causing any local damage for concrete barrier thickness of 12 in. or more. Due to the insignificant local damage caused by wood missiles, no empirical local damage formulae were established. Since tornado generated steel pipe missiles postulated for the plant require greater missile barrier thicknesses than those necessary for wood missiles, no further investigation of local damage due to wood missiles is required.

3) Automobile Missile

The automobile missile is postulated specifically to test the overall capacity of the barrier and the structure, and not penetration resistance or local effects, which are controlled by rigid missiles included in the spectrum of the tornado-generated missiles defined in Section 3.5.1.4. The automobile is a deformable missile with a large frontal area which could penetrate the barrier only through a punching shear failure. Therefore, the available empirical local damage formulas are not applicable to automobile missiles. However, the barriers are designed against punching shear failure and overall failure from impact of an automobile.

Application of the above methods for predicting local damage yield the following required minimum concrete thicknesses for the missiles listed in Table 3.5.1-3:

Concrete Strength	Wall and Roof Thickness (inches)
3000	27
4000	24
5000	21

b) Punching Shear Failure

Since perforation criteria will implicitly satisfy punching shear requirements, design for punching shear is not required in the case of hard missiles, such as solid steel missiles, whenever the barrier thickness is greater than the minimum required perforation thickness. In the case of pipe missiles, design for punching shear is not required since the barrier thickness provided to prevent spalling is based on actual test results. Similarly, punching shear is not checked for wood missiles because actual test results show no local damage.

For the automobile missile, a punching shear failure is prevented by assuring that the punching shear capacity exceeds the punching shear resulting from the applied loads. The punching shear capacity of walls and slabs is established in accordance with Sections 11.10 or 11.15.7 of the ACI 349-76 Code (Reference 3.5.3-15), with the allowables increased by a Dynamic Increase Factor as specified in Appendix C of the ACI 349-76 Code (Reference 3.5.3-16). Also, Long's criteria for punching shear capacity, presented in Reference 3.5.3-34, is used whenever applicable.

3.5.3.1.2 Steel Barriers

Steel barriers are designed to prevent perforation of the barrier. Local damage resulting rom cylindrical missile impact is predicted using the following formulas:

Stanford Research Formula:

$$\frac{E}{D} = \frac{S}{46{,}500}\left[16{,}000\,T^2 + 1500\left(\frac{WT}{W_S}\right)\right] \tag{3}$$

where:

T	=	steel thickness to be penetrated (in.)
E	=	critical kinetic energy required for penetration (ft.-lb.)
W	=	length of a square side between rigid supports (in.)
W_s	=	length of a standard width of 4 in.
D	=	missile diameter (in.)
S	=	ultimate tensile strength of the target steel plate (psi)
L	=	missile length (in.)
V_c	=	missile velocity (fps)

This formula is good for the following ranges:

$0.01 < T/D < 0.8$

$0.002 < T/L < 0.05$

$10 < L/D < 50$

$5 < W/D < 8$

$8 < W/T < 100$

$70 < Vc < 400$

Rewritten, the Stanford formula becomes:

$$T = \left(\sqrt{\frac{2.9063E}{DS} + (0.0022)\,\frac{W^2}{(W_S)^2}}\right) - 0.0469\left(\frac{W}{W_S}\right) \tag{4}$$

Ballistic Research Laboratory Formula:

$$T^{3/2} = \frac{0.5\,MV^2}{17400\,K^2 D^{3/2}} \quad = (5)$$

where:

T	=	thickness to be penetrated (in.)

M = mass of missile $\frac{(wt.)}{g}$ $\frac{(lb.-sec.^2)}{ft.}$

V = velocity of missiles (fps)

D = diameter of missile (in.)

K =constant depending on the grade of steel (usually about one).

3.5.3.1.3 Multiple Element Barriers

For multiple element missile barriers, residual velocity of the missile perforating the first element is considered as the striking velocity for the next element. The residual velocity is obtained from the difference between the kinetic energy of the missile before impact and the energy required to perforate the first barrier. Based on References 3.5.3-14 and 3.5.3-32, the residual velocity is calculated as:

$$V_r = \left(V_s^a - V_p^a\right)^{1/a} \quad for\ V_p < V_s \qquad (6a)$$

$$V_r = 0 \quad for\ Vp > Vs \qquad (6b)$$

where:

V_r = residual velocity of missile after perforation of an element (fps)

V_s = striking velocity of missile (fps)

V_p = velocity required to just perforate an element (fps)

a = power of velocity in the equation for penetration (1.8 for a multiple element barrier where the first element is steel and 2.0 where the first element is concrete are considered acceptable values).

3.5.3.2 Overall Damage

3.5.3.2.1 Impactive Load Analysis

The overall structural capacity of both concrete and steel barriers is determined to preclude structural collapse of the barrier under missile impact loading.

For all reinforced concrete or steel structural elements or systems of elements subjected to impactive loads (i.e., tornado-generated missiles), the structural response is determined by using one of the following methods:

a) The dynamic effects of the impactive loads is considered by calculating a dynamic load factor (DLF).

The available resistance to the impactive load, Rm, must be at least equal to the peak of the impactive load multiplied by the DLF, as described in References 3.5.3-20 and 3.5.3-21. Use of these references to determine the DLF was recommended in the Second ASCE Conference, Civil Engineering and Nuclear Power.

b) Dynamic effects of impactive loads are considered by using impulse, momentum, and energy balance techniques, as detailed in Williamson and Alvy "Impact Effects of Fragments Striking Structural Elements" (Reference 3.5.3-23). For concrete barriers, strain energy capacity is limited by the ductility criteria specified in Reference 3.5.3-16 and Table 3.5.3-1. For steel barriers, the maximum allowable ductility is the extreme fiber strain at the onset of strain hardening divided by the extreme fiber strain at the yield point.

A simplified method based on idealization of the actual structure to an equivalent single-degree-of-freedom system and of the impulse load time history to a simple mathematical form, is used to analyze the Seismic Category I structures.

In defining the equivalent single-degree-of-freedom system, References 3.5.3-20 and 3.5.3-21 are used to determine the load, mass, loadmass factors, and the parameters involving the maximum resistance, spring constant, and dynamic reactions of the systems under various loading conditions.

In calculating the stiffness of concrete sections, the moment of inertia must account for cracking of the concrete in accordance with ACI Publication SP 17 (1973) "Design Handbook" and the Strength Design Method of ACI 318-71.

The ultimate load capacity of concrete barriers is based on the yield line theory of reinforced concrete slabs. The collapse mechanism is a circular fan yield line pattern based on the impact of a concentrated load. The yield displacement values for structural elements are shown in Tables 3.5.3-2 and 3.5.3 3.

For a ductile missile, characterized by significant local deformation of the missile during impact (wood plank, utility pole, or steel pipe), the peak of the impactive force is determined by the formula:

$$F\ crushing = \sigma\ crushing \times A_{net}$$

where:

σ crushing = 3750 psi for wood missiles

= 60,000 psi for solid steel missiles

= 80,000 psi for steel pipe missiles

Anet = net cross sectional area of the missile

Assuming a rectangular impulse for the force function, the duration of the impulse, t_d is determined by the formula:

$$t_d = \frac{mV_m}{F\ crushing} \qquad (8)$$

t_d = Time duration of impact

where:

m = mass of missile

V_m = striking velocity of the missile

A representative forcing function for frontal impact of an automobile striking a rigid barrier (Reference 3.5.3-32), is

$F(t) = 0.625\ V_s\ W \sin 20t \quad 0 \leq t \leq 0.0785 \text{ sec}$ (9a)

$F(t) = 0 \quad t > 0.0785 \text{ sec}$ (9b)

where:

F (t) = amplitude of the force

V_s = striking velocity of the automobile

W = weight of the automobile

t = time after impact (seconds)

20t = (20 radians/sec.) (t)

Based on the above formula, the forcing function for the automobile is approximated as a rectangular shape of magnitude:

$F = 0.625\ V_s W$ (10)

and total time duration, t_d, of

$t_d = \frac{M\ V_S}{F}$ (11)

where M is the mass of the automobile.

3.5.3.2.2 Design of Concrete Barriers

When a missile impacts near the middle of a two-way slab, the analytical approach shown in Table 3.5.3-3 is utilized. The resistance and yield displacement values are calculated in accordance with the boundary conditions and long/short sides ratio of the two-way slab. The ductility factors are shown in Table 3.5.3-1

In the case where a missile strikes a beam, conventional analysis is performed as shown in Table 3.5.3-2. The ductility factors are shown in Table 3.5.3-1. The transfer of the local loadings to the rest of the structure is treated on an elastic basis in accordance with the acceptance criteria presented in Sections 3.8.1 through 3.8.5.

3.5.3.2.3 Design of Steel Barriers

In the analysis of impact effects on steel plate barriers when a missile hits near the center of the plate, the resistance function specified in Table 3.5.3-3 is used in conjunction with the allowable ductility factors in Table 3.5.3-1.

For impact effects in the vicinity of a support, the Stanford Research penetration formula is used. This automatically precludes punching shear failures.

REFERENCES: SECTION 3.5

3.5.1-1 CT-27553 Revision 0 Dated 11/25/2016 Siemens Missile Analysis Report (Calculation HNP-MECH/M-1239).

3.5.1-2 TP-04124 “Missile Probability Analysis for the Siemens 13 9m2 Retrofit Design of Low-Pressure Turbine by Siemens AG”. Submitted to the Nuclear Regulatory Commission as Topical Report TP-04124-NP-A. For Public Record, June 7, 2004, Siemens Westinghouse Power Corporation.

3.5.1-3 WCAP-16501-P Revision 0 Dated, February 2006, “Extension of Turbine Valve Test Frequency Up to 6 Months for BB-296 Siemens Power Generation (Westinghouse) Turbines with Steam Chests” Westinghouse Electric Company LLC, Westinghouse Proprietary Class 2.

3.5.1-4 Deleted by Amendment 62.

3.5.1-5 Deleted by Amendment 62.

3.5.1-6 IEEE Trial-Use Guide: General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems, ANSI N41.4, IEEE Std. 352-1972.

3.5.1-7 W. H. Van Allen, Reliability Engineering, ARINC Research Corporation, Englewood Cliffs, New Jersey, Prentice Hall, 1964, Chapter 5.

3.5.1-8 A. Amirikian, "Design of Protective Structures," NAVDOCKS P-51, August 1950 (Presented at annual meeting of the ASCE, Chicago, Illinois, October 11-14, 1950).

3.5.3-1 NRC Regulatory Guide 1.76 "Design Basis Tornado for Nuclear Power Plants"

3.5.3-2 NRC Regulatory guide 1.70.16 "Information for Safety Analysis Reports. Missile Barrier Design Procedures"

3.5.3-3 NRC Regulatory Guide 1.13 "Fuel Storage Facility Design Basis"

3.5.3-4 NRC Regulatory Guide 1.117, "Tornado Design Classification"

3.5.3-5 NRC Standard Review Plan Sec 2.3.1 "Regional Climatology"

3.5.3-6 NRC Standard Review Plan Sec 2.3.2 "Local Meteorology"

3.5.3-7 NRC Standard Review Plan Sec. 3.5.1.4 "Missiles Generated by Natural Phenomena"

3.5.3-8 NRC Standard Review Plan Sec 3.5.2 "Structures, Systems, and Components to be Protected from Externally Generated Missiles"

3.5.3-9 NRC Standard Review Plan Sec 3.5.3 "Barrier Design Procedures"

3.5.3-10 NRC Standard Review Plan Sec 3.8.4 "Other Seismic Category I Structures"

3.5.3-11 NRC Standard Review Plan Sec 3.8.5 "Foundations"

3.5.3-12 NRC Branch Technical Position AAB-32 "Tornado Design Classification"

3.5.3-13 ACI 318-71 "Building Code Requirements for Reinforced Concrete"

3.5.3-14 Recht, R. F., and T. W. Ipson, "Ballistic Perforation Dynamics," "Journal of Applied Mechanics, Transactions of ASME, Vol 30, Series E, No. 3 (September 1963)

3.5.3-15 ACI 349-76 "Code Requirements for Nuclear Safety Related Concrete Structures"

3.5.3-16 ACI 349 Committee - Appendix C "Special Provisions for Impulsive and Impactive Effects"

3.5.3-17 Deleted by Amendment 62.

3.5.3-18 Manual, S. A., "Impactive Dynamic Analysis," ASCE National Structural Engineering Meeting, Baltimore, Maryland, (April 19 23, 1971)

3.5.3-19 ASCE "Manual of Standard Practices for Design of Nuclear Power Plant Facilities - Chapter 6 - Design Against Impulse and Impact Loads."

3.5.3-20 Biggs, J. M. "Introduction to Structural Dynamics" McGraw Hill 1964

3.5.3-21 Norris C. H. et. al. "Structural Design for Dynamic Loads" McGraw Hill 1969

3.5.3-22 Amerikian, A. "Design of Protective Structures" Report NP-3726, Bureau of Yards and Docks, Dept of Navy 1950

3.5.3-23 Williamson, R. A. and Alvy, R. R. "Impact Effect of Fragments Striking Structural Elements" Holmes and Narver, Inc 1973

3.5.3-24 Samuely, F. J. and Himann, C. W. "Civil Protection" The Architectural Press (London, 1939)

3.5.3-25 Zaghool, E. E. F. and Rawdon de Paiva, H. A., "Test of Reinforced Concrete Flat Plate Floors," Journal of the Structural Division, ASCE Vol 96 ST3 (March 1970)

3.5.3-26 Moe, J., "Shearing Strength of Reinforced Concrete Slabs and Footings Under Concentrated Loads" Portland Cement Association Bulletin D47 (April 1961)

3.5.3-27 Newmark, N. M., "An Engineering Approach to Blast-Resistant Design" ASCE Transactions Paper No. 2786

3.5.3-28 Blume, T. A., Newmark N. M., and Corning L. H., "Design of Multistory Reinforced Concrete Building for Earthquake Motions" PCA 1961

3.5.3-29 Kennedy, R. P., "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects" Holmes and Narver, Inc., Anaheim, California

3.5.3-30 Stevenson, J. D., "Simplified Procedures for Analysis and Design of Concrete Structures Subjected to Missile Loads with Particular Application to Nuclear Power Plant Facilities"

3.5.3-31 Gwaltney, R. C., "Missile Generation and Protection in Light Water Cooled Power Reactor Plants" USAEC Report ORNL NSIC 22 (September 1968)

3.5.3-32 Rotz, J. V., "Symposium on Tornadoes; Assessment of Knowledge and Implications for Man" Texas Tech University, Lubbock, Texas, June 22 24, 1976

3.5.3-33 "Full-Scale Tornado-Missile Impact Tests" EPRI NP-148, Project 399, by Sandia Laboratories, Tonopah Test Range, Tonopah, Nevada for Electric Power Research Institute, Interim Report, April 1976.

3.5.3-34 Long, A. E., "A Two-Phase Approach to the Prediction of the Punching Strength of Slabs" ACI Journal, February 1975

3.5.3-35 Timoshenko S., Woinowsky - Krieger S., "Theory of Plates and Shells" McGraw-Hill 1959.

3.5.3-36 NRC Regulatory Guide 1.115, Rev. 1 "Protection Against Low Trajectory Turbine Missiles."

3.5.3-37 Sliter, G. E., "Residual Velocity of Missiles Perforating Concrete, Second ASCE Conference on Civil Engineering and Nuclear Power; Vol IV: Impactive and Impulsive Loads."

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

This section describes the design bases and measures that are taken to demonstrate that the systems, components and structures required to safely shutdown and maintain the reactor in a cold shutdown condition are adequately protected against the effects of blowdown jets, reactive forces, and pipe whip resulting from postulated rupture of piping both inside and outside Containment.

The following criteria are used for the protection against dynamic effects associated with postulated pipe rupture:

a) Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" (3/75),

b) Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment" (5/73): Used for analysis of ASME Code Class 1 piping inside the Containment, as explained in Section 3.6.2.1.1.2. For ASME III, Code Class 2 and 3 piping inside Containment, Branch Technical Position MEB 3-1 is used.

c) Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment" (3/75): Used for analysis of ASME III Code Class 2 and 3 piping both inside and outside Containment. Also used for ASME Code Class 1 piping inside Containment as explained in Section 3.6.2.1.1.2.

d) Structural Engineering Branch Document B, "Structural Design Criteria for Evaluating the Effects of High Energy Pipe Breaks on Seismic Category I structures outside the Containment"

   This criteria also applied to Seismic Category I structures inside Containment.

e) Guidance received from the NRC staff and documented in the correspondence listed below is followed for piping other than reactor coolant loop piping:

   1) April 25, 1975, meeting minutes transmitted by Mr. A. Barchas (Ebasco Services, Inc.) to Mr. P. Mattews (NRC) in his letter of May 8, 1975.

   2) Comments on the April 25, 1975, meeting minutes by Mr. J. P. Knight (NRC) to Mr. A. Barchas in his letter of May 28, 1975.

   3) August 5, 1975, meeting minutes transmitted to Mr. B. Rusche (NRC) by Mr. J. A. Jones (CP&L) in September, 1975.

   4) Letter from T. M. Novak of NRC to E. E. Utley of CP&L Dated 8/15/85 Regarding Elimination of Arbitrary Intermediate Breaks.

f) Guidance received from the NRC staff and documented in the correspondence listed below is followed for reactor coolant loop piping:

   1) Letter from Mr. George W. Knighton of NRC to Mr. E. E. Utley of CP&L dated June 5, 1985 regarding exemption from a portion of General Design Criterion 4 of Appendix A to 10 CFR Part 50.

   2) General Design Criterion 4 of Appendix A to 10 CFR Part 50 final rule making change per 52 FR 41288 dated October 27, 1987.

After the pipe break event, the consequences are considered to be the following:

a) pipe whip,

b) jet impingement,

c) compartment pressurization,

d) compartment flooding, and

e) high temperature/high humidity environment.

f) radiation

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS BOTH INSIDE AND OUTSIDE OF CONTAINMENT

3.6.1.1 Design Bases

The following design bases are considered in determination of the dynamic effects associated with the pipe rupture:

a) The assumptions (i.e., loss of offsite power, single active failure) used in conducting pipe break analyses are listed in Section 3.6.1.3.

b) The effects of each postulated piping failure will result in offsite releases less than 10 CFR 50.67 limits.

c) The functional capability of systems and equipment required to assure safe shutdown and the ability to maintain a cold shutdown condition after a given break must not be impaired by the pipe whip, jet impingement or environmental conditions resulting from that break.

d) Damage to any structure, directly caused by the pipe whip, jet impingement or environmental consequences of a given break, must not impair the function of any system or equipment required to assure safe shutdown and the ability to maintain a cold shutdown condition.

e) The effects of a postulated failure, including radiation and environmental conditions do not preclude habitability of the Control Room or any location where manual action is required to achieve and maintain a cold shutdown condition. The structural integrity of these areas shall be preserved.

f) The design leak-tightness integrity of the Containment is preserved.

   Essential systems are those that are needed to shutdown the reactor and to mitigate the consequences of a postulated pipe break, without offsite power. However, depending upon the type and location of a postulated pipe break, certain safety equipment may not be classified as essential for that particular event.

The essential systems required for each postulated piping failure are identified below.

a) The following systems or portions of these systems are required to mitigate the consequences of postulated breaks of reactor coolant pressure boundary piping that will result in a loss-of-coolant accident (LOCA). For the definition of reactor coolant pressure boundary (RCPB), see Figure 3.6.1-1.

1) Reactor Protection System

2) Engineered Safety Features Actuation System

3) Safety Injection System

4) Containment Spray System

5) Reactor Coolant System

6) Main Steam Supply and Feedwater Systems (from the steam generator out to the containment isolation valves)

7) Auxiliary Feedwater System

8) Class IE electrical systems, AC and DC (including switchgear, batteries, and distribution systems)

9) Diesel generators (including jacket water cooling and lube oil)

10) Diesel Generator Fuel Oil Storage and Transfer System

11) Containment Cooling System

13) Component Cooling Water System (portions required for operation of other listed systems)

14) Heating, ventilation, and air conditioning systems required for operation of other listed systems and Control Room including Essential Services Chilled Water System

15) Containment Isolation System

16) Post-accident monitoring instruments

17) Containment area radiation monitors (four area monitors in the Containment)

18) Residual Heat Removal System

19) Control rod drive mechanisms (CRDM) (protection of the CRDMs is required to the extent that the control rods will be released for insertion into the core during accident conditions; protection of power and control circuitry is not required)

20) Service Water System (safety related portions)

b) The following systems or portions of these systems are required to mitigate the consequences of postulated breaks of main steam, feedwater or steam generator blowdown piping.

1) Reactor Protection System

2) Engineered Safety Features Actuation System

3) Auxiliary Feedwater System

4) Safety Injection System

5) Containment Spray System (for breaks inside the Containment only)

6) Reactor Coolant System (maintain RCPB)

7) Main Steam Supply and Feedwater System (from the steam generator out to the containment isolation valves)

8) Residual Heat Removal System

9) Class IE electrical systems, AC and DC (including switchgear, batteries, and distribution systems)

10) Diesel generators (including jacket water cooling and lube oil)

11) Diesel Generator Fuel Oil Storage and Transfer System

12) Containment Cooling System (for breaks inside the Containment only)

13) Component Cooling Water System (portions required for operation of other listed systems)

14) Heating, ventilation, and air conditioning systems required for operation of other listed systems and Control Room (including Essential Services Chilled Water System)

15) Instrumentation required for post-accident monitoring

16) Control rod drive mechanisms (CRDM) (protection of the CRDMs is required to the extent that the control element assemblies will be released for insertion into the core during accident conditions; protection of power and control circuitry is not required)

17) Containment Isolation System

18) Service Water System (safety related portion)

c) Other postulated high energy line breaks not included in listings a) and b) above. These breaks are evaluated on a case-by-case basis to ensure that the design bases a) through f) listed above are met.

Piping isometrics with rupture points indicated are provided in Appendix 3.6A. Appendix 3.6A also identifies those systems or components that are required for plant safety or shutdown and must be protected by a barrier from the effects of the postulated break.

3.6.1.2 Description

3.6.1.2.1 High energy piping

Piping system which, during normal operating conditions, exceed 200 F and/or 275 psig are considered to be high energy. The following systems, or portions of systems, are evaluated as high energy pipelines for pipe rupture.

1. Inside Containment (High Energy)

   a. Reactor Coolant System, including hot, crossover and cold legs, pressurizer surge, spray and auxiliary spray, safety and relief lines, and drain lines

   b. Chemical and Volume Control System (letdown, charging and seal injection lines)

   c. Safety Injection System (from the reactor coolant loop up to the first normally closed valve)

   d. Residual Heat Removal System (portions only)

   e. Main Steam Supply System

   f. Feedwater System

   g. Steam Generator Blowdown System

   h. Auxiliary Feedwater System

2. Outside Containment (High Energy)

   a. Main Steam Supply System

   b. Extraction Steam System

   c. Auxiliary Steam System (including condensate return)

   d. Feedwater System

   e. Chemical and Volume Control System (charging pump discharge lines, i.e. charging line and seal injection lines; letdown line to downstream of letdown heat exchanger)

   f. Steam Generator Blowdown System

   g. Auxiliary Feedwater System

   h. Safety Injection

Portions of the main steam and feedwater piping are routed through the Steam Tunnel in the RAB. For further information refer to Section 3.6.2.1.4.

The Main Steam and Feedwater Tunnel (MSFT) is located in the Reactor Auxiliary Building (RAB) adjacent to the Reactor Containment Building at azimuth 270°. The floor of the tunnel is at elevation 263.00, and the penthouse roof is at elevation 318.00. The tunnel is bounded on the west side by the containment wall, the north side by a wall at column 29, the east side by a wall at column D, and the south side by a wall at column 25. All walls, floors, and roofs are constructed of concrete and are at least 4 ft. thick. Figures 1.2.2-27, 1.2.2-31, 1.2.2-35, 1.2.2-39, and 1.2.2-43 provide plan and elevation views of this area.

The main steam, main feedwater, and auxiliary feedwater piping are located within the steam tunnel. The main steam isolation valves, code safety valves, main steam power operated relief valves, feedwater isolation valves, and the auxiliary feedwater isolation valves are located within the steam tunnel. Associated power and control cables are also located within this area. Also in the compartment are various instrumentation, branch piping lines, hydrazine and ammonia chemical addition lines, steam supply to the turbine driven auxiliary feedwater pump, and bypass loops of the main steam isolation valves.

Located adjacent to the north, east and south walls of the tunnel at elevation 286.00 are compartments housing battery rooms, safety related switchgear, safety related motor control centers, safety related air handling units, hydrogen recombiner power panel abandoned in place, the auxiliary control panel, and the diesel sequencer. Located adjacent to the north, east, and south walls at elevation 261.00 are compartments housing the electrical penetration area, volume control tank, RHR heat exchangers, auxiliary feedwater piping and control valve gallery, HVAC safety related chillers and air handling units, and various motor control centers and instrument racks. Located below the steam tunnel floor at elevation 236.00 is the mechanical penetration area, auxiliary feedwater pumps and associated piping, component cooling water heat exchangers and pumps, high head safety injection pumps (charging pumps), safety related HVAC equipment, associated motor control centers, instrument racks and safety related piping.

3.6.1.2.2 Moderate energy piping. Moderate energy systems are defined as follows:

a) Systems where <u>both</u> of the following apply:

    (1) Maximum normal operating pressure is 275 psig or less, and

    (2) Maximum normal operating temperature is 200 F or less.

b) Systems which exceed either or both of the above conditions, but only for <u>less than two percent of system</u> normal operating time (not including testing).

c) Systems which exceed either or both of the above conditions, but which are in operation <u>less than one percent of plant</u> normal operating time (not including testing).

The following systems, or portions of systems, are evaluated as moderate energy pipelines for pipe rupture:

a) Inside Containment (Moderate Energy)

    1) Component Cooling Water System

2) Service Water System

3) Residual Heat Removal System (downstream of the second isolation valve to the containment penetration).

4) Safety Injection System (portions which are in operation as part of the Residual Heat Removal System).

5) Waste Processing System

6) Instrument Air System

7) Miscellaneous Drains System

8) Nitrogen System

9) Reactor Makeup Water System

10) Service Air System

11) Sampling System

12) Valve Leakoff System

b) Outside Containment (Moderate Energy)

1) Component Cooling Water System

2) Fire Protection System

3) Residual Heat Removal System

4) Containment Spray System

5) Chilled Water Supply System (Nonessential and Essential)

6) Chilled Water Return System (Nonessential and Essential)

7) Process Sampling System

8) Demineralized Water System

9) Circulating Water System

10) Fuel Pool Cooling and Cleanup System

11) Waste Processing System

12) Condensate System

13) Diesel Generator Fuel Oil transfer System

14) Potable and Sanitary Water System

15) Chemical and Volume Control System

16) Safety Injection System

17) Boron Recycle System

18) Condenser Evacuation System

19) Caustic and Acid System

20) Chemical Feed System

21) Carbon Dioxide System

22) Service Air System

23) Primary Filtered Makeup Water System

24) Auxiliary Boiler Fuel Oil System

25) Hydrogen System

26) Instrument Air System

27) Lube Oil System

28) Miscellaneous Drains System

29) Make-up Plant Water

30) Nitrogen System

31) Oxygen System

32) Reactor Makeup Water System

33) Screen Wash System

34) Sampling System

35) Cooling Tower Blowdown System

36) Valve Leakoff System

37) Waste Processing Building Cooling Water System

38) Secondary Waste Treatment System

39) Alternate Seal Injection System

The pipe rupture isometrics provided in Appendix 3.6A show the separation of high and moderate energy piping. Moderate energy pipe rupture sketches provided in Appendix 3.6A indicate the extent of moderate energy fluid systems.

3.6.1.2.3 Non-High and Non-Moderate Energy Piping

Piping systems which are neither in operation during normal plant conditions nor whose pressure and temperature exceed atmospheric and ambient conditions, respectively are classified as neither high-energy nor moderate-energy systems. No breaks or through-wall leakage cracks are postulated.

Systems in which the entire piping or portions of the piping meet the above criteria are as follows:

a) Inside Containment

1) Containment Spray System

2) Fire Protection System

3) Safety Injection System (the portions not normally pressurized)

(The Safety Injection piping inside containment from the containment penetrations (M-17, M-20, M-21, and M-22) to the first check valves (1SI-V17SA, 1SI-V23SB, 1SI-V29SA, 1SI-V63SA, 1SI-V69SB, 1SI-V75SA, 1SI-V39SA, 1SI-V45SB, 1SI-V51SA, 1SI-V84SA, 1SI-V90SB, and 1SI-V96SA) at the safety class break is not normally pressurized, and is therefore, excluded from break and through wall crack evaluation.)

b) Outside Containment

1) Containment Spray System

2) Auxiliary Feedwater System (pump suction to condensate storage tank)

3) Fire Protection System

4) Diesel Generator Fuel Oil and Transfer System

5) Leak Rate Testing System

An analysis of pipe break events postulated in the design was performed to determine the effect on those safety related systems and components that provide protective actions required to mitigate the consequences of the postulated pipe break event.

3.6.1.2.4 Protective Measures

Whenever the separation inherent in the plant design is shown to assure the functional capability of the essential systems required following a postulated pipe break event, no additional protective measures are required.

When necessary, additional protective measures such as those described below are incorporated into the design to assure the functional capability of essential systems following the postulated pipe break event.

a) Separation - The plant arrangement provides separation where practical between redundant safety systems in order to prevent loss of safety function resulting from the dynamic effects of the rupture event. Separation between redundant safety systems with their related auxiliary supporting features, therefore, is the basic measure incorporated in the design to protect against the dynamic effects of postulated pipe break events.

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

1) Safety related systems are located away from most high energy piping.

2) Redundant (i.e., "A" and "B" trains) safety subsystems and components are located in separate compartments.

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

b) Barriers, Shields, and Enclosures - Structures required to provide protection against the effects and consequences of the pipe break are evaluated to determine that these structures are designed to accomplish this function. Damage to any structure caused by pipe rupture, jet impingement, missiles or environmental consequences will not impair the capability to safely shutdown the plant. Structures providing barrier protection are designed to withstand the pressure, humidity and temperature transients which result from a high energy piping system break plus normal operating loads plus SSE loads.

   Where it is not feasible or practical to isolate the Seismic Category I piping, the adjacent non-seismic Category I piping was seismically designed in accordance with C.2 and C.4 of Regulatory Guide 1.29. Refer to FSAR Section 3.7.3.13 on the interaction of other piping with Seismic Category I piping.

c) Piping Restraint Protection - Where adequate protection does not already exist due to separation, barriers or shields, piping restraints are provided as necessary to meet the functional protection requirements.

Restraints are not provided when it can be shown that the broken pipe will not cause unacceptable damage to essential systems or components.

The design criteria for restraints are given in Section 3.6.2.3.

The effects of steam environment on safety related equipment are discussed in Section 3.11. An analysis of the potential effects of missiles is discussed in Section 3.5.

3.6.1.3 Safety Evaluation

By means of the design features such as separation, barriers, and pipe whip restraints, all of which are discussed in Section 3.6.1.2, the effects of pipe break do not damage essential systems to an extent that would impair their functional capability.

Specific design features listed in Section 3.6.1.2, used for protecting the essential systems, are identified in Appendix 3.6A.

In conducting the pipe rupture analyses, the following assumptions are used:

a) If the postulated pipe failure results in an automatic separation of the turbine generator from the power grid, or results in an automatic reactor trip, then offsite power is assumed to be unavailable.

b) If the postulated pipe failure requires safety system response to the event, the analysis assumes a single active component failure in either the safety systems required to mitigate the consequences of the event or their auxiliary supporting features except as noted in d) below. This single active failure is in addition to the postulated pipe failure and any direct consequences of the piping failure.

c) Operator action to mitigate the consequences of the postulated pipe failure is analyzed for each specific event. The feasibility of initiating operator actions on a timely basis, as well as the accessibility provided to allow the operator actions, is demonstrated.

d) 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 shutdown the reactor and mitigate the consequences of the piping failure), single failures of components in the other train or trains of that system only are not 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.

e) An unrestrained whipping pipe is considered capable of

    1) rupturing impacted pipes of smaller nominal pipe sizes, and

    2) developing through-wall leakage cracks in equal or larger nominal pipe sizes with thinner wall thicknesses

f) Jet impingement forces from a given pipe of specified nominal pipe size and wall thickness are considered capable of:

    1) rupturing targeted pipes of smaller nominal pipe size, and

    2) developing through-wall leakage cracks in pipe of larger nominal pipe size and thinner wall thickness.

The energy level in a whipping pipe is considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

The high energy lines, inside and outside Containment, evaluated in the analysis are described in Sections 3.6.1.1 and 3.6.1.2. The results of these analyses are presented in Appendix 3.6A.

Given the separation criteria in Section 3.6.1.2, and the pipe break criteria in Section 3.6.2, the effects of high energy pipe breaks are generally not analyzed where it is obvious that all essential systems, components, or structures are physically remote from a break in that piping run.

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

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

3.6.2.1.1 High Energy Piping Systems (Inside Containment)

This section provides the criteria used to locate the postulated break points for high energy fluid systems inside Containment.

3.6.2.1.1.1 Reactor Coolant System (RCS) Loop Piping

Breaks are not postulated in the ASME Section III Class 1 RCS main loop piping based on leak before break (LBB) analysis, as discussed in References 3.6.2-1, 3.6.2-9, 3.6.2-10, 3.6.2-11, 3.6.2-12, and 3.6.2-13 which projects the LBB analysis through the period of extended operation for License Renewal. Additional details of the analysis applicable to License Renewal are provided in Chapter 18. The LBB analysis results in postulated breaks in attached auxiliary branch line connections (Table 3.6.2-1).

3.6.2.1.1.2 ASME Section III, Code Class 1 Piping (Excluding RCS Main Loop Piping)

Regulatory Guide 1.46 has been followed in all matters except for the postulation of break points. The criteria of MEB 3-1 for Class 1 piping has been adapted such that pipe breaks are postulated to occur at:

a) terminal ends.

b) intermediate locations where the maximum stress range as calculated by Eq. (10) and either (12) or (13) exceeds 2.4 S_m.

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

A minimum of two arbitrary intermediate breaks (AIBs) are postulated between the terminal ends of ASME Section III, Code Class 1 piping which fail to meet all the following criteria:

a) Large dynamic loads on the piping system are not anticipated.

b) The piping system material or service condition is not susceptible to stress corrosion cracking.

c) Thermal fatigue, such as in mixing situations, is not present.

d) All safety related equipment in the vicinity of the piping system has been environmentally qualified for the nondynamic effects of a nonmechanistic pipe break.

AIBs are those locations which, based on piping stress analysis results, are below the stress and fatigue limits specified in MEB 3-1, but are selected to provide a minimum of two postulated breaks between the terminal ends of a piping system. AIBs are located at those places in the line that have stresses closest to the stress and fatigue limits specified in MEB 3-1.

The Summer 1979 Addenda of the ASME Code will be used for fatigue evaluation.

Terminal ends are extremities of piping runs that connect to structures, components (i.e., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping thermal expansion.

A branch connection to a main piping run is a terminal end of the branch run, except if the branch nominal size is greater than or equal to 75 percent of the main run size and the stress analysis model included the branch run in the same model.

In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up the first normally closed valve), the terminal end of such runs is the piping connection to this closed valve.

3.6.2.1.1.3 ASME Section III, Code Classes 2 and 3 Piping

Rupture locations are postulated to occur in any piping or branch run, at terminal ends and intermediate locations as per Branch Technical Position MEB 3-1 and Generic Letter 87-11. Break locations are postulated on the aforementioned piping by the following:

a) At terminal ends. Terminal ends are as previously defined in Section 3.6.2.1.1.2.

b) At other locations between terminal ends where stresses under normal and upset plant conditions and an OBE event as calculated by Equations (9) and (10) of Paragraph NC 3652 ASME Code, Section III, exceed 0.8 ($1.2S_h+S_A$).

Per Generic Letter 87-11, a minimum of two AIBs (as defined in 3.6.2.1.1.2) are no longer required to be postulated between the terminal ends of ASME Section III, Code Classes 2 and 3 piping which fail to meet all the following criteria:

a) Large dynamic loads on the piping system are not anticipated.

b) The piping system material or service condition is not susceptible to stress corrosion cracking.

c) Thermal fatigue, such as in mixing situations, is not present.

d) All safety related equipment in the vicinity of the piping system has been environmentally qualified for the nondynamic effects of a nonmechanistic pipe break.

e) The effects of local welded attachments have been considered as required by ASME Code Article NC/ND-3645.

3.6.2.1.2 High Energy Piping Systems (Outside Containment)

This section discusses the criteria used for location of postulated break points in ASME Section III, Code Class 2 and 3 high energy fluid systems outside the Containment. The postulated rupture points are located as previously discussed in Section 3.6.2.1.1.3.

Refer to Figure 3.6.2-1 which illustrates the main steam supply and feedwater piping outside Containment.

The main steam lines are classified as Safety Class 2/Seismic Category I from the steam generators inside Containment up to and including the main steam isolation valve on each line. From this point, running downstream horizontally to the end of the pipe tunnel, through a 90 degree vertical elbow, through risers into the main steam header, to the end of the pipe rupture restraint system, the piping is classified Non-Nuclear Safety/Seismic Category I. The piping described above (between the containment penetration and the pipe rupture restraint systems outside Containment) is not subject to postulation of pipe breaks for design purposes in accordance with Section 3.6.2.1.4. The piping downstream of the pipe rupture restraint system is classified as Non-Nuclear Safety/Seismic Category I up to the last seismic restraint as detailed in Figure 3.6.2-1, and designed to appropriate code stress limits and 10 CFR 50 Appendix B QA requirements. The location of the first main steam line break in this section of piping is postulated at the elbow in the Turbine Building (adjacent to the Reactor Auxiliary Building) where the steam lines are non-Seismic Category I. The description of the first postulated break in the main steam lines is consistent with FSAR Figure 3.6.2-1. However, the assumed crack in the main steam line for steam tunnel subcompartment analysis is not shown. This is due to the fact that Figure 3.6.2-1 and associated description was intended to describe compliance with SRP 3.6.2 for mechanistic pipe ruptures i.e., those breaks postulated based upon stress and/or fatigue analysis.

The feedwater piping is classified as Safety Class 2/Seismic Category I from the steam generators inside Containment up to and including the feedwater check valve outside Containment. The piping passes through the pipe tunnel, (each feedwater line routed below its corresponding main steam supply line) continues through the Reactor Auxiliary Building with only large radius bends (greater than five diameters) until the Reactor Auxiliary Building-Turbine Building interface. The portion of piping from the check valve up to this interface is classified as Non-Nuclear Safety/Seismic Category I (with 10 CFR 50, Appendix B QA Program being applied). The balance of piping downstream of the last seismic restraint routed in the Turbine Building is classified as Non-Nuclear Safety/Non-Seismic Category I.

The feedwater piping between the containment penetration and the end of the Reactor Auxiliary Building is not subject to postulation of pipe breaks in accordance with Section 3.6.2.1.4 and points of clarification noted above.

The location of the first postulated feedwater piping failure outside Containment is taken at the Reactor Auxiliary Building-Turbine Building interface.

3.6.2.1.3 Moderate Energy Piping Systems (Both Inside and Outside Containment)

For ASME Code Class 2, 3 and non-nuclear safety, moderate energy piping systems routed in areas containing no high energy piping subjected to postulated piping failures, but which are located near components or structures required for safe shutdown, through wall leakage cracks are postulated to occur at any location.

Through wall leakage cracks are postulated in Seismic Category I fluid system piping located within, or outside and adjacent to, protective structures, except where the maximum stress range, as calculated by Equations (9) and (10) of paragraph NC-3652 of the ASME Code Section III, in these portions of Code Class 2 or 3 piping, or Non-Nuclear piping is less than 0.4 (1.2 S_h+S_A).

Through wall leakage cracks are postulated in fluid system piping designed to non-seismic standards so as not to result in any loss of capability of essential systems and component failure and still perform all functions required to shutdown the reactor and mitigate the consequences of the postulated piping failure.

Through wall leakage cracks are not postulated in moderate energy fluid piping systems that are located in the same area as high energy fluid piping systems which have previously determined rupture locations, provided such cracks would not result in more limiting environmental conditions.

Inside Containment, high energy piping such as main steam supply, feedwater and reactor coolant loop piping breaks are postulated and evaluated for environmental conditions which result in more severe environmental conditions, and on this basis, no moderate energy cracks are postulated inside Containment.

Leakage cracks in fluid system piping between containment isolation valves is described in Section 3.6.2.1.4.

3.6.2.1.4 High and Moderate Energy Piping Between the Containment Isolation Valves

a) High Energy Systems - Breaks are not postulated in those portions of piping passing through Containment provided the requirements of ASME Section III, Subarticle NE-1120 and the following additional design requirements are met.

    1) All piping between containment isolation valves is ASME Code Class 2 piping. The following design stresses are not exceeded for ASME Code, Section III, Class 2 piping:

    a. The maximum stresses as calculated by the sum of Equations (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 do not exceed 0.8 (1.2 S_h+S_A).

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

2) To the extent practicable, welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses, or tests are performed to demonstrate compliance with the limits of item 1) above.

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

4) The length of these portions of piping is reduced to the minimum length practical.

5) To the extent practicable, the design of pipe anchors or restraints (i.e., connections to containment penetrations and pipe whip restraints) will not require welding directly to the outer surface of the piping (i.e., flued integrally forged pipe fittings may be used) except where welds are 100 percent volumetrically examinable in service and detailed stress analysis is performed to demonstrate compliance with the limits of item 1) above.

6) Guard pipes have not been utilized because of the single barrier containment design.

7) Inservice examination of all pipe welds in the break exclusion region is conducted during each inspection interval as defined in Section 6.6.8 herein.

Breaks are postulated in the main steam supply and feedwater lines in accordance with Reference 3.6.2-2 and the clarifications of Reference 3.6.2-3. These references require that a crack equivalent to the flow area of a single ended pipe break, non-mechanistic in nature (i.e. not based on stress and fatigue analysis) be considered in the steam tunnel area. The analysis is limited to the pressurization and environmental effects of the break only. Refer to Section 3.6A.2.3 for the analysis. Pipe whip and jet impingement effects are excluded.

As a result of the above considerations, the tunnel design includes a penthouse which provides adequate ventilation to prevent overpressurization of the tunnel compartment. Water-tight doors have been provided which prevent spillage of water beyond the tunnel boundaries onto any essential equipment. Essential equipment within the tunnel is located above the maximum attainable flood level or is not impacted by the flood to a degree that would challenge a function required to shut down the reactor and mitigate the consequences of a postulated pipe failure. See Section 3.6A.3.2. All essential equipment within the tunnel has been environmentally qualified to withstand the effects of the worst-case break so as to not impact a required essential function (see Section 3.11).

b) Moderate Energy System - Leakage cracks are not postulated in those portions of ASME Code Class 2 fluid system piping between containment isolation valves provided they meet the requirements of the ASME Code Section III, Subarticle NE-1120, and are designed such that the maximum stress range as calculated by Equations (9) and (10) of Paragraph NC-3652 of the ASME Code, Section III, does not exceed 0.4 (1.2 S_h+S_A) for ASME Code, Section III, Code Class 2 piping.

3.6.2.1.5 Types of Breaks and Leakage Cracks in Fluid System Piping

a) Circumferential Pipe Breaks - The following circumferential breaks are postulated in high energy fluid system piping:

1) Circumferential breaks are postulated in fluid system piping and branch runs exceeding a nominal pipe size of one inch, except where the maximum stress range exceeds 0.8 (1.2 S_h+S_A) but the circumferential stress range is at least 1.5 times the axial stress range.

2) Where break locations are selected without the benefit of stress calculations, breaks are 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 determine by detailed stress analyses (e.g., finite element analyses) or tests on a pipe fitting.

3) Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by analysis.

4) The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically 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 is taken into account, as applicable, in the reduction of jet discharge.

5) Pipe whipping is assumed to occur in the plant defined by the piping geometry and configuration, and to cause pipe movement in the direction of jet reaction.

b) Longitudinal Pipe Breaks - The following longitudinal breaks are postulated in high-energy fluid system piping:

1) Longitudinal breaks in fluid system piping and branch runs are postulated in nominal pipe size four in. and larger, except where the maximum stress range exceeds 0.8 (1.2 S_h+S_A) but the axial stress range is at least 1.5 times the circumferential stress range.

2) Longitudinal breaks are not 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 item (1) above apply).

   b) At intermediate locations where the criterion for a minimum number of break locations must be satisfied.

3) Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are oriented (but do not occur 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).

4) The dynamic force of the fluid jet discharge is based on a circular or elliptical 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 are taken into account, as applicable, in the reduction of jet discharge.

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

c) Through Wall Leakage Cracks - The following through wall leakage cracks are postulated in moderate energy fluid system piping:

1) Cracks are postulated in moderate energy fluid system piping and branch runs exceeding a nominal pipe size of one inch.

2) Fluid flow from a crack is 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.

$$\left[\left(\frac{1}{2}\right)D \times \left(\frac{1}{2}\right)WT\right]$$

3) The flow from the crack is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects are determined on the basis of a conservatively estimated time period required to effect corrective actions.

3.6.2.1.6 Containment Penetration Piping

The details of the containment penetrations, identifying all process pipe welds, access for inservice inspection of welds, points of fixity, and points of geometric discontinuity are discussed in Section 3.8.1.1.3.3 and detailed on Figures 3.8.1-16 and 3.8.1-17.

3.6.2.1.7 Diesel Generator Room Piping

Moderate energy systems installed in the diesel generator room are: Emergency Air, Fire Protection, Fuel Oil, Lube Oil, Miscellaneous Drains, Service Water, Potable Water and Station Air.

There are no high energy lines in the Diesel Generator Building. The systems listed above were eliminated from high energy lines since they are not utilized during normal plant operation or on the basis of their size (i.e. less than one inch).

The large diesel exhaust lines are open to the atmosphere, and the available energy in these lines is negligible. As a result, a failure in one of these lines, or any of the smaller sized lines, would not have adverse consequences nor be capable of breaching the twenty-four inch reinforced concrete wall separating the diesel generator rooms.

Moderate energy fluid system cracks were postulated in the Diesel Generator Building. It has been demonstrated, by calculation, that internal flooding from the moderate energy fluid system failure (crack) will not adversely affect the diesel generator.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Model

3.6.2.2.1 RCS Main 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 RCS. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flowrates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, and calculates the time-history of forces at appropriate locations, (e.g., elbows) in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the RCS. Key parameters calculated by the hydraulic model are pressure, mass flowrate, and density. These are supplied to the thrust calculation, together with plant layout information, to determine the time-dependent loads exerted by the fluid of 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.

MULTIFLEX (Reference 3.9.1-3) is a computer program for analyzing thermal-hydraulic structure system dynamics. It includes mechanical structure models and their interactions with the thermal-hydraulic system. The thermal hydraulic portion of MULTIFLEX is based on the 1-dimensional homogeneous flow model which is expressed as a set of mass, momentum and energy conservation equations. In MULTIFLEX, the structural walls surrounding a hydraulic path may deviate from their neutral positions depending on the force differential on the wall.

MULTIFLEX computes the pressure response of a system during a decompression transient. The transient pressure response can then be used to evaluate the system's overall dynamic structural response. The pressure distributions computed by MULTIFLEX can be used to evaluate the reactor core assembly and other primary coolant loop component supports integrity.

The analysis is performed for the subcooled decompression period of the transient where hydraulic loads are greatest. These loads are used in the structrual evaluation of the reactor pressure vessel support system, in conjunction with other loads associated with a LOCA and with a safe shutdown earthquake (SSE).

The THRUST (STHRUST in Reference 3.9.1-1) computer program was developed to compute the transient (blowdown) hydraulic loads resulting from a LOCA.

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

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

The symbols and units are:

F	=	Force (lb_f)
A	=	Aperture area (ft.^{2})
P	=	System pressure (psia)
$\dot{m}$	=	Mass flowrate, (lbm/sec.)
ρ	=	Density, (lbm/ft.^{3})
g	=	Gravitational constant (32.174 ft.-lb_m/lb_f - sec.^{2})
A_m	=	Mass flow area (ft.^{2})

In the model to compute forcing functions, the Reactor Coolant System is represented by a similar model as employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each 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.

Calculation of Jet Thrust Force Resulting from LOOP Pipe Ruptures

When fluid is discharged from a ruptured pipe, it will exert a reaction force (thrust load) on the piping system. The magnitude of this thrust load is a function of the pressure and temperature of the fluid, the break flow area, piping geometry, and other factors.

For steady state conditions the thrust load, F_T, can be represented as:

$$F_T = (P_e - P_a) A_e + \rho_e \frac{V_e^2}{g_c} A_e \quad (1)$$

where:

A_e	=	area at exit plane
g_c	=	Newton's conversion constant
P_a	=	ambient pressure
P_e	=	pressure at exit plane

V_e = fluid velocity at exit plane

ρ_e = fluid density at exit plane

Steady state conditions are assumed to be reached in one milli-second (.001 seconds) following the rupture.

Equation (1) can be rewritten in terms of the jet thrust coefficient, C_T, as:

$$F_T = C_T P_O A_e$$

where:

$$C_T = \frac{P_e - P_a}{P_O} + \frac{\rho_e V_e^2}{g_c P_o} \qquad (2)$$

and:

P_o = initial total (stagnation) pressure in the pipe.

For a system with negligible frictional and form losses, CT is a function only of the stagnation pressure and enthalpy of the fluid in the reservoir. For the pipe breaks considered in the RCL, the frictional and form loss effects on CT are negligible.

The fluid in the reactor coolant loop will flash when discharged to ambient conditions. The water in these lines is below saturation temperature (that is, subcooled) and its discharge through the break opening may occur under nonequilibrium thermodynamic conditions; that is, its flashing, as it flows through the break opening, may be somewhat suppressed. For the discharging of subcooled flashing water, the jet thrust coefficient is determined from ANS 58.2 (ANSI N176), Appendix B, which was developed, in part, using the Henry Fauske model for subcooled water blowdown.

Although equation (2) is for steady-state conditions, it may be employed to compute the jet force for time-varying reservoir conditions and for break opening areas by varying the appropriate terms with time. This is conservative because, for a system with negligible friction, the thrust load is at a maximum during steady state.

3.6.2.2.2 RCS Pressurizer Surge Line, Main Steam Supply, Feedwater and Other High Energy Systems

Analysis on the Feedwater piping outside containment was performed with the use of computer program PIPERUP described in Reference 3.6.2-8.

The methods to describe forcing functions and response models for these systems are described in Reference 3.6.2-4 (ETR-1002).

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

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

A loss of reactor coolant accident is assumed to occur for a branch line break down to the restraint of the second normally open automatic isolation valve (Case II on Figure 3.6.1-1) on outgoing lines and down to and including the second check valve (Case III on Figure 3.6.1-1) on incoming lines normally with flow. It is assumed that motion of the unsupported line containing the isolation valves could cause failure of the operator of both valves to function. A pipe break beyond the restraint or second check valve does not result in an uncontrolled loss of reactor coolant if either of the two valves in the line close.

Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint does not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV on Figure 3.6.1-1) a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the Reactor Coolant System are defined as "large" for the purpose of this criteria when the inside diameter is greater than four in. up to the largest connecting line, the 14 in. pressurizer surge line. Rupture of these lines results in a rapid blowdown from the Reactor Coolant System and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the Reactor Coolant System are defined as "small" if they have an inside diameter equal to or less than four in. 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 sq. in. corresponding to four in. inside diameter piping.

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a loss of reactor coolant or steam or feedwater line break accident to ensure that the public is protected in accordance with 10 CFR 50.67 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 the reactor coolant system main 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:

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

b) The containment leak tightness is not decreased below the design value, if the break leads to a loss of reactor coolant; (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) and

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

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

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

3.6.2.3.1.1 Reactor Coolant Loop Piping and Large Branch Lines

Propagation of damage resulting from rupture of large branch lines connected to the main reactor coolant loop is permitted to occur but does not exceed the design basis for containment and subcompartment pressures, loop hydraulic forces, reactor internals reaction loads, primary equipment support loads, or emergency core cooling system performance. Rupture of the main reactor coolant loop is excluded for dynamic effects since the application of leak-before-break (LBB). Refer to Section 3.6.2.1.1.1.

Large branch line piping must be restrained to meet the following criteria in addition to items a) through c) in Section 3.6.2.3.1 for a pipe break resulting in a loss of reactor coolant:

a) Propagation of the break to the unaffected loops must be prevented to ensure the delivery capacity of the accumulators and residual heat removal pumps.

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

3.6.2.3.1.2 Small Branch Lines

In the unlikely event that one of the small pressurized lines should fail and result in a loss-of-coolant accident, the piping is restrained and/or arranged to meet the following criteria in addition to items a) through c) of Section 3.6.2.3.1.

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

b) Propagation of the break in the affected leg is permitted but is limited to a total break area of 12.5 sq. in. (four in. 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) Pipe whip of a small branch line causing a break in the high head safety injection line connected to the affected leg is prevented if the high head safety injection line break would result in a loss of core cooling capability.

3.6.2.3.2 High Energy Piping Dynamic Analysis

Pipe breaks are postulated in high energy piping in accordance with the criteria in Section 3.6.2.1. The analyses for determining the dynamic effects of pipe break are performed using the techniques discussed in Section 3.6.2.3. Descriptions of the applicable measures to protect against pipe whip and blowdown jet impingement forces are given in Section 3.6.1.2.

3.6.2.3.3 Pipe Whip Restraint Design Criteria

a) Design Bases - The pipe break locations and orientation are determined in accordance with Section 3.6.2.1. For each postulated pipe break, the possible effects of the break are investigated and, if necessary (per Section 3.6.1.2), restraints are provided to prevent pipe whip.

b) Functional Requirements - High energy pipe whip restraints are designed to ensure that the pipe whip will be eliminated or minimized. All restraints are designed to permit the predicted thermal and seismic movements of the pipes.

c) Design Parameters - After the pipe restraint locations are identified, the following design parameters are determined:

    1) Jet thrust force

    2) Pipe seismic displacements

    3) Pipe thermal displacements

    4) Pipe insulation thickness

    5) Maximum allowable pipe travel

The jet thrust force and maximum allowable pipe travel are used in the analysis process.

Insulation, and seismic and thermal movements are used in determining the minimum gap between the restraint and pipe surfaces.

Typical sketches of the various pipe whip restraints to be used are shown on Figures 3.6.2-2 through 3.6.2-4.

The design loading combination considered for the design of pipe whip restraints, which are within ASME Code allowables is as follows:

Load Combination	Dead Weight of Restraint + Pipe Rupture Load + SSE

3.6.2.3.4 Jet Impingement

3.6.2.3.4.1 RCS Main Loop Piping

The basis for eliminating jet impingement from consideration in the RCS Main Loop Piping is provided in 3.6.2.1.1.1.

3.6.2.3.4.2 Piping Other Than RCS Main Loop Piping

The geometry of the jet stream, its pressure distribution and the temperature distribution, depend on the properties of the discharged fluid, the surrounding medium, and the fluid conditions at the exit plant, i.e., choked or unchoked flow.

Two types of breaks and three kinds of jet development are considered. For jets emerging out of breaks where the flow is unchoked, (typically non-flashing water) the theory of free submerged jets applies (Reference 3.6.2-5). For choked flow at the exit plane (typically flashing water and/or wet steam) the jet streams are based on work by F.J. Moody (Reference 3.6.2-6).

a) Guillotine break jet development: This break is perpendicular to the pipe axis with complete severance and lateral separation of at least one pipe diameter between the two ends. This results in the development of two free and clear jets whose shape is dependent on the fluid phase (see Figures 3.6.2-5 and 3.6.2-6).

b) Guillotine break with limited separation jet development: This special case of a jet development is due to the relative position of the two pipe ends. For this kind of jet development, dynamic analysis must prove that with respect to each other, the pipe ends remain within the following bounds:

axial separation	≤ 0.5 inside diameter
lateral separation	≤ 0.5 inside diameter
area of break	≤ twice the pipe flow cross-sectional area

The jet which develops from this case is shown on Figure 3.6.2-7. This type of jet will have a lower impingement pressure than the jets described in items a) and c) at equal distances from the break plane because the jet area perpendicular to the pipe centerline is much larger.

c) Longitudinal break jet development: This break is an axial split of circular or elliptical shape whose break area is equivalent to the effective cross sectional flow area of the pipe at the break location. The resultant jet development is dependent on the fluid phase (see Figures 3.6.2-5 and 3.6.2-6).

Only two non-concurrent jets are assumed at longitudinal break locations, both for analyzing jet effects and for the design of pipe whip restraints. These jets will be diametrically opposed and cause the worst out of plane bending.

The jet impingement force on a target is then calculated from:

$$F = \frac{F_j}{A_j} \; A_x G \; (DLF)$$

where:

DLF = 2 = dynamic load factor

A_x = impacted area of the target (sq.ft.)

G = geometric shape factor

A_j =cross sectional area of jet at the target distance from break plane (sq. ft.)

F_j = total jet impingement force at the break plane (lbf.)

The jet stream for wet steam and flashing water will be divided into three regions, as indicated on Figure 3.6.2-5. In region 1, the jet opens up with a half angle of 45 degrees for a distance of five D, where D is the inside diameter of the broken pipe. Region 2 extends uniformly from five D to approximately 25.5 D, at which point region 3 begins as a cone whose apex at the break plane has a half-angle of 10 degrees. Region 3 extends from 25.5 D to the end of the jet.

For region 1:

F_j = KPA where: K = 2.0
P = operating fluid pressure
A = cross-sectional flow area

For regions 2 & 3:

$F_j = KP_{sat}A$ where: K = 1.26
Psat = saturation pressure at operation fluid temperature
A = cross-sectional flow area

The jet streams developed by dry steam and nonflashing water are indicated on Figure 3.6.2-6. For these two types of fluids:

F_j = KPA where: K = 1.26 for dry steam or 2.0 for nonflashing water
P = operating pressure
A = cross-sectional flow area

Fluid	Apex Angle
Dry Steam	10 degrees
Nonflashing Water	10 degrees

For guillotine breaks which have limited separation (see Figure 3.6.2-8):

Steam of Quality (EB) 99 Percent (For all regions, from R = 0 to R = ∞)

$F_j = KP_oA$	where:	K	= 1.26
		P_o	= operating pressure
		A	= break area

Flashing Fluids and Steam-Water Mixtures (Quality <99 percent)

In the initial region, from $R = 0$ to $R = R^{jet}$ asymptotic

$F_j = KP_oA$	where:	K	= 2.0
		P_o	= operating pressure
		A	= break area

In the subsequent region, from $R = R^{jet}$ asymptotic to $R = \infty$

$F_j = KP_{sat}A$	where:	K	= 1.26
		P_{sat}	=saturation pressure at operating fluid temperature
		A	= break area

These jet streams are then analyzed to determine what components are being hit. Those pieces of equipment which are essential for the safety of the plant are then either qualified with the jet loading or are protected from the jet impingement forces by the method described in Section 3.6.1.2

The design loading combinations and allowable stress limits for essential components which fall under the ASME Code are as follows:

Load Combination	Plant Operating Condition	Allowable Stress
Normal Operating + SSE + Jet Impingement	Faulted	$\leq 3\ S_m$ (Safety Class 1) $\leq 2.4\ S_h$ (Safety Classes 2 & 3)

where: S_m = design stress intensity (NB-3600 of ASME III)

S_h = allowable stress at maximum (hot) temperature (NC-3600 of ASME III)

All other essential components (i.e., conduit, junction boxes, instrumentation cabinets) with jet impingement loads are evaluated on a case by case basis. For results of the jet impingement analysis, see Appendix 3.6A.

3.6.2.4 Guard Pipe Assembly Design Criteria

There are no guard pipes used between containment isolation valves because of the single barrier containment design.

3.6.2.5 Materials to Be Submitted at the Operating License Review

3.6.2.5.1 Nuclear Steam Supply System Vendor Scope for Reactor Coolant Pipe

a) Table 3.6.2-1 identifies the design basis branch line break locations and orientations for the main reactor coolant loop.

There is negligible impact on the RCS Main Loop Piping due to the RSG/Uprating Program and, therefore, there is no change in the existing LBB Analysis (Reference 3.6.2-12).

b) Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Section 3.9.1.4. 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.

3.6.2.5.2 Other Piping

The results of the analyses performed on piping systems are contained in Appendix 3.6A.

3.6.2.5.3 Energy Dissipating Crushable Material

The allowable energy dissipating capacity of one crushable material used is based on the results of typical static tests of the material. The allowable design energy capacity is based on a maximum permissible crushing of the material to 50 percent of its original thickness. By specifying material such that its maximum thickness after being completely crushed to its maximum energy absorbing capacity does not exceed 35 percent of its original thickness, the allowable capacity is limited to a maximum of 77 percent of the energy dissipating capacity determined by the typical static test results.

NOTE: During the steam generator replacement outage (RF10) a new whip restraint was installed for the rerouted feedwater piping inside the steam generator subcompartments. The new whip restraint design utilized stainless steel crushable material with a crushing strength in the designated direction as shown in rupture restraint structure drawing 2168-G-0236 S06 or 6500 psi at 400°F with a permissible variation of ± 10%.

REFERENCES: SECTION 3.6

3.6.2-1 "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," - WCAP-8082-P-A, January 1975 (Proprietary Version); WCAP 8172 A, January 1975 (Non-Proprietary Version).

3.6.2-2 NRC letter from Olan D. Parr, Chief-Light Water Reactors Branch No. 3, Division of Project Management to Mr. J. A. Jones, Executive Vice President - Engineering Construction and Operations, Carolina Power and Light Company, dated April 10, 1978 - "Design of the Pipe Tunnel for Main Steam Line and Feedwater Line in the Shearon Harris Plant."

3.6.2-3 NRC meeting minutes dated July 24, 1978 - "Summary of Meeting Held on June 15, 1978, to Discuss the Design of the Pipe Tunnel for Main Steam and Feedwater Lines."

3.6.2-4 "Design Considerations for the Protection from the Effects of Pipe Rupture," - ETR-1002-P (Proprietary version) and ETR-1002 (Non-proprietary version), by Ebasco Services, Inc. November, 1975.

3.6.2-5 "The Theory of Turbulent Jets" - N.G. Abramovich, MIT Press, 1963.

3.6.2-6 "Prediction of Blowdown Thrust and Jet Forces" - F.J. Moody, ASME pages 69-HT-31.

3.6.2-7 "ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, Summer 1973 Addenda.

3.6.2-8 Letter from T. M. Novak of NRC to E. E. Utley of CP&L, dated August 15, 1985, regarding Elimination of Arbitrary Intermediate Breaks.

3.6.2-9 "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Shearon Harris Unit 1" - WCAP 10699, September 1984 (Proprietary Version); WCAP 10700, September 1984 (Nonproprietary Version).

3.6.2-10 Letter from Mr. George W. Knighton of NRC to E. E. Utley of CP&L dated June 5, 1985, regarding Request for Exemption from a Portion of General Design Criterion 4 of Appendix A to 10 CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as a Structural Design Basis for Shearon Harris Nuclear Power Plant, Unit 1.

3.6.2-11 "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for The Shearon Harris Unit 1 Nuclear Plant" - WCAP-14549, December 1996 (Proprietary Version); WCAP-14550, December 1996.

3.6.2-12 WCAP-14778, Revision 1, "Shearon Harris Nuclear Plant SG Replacement Uprating Engineering Report."

3.6.2-13 WCAP-14549-P, Addendum 1, "Technical Justification for Eliminating Primary Loop Rupture as the Structural Design Basis for Harris Nuclear Plant for the License Renewal Program," Revision 0, January 2005.

3.6.2-14 Generic Letter 87-11, “Relaxation in Arbitrary Intermediate Pipe Rupture Requirements,” June 1987.

APPENDIX 3.6A PIPE RUPTURE REPORT

3.6A PIPE RUPTURE ANALYSIS

Section 3.6 describes the design bases and measures that are taken on SHNPP to demonstrate that the systems, components, and structures required for safe shutdown and maintaining the reactor in a cold shutdown condition, are adequately protected against the dynamic effects associated with pipe rupture. This appendix presents the results of the pipe rupture analysis. Section 3.6A.1 discusses the results of high energy pipe break evaluation inside Containment. High energy pipe breaks outside Containment are discussed in Section 3.6A.2. Subcompartment Pressure Analyses both inside and outside Containment are addressed in Section 3.6A.3. Moderate Energy Piping System analyses are described in Sections 3.6A.4 and 3.6A.5. The Flooding Analysis is presented in Section 3.6A.6.

3.6A.1 HIGH ENERGY PIPE BREAK INSIDE CONTAINMENT

The high energy piping systems or portions of systems which are considered for pipe rupture analysis inside containment are:

1. Steam Generator Blowdown System

2. Reactor Coolant Drain Lines

3. Safety Injection System (from the reactor coolant loop up to the first normally closed valve)

4. Chemical and Volume Control System (Letdown, Charging, and Seal Injection)

5. Reactor Coolant Loop

6. Pressurizer Safety and Relief

7. Pressurizer Spray and Auxiliary Spray

8. Pressurizer Surge

9. Main Steam and Feedwater

10. Residual Heat Removal System (portion only)

11. Auxiliary Feedwater

The criteria used to locate the postulated break points for high energy piping systems is described in Section 3.6.2. The various protective methods used to mitigate the consequences of the postulated pipe break are given in Section 3.6.1.2.

3.6A.1.1 Pipe Whip Analysis

This section describes the method of protection used against pipe whip for each pipe break in the systems listed in Section 3.6A.1. The methodology used to evaluate the design loads for pipe whip restraints is also included.

3.6A.1.1.1 Steam Generator Blowdown System

3.6A.1.1.1.1 General description

The Steam Generator Blowdown System (SGBS) operation and design bases are described in Section 10.4.8. The SGBS lines are 4-inch nominal size Schedule 40, 3-inch nominal size Schedule 40 and a short segment of 2-inch Schedule 80 carbon steel or low alloy steel pipe. The portion of the SGBS from the steam generators up to and including the isolation valves outside the Containment comprises an extension of the steam generator boundary. This portion of the system is designed in accordance with the ASME Code, Section III, Class 2 and Seismic Category I requirements. Pneumatically operated isolation valves are also located inside Containment. The high energy portion of the SGBS inside Containment extends from the steam generator to the penetration.

3.6A.1.1.1.2 Pipe whip analysis

Postulated break locations and pipe whip restraints are indicated in Figure 3.6A-24. All breaks in the SGBS are a double ended guillotine (circumferential) type. Terminal end breaks are postulated at the steam generator nozzles and at the penetrations.

The pipe whip restraints provided for the system are identified in Table 3.6A-4. The restraints are designed to permit the predicted thermal and seismic movements of the piping. Typical sketches of the various types of pipe whip restraints are shown on Figures 3.6.2-2 through 3.6.2-4. The pipe whip restraints are designed to withstand pipe rupture thrust load which includes a dynamic load factor appropriate for the gap between the pipe and the restraint.

For breaks where pipe whip protection is not provided by means of restraints, a detailed study was conducted to evaluate the effects of the whipping pipe on essential systems and components; it was observed that the whipping pipe would not compromise the function of any essential systems.

3.6A.1.1.2 Reactor coolant drain lines

3.6A.1.1.2.1 General description

The reactor coolant loop drain lines are connectable to the Waste Processing System. The drain connections are shown on flow diagram, Figures 5.1.2-1 and 5.1.2-2. The drain lines are 2-inch nominal size, Schedule 160, stainless steel piping. That portion of piping from the reactor coolant loop to the second normally closed valve comprises an extension of the reactor coolant pressure boundary. This portion of the system is designed in accordance with the ASME Code Section III, Class 1 and Seismic Category I requirements. The remainder of the piping system (Waste Processing) is non-seismic.

3.6A.1.1.2.2 Pipe Whip Analysis

Postulated break locations for reactor coolant loop drain lines are indicated in Figure 3.6A-14.

The short length of pipe will not whip against any safety related equipment or components except the reactor coolant loop piping. Thus, not providing pipe whip restraints for these lines is justified.

3.6A.1.1.3 Safety Injection System & Accumulator Discharge Lines

3.6A.1.1.3.1 General Description

The Emergency Core Cooling System (ECCS) or the Safety Injection System (SIS) operation and design bases are described in Section 6.3. The flow diagram for the Safety Injection System is shown on Figures 6.3.2-1 through 6.3.2-3 and Figure 5.1.2-1.

The cold leg injection piping from the accumulators is classified as high energy piping. The SIS Hi head and Low head injection piping between the RC Loop and the first check valve near the RC Loop is classified as high energy piping. The remainder of the SIS piping inside Containment is classified as moderate energy piping by the two percent rule explained in Section 3.6.1.

3.6A.1.1.3.2 Piping Whip Analysis

The location of the SIS postulated break points and pipe whip restraint locations are provided on Figures 3.6A-20 and 3.6A-22. The pipe whip restraints provided for the system are identified in Table 3.6A-8. The restraints are designed to ensure that the pipe whip will be minimized. The type of the restraint, coordinate directions in which the restraints are capable of supporting the pipe whip load, and the identification of breaks which activate each restraint are given in Table 3.6A-8.

3.6A.1.1.4 Chemical and Volume Control System

3.6A.1.1.4.1 General Description

The Chemical and Volume Control System (CVCS) operation and design bases are described in Section 9.3.4. The flow diagram for the CVCS is shown on Figures 9.3.4-1 through 9.3.4-5.

The letdown piping inside Containment from the Reactor Coolant Loop to the containment penetration is three-inch nominal size with schedule 160 and 40S, and two-inch nominal size with schedule 40S. The portion of piping from the Reactor Coolant Loop up to the second isolation valve, located upstream of the regenerative heat exchanger, is designed in accordance with the ASME Code, Section III, Class 1 and Seismic Category I Criteria. The remainder of the line to the containment penetration is ASME Code, Section III, Class 2, Seismic Category I pipe. The letdown line from the Reactor Coolant Loop to the letdown heat exchanger downstream piping is classified as high energy.

The charging lines inside Containment are Schedule 160, three-inch nominal size. The two charging lines are ASME Code, Section III, Class 1 from the Reactor Coolant Loop to the second isolation valve in each line. The remainder of the charging line is Class 2. The charging system piping inside Containment is designed to seismic Category I criteria and is all considered high energy.

The seal injection piping inside Containment from penetration to reactor coolant pump 1 1/2 inch nominal size Schedule 160 stainless steel piping. The portion of the piping from reactor coolant pump up to the second check valve is Safety Class 1/Seismic Category I. The remainder of the line to the containment penetration is Safety Class 2/Seismic Category I.

3.6A.1.1.4.2 Pipe Whip Analysis

Figures of the letdown lines inside Containment indicating the locations of postulated break points and pipe whip restraints are provided in Figures 3.6A-9 through 3.6A-11. All breaks in the letdown line are double ended guillotine (circumferential) type only. Breaks are postulated at the reactor coolant loop nozzle and inlet and outlet nozzles of the Regenerative Heat Exchanger and the penetration. The pipe whip restraints provided for the letdown piping inside Containment are identified in Table 3.6A-5.

Figures of the charging lines inside Containment indicating the locations of postulated break points and pipe whip restraints are provided in Figures 3.6A-9 through 3.6A-11. All breaks in the charging lines are double ended guillotine (circumferential) type only. Breaks are postulated at the reactor coolant loop nozzles, regenerative heat exchanger nozzles, and the penetration. Intermediate breaks are postulated based on stress intensity or usage factor. The pipe whip restraints provided for the charging lines inside Containment are identified in Table 3.6A-5.

Figures of the seal injection lines inside Containment indicating the locations of break points are provided in Figures 3.6A-9 through 3.6A-11. All breaks in the seal injection lines are circumferential type only. Breaks are postulated at the reactor coolant pump nozzles and the penetrations.

3.6A.1.1.5 Reactor Coolant System

3.6A.1.1.5.1 General Description

The Reactor Coolant System (RCS) operation and design bases are described in Section 5.1. The flow diagram of the system is shown on Figure 5.1.1-1, 5.1.1-2, and 5.1.1-3. The reactor coolant loop piping is stainless steel piping designed and fabricated in accordance with the ASME Code, Section III, Class I Criteria.

3.6A.1.1.5.2 Pipe Whip Analysis

No break points at the RCS loops are considered for pipe whip analysis. The break locations described in Section 3.6.2 are considered for containment environmental analysis. Pipe stops provided in the system are described in the Reference 1 of Section 3.6. The results of containment and subcompartment pressure analyses are provided in Section 6.2.1. The systems and equipment necessary to mitigate the consequences of an RCS break (LOCA) are described in Section 3.6.1.

3.6A.1.1.6 Pressurizer Safety and Relief Piping

3.6A.1.1.6.1 General Description

The pressurizer safety and relief valve discharge lines are three- and six-inch nominal size schedule 160 stainless steel pipe. The portion of the piping between the pressurizer nozzle and the safety and relief valve is kept pressurized during normal plant operating condition. This piping is classified as high energy and designed in accordance with ASME Code, Section III, Class 1 and Seismic Category 1 criteria. The piping downstream of the valve is non-seismic. The pressurizer safety and relief system operation and design bases are described in Section 5.2.2.

3.6A.1.1.6.2 Pipe Whip Analysis

Postulated pipe break locations for pressurizer safety and relief lines are provided on Figure 3.6A.15. A summary of a pipe break calculation is provided on Figure 3.6A.15 (3 sheets). Guillotine (circumferential) as well as slot breaks have been postulated for 4-inch nominal pipe size and larger piping. No pipe whip restraints are needed for the piping since high energy portions of these piping systems are completely enclosed within pressurizer cubicle and no essential components are impacted by pipe whip.

Piping downstream of safety and relief valves are not classified as high energy, and thus, breaks are not postulated in this portion of the piping system.

3.6A.1.1.7 Pressurizer Spray and Auxiliary Spray

3.6A.1.1.7.1 General Description

The pressurizer spray system operation and design bases are described in Section 5.4.10. The spray lines from the Reactor Coolant Loop to the pressurizer nozzle are four-inch and six-inch nominal size schedule 160, stainless steel piping. The Auxiliary Spray line connected to the spray line is two-inch nominal size schedule 160-stainless steel piping. The entire spray piping and a portion of the Auxiliary Spray piping from the main spray line to the second check valve are designed in accordance with the ASME Code, Section III, Class 1 and Seismic Category I requirements. The remainder of the auxiliary spray line is designed per ASME Code, Section III, Class 2 and Seismic Category I requirements.

3.6A.1.1.7.2 Pipe Whip Analysis

Guillotine (circumferential) as well as slot breaks are postulated for the 4-inch and 6-inch nominal size piping. A summary of pipe break calculations and pipe whip restraint locations are provided on Figures 3.6A-14A and 3.6A-15. Table 3.6A-7 identifies the pipe whip restraints provided for the spray lines. For breaks where pipe whip protection is not provided by means of restraints, a detailed study was conducted to evaluate the effects of the whipping pipe on essential systems and components. It has been determined that the whipping pipe would not compromise the function of any essential components and, thus, safe plant shutdown capability is maintained.

3.6A.1.1.8 Pressurizer Surge

3.6A.1.1.8.1 General Description

The pressurizer surge system operation and design bases are described in Section 5.4.10. The surge line is a 14-inch nominal size Schedule 160 stainless steel pipe. The surge line is designed in accordance with ASME Code, Section III, Class 1 and seismic Category I requirements.

3.6A.1.1.8.2 Pipe Whip Analysis

Postulated break points and pipe whip restraint locations of the pressurizer surge line is provided on Figure 3.6A-23. Pipe breaks are postulated at terminal ends and at four intermediate points. Rupture restraints are located in the piping to prevent adverse pipe whip effects on essential systems and components. Table 3.6A-6 lists all rupture restraints provided in the system.

3.6A.1.1.9 Main Steam and Feedwater

3.6A.1.1.9.1 General Description

The piping systems considered for pipe rupture analysis are three main steam and three feedwater lines. Both the main steam and feedwater lines are carbon steel ASME SA-106 GRC piping, designed in accordance with ASME Code, Section III, Class 2 and seismic Category I Criteria. Feedwater lines inside SG subcompartments were replaced with chrome-moly ASME SA-335, GR. PII piping when the SGs were replaced.

The main steam lines are 32 inches in size with 29.625 inch minimum I.D. The design bases and operation of the main steam lines are described in Section 10.3.1.

The feedwater lines are 16 inches in size with between 14.5 inch and 13.56 inch I.D. Their design bases and operation are described in Section 10.4.7.

3.6A.1.1.9.2 Pipe Whip Analysis

Selection of break locations in the main steam and feedwater piping was accomplished in accordance with the criteria presented in Section 3.6.2. Resulting break locations for the main steam and feedwater piping inside Containment are shown on Figures 3.6A-1 through 3.6A-7. For the main steam and feedwater piping where a break is postulated to occur at an elbow tangency point, the breaks are postulated at both elbow tangency points (due to the uncertainty in predicting where on the elbow the break would occur). The rationales for break selection along with break locations are shown in Tables 3.6A-19 through 3.6A-21. Only circumferential breaks were postulated at these locations.

Pipe whip restraints provided for the main steam and feedwater piping are shown on Figures 3.6A-1 through 3.6A-7. Restraint systems are selected to prevent unacceptable pipe whip resulting from the identified break locations. The directions in which the restraints are designed to support the pipe rupture load are given in Tables 3.6A-1 and 3.6A-2. Adequacy of the restraint systems to prevent pipe whips was demonstrated using computer programs RELAP and PLAST. Analysis was performed for loop-1 and loop-2 piping systems only. Loop-3 is considered identical but opposite hand to loop-1.

High energy pipe breaks were postulated at the FW nozzle. The results of the full FW circumferential break impacted the supcompartment pressure and temperature analyses which resulted in higher ΔP across the bio-wall than allowed in Tables 6.2.1-2 and 6.2.1-3. Therefore, the pipe break axial opening was restricted to 2.5" by the installation of a pipe whip restraint, and the resultant jet was reduced to a disc type rather than a full circumferential break type. This whip restraint limits the pipe motion and therefore, no full circumferential breaks (forward or reverse) had to be postulated for the FW system with the new RSGs.

3.6A.1.1.10 Residual Heat Removal System (RHRS)

3.6A.1.1.10.1 General Description

The RHRS operation and design bases are described in Section 5.4.7. The flow diagram for the RHR system is shown on Figures 5.4.7-1 and 5.4.7-2. The RHR discharge piping between the RC Loop and the first check valve near the RC Loop is classified as high energy piping. This portion of piping is common to safety injection and is described in Section 3.6A.1.1.3. The RHR suction piping between the RC loop and the first normally closed motor operated valve near the RC loop is classified as high energy piping. The remainder of the piping system is considered moderate energy piping by the two percent rule explained in Section 3.6.1.

3.6A.1.1.10.2 Pipe Whip Analysis

The location of the RHR lines postulated break points and pipe whip restraint locations are provided on Figures 3.6A-21 and 3.6A-22. The piping whip restraints provided for the system, the type of the restraints, the direction of the restraints, and corresponding breaks are identified in Table 3.6A-9.

3.6A.1.1.11 Auxiliary Feedwater System

3.6A.1.1.11.1 General Description

Auxiliary feedwater system (AFWS) operation and design bases are described in Section 10.4.9. The high energy portion of the AFWS inside Containment extends from the penetration to the steam generator. The AFWS lines inside Containment are six-inch nominal size Schedule 80, carbon steel piping. Schedule 120 and 160 piping is used at the steam generator AFW nozzles.

3.6A.1.1.11.2 Pipe Whip Analysis

Break locations and pipe whip restraints for the auxiliary feedwater lines are provided on Figure 3.6A-8. All breaks in the AFWS are double-ended guillotine (circumferential) type. Terminal end breaks are postulated at the steam generator nozzles and at the penetrations. Pipe whip restraints provided for the system are listed in Table 3.6A-3.

3.6A.1.2 Jet Impingement Analysis Inside Containment

The essential components and systems located inside the Containment required for safe shutdown of the plant are evaluated for the effects of jet impingement. The jet impingement analysis procedure and the results of the analysis are presented here.

The jet envelopes were drawn on the system piping drawings for all high energy pipe breaks inside Containment (identified for the various piping systems in Section 3.6A.1). The shape of the jets is dependent on the fluid phase. The various jet shapes used for the analysis are described in Section 3.6.2 The jet envelope drawings along with the components system layout drawings for Electrical, Instrumentation and Control, HVAC and Mechanical Systems are used to identify all component-jet interactions for each high energy pipe break. Component-jet interactions were judged acceptable or unacceptable according to the plant shutdown logic, single active failure criteria, and environmental effects. The interactions identified as unacceptable were either moved out of the jet completely or were further analyzed for operability of the components under the jet impingement loading. The procedure for calculating jet impingement forces on a target is given in Section 3.6.2.3.4. If the equipment/component cannot be qualified with the jet loading, they are protected from the jet impingement forces by the methods described in Section 3.6.1.2

3.6A.1.2.1 HVAC jet interactions

Jet impingement interactions with safety related HVAC components located inside the reactor containment building were found to be acceptable, since loss of function of these components for specific high energy pipe failures does not jeopardize the safe shutdown of the facility.

3.6A.1.2.2 Electrical jet interactions

An analysis was conducted to determine if lost power to safety-related mechanical and instrumentation and control equipment would effect safe shutdown of the facility. In general, jet impingement interactions with safety related electrical cables or conduits located inside the reactor containment building were found to be acceptable.

In those instances where unacceptable interactions were identified, protection was provided or the cables/conduits were rerouted.

3.6A.1.2.3 Structural jet interactions

Structural steel components identified as jet targets and supporting safety related components were evaluated. It was determined that the structure was either capable of sustaining the jet loads or that failure of the structure did not jeopardize the function of essential components. Certain other structures (such as staircases, platforms, etc.) are assumed to be distorted only by the jet and that any distortion of these structures would not affect essential components or create missiles. All concrete structures intercepting jet envelopes are designed for the resulting jet impingement loads.

3.6A.1.2.4 Instrumentation and control jet interactions

An analysis was conducted to determine if loss of instrumentation and control components would affect safe shutdown of the facility. In general, jet impingement interactions with safety related instrumentation and controls inside the reactor containment building were found to be acceptable. In those instances where unacceptable interactions were identified, protection has been provided or instrumentation has been relocated out of the jet envelope.

3.6A.1.2.5 Mechanical systems

All mechanical component-jet impingement interactions were found to be acceptable.

Pipe hangers and supports are generally overdesigned and subsequently assumed to be functional under jet impingement load on piping. It is also assumed that local failure of a support would not jeopardize the function of the system.

3.6A.1.3 Environmental Effects of High Energy Breaks Inside Containment

As discussed in Section 6.2.1.1.3, the environmental conditions (pressure, temperature, humidity, and radiation) inside the Containment are the most severe after a design basis accident (DBA). The DBAs include LOCA and Main Steam Line Break (MSLB). All safety-related mechanical and electrical equipment located inside Containment is capable of functioning under the environmental conditions resulting from the design basis accident.

Furthermore, as discussed in Section 3.6A.6, flooding will not affect the operation of the safety-related equipment located in the RCB. Thus the environmental conditions resulting from all high energy pipe breaks (discussed in Section 3.6A.1) will not affect the operation of the safety related equipment.

3.6A.2 HIGH ENERGY PIPE BREAK OUTSIDE CONTAINMENT

The high energy piping systems which are considered for pipe rupture analysis outside Containment are:

a) Chemical and Volume Control System (charging pump discharge lines; i.e., charging line and seal injection lines; letdown line to downstream of letdown heat exchanger).

b) Steam Generator Blowdown System (SGBS)

c) Main Steam (MS) and Feedwater (FW)

d) Auxiliary Feedwater

e) Extraction Steam and Auxiliary Steam Lines

f) Safety Injection System

The criteria used to locate the break points for high energy piping outside Containment are described in Section 3.6.2. The various protective methods used to mitigate the consequences of the postulated pipe break are given in Section 3.6.1.2.

3.6A.2.1 Pipe Whip Analysis

This section describes the method of protection used against pipe whip for each pipe break in the system listed in Section 3.6A.2.

3.6A.2.1.1 Chemical and volume control system.

3.6A.2.1.1.1 General description

The Chemical and Volume Control System (CVCS) operation and design bases are described in Section 9.3.4. The flow diagram for the CVCS is shown on Figures 9.3.4-1 through 9.3.4-5.

The high energy portion of the charging system outside Containment extends from the charging pump discharge to the penetration. This portion of charging line is three-inch, four-inch, and two-inch Schedule 160 pipe. The charging pipe is made of stainless steel and is designed in accordance with the ASME Code, Section III, Class 2, and seismic Category I Criteria.

The letdown piping from the penetration to a valve in the piping downstream of the letdown heat exchanger is classified as high energy. The letdown line outside Containment is three-inch Schedule 40S. The entire high energy portion of the system is made of stainless steel and designed in accordance with the ASME Code, Section III, Class 2 and seismic Category I Criteria.

3.6A.2.1.1.2 Pipe whip analysis

Figures of the charging lines outside Containment indicating postulated break points and pipe whip restraint locations are provided on Figures 3.6A-12, 3.6A-13, 3.6A-18, 3.6A-26, and 3.6A-27. All breaks in this line are a double-ended guillotine (circumferential) type. Terminal end breaks are postulated at the penetration, the intermediate anchor points, the closed valve end mini-flow orifice end, and the charging pumps discharge nozzles. The intermediate break locations are selected based on the stress criteria given in Section 3.6.2.1. The type of restraint, coordinate direction in which the restraints are capable of supporting the pipe whip load, and the identification of breaks which activate each restraint are given in Table 3.6A-14.

Figures of the letdown lines outside Containment indicating postulated break points and pipe whip restraint locations are provided on Figures 3.6A-12, 3.6A-13, 3.6A-26, and 3.6A-27. All breaks postulated for the piping are a double ended guillotine (circumferential) type. Terminal end breaks are postulated at the penetration, the normally closed valves, and the inlet and outlet nozzles of the letdown heat exchanger. The intermediate break points are selected based on the stress criteria given in Section 3.6.2.1.

Pipe whip restraints provided for the system are listed in Table 3.6A-14.

3.6A.2.1.2 Steam Generator Blowdown System

3.6A.2.1.2.1 General description

The Steam Generator Blowdown System (SGBS) operation and design bases are described in Section 10.4.8. The flow diagram of the SGBS is shown on Figure 10.1.0 6.

The high energy portion of the SGBS outside Containment extends from the penetration to the blowdown tank. Piping from penetration to the isolation valve has been designed in accordance with Safety Class 2 and seismic Category I requirements. The piping from the isolation valve outside Containment up to Reactor Auxiliary Building-Turbine Building interface wall is designed in accordance with Safety Class 3, seismic Category I requirements. The remainder of the system is non-seismic and non-safety related.

The SGBS lines outside Containment are four-inch nominal size Schedule 40 and six-inch Schedule 40 pipe. The entire SGBS piping outside Containment is made of carbon steel.

3.6A.2.1.2.2 Pipe whip analysis

Break locations are selected in the SGBS lines in accordance with the stress criteria for Class 2 pipe presented in Section 3.6.2. Figures of the SGBS lines outside Containment indicating postulated break points and pipe whip restraint locations is provided in Figure 3.6A-25. All breaks in the SGBS piping are a double-ended guillotine (circumferential) type. Terminal end breaks are postulated at the penetrations and the anchor point at the Reactor Auxiliary and Turbine Building interfaces.

The type of restraint, coordinate direction in which the restraints are capable of supporting the pipe whip load, and the identification of break which activated each restraint are given in the Table 3.6A-13.

For those breaks which have no pipe whip protection, a detailed study was conducted to evaluate the effects of the whipping pipe on the essential systems and components. It was found that the whipping pipe will not affect the operation of any essential systems needed to mitigate the consequences of the break and to shut down the plant.

3.6A.2.1.3 Main Steam and Feedwater

3.6A.2.1.3.1 General Description

General description and the criteria for break selection for the main steam and feedwater piping outside Containment is given in Section 3.6.1.2. Refer to Figure 3.6.2-1 which illustrates the main steam and feedwater piping outside Containment.

The main steam line between penetration and the isolation valve is a 34 inch line with a 29.625 inch minimum I.D. Downstream isolation valves to the main steam header are 32 inches with 29.625 inch minimum I.D. The header is 50 inch with 3.279 wall thickness. Two 44 inch lines run on the roof from the header to the turbine building.

The mathematical model for main steam outside Containment is given in Figures 3.6A-32, 3.6A-32.1 and 3.6A-32.2.

The mathematical model for feedwater outside Containment is given in Figure 3.6A-33.

The main steam and feedwater piping systems located in the Turbine Building are classified as ANSI B31.1, non-seismic. In accordance with the criteria of SRP 3.6.2, breaks are postulated to occur at terminal ends and at each intermediate pipe fitting, welded attachment, and valve. Break and restraint locations and jet impingement envelopes are shown in Figures 3.6A-2, 3.6A-6, and 3.6A-7.

3.6A.2.1.3.2 Pipe Whip Analysis

As discussed in Section 3.6.2.1.4, no breaks are postulated in the area near the containment isolation valves. Tables 3.6A-15 through 3.6A-18 present comparisons of the combined pipe stresses versus the required allowables for the operating and pipe rupture conditions. The reported pipe rupture stress at each node was individually determined from the various pipe whip analyses. Lengths of the main steam and feedwater piping from penetration to isolation valves where no breaks are postulated are shown on Figure 3.6.2-1.

At each of the postulated break locations, dynamic analyses were performed to determine the fluid thrust forcing functions and pipe whip restraint loads. The coordinate directions in which the restraints are capable of supporting the pipe whip loads are given in Tables 3.6A-10 and 3.6A-11.

3.6A.2.1.4 Auxiliary Feedwater

3.6A.2.1.4.1 General Description

Auxiliary Feedwater System (AFW) operation and design basis are described in Section 10.4.9. The high energy portion of the AFW outside Containment extends from the steam generator auxiliary feed pumps to containment penetrations.

3.6A.2.1.4.2 Pipe Whip Analysis

The AFW lines indicating break locations and pipe whip restraints are provided in Figures 3.6A-5 through 3.6A-8. All breaks in this line are a double-ended guillotine type. Terminal end breaks are postulated at the penetration, the intermediate anchor points, and the pump discharge nozzles. The intermediate break locations are selected based on the stress criteria given in Section 3.6.2.1. The type of restraint, direction in which the restraints are capable of supporting the pipe whip load, and the identification of breaks which activate each restraint are given in Table 3.6A-12.

For breaks where pipe whip protection is not provided by means of restraints, a study was conducted to evaluate the effects of the whipping pipe on essential systems and components, it was observed that the whipping pipe would not compromise the function of any essential systems.

3.6A.2.1.5 Extraction and Auxiliary Steam Lines

3.6A.2.1.5.1 General Description

The extraction steam system operation and design bases are described in Section 10.2.2.4. The flow diagram for this system is shown on Figure 10.1.0 2.

The auxiliary steam system consists of steam supply from the auxiliary boiler to all equipment requiring auxiliary steam. Various equipment is located in the Reactor Auxiliary Building, Waste Processing Building, and Turbine Building. This system is not a safety-related system and is not required to operate during design bases accidents.

3.6A.2.1.5.2 Pipe Whip Analysis

Pipe breaks are postulated at every weld point in the Extraction Steam and Auxiliary Steam lines. The entire Extraction Steam System is located in the Turbine Building, therefore, no essential components are affected by pipe whip. One auxiliary steam header is routed through the RAB which supplies steam to the boric acid batch tank waste cycle evaporator.

The line is routed through cubicles containing non-essential components and is equipped with automatic isolation devices (i.e., Safety Class 3, Seismic Category I excess flow check valves) which would limit blowdown due to a postulated pipe rupture. No pipe whip protection is required since no essential component is impacted by auxiliary steam line breaks.

3.6A.2.1.6 Safety Injection System

3.6A.2.1.6.1 General Description

The Emergency Core Cooling System or the Safety Injection System (SIS) operation and design basis are described in Section 6.3. The flow diagram for the Safety Injection System is shown on Figures 6.3.2-1 through 6.3.2-3 and Figure 5.1.2-1. The high energy portion of the SIS outside the Containment Building extends from the discharge of the Charging/Safety Injection pumps to the first normally closed valves which are 2SI-V500SA, 2SI-V501SB, 2SI-V502SA, 2SI-V505SB, and 2SI-V506SA.

3.6A.2.1.6.2 Pipe Whip Analysis

The SIS lines indicating break locations and pipe whip restraints are provided in Figures 3.6A-17 through 3.6A-19. All breaks in these lines are a double-ended guillotine type. Terminal end breaks are postulated at RAB penetrations, the intermediate anchor points, normally closed valves and the pump discharge nozzles. The intermediate break locations are selected based on the stress criteria given in Section 3.6.2.1. The type of restraint, direction in which the restraints are capable of supporting the pipe whip load, and the identification of breaks, which activate each restraint are given in Table 3.6A-14.

For breaks where pipe whip protection is not provided by means of restraints, a study was conducted to evaluate the effects of the whipping pipe on essential systems and components. It was observed that the whipping pipe would not compromise the function of any essential systems.

3.6A.2.2 Jet Impingement Analysis Outside Containment

The essential components and systems located outside containment required for safe shutdown of the plant located outside Containment are evaluated for the effects of jet impingement. The jet impingement analysis procedure and the results of the analysis are presented here.

The jet envelopes were drawn on the system piping drawings for all high energy pipe breaks outside Containment (identified for the various piping systems in Section 3.6A.2). The shape of the jets is dependent on the fluid phase. The various jet shapes used for the analysis are described in Section 3.6.2. The jet envelope drawings along with the Component/System layout drawings for Electrical, Instrumentation and Control, HVAC, and Mechanical Systems are used to identify all component-jet interactions for each high energy pipe break. Component-jet interactions were judged acceptable or unacceptable according to the plant shutdown logic, single active failure criteria, and environmental effects. Components involved in interactions identified as unacceptable were either moved out of the jet completely or were further analyzed for the operability of the components under the jet impingement loading. The procedure for calculating jet impingement force on a target is given in Section 3.6.2.3.4. If the equipment/component cannot be qualified with the jet loading, they are protected from the jet impingement forces by the methods described in Section 3.6.1.2.

The high energy piping located outside Containment are: 1) Chemical and Volume Control, 2) Steam Generator Blowdown, 3) Main Steam and Feedwater, and 4) Auxiliary Feedwater.

a) Major portions of the Chemical and Volume Control System are routed through the shielded pipe tunnel in the RAB. The letdown heat exchanger and the charging pumps are located in the individual compartments. The fluid jets from pipe breaks will be contained in these compartments. The compartments are designed for the resulting jet loads. Jet impingement interactions for the remainder of the piping that is not routed through the shielded pipe tunnel were found to be acceptable. Unacceptable interactions are being protected or relocated outside the jet envelopes.

b) Steam generator blowdown interactions with safety related components required for safe plant shutdown were found to be acceptable. Unacceptable interactions are being protected or moved out of the jet envelopes.

c) The first postulated break on main steam and feedwater piping outside Containment is located in the Turbine Building where no safety related components are located. The Reactor Auxiliary Building Wall-B is designed for steady state jet impingement loads from the time history blowdown analysis.

d) Auxiliary feedwater jet impingement interaction with safety related components required for safe plant shutdown were found to be acceptable. Unacceptable interations are being protected or moved out of the jet envelope.

3.6A.2.3 Environmental Effect of High Energy Breaks Outside Containment

The high-energy piping systems located in the Reactor Auxiliary Building are given in Section 3.6A.2. The first postulated break on main steam and feedwater piping outside Containment is located in the Turbine Building. Therefore, the environmental conditions provided by the pipe breaks in these systems cannot affect safety-related components located in other buildings. Portions of the main steam and feedwater piping is routed through the Steam Tunnel in the RAB where environmental effects were considered as given in Section 3.6.2.1.4.

The environmental effects of high-energy line breaks originating from Steam Generator Blowdown, CVCS Letdown, CVCS Charging, CVCS Seal Injection, and Safety Injection piping within the RAB were evaluated for temperature, pressure, humidity, flooding, and radiation. Breaks from these systems would not inhibit the ability to initiate or maintain safe shutdown of the plant.

The environmental effects of breaks originating from Extraction Steam piping do not require evaluation as this piping is entirely contained in the Turbine Building. The Auxiliary Steam and Auxiliary Feedwater systems meet the time criteria of Section 3.6.1.2.2 and are excluded from high-energy pipe rupture environmental analysis.

3.6A.3 SUBCOMPARTMENT PRESSURE ANALYSIS

This section presents the results of the subcompartment pressure analysis in which the integrity of subcompartments is evaluated for the differential pressure loading resulting from high energy piping failures.

3.6A.3.1 Subcompartment Pressure Analysis - Inside Containment

The subcompartments inside Containment which are subject to pressure transients caused by the mass and energy releases from postulated high energy pipe breaks within their boundaries were analyzed. Section 6.2.1 presents a complete description and results of this analysis.

3.6A.3.2 Subcompartment Pressure Analysis - Outside Containment

The subcompartment pressure analysis in the Reactor Auxiliary Building (RAB) is limited to the high energy piping failures. The high energy piping located in the RAB are given in Section 3.6A.2.

Due to the large compartment size of the RAB and the relatively large net free volume, the internal pressure buildup due to postulated high energy pipe breaks is not expected. Therefore, the only cases considered for analysis were the main steam and feedwater line breaks in the Steam Tunnel and CVCS pipe break in the shielded pipe tunnel.

A detailed blowdown analysis has not been performed for the charging pump and RHR heat exchanger compartments. It was assumed that sufficient vent area to the larger RAB volume existed such that overpressurization of the compartments was not expected.

In accordance with the criteria presented in Standard Review Plans 3.6.1 and 3.6.2, no specific pipe breaks are postulated in the main steam, main feedwater, and branch piping up to the first isolation valve. However, in order to provide an additional level of assurance of operability of equipment required to achieve/maintain safe shutdown located within the steam tunnel and adjacent to tunnel, the building structure and essential equipment are designed for the environmental conditions (pressure, temperature, and flooding) that would result from a crack, equal in area to one cross sectional pipe area of either the largest main steam or feedwater line. The postulated crack is considered to be nonmechanistic in nature (i.e., not based upon stress and/or fatigue analysis). Pipe whip and jet impingement effects are excluded from this analysis. Qualification of equipment required to achieve/maintain safe shutdown is performed to the resulting environmental conditions (refer to Section 3.11).

The following cases were analyzed to determine the worst environmental conditions for the main steam tunnel compartment.

Case 1: Blowdown from a main steam line break crack, equivalent to the flow area of a single ended rupture (1.4 ft^2 - corresponds to the total flow area of the SHNPP steam generator flow restrictor located in the steam generator discharge nozzle - refer to Sections 5.4.4 and 15.1.5).

Case 2: Blowdown from a main feedwater line break equivalent to the flow area of a single ended rupture (1.03 ft^2).

Case 1, Main Steam Line Break, resulted in the maximum compartment temperature, 441°F, and the maximum compartment pressure, 19.8 psia, which reflects the results of the SGR/PUR evaluation.

The pressure and temperature subcompartment analysis was performed for main steam line break considering superheat (Case 1) using the COMPRESS computer code (Reference

3.6A-5). The steam tunnel was subdivided into subcompartment volumes and connecting junctions as presented in the nodalization model provided in Figures 3.6A-34a and 34b. The tunnel was divided into twelve (12) subcompartments for the pressurization analysis and five (5) subcompartments for the temperature analysis. The division of the subcompartments was based upon the physical structure which makes up the tunnel and the arrangement of the main steam piping.

The steam tunnel is a large room adjacent to the steam line containment penetrations where the main steam line isolation valves are located. This room is vented to atmosphere and is cooled by two safety-related 40,000 cfm fans.

For case 1 the worst environmental temperature effects were calculated based on Mass & Energy releases provided by Westinghouse that included superheat blowdown (i.e., the effects of steam generator tube bundle uncovery). These Mass & Energy releases reflect the SGR/PUR configuration and are specific to SHNPP.

References 3.6A-1 through 3.6A-4 provide the inputs and methodology for the Mass & Energy release applicable at SGR/PUR conditions and for the main steam line tunnel temperature/pressure profiles.

Plots of the Case 1 MSLB time history of the tunnel pressure and temperature analysis are presented in Figures 3.6A-35 and 3.6A-40, respectively.

The tunnel is designed to withstand the resulting peak pressure to assure structural stability and to prevent adjacent areas from being affected by a break in the tunnel. Equipment required to achieve/maintain safe shutdown located within the steam tunnel are qualified to resulting environmental conditions. A discussion of the thermal lag effects of the MSLT environment on safety related equipment is discussed in FSAR Section 3.11E.

The original pressure and temperature subcompartment analysis was performed for the main feedwater line break (Case 2) using the RELAP IV computer code. The compartment was subdivided into subcompartment volumes and vent areas as presented in the nodalization model provided in Figure 3.6A-34. The steam tunnel was divided into twelve (12) subcompartments based upon the physical structure which makes up the tunnel, and the arrangement of the main steam, main feedwater piping, and associated isolation valves.

Plots of the Case 2 MFLB time history of tunnel pressure and temperature analysis are presented in Figures 3.6A-41 and 3.6A-42, respectively.

The Case 2 MFLB pressure and temperature responses were not updated for SGR/PUR since the original MFLB pressure and temperature profiles were enveloped by the MSLB profiles in the MSLT.

A flooding analysis was performed assuming a single ended rupture of the largest feedwater line located in the steam tunnel. The design basis analysis concluded that the maximum calculated flood level would be less than 7 ft. above the steam tunnel floor elevation of 263 ft. All equipment required to achieve/maintain safe shutdown within the steam tunnel is located above the flood level to preclude damage to the equipment, or does not suffer a loss of an essential function from the flooding. The AFW isolation valve actuators remain above the maximum postulated flood level (267 ft) for those events which require AFW isolation.

The main steam and feedwater line breaks' environmental effects are described in Sections 3.6.2.1.4 and 3.11.

The maximum internal pressure build-up in the shielded pipe tunnel, due to a break in the three-inch CVCS letdown piping was found to be 0.6 psig. The shielded pipe tunnel was designed to withstand this differential pressure loading.

3.6A.4 MODERATE ENERGY PIPING FAILURES - INSIDE CONTAINMENT

The high energy pipe breaks inside Containment were determined to be the enveloping design bases breaks; therefore, the effects of moderate energy piping failures were not evaluated. See Section 3.6.2.1.3.

3.6A.5 MODERATE ENERGY PIPING FAILURES - OUTSIDE CONTAINMENT

The flooding and environmental conditions resulting from moderate energy piping failures were considered for evaluating the availability of essential systems and components. The flooding analysis in RAB due to moderate energy piping failures is described in Section 3.6A.6. The environmental conditions due to moderate energy piping failure in RAB were considered for evaluating the functional capability of safety related equipment and components. The environmental conditions for which the equipment is qualified are given in Section 3.11.

3.6A.6 FLOODING ANALYSIS

3.6A.6.1 Scope

The following sections present the results of an evaluation of systems in the Reactor Containment Building (RCB) and Reactor Auxiliary Building (RAB) associated with the effects of flooding resulting from postulated piping failures. The areas investigated and the features incorporated in the plant design in order to comply with these criteria are also described.

3.6A.6.2 Criteria and Assumptions

The criteria employed in the flooding analysis are based on Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment."

3.6A.6.3 Reactor Containment Building

The Safety Injection System Sump and Containment Sump in the Reactor Containment Building (RCB) are designed to collect the fluid due to the design basis Loss-of-coolant accident. All safety-related equipment is located above the highest water level in the RCB. Thus, the flooding analysis for the RCB is not required. The consequences of flooding in the Turbine Building and the Fuel Handling Building are not addressed because no equipment essential for safe shutdown is located in these buildings.

3.6A.6.4 Reactor Auxiliary Building

In general, the worst flooding in the RAB results from postulated cracks in moderate energy piping. The exceptions to this are the CSIP rooms for which high-energy breaks at the CSIP discharge result in greater water depth. However, such high-energy breaks will, by default, render the affected CSIP immediately inoperable such that the final water depth in a single-train CSIP room is inconsequential.

The assumptions and guidelines used in the moderate energy flooding evaluation are as follows:

a) No earthquake is postulated concurrent with a crack in moderate energy piping.

b) Offsite power is assumed to be unavailable where a postulated crack results in a direct reactor trip or turbine generator trip causing automatic separation of the turbine generator from the power grid.

c) A single active failure is assumed in systems used to shut down the plant or to mitigate the consequences of the crack.

d) All available systems including those actuated by operator actions are used to mitigate the consequences of the crack.

e) Operator action is based upon supervisory information, response time, and access to equipment for the proposed actions. Redundant Class IE level switches and associated main control room indicating lights warn the operators of excessively high RAB Equipment Drain Sump Water Levels. Thirty minutes from event initiation to manual initiation of protective action such as closing or opening a valve, or shutting off or starting a pump, has generally been assumed in the analysis and is considered to be ample time. For those cases in which operators would likely not take action within thirty minutes, the analysis shows that safety-related equipment is not affected by the flooding.

f) A moderate energy fluid system pipe failure is considered separately as a single postulated initial event occurring during normal plant operation.

g) Rate of flow from cracks is assumed to be from an infinite reservoir.

3.6A.6.4.1 Evaluation Technique

All compartments in the RAB were modeled on general arrangement drawings. Each compartment identifies the piping systems routed through these compartments. The moderate energy piping failure in the compartment or in the communicating compartment which produces the worst flooding condition for each compartment was selected for the evaluation.

From the analysis, it was determined that the limiting break is a break on a component cooling water line in a RHR heat-exchanger room which would deprive that RHR heat-exchanger of its heat removal capability. This is acceptable based on the guidance of section B.3.b(3) of Branch Technical Position SPLB 3-1. All other flooding scenarios do not adversely affect any safety-related equipment.

REFERENCES: SECTION 3.6A

3.6A-1 Carolina Power and Light Company letter HR/99-105, dated November 1, 1999, Main Steam Line Break Mass & Energy Release Data for the Main Steam Line Tunnel.

3.6A-2 WCAP-10961, Rev. 1, (Proprietary), "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment, Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup," October 1985.

3.6A-3 Carolina Power and Light Company letter HR/00-054, February 18, 2000, Main Steam Line Break Mass & Energy - Outside Containment.

3.6A-4 Carolina Power and Light Company letter HR/00-073, March 9, 2000, Main Steam Line Break Mass & Energy - Outside Containment.

3.6A-5 COMPRESS M-1, RE&C SW#MC-095, October 1991, "A Code for Computing Sub-Compartment Pressure Responses". [Topical Report VEC-TR-004-0 submitted to NRC December, 1975].

3.7 SEISMIC DESIGN

3.7.1 SEISMIC INPUT

3.7.1.1 Design Response Spectra

Two earthquake motions were considered in the dynamic analyses of all Seismic Category I structures, systems, subsystems, and equipment: the operating basis earthquake (OBE) and safe shutdown earthquake (SSE). Definitions and peak accelerations associated with the OBE and SSE were established according to the seismicity evaluation described in Section 2.5.2.

The design value of the maximum horizontal ground acceleration is 0.15g for the safe shutdown earthquake and .075g for the operating basis earthquake.

The design response spectra used for all Seismic Category I structures, systems, and components, except dams and dikes, were developed in accordance with Regulatory Guide 1.60. The horizontal and vertical design response spectra, normalized to 0.15g for the SSE and 0.075g for the OBE, are presented on Figures 3.7.1-1 through 3.7.1-4 and were applied at the foundation level.

The design response spectra used for the Seismic Category I dams and dikes were based on a modified form of a smoothed response spectra developed from the strong motion record of the 1935 Helena, Montana earthquake, normalized to the maximum horizontal ground accelerations of the safe shutdown earthquake and the operating basis earthquake. This record was obtained from a seismograph that was established on competent bedrock and is, therefore, considered appropriate for the proposed plant site.

The horizontal design response spectra for the dams and dikes, normalized to 0.15g for the SSE and 0.075g for the OBE, are presented on Figures 3.7.1-5 and 3.7.1-6, respectively. The vertical design response spectra for the dams and dikes, normalized to 0.10g for the SSE and

0.05g for the OBE, are presented on Figures 3.7.1-7 and 3.7.1-8, respectively. The seismic analysis of the dams and dikes, based on the design response spectra presented on Figures 3.7.1-5 through 3.7.1-8, is discussed in Section 2.5.6. An evaluation of the behavior of the dams and dikes during an earthquake whose response spectra were developed using the Regulatory Guide 1.60 methodology, is also presented in Section 2.5.6.

3.7.1.2 Design Time-History

Earthquake synthetic time histories, consistent with the horizontal and vertical design response spectra presented in Section 3.7.1.1, were generated.

The earthquake synthetic time histories used in the analysis of all Seismic Category I structures, systems, and components, except dams and dikes, were developed for the horizontal and vertical components of SSE and OBE by using the Ruiz and Penzien procedure. This procedure utilizes a linear stochastic model to generate records of filtered nonstationary shot noise to simulate ground motion accelerograms recorded during strong motion earthquakes, as described in Reference 3.7.1-1. To assure that the spectra of the generated accelerogram envelops the specified design spectra, Tsai's procedure was utilized. Tsai's procedure consists of a deterministic technique which modifies the accelerograms by passing the motion successively through a set of frequency filters to suppress or raise any local portion of the response spectra to match the design response spectra, as described in Reference 3.7.1-2. The horizontal and vertical earthquake accelerograms for the maximum horizontal ground accelerations of 0.5g for SSE and 0.075g for OBE are presented on Figures 3.7.1-9 through 3.7.1-12. The time histories were derived at time steps of 0.005 seconds and have a duration of 10 seconds.

A comparison of the spectral values that were derived from the time histories of the components of the SSE and OBE and the design response spectra, was made at the following frequencies:

FREQUENCY INTERVAL (CPS)	INCREMENT (CPS)
0.2 - 3.0	.10
3.0 - 3.6	.15
3.6 - 5.0	.20
5.0 - 8.0	.25
8.0 - 15.0	.50
15.0 - 18.0	1.0
18.0 - 22.0	2.0
22.0 - 34.0	3.0

This comparison is presented on Figures 3.7.1-13 through 3.7.1-24, for two, four, and seven percent critical damping.

Three statistically independent excitations were developed for three different directions, and were used for the dynamic analysis of the Nuclear Steam Supply System and for the stability analysis of the Seismic Category I structures, in order to combine the three representative components of the earthquake motion, at each time algebraically, as described in Section 3.7.2.

The statistical independence of the three synthetic free-field acceleration time-histories was established by comparing statistical properties of the synthetic time-histories with properties derived from recorded earthquake accelerograms. In particular, the values of the normalized correlation coefficient at zero time delay and the average value of the coherence function over the seismic frequency range have been calculated for the synthetic and real time-histories and shown to be comparable.

The earthquake accelerograms, shown on Figures 3.7.1-9 and 3.7.1-10, were used for the SSE and OBE, respectively, for the north-south direction. Two additional sets of statistically independent accelerograms, developed for the east-west and vertical directions, are presented on Figures 3.7.1-25 through 3.7.1-28.

A comparison of the spectral values of the SSE statistically independent horizontal east-west and vertical time histories, and the corresponding design response spectra, is presented on Figures 3.7.1-29 through 3.7.1-34, for two, four, and seven percent damping, using the frequency intervals discussed above. The comparisons discussed above show that none of the points fall below ten percent of the design response spectrum, and no more than five points fall below the design response spectrum.

The earthquake accelerograms used in the analysis of the Seismic Category I dams and dikes envelop the horizontal and vertical design response spectra presented on Figures 3.7.1-5 through 3.7.1-8. Figures 3.7.1-35 through 3.7.1-37 show the SSE horizontal accelerograms for one, two, and five percent damping.

To demonstrate that these time histories envelop the design response spectra, a high resolution response spectra analysis was performed. Each time history was analyzed at 247 discrete period points between the period range of 0.014 to 3.000 sec. These period points were spaced at 0.0005 sec. intervals at the short period end and at 0.1 sec. intervals at the long period end. These period intervals were established by performing response analysis at both half resolution (124 period points) and full resolution (247 period points). It was found that there was essentially no change in the general shape of the response spectra. Therefore, these 247 closely spaced period points are considered to be sufficient to detect all the peaks and valleys of the response spectra.

Comparison of these time histories with the horizontal design response spectra for the SSE are indicated on Figures 3.7.1-38, 3.7.1-39 and 3.7.1-40, for one, two, and five percent damping, respectively.

3.7.1.3 Critical Damping Values

The damping ratios, which are expressed as percentages of critical damping and used in the dynamic analysis of Seismic Category I structures, are consistent with those of Regulatory Guide 1.61, and are shown in Table 3.7.1-1.

For the Seismic Category I Main Dam, Auxiliary Dam and Auxiliary Separating Dike, the seismic analysis is presented in Section 2.5.6.

For Seismic Category I cable tray supports, damping ratios per 1978 Bechtel Power Corporation Cable Tray and Conduit Test Program (Report No. 1053-21.1-4) are to be utilized.

For the Seismic Category I reactor coolant loop system, Seismic Category I piping systems, and Seismic Category I equipment not purchased as of March 1, 1977, the SHNPP complies with the damping values of Regulatory Guide 1.61. In accordance with the provision of Regulatory

Position C2, documented test data have been provided to and approved by the NRC which justifies the use of a damping value higher than three percent critical for large piping systems under the faulted condition. A conservative value of four percent critical has been justified by testing for the Westinghouse reactor coolant loop, as presented in WCAP-7921-AR "Damping Values of Nuclear Power Plant Components", May, 1974 (Reference 3.7.1-3). The damping values for control rod drive mechanisms and the fuel assemblies of the NSSS, used in the reactor coolant system analysis, are in conformance with the values for welded and/or bolted steel structures (as appropriate) listed in Regulatory Guide 1.61. Tests on fuel assembly bundles justified conservative component damping values of seven percent for OBE and ten percent for SSE in the fuel assembly (see Reference 3.7.1-4).

The damping values in Code Case N-411 may be used for piping as an option to the values in Regulatory Guide 1.61 (Table 3.7.1-1) for Balance of Plant piping and Table 3.7.2-16 for Class 1 piping 12 inches and larger (including reactor coolant loop). The Code Case N-411 values will be used for reanalysis of any calculation representing a piping system or portion of a piping system designed for seismic loads for IE Bulletin 79-14 as-built reconciliation, plant modifications, or support/snubber optimization. A combination of Regulatory Guide 1.61 values and those in Code Case N-411 may not be used. Code Case N-411 may not be used for time history analysis.

The damping values used in the stress analysis for the steam generator replacement were based on Regulatory Guide 1.61 damping. Values used were 2% and 3% damping for OBE and 3% damping for SSE.

Seismic Category I equipment already purchased as of March 1, 1977 has been reevaluated using the damping values of Regulatory Guide 1.61, except as indicated in the preceeding paragraph, and requalified where necessary.

The damping characteristics for the supporting media at the site were determined by laboratory tests on representative samples of applicable soil strata, as described in Section 2.5.4. The rock damping values used in the dynamic analysis are two and five percent for OBE and SSE, respectively.

3.7.1.4 Supporting Media for Category I Structures

Except as discussed below, all Seismic Category I structures are founded on sound rock which has a shear wave velocity of 5600 ft/sec. Section 2.5.4.3 shows structural locations, depth, and height. It is, therefore, concluded that the lumped spring approach is suitable, and that the interaction between the Seismic Category I structure foundations and the surrounding soil is represented by springs (rotational, torsional, and translational) which were developed according to the procedures described in Reference 3.7.1-5. The spring constants were calculated by using the soil parameters and formulas presented in Table 3.7.1-2 and Reference 3.7.1-5 (Tables 10-13 and 10-14, and Figure 10-16).

The translational spring Kx, which represents the compressibility of the soil on the sides of the Seismic Category I structure foundations, was determined by using the procedures described in Reference 3.7.1-6.

For the vertical model, only a vertical translation spring was introduced for consideration of the soil-structure interaction. This spring, which represents the compressibility of the rock directly beneath the building foundations, was determined by using the procedures described in Reference 3.7.1 6.

The Seismic Category I underground piping systems, underground electrical conduits and electrical manholes, founded on soil, compacted fill, or weathered rock, are the only Seismic Category I structures that are not founded on sound rock. The dynamic analysis procedures and structure-soil interaction of these structures are discussed in Section 3.7.2.4A and 3.7.3.12. The foundation supporting media of the dams and dikes, their seismic analysis, and the soil-structure interaction are discussed in Section 2.5.6.

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2A BALANCE OF PLANT SCOPE

3.7.2.1A Seismic Analysis Methods - Balance of Plant Scope

The seismic analyses of the Seismic Category I structures were performed by using the normal mode time-history technique. The structures, considered as seismic systems and analyzed in this manner, are identified in Table 3.7.2-1. The analyses of the Main Dam and Spillway, the Auxiliary Dam and Spillway, and the Auxiliary Separating Dike are discussed in Section 2.5.6. Seismic analyses of components and equipment that are provided by Westinghouse are discussed in Sections 3.7.2B and 3.7.3.

All Seismic Category I structures, except Seismic Category I underground piping systems, underground electrical conduits, and electrical manholes, are founded on sound rock; therefore, the lumped mass-spring approach was used to develop the mathematical model for the dynamic analyses of the structures. The mathematical model assumes a single cantilever or multi-cantilever lumped mass system. The lumped masses are connected by weightless elastic bars which represent the stiffness of structural walls and/or columns. Each mathematical model is supported by a mass which represents the foundation mat; the interaction of the foundation mat with the supporting rock medium is represented by linear elastic springs.

The lumped masses are located at floor levels and at any other points where the dynamic responses are important. The dead weights of the structural floor system, steel framing, grating, miscellaneous steel, equipment, piping, and electrical cables and trays (considered as a uniform load distributed over the floor) are included in the lumped mass at the corresponding level. The dead weights of columns and structural walls are evenly distributed between the levels over which they span. The dead weights of block walls are lumped at the levels at which they are supported.

The mathematical model adopted for the Containment Building dynamic analyses consists of two individual cantilevers representing the containment structure and the internal structures, respectively. The two cantilevers are founded on the same base which, in turn, is supported by rotational and translational springs due to the foundation-rock interactions. The mathematical model is shown on Figure 3.7.2-1. The wall of the Concrete Containment Structure does not support any substantial floors. Interior grating platforms are supported by steel beams and independent steel columns in order to preclude interaction between the internal structures and the containment structure. Mass points of the containment structure are located at the platform elevations and at the locations where floor response spectra are required, such as the equipment hatch, personnel air locks, polar crane girder, and main steam and feedwater lines, and at locations of seismic instruments. In addition, mass points are provided at intermediate locations so that they are no more than 50 ft. apart.

The mathematical model for the vertical seismic analyses is slightly different from the model used for the horizontal analyses. In the vertical model, branch mass points maybe provided which are off the center line of the cantilever. A branch point may represent that portion of the floor where major equipment is located. Floor response spectra at branch points include the effects of stiffness of floor slabs in the vertical direction. The branch stiffness is calculated in accordance with the guidelines stipulated in References 3.7.2-1 and 3.7.2-2.

Two dynamic degrees of freedom, one for horizontal translation and one for rotation, are allowed for each mass point for the horizontal seismic analysis. The rocking motion due to the vertical excitation is negligibly small, and is therefore ignored. Vertical translation is the only response considered in the vertical seismic analysis.

Torsional effects have been taken into account by incorporating a torsional degree of freedom at the base of the three-dimensional lumped-mass model. This is discussed further in Section 3.7.2.11A.

An adequate number of masses and degrees of freedom in each dynamic structural model has been taken into consideration. In all cases, either the number of degrees of freedom has been chosen to be more than twice the number of modes with frequencies less than 33 cps, or so that the inclusion of additional modes does not result in more than a 10 percent increase in responses.

In the modal analysis, modes up to at least 33 cps frequency were considered for seismic responses of the structures.

Criteria for differential seismic movements between interconnecting piping, component, and equipment supports are discussed in Sections 3.7.3.8, 3.7.3.9, and 3.9.3. Differential seismic movements between Seismic Category I structures do not cause structural coupling. A sufficient gap is provided between the foundations and superstructures of adjacent buildings at all elevations to preclude any pounding by one another due to seismic events up to and including the SSE.

The seismic analysis of the Nuclear Steam Supply System (NSSS) utilized nonlinear, three-dimensional, time-history dynamic analysis methods; the NSSS is coupled with the building internals structural model (see Figure 3.7.2-2). The coupled model is subjected to three components of earthquake forces simultaneously at the ground node point. The three free-field time-history components, the N-S and E-W horizontal directions and the vertical direction, are statistically independent and are applied simultaneously for ten seconds of OBE and SSE. The seismic analysis of the NSSS was performed using the Westinghouse computer code WECAN. The methodology and verification of this code is presented in WCAP 8929 "Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program."

Typical mathematical models for the dynamic analyses of horizontal and vertical excitations for Seismic Category I structures are shown on Figures 3.7.2-3 through 3.7.2-6 and 3.7.2-13 through 3.7.2-16.

As described in Sections 3.7.2.1A and 3.7.2.3A, three and/or two dimensional, lumped mass models have been used in the dynamic analyses of Seismic Category I structures.

For such structural models, the equations of dynamic equilibrium can be expressed in a matrix form as:

$$[M]\ \{\ddot{x}\}\ +\ [C]\ \{\dot{x}\ \}\ +\ [K]\ \{x\ \}\ =\ -[M]\ \{I\ \}\ \ddot{x}_g \qquad (1)$$

where:

[M] = diagonal mass matrix

[C] = damping matrix

[K] = stiffness matrix

{I} = unit vector

$\{x\}$ = displacement vector

$\{\dot{x}\}$ = velocity vector

$\{\ddot{x}\}$ = acceleration vector

$\ddot{x}_g$ = seismic ground acceleration

Equation (1) is the coupled-system equation; in the dynamic analysis, the terms are decoupled and solved by modal analyses procedures.

The modal analysis is based on the assumption that [M], [C], and [K] can be diagonalized by a common transformation of the coordinate system. Let

$$\{x\} = [\varphi]\ \{y\} \qquad (2)$$

where: $[\varphi]$ = Transformation matrix and $[\varphi]^T$ = Transpose of $[\varphi]$

Substituting Equation (2) into Equation (1) and premultiplying by [φ]T, the transpose of [φ] gives:

$$[\varphi]^T\ [M]\ [\varphi]\ \{\ddot{y}\} + [\varphi]^T\ [C]\ [\varphi]\ \{\dot{y}\} + [\varphi]^T\ [K]\ [\varphi]\ \{y\} = -\ [\varphi]^T\ [M]\ \{I\}\ \ddot{x}_g \qquad (3)$$

This is a classical eigenvalue problem. When the transformation matrix is chosen to be the eigenvectors, [M], [C], and [K] are diagonalized as follows:

$$[\cancel{I}] = [\varphi]^T\ [M]\ [\varphi] \qquad (4a)$$

$$[\cancel{d_2}] = [\varphi]^T\ [C]\ [\varphi] \qquad (4b)$$

$$[\cancel{\omega}] = [\varphi]^T\ [K]\ [\varphi] \qquad (4c)$$

ω in Equation 4(c) is the natural angular frequency.

d in Equation (4b) is the damping value

The above transformation reduces Equation (3) to:

$$\lceil \bcancel{I} \rfloor \{\ddot{y}\} + \lceil \bcancel{d} \rfloor \{\dot{y}\} + \lceil \bcancel{\omega^2} \rfloor \{y\} = - [\varphi]^T [M] \{ I \} \ \ddot{x}_g \quad (5)$$

Equation (5) consists of n single-degree equations; the solutions of these equations are readily available (Reference 3.7.2-3).

Since Equation (5) is a set of single-degree modal equations, these equations may be compared with the free damped single-degree equation

$$d_j = 2 \beta_j \omega_j \quad (6)$$

where β_j is the modal damping ratio of the j-th mode, also called the modal damping factor. In this comparison, the assumption is made that $[\varphi]^T$, [C], and $[\varphi]$ yield diagonal terms only. Generally, if material damping of the system does not vary widely, the off-diagonal terms are comparatively negligible and Equation (4b) is then an acceptable assumption.

The single-degree equation of motion is (See Regulatory Guide 1.61):

$$\ddot{u} + 2 \beta \omega \dot{u} + \omega^2 u = - \ddot{x}_g \quad (7)$$

Utilizing the solution of Equation (7), the solution of Equation (5) is simply:

$$y = \lceil \bcancel{u} \rfloor [\varphi]^T [M] \{I\} = [u] \{\Gamma\} \quad (8)$$

where:

$$\{\Gamma\} = [\varphi]^T [M] \{I\} \quad (8a)$$

is defined as the participation factor. Substituting {y} into Equation (2) yields the displacement vector,

$$\{x\} = [\varphi] \lceil \bcancel{u} \rfloor \{\Gamma\} \quad (9)$$

The velocity vector,

$$\{\dot{x}\} = [\varphi] \lceil \bcancel{\dot{u}} \rfloor \{\Gamma\} \quad (10)$$

and the acceleration vector,

$$\{\ddot{Z}\} = \{\ddot{x} + \ddot{x}_g\} = -[M]^{-1} ([C] \{\dot{x}\} + [K] \{x\}) \quad (11)$$

$\ddot{Z}$ is the absolute acceleration of mass.

as well as forces, shears, and moments are also determined.

In the time history analysis, u is the solution of Equation (7) as a function of time. In the spectral analysis method the same equation of motion governs except u is the input spectrum value.

Since spectrum values are the maximum modal responses without phasing, the statistical average sum instead of the algebraic sum is used in the modal superposition. The most generally accepted statistic average sum is the square root of the sum of the squares (SRSS). Therefore, instead of using Equations (9) and (11) to obtain displacements and accelerations, the following equations are used:

$$X_i = \left[\sum_{k=1}^{m} (\varphi_{ik} u_k \Gamma_k)^2\right]^{1/2} \quad (12)$$

$$\ddot{x}_i = \left[\sum_{k=1}^{m} (\varphi_{ik} \ddot{u}_k \Gamma_k)^2\right]^{1/2} \quad (13)$$

Where m denotes the number of modes taken in the spectrum response analysis computations.

The forces, shears, and moments are obtained in a similar manner. However, an exception to this procedure is that portion of the response which is due to closely spaced modes. Two consecutive modes are defined as closely spaced in a response spectrum modal dynamic analysis if their frequencies differ from each other by less than 10 percent of the lower frequency. The values of the response of these modes are combined using criterion specified in Regulatory Guide 1.92.

The modal damping expression is the same in all the computer codes including STARDYNE and DYNAMICS.

The damping value used for the containment structure is a constant, since only one material is used in the structure. Therefore, the resultant matrix $\varphi^T C \varphi$ is a diagonal matrix with zero off-diagonal terms, when the transformation matrix φ is chosen to be the eigenvectors. In fact, the modal damping factor is calculated as a proportion of modal dissipated energy to the total modal energy (see Reference 3.7.2-10), or

$$D_r = \frac{\sum_{i=1}^{N} E_{ir} d_i}{E_r} \quad (13a)$$

in which

D_r = modal damping factor at r-th mode

E_{ir} = energy in i-th mass or member at r-th mode

d_i = fraction of critical damping of i-th mass or member

E_r = total energy at r-th mode

In the strain energy approach

$$D_r = \frac{\sum_{i=1}^{N} \{\varphi_r\}^T d_i [K]_i \{\varphi_r\}}{\{\varphi_r\}^T [K] \{\varphi_r\}} \quad (13b)$$

where

$\{\varphi_r\}$ = mode r eigenvector

[K] = total system stiffness matrix

$[K]_i$ = stiffness matrix for component i

N = number of degrees of freedom with component r

Since constant damping value is used for the containment structure (d_i is constant in Equation (13b)), D_r becomes:

$$D_r = \frac{d_i \sum_{i=1}^{N} \{\varphi_r\}^T [K]_i \{\varphi_r\}}{\{\varphi_r\}^T [K] \{\varphi_r\}} = d_i = \text{constant}$$

which says the modal damping is constant for all modes, and {φ} matrix is not required in the computation. This equation also says that the damping matrix [C] is proportional to the stiffness [K], or

$$[C] = \beta[K]$$

and

$$[\varphi]^T [C][\varphi] = \beta[\varphi]^T [K][\varphi] = \beta[\omega^2_n]_D$$

This shows that $[\varphi]^T [C][\varphi]$ is a matrix without off-diagonal terms, since $[\omega^2_n]_D$ is a diagonal matrix.

3.7.2.2A Natural Frequencies and Response Loads - Balance of Plant Scope

Natural frequencies of all Seismic Category I structures listed in Table 3.7.2 1 are determined by using Ebasco's in house computer program DYNAMIC 2037 for two-dimensional models, and STARDYNE for three-dimensional models.

Natural frequencies, eigenvalues, and participation factors for major seismic Category I Structures are presented in Tables 3.7.2-2 through 3.7.2-6.

The maximum structural responses (displacements, accelerations, shear forces, and bending and torsional moments) for major Seismic Category I structures are given in Tables 3.7.2-7 through 3.7.2-11.

Floor response spectra at selected locations have also been generated. Procedures for development of floor response spectra are discussed in Section 3.7.2.5A.

Figure 3.7.2-7 shows a typical floor response spectra in the Containment Building, at the operating floor elevation, for one percent critical damping.

3.7.2.3A Procedure Used for Modeling - Balance of Plant Scope

The mathematical models for seismic analysis of Seismic Category I structures are described in Section 3.7.2.1A. The Seismic Category I structures that are considered in conjunction with a soil-structure interaction are defined as "seismic systems." Other Seismic Category I systems and components that are not designated as "seismic systems" are considered as "seismic subsystems." In general, the frequencies of system and subsystems alone have negligible effect on the error due to uncoupling. Therefore, the mass ratio, Rm, and the frequency ratio, R_f, govern the results. R_m and R_f are defined as:

$$R_m = \frac{\text{Total mass of the supported subsystem}}{\text{Mass that supports the subsystem}} \tag{14}$$

$$R_f = \frac{\text{Fundamental frequency of the supported subsystem}}{\text{Frequency of the dominant support motion}} \tag{15}$$

The following criteria are used for decoupling:

a) If R_m <0.01, decoupling can be done for any R_f.

b) If $0.01 \leq R_m \leq 0.1$, decoupling can be done if $0.8 \geq R_f \geq 1.25$

c) If R_m > 0.1, an approximate model of the subsystem is included in the primary system model.

If the subsystem is comparatively rigid and also rigidly connected to the primary system, only the mass of the subsystem is included at the support point in the primary system model. In the case of a subsystem supported by very flexible connections, i.e., a pipe supported by hangers, the subsystem is not included in the primary model. The equipment and components which fall under the definition of subsystems are analyzed as systems decoupled from the primary structure. Seismic input for the decoupled subsystem is obtained from the analysis of the primary structure which includes only the mass of the subsystem. The mass and stiffness characteristics of the reactor coolant system components are, however, incorporated as separate cantilevers and connected to the mathematical model of the primary structure to obtain the interaction and decoupling effects.

As described in Section 3.7.2.11A, an investigation was performed for all Seismic Category I structures to determine if the coupling effects of those degrees of freedom that were omitted from the three-dimensional models were significant. Three-dimensional models are used only where torsional and coupling effects are significant.

Where two-dimensional models are used, independent analysis for three orthogonal directions of seismic motions, two horizontal and one vertical, is performed and maximum responses are combined by the square root of the sum of the squares (SRSS) technique, as described in Section 3.7.2.6A.

3.7.2.4A Soil-Structure Interaction - Balance of Plant Scope

All Seismic Category I structures, except the underground piping systems, the underground electrical conduits, and the electrical manholes, are founded on sound rock. The dynamic analysis was performed using the lumped mass-spring approach.

The lumped mass-spring approach, also called the equivalent soil-springs and dash pots method, is based on the analytical solution of the model of a rigid mat resting on the surface of an elastic half-space.

The values of the foundation springs are calculated by using formulas presented in Reference 3.7.1-5. The dash pots of the lumped system represent the damping of the soil in the foundation soil system. There are two types of damping in the real system: one induced by the loss of energy through propagation of elastic waves away from the immediate vicinity of the footing, the other associated with internal energy losses within the soil due to hysteretic and viscous effects. The equivalent damping, corresponding to the elastic-wave propagation, has been designated as "geometric damping", occasionally called "radiation damping." The expressions for the geometric damping ratio to the critical damping obtained through the half-space theory and corresponding analogs for rigid circular footings are summarized in Reference 3.7.1-5. Damping ratios and variation of damping ratio with strain are discussed in Section 2.5D.13.3.2.2.2, and presented in the figures referenced in that section. The equations for geometrical damping, developed by using the theory of vibration of a rigid circular footing on an elastic half space are used to provide estimates for the geometrical damping of footings with rectangular form in plan. This is accomplished by converting the rectangular base of dimensions 2c by 2d into an equivalent circular base having a radius r_o, determined by the following conversions, as suggested in Reference 3.7.1-5.

For translations:

$$r_o = \left[\frac{4cd}{\pi}\right]^{1/2} \qquad (16a)$$

For rocking:

$$r_o = \left[\frac{16cd^3}{3\pi}\right]^{1/4} \qquad (16b)$$

For torsion:

$$r_o = \left[\frac{16cd\,(c^2 + d^2)}{6\pi}\right]^{1/4} \qquad (16c)$$

in which

2c =width of the foundation (along the axis of rotation for the case of rocking)

and

2d = length of the foundation (in the plane of rotation for rocking)

Ebasco's computer program, DYNAMIC 2037 for 2D models, and STARDYNE for 3D models, have been used for seismic analysis by using the lumped mass-spring approach. The output provides maximum responses (acceleration, shear, and moment) as well as the acceleration time history at each mass point. The response spectra for the various damping ratios are then generated from the time histories.

Verification of the bed rock representation as one-way spring by STARDYNE for the stability (overturning moments) analysis was obtained by comparison of the total kinetic energy of the structure and ground directed towards overturning with the potential energy of raising the total weight of the structure from its center of gravity to a height sufficient to rotate around a foundation edge. Buoyancy effects were considered, using the groundwater elevation specified for the site in Section 2.4.13.5.

The Reactor Auxiliary Building was used for the comparison. The mathematical models for the two horizontal directions and the vertical direction, and the value of mass at each mass point are shown in Figure 3.7.2-3. The structure was assumed to be a flexible cantilever for the evaluation. The overturning of the structure requires that it rotate about a foundation edge, with the structure acting as an inverted pendulum. The natural period of the structure as an inverted pendulum is much longer than that of the linear elastic structural response.

The maximum kinetic energy for overturning was obtained for the two principal horizontal orthogonal directions and the vertical direction using the velocities from the safe shutdown earthquake 2 percent damping response spectra, for the first mode frequency (Figures 3.7.1-1, 3.7.1-13; and 3.7.1-19). Comparison of the kinetic energy acting to overturn the structure with the potential energy required to rotate the structure about a foundation edge to reach the point of instability indicated that overturning moments are negligible.

Eccentricity time-history curves for the Reactor Auxiliary Building mat considering both N-S plus vertical, and E-W plus vertical safe shutdown earthquake were prepared, and the 100 percent mat contact area line was placed on the curves. For the N-S direction plus vertical, there is only one peak outside the line, and for the E-W direction plus vertical, there are relatively few widely scattered spikes which, in general, last relatively short times. The spikes are all within at least 75 percent mat contact area, and are considered to have a negligible effect on the results of the dynamic analysis of the structure.

Factors of safety were calculated for all seismic Category I structures according to FSAR Section 3.8.5.5. A tabulation of the factors of safety for overturning, for the Category I structures is presented in Table 3.7.2-18.

The factor of safety against overturning was computed by dividing the resisting moments by the overturning moments, as stated in FSAR Section 3.8.5.5.

The overturning moment for the SHNPP was calculated by the square root of the sum of the squares (SRSS) of the horizontal acceleration only in both the E-W and N S directions. The vertical seismic overturning moment was added directly, as 0.4 times the actual moment.

$$M = \sqrt{M^2EW + M^2NS} + 0.4M\ V \qquad (1)$$

A study was also conducted for the SHNPP to compute the overturning moment by the SRSS method by taking the square root of the sum of the square of the overturning moments due to the horizontal acceleration in both the E-W and N-S directions, plus the overturning moments due to the vertical seismic uplift forces.

$$M = \sqrt{M^2EW + M^2\ NS + M^2\ V} \qquad (2)$$

The resisting moment is the vertical load of the building and mat (reduced by buoyancy) multiplied by the distance to the edge of the building.

The factor of safety against overturning computed by both the methods described above resulted in a higher factor of safety than the minimum required by the FSAR.

The factor of safety for the RAB calculated in accordance with Method 1 above is 1.25 and for Method 2 is 1.33. Both are higher than the acceptable limit of 1.10.

For all computer codes used for seismic dynamic analysis, the model used a spring that was always in contact with the foundation rock. The output from the computer consisted of a shear, moment, and reaction at the foundation level. Hand computations were made to determine the stability of the structure and the safety factor for sliding and overturning.

The Seismic Category I electrical manhole structures consist essentially of very rigid reinforced concrete boxes fully buried in the surrounding soil. Because of their relatively small sizes, the individual structures are assumed to be single mass points excited by the same accelerations as those of the surrounding soil mass. The ground acceleration at the level of individual manholes was determined by an amplification analysis of ground motion through a vertical soil column between the bedrock and the manholes by using the computer program SHAKE developed by the University of California, Berkeley.

The program SHAKE is based on the assumption that the main response in a soil deposit is caused by the upward propagation of shear waves from the underlying rock formation. Surface waves and pressure waves were not considered to be significant for determining the acceleration amplification for the simplified design approach for electrical manholes which are buried in soil. All other Seismic Category I structures are founded directly on rock.

The program SHAKE computes the responses associated with vertical propagation of shear waves through a linear viscoelastic system. The program is based on a continuous solution to the wave equation adapted for use with transient motion through the fast Fourier transform algorithm. The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent linear soil properties and an iterative procedure to obtain values for modulus and damping that are compatible with the effective strain in each layer.

The input soil properties were obtained from field geophysical measurements and dynamic laboratory testing. The model is divided into a layered system; each layer is completely defined by its characteristics of shear modulus, critical damping ratio, unit weight, and thickness. The horizontal and vertical safe shutdown earthquake motions were inputted separately at the base of the model (bedrock). Through the iterative process, the strain compatible solutions were obtained, and the new motions at the top of each layer were computed. The vertical soil column models are shown on Figure 3.7.2-8. The values of shear modulus and dampings are shown as a function of layer for each model. The height of soil column for the seismic design of electrical manholes varies from 15 feet to 25 feet. Since the ground acceleration obtained for the worst column height has been used for the design of all the manholes, one to one correspondence between all the manholes to the models in Figure 3.7.2-8 is illustrated by showing one typical manhole.

The non-linearity of shear modulus and damping is accounted for by the use of equivalent linear soil properties (Idriss & Seed 1968 and 1970) using an iterative procedure to obtain values of shear modulus and damping compatible with the effective strain in each layer.

The maximum ground accelerations at the level of the manholes, obtained through the above analysis, are further increased by 50 percent for the equivalent static analysis of each structure.

The accelerations used for design purposes, as obtained from the above procedure, are as follows:

Horizontal SSE Acceleration: 0.25g
Vertical SSE Acceleration: 0.19g
Horizontal OBE Acceleration: 0.14g
Vertical OBE Acceleration: 0.10g

Other buried Category I structures are:

a) ESWS Screen Structure

b) ESWS Intake Structure

c) ESWS Discharge Structure

d) Diesel Fuel Oil Storage Tank Building

These structures are founded directly on sound rock. The design approach used in their dynamic analysis is discussed above.

3.7.2.5A Development of Floor Response Spectra - Balance of Plant Scope

Ebasco's in-house computer program DYNAMIC 2037 was used to generate floor response spectra for the damping values specified in Table 3.7.1-1. The floor response spectra calculations were based on the exact analytical solutions of the governing differential equations for the successive linear segments of the excitation, specified at equal time intervals. The method is described in detail in Reference 3.7.2-3.

The floor response spectra were generated separately for three directions of earthquake motion. A decoupled analysis of the subsystems was then performed using the floor response spectra as described in Section 3.7.3. The floor response spectra for simultaneous action of three directions of earthquake were not generated.

The response spectrum method for the development of floor response spectra for Seismic Category I structures was not used.

Each floor response spectrum was also broadened on the frequency axis in order to take into account any parametric variations in properties, such as shear modulus, damping, and material, as discussed in Section 3.7.2.9A.

In general, floor response spectra have been constructed for one percent, two percent, and four percent damping values for the OBE, and two percent, four percent, and seven percent for the SSE.

3.7.2.6A Three Components of Earthquake Motion - Balance of Plant Scope

The seismic analysis of all Seismic Category I structures, systems, and components takes into consideration three orthogonal directions of seismic motions; two horizontal and one vertical. The maximum responses to each of the three components of motion are determined separately and combined by the square root of the sum of the squares (SRSS) method to obtain the total seismic responses in accordance with Regulatory Guide 1.92. The simultaneous application of time histories or linear summation of responses are not performed. the SRSS in mathematical form is:

$$R_j = \pm\left[\left(R_{j^1}^2\right) + \left(R_{j^2}^2 + R_{j^3}^2\right)\right]^{1/2} \quad (17)$$

in which R_j denotes the most probable response in the j th direction, considering three directional earthquake effects. Rjk (k = 1,2,3) denotes the response in the j-th direction resulted from the earthquake component in the k th direction. The R can be displacements, velocities, accelerations, forces, moments, or stresses.

The SRSS method is not used in design of foundation mats of Seismic Category I structures. When part of the mat is lifted from the supporting ground, the stresses are redistributed due to the fact that the soil does not take any tension. This foundation separation problem is non-linear in nature and the dynamic stresses are no longer separable from the stresses due to other loading conditions. The inseparable condition makes the SRSS method impossible to apply.

For design of the foundation mats of some of the Seismic Category I structures (identified in Table 3.7.2-1), the time history responses from each of the three components of the earthquake motion were combined at each time step algebraically by using the three statistically independent earthquake excitations described in Section 3.7.1.2.

An alternative method was used for the design of the mats of some buildings. These buildings are identified in Table 3.7.2-1. In this method, the principal orthogonal direction horizontal motion, in each direction separately, was amplified by a factor of 1.2 and combined with the vertical direction earthquake, and other loadings considered to be in effect. Amplifications of horizontal direction earthquake motion was applied only to rectangular mats. No amplification was considered for circular foundations.

Another method suggested by N M Newmark in Reference 3.7.2-5 was also used in modified form. This procedure involves taking the seismic forces corresponding to 100 percent of the motion in two horizontal directions, combining them by the SRSS method, and adding the absolute value with 40 percent of the motion in the vertical direction, then adding the absolute values of these to obtain the maximum resultant force at a point in a particular direction. Stresses corresponding to the combined effect are then computed. Table 3.7.2-1 indicates which method was used for each structure.

3.7.2.7A Combination of Modal Responses - Balance of Plant Scope

Modal responses in the spectral analysis were combined according to the guidelines contained in Regulatory Guide 1.92, as described in Section 3.7.2.1.

3.7.2.8A Interaction of Non-Category I Structures with Category I Structures - Balance of Plant Scope

The following criteria were used to assure that the collapse of non-Seismic Category I structures would not impair the integrity of adjacent Seismic Category I structures or components.

a) Sufficient separation has been maintained between Seismic Category I and non-Seismic Category I structures, or

b) The partial or complete collapse of these structures will not impair the integrity of any of the neighboring Seismic Category I structures or components, or

c) The failure or collapse of non-Seismic Category I structures is prevented under SSE conditions.

Except for the Turbine Building, the retaining wall west of the Fuel Handling Building, and the retaining wall east of Fuel Handling Building by the Unit 2 Reactor Auxiliary Building which do not satisfy condition a) above, the plant arrangement provides for sufficient distance between Seismic Category I structures, systems, and components and non-Seismic Category I structures. The failure or collapse of non-Seismic Category I structures cannot impair the ability of Category I Structures or systems to perform their intended design functions.

The non-Seismic Category I structures or components whose failure could jeopardize Seismic Category I structures or components, including the Turbine Building, the retaining wall west of the Fuel Handling Building, and the retaining wall east of Fuel Handling Building are designed to prevent failure by adopting the following criteria for SSE loads combined with dead and live loads, postulated to be present during the event, using unit load factors:

a) Structural Steel - Structures are seismically designed in accordance with Regulatory Guide 1.29, Positions C2 and C4, with the exception of these items within the structure whose collapse would not impair the ability of the structure to withstand the SSE or the integrity of adjacent Seismic Category I structures or components. For the applicable stresses see Section 3.8.3.3.3 and load combination b-1 of that section.

b) Reinforced Concrete - Structures including the retaining walls west and east of the Fuel Handling Building are seismically designed in accordance with Regulatory Guide 1.29, Positions C2 and C4. The design allowables for reinforced concrete structures are those specified for the Strength Design Method in ACI 318-71 "Building Code Requirements for Reinforced Concrete," except that the capacity reduction factors φ specified by the ACI code are permitted to reach unity. For applicable load combinations, see Section 3.8.4.3.2(a) and (d). For additional details of the retaining walls of the Fuel Handling Building, see FSAR Section 3.8.4.9.

The interface criteria between Seismic Category I and non-Seismic Category I piping are discussed in Section 3.7.3.

3.7.2.9A Effects of Parameter Variations on Floor Response Spectra - Balance of Plant Scope

Floor response spectra for Seismic Category I structures are determined from the in-structure acceleration time histories. The peaks of the floor response spectra are broadened plus or minus fifteen percent in frequency, according to the example shown on Figure 1 of Regulatory Guide 1.122, to account for variation of parameters, such as the material properties of the structure and soil, damping values, soil-structure interaction techniques, and approximations in the modelling techniques.

3.7.2.10A Use of Constant Vertical Static Factors - Balance of Plant Scope

Constant vertical load factors, as vertical response loads for seismic design of Seismic Category I structures, systems, and components, are not used. The vertical responses are obtained from a vertical seismic system multimass dynamic analysis.

3.7.2.11A Method Used to Account for Torsional Effects - Balance of Plant Scope

Where preliminary analyses shows that torsional effects are significant, analyses are performed to take into account torsional effects from the following sources:

a) The general layout of the building is not symmetrical in geometry.

b) The structure is symmetrical in geometry, but the distribution of masses within the building is not symmetrically arranged.

The nuclear power plant structures are shear-wall type construction, and bending deformations are comparatively small; therefore, the rigidity centers approximately coincide with the centers of the effective shear areas. Since the mass point includes both walls and slab, the center of gravity of the mass point will not coincide with the rigidity center. Although some torsional movement due to this difference will occur under horizontal seismic excitation, all the Seismic Category I structures have negligible torsional movement, due to their close-to-symmetrical configuration.

For torsional seismic analysis of the Seismic Category I structures, when torsional effects are significant, three-dimensional lumped-mass cantilever models are utilized. Cantilevers are centered at rigidity centers and lumped masses, located at the mass centers, are connected to the cantilever with rigid links, representing rigid floors. Each node point is provided with two orthogonal horizontal degrees of freedom in the principal directions, and a third rotational degree of freedom in the plane of the two principal orthogonal axes. The section properties, used in 2D horizontal seismic analysis, described in Section 3.7.2.1A, are used for this torsional analysis. In addition, torsional rigidity of the building floors, in the plane of the rotational degree of freedom, is provided.

Soil-structure interaction is considered by including translational and rotational springs and dash-pot dampers. The numerical values of spring constants and dampers are calculated by using formulas presented in Reference 3.7.1-5, as described in Section 3.7.2.4A, for "Lumped Mass-Spring Approach." Torsional dynamic analysis for the two orthogonal horizontal directions of seismic motions is carried out independently. Typical mathematical models for torsional dynamic analysis of the Containment Building are shown on Figure 3.7.2-9 for the three-dimensional torsional model and on Figure 3.7.2-10 for the two dimensional torsional models.

The seismic analysis of the Containment Building, using a three-dimensional mathematical model, was performed by using the normal mode time history technique. The industry-proven computer program STARDYNE was used for this analysis, using as input the simulated time histories, developed as described in Section 3.7.1, to envelop the design response spectra.

Table 3.7.2-12 shows the comparison of the natural frequencies and participation factors for the Containment Building dynamic analysis, using three and two-dimensional torsional mathematical models and a two-dimensional dynamic model. By comparing the natural frequencies and analyzing the participation factors of the coupled and uncoupled models, the comparison shows that the coupling among omitted degrees of freedom was not significant, therefore, the two-dimensional dynamic model was used in the dynamic analysis of the Containment Building.

Table 3.7.2-13 shows the comparison of the maximum structural responses for the dynamic analysis of the Containment Building using three and two dimensional models.

Similar investigations were performed for all major Seismic Category I structures to determine if the incorporation of torsional effects is significant. Except for the Tank Building, it was determined that, for the rest of the Seismic Category I structures, torsional effects are not significant.

Torsional effects are significant for the Tank Building, due to a comparitively large eccentricity between the shear center and the mass center in one direction. Therefore, a three-dimensional torsional mathematical model was used to include the torsional effects. Figure 3.7.2-11 shows the mathematical model for the Tank Building. Table 3.7.2-14 presents the natural frequencies, eigenvalues, and participation factors for different modes, and Table 3.7.2-15 presents the maximum structural responses from the dynamic analysis of the Tank Building.

Although these investigations show that coupling among omitted degrees of freedom is not significant, in order to account for torsional accelerations in the seismic analysis, shear forces acting with an eccentricity of not less than five percent of the maximum dimension at that level were considered in the seismic analysis of the structure.

3.7.2.12A Comparison of Responses - Balance of Plant Scope

Since only one method of analysis has been used for seismic analysis of each structure, no comparison of responses by other methods has been made. All of the Seismic Category I structures have been analyzed by using the modal analysis time-history method. Since the time history method involves direct integration at each time step, time-phase relationships between various modal responses are taken into account, which results in more reliable and accurate structural responses than is obtained by the response-spectrum method. Thus, no comparison of responses was considered to be necessary.

3.7.2.13A Methods for Seismic Analysis of Dams - Balance of Plant Scope

The seismic stability analysis of the Seismic Category I dams and dikes is presented in Section 2.5.6.

3.7.2.14A Determination of Category I Structure Overturning Moments - Balance of Plant Scope

The seismically induced overturning moments for Seismic Category I structures are obtained from the methods of analysis listed for the various buildings in Table 3.7.2-1. As discussed in Section 3.7.2.6A, three statistically independent earthquake excitations were independently applied on the mathematical models of the Seismic Category I structures in order to determine the overturning moments and the shear forces from three orthogonal components of seismic motions. At each time interval, in order to determine the maximum bearing pressure arising from three orthogonal components of seismic motions, the shear forces and overturning moments were algebraically combined. For stability analysis, the vertical inertia force was assumed to act upward, reducing the effective downward weight of the structure. Table 3.7.2-1 indicates methods used in the determination of overturning moments for Seismic Category I structures. The evaluation of the factors of safety against overturning, flotation, and sliding is discussed in Section 3.8.5.

3.7.2.15A Analysis Procedure for Damping - Balance of Plant Scope

The equivalent modal damping ratio is a function of individual material damping of the structure. For the lumped mass-spring approach, the equivalent modal damping, also called composite damping, is evaluated as a weighted average by strain energy proportions based on the stiffness of the elements. The equivalent modal damping ratio of the r-th mode is

$$\beta_r = \frac{\sum_{i=1}^{m}\sum_{\ell=1}^{n}\sum_{j=1}^{n}\varphi_{\ell r}\, k_{\ell j}^{(i)}\ \gamma_i\, \varphi_{jr}}{\sum_{\ell=1}^{n}\sum_{j=1}^{n}\varphi_{\ell r}\, k_{\ell j}\, \varphi_{jr}} \qquad (18)$$

in which:

$ß_r$ = modal damping ratio of the r-th mode

$\varphi_{\ell r}$ = modal displacement of 1-th mass at r-th mode

$k_{\ell j}^{(i)}$ = stiffness of the i-th member

γ_i = fraction of critical damping factor of i-th mass or member

n = number of dynamic degrees of freedom

m = number of masses or members

If γ_i in the above equation is a constant, then $\beta_r = \gamma_i$.

β is also related to C damping matrix in the equation, $[C] = [\varphi^T]^{-1}[2\beta\omega][\varphi]^{-1}$. The values for γ are selected from those specified in Table 3.7.1-1, and are based on Regulatory Guide 1.61.

This methodology and damping expression are the same as that presented in Reference 3.7.2-11. The information therein has been the industry standard found acceptable by the NRC for over a decade.

3.7.2B NSSS SCOPE

3.7.2.1B Seismic Analysis Methods - NSSS Scope

Those components and systems that must remain functional in the event of the SSE are identified by applying the criteria of Section 3.2.1.

In general, the dynamic analyses are performed by using response spectrum analysis, or integration of the uncoupled modal equations as direct integration of the coupled equations of motion, or nonlinear modal superposition.

Dynamic Analysis - Mathematical Model

The first step in the 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 inertial 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 3.7.2-7.

Equations of Motion

Consider the multi-degree of freedom system shown in Figure 3.7.2-12. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r\ddot{y}_r + \sum^i c_{ri}\dot{u}_i + \sum^i k_{ri}u_i = 0 \qquad (19)$$

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 a 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

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad (20)$$

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 (19) can be written as:

$$m_r\ddot{u}_r + \sum^i c_{ri}\dot{u}_i + \sum^i k_{ri}u_i = -m_r\ddot{y}_s \qquad (21)$$

For a single degree of freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad (22)$$

Modal Analysis Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which were determined by eigen solution of Equation (21). The right hand side and the damping term are set equal to zero for this purpose, as illustrated in Reference 3.7.2-1 (pages 83 through 111). Thus, Equation (21) becomes:

$$m_r\ddot{u}_r + \sum^i k_{ri}u_i = 0 \qquad (23)$$

The equation given for each mass point, r, in Equation (23) can be written as a system of equations in matrix form as:

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = 0 \qquad (24)$$

where:

[M]	=	mass and rotational inertia matrix
{Δ}	=	column matrix of the general displacement and rotation at each mass point relative to the base
[K]	=	square stiffness matrix
$\{\ddot{\Delta}\}$	=	column matrix of general translational and angular acceleration at each mass point relative to the base, $d^2 [\Delta]/dt^2$

Harmonic motion is assumed and the {Δ} is expressed as

$$\{\ddot{\Delta}\} = \{\delta\} \sin \omega t \qquad (25)$$

where:

$\{\delta\}$	=	column matrix of the spatial displacement and rotation at each mass point relative to the base
ω	=	natural angular frequency of harmonic motion in radians per second
t	=	time in seconds

The displacement function and its second derivative are substituted into Equation (24) and yield:

$$[K]\{\delta\} = \omega^2 [M]\{\delta\} \qquad (26)$$

The determinant [K] - ω^2 [M] is set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation (26). 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 [K] - ω^2[M] when set equal to zero yields simply:

$$k - \omega^2 m = 0 \tag{26a}$$

or:

$$\omega = \frac{k}{m}^{1/2} \tag{27}$$

where 1 is the natural angular frequency in radians per second.

The natural frequency in cycles per second is therefore:

$$f = \frac{1}{2\pi} \left(\frac{k}{m}\right)^{1/2} \tag{28}$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation (26).

Modal Equations

The response of a structure or component is always some combination of its normal modes. Good accuracy can usually be obtained by using only the first few modes of vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as (Reference 3.7.2-1 pages 116 through 125):

$$\ddot{A}_n + 2\omega_n p_n \dot{A}_n + \omega^2 A_n = -\Gamma_n \ddot{y}_s \tag{29}$$

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u_{rn}, by the equation:

$$u_{rn} = A_n \varphi_{rn} \tag{30}$$

where:

ω_n = natural angular frequency of mode n in radians per second

p_n = critical damping ratio of mode n

Γ_n = modal participation factor of mode n given by:

$$\Gamma_n = \frac{\sum^n m_r \varphi'_{rn}}{\sum^n m_r \varphi^2_{rn}} \tag{31}$$

φ_{rn} = ratio of displacement of mass point r in mode n to maximum modal displacement

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 (29) is analogous to the equation of motion for a single degree of freedom system that can be developed from Equation (22). Dividing Equation (22) by m gives:

$$\ddot{u} + \frac{c}{m}u + \frac{k}{m}u = -\ddot{y}_s \qquad (32)$$

The critical damping ratio of the single degree of freedom system, p, is defined by the equation:

$$p = \frac{c}{c_c} \qquad (33)$$

where the critical damping coefficient, c_c, is given by the expression:

$$c_c = 2m\omega \qquad (34)$$

Substituting Equation (34) into Equation (33) and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \qquad (35)$$

Substituting this expression and the expression for k/m given by Equation (27) into Equation (32) gives:

$$\ddot{u} + 2\omega p\dot{u} + \omega^2 u = -\ddot{y}_s \qquad (36)$$

The similarity of Equations (29) and (36) allows each mode to be analyzed as if 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 is not appropriate for a slightly damped structure (e.g., steel) supported by a massive moderately damped structure (e.g., concrete). There are several methods which can be used to overcome this difficulty.

One method is to develop and analyze separate mathematical models for both structures by using their respective damping values. The massive, moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures is used as a forcing function for the subsequent detailed analysis. A second method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion. A third method is to use the Rayleigh damping method based on computed modal energy distribution. In yet another method, which is utilized in the coupled building/loop analysis, the damping value for a given mode is derived from the calculation of the composite modal damping which is based on the distribution of the strain energy in the structure for that mode.

Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (displacement, velocity, and acceleration) of a single degree of freedom system versus its natural frequency of vibration when subjected to a time history motion of its base.

The response spectrum concept can be best explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree of freedom system with a given damping and natural frequency is calculated for a given base motion.

The variations in response are established and the maximum absolute value of each is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency in sufficient detail to establish the complete curve. Other curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis, and the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with the base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$S_{a_n} = \omega_n S_{v_n} = \omega_n^2 S_{d_n} \tag{37}$$

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.

Integration of Modal Equations

This method can be separated into the following two basic parts:

a) Integration procedure for the uncoupled modal Equation (29) 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.

Integration Procedure

Integration of these uncoupled modal equations is done by step-by-step numerical integration. The step-by-step numerical integration procedure consists of selecting a suitable time interval, Δt, and calculating modal acceleration, An, modal velocity, An, and modal displacement, An, 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 (30) as:

$$u_{rn} = A_n \varphi_{rn} \quad (38)$$

with the corresponding acceleration of mass point r in mode n as:

$$\ddot{u}_{rn} = \dot{A}_n \varphi_{rn} \quad (39)$$

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.

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:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} \quad (40)$$

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 inertial force term of - M (Z) to the right hand side of the basic Equation (40), i.e.,

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} - [M]\left(\ddot{Z}\right) \quad (41)$$

The vector $\{\ddot{Z}\}$ is defined by its components $\ddot{Z}_i$ where i refers to each degree of freedom of the system. Z_i is equal to a_1, a_2, or a_3, if the i-th degree of freedom is aligned with the direction of the system translational acceleration a_1, a_2, or a_3, respectively. $Z_i = 0$ if the i-th degree of freedom is not aligned with any direction of the system translational acceleration. A 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 (41) is the nodal 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 the transient analysis, the portion of internal forces due to static deformation is always balanced by the portion of the forces are statically applied. Hence, on the portion of the forces which deviate from the static loads will produce dynamic effects. The output of this analysis is the total result due to statically and dynamically applied loads.

Nonlinear Modal Superposition

In the nonlinear modal superposition method, the nonlinearities are presented as pseudo forces. The mass and stiffness matrices are calculated only once and the corresponding mode shapes and natural frequencies are associated with the linear system simulating the initial state of the undamped structure with external force acting on it. This state of the structure is hereafter referred to as the reference state. During the time history analysis, as the nonlinear behavior comes into action, the true frequencies and mode shapes change. The effect of the variation of the true frequencies and mode shapes from the original ones is represented by pseudo forces on the right-hand side of the equation of motion.

The generalized equation of motion for a nonlinear structure is:

$$[M]\{\ddot{x}\} + [C_{nl}]\{\dot{x}\} + [K_{nl}]\{x\} = \{F(t)\} \quad (42)$$

where

[M] = mass matrix

$[C_{nl}]$ = nonlinear damping matrix, dependent upon velocity and displacement

$[K_{nl}]$ = nonlinear stiffness matrix, dependent upon displacement

{x}, $\{\dot{x}\}$, $\{\dot{x}\}$, and {F(t)} = displacement, velocity, acceleration and applied force vectors.

Let $[C_{nl}] = [C] + [\bar{C}]$ (43a)

and $[K_{nl}] = K + [\bar{K}]$ (43b)

where [C] and [K] are the damping and stiffness matrices representing the reference state of the structure, $[\bar{C}]$ and $[\bar{K}]$ [K] are the damping and stiffness matrices, dependent on velocity and displacement. Substitution of Equation (43) into Equation (42) gives:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} - \{F_{n1}\} \quad (44)$$

where the pseudo-force vector is defined by

$$F_{nl} = [\bar{C}]\{\dot{x}\} + [\bar{K}]\{x\} \quad (45)$$

The homogeneous, undamped equation of motion representing the reference state of the structure is:

$$[M]\{\ddot{x}\} + [K]\{x\} = 0 \qquad (46)$$

Let [ω] and [φ] be the natural frequency and normalized mode shape matrix; {q} is the modal displacement vector. The following transformation:

$$\{x\} = [\varphi]\{q\} \qquad (47)$$

is substituted in Equation (44), resulting in the following uncoupled modal equation:

$$\{\ddot{q}\} + [2\,\zeta_j\omega_j]\{\dot{q}\} + [\omega_j^2]\{q\} = \{Q\} = \{Q_{n1}\} \qquad (48)$$

where

ζ_j = percentage of the critical damping for jth mode

$\{Q\} = [\varphi]^T\{F(t)\}$ = generalized applied force vector. (48a)

$\{Q_{nl}\} = [\varphi]^T \{F_{nl}\}$ = generalized pseudo-force vector.

Arrays {q}, $\{\dot{q}\}$, $\{\ddot{q}\}$, are the modal displacement, velocity, and acceleration vectors, respectively. The generalized pseudo-force vector is a function of displacement and velocity. For a given time step, it can be approximated by the Taylor series for a given time step; modal equations of motion are integrated analytically. Then the displacement and velocities of the nodes associates with nonlinear elements are calculated. This information is used to calculate the generalized pseudo-force vector and its time derivatives. Then the nodal equations are integrated for the next time step.

3.7.2.2B Natural Frequencies and Response Loads - NSSS Scope

Natural frequencies and maximum structural responses of the Seismic Category I structures are presented in Section 3.7.2.2A.

3.7.2.3B Procedures Used for Modeling - NSSS Scope

Procedures used for modeling are discussed in Section 3.7.2.1B and 3.7.2.3A.

3.7.2.4B Soil-Structure Interaction - NSSS Scope

The soil-structure interaction is discussed in Section 3.7.2.4A.

3.7.2.5B Development of Floor Response Spectra - NSSS Scope

The development of the floor response spectra is discussed in Section 3.7.2.5A.

3.7.2.6B Three Components of Earthquake Motion - NSSS Scope

The seismic design of the piping and equipment includes the effect of the seismic response of the supports, equipment, structures, and components. The system and equipment response is determined by using three earthquake components, two horizontal and one vertical. The design ground response spectra are the bases for generating these three input compartments. Floor response spectra are generated for two perpendicular horizontal directions and the vertical direction. System and equipment analysis is performed with these input components. The damping values used in the analysis are those given in Table 3.7.2-16.

In computing the system and equipment response by response spectrum modal analysis, the methods of Section 3.7.2.7B are used to combine all significant modal responses to obtain the combined unidirectional responses.

The combined total response is then calculated by 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, and stress, 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:

$$R_C = \left(\sum_{T=1}^{3} R_T^2\right)^{1/2} \tag{49}$$

where:

$$R_T = \left(\sum_{i=1}^{N} R_{Ti}^2\right)^{1/2} \tag{50}$$

where:

R_C = total combined response at a point

R_T = value of combined response of direction T

R_{Ti} = value of response for direction T, mode i

N = total number of modes considered

The subscripts can be reversed without changing the results of the combination.

For the case of closely spaced modes, R_T in Equation (49) shall be replaced with R_T as given by Equation (51) in Section 3.7.2.7B.

The system and equipment response can also be determined by using time history analyses.

If time history analysis is performed, the two horizontal and vertical time history components are applied simultaneously.

3.7.2.7B Combination of Modal Responses - NSSS Scope

The total unidirectional seismic response is obtained by combining the individual modal responses by 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 towards successively higher frequencies. No one frequency is in more than one group. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor e. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{i=1}^{S} \sum_{K=M_j}^{N_{j-1}} \sum_{\ell=K+1}^{N_j} R_K R_\ell \varepsilon_{K\ell} \quad (51)$$

where:

R_T = total unidirectional response

R_i = absolute value of response of mode i

N = total number of modes considered

S = number of groups of closely spaced modes

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

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

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

$$\varepsilon_{K\ell} = \left[1 + \left[\frac{\omega'_K\, \omega'_\ell}{\beta'_K\, \omega_K\, \beta'_\ell\, \omega_\ell}\right]^2\right]^{-1} \quad (52)$$

and

$$\omega'_K = \omega_K \left[1 - (\beta'_K)^2\right]^{1/2} \quad (53)$$

$$\beta'_K = \beta_K + \frac{2}{\omega_K\, t_d} \quad (54)$$

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

3.7.2.8B Interaction of Non-Category I Structures with Seismic Category I Structures - NSSS Scope

Interaction of non-Seismic Category I structures with Seismic Category I structures is discussed in Section 3.7.2.8A.

3.7.2.9B Effects of Parameter Variations on Floor Response Spectra - NSSS Scope

The effects of parameter variations on the response spectra is discussed in Section 3.7.2.9A.

3.7.2.10B Use of Constant Vertical Static Factors - NSSS Scope

Constant vertical static factors are not used as the vertical floor response load for the seismic design of safety class systems and components within Westinghouse's scope of responsibility. All such systems and components are analyzed in the vertical direction.

3.7.2.11B Methods Used to Account for Torsional Effects - NSSS Scope

The seismic analysis of the NSSS equipment is based on a three dimensional coupled mathematical model, using the time-history dynamic analysis method and subjected to three statistically independent earthquake components, applied simultaneously at the base of the NSSS-containment coupled model, and, therefore, the torsional effects are automatically considered.

3.7.2.12B Comparison of Responses - NSSS Scope

Since the dynamic analysis of the NSSS equipment is based on a three dimensional mathematical model using the model analysis time-history method no comparison with other methods has been made.

3.7.2.13B Methods for Seismic Analysis of Dams - NSSS Scope

The methods for seismic analysis of dams and dikes are discussed in Section 3.7.2.13A.

3.7.2.14B Determination of Seismic Category I Structure Overturning Moments - NSSS Scope

Seismic Category I structure overturning-moment determination is discussed in Section 3.7.2.14A.

3.7.2.15B Analysis Procedure for Damping - NSSS Scope

Procedures for damping are discussed in Section 3.7.2.1B.

3.7.2.16B Comparison of Response Spectra Using a Soil Structure Interaction Spring Constant on the Spectra Generated by Replacing the Spring Constant With a Fixed Coupling

The response spectra utilized for equipment procurement and structural design, broadened in accordance with Regulatory Guide 1.60, and using a soil structure interaction spring constant, was compared with the spectra generated by replacing the spring constant with a fixed coupling and using actual average concrete strengths.

This comparison showed the acceleration vs. frequency curves for the fixed base case to be enveloped by the plant design basis with minor exceptions. In those few areas where the fixed base acceleration was locally greater than the spring base, an examination of potential impact was conducted. In all cases, the fixed base peak was lower than the spring base peak, and occurred at a higher frequency. A review of the few items of potential safety significance located at these areas confirmed that there would have been no impact on design even if the fixed base response were utilized. This confirmed the conservatism of the lumped-mass, spring base model.

The "f'c" values for concrete as used in the comparison of fixed base vs. spring base model were the average of concrete cylinder strength values obtained by testing the concrete for each structure.

The average "f'c" values for each structure, obtained as described above, are summarized in Table 3.7.2-17, attached.

The following specific equipment was reviewed in the following area for the comparison study:

a) Fuel Handling Building - All safety-related equipment above El 286 was reviewed for possible impact:

    1) Radiation Monitors
    2) Low Range Flow Switches
    3) Level Switches
    4) Thermocouple Assemblies and Test Thermowells
    5) Butterfly Valves
    6) HVAC Duct Supports
    7) Miscellaneous Electrical Mounting Details (dwg B-060)
    8) Electrical Boxes (dwg B-044)

In cases where the peak of the fixed base analysis exceeds the spring base, representative node points were selected and a frequency analysis performed. The only two pieces of equipment for which this was necessary were the cask crane and the auxiliary crane, both in the Fuel Handling Building.

Two (2) representative node points on the cask crane dynamic analysis model and one (1) representative node point on the auxiliary crane dynamic analysis model were selected. Response in the vertical direction due to the first 12 modes (13 modes for the auxiliary crane) were combined for these three node points; and, in all cases studied, the acceleration due to the fixed base model floor response spectrum was less than the acceleration for the spring base model. No changes in the equipment design were therefore necessary.

For all other equipment checked, the broadened spring base response spectra curves enveloped and the fixed base response spectra curves; hence, no further analysis was necessary.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

3.7.3.1 Seismic Analysis Method

3.7.3.1.1 Balance of Plant Scope

Seismic Category I piping systems were analyzed as discussed in Section 3.7.3.8.

Seismic Category I cable trays, conduits, HVAC duct systems and equipment supports were analyzed by the method of modal response spectra. This accounted for the effects of multiple spans and multiple modes on seismic response.

A three dimensional mathematical model was constructed with a sufficient number of dynamic degrees of freedom to closely simulate the dynamic behavior of the subsystems.

The effects of foundation torsion, rocking, and translation were included in the seismic response analysis of the Seismic Category I structures so that the effects of these parameters are reflected in the floor response spectra, which are used as the seismic input data for the seismic response analysis of the subsystems.

All of the significant modes of the subsystems were selected for the determination of the seismic response. In general, modes from 1 to 33 Hz were selected, but modes higher than 33 hz were also included where the first mode of the subsystem in some direction had a frequency greater than 33 Hz.

When the supports for a subsystem were all mounted at the same floor, the relative displacement among supports was not considered. This relative displacement was considered where the supports of the same subsystem were located at different floors.

For the case where the supports of the same subsystem were located in different buildings, the maximum relative displacements among the different supports were considered in the seismic dynamic analysis of the subsystem.

Relative displacements within a structure were assumed to be in phase relative to the mat. Relative displacements between buildings were assumed to be totally out of phase.

3.7.3.1.1.1 Modal Response Spectra Method

The method of dynamic analysis by the modal response spectra method is as follows:

a) Basis of Analysis

1) The system is linearly elastic.

2) Masses are lumped at discrete intervals and are connected by weightless elastic members. The maximum spacing between mass points does not exceed one-half of the distance for which the frequency of a simple support beak would be 33 cps. Furthermore, it is verified that the number of degrees of freedom considered in the analysis are equal to or more than twice the number of modes with frequencies less than 33 Hz.

3) Each mass point has up to six degrees of freedom except for points indicated as restrained in a given direction.

4) The system is anchored at two or more positions and these anchor points are assumed fixed for the determination of natural frequencies and mode shapes.

5) Dynamic loadings in the three orthogonal coordinate directions are determined separately and are combined on the basis of excitation occuring in the vertical and two mutually perpendicular horizontal directions at the same time by the square root of the sum of the squares method.

6) The mass polar moment of inertia, i.e., the mass component involved in rotation, is negligible.

7) Damping is viscous and assumed constant for all modes.

8) For piping, increased flexibility due to bends is included in the analysis. Pressure is included according to applicable codes.

b) Method of Analysis

1) Frequency Analysis - The stiffness matrix method of natural mode analysis is employed to determine natural frequencies and associated mode shapes.

The equations of motion for the subsystem may be written as:

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = \{F\} \qquad (1)$$

where:

[M] = Diagonal matrix of lumped masses, the rows and columns of which are arranged to correspond to the components of the stiffness matrix. The masses that are effective in the three coordinate directions are taken to be equal to the total mass assumed to be lumped at the point under study.

[K] = Square, symmetric matrix of stiffness coefficients including the effects of axial deformation, bending, torsion, and shear in the three coordinate directions.

$\{\Delta\}$ = Column matrix of displacement.

$\{\ddot{\Delta}\}$ = Column matrix of acceleration.

{F} = Column matrix of external loads.

The stiffness matrix [K] is assembled as follows:

Each subsystem section has the following properties:

E = Modulus of elasticity

μ = Poissons ratio

I = Moment of inertia

A = Cross-sectional area

L = Length

From these properties the characteristics of the section are computed:

$G = \frac{E}{2\,(1+\mu)}$; $GJ = \frac{EI}{1+\mu}$; $\alpha = \beta = 2\frac{(1+\mu)}{AE}$

$\varepsilon = n = \frac{1}{EI}$; $\gamma = \frac{1}{AE}$; $\lambda = \frac{1}{GJ} = \frac{1+\mu}{EI}$

The end flexibility of the section is contained in the 6 x 6 matrix φ:

$$\varphi = \begin{bmatrix} \alpha L + \frac{\varepsilon L^3}{3} & 0 & 0 & 0 & \frac{\varepsilon L^2}{2} & 0 \\ 0 & \beta L + \frac{nL^3}{3} & 0 & \frac{-nL^2}{2} & 0 & 0 \\ 0 & 0 & \gamma L & 0 & 0 & 0 \\ 0 & \frac{-nL^2}{2} & 0 & nL & 0 & 0 \\ \frac{\varepsilon L^2}{2} & 0 & 0 & 0 & \varepsilon L & 0 \\ 0 & 0 & 0 & 0 & 0 & \lambda L \end{bmatrix}$$

A transformation matrix [R] is established to bring the pipe section into the general coordinate system. This matrix is based on the orientation and location of the section in the overall system. The flexibility in the generalized coordinate system is obtained from the element flexibility matrix [φ] in the local coordinate system as:

$$[\varphi_G] = [R]\,[\varphi]\,[R]^T$$

The flexibilities [φ_G] are accumulated for each element and the stiffness coefficients are computed as $K_A = [\varphi_G]^{-1}$ and assembled into the overall stiffness matrix [K]. For the determination of natural frequencies and mode shapes, Equation (1) is solved by first setting the external loads {F} equal to zero and the displacement vector {Δ} = {δ} sin ωt.

Then:

$$\{\ddot{\Delta}\} = -\{\delta\}\,\omega^2 \sin \omega t$$

Equation (1) becomes:

$$[K]\{\delta\} = \omega^2 [M]\{\delta\}$$

This generalized eigenvalue equation is solved by iterative techniques to determine the natural frequencies and mode shape vectors {δ_n} of the system.

This generalized procedure permits the K and M Matrices to be assembled for systems containing multiple constraints, multiple loops, and multiple lumped masses as well as simple single branch systems. The matrices are assembled using conventional finite element techniques. A sample calculation is provided in Table 3.7.3-2.

For the case of multiple fixed branches and looped systems, the input motion is transmitted to the piping system through the piping supports. The piping supports are modeled as springs acting in the appropriate directions. Spring rates are based on characteristic values for various rigid restraints and snubbers, as stated in FSAR Section 3.7.3.8.1. Since the system may be situated at various elevations and/or structures, the input motion is resolved into two distinct effects. A response spectrum analysis is performed to determine inertia effects. For this analysis, as stated in FSAR Section 3.7.3.9.1, an enveloped response spectrum is used. The effect of differential seismic movement between floors and/or building is considered statically in an integrated system analysis, as stated in FSAR Section 3.7.3.1.1. Relative displacements within a structure are assumed to be in phase relative to the mat, while relative displacements between structures are assumed to be totally out of phase.

c) Modal Analysis

Let N = Total number of lumped masses

d = Direction X, Y, Z

n = Mode number

δ_{idn} = Shape factor (i^th component for the n^th mode and direction d)

$SA_{nd}(t)$ = Instantaneous acceleration response in d^th direction for mode n

$\overline{SA}_{nd}$ = Floor response spectral acceleration in d^th direction for mode n [i.e., equal to maximum of SA_{nd} (t)]

then

M_i = mass at i

R_{nd} = $\sum_{i=1}^{N} M_i \, \delta_{idn}$

M_n = Effective mass = $\sum_{d=d_1}^{d_3} \sum_{i=1}^{N} M_i \, \delta_{idn}^2$

PF_{nd} = Participation factor for the n^th mode and d^th direction

= R_{nd}/M_n

The instantaneous modal inertia force for mass point i, direction d, and mode n is given by:

$F_{idn}(t) = M_i \, \delta_{idn} \, [PF_{nd} \, SA_{nd}(t)]$

The maximum value is given by:

$F^*_{idn} = M_i \, \delta_{idn} \, (PF_{nd} \overline{SA}_{nd})$

d) Dynamic Modes - The modes are divided into two groups: the lower modes and higher "rigid" modes. The rigid modes are those whose natural frequencies lie outside the range where the support movement has significant energy. For earthquakes, this corresponds to frequencies above 33 Hz. Dynamic response analysis includes all modes below 33 Hz, however, additional calculations are made to account for all the rigid modes combined. The total solution includes the contributions of non-rigid as well as rigid modes added in the square root of the sum of the squares manner.

e) Stress and Displacement Analysis - The modal inertia forces F_{idn} are utilized as response loads in a static analysis to generate the maximum modal internal forces F^*_{idn}, moments M^*_{idn}, and displacement Δ^*_{idn}. The maximum responses such as forces, moments, and stresses due to earthquake disturbance in each direction are obtained separately by combining the modal responses. The method of combining modal responses is described in Section 3.7.3.7.

f) Three Dimensional Earthquake - The method of obtaining the maximum structural response due to the simultaneous action of earthquake motion in three orthogonal directions (two horizontal and vertical) is discussed in Section 3.7.3.6.

3.7.3.1.1.2 NSSS Vendor Scope

Seismic analysis methods for subsystems within Westinghouse's scope of responsibility are given in Section 3.7.2.1B and 3.7.3.5.

3.7.3.1.2 Modal Time-History Analysis Method

The method of modal time-history analysis is discussed in Section 3.7.2.1A.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Balance of Plant Scope

Ten equivalent maximum stress cycles per earthquake were considered for fatigue analysis of the subsystem according to applicable code criteria. One Safe Shutdown Earthquake (SSE) and five Operating Basis Earthquakes (OBE) were considered.

3.7.3.2.2 NSSS Vendor Scope

Where fatigue analyses of mechanical systems and components are required, Westinghouse specifies in the equipment specification the number of cycles of the operating basis earthquake (OBE) to be considered. The number of cycles for NSSS components is given in Table 3.9.1-1.

3.7.3.3 Procedures Used for Modeling

3.7.3.3.1 Balance of Plant Scope

The mathematical model used in all seismic subsystems (see Section 3.7.3.1.1) included sufficient mass points and corresponding degrees of freedom to provide a three dimensional representation of the dynamic characteristics of the subsystems.

The flexibility of platform structures was considered in the dynamic modeling for any subsystem mounted from platforms.

3.7.3.3.2 NSSS Vendor Scope

Refer to Section 3.7.2.1B for modeling procedures for subsystems in Westinghouse's scope of responsibility.

3.7.3.4 Bases for Selection of Frequencies

3.7.3.4.1 Balance of Plant Scope

Where feasible and practical, subsystems were designed to avoid the resonant frequency region of the supporting structure. Shifting of the subsystems away from the resonant region was achieved by modifying mass-stiffness characteristics.

Because of practical limitations, subsystems were, in some cases, designed in such a way that the frequencies fell into the resonant region of the supporting system. The amplified seismic response of the subsystem was then evaluated by proper consideration of total modal contribution from all modes within the frequency range of 1 to 33 Hz as a minimum. In some cases, the modes with frequencies higher than 33 Hz were also included. See Section 3.7.3.1.

3.7.3.4.2 NSSS Vendor Scope

There is no specific design criterion which ensures that the fundamental frequencies of NSSS piping and equipment are different from the forcing frequencies of the supporting structures. The effect of the equipment fundamental frequencies relative to the supporting structure forcing frequencies is, however, considered in the analysis of the NSSS piping and equipment.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

3.7.3.5.1 Balance of Plant Scope

The equivalent static load method was used to evaluate the seismic responses (member stresses and embedment loading) of applicable subsystems. This method was not used for piping, which is discussed in Section 3.7.3.8.

The equivalent static seismic acceleration coefficients which were used for the static analysis of the subsystem were obtained from the dynamic seismic analysis performed on the three dimensional model of the subsystem.

Design criteria specified the required range of support frequencies. Using the range of frequencies, the equivalent static seismic acceleration coefficients were calculated from the dynamic analysis of each representative subsystem.

The coupling effect from the multiple spans of a subsystem and the response contributions from all significant modes of a flexible subsystem were considered in the dynamic analysis. The equivalent static seismic coefficients included effects of such contributions.

3.7.3.5.2 NSSS Vendor Scope

The static load equivalent or static analysis method involves multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as single degree of freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi degree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50 percent to account conservatively for the increased modal participation.

3.7.3.6 Three Components of Earthquake Motion

Refer to Section 3.7.2.6A for a discussion on the procedure to account for the three components of earthquake motion for both BOP and NSSS vendor scope of supply.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Balance of Plant Scope

Modal responses were combined in the square root of the sum of the squares manner except for the responses of the closely spaced modes which were combined by the summation of the absolute values method. The latter were then combined with the responses of the remaining significant modes by the square root of the sum of the squares method. Closely spaced modes were ascertained utilizing the criterion of Regulatory Guide 1.92.

3.7.3.7.2 NSSS Vendor Scope

Methods used to combine modal responses for subsystems in Westinghouse's scope of responsibility are given in Section 3.7.2.7B.

3.7.3.8 Analytical Procedures for Piping

3.7.3.8.1 Balance of Plant Scope

The stress analysis of Seismic Category I, ASME, Section III, Safety Class 2 and 3 piping is in accordance with ASME B & PV Code, Section III, subarticles NC/ND, and is described below. The design criteria was in accordance with formulations given in subarticle NC/ND 3600.

The seismic analytical procedure using the computer methods described in Section 3.7.3.8.1.1 involves an analysis of the piping systems using characteristic spring rates of various rigid constraints and snubbers. Restraint loads based on this analysis were used to design particular controls (i.e., rigid restraints or snubbers). The final design analysis is based on characteristic spring rates.

Equipment having frequencies 33 Hz or higher was assumed rigid for the purpose of analyzing the connected piping. Where the frequency search of equipment indicated a frequency less than 33 Hz, the equipment was considered nonrigid. In such a case, the equipment vendor was asked to provide a dynamic model of the equipment having the same response in two orthogonal horizontal directions and the vertical direction. This dynamic model of the equipment was included in the stress analysis of the piping.

Welded attachments were avoided to the degree practicable. However, where integral attachments could not be avoided, local stress generated in the pipe due to their presence were considered.

Design criteria was based on formulations and allowable stress limits given in ASME Section III, subarticle NC/ND-3600, with load combinations which consider OBE and SSE effects along with other coincident loading conditions as delineated in design specifications.

For the seismic design of piping, the two orthogonal horizontal and vertical loadings were obtained from the floor response spectra that were generated for the appropriate structures and elevations with damping factors given in Table 3.7.1 1 of Section 3.7.1. The floor response spectra were based upon a dynamic seismic analysis method that considered the foundation torsion, rocking, and translation in the structural model as described in Section 3.7.2.

The effects of parameter variations on the floor response spectra were considered as discussed in Section 3.7.2.9.

An alternative method for broadening of the structured peaks can be based on a probabilistic approach. In the particular case where there is more than one piping frequency located within the frequency range of a widened spectrum peak that is associated with a structural frequency, the floor spectrum curve may be more realistically applied in accordance with the following criterion. Based on the fact that the actual natural frequency of the structure can possibly assume only one single value within the frequency range defined by $f_j \pm \Delta f_j$, but not a range of values, only one of these piping modes can respond with a magnitude indicated by the peak spectral value.

Therefore seismic analysis of piping systems using the broadened floor design response spectra may be accomplished by applying the method of peak shifting as described in the Summer 1984 Addendum of ASME Section III, Appendix N, paragraph N 1226.3(d).

Where more than one single floor response spectra were applicable, the analysis was done according to the criteria in Section 3.7.3.9. Seismic analyses considered three component motion as discussed in Section 3.7.3.6 and all dynamic modes were combined as discussed in Section 3.7.3.7.

Relative displacements within buildings and between buildings due to the seismic response were considered as described in Section 3.7.3.1.

All Seismic Category I, Safety Class 2 and 3 piping systems were seismically analyzed utilizing the methods in Sections 3.7.3.8.1.1 or 3.7.3.8.1.2. Piping 2 1/2 in. nominal size or larger with design temperature above 275 F, were analyzed by the computer method described in Section 3.7.3.8.1.1. All other piping subsystems were analyzed by either the computer method described in Section 3.7.3.8.1.1 or by the simplified method described in Section 3.7.3.8.1.2.

3.7.3.8.1.1 Computer Method

The computer method involves the use of computer programs to analyze the piping using the modal response spectra method described in Section 3.7.3.1.1 or the frequency based static method described below. When the computer method is used, systems are normally analyzed by the modal response spectra method. If the first mode period of the piping was 70 percent or less than the period corresponding to the peak of the applicable response spectrum, then the frequency based static method was used. If the applicable response spectrum shows more than one meaningful peak, the peak with lowest period was used.

In the frequency based static method, a static analysis is directly made using an acceleration value of 1.5 times the maximum value of the floor response spectrum in the period range equal to or less than the period of the piping.

The computer program used for the static analysis utilizes the same stiffness matrix method as that described for the dynamic analysis. The program automatically determines forces, moments, and deflections in the three coordinate directions in the piping system.

To justify this procedure for seismic analysis of piping, three sample problems (Figures 3.7.3-2 through 3.7.3-5) are presented using both modal response spectra and frequency based static methods. The frequency based static method used 1.5g in each of the three orthogonal directions. The modal response spectra method utilized 18, 14, and 16 modes for sample problems 1, 2, and 3, respectively. In each case, the analysis included one mode higher than the number of modes required to reach 33 Hz. For all modes, the acceleration in two horizontal and one vertical direction was taken as 1.0g. The periods for the analyzed modes for all sample problems were between 0.19 seconds and 0.02 seconds.

In all cases, the maximum computed stress was higher for the frequency based static analysis than for the modal response spectra analysis. Table 3.7.3-1 shows the computed stress values at the point of maximum stress for each problem and method.

3.7.3.8.1.2 Simplified Method

The simplified method of analysis consists of locating restraints such that the period of the first mode of vibration will not exceed 70 percent of the period corresponding to the peak of the applicable floor response spectra, as defined in Section 3.7.3.8.1.1. This method involves the use of appropriate and comprehensive charts and tabulations that include correction factors for the effects of concentrated loads, branch connections, changes in pipe size, changes in direction, offsets and various combinations of these effects. The piping system is studied for loading effects in each of the three coordinate directions to assure that it is adequately restrained in all directions. An additional analysis is performed to evaluate the thermal effects of the restraints on the system. This is done by means of charts that define the minimum distance required for placing restraints adjacent to any expanding leg in order to stay within allowable stress limits.

In analyzing a piping system by the simplified method, the first step is to make a computer run, using a program that was written for this specific purpose, that calculates the maximum allowable span for any preset period of vibration. Along with the period of vibration, the g factors in the horizontal and vertical directions are specified and the output is the maximum span, the bending moments, and the forces in both directions for all pipe sizes and schedules, empty or filled with water, with or without insulation. Some conservatism is included in the program since the span is based on pinned end conditions which is a more conservative approach than a continuous beam calculation, as consecutive uniform spans are avoided. Also the program internally multiplies the calculated span by 0.9 as an added safety factor.

The calculated span applied only to uniform, straight runs of piping, and is the basic span length to which appropriate correction factors are applied for determining the allowable span for various configurations of piping.

Figure 3.7.3-6 shows various pipe configurations for which correction factors have been calculated, including the effects of changes in pipe size, concentrated loads, elbows, offsets, loops, and branch connections. Correction factors have been derived for combinations of loops and offsets, with and without concentrated loads along the pipe runs, and include the flexibility factors of elbows.

The vibrational effects of concentrated loads, elbows, and various pipe shapes were investigated and exact formulae for each of these effects were derived and checked by computer analysis. These formulae were simplified by inserting conservative correction factors, again verified by computer analysis for determining the adjusted span length for the various pipe shapes.

In applying the simplified method, the appropriate floor response spectrum is examined to determine the maximum period of vibration and acceleration values for which the piping is to be designed. The computer program uses this information to calculate the forces, stresses, and the basic allowable span. The analyst then studies an isometricsketch and tentatively locates restraints along the pipe run, based on the correction factor criteria, for loadings in the three directions. An evaluation is then made of the thermal effects caused by the restraints. This is done by means of thermal charts made for this specific purpose that give forces and stresses in the pipe run, based on the expansion between restraints, and for all pipe sizes and schedules.

3.7.3.8.2 NSSS Vendor Scope

The Class I piping systems were analyzed to the rules of the ASME Code, Section III. When response spectrum methods were used to evaluate piping systems supported at different elevations, the following procedures were used. The effect of differential seismic movement of piping supports was included in the piping analysis according to the rules of the ASME Code, Section III. According to ASME definitions, these displacements cause secondary stresses in the piping system.

In the response spectrum dynamic analysis for evaluation of piping systems supported at different elevations, spectra which envelope the floor response spectra corresponding to the applicable support locations were used. Westinghouse does not have in their scope of analysis any piping systems interconnected between buildings.

3.7.3.8.3 Buried Seismic Category I Piping

The analysis of buried Seismic Category I piping is described in Section 3.7.3.12.

3.7.3.9 Multiple Supported Equipment Components with Distinct Inputs

3.7.3.9.1 Balance of Plant Scope

Where the location of the subsystem is such that more than a single floor response spectrum was applicable, the enveloped response spectrum was used.

3.7.3.9.2 NSSS Vendor Scope

When response spectrum methods were used to evaluate reactor coolant system primary components interconnected between floors, the procedure described in the following paragraphs was used. The primary components of the Reactor Coolant System are supported at no more than two floor elevations.

A dynamic response spectrum analysis was first made assuming no relative displacement between support points. The response spectra used in this analysis is the most severe floor response spectra.

Per ASME Code rules, the stress caused by differential seismic motion is clearly secondary for piping (NB-3650). For component supports differential seismic motions are considered as primary stresses. For components, the differential motion was 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, causes stresses which were evaluated by 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, were combined absolutely with due consideration for the ASME classification of the stresses.

3.7.3.10 Use of Constant Vertical Static Factors

3.7.3.10.1 Balance of Plant Scope

A single constant seismic vertical load factor was not used for the seismic design of seismic subsystems. The vertical load factor was determined from the analysis.

3.7.3.10.2 NSSS Vendor Scope

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

3.7.3.11 Torsional Effects of Eccentric Masses

3.7.3.11.1 Balance of Plant Scope

Torsional effects of all valves and other significant eccentric masses were included in the analysis of all Seismic Category I piping systems by taking into account the mass and eccentricity in the mathematical model.

3.7.3.11.2 NSSS Vendor Scope

The effect of eccentric masses, such as valves and valve operators, was 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 met the limits of the criteria applicable to the safety class of the piping.

3.7.3.12 Buried Seismic Category I Piping System, Electrical Conduits, and Tunnels

The location of all Seismic Category I pipes and conduits in the yard, fill areas in the yard, and the backfill areas against plant buildings are shown on Figure 3.7.3-7.

The fill in the yard for support of Seismic Category I pipes and conduits was placed from 1974 onwards and was classified into five different types according to the gradation of the material and method of placement and control. A description of the five types of random fill is below.

Type I: Overburden material with 90 percent passing a 3/4 in. screen, was compacted by a sheepsfoot roller in 8 in. layers to at least 95 percent of maximum standard proctor density, and controlled by field density tests.

Type 2: Overburden material with less than 90 percent passing a 3/4 in. screen, was compacted by six passes of a sheepsfoot roller in 12 in. layers. The method was developed by in-place test fill sections to achieve at least 95 percent of maximum standard proctor density.

Type 3: Overburden material, with 90 percent passing a 3/4 in. screen, was compacted by eight passes of a sheepsfoot roller in 12 in. layers. The method was developed by in-place test fill sections to achieve at least 95 percent of maximum standard proctor density.

Type 4: Ripped rock material, with less than 90 percent passing a 3/4 in. screen, was compacted by eight passes of a sheepsfoot roller in 12 in. layers. The method was developed by in-place test fill sections to achieve at least 95 percent of maximum standard proctor density.

Type 5: Blasted rock material was placed in 24 in. lifts and compacted by six passes of a vibratory roller. The method was developed by a test fill section.

Random Fill Types 1, 2, 3, and selected backfill materials are fine materials obtained from excavations of residual soils (overburden material) from the plant area or its vicinity. This material is similar to Material "Z" discussed in Appendix 2.5D. Figure 2.5D-54 shows Shear Moduli at Mean Normal Effective Stress of 1000 psf obtained from laboratory tests conducted on Material "Z". Laboratory tests (consolidation test) were performed on residual soils to determine Time-Settlement curves shown on Figure 3.7.3-8.

Random Fill Types 4 and 5 are similar to the random rockfill used for the Auxiliary Dam and Separating Dike. No dynamic laboratory testing was done for random fill Types 4 and 5. However, published data, as discussed in Section 2.5D.14.3.2 was utilized to determine material properties for design.

For the analysis of Seismic Category I pipes and conduits buried in fill in the yard the following properties were used:

Unit Weight pcf	135
Soil Subgrade Modulus (lb./in.2/in.)	50
Pressure Wave Velocity (fps)	1,500

Field tests to determine the moisture content and density of the fill were performed on the five types of random fill prior to construction in order to develop the construction procedure and also during actual fill placement. Average values of the test results are given below:

Type 1:	Brown clayey silt with yellow silty clay:	
	Dry Density	=114 pcf
	Moisture Content	=10 percent
	Compaction	=97 percent maximum standard proctor density
Type 2:	Brown clayey (sandy) silt with pieces of siltstone and some sandstone:	
	Dry Density	=125 pcf
	Moisture Content	=7 percent
	Compaction	=98 percent maximum standard proctor density
Type 3:	Brown clayey silt with yellow silty clay:	
	Dry Density	=117 pcf
	Moisture Content	=13 percent
	Compaction	=99 percent maximum standard proctor density
Type 4:	Brown clayey silt with pieces of siltstone (rock sizes up to 10 in):	
	Dry Density	=127 pcf
	Moisture Content	=7.5 percent
	Compaction	=98 percent maximum standard proctor density
Type 5:	Brown siltstone (rock sizes up to 21 in.):	
	Dry Density	=136 pcf from test fill section
	Moisture Content	=4.5 percent

A profile along the 30 in. service water line in the yard from the Emergency Service Water Intake Structure to the Tank Building, which is also representative of the electrical conduits running parallel to the service water line, is shown on Figure 3.7.3-9 through 3.7.3-11. The type of fill and the year of placement is indicated on the figure. A section near the Tank Building, indicating concrete backfill under the 30 in. service water pipe line, is shown on Figure 3.7.3-7.

The natural topography of the area had gradual changes in the ground elevations, which is evident from the profile along the 30 in. service water pipe line. The area was graded to a nominal elevation 260 ft. in 1974 by filling the low areas with the materials excavated from the plant area. The pipe line and the electrical conduit was placed in trenches excavated to the required elevations. There were no abrupt changes in the depth of fill. Therefore, the pipe and the conduit will bend and follow local settlement, if any.

Since the yard fill was placed and compacted in 1974-75, the maximum anticipated settlement of all Seismic Category I piping and conduits buried in the yard area is less than 1/8 in. This settlement was calculated using a one dimensional consolidation theory and the consolidation test data shown on Figure 3.7.3 8.

The following assumptions were made for the settlement calculation:

a) Plant grade at Elevation of 260 ft.

b) Top of rock at Elevation of 225 ft. (the deepest cut section provides a maximum fill section of 35 ft.)

c) Settlement caused by the weight of fill

d) All fill conservatively estimated to be selected backfill material similar to Type I material. A typical consolidation test for this material was used.

As an extreme case, the total settlement computed for a 35 ft. fill section of Type I material under its own weight was approximately 3 3/4 in. This represented the maximum settlement that would occur. It was determined by a time settlement analysis that 97 percent of this maximum settlement, or 3 5/8 in., would occur within 1 1/2 years after fill placement. Since the fill in these areas had been in place for nearly four years, a prudent and conservative settlement of 1/8 in. was considered.

The entire fill section was assumed to consist of Type 1 material, but in reality, much of the fill, particularly in the deep cut sections, consists of Type 3 and Type 4 materials. The Type 4 material consists of ripped rock material that is well graded and of high density and low voids. Type 4 material is not prone to consolidation related settlement, and any settlement that was experienced in Type 4 material occurred nearly instantaneously. This material is of such high density and its gradation is compatible with the finer grained material in the fill (Type 1, 2, & 3) that the possibility of any non-settlement related fill movement was precluded. The Type 3 material was compacted to 99 percent of maximum standard proctor density.

An abrupt change in the depth of backfill occurred near the building where the lines leave or enter such buildings. To avoid any settlement and restriction on the construction activities around the plant buildings, the excavated area under the 30 in. and 8 in. service water pipe lines between the Tank Building and Turbine Building walls and the rock or natural ground at the locations shown on Figure 3.7.3-7 was backfilled with 2000 psi concrete. The concrete fill was separated from the building wall by a 2 in. thick styrofoam board. The excavated area under the 30 in. service water pipe lines adjacent to the Emergency Service Water and Cooling Tower Make-Up Intake Structure was backfilled with crushed rock to avoid settlement.

The backfill under the 8 in. service water piping at other locations, other small diameter piping, and under electrical conduits was selected impervious material. The backfill was placed as the construction of the building or structure progressed and was controlled by density, moisture content, and permeability tests. The backfill was compacted to at least 95 percent of maximum standard proctor density.

Time-settlement calculations for selected impervious materials (which are residual soils) at building line interfaces indicated that most of the settlement in this type of material would occur during the first nine months after placement. The backfill near the building was placed to grade level and then a trench was excavated more than a year later to complete the installation of the pipe line.

Calculations similar to those described above for Seismic Category I systems in random fill were performed where the systems enter selected fill adjacent to the plant buildings. The assumptions were that plant grade is at Elevation 260 and the top of the concrete mat is at Elevation 236 ft. It was also assumed that settlement was caused by the weight of the fill.

The total settlement of the selected fill in this area would be 1 1/2 in. with 97 percent of this occurring in the first nine months after fill placement. This would result in approximately 1/16 in. of settlement remaining nine months after fill placement.

Ebasco's design procedure for seismic analysis of Seismic Category I buried piping was based upon Newmark's method (Reference 3.7.3-1) and Hetenyi's theory in beams on elastic foundations (Reference 3.7.3-2). The analysis procedure included calculation of stresses in the buried portion of the piping due to loads acting on the nonburied portion of the piping inside the building (interaction effect), superimposed on the stresses due to various loads acting on the buried portion of the piping. The resultant stresses were within allowable stress criteria based on the applicable ASME Section III Code.

The buried piping in the yard was analyzed using the above procedure and the soil properties stated above. It was assumed that the piping would be distorted in the same fashion as the earth and, therefore, would assume a sinusoidal wave shape. The wave length and maximum displacement were calculated and the bending moment and stress effects on the piping were obtained. Settlement in the fill along the piping due to differential depth of backfill did not cause any significant stresses in the piping and the resultant stresses were still within allowable stress limits.

At points where piping leaves the ground and is attached to structures, the maximum possible differential movement between the ground and the structure was determined. The differential movement was absorbed either by providing sufficient flexibility in the piping from the ground to the structure or by the use of flexible joints in the piping such as ball joints.

In certain instances, piping which enters structures is supported or anchored within the structure and not at the wall penetration. Wall penetrations were sized to provide sufficient room for differential pipe movement. Flexible membranes provided a moisture seal between the pipe and the structure wall.

The excavated area under the 30 in. and 8 in. service water pipe lines between the Tank Building and Turbine Building walls and the rock or natural ground was backfilled with concrete which will have insignificant differential settlement.

Seismic Category I electrical conduits in the yard were also analyzed by Newmark's method. The electrical conduits and electrical manholes were both buried in fill or backfill. Both of them would move with the fill with no local differential settlement to cause shear in the conduit. Moreover, the ends of conduits are not anchored in the wall of manholes and pass through sleeves with elastic boots which permit free movement of the conduits in any direction. Settlement in the fill or backfill along the conduit due to differential depth of fill did not cause any significant stresses in the conduit and the resultant stresses were well within the allowable stress limits.

The sleeves at the electrical manhole will permit rotation of the conduit end due to differential settlement of the manhole and the adjacent soils, if any.

The fill in the yard area supporting Seismic Category I piping and conduits is not subject to liquefaction during a seismic event.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

Non-Seismic Category I piping systems in close proximity to Seismic Category I piping systems (not attached to Seismic Category I piping systems) are constrained, physically separated by barriers, remotely located, or evaluated to assure that failure of any non-Seismic Category I piping system would not cause failure of Seismic Category I structures, piping, or equipment essential for safe shutdown.

If it was not feasible or practical to isolate the Seismic Category I piping, the adjacent non-Seismic Category I piping was seismically designed to assure that faulted limits were met during an earthquake of SSE intensity.

In the case of non-Seismic Category I piping systems attached to Seismic Category I piping systems, the dynamic effects were included in the modeling of the Seismic Category I piping up to the first anchor or system of restraints which decouples the piping.

For the anchor or system of restraints separating the Category I piping from the non-Category I piping, the seismic loads are either calculated based on the actual piping and support configuration or taken as twice the load from the Category I portion plus the load from the non-Category I portion.

It should be noted that, except for Main Steam and Feedwater Systems, all seismic/non-seismic interface restraints are located in seismically analyzed structures thereby assuring that collapse of the restraint structure will not occur. Restraints for the Main Steam and Feedwater Systems are discussed in Section 3.6.2.1.2. Actual loads transmitted to the interface restraint from the non-safety piping are limited by the following practical considerations:

a) Small size lines are typically supported by use of U-bolts which limit the vertical and lateral excitation of the piping.

b) Large size lines are supported based on span criteria which limit the ertical excitations that are generally predominate. These lines exhibit large structural damping and the piping runs are relatively short between anchor points, e.g., sleeves, penetrations, or equipment nozzles.

Several anchors of various pipe sizes and materials were analyzed and found to maintain structural integrity when the piping reached its yield strength.

Seismic/non-seismic anchors on small bore piping, here defined as piping of diameter up to but excluding 6 inches, are capable of accommodating loads resulting from either portions of piping by reason of the manner in which the non-seismic portion of the piping is supported, i.e., both vertically and horizontally per ANSI B31.1 spacing. This spacing has been shown to result in acceptable result reactions and acceptable load on the anchors. Large bore piping which is supported both horizontally and vertically per ANSI B31.1 will also transmit seismic loads from the non-seismic portion which are acceptable to the anchor.

3.7.3.14 Seismic Analysis for Reactor Internals

Fuel assembly component stresses induced by horizontal seismic disturbances were 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, was used as the seismic input. The reactor internals and the fuel assemblies were modeled as spring and lumped mass systems or beam elements. The component seismic response of the fuel assemblies was analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies provided by Westinghouse is contained in Reference 3.7.3-3.

Fuel assembly lateral structural damping obtained experimentally is presented in Reference 3.7.3-3, Figure B-4. The distribution of fuel assembly amplitudes decreases as one approaches the center of the core. Fuel assembly displacement time history for the SSE seismic input is illustrated in Reference 3.7.3-3, Figure 2-3.

For SGR/PUR conditions, Westinghouse performed analyses to evaluate the effect on the reactor pressure vessel system and reactor internal components as described in Reference 3.7.3-4. These analyses show that the non-fuel related reactor pressure vessel system and reactor internal components design criteria continue to be met for SGR/PUR operations. Based on these analysis results, Siemens performed evaluations and determined that the analysis of record continues to be bounding as described in Reference 3.7.3-5. These analyses show that Siemens fuel assembly mechanical design criteria continue to be met for SGR/PUR operations.

The CRDMs were seismically analyzed to confirm that system stresses under the combined loading conditions, as described in Section 3.9.1, do not exceed allowable levels as defined by the ASME Code, Section III. The CRDM was mathematically modeled as a system of lumped and distributed masses. The model was analyzed under appropriate seismic excitation and the resultant seismic bending moments along the length of the CRDM are calculated. The corresponding stresses were combined with the stresses from the other loadings required and the combination was shown to meet ASME Code, Section III requirements.

3.7.3.15 Analysis Procedure for Damping

3.7.3.15.1 Balance of Plant Scope

The procedure is discussed in Section 3.7.2.15A.

3.7.3.15.2 NSSS Vendor Scope

Analysis procedures for damping for subsystems in Westinghouse's scope of responsibility are given in Section 3.7.2.1B.

3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison with Regulatory Guide 1.12

The seismic instrumentation program for the Shearon Harris Nuclear Power Plant is designed to meet the guidance specified by Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 1, April 1974. A comprehensive seismic instrumentation program is provided to record any seismic disturbances at the site. The discussions and descriptions below provide Duke Energy's interpretations and clarifications to the regulatory position of Regulatory Guide 1.12 as it pertains to a single-unit site.

Three triaxial time-history accelerometers (T/A's) are provided to interface with the digital time-history accelerograph system. One T/A (SE-*1SM-5200A) is located on the containment mat, while another T/A (SE-*1SM-5200B) is located at a higher elevation on the containment structure, as shown on Figure 1.2.2-3. These two T/As transmit signals to the triaxial time-history accelerograph digital recorder (D/TR) in the Control Room. The D/TR provides a record of frequency, amplitude, and phase relationship data in the event of a seismic disturbance. The third triaxial time-history accelerometer (T/A) (SE-*1SM-5200C) is located in the Diesel Fuel Oil Storage Tank Building foundation at Elevation 242.25 ft. MSL. This T/A will also transmit signals to a D/TR recorder in the Control Room during a seismic event.

The two containment T/As, located for easy access, maintenance, and inspection, are rigidly mounted on structures directly connected to the containment structure so that the accelerograph records movements corresponding to the containment structure movement.

The Control Room is provided with a comprehensive recording and playback (D/PB) system whereby the digital recordings are graphically displayed on a computer display mounted within the cabinet. A seismic switch (S/S) or the alarm panel within the seismic monitor cabinet will activate an alarm to alert the operator when the seismic event occurs.

Three triaxial peak acceleration recorders (P/A's) are located at the following locations in SHNPP Unit 1: one is located on the reactor coolant pipe connecting to the reactor vessel cold leg at Elevation 253.75 ft. MSL, another one is located on the steam generator 1A-SN pedestal at Elevation 238.0 ft. MSL, and a third one is located at the component cooling water pump pad, (Seismic Category I) outside the containment structure at Elevation 236.0 ft. MSL.

Three passive triaxial response spectrum recorders (TR/SR) are provided to permanently record the peak accelerations at 16 discrete frequencies. The active peak acceleration function (ATR/SR) is performed by a combination of the T/A, recorder and computer. Peak accelerations are recorded by the D/TR and the computer is programmed to alarm the main control room when a select number of predetermined acceleration limits have been exceeded at certain frequencies. Thus, the ATR/SR will initiate visual annunciation of peak acceleration in the Control Room.

3.7.4.2 Location and Description of Instrumentation

Based on the provisions outlined in Regulatory Guide 1.12, the selection of the seismic instrumentation is described in the following paragraphs.

Table 3.7.4-1 summarizes the locations for the Seismic Instrumentation System.

Triaxial Time-History Accelerograph System

The following are the components of the Triaxial Time-History Accelerograph System:

a) Triaxial time-history accelerometer (T/A):

Each T/A continuously senses triaxial acceleration greater than 0.01g.

b) Triaxial time - history accelerograph system computer:

The computer is programmed with one vertical and two horizontal triggers. The computer actuates the triaxial time-history accelerograph digital recorder upon accelerations exceeding a preset threshold of 0.01g and will continue to run until approximately ten seconds (adjustable) after the last acceleration above the threshold.

c) Triaxial time-history accelerograph digital recorder (D/TR):

When actuated by the computer within 100 msec of signal initiation, the D/TR records the damped response spectra inputs and interfaces with the computer and alarm panel to initiate control room annunciation when the OBE containment foundation design values have been exceeded at any of the frequencies monitored. The recorders are mounted in the Control Room.

d) Triaxial time-history accelerograph control unit:

The control unit is comprised of a keyboard, mouse, display and computer. It has provisions for inplace testing and calibration as a permanent part of the acquired record.

e) Triaxial time-history accelerograph digital recorder and playback (D/PB).

This D/PB is part of the control unit and has the capability to play back the recordings produced by the D/TR as digital graphs of seismic event data.

Triaxial Peak Accelerograph System

The triaxial peak acceleration recorder (P/A) is a self-contained passive device requiring no internal or external power or control connections. The P/A records triaxial peak accelerations. Each peak acceleration axis record is scratched permanently on the recording plate which is removed after the seismic disturbance for data reduction and evaluation.

Triaxial Response Spectrum Recorder System

The following are the components of the Triaxial Response Spectrum Recorder System:

a) Triaxial peak shock recorder (passive) TR/SR:

The TR/SR is a passive device, which requires no internal or external power. It records peak accelerations at a number of discrete frequencies.

After the seismic event, the data records from the TR/SR are used to develop a response shock spectrum.

b) Triaxial peak shock recorder (active) ATR/SR:

The ATR/SR function is to permanently record peak accelerations at a number of discrete frequencies. The ATR/SR function is performed by a combination of the T/A, recorder and computer software. Peak accelerations are recorded by the D/TR and the computer is programmed to alarm the main control room when a select number of predetermined acceleration limits have been exceeded at certain frequencies. After the seismic event, the data records from the ATR/SR are used to develop a response shock spectrum.

c) Triaxial peak shock annunciator TR/SA:

The TR/SA function is performed by a combination of the T/A, recorder and computer software located in the Control Room, visually annunciates when a triaxial acceleration limit has been exceeded at certain frequencies. The limit and frequency are based on the HNP response spectrum curves specified by HNP-D-0071.

Triaxial Seismic Switches

The following are the components of the Triaxial Seismic Switch System:

a) Triaxial seismic switch unit (S/S):

The S/S is of triaxial trigger configuration and provides control room annunciation whenever the OBE zero period acceleration on at least one axis has been exceeded.

3.7.4.3 Control Room Operator Notification

The seismic instrumentation/monitoring system panel is located in the Control Room. The panel contains the digital recorder, digital playback system, the computer, and power supply. All these are described in Section 3.7.4.2.

The design limit annunciation in the Control Room is used by the control room operator as a warning of the potential extent of the seismic event. At this point, the operator shall take the appropriate action commensurate with the requirements of 10 CFR 100 Appendix A.

3.7.4.4 Comparison of Measured and Predicted Responses

The plant operators are provided with a procedure and criteria to review the accelerations recorded in the Control Room.

Verification of the seismic design analyses for the Containment Building is made by comparing the measured seismic motion at the upper accelerograph location with that calculated from inputting the measured seismic motion at the top of the mat into the mathematical model used for design.

First, the time history records are digitized and corrected for time signal variations and baseline variations. The time history records from the triaxial sensors located at the foundation of the Containment Building are used to calculate response spectra at appropriate critical damping values. The response spectra thus obtained are compared with the design response spectra. Amplified response spectra are then calculated at the locations of the other sensors in the Reactor Auxiliary Building and Diesel Fuel Oil Storage Tank Building (DFOSTB) for comparison and correlation with the response spectra directly measured. Structural responses and amplified response spectra are calculated using time history records with the dynamic model for comparison with the original design and analysis parameters. This comparison permits evaluation of seismic effects on structures and equipment and forms the basis for remodeling, detailed analyses, and physical inspection.

Direct verification of the seismic responses of Seismic Category I systems and components is not intended; however, the measurement taken from the proposed instrumentation will provide sufficient information to verify the input used for design analyses of Seismic Category I systems and components.

REFERENCES: SECTION 3.7

3.7.1-1 Ruiz, P., Penzien, J., "Probabilistic Study of the Behavior of the Structures During Earthquakes" Report No. EERC 69-3, VC Berkeley (1969)

3.7.1-2 Tsai, N.C., "Spectrum Compatible Motions for Design Purposes" JEMD, ASCE 98 NO. EM 2(1972)

3.7.1-3 Damping Values of Nuclear Plant Components" WCAP 7921 AR (May 1974).

3.7.1-4 Gesinski, T.L., and Chiang, D., "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-coolant accident" WCAP 8236, December 1973 (Proprietary) and WCAP-8288, January 1974 (Nonproprietary).

3.7.1-5 Richard, F.E., Jr., Hall, J.R., and Woods, R.D., "Vibrations of Soils and Foundations," Prentice-Hall, Inc. (Englewood Cliffs, New Jersey, 1970), pp. 334-385.

3.7.1-6 Barkan, D. D., "Dynamics of Bases and Foundations," McGraw-Hill, 1962.

3.7.1-7 Newmark, Nathan M., Blume, John A., and Kapur, Kmwar K., "Design Response Spectra for Nuclear Power Plants," presented at ASCE National Structural Engineering Meeting, April 1973.

3.7.1-8 Newmark, N.M., and Rosenbleuth, E., "Fundamentals of Earthquake Engineering," Prentice Hall, 1971.

3.7.1-9 Terzaghi, Karl, "Evaluation of Coefficients of Subgrade Reaction," Geotechnique, Vol 5, 1955 pp 297-325.

3.7.1-10 Seed, H. Bolton, Idriss, I.M., "Soil Moduli and Damping Factors for Dynamic Response Analysis," Earthquake Engineering Center, University of California, Berkely, California Report No. EERC 70 10, December 1970.

3.7.1-11 Newmark, N.M., "Design Criteria for Nuclear Reactors Subjected to Earthquake Hazards," Urbana, Ill., May 25, 1967.

3.7.1-12 Whitman & Richart, "Design Procedures for Dynamic Loaded Foundations," Journal of Soil Mechanics and Foundation Division, ASCE, November 1967.

3.7.1-13 Isao Toriumi, "Model Analysis of a Seismic Design of a Nuclear Reactor Building," Nuclear Structural Engineering 2, 1965, pp 301 305.

3.7.1-14 Newmark, N.M., "Torsion in Symmetrical Buildings," Fourth World Conference on Earthquake Engineering, Santiago, Chile 1969.

3.7.1-15 Whiteman, R.V., "Soil Structure Interaction" Seismic Design for Nuclear Power Plants, M.I.T. Press, R.J. Hansen, Editor 1970, pp 245 269.

3.7.1-16 Lysmer, J., Seed, H.B., Udaka, T., Hwang, R.N., and Tsai, C.F., "Efficient Finite Element Analysis of Seismic Structure-Soil Structure Interaction" Second ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, New Orleans, Louisiana December 8-10, 1975.

3.7.1-17 Idriss, I.M., Seed, H.B., "Seismic Response of Horizontal Soil Layers" Proceedings of the ASCE, Journal of the Soil Mechanics and Foundation Division, SM-4, July 1968.

3.7.1-18 Seed, H.B., Lysmer, J., and Hwang, P., Soil Structure Interaction Analyses for Evaluating Seismic Response," Report No. EERC 74-6, University of California, Berkeley, 1974.

3.7.1-19 Seed, H.B., Whitman, R.V., "Design of Earth Retaining Structures for Dynamic Loads," ASCE Specialty Conference on Lateral Stresses in the Ground, and the Design of Earth-Retaining Structures, 1970, pp. 103 147.

3.7.2-1 Biggs, J.M., "Introduction to Structural Dynamics" McGraw Hill Co., 1964.

3.7.2-2 Norris, Chetal, "Structural Design for Dynamic Loads" McGraw Hill, 1959.

3.7.2-3 Nigam, Navin C. and Jennings, Paul C., "Digital Calculation of Response Spectrum for Strong-Motion Earthquake Records," National Science Foundation, June 1968.

3.7.2-4 REFERENCE DELETED

3.7.2-5 Newmark, N.M., "Seismic Design Criteria for Structures and Facilities, Trans-Alaska Pipeline System," Proceedings, U.S. National Conference on Earthquake Engineering Research Institute.

3.7.2-6 Lin, C. W., "How to Lump the Masses - A Guide to the Piping Seismic Analysis," ASME Paper 74-NE-7 presented at the Pressure Vessels and Piping Conference, Miami, Florida, June, 1974.

3.7.2-7 "Damping Values of Nuclear Plant Components," WCAP-7921-AR, May, 1974.

3.7.2-8 "Safety Analysis of the 8-Grid 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-coolant accident," WCAP-8236, Addendum 1 (Proprietary), March, 1974 and WCAP-8288, Addendum 1 (Non-Proprietary), April, 1974.

3.7.2-9 Thomas, T. H., et al., "Nuclear Reactors and Earthquakes," TID 7024, U. S. Atmoic Energy Commission, Washington, D. C., August, 1963.

3.7.2-10 K. D. Chiu and H. J. Kuo, Ebasco Services, Inc., "Modal Damping in Dynamic Analysis of Structures", Specialty Conference on Structural Design of Nuclear Facilities, Vol. II, 1973, Pp. 499-514.

3.7.2-11 Whitman, R. V., "Soil Structure Interaction," Seismic Design for Nuclear Power Plants, MIT press, Cambridge, Mass., 1970.

3.7.3-1 Newmark, N. M., "Earthquake Response Analysis of Reactor Structures," Nuclear Engineering and Design, Vol. 20, pp. 303 to/incl 322 (1972).

3.7.3-2 Hetenyi, M., "Beams on Elastic Foundations," Univ. of Michigan Press, Ann Arbor, Michigan, 1967.

3.7.3-3 "Safety Analysis of the 8-Grid 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-coolant accidents," WACP-8236, Addendum 1 (Proprietary), March, 1974 and WCAP-8288, Addendum 1 (Non-Proprietary), April, 1974. [Applicable to original fuel provided by Westinghouse.]

3.7.3-4 Harris Nuclear Plant, Steam Generator Replacement/Uprate Analysis and Licensing Project, NSSS Engineering Report, WCAP-14778 Revision 1, Westinghouse Propriety Class 2C, June 2000.

3.7.3-5 "Generic Mechanical Licensing Report for Advanced 17 x 17 Fuel Design", EMF-93-074 (A) & Supplement 1, June 1994.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAIMENT

3.8.1.1 Description of the Containment

3.8.1.1.1 General description

The Concrete Containment Structure (CCS) is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome and a flat base with a recess beneath the reactor vessel.

The structure, shown on Figure 3.8.1-1, consists of a cylindrical wall measuring 160 ft. in height from the liner on the base to the springline of the dome and has an inside diameter of 130 ft. The cylinder wall is 4 ft. 6 in. thick. The inside radius of the 2 ft. 6 in. thick dome is equal to that of the cylinder so that the discontinuity at the spring line due to the change in thickness is on the outer surface. The base mat consists of a 12 ft. thick structural concrete slab and a metal liner. The liner is welded to inserts embedded in the concrete slab. The base liner is covered with concrete, the top of which forms the floor of the containment. The base mat is supported by sound rock.

The basic structural elements considered in the design of the containment structure are the basemat, cylinder wall, and dome. These act essentially as one structure under all loading conditions. The nominal liner plate is 3/8 in. thick in the cylinder, 1/4 in. thick on the bottom, and 1/2 in. thick in the dome. The liner is anchored to the concrete shell by means of anchor studs fusion welded to the liner plate so that it forms an integral part of the containment structure. The liner functions primarily as a leak tight membrane. An impervious plastic waterproofing membrane is placed between the containment foundation mat and the ground. Before laying the membrane, a concrete leveling surface is placed on the rock. After installing the membrane, a concrete protective layer is installed before placing reinforcement for the foundation mat. The waterproofing membrane for the Containment Building is continuous under the containment foundation mat and terminates into waterstops at the joints with adjacent structures.

The arrangement of the Containment and the relationship and interaction of the shell with the interior compartment shielding walls and floors are shown on Figures 3.8.1-1, 3.8.3-1, and 3.8.3-2.

The containment wall is independent of adjacent interior and exterior structures; sufficient space is provided between the containment wall and adjacent structures to prevent contact under any combination of loading. The interior grating platforms and concrete slabs are supported on steel beams which span between the secondary shield wall and the containment wall. These beams are independently supported, near the containment wall, by steel columns resting on the concrete mat.

The circular polar crane runway girder is supported by a series of uniformly spaced steel plate brackets which extend from the inside face of the containment wall and are attached to the liner plate. The crane runway circle is not concentric with that of the Containment, but is offset to provide a passageway on one side of the pipe runs of the containment spray header piping mounted in the dome. The liner plate is thickened to one inch to support the brackets and is anchored to the concrete containment wall.

The Concrete Containment Structure and associated parts and appurtenances were originally designed for an expected operating life of 40 years.

Basically three materials - concrete, reinforcing steel, and steel liner plate, are used for construction of the Containment.

The concrete has a compressive strength of 4000 psi at 28 days after placement except at the bottom portion of the cylindrical wall and around major penetrations where the concrete has a compressive strength of 5000 psi at 28 days after placement. The reinforcing steel is new billet steel in accordance with ASTM A615 Grade 60. Where called for on the design drawings, weldable grade reinforcing steel in accordance with ASTM A706 was used.

The steel liner plate is carbon steel conforming to ASTM A 516 Grade 70. This steel has a minimum yield strength of 38,000 psi and a minimum ultimate tensile strength of 70,000 psi.

The Containment encloses the reactor pressure vessel, pressurizer, steam generators, reactor coolant pumps, and piping, and portions of the Engineered Safety Features Systems. The containment wall protects the Reactor Coolant System from site environmental conditions. It is designed as a Seismic Category I structure for earthquake, tornado, and external missile loading conditions. It also limits the release of radioactive fission products to the environment in the unlikely event of a loss-of-coolant accident (LOCA), and in addition, provides biological shielding for both normal and accident conditions. The functional requirements of the Containment are discussed in detail in Section 6.2.1.

The cylindrical section of the containment shell includes large openings for access hatchways and penetrations. The concrete wall is locally thickened and additional reinforcement is provided at these large penetrations. Penetrations are anchored in the containment wall.

A permanent steel ladder with a safety cage is provided on the exterior cylinder portion of the Containment Building for access to the bottom portion of the dome. Another ladder with a safety rail is provided on the exterior of the dome for access to the top. A guard rail is provided around the entire springline of the Containment. U-shaped steel bolts are embedded in the top and bottom of the dome to allow for the hanging of scaffolding to inspect the entire dome and cylinder portion of the Containment Building.

3.8.1.1.2 Foundation Mat

The foundation mat is a conventionally reinforced concrete mat of circular shape and 12 ft. uniform thickness. The top of the mat is 44 ft. below finished grade.

The entire mat is structurally independent of adjacent Seismic Category I foundations. The mat has a recess in the central portion to house the reactor pressure vessel, and in the engineered safety features (ESF) area, there is a recess to house the ESF system sumps for the containment spray header water which exits the Containment through two collection sumps and embedded drain pipes.

The foundation mat, inside the Containment and including the reactor cavity, is covered with 1/4 in. thick carbon steel liner plate, except at the connection with the wall liner plate, where a 3/8 in. (nominal) thick liner plate is provided. A five ft. thick concrete internal mat is provided over the liner for protection and support of internal primary and secondary shield walls.

In order to protect the mat liner plate against groundwater hydrostatic pressure, an impervious waterproofing membrane is placed continuously below the foundation mat and terminates into waterstops at the joints with adjacent structures. The seismic gaps between adjacent structures are cut off from groundwater by double rows of horizontal waterstops. As described in Section 3.4.1.1, any leakage through the waterproofing membrane will be drained through porous concrete drains placed between the membrane and the concrete mat.

The primary and secondary shield walls are supported by the internal foundation mat which in turn is resting on the external foundation mat. No anchorage of the interior structures through the liner plate and into the external mat is provided.

The reinforcing steel of the foundation mat, shown on Figure 3.8.1-2, consists of radial and circumferential reinforcement placed at the top and bottom of the mat. Radial bars have no splices; circumferential bars utilize the longest length possible so that the number of splices is minimized. Splices are staggered whenever practical. Shear reinforcement is provided whenever required by design. The base mat is considered a circular flat slab resting on an elastic foundation and the finite element approach was used for analysis. The mat is designed to withstand the loading defined in Section 3.8.1.3.

3.8.1.1.3 Cylindrical wall

3.8.1.1.3.1 Reinforcing steel arrangement

The reinforced concrete cylindrical wall is designed to withstand the loadings and stresses anticipated during the operating life of the plant, as defined in Section 3.8.1.3. The steel liner is attached to, and supported by, the concrete. The liner functions primarily as a gas-tight membrane and also transmits loads to the concrete. During construction, the steel liner serves as the inside form for the concrete wall and dome. The containment structure does not require the participation of the liner as a structural component.

Hoop tension in the cylindrical concrete wall is resisted by horizontal reinforcing bars near both the outer and inner surfaces of the wall.

Horizontal circumferential bars, including those in the dome, have their splices staggered wherever possible.

Longitudinal tension in the cylindrical wall is resisted by rows of vertical reinforcing bars placed near the interior and exterior faces of the wall, with cadweld splices staggered whenever practical.

Figure 3.8.1-3 shows typical reinforcing steel for the cylindrical wall.

Reinforcing steel which terminates in locations where biaxial tension is predicted, such as at penetrations, is anchored by hooks, bends, or positive mechanical anchorage in such a manner that the force in the terminated bar is adequately transferred to other reinforcement. Also, bar development length at such location is increased.

The main vertical and hoop reinforcing steel in the containment wall and dome have a concrete cover of 6 1/2 inches and 2 3/4 inches respectively. Concrete cover for reinforcing steel other than these are governed by provisions listed in the ASME/ACI 359 code.

The juncture of the cylinder to the base slab is considered to be rigidly connected. The cylinder at this point cannot expand but joint rotation is considered as the wall deforms under the internal pressure, temperature, and dead load conditions; hence, radial shear and moments are introduced into the cylinder wall. All the radial shears at the base of the cylinder wall are resisted by reinforcing steel. This shear reinforcing is horizontal.

The nonaxisymmetric loads, such as wind, tornado, and seismic excitations, induce tangential shears into the cylindrical concrete wall and concrete dome. Although the liner plate in the cylindrical wall and dome has shear capacity available to resist targential shear, no credit was taken for this capacity. The tangential shear carried by the concrete does not exceed 60 psi and 40 psi for abnormal load combinations associated with the safe shutdown earthquake and operating basis earthquake, respectively, as required by Standard Review Plan 3.8.1. The excess tangential shear is taken by diagonal seismic reinforcing bars. The seismic reinforcement, shown on Figures 3.8.1-4 and 3.8.1-5, extends diagonally into the dome until a point is reached where the concrete alone can resist the tangential shear. Sufficient overlap is made between the linear and diagonal reinforcing to allow transfer of shears. At the major penetrations, the seismic reinforcement is either bent around the penetration or is cut off, in which case a mechanical embedment, consisting of a cadweld sleeve welded to an anchorage, is provided.

The concrete thickness of the wall is increased from 4 ft. 6 in. to 6 ft. 6 in. around the major penetrations such as the equipment hatch, personnel lock, emergency air lock, main steam penetrations, and feedwater penetrations. In all of these areas, the main hoop and vertical reinforcement are bent around openings, hooked into the wall, or terminated using a mechanical embedment. Additional circular radial and shear reinforcement is provided to withstand stress concentrations and additional radial and in-plane shear developed in these areas by the loading combinations described in Section 3.8.1.3.

Figure 3.8.1-6 shows the reinforcement in the equipment hatch area of the containment structure.

Figures 3.8.1-7 and 3.8.1-8 show the reinforcement in the personnel air lock, emergency air lock, and HVAC penetrations areas. In all of these areas, the anchorage of the steel penetration into the concrete wall is provided by steel anchorages welded to the penetrations sleeves. For all penetration sleeves designed in accordance with requirements of the ASME Code Section III Division 1, such as the equipment hatch, personnel air lock, emergency air lock, and Type I penetration sleeves, special anchorages were provided using ASME Code material and manual welding. For all penetration sleeves designed in accordance with requirements of the ASME Code Section III Division 2, in the portion backed by concrete, such as Type II and Type III penetration sleeves, double headed machine welded Nelson Studs were provided.

Figure 3.8.1-9 shows the reinforcement in the main steam and feedwater penetration area. In addition to the main circumferential and vertical reinforcement bent around penetrations, additional circular reinforcement is provided around each individual penetration and radial interconnecting reinforcing bars. In order to provide for sufficient resistance against excessive rupture loads and to accommodate the interaction between the concrete structure and steel penetrations, the attachments of the penetration sleeves are directly connected with the radial reinforcing bars transferring the loads into the concrete wall.

Figure 3.8.1-10, Section P-P, shows the 6 in. attachments shop welded to the penetration sleeve. No. 18 radial reinforcing bars are connected through a cadweld mechanical connection to 9 in. attachments, which in turn are field welded to the 6 in. attachments connected to the sleeves.

The reinforcement arrangement around penetrations smaller than 18 in. is shown on Figure 3.8.1-11. Structural built-up steel members are provided to transfer the forces from the main circumferential and vertical reinforcing bars to special bars, closely spaced, or reinforcing bars were bent around openings. Additional inclined reinforcement is provided when required.

3.8.1.1.3.2 Liner plate

A continuous welded steel liner plate is provided on the entire inside face of the concrete containment cylindrical wall to limit the release of radioactive materials into the environment. The thickness of the liner in the cylindrical wall area is 3/8 in. nominal. A one inch thick liner plate is provided at the crane girder brackets elevation. Ring collars up to 2 in. thick are provided around all penetrations and shop welded to the penetration sleeves, as required by ASME Section III Division 2/ACI 359 Code, Section CC4552.2.1.

Figures 3.8.1-12 and 3.8.1-13 show liner plate details. An anchorage system, consisting of Nelson Studs 5/8 in. diameter by 4 in. long, is provided to prevent instability of the liner for all load combinations described in Section 3.8.1.3.

In order to minimize liner stresses, strains and deformations under the design loading condition described in Section 3.8.1.3, the cylindrical wall liner plate connection with the foundation mat lower plate is an unanchored embedded 90 degree free-standing welded connection. No anchor studs are provided on a 5 ft. vertical portion and on a 3 ft. horizontal portion of the liner plate. In order to allow free deformation of the liner plate during test pressure conditions, an inch of ethafoam is provided on the inside face of the liner plate facing the concrete of the internal mat. In order to allow vertical movement at the concrete connection during the same test pressure conditions, ethafoam is also provided against the back up plate and the end of the horizontal liner plate, as shown on detail X on Figure 3.8.1-12.

The one inch liner plate at the crane girder brackets area is anchored into the concrete wall with shear lugs, anchor bolts connected to embedded plates, special anchorages, and Nelson studs, as shown on detail Y and Section A-A on Figure 3.8.1-12, in order to withstand the complexity of loading induced during operation of the crane and/or seismically induced loads.

Figure 3.8.1-13 shows the arrangement of anchor studs around different types of containment penetrations.

Leak chase channels or angles are provided at the liner seams for leak tightness examination.

There are no through liner attachments. The supports for HVAC ducts, piping hangers, and ladders, are welded to the liner plate, which is locally reinforced with additional studs in the region of surface attachments.

A yield strength of 45.6 ksi (70°F-100°F) was used for the 3/8 inch thick plate. This yield strength is the basis for considering that, for both service and factored load conditions, the yield stress is not exceeded in the regions identified as overstressed (if plate yield stress is 38 ksi (70-100°F)). This is a conservative value and was obtained as follows:

a) All certified mill test reports for the 3/8 inch thick plate that was supplied were reviewed. The least yield stress value from all reports for that thickness plate is 45.6 ksi. This is the value that was used. It was reduced for higher temperatures (temperatures from 100°F to 240°F) by the application of ASME Section III Division I Appendix Table I-2.1 "Yield Strength Values Sy for Ferritic Steels", values for SA 516 Grade 70. Two straight line reductions in yield strength were obtained from the table, the first, for reduction in strength between 100°F and 200°F, and the second, for reduction in strength between 200°F and 300°F. The slopes of the two lines were expressed in terms of reduction in strength, ksi, per degree F temperature increase and applied to the 45.6 ksi least yield strength value to obtain reduced yield strength values for temperatures up to 240°F.

b) Reductions in modulus of elasticity for the material due to increase in temperature were also evaluated, based on ASME Section III Appendix I Table I-6.0 "Moduli of Elasticity E of Materials for Given Temperatures", and considered in the determination of strain at various temperatures.

c) The certified test reports of all the welding electrodes for the liner plate joining welds were also reviewed. The least value of yield was found to be 58.0 ksi. It was concluded that the electrodes supplied do not adversely affect the yield strength of the liner plates.

d) Verification of liner strains due to containment pressurization is obtained from the liner strains measurements made for the containment building structural integrity test. The test is described in Section 3.8.1.7.1. The liner strain gage locations are shown in Figures 3.8.1-47, 48, and 49.

3.8.1.1.3.3 Containment penetrations

Access into the Concrete Containment Structure is provided by an equipment hatch, a personnel air lock, and an emergency air lock.

The equipment hatch is a welded steel assembly having an inside diameter of 24 ft. 0 in. with a weld-on cover with sufficient material to initially allow for six removals and rewelding. Activities to remove the equipment hatch weld-on cover to provide for the replacement of the reactor vessel head and the steam generators have resulted in the weld-on cover being removed and rewelded twice, leaving four remaining possible removals and reweldings. A 15 ft. 0 in. inside diameter bolted cover is provided in the equipment hatch cover for passage of smaller equipment during plant operation. Provision is made to pressurize the space between the gaskets of the bolted hatch cover to meet the requirements of Appendix J of 10 CFR 50 as discussed in Section 6.2.6.
Figure 3.8.1-14 shows the equipment hatch.

The containment equipment hatch is provided with external missile protection as described in Table 3.5.1-1.

One breech type personnel air lock (Figure 3.8.1-15) and one personnel emergency air lock (Figure 3.8.1-16) are provided. Each lock is a welded steel assembly having two doors which are double-gasketed with material resistant to radiation. Provisions are made to pressurize the space between the gaskets. The doors of each lock are equipped with quick acting valves for equalizing the pressure across each door and the doors are not operable unless pressure is equalized. There is visual indication outside each door showing whether the opposite door is open or closed and whether its valve is open or closed. Provisions have been made outside each door for remotely closing the opposite door so that in the event that one door is accidently left open it can be closed by remote control. Interior lighting and communications systems were installed. These systems are not capable of operating from emergency power supply.

Two pressure gages are placed at each end of the personnel locks, one reads from outside the lock and measures lock pressure. The other reads from inside the lock and measures containment pressure. Nozzles are installed which permit pressure testing of the locks at any time.

The breech-type personnel air lock has a 9ft.-0in. inside diameter with full diameter breech doors to open outwardly from each end of the lock. Doors for the lock are hydraulically sealed and electrically interlocked. During plant shutdown, it will be necessary to open both doors at the same time; therefore, a means of defeating the interlock is provided. The controls to override the interlock are kept locked and under strict administrative control outside of Containment. The controls inside the air lock and inside Containment are also administratively controlled and are rarely accessible during periods when containment integrity is required. Opening of the doors after unsealing will be done with a hydraulic motor, as will closing before sealing. Manual (hand pump) operation of the sealing ring and door swing mechanism is provided in case of a power failure.

All leakage and pressure testing on the breech-type personnel air lock will be done without the use of the test clamps since sealing is accomplished by forcing the doors against the seals when the rotating third seal ring is rotated into the breech locked position. Since the pressure applied to the double seals of the lock during testing is exerted by the third ring, the effectiveness of the seal cannot be increased beyond that seen during operating or accident condition. Test connections are provided for continuous testing between the double seals of each door for leakage.

The personnel emergency air lock has an outside diameter of 5 ft. - 0 in. with a 2 ft. - 6 in. diameter door located at each end of the lock. The doors of the lock are in series and are mechanically interlocked to ensure that one door cannot be opened until the second door is sealed. Violation of the interlock can only be made by use of special tools and procedures under strict administrative control.

Test clamps are provided for leakage and pressure testing of the personnel emergency air lock. This set of clamps fits either door and is designed to withstand, as a minimum, the full peak containment internal pressure. Compression of the double seals on each of the doors is limited to that which occurs before a metal to metal seat is achieved between the door and the protruding metal flange adjacent to the seals on the lock barrel. The internal containment pressure (or pressure exerted by the test clamps) necessary to achieve the metal to metal seat is approximately 3 PSI over the surface of the door. Effectiveness of the seals during testing, therefore, cannot be artificially increased beyond that seen during operating or accident conditions by overtightening of the clamps. Mechanical and electrical penetrations are provided in the cylindrical wall of the containment structure to provide access for mechanical piping and electrical cables.

Mechanical penetrations are divided into two general types:

a) Type I - High pressure, high temperature piping (above 200°F).

b) Type II - General piping (penetrations which are subject to only relatively small pipe rupture forces and temperatures up to 200°F).

Type I mechanical piping hot penetrations are provided for high pressure and high temperature (above 200 F) lines which penetrate the concrete containment structure. The process pipe is connected to a containment penetration sleeve (which is partially embedded in the concrete wall) by a forged flued head fitting. The flued head fittings are designed to carry the forces and moments due to the normal operating conditions and due to the postulated pipe rupture loads by transferring these forces to the containment penetration sleeves and further into the concrete containment wall.

Figure 3.8.1-17 shows a Type I mechanical penetration.

Type II mechanical piping cold penetrations are provided for low temperature (below 200 F) lines which penetrate the concrete containment structure. As shown on Figure 3.8.1-18, the process pipe passes through a containment penetration sleeve which is partially embedded and anchored into the concrete wall. The annular gap between the process pipe and the sleeve is sealed on both the inside and outside faces of the concrete wall. The inside plate is designed to withstand the internal pressure and to transfer all of the normal operating loads and/or the postulated accident piping rupture loads from the piping system to the penetration sleeve and then into the concrete wall. The outside seal is flexible to accommodate thermal expansion movements.

Type II penetrations also include HVAC penetrations and groups of small diameter lines (instrument, sampling lines) which incorporate socket weld couplings welded to closure plates. Two categories of penetration are included in Type II penetration: Type IIA for single tubing or multiple pipes and/or tubings and Type IIB for single pipe.

Electrical penetrations are included within the Type III penetrations. Modular type penetrations are used for all electrical conductors passing through the containment wall. Each penetration assembly consists of a stainless steel header plate attached to a carbon steel welded ring which is in turn welded to the pipe sleeve. The header plate accepts either three or six modules depending on the penetration diameter and voltage classification. The modules are held in the header plates by means of retaining clamps. Each module is a hollow cylinder through which the conductors pass. The conductors are hermetically sealed into the module with an epoxy compound. Each module is provided with a pressure connection to allow pressurization for testing. Figure 3.8.1-19 shows typical electrical penetrations. The header plates are attached to penetration sleeves located in the wall of the containment vessel and welded to the containment liner. Sealing between the header plates and the sleeves is accomplished by welding. All materials used in the design are selected for compatibility with all possible environmental conditions during normal, accident, or post-accident periods. Spare electrical penetration sleeves are provided for possible future uses. Each penetration is sealed and tested at the factory for leakage. The only seals that need to be made in the field are the welds attaching the header plates to the sleeves.

HVAC penetration sleeves, 48 in. and 24 in. diameter, are similar to the mechanical Type II penetration sleeves.

A fuel transfer penetration is provided to transport fuel assemblies between the refueling cavity in the Containment and the fuel transfer canal in the Fuel Handling Building. This penetration consists of a 20 in. diameter stainless steel pipe installed inside a 26 in. pipe. The inner pipe acts as the transfer tube and is fitted with a double-gasketed blind flange in the refueling cavity and a standard gate valve in the fuel transfer canal. This arrangement prevents leakage through the transfer tube in the event of an accident.

The penetration sleeve is welded to the steel liner and anchored into the concrete wall.

Provision is made for testing welds essential to the integrity of the liner. Bellows expansion joints are provided to compensate for any differential movement between the structures, due to operating thermal expansion and seismic movements.

The fuel transfer tube expansion joints are not part of the containment pressure boundary. Rather the transfer tube is rigidly attached to the containment penetration sleeve. Two bellows type expansion joints are installed, the first forming a flexible joint between the transfer tube and the transfer canal inside the Containment; the second forming a flexible joint between the transfer tube and the Fuel Handling Building fuel transfer canal. Figure 3.8.1-20 shows the design of the fuel transfer tube.

The expansion joint inside the Containment is accessible for visual inspection at any time. The expansion joint in the Fuel Handling Building is also accessible for inspection at any time except when the transfer canal is flooded during the actual fuel transfer period.

Also included are four valve chambers and their appurtenances. The valve chambers and their appurtenances, shown on Figure 3.8.1-21, are 9 ft - 0 in. diameter by 19 ft. - 0 in. long airtight enclosures which function as a secondary containment boundary to completely enclose the containment sump lines and isolation valves.

3.8.1.1.4 Containment Dome

The containment dome is a lined reinforced concrete hemispherical dome of 2 ft. 6 in. uniform thickness. A continuous welded steel liner plate, one half inch thick, is provided on the inside face of the dome. The arrangement of the studs in the dome is shown on Figure 3.8.1-12. Nelson studs 5/8 in. diameter by 4 in. long are used to connect the liner to the concrete.

The reinforced concrete dome is designed to withstand the loads anticipated during the operating life of the plant and postulated accidents and events described in Section 3.8.1.3. Meridional and circumferential reinforcing bars are provided to resist the refueling tensile forces and bending moments.

Figures 3.8.1-22 and 3.8.1-23 show the arrangement of the reinforcement in the dome.

The dome reinforcement consists of layers of reinforcing steel placed meriodionally, extending from the vertical reinforcing of the cylindrical wall, and horizontal layers of circumferential bars. The layers are located near both the inner and outer faces of the concrete. The radial pattern of the meridional reinforcing steel, terminating in the containment dome, results in a high degree of redundancy of reinforcing steel in the dome. Bars are terminated beyond a point where there is more than twice the amount of steel required for design purposes. The rate of convergence of these bars, and the low stress requirements dictated by this arrangement, results in a satisfactory development length of the meridional reinforcing bars. Near the crown of the dome, the meridional reinforcing bars are welded to a steel hub plate, cast in the concrete, concentric with the dome centerline.

Although the liner plate is not considered as a structural element to sustain the loading imposed on the dome, during construction the liner plate is used as a form to withstand the weight of the reinforcing steel and fresh concrete during placement. To minimize the locked in stresses during construction, the placement of the concrete in the dome area is made in lifts of 4 to 5 ft. of concrete. The next placement of the concrete is added only when the concrete previously placed has enough strength to take additional construction loads.

Ventilation openings are provided at the top of the dome to be used during construction. These openings are filled with concrete when construction is finished.

3.8.1.1.5 Containment Structural Boundaries

The Containment is a composite steel and reinforced concrete assembly that is designed as an integral part of the containment's pressure-retaining barrier to retain and control the release of radioactive or hazardous effluents released from the nuclear power plant equipment which the containment encloses.

The design, materials, fabrication, construction, testing, examination, structural integrity test, and quality assurance for the Containment Building, consisting of a reinforced concrete mat, cylindrical wall and dome, lined with steel liner, and associated materials, parts, and appurtenances, are in accordance with ASME Code Section III Division 2/ACI 359 Code winter 75 addendum, with the exceptions listed in Appendix 3.8A.

All pressure-retaining, leak-resisting, and load-bearing concrete and steel portions of the Containment and all parts or appurtenances that act integrally with the pressure-retaining portion to carry the fluid pressure loads are covered by the ASME Code Section III Division 2/ACI 359 Code, except that:

a) Parts and appurtenances under the jurisdiction of Section III Division 1 are considered only with respect to their functional collaboration with the concrete and steel portions of the component in carrying loads.

b) Parts and appurtenances under the jurisdiction of Section III Division 1, whose directional loadings can be described by moments and forces acting on portions of the concrete component for design purposes, are characterized by such loading conditions which for the concrete containment can be shown to be functionally acceptable.

c) Parts and appurtenances specified to meet the requirements of Section III Division 1 and furnished before April 29, 1977, meet the requirements of Subsection NA of Division 1. Parts and appurtenances furnished after April 29, 1977, meet the requirements of Subsection NA of Division 2. The parts and appurtenances which are designed under the jurisdiction of Section III Division 1 are presented in Table 3.8.1-1.

   The boundaries of the Containment Building and the different parts and appurtenances are shown on Figures 3.8.1-24 and 3.8-1-25.

For the design of the Equipment Hatch, Personnel Air Lock, Emergency Air Lock, and all penetrations, at the transition portion from concrete to steel, the following aspects are considered:

a) Metal sections not backed by concrete meet the requirements of Division 1 and consider the concrete confinement except that proof testing is in accordance with CC-6000 of the ASME Code Section III, Division 2/ACI-359 Code.

b) Metal sections are attached to concrete sections by one of the following:

   1) Tension attachment to the primary reinforcement of the concrete containment.

   2) Anchorage system attached to the metal shell and extended into the concrete. The metal shell is not reduced below the minimum thickness required for primary

mechanical loads for a distance of 25 t from the point where the concrete-to-metal junction occurs, where t is the thickness of the metal penetration sleeve at the transition section.

Where the penetration sleeves or the liner is backed by compressible material to provide local flexibility, the penetration sleeves or the liner meet all requirements for material, design, fabrication, and examination of the ASME Code, Section III, Division 1 in the region where compressible material is present. Where penetration sleeves or liner are attached to concrete directly or to embedded members, only the requirements for liner apply.

3.8.1.2 Applicable Codes, Standards, and Specifications

The structural design, materials, fabrication, construction, testing, inservice surveillance, and quality assurance for the Containment conform to the codes, standards, regulations, and specifications listed below, except where specifically stated otherwise.

General Codes and Standards

OSHA Occupational Safety and Health Administration, Federal Safety Regulations (1975 listing)

North Carolina State Building Code, 1969 Edition

ACI American Concrete Institute Standards

211.1-1974 Recommended Practices for Selecting Proportions for Normal and Heavy Weight Concrete

301-1975 Specifications for Structural Concrete for Buildings

304-1973 Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete

305-1972* Recommended Practice for Hot Weather Concreting

306-1966 Recommended Practice for Cold Weather Concreting

309-1974 Recommended Practice for Consolidation of Concrete

315-1974 Manual of Standard Practice for Detailing Reinforced Concrete Structures

318-1971 Building Code Requirements for Reinforced Concrete

347-1968 Recommended Practice for Concrete Formwork

SP-2-1975 Manual of Concrete Inspection

AISC American Institute of Steel Construction

Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings (AISC Specification) (2/12/69, with Supplements 1- 11/1/70, 2- 12/8/71, and 3 6/12/74)

ASME American Society of Mechanical Engineers

ASME Section III Code for Concrete Reactor Vessels and Containments (ASME Division 2 Boiler and Pressure Vessel Code, Section III, Div 2) 1975 Edition, with winter 1975 Addenda and other ASME Code Sections as required by ASME Section III, Division 2/ACI 359 Code.

Exceptions to the ASME Section III, Division 2/ACI 359 74 Code are listed in Appendix 3.8A.

Section II Material Specifications

Section III Nuclear Power Plant Components, Subsection NE for Class MC

Division 1 Components

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*Use this edition except Paragraph 4.4.3. Comply with ACI 305-1974 Paragraph 4.4.3 only.

Section IX 1971 Edition with Summer 73 Addenda. Welding and Brazing Qualifications. Field welding is performed to 1971 Edition with Winter 1976 Addenda, Welding and Brazing Qualifications.

AWS American Welding Society

D 2.0 Welded Highway and Railway Bridges with 1967 and 1970 revisions, for services performed prior to 4/29/77

D 1.1-75 Structural Welding Code, with Revisions 1 (1976) and 2 (1977) for services performed after 4/29/77

D 12.1-75 Recommended Practices for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction

SSPC Steel Structures Painting Council

SP-6 Commercial Blast Cleaning

USNRC United States Nuclear Regulatory Commission

The following NRC Regulatory Guides as identified in Section 1.8 are applicable:

1.10 Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

1.15 Testing of Reinforcing Bars for Category I Concrete Structures

1.18	Structural Acceptance Test for Concrete Primary Containment
1.19	Nondestructive Examination of Primary Containment Liner Welds
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
1.55	Concrete Placement in Category I Structures
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants
1.61	Damping Values for Seismic Design of Nuclear Power Plants
1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants
1.76	Design Basis Tornado for Nuclear Power Plants
1.92	Combination of Modes and Spatial Components in Seismic Response Analysis
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

<u>ANSI American National Standards Institute</u>

N6-2	Safety standard for the design, fabrication and maintenance of steel containment structures for Stationary Nuclear Power Reactors.
N-101.2-1972	"Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities."
N-101.4-1972	"Quality Assurance for Protective Coatings Applied to Nuclear Facilities."
N512-1974	"Protective Coatings (Paints) for the Nuclear Industry."
N45.2.2-1972	"Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants." (During the construction phase of SHNPP) and associated Amendments.
N45.2.5-1974	"Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants," except that bolt threads will be

allowed to be flushed with the top of the connecting nut in accordance with ANSI N45.2.5-1978.

Industry Standards

Industry standards, such as those published by the American Society for Testing and Materials (ASTM) or the American Association of State Highway and Transportation Officials (AASHTO), are used whenever possible to describe material properties, testing procedures, fabrication, and construction methods. Exception: The following ASTM standards pertaining to portable weighing devices will be site calibrated to a 0.5 percent accuracy through the range of use: ASTM-C-29; ASTM-C-88; ASTM-C-117; ASTM-C-125; ASTM-C-127; ASTM C 128; ASTM-C-131; ASTM-C-136; ASTM-C-138; ASTM-C-142; ASTM-C-535; and ASTM C 566.

Specifications

The following specifications specify the requirements for materials, design criteria, fabrication, erection, inspection, and quality assurance. These specifications, in general, reflect and expand on the requirements set forth in ASME Section III, Division 2/ACI 359 Code.

1. Ebasco Specification CAR-SH-AS-1 "Containment Liner, Air Locks, and Hatch"

2. Ebasco Specification CAR-SH-AS-7 "Structural Steel"

3. Ebasco Specification CAR-SH-M-54 "Mechanical Penetrations"

4. Ebasco Specification CAR-SH-E-30 "Electrical Penetrations"

5. Ebasco Specification CAR-SH-CH-6 "Concrete"

6. Ebasco Specification CAR-SH-CH-7A "Concrete Reinforcing Steel"

7. Ebasco Specification CAR-SH-CH-7 "Weldable Concrete Reinforcing Steel"

8. Ebasco Specification CAR-SH-CH-12 "Waterstops"

9. Ebasco Specification CAR-SH-CH-13 "Waterproofing"

10. Ebasco Specification CAR-SH-CH-15 "Mechanical Splicing of Concrete Reinforcing Steel"

11. Ebasco Specification CAR-SH-CH-16 "Dome Hub Plates and Reinforcing Steel Splice Assembly"

12. Ebasco Specification CAR-SH-CH-22 "Structural Integrity Test of Concrete Containment Building"

3.8.1.3 Loads and Loading Combinations

3.8.1.3.1 Definitions of loads

The following nomenclature and definitions apply to all loads encountered and/or postulated for the design of the Containment:

1. Dead Loads, (D) - Dead load consists of the weight of the concrete wall, dome, base slab, equipment deadweight, and all internal concrete, including hydrostatic loads. Uplift forces which are created by the displacement of groundwater, assumed to be at Elevation 251 ft., are accounted for in the design of the structure. Included are the weights of piping, cable trays, and ductwork.

   A reinforced concrete density of 143 pcf, with a possible minimum of 137 pcf, was used in the design. The density of the steel reinforcing and liner plate used in the design was 489 pcf.

   The deadweight of the crane bridge and trolley was also considered in the design. Equipment permanent operating loads as specified by the equipment manufacturers were included in the dead loads of the structure.

2. Live Loads, (L) - Live load consists of loads on the dome which are uniformly applied to the top surface of dome at an assumed value of 20 psf of horizontal plan projection to assure a strength adequate to support snow loading. A random temporary loading condition during construction or maintenance was assumed to be 50 psf. The design also accounts for a load of 250 tons supported by the polar crane during construction and maintenance operation (load combination 2) and 175 tons during load combinations 4, 5, 6, 7, and 9, described in Section 3.8.1.3.2.

3. Normal Operation Temperature Load (To) - Normal operation temperature loads consist of the loads induced by thermal gradients existing through the concrete wall and dome and those exerted on the concrete by the liner under normal operating conditions. The temperature gradient through the wall is essentially linear and is a function of the operating temperature internally and the average ambient temperature externally. The temperature gradient between the outside and inside of the Containment during operation induces stresses in the structures which are of internal nature, tension outside and compression inside the shell. Both summer and winter operating conditions are considered. In all cases the conditions assumed are considered of long enough duration to result in a straight line temperature gradient. The gradients considered are:

Summer Operation	Operating	Shutdown
Operating temperature inside building	120 F	65 F
Exterior sustained concrete temperature	90 F	90 F

Winter Operation	Operating	Shutdown
Operating temperature inside building	120 F	50 F
Exterior sustained concrete temperature	20 F	20 F

For all cases, the "as constructed" temperature is assumed to be 60 F.

Transient thermal gradients during startup and shutdown are considered in the analysis.

4. Operating Pipe Loads (Ro) - The pipe reaction anchor loads during normal operating or shutdown conditions are the loads exerted upon the containment structure by pipe restraints under the normal operating or shutdown thermal conditions of various piping systems.

5. Internal Pressure Load (Pv) - An internal negative pressure (other than due to a LOCA) of 2 psig is considered in the design.

   An internal positive pressure (other than due to a LOCA) of 3 psig is considered in the design.

   Either the negative pressure or the positive pressure is used in the load combination, whichever is more critical for the particular item of interest.

6. Test Pressure Load (Pt) - Section CC 6211 of the ASE Section III, Division 2/ACI 359 Code, requires that an internal pressure of up to 115 percent of the design pressure be considered in the design of the Containment. To meet this requirement, an internal pressure of 51.75 psig was used in the design.

7. Test Thermal Load (Tt) - Thermal loads during pressure tests, including liner expansion and temperature gradients in the wall and dome, are considered in the design of the Containment.

8. Design Basis Accident Pressure Loads (P) - The design basis accident pressure loads, due to a loss-of-coolant accident or other postulated pipe breaks, are considered in the design of the Containment. An equivalent static design pressure of 45 psig was used in the design of the Containment Structure. The use of an equivalent static load in the design of the containment for LOCA loadings is justified. Comparison of the time of LOCA pressure rise to the initial peak value, and the natural period for the first circumferential ("breathing") mode indicates that the ratio of the time of LOCA pressure rise to the first period of vibration is on the order of about 500:1. Therefore, the load can be considered to be statically applied, and the dynamic load factor for the LOCA pressure loading is essentially unity.

   Axisymmetric dynamic analysis studies indicated that the contributions of the higher (oval) modes to the maximum responses are relatively small. Therefore, these modes were not considered in the dynamic analysis of the containment building.

9. Design Basis Accident Thermal Loads (Ta) - Thermal stresses due to an internal temperature increase caused by the design basis accident are considered.

   The containment liner design average temperature under the design basis accident is assumed equal to 255.3°F*, associated with 1.0, 1.25, and 1.5 times the accident pressure, as described in Section 3.8.1.3.2. The containment liner design accident temperature for liner immediately adjacent to the main steam and feedwater penetrations is calculated to be 244°F when main steam pipe rupture is postulated to occur near the penetration. Accident temperatures mainly affect the liner, rather than the concrete and reinforcing bars, due to the insulating properties of the concrete. By the time the temperature of the concrete within the interior of the concrete begins to rise significantly, the internal pressure and temperature in the Containment due to the accident have been drastically reduced from their maximum.

Except for the wall at the main steam and feedwater penetration sleeves, the concrete wall is designed for a steady-state temperature gradient, with the interior face subjected to the maximum indicated temperature of 120°F and the exterior face subjected to summer or winter operation temperature, as specified in Section 3.8.1.3.1.C). The maximum steady state temperature of concrete at the main steam and feedwater penetration sleeves is 196.6°F, for concrete directly in contact with the sleeves. In addition, due to the interaction between the liner which is subjected to the containment design accident temperatures, and the concrete wall which is subjected to a steady state temperature gradient, increased stresses induced in the reinforcing steel and concrete are considered in the design.

10. Earthquake Loads (E, E') - Earthquake loads are computed using the following:

*Note: This value was derived with the considerations of SG Replacement and Power Uprate Projects in Reference 3.8.1-91.

a. Operating Basis Earthquake (E) horizontal ground acceleration is 0.075g.

b. Safe Shutdown Earthquake (E') horizontal ground acceleration is 0.15g.

c. To account for the simultaneous action of the three spatial components of the earthquake, the representative maximum value of a particular response is obtained by taking the square root of the sum of the squares of the corresponding maximum values of the response to each of the three spatial components calculated independently.

Specific loads resulting from the application of the above accelerations are obtained by the seismic dynamic analysis, as described in detail in Section 3.7.2.

11. Wind Load (Hu) - As described in Section 3.3.1, wind loading for the containment structure is based on a 179 mph wind, with gust factors included, at 30 ft. above ground level. Distribution of the wind load is made in accordance with References 3.8.1-39, -40, and -41, as described in Section 3.3.1.3.

12. Tornado Load (W) - As described in Section 3.3.2 tornado loading for the containment structure is based on the following characteristics:

a. External wind forces resulting from a tornado funnel with a horizontal peripheral tangential velocity of 290 mph and a horizontal translational velocity of 70 mph, W. Conservatively, this is taken as 360 mph wind applied uniformly over the entire height of the Containment. The loading distribution around the structure is in accordance with References 3.8.1-39,-40, and -41; gust factors are taken as unity.

b. Decrease in atmospheric pressure of 3 psi at a rate of pressure drop of 2 psi/sec., Wp. Venting of the structure is not considered.

c. The missile spectra given in Section 3.5.1.4 is used in the design of the containment structure, W_m. The methods of establishing the overall structural response due to missile impact are described in Section 3.5.3.2.

d. In determining the total tornado load, W, the effects of the uniform tornado wind load, W_w, the tornado differential pressure load, Wp, and the tornado missile load, Wm, are considered by using the combinations listed in Section 3.3.2.2.4.

13. Design Basis Accident Piping Loads (Ra) - The pipe reaction anchor loads during accident conditions are the loads exerted upon the containment structure by pipe restraints under the thermal conditions generated by the design basis accident, including Ro.

14. Pipe Accident Loads (Rr) - The pipe accident loads are the loads exerted upon the containment structure due to local effects of the design basis accident and include:

    a. Rrr = Equivalent static load on the structure generated by the reaction of a broken high-energy pipe during the postulated break, including an appropriate dynamic load factor to account for the dynamic nature of the load.

    b. Rrj = Jet impingement equivalent static load on a structure generated by or during the postulated break, including an appropriate dynamic load factor to account for the dynamic nature of the load.

    c. Rrm - Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping or small pieces of equipment travelling at high velocities, including an appropriate dynamic load factor to account for the dynamic nature of the load.

15. Post-LOCA Flooding (Hq) - Post-LOCA flooding of the Containment for the purpose of fuel recovery is not a design condition. When access to the Containment is required following a LOCA, all necessary repairs will be made to permit fuel recovery.

3.8.1.3.2 Load combinations

The design of the Concrete Containment Structure incorporates two general loading categories, the Service Load Category and the Factored Load Category.

3.8.1.3.2.1 Service load combinations

Service load combinations are any conditions encountered during construction and normal operation of the plant. Included in such conditions are any anticipated transient or test conditions during normal and emergency startup and shutdown of the nuclear steam supply, safety, and auxiliary systems. Also included in this category are those severe environmental conditions (operating basis earthquake and wind load) which may be anticipated during the life of the facility. The service load combinations are presented in Table 3.8.1-2.

3.8.1.3.2.2 Factored load combinations

Factored loads include loads encountered in the life of the facility such as severe environmental loads (wind loads, operating basis earthquake), extreme environmental loads (tornado loads, safe shutdown earthquake), and abnormal loads (loads generated by the design basis accident, P, Ta, Ra, and Rr). The factored load combinations are presented in Table 3.8.1-2.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General considerations

The analysis of the containment shell is based on the classical theory of thin elastic shells of revolution in accordance with Section CC-3300 of the ASME Code, Section III, Division 2. The shell is assumed to be ideally elastic, homogeneous, and isotropic. Reinforcement and the steel liner are neglected in calculating the member stiffness.

The design of the Containment demonstrates that, for factored load conditions, the following requirements are met:

1. The summation of external and internal forces and moments satisfies the laws of equilibrium and does not bring any structural section to a general yielding state.

2. Tensile yielding in the reinforcement is acceptable when thermal gradient temperature effects are combined with other applicable loads, provided that the temperature induced forces and moments are reduced as yielding in the reinforcement occurs, and the increased concrete cracking does not cause deterioration of the Containment.

The liner plate is not used as a strength element. Interaction of the liner with the Containment is considered in determining liner behavior.

The general requirements used in the design of the metallic liner are as follows:

1. The liner plate is designed to withstand the effects of imposed loads and to accommodate deformation of the concrete containment without jeopardizing leak tight integrity.

2. The liner plate is welded using weld details which do not jeopardize leak tight integrity of the Containment.

3. The liner plate is anchored to the concrete containment. This does not preclude local flexural deformation between anchor points.

4. The liner plate is designed within the limits of stress, strain, and deformation specified in Table 3.8.1-3.

The liner anchorage system is designed to accommodate all design loads and deformations without loss of structural or leak tight integrity.

The anchorage system is designed to that a progressive failure of the anchorage system is precluded in the event of a defective or missing anchor.

Penetration assemblies, including sleeves, reinforcing plates, and penetration anchors, are designed to accommodate all design loads and deformations without loss of structural or leak tight integrity. Effects such as temperature and shrinkage are considered.

Temporary or permanent brackets and attachments are designed to resist the design loads without loss of the liner integrity due to excessive deformation from bracket or attachment loads.

The design of penetration sleeves, not backed by concrete and designated as Class MC components, as defined in Section 3.8.1.1, is covered in Section 3.8.2.

3.8.1.4.2 Assumptions on boundary conditions

Basically three structural components are analyzed by assuming that each is in equilibrium with loads applied to it and compatible with deformations at the juncture of the structures. The three structures are:

1. The 130 ft. I.D. Hemispherical dome.

2. The 130 ft. I.D. and 160 ft. high cylindrical wall.

3. The circular foundation mat.

Mathematically, the dome and cylinder are considered as thin-walled shells in the form of surfaces of revolution. The classical theory of thin shells is used to determine both membrane and bending stress resultants due to each individual load, but redistribution of moments and forces is considered due to the cracking of concrete of these statically indeterminate structures, as described in Section 3.8.1.4.4.4.

3.8.1.4.3 Circular foundation mat analysis

The concrete foundation mat which supports the Concrete Containment Structure and the internal structures is designed in accordance with the ASME Section III, Division 2/ACI 359 Code, Winter of 1975 Addenda.

The analysis of the foundation is concerned primarily with the determination of shear and moment in the reinforced concrete foundation mat and the determination of the interaction of the mat with the underlying bearing material.

For this foundation supported by rock, the pertinent requirements of the design are the maintenance of bearing pressures within allowable limits, particularly due to overturning moments, and the assurance that there is adequate resistance to sliding of the structure if its is subjected to lateral loads. The stability of the foundation mat is further discussed in Section 3.8.5.

The design loads considered for the analysis of the foundation mat are the maximum resulting forces from the superstructure due to static and dynamic load combinations and those loads directly applied on the base slab, such as dead, live, hydrostatic, internal pressure, temperature, and equipment loads.

In the analysis, the foundation mat is treated as a plate supported on an elastic foundation; the finite element method of analysis is used, employing proven, industry accepted computer programs. The subgrade modulus considered in the analysis is determined by using appropriate correlations with the engineering properties of the foundation materials used at the site, as described in Section 3.7.2.4.

The containment and internal structure walls supported by the foundation mat are represented by force boundary conditions and appropriate nodal displacement restraints in the finite element mathematical model.

The rock foundation is simulated by discrete springs acting at the grid points of the mat elements. For the initial step of analysis all springs are assumed active. The resulting forces in the springs for the critical load combinations indicate which springs are under tension and should be eliminated.

Thereafter, the second step is commenced by deactivating the resulting tension springs for the most severe load combinations. The final check of the assumed spring supports is required to demonstrate that no tension exists in the springs. The analysis then supplies the required forces and moment for the final design of the mat.

The STARDYNE computer program is used for this analysis.

The cavities in the mat are analysed together with the base mat. Figure 3.8.1-26 shows the foundation mat shear forces and bending moments for the most critical load combinations, which govern the design.

3.8.1.4.4 Cylindrical wall and dome analysis

The analysis of the hemispherical dome and cylinder due to axisymmetric loads, such as accident pressure, test pressure, gravity, and temperature loads, is based on the primary membrane theory. In addition, the local bending moment and radial shear in the vicinity of the dome-cylinder juncture, as well as the cylinder-base juncture, are also analyzed by applying the condition of compatibility at the junctures. The analytical procedure and formulations are based on those contained in References 3.8.1-42, 3.8.1-43, 3.8.1-44, and 3.8.1-45. The change of the sectional properties due to cracking (or partial cracking) of concrete under accident pressure and test pressure are considered in the analysis.

The expansion of the liner in the dome and cylinder due to an increase in temperature creates tension in the reinforcing and compression in the liner. Compatibility of these thermal strains, and the mechanical strains due to pressure, are accounted for in the analysis.

The effects of concrete cracking are carefully considered in the containment analysis. The following three types of cracking are considered:

1. Membrane crack, an axisymmetrical crack or a crack formed around the whole circumference. This crack could result from internal pressure loading, or from internal pressure combined with other unsymmetric loads.

2. Local membrane crack, an unsymmetrical local crack constituting only a part of the circumference. This crack could result from a seismic load in the normal operating condition.

3. Partial bending crack, only a portion of the section (along the thickness) is assumed to be cracked. For example, a horizontal crack due to the discontinuity moment at the lower portion of the containment wall under the accident pressure load is considered to be a partial bending crack.

3.8.1.4.4.1 Treatment of axisymmetric and non-axisymmetric loads

The concrete containment structure loading cases under axisymmetric loads are analyzed by the finite element method, by using the beam on elastic foundation approach to represent the actual cylindrical shell of revolution, and taking into consideration the effects of cracking of concrete, as described in Section 3.8.1.4.4.4. A three-dimensional model, using the STARDYNE computer program, was also used to verify the results of the Ebasco computer program.

The Concrete Containment Structure loading cases under non-axisymmetric loads are analyzed by the finite element method, using a three-dimensional finite element model and industry proven computer programs, as described in Section 3.8.1.4.4.4.

3.8.1.4.4.2 Treatment of transient loads

As presented in Section 3.8.1.3.1 c), during normal operating conditions of the plant a linear temperature gradient across the containment wall thickness develops, with the inside face of the wall subjected to the maximum indicated temperature of 120°F and the outside face of the wall subjected to a temperature of 90°F and 20°F in summer and winter conditions, respectively.

The normal operation thermal loads are determined considering the thermal gradients in summer and winter which are adjusted by subtracting the construction temperature from the surface temperatures for the thermal input into the containment analysis.

The design accident thermal loads consist of normal operation thermal gradient and the temperature increment generated by the postulated accident.

As described in Section 3.8.1.3.1i), the accident temperature mainly affects the liner, rather than the concrete and reinforcing bars, since the concrete has a much lower thermal conductivity than the steel liner and the accident temperature drops off very rapidly. Therefore, the accident thermal increment cannot penetrate very far into the concrete, as evidenced by numerous transient thermal analyses. Thus, at the moment of the higher accident temperature, the Containment is subjected to a "skin temperature effect" imposed by the liner plate. Due to the interaction between the concrete wall, subjected to a steady-state temperature gradient, and the liner plate, subjected to containment design accident temperatures, increased stresses induced into the reinforcing steel and into the concrete are determined.

3.8.1.4.4.3 Treatment of localized loads

The Concrete Containment Structure is designed for localized loads, such as jet impingement loads and tornado generated missile loads.

The Concrete Containment Structure is designed to withstand, without loss of function or perforation, a representative tornado-driven missile spectra as described in Section 3.5.1.4, using the combinations of loads listed in Section 3.3.2.2.4 and in Table 3.8.1-2.

An impactive dynamic analysis is performed in order to investigate the following aspects of the problem:

1. The penetration of the target by a missile, local damage to the impact area, estimation of the depth of penetration, and the potential generation of secondary missiles by spalling or scabbing, as described in Section 3.5.3.1.

2. The structural response of the member, the overall response of the structure to missile impact, assuming acceptable ductility ratios and estimates of forces, moments, and shears induced in the structure by the impact force of the missile to check for structural integrity, as described in Section 3.5.3.2.

A three-dimensional finite element model is used for this investigation, with the tornado-generated missile connected load applied independently at different locations on the outside face of the containment wall in order to determine the equivalent spring constants, equivalent masses, and the natural frequencies of the equivalent simplified dynamic models used in the investigation of the structural responses.

3.8.1.4.4.4 Effects of cracking of concrete

The following considerations are included in evaluating the effects of concrete cracking:

1. Analysis for Axisymmetric Loads - When the Containment is subjected to axisymmetric loads, the shell is analyzed by the methods specified below. The accident pressure is the load that causes membrane cracks in the shell and partial bending cracks at the boundaries, as described in Section 3.8.1.4.2. The membrane stress resultants are not affected by the sectional properties of the shell; however, the boundary discontinuity moments are affected by the sectional properties of the shell. Since this is a non-linear material problem, an iteration process is employed to obtain reliable results.

   The containment crack modeling is shown on Figure 3.8.1-27. The containment analysis used to account for section property variations and changes due to concrete cracking is a finite element analysis which includes the beam on elastic foundation approach to represent the actual cylindrical shell of revolution. The finite element analysis and the Ebasco computer program used in the analysis are further described in Appendix 3.8B. The finite element analysis consists of the following procedures:

   a. The meridian and circumferential membrane force resultants which are independent of the sectional properties are first calculated by the classical membrane theory.

$$N_x = \frac{PR}{2} \qquad (1)$$

$$N_\theta = PR \qquad (2)$$

Where:

P = pressure load, PSF

R = Radius of the containment in ft.

N_x = Meridian membrane force in k/ft.

N_θ = Circumferential membrane force in k/ft.

b. The radial displacements are calculated by the membrane theory, with a free boundary condition and completely cracked section.

$$d_i = \frac{P}{K_i} \qquad (3)$$

$$K_i = \frac{Et_i}{R^2} \qquad (4)$$

Where:

d_i = free boundary radial displacement for i^{th} element.

E = Young's modulus

t_i = Equivalent thickness of the reinforced steel for i^{th} element.

K_i = Shell equivalent modulus of elastic foundation.

c. At the vicinity of the boundary, where discontinuity moments and radial shear develop, the axisymmetrical bending theory is used and its closed form solution (see Reference 3.8.1-45) is employed to construct the flexibility matrix. It is shown on Figure 3.8.1-27 that a finite number of elements can be subdivided, each of which may be assigned different sectional properties based on the presumed compression uncracked zone. The equation is written in matrix form;

$$[f]\ \{F\} = \{d\} \qquad (5)$$

where:

[f] is the flexibility matrix size 2N x 2N (detailed in Appendix 3.8.B)

{F} is the generalized forces, including 2N elements.

{d} is the relative incompatible displacement, which is obtained as described in equation (3) above.

d. After the shears and moments are computed from equation (5), the total moments and meridianal membrane forces for each specific loading combination are obtained by summing up all the moments and meridianal membrane forces due to the individual factored loads.

e. When the total meridianal membrane forces and moments at each node are determined, the compression zone at each node point is computed to check with the presumed compression zone at each node point. If they are sufficiently close, the iteration process is completed and the final stresses are reached. If they are not close, another trial is attempted.

f. Superposition is not valid in this process; complete cycle iteration is performed for each load combination case.

2. Analysis for Asymmetric Loads - When the Containment is subjected to asymmetric loads (seismic and wind loads), the stress resultants of major concern are the vertical (meridian) membrane and the tangential shear. There are local bending moments which are considered to be minor. The type of cracks expected is dependent upon the load combinations. Both membrane cracks and local membrane cracks could develop. Membrane cracks could form under accident conditions and local membrane cracks could form under normal operating conditions.

The structural analysis for asymmetric loads is performed by using a finite element computer program which has been developed primarily for analyzing uniform and isotropic linear elastic material. For the accident condition with membrane cracking and a uniform section, the major analysis results are reliable. In the normal operating condition with local membrane cracks, the results are affected by discrepancies in the sectional properties. However, the shear forces and bending moments developed in the Concrete Containment Structure due to the axisymmetric loading conditions are less than 10 percent of the shear forces and bending moments developed in the concrete cylindrical wall due to the axisymmetric loading conditions generated during the postulated accident. Therefore the discrepancies in the sectional properties for the normal operating conditions are insignificant.

A three-dimensional finite element approach is used to analyze the hemispherical dome and cylinder due to non-axisymmetric loads such as wind, tornado, and seismic loads. The CDC "ANSYS" or "STARDYNE" finite element computer program is used to perform the analysis. Elements are refined at the vicinity of the junctures where change of stress resultants are expected. These programs are developed based on a linear material properties assumption. No iteration is performed to consider concrete cracking automatically. Therefore, the cracked sections are predicted as an input to account for concrete cracking.

An equivalent thickness of the shell is used to modify concrete cracking. The stress resultants which are used in the design are not significantly affected by the change of section rigidity.

A comparative study was performed using the finite element analysis (described in the axisymmetric load analysis) and other industry proven computer programs such as ANSYS and STARDYNE. Figure 3.8.1-28 shows the results of the comparative study. ANSYS was used for the computer program analysis of the polar crane region because the polar crane runway girder and support brackets were represented in the ANSYS model. The design of the containment structure wall, dome, and penetrations used the results from the STARDYNE output because the penetrations were included in the STARDYNE model. The in-house finite element analysis program was used to verify the results obtained for the design of the cylindrical wall.

The containment STARDYNE model used triangular plate elements for the static analysis of the building. The elements were assumed to be homogeneous and isotropic. For that type of element, two factors determine the element properties: the modulus of elasticity (E) of the material of the element and the element moment of inertia (I) derived from the thickness of the element.

Cracked section properties were accounted for in the model by modifying the value of E in the inputs such that the product EI corresponded to the cracked condition of the wall at the location of the element. This was done to the EI for both the vertical and horizontal directions, using the cracks determined from the cracking analysis by the SHELL computer program. For the vertical direction, the wall was divided into zones, and the average crack size for zone was used for the zone.

Figure 3.8.1-27 illustrates the wall finite element model and modeling of cracks.

Figures 3.8.1-29 through 3.8.1-32 show the cylindrical wall and dome shear forces, bending moments, and displacements from the most critical load combinations, which govern the design.

The Concrete Containment Structure is a conventionally reinforced concrete structure in which shrinkage tends to develop stresses in a reverse direction from that developed by the design basis accident; therefore shrinkage is not considered in the design. During construction of the containment structure, construction techniques, as described in Section 3.8.1.6.3 (a), are used in order to minimize the effects of shrinkage.

3.8.1.4.4.5 Description of the Computer Programs Utilized

Descriptions of the computer programs utilized in the analyses and design of the Concrete Containment Structure are presented in Appendix 3.8B. Basically they are industry proven computer programs, such as STARDYNE, NASTRAN, and ANSYS. For the dynamic analysis of the containment structure, the STARDYNE computer program was used for the three-dimensional dynamic model and an Ebasco computer program was used for the two-dimensional dynamic model.

The finite element computer program used to account for the effects of concrete cracking is an Ebasco computer program, which uses the beam on elastic foundations approach to represent the real cylindrical shell. In order to demonstrate that the results obtained by using this computer program are substantially identical with the results obtained by using industry proven computer programs, a comparative study was performed, as described in Section 3.8.1.4.4.4; the results are presented on Figure 3.8.1-28.

3.8.1.4.4.6 Treatment of the Effects of Induced Shears

a) Tangential Shear - The tangential shear force, Vu, is due primarily to earthquake, wind, or tornado loading. For earthquake loading, the tangential shear force is determined from the square root of the sum of the squares of the multiple components of earthquake loading. For wind or tornado loading, the tangential shear forces are determined based on the direction of loading under consideration and are compatible with the determination of N_{he} and N_{ve}, defined in this Section.

The criteria for tangential shear are as follows:

1) All membrane forces, including thermal effects, N_{ht} and N_{vt}, are considered.

2) The allowable tangential shear force, V_c, is defined in Section 3.8.1.5.1.1.c)2).

3) The meridional and hoop reinforcing with or without diagonal reinforcing is proportioned for the vertical and horizontal forces respectively plus that portion of the shear force not carried by the diagonal reinforcing.

4) When diagonal reinforcing is required by Section 3.8.1.5.1.1.c)2) the following equations are used for a four (4) way reinforcing system with 45° inclined bars, for factored load combinations presented in Table 3.8.1 2:

$$A_{sh} = \frac{N_h + N_{he} + V_u}{0.9 f_y} \quad (6)$$

$$A_{sv} = \frac{N_h + N_{ve} + V_u}{0.9 f_y} \quad (7)$$

$$A_{si} = \frac{V_u - V_c}{0.9 f_y} \quad (8)$$

where:

A_{sh}	=area of reinforcing steel in the horizontal direction (in.^{2}/ft.)
A_{sv}	=area of reinforcing steel in the vertical direction (in.^{2}/ft.)
A_{si}	=area of reinforcing steel in the inclined direction (in.^{2}/ft.)
Nv and Nh	=Membrane force in the horizontal and vertical direction due to loads other than earthquake, wind, and tornado (such as pressure and dead load).
N_{ve}	=Membrane force in the vertical direction due to earthquake, wind, or tornado loading. When considering earthquake loading, the force is based on the square root of the sum of the squares of two horizontal and one vertical component of earthquake loading. When considering wind or tornado load, the force is based on the absolute sum of the horizontal and vertical components of loading. The force is always considered as positive.
N_{he}	=Membrane force in the horizontal direction due to earthquake, wind, or tornado loading. The forces are determined on the same basis as Nve. The force is always considered as positive.
f_y	=Specified tensile yield strength of reinforcing steel, psi.
V_u	=Maximum tangential shear at the section under consideration.
V_c	=Tangential shear force carried by the concrete. The strain compatibility of the concrete and reinforcing system along the minor principal axis (concrete compression strut) may be used in verifying that the strain in the tension diagonal does not exceed the strain allowable of $2E_y$.

5) When diagonal reinforcing is not required, the following equations are used for factored load combinations presented in Table 3.8.1-2.

$$A_{sh} = \frac{N_h + N_{he} + V_u}{0.9 f_y} \quad (9a)$$

$A_{sv} = \frac{N_h + N_{ve} + V_u}{0.9 f_y}$ (9b)

6) For service load combinations presented in Table 3.8.1-2, the equations (6) through (9) are used to design the meridional hoop, and inclined reinforcing steel, but 0.9 f_y is replaced by the reinforcing stress allowable listed in Section 3.8.1.5.2.2 and V_u is replaced by V, the applied shear load at the section under consideration.

b) Radial Shear - An example of this type of shear is the shear force caused by self-constraint of a cylinder and base slab during pressurization of the Containment, V_u.

1) Factored Load Design - The nominal shear stress, V_u, is computed by:

$V_u = \frac{V_u}{0.85bd}$ (10)

where:

d = Distance from the extreme compression fiber to the centroid of the tension reinforcement, in.

b = Unit length of section.

When shear reinforcement perpendicular to the containment surface is used, the required area of shear reinforcement is not less than:

$A_v = \frac{(v_u - v_c)bs}{f_y}$ (11)

where:

s = Spacing of shear reinforcement in a direction parallel to the longitudinal reinforcement. The perpendicular shear reinforcement is not spaced further apart than 0.50d.

vc = Nominal permissible shear stress carried by concrete, psi, as defined in Section 3.8.1.5.1.1 (c).

When inclined stirrups are used, the required area is not less than

$A_v = \frac{(v_u - v_c)bs}{f_y (\sin\alpha \pm \cos\alpha)}$ (12)

When shear reinforcement consists of a single bar or a single group of parallel bars, all bent upward at the same distance from the support, the required area is not less than

$A_v = \frac{(v_u - v_c)bs}{f_y (\sin\alpha)}$ (13)

in which (v_u - v_c) does not exceed $3\sqrt{f'c}$

where f'c is the specified compressive strength of concrete.

When shear reinforcement consists of a series of parallel bent-up bars, or groups of parallel bent-up bars at different distances from the support, the required area is not less than that computed by equation (13).

Only the center three fourths of the inclined portions of any longitudinal bar that is bent is considered effective for shear reinforcement.

Where more than one type of shear reinforcement is used to reinforce the same portion of the web, the required area is computed as the sum of the various types separately. In such computations, v_c is included only once. The value of (v_u - v_c) does not exceed $8\sqrt{f'c}$.

Inclined stirrups and bent bars are spaced so that every 45-degree line extending toward the reaction from the mid-depth of the member, 0.50d, to the longitudinal tension bars are crossed by at least one line of shear reinforcement.

Shear reinforcement extends to at least a distance, d, from the extreme compression fiber and is anchored at both ends to develop the design yield strength of the reinforcement.

2) Service Load Design - The same requirements stated in Section 3.8.1.5.1.1.C)2) are used to design shear reinforcement for service loads, with the following modifications:

a) Equation (10) is replaced by $v = \frac{V}{bd}$ (14)

b) The reinforcement steel allowable stress from ASME Code Section III, Division 2/ACI 359 Code CC-3032.1 replaces f_y in Equations (11), (12), and (13).

3.8.1.4.4.7 Variation in Physical Material Properties

The basic assumptions used in the static analysis are in accordance with the ASME Section III, Division 2/ACI 359 Code. Quality control assures that material properties are within the ranges of values anticipated by the analysis and the ASME Section III, Division 2/ACI 359 Code.

In addition, the safety factors included in the allowable stresses provide a safeguard against small adverse variations in material properties and strength.

The effects of the penetrations of the containment shell are taken into account by utilizing a finite element technique to determine the increased forces and moments of the shell in the area of the penetrations. The redistribution of stresses due to containment concrete cracking is also investigated.

Variations in the foundation rock parameters have a negligible effect on the overall analysis of the structure for combined loads since the seismic loads used in the analysis are based on the most critical rock properties.

Concrete temperatures do not exceed the values indicated in the ASME Code Section III, Division 2/ACI 359 Code, Section CC-3440 (a), for normal operation and Section CC-3440 (b) for accident condition.

3.8.1.4.4.8 Treatment of Large Thickened Penetration Regions

Large openings are provided for the equipment hatch, personnel airlocks, main steam penetrations, and feedwater penetrations. In all of these areas, the thickness of the wall is increased from 4 ft. 6 in. to 6 ft. 6 in. in order to accommodate the concentration of stresses and to allow the introduction of additional reinforcement required by special analysis.

All of the large penetrations are incorporated into a three-dimensional finite element model in which a finer mesh around the penetrations is provided in order to obtain reliable stress information. The effect of eccentricity due to the fact that the increase of wall thickness is extended only on the outside face of the wall is considered. The STARDYNE computer program is used for this analysis and the investigation is performed for all load combinations listed in Table 3.8.1-2.

As described in Section 3.8.2, the interaction between the cylindrical concrete wall and steel penetrations is considered and the interaction forces are introduced at the nodal points around the openings.

To account for the effects of concrete cracking, the cracking pattern determined in the finite element analysis described in Section 3.8.1.4.4.4 is used as an input in the finite element analysis used for the large openings.

The results of the analysis include biaxial bending moments and shears, axial force, and torsion. These are used in the design of the reinforced concrete around the penetration openings. Conventional reinforcement, consisting of circular bars around the openings for moments and tensions and stirrups for shear and torsion, is provided.

3.8.1.4.4.9 Liner Plate Analysis and Liner Anchorage System

The purpose of the liner plate is to provide a leak-tight membrane. As such, it is not designed as a component of the Containment to resist design loads, but the stresses and strains in the liner are determined considering the wall and liner as a composite section to assure that the leak-tight integrity of the Containment is not jeopardized.

An anchorage system consisting of headed studs is used to retain the liner and concrete shell as a composite section. The studs are fusion-welded to the liner plate. The headed studs are 5/8 in. diameter by 4 in. long. The mat liner is anchored by welding it to embedded steel members which are anchored in the concrete mat. At the mat-wall intersection, the vertical wall liner is continuously welded to the mat liner.

The liner is analyzed for the loads and load combinations shown in Table 3.8.1-2, except that all load factors in all factored load combinations are equal to 1.0. The calculated stresses and strains do not exceed the values shown in Table 3.8.1-3.

The size and spacing of liner anchorages are chosen such that the response of the liner will be predictable for all of the loads and load combinations shown in Table 3.8.1-2 and keep the liner in contact with concrete for those conditions of loading. The anchorage system is designed to accommodate the design in-plane shear loads or deformations exerted by the liner and loads applied normal to the liner surface. The forces and displacements do not exceed the allowables listed in Table 3.8.1-3. The containment vacuum load of 2 psi, with a load factor of 1.0, is considered in combination with other loads. Liner anchorages and welds are designed to withstand this load condition.

In general, the design of the liner is not fatigue-controlled, since most stress and strain changes will occur only a small number of times and produce only minor stress-strain fluctuations. Earthquake and design basis accident strains occur too infrequently, and with too few cycles, to generally be controlling. Nevertheless, because of the critical nature of the liner, the design assures the suitability of the liner for the following specific operating conditions involving cyclic applications of load and the thermal condition specified in the design specification for the Containment.

The fatigue evaluation of the liner considered the following cyclic loading conditions based on the original plant design life:

a) Thermal cycling due to variations of temperature between cold shutdown and operating conditions of the reactor. The number of cycles was postulated as 500.

b) Thermal cycling due to variations of temperature between summer and winter operating conditions were considered.

c) Thermal cycling associated with the postulated LOCA is one event.

The fatigue methods and limits established by ASME Boiler and Pressure Vessel Code Section III, Division 1, Article NB 3222.4 apply.

Since the liner is anchored at relatively close intervals compared to its diameter, the analysis is based on plate or beam theory, as appropriate.

The anchor studs are analyzed assuming that the liner remains elastic under all conditions, that the liner strains are converted to stresses using Hook's Law, and that the modulus of elasticity and Poisson's ratio do not exceed yield.

The anchor design and analysis considers the effects of the following:

a) The unbalanced loads resulting from variations of liner curvature. Some areas of the liner may have inward curvature between the anchors, whereas other areas may have outward curvature. The variations result in shear load and displacement at the anchor;

b) Liner thicker than nominal due to the rolling tolerances given in SA 20. The thicker plate may impose greater forces and displacements on the anchorage system than a nominal thickness liner;

c) Yield strength higher than the minimum specified due to the rolling processes and biaxial loading;

d) Weld offset, structural discontinuities, and concrete voids behind the liner;

e) Variation in anchor spacing;

f) Variation in anchor stiffness due to variations of the concrete modulus;

g) Local concrete crushing in the anchor zone; and

h) Stud anchors that are designed to fail before tearing the liner.

Due to the nature of the loading and types of components, the allowable capacity of the components is specified in terms of stresses and strains for liner plate and in terms of forces and displacements for the concrete anchorages.

In order to determine the ultimate capacity (force and displacement) and the spring constants of the anchorages, which are required in the analysis of the liner and anchorage, tests were performed at Lehigh University's Fritz Engineering Laboratory in Bethlehem, Pennsylvania. The anchor studs were embedded into a concrete disc, which was subjected to bending in order to create biaxial tension similar to the actual state of stresses that would exist in the actual containment wall during an accident condition. The anchorages were tested in tension and shear both in the region where there is biaxial tension and in the region where there are no stresses.

The results of the tests are shown on Figures 3.8.1-32 through 3.8.1-35. Figures 3.8.1-32 and 3.8.1-33 show the results for studs subjected to tension and shear, respectively, with concrete in biaxial tension; Figures 3.8.1-34 and 3.8.1-35 show the results for studs subjected to tension and shear, respectively, with the concrete unloaded. The tests show that the ultimate capacity of the anchorages is not influenced by biaxial tension. The slope of the curve for the anchorages tested in the region with biaxial tension is smaller than the slope of curve for the anchorages tested in the region with no stresses. Although the ultimate force and displacement capacity is not changed for concrete in biaxial tension or unloaded, the biaxial tension state has an important impact on the analysis, since the slope of the load deformation curve determines the spring constants used in the analysis.

The results of the test for the biaxial tension state were included in a finite element model to determine the behavior of the liner anchorages interaction. Figure 3.8.1-36 shows the finite element model used for the analysis of the liner plate.

To minimize stresses and strains at the junction between the mat liner plate and the cylindrical wall liner plate, an unanchored 90 degree, free standing welded connection was selected, as described in Section 3.8.1.1.3.2. The analysis of this connection is performed using a finite element model as shown on Figure 3.8.1-37. The ANSYS computer program is used for this investigation. The results of the investigation are shown on Figure 3.8.1-38.

The 1 in. thick continuous liner plate which supports the crane brackets is anchored to the concrete containment with anchor bars, plates, and headed steel studs. To determine the behavior of the liner plate in this region and the forces induced in different types of anchorages with different structural rigidity, a finite element model is used, as shown on Figure 3.8.1-39. The external loads used as an input in the finite element analysis are the output forces obtained from the special investigation of the crane girder-cylindrical wall interaction described in Section 3.8.1.4.4.12.

Temporary or permanent brackets and attachments connected to the containment liner plate to support mechanical pipe systems or small equipment are designed to resist the design loads without loss of liner integrity due to excessive deformation from bracket or attachment loads. In order to accommodate the additional loads, the liner plate is locally reinforced with additional studs in the area of surface attachments.

Brackets and attachments connected to the liner are designed and analyzed by using accepted techniques in accordance with the AISC Manual for Steel Construction, Part 5, "Design, Fabrication, and Erection of Structural Steel for Buildings." The design allowable stresses for mechanical loads in the construction, test, and normal load categories for brackets and attachments are in accordance with the AISC Manual. For all other categories of loading, brackets and attachments have been sized for the required section strength as specified in Section 3.8.3.3.3. For brackets and attachments which resist external mechanical loads and are not continuous through the liner, the liner stress in the through-thickness direction is taken as one-half of that in the as rolled direction.

Due to the internal pressure and/or accident differential temperature between the liner surface and concrete wall, the liner plate may be subjected to membrane stresses (tension during the test pressure and compression during accident conditions). As shown on Figure 3.8.1-40, the connection of the pads for mechanical supports induces additional bending stresses into the liner plate. Additional bending stresses are also induced by the locked in stresses produced during placement of fresh concrete during construction, if the liner plate is used as an internal form. All of the combined membrane and bending stresses are calculated and superimposed in order to verify that the stresses and strains are within the ASME Section III, Division 2/ACI 359 Code limits.

If the yield strength capacity of the liner is not exceeded, the analysis is a linear problem and the superposition principle is valid. Therefore the stresses induced into the liner plate by the containment structure loading and the stresses induced by the mechanical loads are determined separately and superimposed after that.

The regions where the stress in the liner exceed yield stress, the load combinations (mechanical loadings included) that produce such state of stress, and by how much the stress and strain are in the inelastic region are:

a) In the cylindrical wall liner from one to five feet (Elevations 217 221) above the top of the foundation mat. The load combination is a) Service Load Combination, 1) Test Pressure, of Table 3.8.1-2. There are no mechanical loadings. The values for how much stress exceeds yield, and strain are given below.

    1) A finite element analysis was performed, using the ANSYS program. The finite element model is shown in Figure 3.8.1-37. The deformation of the wall inside face vertical reinforcement was calculated, based on maximum allowable reinforcement stress for the load case, and used to calculate the liner displacement. The value obtained of 0.0828 inches over the distance of 5 feet was conservatively rounded off to 0.1 inches. The liner stress for this value is 10.5 percent higher than the material specified minimum yield stress.

    2) Figures 3.8.1-41, -42, and -43 were used to determine wall liner strain for load cases with normal operating temperatures. Plastic design theory concepts used in

the preparation of the figures are outlined in Figures 3.8.1-40. Figures 3.8.1-41 and -42 present the relationships among membranes forces, strains, and eccentricities in load application (Figure 3.8.1-41), and among bending moments, strains, and eccentricities (Figure 3.8.1-42). Figure 3.8.1-43 furnishes the value of strain for various combinations of axial force and bending moment.

3) Since the analysis gave a combination of axial force and bending moment in the liner plate, Figure 3.8.1-43 was used to determine the strain due to axial forces combined with bending moment. The value obtained was 0.00154 in./in.

b) In the cylindrical wall liner between Elevations 226 and 256. The load combination is b) Factored Load Combinations, 14) Loss-of-coolant accident, of Table 3.8.1-2. The maximum overstress occurs at Elevation 236. The stress is 4.8 percent higher than the material specified minimum yield strength. The strain is 0.00145 inches per inch.

The value of strain was obtained by combining the strains due to deformation of the wall, the short-term temperature differential between the liner and concrete, and the mechanical loading. At the critical section, a liner strain of 0.00050 in./in. due to the deformation of the wall was calculated from the force-equilibrium of the cross-section, using the assumption that a plane section remains plane after deformation. All loads of the load combination were considered except the short-term temperature differential. A liner strain of 0.00090 in./in. due to the temperature differential was calculated, using the assumption that concrete and liner deformation are equal. A liner strain of 0.00005 in./in. due to the mechanical loading was developed from the STARDYNE computer program analysis of the liner-anchorage system. The three strains were combined to obtain the total strain.

The overstresses reported are in the 3/8 inch thick liner plate on the cylindrical wall, and are based on the minimum yield strength of 38 ksi at 70 F - 100 F specified for the material. The mill test reports for the 3/8 inch plates that were supplied give a lowest value of yield strength of 45.6 ksi. This value, when adjusted for the highest temperatures postulated for the liner reduces the yield strength to 41.7 ksi at 250 F (the maximum liner temperature reported in Section 3.8.1.3.1.1 conservatively rounded off). Therefore, for both the service and factored load cases, wall liner stresses do not exceed yield. Since the liner anchors were analyzed assuming the liner remained elastic under all conditions (ASME Section III Division 2, CC 3630), the higher yield strength properties of the wall liner plate have no effects on the liner anchors.

The liner and liner studs were evaluated for the increase in the Design Basis Accident temperature associated with the Steam Generator Replacement and Power Uprate Projects. The maximum liner temperature for the Design Basis Accident conditions is 255.3°F (Section 3.8.1.3.1.9). The governing load case for the accident conditions is the Load Combination 14 of Table 3.8.1-2. The maximum liner strain due to the accident temperature load, which occurs at Elevation 236, was determined to be 0.0009 in/in after the round off. The total liner strain, which included 0.00055 in/in strain due to other concurrent loads, is equal to 0.00145 in/in. This total strain is well below the allowable strain of 0.005 in/in in accordance with Table 3.8.1-3. The forces in liner studs for the accident conditions were also determined to be within allowables.

3.8.1.4.4.10 Containment Penetrations Analysis

The penetration assemblies are analyzed using the same techniques and procedures used for metal containments, as described in ASME "Boiler and Pressure Vessel Code" Section III, Division 1, Subsection NE, "Class MC Components". The analysis considers concrete confinement of the penetration sleeves, as described in Section 3.8.2.4.1.

Each penetration is provided with an anchorage system capable of transferring pressure loads and other mechanical loads, such as piping restraints, into the concrete. The design allowables for the penetrations are the same as those used in ASME Section III, Division 1. For penetration nozzles which are not continuous through the liner, the liner stress in the through-thickness direction is taken as one-half of that in the as-rolled direction.

The analysis of containment penetrations, designated as Class MC Components, is presented in Section 3.8.2.

3.8.1.4.4.11 General Design Considerations

Design details of the Concrete Containment Structure for flexure, axial, and shear loads, reinforcing steel design requirements (splicing, development length, and anchorages), reinforcing steel fabrication and construction requirements (spacing, cover, tolerances, and bending), and concrete crack control are in accordance with the requirements of ASME Code Section III, division 2/ACI 359 Code.

3.8.1.4.4.12 Special Investigations

a) Cylindrical Wall - Crane Girder Interaction - The polar crane girder is supported on brackets attached to the liner plate. The concrete cylindrical wall and the steel crane girder, which have different thermal expansion coefficients and different thermal gradients during various load combinations, could have differential displacements which could induce large bending moments into the concrete wall and excessive stresses into the crane girder. In order to minimize the interaction forces, moments, and thermal stresses, the crane girder is segmented. The supports of the girder are designed to allow free movement of the crane girder due to the differential thermal gradients and to provide seismic restraint at the same time.

   A three-dimensional finite element analysis is performed to investigate the interaction between the concrete wall and the steel crane girder and to determine the interactive forces, moments, and shears developed in the crane girder, brackets, liner, liner anchorages, and concrete wall. The ANSYS computer program is used for this analysis.

b) Dome Construction Sequence - During construction of the concrete containment dome, the liner plate is used as a form, sustaining the weight of the reinforcing steel and fresh concrete without additional support.

   As shown in Figure 3.8.1-23, placement of the concrete proceeds in successive lifts of 4 to 5 ft. of concrete, with pours up to 20 in. applied symmetrically and continuously around the entire circumference of the dome. The next placement of concrete is added only when the concrete previously placed is strong enough to work in conjunction with the liner plate as a composite section to take the additional construction loads.

   As a result of this construction procedure, additional construction locked-in stresses and displacements occur. In order to account for all of the additive stresses and displacements during construction, and to verify that the allowable stresses and strains in the liner are not exceeded, a finite element analysis is performed, using the NASTRAN computer program. In this finite element analysis, each placement of concrete is modeled in order to account for stresses and strains associated with the additional concrete placed.

3.8.1.5 Structural Acceptance Criteria

The Containment is designed to perform within the elastic range for service loads and is essentially elastic under factored loads.

In order to keep the Containment elastic under service loads and below the range of general yield under factored loads, the allowable stresses and strains specified below are used.

Yield strength reduction factors are used to provide stress margins in order to allow for small variations in homogeneity of material and workmanship.

The tabulated values (d) of yield strength reduction factors, contained in Table 3.8.1-4, are defined as non-dimensional stress limits which are used for designing the containment shell structure against those load combinations specified in Table 3.8.1-2, for both service and factored load combinations.

3.8.1.5.1 Allowable Stresses for the Factored Load Category

3.8.1.5.1.1 Concrete Allowable Stresses

1) Concrete Compressive Stresses

   1) Primary compressive stresses:

Membrane stress	=	0.6	f'c
Membrane plus bending	=	0.75	f'c

   2) Primary-plus-secondary compressive stresses:

Membrane stress	=	0.75 f'c
Membrane plus bending	=	0.85 f'c with a limit of 0.002 strain

The stresses given above in items 1 and 2 are reduced, if necessary, to maintain structural stability.

2) Concrete Tensile Stresses - Concrete tensile strength is not relied upon to resist flexural and membrane tension.

   Table 3.8.1-4 shows the strength reduction factors for concrete.

3) Concrete Shear Stresses - Radial, tangential, peripheral, and torsional shears are considered in the design of the containment structure.

1) Radial Shear - An example of the shear caused by self-constraint of the cylinder and base slab during pressurization of the Containment.

   (a) The nominal shear stress, vc, does not exceed the lesser of:

$$v_c = 3.5\sqrt{f_c'} \quad (15a)$$

$$v_c = \frac{1.9p\sqrt{f_c'}}{0.015} + 2500p\,(\frac{V_u d}{M_u}) \quad (15b)$$

   where Mµ is the applied design load moment at the section under consideration

   for $p < 0.015$

$$v_c = 1.9p\sqrt{f_c'} + 2500p\,(\frac{V_u d}{M_u}) \quad (15c)$$

   for $p \geq 0.015$

   where $(V_u d/M_u)$ does not exceed 1.0

   (b) For sections subjected to membrane composition, either Eq (16) or (17) are used, but v_c shall not be larger than the value given by (18):

$$v_c = 1.9p\sqrt{f_c'} + 2500p\,(\frac{V_u d}{M'}) \quad (16)$$

   where $M = M_u - N_u$ [(4t-d)/8] then M' shall be less than V_ud.

   If M is negative, Eq (18) is used

$$v_c = 2(1 + 0.0005\,Nu/Ag)\sqrt{f_c'} \quad (17)$$

$$v_c = 3.5\sqrt{f_c'}\,\sqrt{1 + 0.002\,N_{u/Ag}} \quad (18)$$

   When Nu = the axial design load normal to the cross section occurring simultaneously with Vu

   Ag = gross area of section

   The units for N_u /Ag are psi.

   (c) For sections subjected to membrane tension, Eq (19) is used with N_u negative for tension:

$$v_c = 2.0\,\sqrt{f_c'}\,(1 + 0.002N_u\,/Ag) \quad (19)$$

2) Tangential Shear - An example is the shear force resulting when the Containment is subjected to earthquake motion.

The allowable tangential shear force is:

$$V_c = v_c bt \tag{20}$$

Where:t = thickness of concrete section

a) The tangential shear stress, v_c, carried by the concrete does not exceed 40 psi and 60 psi for load combinations 11 and 14 respectively, presented in Table 3.8.1-2. When V_u exceeds V_c, diagonal reinforcing is provided.

b) The tangential shear stress v_c, carried by the concrete does not exceed 160 psi for load combinations (6) through (9), presented in Table 3.8.1 2. For these load combinations, a meridional and hoop reinforcing system may be used provided that V_u does not exceed 8.5 $\sqrt{f'c}$. If V_u exceeds this limit a diagonal reinforcing system is provided.

c) The tangential shear stress, V_c, carried by the concrete does not exceed 80 psi for service load combinations presented in Table 3.8.1-2. For these load combinations a meridional and hoop reinforcing system may be used provided that V_u does not exceed 4.2 $\sqrt{f'c}$. If V_u exceeds this limit a diagonal reinforcing is provided.

3) Peripheral Shear

(a) The peripheral or punching shear stress taken by the concrete on the assumed failure surface does not exceed Vc as obtained below:

$$V_{ch} = 4\sqrt{f'c}\,\sqrt{1 + \left(f_m / 4\sqrt{f'c}\right)} \tag{21}$$

$$V_{cm} = 4\sqrt{f'c}\,\sqrt{1 + \left(f_h / 4\sqrt{f'c}\right)} \tag{22}$$

where:

V_{ch} = the allowable shear stress on a failure surface perpendicular to a meridional line.

V_{cm} = the allowable shear stress on a meridional failure surface perpendicular to the plane of the shell.

f_m = membrane stress in the meridional direction, compression is positive.

f_h = membrane stress in the hoop direction, compression is positive.

(b) The value of V_c is calculated as a weighted average of V_{ch} and V_{cm}. For a circular failure surface, V_c is the average of V_{ch} and V_{cm}.

The failure surface for peripheral shear is considered to be perpendicular to the surface of the Containment and located so that its periphery is at a distance d/2 from the periphery of the concentrated load or reaction area.

For failure due to impact loads, local areas for missile impact are defined as having a maximum diameter equal to 10 times the mean diameter of the impacting missile, or 5 $\sqrt{t}$ plus the mean diameter of the impacting missile (where t is defined as the total section thickness in feet), whichever is smaller.

4) Torsion - The shear stress taken by the concrete resulting from pure torsion does not exceed vct as calculated from the following equation:

$$v_{ct} = 6\sqrt{f_c' \sqrt{1 + \frac{fh+fm}{6\sqrt{f_c'}} - \frac{fh\,fm}{6\sqrt{f_c'}}}} \qquad (23)$$

5) Brackets and Corbels - These provisions apply to brackets and corbels having a shear span to depth ratio, a/d, of unity or less. The distance d is measured at a section adjacent to the face of the support but is not taken greater than twice the depth of the corbel or bracket at the outside edge of the bearing area.

(a) The shear stress does not exceed:

$$v_u = \left[6.5 - 5.1\sqrt{\frac{N_u}{V_u}}\right]\left[1 - 0.5\left(\frac{a}{d}\right)\right] \times \left\{1 + \left[64 + 160\sqrt{\left(\frac{N_u}{V_u}\right)^3}\right]\rho\right\}\sqrt{f_c'} \qquad (24)$$

where $\rho = A_s/B_d$ does not exceed 0.13 f_c'/f_y and N_u/V_u is not taken less than 0.20, and where N_u is the design tensile force on a bracket or corbel acting simultaneously.

(b) When provisions are made to prevent tension due to restrained shrinkage and creep so that the member is subject to shear and moment only, the shear stress does not exceed

$$v_u = 6.5\,[1 - 0.5\,(a/d)]\,[1 + 64\,\rho v]\,\sqrt{f_c'} \qquad (25)$$

where $\rho v = (A_s + A_{vh})/bd$ but is not greater than 0.20 f_c'/f_y, and A_{vh} does not exceed A_s.

(c) Closed stirrups or ties that are parallel to the main tension reinforcement and have a total cross-sectional area A_{vh} not less than $0.50A_s$ are uniformly distributed within two-thirds of the effective depth and adjacent to the main tension reinforcement.

(d) The ratio $\rho = A_s/bd$ is not less than 0.04 f_c'/f_y.

d) Concrete Bearing Stresses - Bearing stresses do not exceed 0.6 f_c' except as provided below:

1) When the supporting surface (A_2) is wider on all sides than the loaded area (A_1), the permissible bearing stress on the loaded area may be multiplied by $\sqrt{\frac{A_2}{A_1}}$, but this factor may not exceed two.

2) When the supporting surface is sloped or stepped, A_2 is taken as the area of the lower base of the largest frustum of a right pyramid or cone contained wholly within the support, with its upper base as the loaded area and side slopes of one vertical to two horizontal.

3.8.1.5.1.2 Reinforcing Steel Allowable Stresses

a) Reinforcing Steel Tensile Stresses

1) Average tensile stress is 0.9 f_y.

2) The design yield strength of the reinforcement is 60,000 psi.

3) The tensile strain may exceed yield when the effects of thermal gradients through the concrete section are included, provided that the temperature-induced forces and moments reduce as yielding in the reinforcement occurs and the increased concrete cracking does not cause deterioration of the Containment. Maximum tensile strain is limited to twice the corresponding yield strain.

b) Reinforcing Steel Compressive Stresses

1) For load-resisting purposes, the allowable stress is 0.9 f_y.

2) The strains may exceed yield when acting in conjunction with the concrete if the concrete requires strains larger than the reinforcing yield to develop its capacity.

Table 3.8.1-4 shows the allowable stresses for reinforcing steel.

3.8.1.5.2 Allowable Stresses for the Service Load Category

3.8.1.5.2.1 Concrete Allowable Stresses

a) Concrete Compressive Stresses

1) Primary compressive stresses (as defined in Section CC-3136 of the ASME Code Section III, Division 2/ACI 359 code)

Membrane stress = 0.3 f_c'

Membrane stress for load combinations including wind or earthquake = 0.40 f_c'

2) Primary-plus-secondary compressive stresses (as defined in Section CC 3136 of the ASME Code Section III, Division 2/ACI 359 code)

Membrane stress = 0.45 f_c'

Membrane plus bending = 0.6 f_c'

3) Local compression at discontinuities and in the vicinity of liner anchors = 0.6 f_c'

b) Concrete Tensile Stresses - Concrete tensile strength is not relied upon to resist flexural and membrane tension.

c) Concrete Shear Stresses - The allowable concrete stresses and the limiting maximum stresses for shear and torsion are 50 percent of the values given for factored loads, except for the following, in which 67 percent of the values given for factored loads are used:

1) temporary pressure loads during test conditions.

2) thermal loads combined with other loads, provided that the section thus required is not less than that required for the combination of the other loads in the loading combination.

The computed membrane stress on the gross section resulting from service loads are multiplied by 2 and substituted for N_u/A_g, f_m, or f_h invoking the provisions of Sections 3.8.1.5.1.1 (c) 1), c) 3), and c) 4)).

d) Concrete Bearing Stresses - The allowable stresses for bearing are 35 percent of the stresses given in Section 3.8.1.5.1.1d).

3.8.1.5.2.2 Reinforcing Steel Allowable Stresses

a) Reinforcing Steel Tensile Stresses

1) average tensile stress= $0.5f_y$

The values given above may be increased by 33-1/3 percent when temperature effects or temporary pressure loads during test conditions are combined with other loads.

b) Reinforcing Steel Compression Stresses

1) For load-resisting purposes, the allowable stress is 0.5 f_y.

2) The stress may exceed that given in Item b.1 for compatibility with the concrete, but this stress may not be used for load resistance.

3.8.1.5.3 Allowable Stresses and Strains for Liner Plate and Anchorages

3.8.1.5.3.1 Liner Plate Allowable

The allowable stresses and strains of the liner plate for construction, service, and factored loads are presented in Table 3.8.1-3.

3.8.1.5.3.2 Liner Anchors Allowable

The allowable forces and displacements of the liner anchors for service and factored load combinations are presented in Table 3.8.1-3.

3.8.1.5.4 Concrete Containment Design Considerations

Assumptions, details, and procedures used in the design for flexure, axial, and shear loads are in accordance with the requirements of ASME Code

Reinforcing steel requirements regarding splices, development length, hooks, anchorages, and cover are in accordance with the requirements of ASME Code Section III, Division 2/ACI 359 Code, Section CC 3530.

The requirements for crack control are in accordance with Section CC 3534 of ASME Code Section III, Division 2/ACI 359 Code.

Concrete temperatures do not exceed the values indicated in the ASME Code Section III, Division 2/ACI 359 Code Section CC 3440(b) for accident or short term loading.

Corrosion protection for the reinforcing steel in the containment structure is provided by positioning reinforcing steel to allow clearance between the steel and any concrete face on the containment wall in accordance with ASME Code Section III, Division 2/ACI 359 Code. The alkaline environment of the concrete adequately protects embedded steel parts from corrosion.

Exposed surfaces of the liner walls, domes, air lock, and hatch are protected against corrosion. After suitable surface preparation, rust-inhibiting base coat is applied. Finish coats are nonmetallic with smooth nonporous surfaces suitable for loss-of-coolant accident conditions. Surfaces in contact with concrete are not painted because of the alkaline environment of the concrete.

The radiation sources used for the original plant design and analysis of the shielding requirements are based on the core power level (2900 MW) for each Unit. These are given in Section 12.2.1 and include radiation sources for all phases of plant operation including full power operation, shutdown conditions, and refueling operations, and for various postulated accidents. They include the neutron and gamma fluxes outside the reactor vessel, the reactor coolant activation, fission and corrosion product activities, deposited corrosion product sources on reactor coolant equipment surfaces, spent fuel handling sources, and postulated core meltdown sources. In addition, radiation sources for various auxiliary systems are also tabulated.

The Containment is a reinforced concrete structure with a cylindrical wall 4-1/2 feet thick and a 2-1/2 feet thick dome. In conjunction with the primary and secondary shields, the concrete containment structure limits the radiation level outside the Containment from all sources inside the Containment to no more than 0.25 mrem/hr. at full power operation, based on the original plant design.

The concrete containment structure provides protection to plant personnel from radiation sources inside the Containment following a Design Basis Accident (DBA).

3.8.1.6 Materials, Quality Assurance, and Special Construction Techniques

3.8.1.6.1 Materials

The materials for the Concrete Containment Structure and foundation mat are in accordance with Article CC-2000 of the ASME Code Section III, Division 2/ACI 359 Code, and as specified hereunder. The materials are selected so that they are compatible with both the normal operating environment and the post-accident conditions described in Section 3.11.1. Exceptions to the ASME Code Section III, Division 2/ACI 359 Code are listed in Appendix 3.8A.

a) Cement - Cement conforms to the requirements of ASTM C150, Specifications for Portland Cement, Type II, with the exceptions listed in Appendix 3.8A. Cement is produced and tested by the manufacturer at intervals in accordance with ASTM C-150.

In addition to the tests required of the cement manufacturers, the following tests are performed by CP&L, or an organization designated by CP&L, once every six months:

1) ASTM C-114 - Chemical Analysis

2) ASTM C-115 - Fineness of Portland Cement by the Turbidimeter or ASTM C-204 - Fineness of Portland Cement by Air Permeability Apparatus

3) ASTM C-151 - Autoclave Expansion of Portland Cement

4) ASTM C-191 - Time of Setting of Hydraulic Cement by Vicat Needle

5) ASTM C-109 - Compressive Strength of Hydraulic Cement Mortars

6) ASTM C-190 - Tensile Strength of Hydraulic Cement Mortars

During construction, if cement has been in storage at the site for 6 months, the following tests are performed by CP&L prior to further use of the cement to check storage environment effects on the cement characteristics:

7) ASTM C-191 - Time of Setting of Hydraulic Cement by Vicat Needle.

8) ASTM C-109 - Compressive Strength of Hydraulic Cement Mortars (using 2 in. (50 mm) cube specimens)

Table 3.8.1-5 shows the summary of acceptance test results for the qualification of cement. The in-process test results are maintained as permanent records in the QA vault.

During construction, the establishment of a new cement source was necessary, and the acceptance tests were performed to approve the cement supplier. These tests were reviewed, approved, and maintained in the QA vault.

b) Aggregates - Aggregates conform to the requirements of ASTM C-33, Specifications for Concrete Aggregate, with the exceptions listed in Appendix 3.8A.

The aggregate is tested by the supplier for gradation and fineness modulus every 500 tons and for specific gravity and absorption every 5000 tons. In addition, aggregate used for concrete for the Concrete Containment Structure is tested by CP&L, or an organization designated by CP&L, during concrete production for the requirements and respective frequencies tabulated below:

Requirements	Test Method	Frequency
1) Gradation	ASTM C136	Once daily during production(*)
2) Moisture Content	ASTM C566	Twice daily during production
3) Material finer than #200 Sieve	ASTM C117	Daily during production
4) Organic Impurities	ASTM C40	Daily during production
5) Friable Particles	ASTM C142	Monthly during production
6) Lightweight Particles	ASTM C123	Monthly during production
7) Specific Gravity and Absorption	ASTM C127 and/or ASTM C128	Monthly during production
8) Los Angeles Abrasion	ASTM C131 or ASTM C535	Every 6 months
9) Potential Reactivity	ASTM C289	Every 6 months
10) Soundness	ASTM C88	Every 6 months
11) Water Soluble Chlorides	ASTM D1411Monthly during production	

(*)Twice daily during production if more than 200 cu.yds. of concrete are placed.

Tables 3.8.1-6 through 3.8.1-11 shows the summary of acceptance tests for the qualification of aggregate. The in-process test results are maintained as permanent records in the QA vault.

During construction, the establishment of a new aggregate source was necessary, and the acceptance tests were performed to approve the aggregate supplier. These tests were reviewed, approved, and maintained in the QA vault.

c) Water - Mixing water conforms to the requirements of Article CC 2223 of the ASME Section III, Division 2/ACI-359 Code.

Water used in concrete mixing is sampled, tested, and analyzed initially for use in trial mixes and monthly thereafter for use in production concrete by CP&L, or an organization designated by CP&L, to assure conformance with the following limits and tests:

1) The mixing water, including that contained as free water in aggregate, does not exceed more than 250 ppm of chlorides as Cl- as determined by ASTM D512, "Chloride Ion in Industrial Water and Industrial Waste Water." The water-soluble chloride content of the aggregate is established by the methods described in ASTM D-1411, "Water Soluble Chlorides Present as Admixes in Graded Aggregate Road Mixes."

2) Sulfates 1000 ppm Maximum

3) The total solids content of the mixing water does not exceed 2000 ppm as measured by American Public Health Association "Standard Method for Determination of Total Solids."

4) In addition to the above, the water is tested monthly in accordance with the indicated tests.

Test Method	Requirement
ASTM C109	Effect on Compressive Strength
ASTM C191	Setting Time
ASTM C151	Soundness
ASTM D512	Chlorides
APHA 208*	Total Solids

*Standard Methods 14th Edition, 1975, American Public Health Association.

Table 3.8.1-12 shows the summary of acceptance tests for the qualification of water. The in-process test results are maintained as permanent records in the QA vault.

d) Admixtures - Where necessary, admixtures are added to entrain air and increase workability, while reducing the water-cement ratio and retarding the initial set time. The particular admixtures utilized are determined by conducting tests to ensure compliance with Article CC 2224 of ASME Code Section III, Division 2/ACI 359 Code.

Admixtures are used for all concrete construction in accordance with the following requirements and tested by the supplier at intervals conforming with ASTM C-260 and ASTM C-494.

1) Air Entraining Agents - Air entraining agents conform to ASTM C-260 and are used in proportions so that air-entrainment specified in ACI-318 is produced, as determined by ASTM C-138, C-233, and C-173 or C-231. In order that proportions may be adjusted to produce the specified percentage of air under varying conditions, the agent is not combined with the cement or other admixtures prior to batching.

2) Water Reducing Agents - Water reducing agents used in the concrete conform to ASTM C-494. Final approval of the admixture is contingent upon satisfactory tests with the cement and aggregates used in the work. A set retarding, water reducing agent is used during hot weather in accordance with ACI-305.

Flyash, if used in concrete, conforms to ASTM-C-618, Class F, and is tested in accordance with ASTM C-311 for every 100 tons of flyash utilized. Flyash does not exceed 25 percent, by weight, of cement in the final mix. Concrete produced with flyash meets all of the requirements specified for standard concrete.

Table 3.8.1-13 shows the summary of acceptance tests for the qualification of admixtures. The in-process test results are maintained as permanent records in the QA vault.

e) Cement Grout - Cement, aggregate, water, and admixtures for grout conform to the requirements stipulated above. The proportions of materials are based upon trial mixes using the same type and brand of ingredients as is used for construction to meet the specified requirements of consistency, shrinkage, and compressive strength. The tests are performed in accordance with ASTM C-109 and Corps of Engineers methods CRD-C-79 and CRD-C-588-76.

f) Concrete - Structural concrete for the Containment and foundation mat is specified to have a minimum design compressive strength of 5000 psi (Class X), or 4000 psi (Class AA), at 28 days after placing. The concrete mixes yield a unit air-dry weight of at least 136 lb. per cu. ft. at 28 days, in accordance with ASTM C-642.

The design of concrete mixes is in accordance with ACI 211.1-74 "Recommended Practice for Selecting Proportions for Normal and Heavy Weight Concrete," and in accordance with Article CC-2232 of the ASME Code Section III, Division 2/ACI 359 Code. The previously specified ingredients are used to obtain material proportions for the specified concrete.

During construction, minor modifications of design mixes may be necessitated by variations in aggregate gradation or moisture content.

Concrete construction procedures, including stockpiling, storing, batching, mixing, conveying, depositing, consolidating, curing, and construction joint preparation are in accordance with the provisions of Article CC-4200 of the ASME Code Section III, Division 2/ACI 359 Code. SHNPP complies with the requirements of NRC Regulatory Guide 1.55, with the clarifications described in Section 1.8.

g) Reinforcing Steel

1) Reinforcing Bars - Reinforcing bars are new billet steel in accordance with ASTM A-615 Grade 60 (60,000 psi minimum yield strength). When called for on the design drawings, weldable grade reinforcing steel in accordance with ASTM A706 is used. The reinforcing steel and Cadweld splice material conforms to the requirements of Article CC-2300 of ASME Code Section III, Division 2/ACI 359 Code, with the exceptions listed in Appendix 3.8A.

   Placing and splicing of No. 11 and smaller bars meet the requirements of Article CC-4330 of ASME Code, Section III, Division 2/ACI 359 Code.

   At least one full diameter reinforcing steel sample of each bar size is tested by the reinforcing steel supplier for each 50 tons or fraction thereof of reinforcing bars produced from each heat. No specific method of sample selection is imposed upon the reinforcing steel supplier. These samples are tested based upon ASTM A-615 specifications. All requirements of NRC Regulatory Guide 1.15 are complied with and the material also conforms to ASME Section III, Division 2/ACI 359 Code, except as noted in Appendix 3.8A.

   All samples are tested for:

   - Tensile yield strength
   - Tensile ultimate strength
   - Elongation in 8 in.

- Unit Weight

Inspections are performed as necessary to verify compliance with specifications.

2) Mechanical Splicing - No. 18 reinforcing bars are spliced with mechanical (Cadweld) splices in accordance with the requirements of NRC Regulatory Guide 1.10, with the clarification and exceptions described in Section 1.8 and Appendix 3.8A. The Cadweld inspection program is also in conformance with NRC Regulatory Guide 1.10.

The average tensile strength of the splices are equal to or greater than the specified ultimate tensile strength of the rebar. The minimum acceptable tensile strength of any splice is 125 percent of the specified minimum yield strength for the particular bar size and ASTM specification.

All completed splices are visually inspected at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Splices that fail to pass the visual inspection are discarded and replaced, or repaired by welding. Splices that have been discarded are not used for tensile testing.

The splice samples are either production or sister splices for straight bars, and they are straight sister splices for all curved bars. Selected splices are tested in accordance with the following schedule for each position, bar size and grade of bar and for each splicing crew as follows:

a) Test frequency where only production splices are tested:

(1) 1 out of first 10 splices

(2) 1 out of next 90 splices

(3) 2 out of the next and each subsequent unit of 100 splices

b) Test frequency where combinations of sister and production splices are tested:

(1) 1 production splice of the first 10 production splices

(2) 1 production and 3 sister splices for the next 90 production splices

(3) 1 splice, either a production or sister splice, for the next and subsequent units of 33 splices. At least one-fourth of the total number of splices tested are production splices.

Straight sister splices are substituted for production samples for splicing sleeves arc welded to structural steel elements.

To be acceptable, sound nonporous filler metal must be visible for the full circumference at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed 1/4 in. from the end of the sleeve due to the packing material. Such indentation is not considered as a poor fill.

The following reasons constitute cause for visual rejection of splices:

1) Slag in the tap hole where the slag exceeds the thickness of the sleeve's wall.

2) Spongy appearance of the filler metal caused by gas blowout.

3) Void areas for each end of splices in any position exceeding the allowable values tabulated below:

Bar Size	Allowable Void Area (Sq. In.)
9	1.02
10	1.03
11	1.53
14	2.15
18	3.00

Joints which do not meet the visual acceptance standards are rejected and either completely removed and replaced, or repaired by welding.

Welding is done by the manual shielded metal arc (SMA) process.

The welding electrode for joining the reinforcing bar to the splice sleeve conforms to AWS Specification A.5.5 Classification E 8018-B2 1/8" or 5/32" diameter. Electrodes for joining the splice sleeves to structural steel components conform to AWS A5.1 Classification E 7018.

All rust, scale, oil, grease, dirt, or other foreign substances are removed from the areas to be welded. All degreasing is done by swabbing the weld area with acetone or other approved solvent or cleaner. No residual cleaning compounds are left on the surface prior to welding.

The welding current is direct current with the electrode positive (reverse polarity). The base material is preheated to 300 F minimum and an interpass temperature of 300 F minimum is maintained during welding.

Amperages and voltages are in accordance with electrode manufacturer's recommendation.

All slag, flux, or foreign materials remaining on any bead of welding are removed before laying down the next or successive bead. Stress relieving is not required.

After completion of welding, a visual inspection is made for the presence of cracks, surface porosity, slag inclusions, undercut, and inadequate weld size.

For test sample splices from the Containment Building that fail to meet the tensile test acceptance standards, the following procedures are used:

1) If any production or sister splice used for testing fails to meet the strength requirements and failure occurs in the bar, the cause of the bar break is investigated. Any necessary corrective action affecting splice samples are implemented prior to continuing the testing frequency.

2) If the running average tensile strength of 15 consecutive samples fails to meet the tensile requirements, splicing is halted. The cause(s) of the failure are investigated and the necessary corrective action(s) are taken. When splicing is resumed, the splicing test frequency is started anew.

3) Welded Splices - Welded Splices, if used, comply with Regulatory Guide 1.94.

h) Steel Liner Plate - The fabrication, testing, and examination of the steel liner is in conformance with Articles CC-4500 and CC-5500 of ASME Code Section III, Division 2/ACI 359 Code, with the exceptions listed in Appendix 3.8A.

The steel liner plate is carbon steel conforming to ASTM A 516 Grade 70. This steel has a minimum yield strength of 38,000 psi and a minimum ultimate strength of 70,000 psi with minimum elongation of 21 percent. Liner plates comply with the requirements of the applicable ASME Code material specification for low temperature service. The impact testing minimum requirement is as follows:

1) As specified in ASME Code Section III, Division 1, paragraph NE 2320, for procurement performed prior to April 29, 1977.

2) As specified in ASME, Section III, Division 2/ACI 359 Code, paragraph CC 2520, for procurement performed after April 29, 1977.

Charpy V-notch specimens (SA-370 Figure 11 - Type A) are used for all impact testing at a maximum temperature of 0 F.

Welding materials (electrodes, filler metals, and/or inserts) are selected in conformance with the code requirements. Only those types of low hydrogen electrodes and combinations of wire and flux that produce welds that at least meet the impact values of the parent material, as specified, are permitted in the construction.

All welding materials are certified (Actual Test Results) to meet the impact test requirements of ASME SFA-5.1. Weld metal test plates are certified to meet impact tests in accordance with the applicable Subsection of the ASME Code Section III, Division 2/ACI 359 Code, employing a maximum temperature of 0 F and using the same material and thickness range as defined by the ASME Code Section III, Division 2/ACI-359 Code.

In manual shielded metal arc-welding, the electrodes are of the low hydrogen type, are analytically compatible with the base metal, and are such that the mechanical properties of the resulting welds meet the full requirements for mechanical properties of the base metal. Electrodes conforming to ASME SFA 5.5, Classification E 7010, are permitted for making test channel attachment welds only. All low-hydrogen electrodes are stored in ovens at 200 to 300 F for approximately 8 hours immediately prior to use. Electrodes removed from storage ovens are not exposed to ambient temperature for more than 4 hours. Electrodes removed from ovens and not used within a 4 hour period are returned to the ovens for 8 hours of redrying at

200 to 300 temperature. The electrode manufacturer's recommended practices are acceptable as an alternate, provided they are proven to yield a moisture content of less than 0.6 percent for E 7018 electrodes when they are consumed.

The procedures, design, methods, and sequence of welding are reviewed prior to performance of welding. All full penetration groove welds made without backing have the root layer gouged, chipped, or ground to sound metal prior to welding the second side. All vertical welding proceeds uphill, except for the following, which can be welded either uphill or downhill:

1) Capping or wash passes

2) Shielded metal arc welding using E 7010 electrodes

3) Double-welded groove joints in the containment liner

4) The remaining weld layers beyond the root of single-welded groove joints in the containment liner.

Prior to welding, all surfaces are properly prepared to be free of oil, grease, rust, pitting, scale, and deleterious matter to ensure satisfactory welding. All protective coatings, if present, are chemically or mechanically removed from all areas within 2 in. of a seam to be welded. Weldable primers, such as Deoxaluminite, need not be removed when welding is performed according to procedures which are qualified for welding over such coatings.

All automatic welding is done by the submerged arc process or the externally supplied gas-shielded arc process. The welds are analytically compatible with the base metal and have mechanical properties that meet the full requirements of the mechanical properties of the base metal.

Preheat at 200 F minimum is applied to all material whose thickness exceeds 1-1/4 in. For material whose thickness is less than 1-1/4 in. preheat at 100 F is applied if the base metal temperature falls below 50 F. The above requirements are minimum unless otherwise specified in ASME Code Section III, Division 2, ACI 359 Code, Table CC-4552-2.

Thermal post weld heat treatment is performed as required by, and in accordance with, the ASME Code Section III, Division 2/ACI 359 Code. Post weld heat treatment procedures are reviewed by the Architect-Engineer. Parts of the liner furnished prior to April 29, 1977, comply with the post-weld heat treatment requirements of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Subsection NE, Winter 1971 Addenda, which requirements are equal to or greater than those of Division 2 of the ASME Code Section III, Division 2/ACI 359 Code.

All longitudinal and circumferential welds in the liner are full penetration bevel butt type. All welders, welder operators, and welding procedures are qualified in accordance with and meet the requirements of, Section IX of the ASME Code. All accessible seam welds are subject to spot radiographic inspection in accordance with ASME Section III, Division 2/ACI 359 Code, paragraph CC-5531. Butt welds are examined per ASME Code Section III Division 2/ACI 359 Code, paragraph CC-5521. Radiographic examination is performed in accordance with the techniques prescribed in Section V, Article 2 of the ASME Boiler and Pressure Vessel Code, Winter 1971 Addenda for services rendered prior to April 29, 1977, such as the shear key and sump pit assemblies of Units 1 and 2. For services rendered subsequent to April 29, 1977, radiographic examination is performed in accordance with the techniques prescribed in Section V, Article 2 of the ASME Boiler and Pressure Vessel Code, Winter 1975 Addenda.

In addition to seam welds with back-up bars, all non-butt and attachment welds to the Containment, except those welds for the leak chase system, non-load bearing plates, and temporary erection attachments, are examined by the magnetic particle or liquid penetrant test per ASME Code Section III, Division 2/ACI 359 Code, paragraphs CC-5521, CC-5522, and CC-5523. For magnetic particle or liquid penetrant inspections performed prior to April 29, 1977, the procedures and acceptance criteria conform to Appendix VI and VIII of ASME Boiler and Pressure Vessel Code, Winter 1971 Addenda. For magnetic particle or liquid penetrant inspections performed after April 29, 1977, the procedures conform to Section V, Articles 7 and 6, respectively, ASME Boiler and Pressure Vessel Code Winter 1975 Addenda. Acceptance criteria for the magnetic particle or liquid penetrant examination is in accordance with ASME Code Section III, Division 2/ACI 359 Code, Paragraphs CC-5545 and CC-5544, respectively.

The root pass and final weld layer for attachments to the Containment using full penetration tee welds are examined by the magnetic particle or liquid penetrant method. In addition, the completed tee weld, where accessible, is ultrasonically inspected in accordance with ASME Section III, Division 1, Paragraphs NE-5111 and NE-5330.

Those areas of liner plates which are loaded during service by load bearing plates (loaded in the through thickness direction as defined in Paragraphs CC-3740 and CC-3750 of Section III, Division 2) are examined by the straight-beam ultrasonic method in accordance with SA-578 and ASME Code Section III, Division 2/ACI 359 Code, Paragraph CC-2533.

The criteria for workmanship and visual quality of welds is in accordance with code requirements, as well as the following:

1) Each weld has the minimum specified size throughout its full length. Each weld is free of linear defects such as slag, cracks, pinholes, and excessive undercut and rounded indications such as pinholes which exceed the acceptable limit as permitted by Paragraph CC-5544.2. In addition, the layer of welds is free of coarse ripples, arc strikes, irregular surface, non-uniform bead pattern, high crown, and deep ridges or valleys between beads. Controlled peening, except for the root pass and final weld bead layer, has been reviewed and approved.

2) Butt welds are multipass construction, slightly convex, of uniform height, and full penetration.

3) Fillet welds are of the specified size, with full throat and legs of uniform length.

Vacuum box testing of the liner is performed in accordance with the applicable requirements of ASME Code Section III, Division 2/ACI 359 Code, paragraph CC-5000. After completion of a successful vacuum box and radiography tests, and subsequent repair and retesting of any defects found, the welds are covered by test channels as indicated on the design drawings. A test channel strength and simultaneous leakage (pressure decay) test is then performed by applying 51.75 psig air pressure to the test channels for at least two hours, after which all welds are solution film tested. For those cases where a vacuum box test is performed on the liner seam welds, these welds are not solution film tested a second time. Where there is any indicated loss of channel test pressure within the two hour period, not allowed by accepted test procedures, the channel sections under test are determined to contain defects. Such defects are repaired. Compensation for change in ambient air temperature is made if necessary. Leak testing is performed in accordance with the requirements of ASME Code Section III, Division 2/ACI 359 Code, paragraph CC-5535.2.

All testing connections and accessories, as applicable, are permanently left in place with all connections properly sealed.

After fabrication, surfaces are cleaned in accordance with SSPC-SP-1 "Solvent Cleaning" to remove oil, grease, dirt, loose rust, loose mill scale, and other foreign substances if necessary before mechanical cleaning is started.

A shop coating of 6548/7107 Epoxy White Primer as manufactured by Keeler & Long, Waterbury, Conn. is applied by the liner manufacturer according to the paint manufacturer's instructions, over steel which has been prepared for coating by commercial Blast Cleaning SSPC-SP-6 as described by the Steel Structures Painting Council. In certain instances, SSPC-SP10 "Near White Blast Cleaning" has been permitted in lieu of SSPC-SP-6.

The corresponding topcoat for this primer is applied in the field and consists of Keeler and Long 7475 Epoxy Enamel Topcoat.

The above coating system meets the criteria outlined in ANSI Standard N512-1974, "Protective Coatings (Paints)for the Nuclear Industry" and ANSI Standard N101.2, 1972 "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities."

Application of the above coating system meets the intent of ANSI N101.4 "Quality Assurance for Protective Coatings Applied to Nuclear Facilities" and Reg. Guide 1.54 "Quality Assurance Requirements for Protection Coatings Applied to Water-Cooled Nuclear Power Plants."

Test results, as indicated in the following documents, were utilized in the selection of these paint systems:

a) ORNL-3589 "Gamma Radiation Damage and Decontamination Evaluation of Protective Coatings," By G. A. West and C. D. Watson February, 1965.

b) ORNL Log Book A7562, 6/27/77.

c) ORNL-TM-2412 "Design Consideration of Reactor Containment Spray Systems - Part V, Protective Coating Systems," J. C. Griess, T. H. Roco, et al, October, 1970.

d) Keeler and Long, Inc., Publication 78-0810-1.

The areas in which the above coatings meet specified criteria are as follows:

1) Radiation Resistance - The protective coating system used on the containment liner is resistant to radiation exposures which would result from normal plant operation followed by the radiation exposure resulting from a postulated Loss-of-coolant accident with TID-14844 source terms assumed. ANSI Standard N-512-1974 Table 2.1 lists as a guide more than $4.5x10^9$ rads for "severe exposure" radiation resistance. Test results submitted by the

above mentioned manufacturer indicate that their referenced coatings have radiation resistances which fall in these ranges.

2) Decontamination Ability - A total decontamination factor of 440 with a percentage activity removal of 99.8 (Ref: ORNL A7562) was achieved by the protective coating system used for the containment liner. Coating systems indicated above meet this criteria using appropriate procedures.

3) Heat Transfer Characteristics - Protective coating systems are required to have a heat transfer coefficient range of 1,000 to 3,000 BTU-mil./hr.-ft.^{2} F. The systems indicated above meet this requirement. Effects of the liner coating systems on containment post LOCA transients are not significant.

4) Hydrogen Generation - Coating systems indicated above have no zinc in their composition. Consequently no hydrogen generation will result from contact between the containment spray solution and the coatings.

5) Temperature, Pressure, and Humidity Conditions - Qualification testing of the coating systems are performed for the coating manufacturer by an independent laboratory. The procedures used in the qualification tests and the evaluation standards applied to the test are specified in ANSI Standard N-101.2-1972.

The tests performed meet the temperature, pressure, and humidity conditions calculated for the Shearon Harris Nuclear Power Plant post-accident containment.

The completed liner is constructed to the following tolerances:

1) The difference between the maximum and the minimum diameter at a specified elevation does not exceed 0.65 ft. and the radius from the theoretical centerline of the Containment does not have a minus dimension in excess of 2-1/4 in. or a plus dimension in excess of 3 in. These measurements are taken in at least 26 different plots at the top of the wind girder. Wind girders are located approximately at the center of each course of plate and are not located more than 10' 0 apart in the vertical direction unless approved by the Architect Engineer.

2) Deviation from a 10 ft. straight edge placed in the vertical direction between circumferential seams does not exceed 3/4 in. Measurements are taken no closer than 12 in. from a welded seam.

3) The maximum deviation from a straight line or from a true circular or spherical form, measured anywhere on the liner in any direction, does not exceed $\pm$ 1/4 in. in a 14 in. span.

4) Elevations are maintained to within 2 in. of theoretical elevations up to and including the spring line of the dome. Penetration positions are within $\pm$ 1 in. tolerances.

5) Flat spots or local out-of-roundness do not exceed 2 in. in 15 ft.

i) Liner Plate Anchorages - Concrete anchor studs for attachment of the liner plate are Nelson studs of low carbon steel ASTM A108 of a grade suitable for end welding to the liner plate, with automatically timed welding equipment.

Welding details, qualifications, and procedures for steel welding are in accordance with AWS D2.0 for requirements for stud welding for services provided prior to April 29, 1977. For services rendered after this date, stud welding meets the requirements of the ASME Code Section III, Division 2/ACI 359 Code.

In order to determine the tensile and shear capacities of the anchors and the stiffness required for the analysis of the liner plate and its anchorages, a test was performed at Lehigh University, by Fritz Engineering Laboratory, Bethlehem, Pennsylvania, Report No. 200.77.477.1. The results of these tests are discussed in Section 3.8.1.4.

The concrete anchor studs used for the connection of the bottom liner plate are bent Nelson studs 3/8 in. diameter x 4 in. long. The concrete anchor studs used for the connection of the cylindrical wall and dome liner are headed Nelson studs 5/8 in. diameter x 4 in. long.

During construction, the following requirements for testing and inspection are observed:

1) Prior to the start of the stud welding operation, two studs are welded in the same general position to separate pieces of material that are of similar thickness and material as the member. After cooling, each stud is bent at an angle of 30 degrees from its original axis by striking the stud with a hammer. If failure occurs in the weld zone of either stud, the procedure is corrected and two additional studs are successfully welded and tested before any studs are welded to the member. The foregoing testing is performed after any change in the welding procedure. If failure occurs in the stud shank, an investigation is made to ascertain and correct the cause before more studs are welded.

2) Studs bent in testing that show no signs of failure are straightened by hammer blows without heating. Studs attached to the embedded angles, structural tees, and liner plate forming the bottom section of the liner are not straightened after being bent for testing.

3) Studs on which a full 360 degrees weld is not obtained are repaired by adding a 3/16 in. fillet weld in place of the lack of weld, using the shielded metal arc process with low-hydrogen welding electrodes.

4) If the reduction in the height of studs as they are welded becomes less than normal, welding is stopped immediately and not resumed until the cause has been corrected.

5) If visual inspection reveals any stud in which the reduction in height due to welding is less than normal, such stud is struck with a lead hammer, or an approved alternate method, and bent 15 degrees off vertical. Studs that crack in the weld, base metal, or the shank, under inspection or subsequent straightening, are replaced.

6) For studs fastened to penetration sleeves, the first two studs welded to each sleeve, after being allowed to cool, are bent 30 degrees by striking the stud with a lead hammer or an approved alternate method. If failure occurs in the weld zone of either stud, the stud is removed, the procedure is corrected, and two additional studs are successfully welded and tested on a sister plate before further studs are attached to the sleeve. Two consecutive

studs are then welded to the member, tested, and found satisfactory before any more production studs are welded to the sleeves. Subsequently, a 10 percent random sample of the studs on each sleeve are bend tested.

j) Penetration Anchorages and Attachments - For all Type II and Type III penetration sleeves designed in accordance with ASME Code Section III, Division 2/ACI 359 Code, in the portion backed by concrete, the concrete anchorages used to connect the sleeve into the concrete wall are double headed Nelson studs 7/8 in. diameter by 8 in. long.

Fabrication, welding details, and welding qualification procedures for stud welding are in accordance with ASME Code, Section III, Division 2/ACI 359 Code, except as noted in Appendix 3.8A.

Special anchorages are used for all Type I penetration sleeves and components, such as the equipment hatch, personnel air locks and emergency air locks, designed in accordance with ASME Code, Section III, Division 1, Subsection NE. The special anchorages are fabricated from SA 105 materials using accepted manual welding procedures. Fabrication welding details, qualification, and procedures for welding anchorages in accordance with ASME Code, Section III, Division 1, Subsection NE.

In order to determine the tensile and shear capacity of the concrete anchorages and the stiffness required for analysis of the concrete containment interaction with steel penetrations, a test was performed at Lehigh University by Fritz Engineering Laboratory, Bethlehem, Pennsylvania, Report No. 200.77.477.2. The results of these tests are discussed in Section 3.8.2.4.

Special attachments are used for the main steam and feedwater penetration sleeves, which are subjected to excessive rupture loads and which are designed in accordance with ASME Code, Section III, Division 1, Subsection NE. The special attachments are fabricated from material similar to the material used for the penetration sleeves. Accepted manual welding procedures are employed.

Fabrication, welding details, and welding qualification procedures for attachment welding are in accordance with ASME Code, Section III, Division 1, Subsection NE.

In order to determine the tensile and shear capacity of the concrete attachments and the stiffnesses required for the analysis of the concrete containment penetration sleeves interaction, tests were performed at Lehigh University, Fritz Engineering Laboratory, Bethlehem Pennsylvania, Report No. 200.77.477:3. The results of these tests are discussed in Section 3.8.2.4.

k) Structural Steel Members and Attachments - Material for liner plate attachments (load bearing), crane brackets, and structural steel members which are attached to the containment liner are in accordance with the ASME Code Section III, Division 2/ACI 359 Code, as described in Appendix 3.8A.

Crane girders, structural steel, stiffener plates, and similar applications not within the scope of the ASME Code conform to the following:

1) Plate material ASTM-A36 or ASTM-A516 GR70

2) Structural Steel ASTM-A36

The following welding inspections are made:

1) All full penetration butt welds are 100 percent radiographed.

2) All full penetration tee welds are tested by magnetic particle or liquid penetrant test of root pass and final weld layer; ultrasonic tests are performed on completed welds where accessible.

3) Fillet welds joining structural members in which either member is greater than 5/8 in. nominal thickness are inspected by liquid penetrant or magnetic particle methods after the final weld layer is applied. All other fillet welds are inspected visually for unacceptable defects using 5X magnification.

4) The above examinations are performed in accordance with the AWS Code specified in Section 3.8.1.2. As an alternate, the above required examination may be performed in accordance with the ASME Code, as follows:

   a. Radiographic, magnetic particle, and/or liquid penetrant examinations may be performed in accordance with the requirements of the ASME Code, Section V and Section III, Division 2/ACI 359 Code, as specified in Section 3.8.1.6, for services after April 29, 1977.

   b. Ultrasonic examination may be performed in accordance with the requirements of the ASME Code, Section III, Divisions 1 and 2 as described in 3.8.1.6.1 h)2) above.

5) All welders, welder operators, and welding procedures are qualified in accordance with either the requirements of the AWS Code or the ASME Code, Section IX, whichever is applicable.

3.8.1.6.2 Quality Assurance

The overall quality assurance program is in accordance with the Engineering and Construction QA program which was approved by the NRC during the Construction Permit review. Materials testing, fabrication, construction, and construction testing and examination are in accordance with applicable provisions of Articles CC-4000 and CC-5000 of the ASME Code Section III, Division 2/ACI 359 Code. The test methods and frequency of testing for concrete and concrete ingredients conform to the requirements stipulated in the ASME Code Section III, Division 2/ACI 359 Code, with the exceptions listed in Appendix 3.8A.

The services of an independent laboratory were obtained prior to commencing concrete work. This laboratory or CP&L produced control mixes with consistencies satisfactory for the work, using the proposed materials, in order to determine suitable mix proportions that are necessary to produce concrete conforming to the specified type and strength requirements.

Proportions for concrete mixes are based on laboratory or CP&L trial batches made of materials specifically approved for use and from which individual water/cement ratio curves were developed. Mix proportions were selected to ensure maximum workability and conformance with the concrete compressive strength requirements.

Proportions for the laboratory or CP&L trial batches and the subsequent mix adjustments were in accordance with ACI 211.1, "Recommended Practice for Normal and Heavyweight Concrete."

Initially, concrete mix proportions were selected from the appropriate water/cement ratio curves, so that the average compressive strength exceeded f_c', i.e., 5000 psi (Class X) and 4,000 psi (Class A), by 1,200 psi. In addition, proportions were selected so that the air-dried hardened unit weight would not be less than 137 lb./ft.^{3} and the slump and air content would be 4 in. and 4 to 8 percent, respectively. A maximum slump of 8 in. is permitted if superplasticizer mix is used.

The initial mix proportions were used until sufficient test data (concrete cylinders tested in accordance with ASTM C39) become available and an over-design considerably less than 1,200 psi could be established.

New mix proportions were selected based on the water-to-cement ratio curves modified by field tests and newly established over-design strength so that the requirements of Sub-Subparagraph CC-2232.2(b) of the ASME Code Section III, Division 2/ACI 359 Code are complied with.

Tables 3.8.1-14 through 3.8.1-16 show a summary of the acceptance tests for qualification of concrete mixes with compressive strengths of 5000, 4000, and 3000 psi. During construction, requalification of specific concrete mixes was performed due to the establishment of a new aggregate source. These test results are maintained in the QA vault.

For concrete used in the Containment Structure, the properties tabulated below are measured - prior to construction - in accordance with the respective specifications and the applicable conditions noted below:

Property	Specification	Age of Sample (Days)	Temperature (°F)
1. Slump	ASTM C143	NA	NA
2. Compressive Strength	ASTM C39	3, 7, & 28	As per ASTM C39
3. Flexural Strength	ASTM C78	28	As per ASTM C78
4. Splitting Tensile Strength	ASTM C496	28	As per ASTM C496
5. Static Modulus of Elasticity	ASTM C469	28	As per ASTM C469
6. Poisson's Ratio	ASTM C469	28	As per ASTM C469
7. Coefficient of Thermal Conductivity	CRD-C44	28	As per CRD-C44
8. Coefficient of Thermal Expansion	CRD-C39	28	As per CRD-C39
9. Creep of Concrete in Compression (*)	ASTM C512	2, 7, 28, 90 days & 1 yr.	As per ASTM C512
10. Shrinkage (*) Coefficient (Length change of cement mortar and concrete)	ASTM C157	4, 7, 14, & 28 days & 8, 16, 32, & 64 weeks	As per ASTM C157
11. Density (Specific Gravity)	ASTM C642	28	As per ASTM 642

* These tests are concurrent with construction.

Concrete slump, temperature, air content, and mechanical properties examinations are performed on a common sample to establish conformance with the provisions listed above.

Concrete is sampled at the point of delivery into the forms.

The methods used in sampling, making, curing, and testing the concrete samples, either in the field or in the laboratory, are in accordance with the appropriate ASTM Standards and include, but are not necessarily restricted to, the following standards:

ASTM C172 - Standard method of Sampling Fresh Concrete

ASTM C31 - Standard method of Making and Curing Concrete Compressive and Flexural Test Specimens in the Field.

ASTM C192 - Standard Method of Making and Curing Concrete Test Specimens in the Laboratory.

ASTM C39 - Standard Method of Test for Compressive Strength of Cylindrical Concrete Specimens.

ASTM C567 - Standard Method of Test for Unit Weight of Structural Lightweight Concrete.

ASTM C138 - Tentative Method of Test for Unit Weight, Yield, and Air Content (Gravimetric) of Concrete.

Three-day, seven-day, and 28-day tests are made on 6 x 12 in. cylinders. For each design mix, a correlation between three-day, seven-day, and 28-day strengths is made in the laboratory. Soon after a job starts, a similar correlation evolves for samples of concrete taken from the mixer. After that correlation has been established, the results of the 7-day tests may be used as an indicator of the compressive strengths which should be expected at 28 days. If 7-day tests show compressive strengths that are too low, corrective measures are taken at once without waiting for the results of the 28-day tests.

The number of test cylinders made under various conditions are as follows:

		Min. No. of Test Breaks			
	Cylinders	3-Day	7-Day	28-Day	Extra
1) Until final determination of each design mix for each class of concrete placed in any one day.*					
Each 100 cu. yd. or fraction thereof	14	4	4	4	2
2) For each class of concrete of determined mix placed in any one day.					
Each 100 cu. yd. or fraction thereof	4	-	1	2	1

*This is intended to cover only those new design mixes, created by modification of determined design mixes, which have not been proven by the lab tests prior to their placement. The number of cylinders may be reduced to a minimum of four per set if a sufficient number of cylinders (e.g., 100) for the modified design mix has proven the mix to be acceptable.

The extra cylinders are tested if it is necessary to substantiate 7 or 28 day test results.

The concrete cylinders are tested for compressive strength in accordance with ASTM C39. The strength level of the concrete is considered satisfactory if:

a) No individual strength test results falls more than 500 psi below the required class strength at 28 days.

b) The averages of all sets of three consecutive strength test results equal or exceed the required class strength at 28 days.

Each 28-day strength test result is the average of two cylinders from the same sample. The variation between the two cylinders must be not more than five percent of their average. A greater variation requires testing of the third (spare) cylinder to determine the average strength. If the third cylinder strength variation is also greater than five percent of the average, CP&L determines the reason for such a wide variation in test results and rectifies it.

The coefficient of variation for the tests on each mix, as determined in accordance with ACI 214, must not be greater than 15 percent. A greater variation will require a review of concrete batching, mixing, transporting facilities, and procedures to assure a reduction in this coefficient to the required 15 percent or lower.

The slump tests are performed as follows:

a) One slump test is performed for the first batch placed each day, and thereafter for each 50 cubic yards of each class of concrete placed.

b) Slump tests are made on each concrete batch used for test cylinders.

c) Slump tests are made at any time the inspector has reason to suspect that the concrete slumps are not within the allowable tolerances.

The concrete air entrainment content and temperature is taken with each slump test.

The concrete unit weight is determined daily during production, in addition to the slump, air content, and temperature.

The batch plant scales are calibrated to ASTM C 94 standard on a monthly basis.

Mixer uniformity tests to the ASTM C 94 standard are performed initially and every six months.

The evaluation of the test results for concrete are in accordance with ACI 214 and ASME Code Section III, Division 2/ACI 359 Code.

During concrete operations, inspectors at the batch plant witness the mix proportions of each batch delivered to construction, and periodically sample and test the concrete ingredients. The inspectors ensure that a ticket is provided for each batch, which documents the time loaded, actual proportions of the mix, amount of concrete, and the concrete design strength. The cleanliness of trucks and the handling and storage of aggregate are checked by the batch plant inspectors. The concrete batch plant complies in all respects, including provisions for storage and precision of measurements, with ASTM C-94, and National Ready Mixed Concrete Association (NRMCA) - Certification of Ready Mixed Concrete Production Facilities. Water and ice additions, if necessary, are modified as required based on measurements of the moisture content and gradation changes of the aggregate.

Other inspectors at the construction site inspect reinforcing and form placement, make slump tests, make test cylinders, check air content, check concrete temperatures, record weather conditions, and inspect concrete placing and curing. The requirements of Regulatory Guide 1.55 are followed with clarifications described in Section 1.8.

The reinforcing steel bars comply with the requirements of Articles CC-4300 and CC-5300 of ASME Section III, Division 2/ACI 359 Code, with the exceptions listed in Appendix 3.8A. The requirements of Regulatory Guides 1.10 and 1.15, with clarifications in Section 1.8 and Appendix 3.8A, are also followed.

The following inspections are performed:

a) Visual inspection of fabricate reinforcement is periodically performed to ascertain dimensional conformance with specifications and drawings.

b) Visual monitoring of in-place reinforcement is periodically performed by the placing inspector to assure dimensional and locational conformance with drawings and specifications.

3.8.1.6.3 Special Construction Techniques

The recommendations of Regulatory Guide 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures," are not applicable to the Shearon Harris containment. The Concrete Containment Structure (CCS) is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome and a flat base with a recess beneath the reactor vessel, as described in Section 3.8.1.1. No prestressing system is employed in the containment design and construction. However, the following special construction techniques were followed.

a) Concrete construction practices, including stockpiling, storing, batching, mixing, conveying, depositing, consolidating, curing, and the preparation of formwork and construction joints, are in accordance with the provisions of Section CC-4200 of the ASME Code Section III, Division 2/ACI 359 Code with the exceptions listed in Appendix 3.A. The requirements of RG 1.55, with the clarification described in Section 1.8, are also followed. No special construction techniques are utilized in the concrete construction.

   In general, concrete lifts in the wall and dome of the containment structure are placed in approximately 10 ft. and 4 ft. high lifts, respectively. Each lift is constructed in not more than 20 in. layers placed at such a rate that concrete surfaces do not reach their initial set before additional concrete is placed. Past experience indicates that the use of properly controlled concrete mixes and placements not exceeding 20 in. high layers, as described above, followed by careful curing at each lift, controls shrinkage sufficiently to provide the necessary stability in the finished concrete.

   The cylindrical wall liner is used as an interior form for placing of concrete in the wall. The liner is connected to the exterior form as shown on Figure 3.8.1-44. Additional vertical channels are provided on the inside face to minimize liner stresses due to the placement of fresh concrete.

Calculations are made in order to determine the stresses induced into the liner during construction. These "locked-in" stresses are super-imposed on all other mechanical and thermal stresses induced into the liner, using the load combinations from Table 3.8.1-2.

The dome liner is used as the sole support for placing reinforcing steel and concrete in the dome. Calculations are made and a sequence of operations is devised to allow this practice with assurance that the liner will not be in jeopardy of buckling. The sequence of concrete placement in the dome is shown on Figure 3.8.1-23.

b) Temporary Construction Openings - Temporary construction openings are provided in the cylindrical wall of the containment structure. Construction joints are provided around the openings and the concrete surface is sufficiently roughened for proper interlocking of the concrete. The reinforcement extends into the opening for sufficient length to enable splicing of bars.

The wall around the opening is designed to provide the necessary reinforced concrete beam section to span the opening, and to provide the necessary column section on either side of the opening to transfer the loads to the foundation mat.

3.8.1.7 Testing and In-Service Surveillance Requirements

3.8.1.7.1 Structural Integrity Pressure Test

The Concrete Containment Structure is subjected to a preoperational structural proof test after the Containment is complete, with liner, concrete structures, all electrical and piping penetrations, equipment hatch, and personnel locks in place.

While the SHNPP's Containment is a non-prototype Containment, the structural acceptance test is performed in accordance with the procedures outlined in Article CC-6000 of the ASME Code Section III, Division 2/ACI 359 Code for a prototype Containment as augmented by the provisions delineated in Regulatory Guide 1.18.

The internal test pressure is increased from atmospheric pressure to 1.15 times the containment design pressure in five approximately equal pressure increments. The Containment is depressurized in the same number of increments. Measurements are recorded at atmospheric pressures and at each pressure level of the pressurization and depressurization cycles. Concrete crack patterns are recorded at atmospheric pressure both before and immediately after the test and at the maximum pressure level achieved during the test. Instrumentation for these tests consists of taut wire extensometers for longer distance and LVDT (linear variable differential transducers) for shorter distances, with automatic data logging systems to measure deflections. The environmental conditions during the test are measured in a manner and to an extent that permits evaluation of their contributions to the response of the Containment. The test is not conducted under extreme weather conditions such as snow, heavy rain, or strong winds.

In order to determine the complete picture of the overall deflection pattern of the Containment, radial and vertical deflections of the Containment are measured in accordance with ASME Code Section III, Division 2/ACI 359 Code, Article CC-6232. The radial deflections are measured at several points along four meridians spaced around the Containment, including locations with varying stiffness characteristics. Vertical deflections of the Containment are measured at the apex, the springline of the dome, and at two intermediate locations between a point near the apex and the springline on two azimuths.

Figure 3.8.1-45 shows the radial displacement measurement locations and Figure 3.8.1-46 shows the vertical displacement measurement locations.

The radial deflections of the containment wall adjacent to the equipment hatch opening are measured at twelve points, as shown on Figure 3.8.1-45.

The pattern of cracks that exceed 0.01 inch in width before, during, or after the test are mapped in accordance with ASME Code Section III, Division 2/ACI 359 Code, Article CC-6233 near the base-wall intersection, at the midheight of the wall, at the springline of the dome, around the equipment hatch opening, and at main steam and feedwater penetrations, as shown on Figure 3.8.1-50.

Strain measurements in the concrete sufficient to permit a complete evaluation of strain distribution are determined in accordance with the requirements of ASME B&PV Code Section III Division 2/ACI 359 Code Article CC-6134 and as shown on Figures 3.8.1-47, 3.8.1-48, and 3.8.1-49.

As a minimum, the following responses of the Concrete Containment Structure to pressurization are established by the tests:

a) Yielding of conventional reinforcement does not develop, as determined from analysis of crack width, strain gage, or deflection data.

b) No visible signs of permanent damage to either the concrete structure or the steel liner that can be detected.

c) The deflection recovery 24 hours after complete depressurization is 70 percent or more.

d) The measured maximum deflections at points of maximum predicted deflection does not exceed predicted values by more than 30 percent. This requirement is waived if the 24-hour recovery is greater than 80 percent.

3.8.1.7.2 Initial and In-Service Leakage Rate Tests

Initial and in-service leakage rate tests are discussed in Section 6.2.6.

3.8.2 STEEL CONTAINMENT

This section is not applicable to the SHNPP since a steel-lined reinforced concrete containment is used (see Section 3.8.1.1).

However, certain steel components in the containment system are designed and fabricated in accordance with the ASME B&PV Code Section III, Division 1, Subsection NE "Class MC Components." These components, as described in Section 3.8.1.1, consist of the following:

a) equipment hatch.

b) personnel air lock.

c) emergency air lock.

d) Type I mechanical penetrations.

e) all portions of the containment penetration sleeves not backed by concrete.

This section concerns the requirements of the ASME B&PV Code, Section III, Division 1, Subsection NE, "Class MC Components." The equipment hatch, personnel air lock, emergency air lock, Type I mechanical penetrations and all other penetration sleeves not backed by concrete are not stamped because they are an integral part of an unstamped containment structure (see Appendix 3.8A).

3.8.2.1 Description of ASME Class MC Components

Access into the Concrete Containment Structure is provided by an equipment hatch, personnel air lock, and emergency air lock.

The containment penetration assemblies provide for the passage of process, service, sampling and instrumentation system piping through the containment wall, and for the transfer of new or spent fuel between the Containment and the Fuel-Handling Building, while providing a leak tight seal to maintain containment integrity. It is the principal objective that these assemblies retain their integrity not only during a postulated accident condition but also during the entire operational life of the plant.

3.8.2.1.1 Equipment Hatch

One equipment hatch, a welded steel assembly with an overall inside diameter of 24 ft. - 0 in., is provided.

A 15 ft. - 0 in. inside diameter bolted cover is provided in the equipment hatch cover for passage of smaller equipment which is too large to pass through the breech-type personnel air lock. Figure 3.8.1-14 shows the equipment hatch.

As required by ASME Code Section III, Division 2/ACI 359 Code, Sub-article CC 4552.2.1, a collar ring is provided at the connection between the liner plate and the penetration sleeve. This collar ring is shop welded and postweld heat treated together with the penetration assembly prior to welding to the liner. Test angles and channels are provided to check the integrity of the welds.

As shown on Figure 3.8.1-14, the penetration sleeve is attached to the concrete wall using anchorages extended into the concrete. These anchorages are capable of withstanding the interaction forces between the steel penetration sleeve and the concrete cylindrical wall. The anchorages are fabricated using ASME SA 105 Code Material. Fabrication, welding details, qualification and procedures for the welding of the anchorages are in accordance with ASME Section III, Division 1, Subsection NE.

In order to determine the tensile and shear capacity of the concrete anchorages used, tests were performed at Lehigh University, Fritz Engineering Laboratory, Bethlehem, Pennsylvania. The results of the tests are presented in Section 3.8.2.4.

3.8.2.1.2 Personnel Access Penetrations

One breech-type personnel air lock (Figure 3.8.1-15) and one personnel emergency air lock (Figure 3.8.1-16) are provided. Each lock is a welded steel assembly having two doors which are double-gasketed with material resistant to radiation.

The breech-type personnel air lock has a 9 ft. - 0 in. inside diameter.

The personnel emergency air lock has an outside diameter of 5 ft. - 0 in. with a 2 ft. - 6 in. diameter door located at each end of the lock.

The personnel air locks are connected by collar rings to the liner plate, as shown on Figures 3.8.1-15 and 3.8.1-16. These collars are shop welded and postweld heat treated together with the penetration assemblies prior to welding to the liner. The penetration sleeves are attached to the concrete wall using anchorages extended into the concrete, similar to those described for the equipment hatch.

3.8.2.1.3 Type I Mechanical Penetrations

Type I mechanical piping hot penetrations are provided for high pressure and high temperature (above 200 F) lines which penetrate the Concrete Containment Structure. Process pipe is connected to containment penetration sleeves by forged flued head fittings, which are designed to carry the forces and moments due to normal operating conditions and due to the postulated pipe rupture loads, and which transfer these forces to the containment penetration sleeves and further into the concrete containment wall.

Figure 3.8.1-17 shows the Type I mechanical piping penetrations. The forged flued head fitting is connected to the containment penetration sleeve with a full penetration weld. A collar ring is provided between the containment penetration sleeve and the liner plate and this collar is shop welded and postweld heat treated together with the penetration sleeve prior to welding to the liner. The penetrations sleeves embedded in the concrete wall are attached to the wall using concrete anchorages, as described in Section 3.8.1.1.3.1, for all Type I penetrations except main steam and feedwater penetrations.

For main steam and feedwater penetrations, as shown on Figure 3.8.1-10, special attachments are used. These attachments are designed to withstand the interactive forces between the steel penetration and the concrete wall and to transfer the normal operating loads and/or pipe accident rupture loads from the piping system into the concrete wall. The attachments are fabricated using code material, similar to the material used for the penetration sleeves. Fabrication, welding details, qualification and welding procedures are in accordance with ASME Section III, Division 1, Subsection NE. In order to determine the tensile and shear capacity of these concrete attachments, tests were performed at Lehigh University, Fritz Engineering Laboratory, Bethlehem, Pennsylvania. The results of these tests are presented in Section 3.8.2.4. In order to provide sufficient resistance against excessive postulated piping rupture loads, the attachments are directly connected, through a cadweld mechanical connection, to No. 18 reinforcing bars which extend radially into the concrete wall.

As shown on Figure 3.8.1-17, for the main steam and feedwater penetrations, collar rings are provided at both faces of the concrete wall in order to withstand the postulated piping rupture loads which are reversible and can act inward or outward on the concrete wall.

The main steam and feedwater penetrations fin cooling system is comprised of fin assemblies attached to the flued heads and/or penetration sleeves of Type 1 containment mechanical penetrations M-1 through M-6. Each fin assembly half consists of ten copper fins which are evenly spaced and perpendicular to the axis of the penetration. The fins are attached to a curved copper band. A lead blanket is placed between each fin assembly and flued head/sleeve to ensure good surface contact. The fin assembly is secured to the flued head/sleeve by joining the assembly halves together and connecting with bolts.

The fin cooling assemblies comprise a passive heat transfer system whose purpose is to remove excess heat from the penetration flued heads. This will limit the temperature of local areas of the reactor containment building concrete wall in the vicinity of the penetrations to that permitted by the governing code (ASME Section III Division 2 as identified in paragraph 3.8.1.2) during normal operation.

No seal is provided at the outside face of the containment wall.

3.8.2.1.4 Penetration Sleeves Not Backed by Concrete

All mechanical, HVAC and electrical containment penetrations, except penetrations described in Sections 3.8.2.1.1, 3.8.2.1.2 and 3.8.2.1.3, are designed as Class MC Components in accordance with ASME Section III, Division 1, Subsection NE, in the portion not backed by concrete, including a portion equal to 25t (where t is the thickness of the penetration sleeve). The rest of the penetration sleeve is designed in accordance with ASME Code Section III, Division 2/ACI 359 Code, but the thickness of the penetration sleeves is not reduced in this area.

3.8.2.1.4.1 Type II Mechanical Penetrations

Type II mechanical piping cold penetrations are provided for low temperature (below 200 F) lines which penetrate the Concrete Containment Structure. As shown on Figure 3.8.1-18, the process pipe passes through a containment penetration sleeve which is partially embedded and anchored into the concrete wall. The annular gap between the process pipe and the sleeve is sealed both at the inside and outside faces of the concrete wall. The inside plate is designed to withstand the internal pressure and to transfer all of the normal operating loads and/or the postulated accident piping rupture loads from the piping system to the penetration sleeve and further into the concrete wall.

Collar rings are provided between the containment penetration sleeves and the liner plate. These collars are shop welded and postweld heat treated prior to welding to the liner plate.

The penetration sleeve is attached and anchored into the concrete wall using concrete anchors in the portion of the sleeve required to comply with ASME Section III, Division 2. Therefore, for these penetration sleeves, pairs of Nelson anchors studs 7/8 in. diameter X 8 in. long are used.

3.8.2.1.4.2 Type III Electrical Penetrations

Modular type penetrations are used for all electrical penetrations. The header plates are attached to penetration sleeves located in the wall of the containment and welded to the containment liner. The connection of the sleeves to the liner plate and the anchorage into the concrete wall are as described in Section 3.8.2.1.4.1.

3.8.2.1.4.3 Fuel Transfer Tube Penetration

A fuel transfer penetration is provided to transport fuel assemblies between the refueling cavity in the Containment and the fuel transfer canal in the Fuel-Handling Building. This penetration consists of a 20 in. diameter stainless steel pipe installed inside a 26 in. containment penetration sleeve.

The connection of the penetration sleeve to the liner plate and the anchorage into the concrete wall are as described in Section 3.8.2.1.4.1.

3.8.2.1.4.4 Refueling Access Penetration

A refueling access penetration is provided at Sleeve S 66. This sleeve is outfitted with flange connections on both sides of the containment wall. During outages, the sleeve is used for eddy current, sludge lancing, and other equipment that must be run from outside the building to a location inside. During operation, blind flanges constitute the containment barrier and during outages vapor seals provide protection against refueling accidents.

3.8.2.1.5 Containment Sump Penetrations

A special type of penetration assembly is provided on the suction lines from the containment sump. These lines are used following a LOCA for recirculation of containment sump water by the containment spray and RHR pumps. Special provisions are made on these lines to reduce the possibility of leakage of sump water during recirculation. Each line consists of a concentric pipe from the sump to a leak tight compartment enclosing a portion of the suction line and the isolation valve outside the Containment. The containment sump penetration is shown on Figure 3.8.2-1.

3.8.2.2 Applicable Codes, Standards, Regulations, and Specifications

The structural design, materials, fabrication, construction, testing, in service surveillance and quality control for the ASME Code Class MC Components conform to the codes, standards, regulations, and specifications listed below, except where specifically stated otherwise.

a) American Society of Mechanical Engineers (ASME):

   1) Boiler and Pressure Vessel Code, Winter 1971 Addenda, for services rendered prior to April 29, 1977.

      Section II — Material Specifications

      Section III-Division 1 — Nuclear Power Plant Components,

	Subsection NA
	General Requirements
	Subsection NE
	Class MC Components
Section V	Nondestructive Examinations
Section IX	Welding and Brazing Qualifications

2) Boiler and Pressure Vessel Code, Winter 1973 Addenda for Type I Penetrations, and Winter 1975 Addenda for Equipment Hatch, Personnel Air Locks, and all other penetrations, for services rendered after April 29,1977.

Section II	Part A - Ferrous
	Part C - Welding Rods, Electrodes and Filler Metals
Section III-Division 1	Nuclear Power Plant Components
	Subsection NA
	General Requirements
	Subsection NC
	Subsection NE
	Class MC Components
Section V	Nondestructive Examinations
Section IX	Welding and Brazing Qualifications

3) Boiler and Pressure Vessel Code, Winter 1975 Addenda

Section III-Division 2	Code for Concrete Reactor Vessels and Containments (ASME/ACI 359), with the exceptions listed in Appendix 3.8A.

b) American Institute of Steel Construction (AISC): - Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings-1973, 7th Edition.

c) American Society for Testing and Materials (ASTM): - Various ASTM specifications, supplemented by the further requirements of ASME Section III, as follows:

ASTM A36 - Structural Steel.

ASTM A516 - Carbon Steel Plates for Pressure Vessels for moderate and lower temperature service.

ASTM A333 - Seamless and Welded Steel Pipe for low temperature service.

ASTM A108 - Cold Finished Carbon Steel Bars and Shafting.

ASTM A350 - Forged or Rolled Carbon and Alloy Steel Flanges, Forged fittings, valves and parts for low temperature service.

ASTM A435 - Straight Beam Ultrasonic Examination of Steel Plates for Pressure Vessels.

ASTM A155 - Electric-Fusion Welded Steel Pipe for High-Pressure Service.

ASTM A105 - Forgings, Carbon Steel for Piping Components.

d) Steel Structure Painting Council (SSPC):

1) SSPC-SP-6 - Commercial Blasting Cleaning

e) NRC General Design Criteria and Regulatory Guides

1) Regulation 10 CFR 50, Appendix J

2) Regulatory Guide 1.57

3) Standard Review Plan 3.8.1 "Concrete Containment"

4) Standard Review Plan 3.8.2 "Steel Containment"

f) American Welding Society (AWS) - AWS D2.0 - 69 and 1970 Revisions - Welded Highway and Railway Bridges for services rendered prior to April 29, 1977.

AWS D1.1 - Structural Welding Code - Rev 2, 1977 - Design of New Bridges, for services rendered after April 29, 1977. Non-destructive examination acceptance criteria is based on all welds being subjected to tensile stress.

g) American National Standards Institute (ANSI) - ANSI N6.2 "Safety Standard for the Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors."

ANSI N-101.2-1972 "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities."

ANSI N-101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities."

ANSI N512-1974, "Protective Coatings (Paints) for the Nuclear Industry."

ANSI N45.2.2-1972 "Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase)" and Amendments thereto.

h) Specifications - The following specifications provide the requirements for materials, design criteria, fabrication, erection, inspection and quality assurance. These specifications, in general, reflect and expand on the requirements set forth in the ASME Code.

1) Ebasco Specification CAR-SH-AS-1 "Containment Liner, Air Locks, and Hatch"

2) Ebasco Specification CAR-SH-AS-7 "Structural Steel"

3) Ebasco Specification CAR-SH-AS-7A "High Strength Bolted Field Connection for Structural Steel"

4) Ebasco Specification CAR-SH-M-54 "Mechanical Penetrations"

5) Ebasco Specification CAR-SH-E-30 "Electrical Penetrations"

6) Ebasco Specification CAR-SH-CH-15 "Mechanical Splicing of Concrete Reinforcing Steel"

7) Ebasco Specification CAR-SH-CH-22 "Structural Integrity Test of Concrete Containment Building"

3.8.2.3 Loads and Load Combinations

The equipment hatch, personnel and emergency air lock penetrations, Type I mechanical penetration sleeves, and all penetration sleeves not backed by concrete, are designed in accordance with the requirements of ASME Section III, Division 1, Subsection NE "Class MC Components" and Regulatory Guide 1.57. The flued-head portion is designed per requirements of Subsection NC.

The loads and the load combinations for design are as follows:

a) Loads

D - Dead loads.

L - Live loads.

P_t - Test pressure load.

T_t - Test temperature load.

T_o - Thermal effects and loads during startup, normal operation or shutdown conditions, based on the most critical transient or steady state condition.

R_o - Pipe reactions during startup, normal operation or shutdown conditions, based on the most critical transient or steady state condition.

E - Loads generated by the operating bases earthquake.

E' - Loads generated by the safe shutdown earthquake.

P_a - Pressure equivalent static load generated by the postulated design basis accident.

T_a - Thermal loads under thermal conditions generated by the postulated design basis accident and including To.

R_a - Pipe reactions under thermal conditions generated by the postulated design basis accident and including Ro.

Y_r - Equivalent static load on the penetration generated by the reaction on the broken pipe during the design basis accident.

Y_j - Jet impingement equivalent static load on the penetration generated by the broken pipe during the design basis accident.

Y_m - Missile impact equivalent static load on the penetration generated by or during the design basis accident, such as pipe whipping. Since the design and location of pipe rupture restraints and missile barriers preclude the possibility of impacting on a penetration, Ym can be assumed equal to zero.

P_v - Sub atmospheric pressure load (external pressure).

b) Load Combinations - The load combinations used in the design of the penetration components are listed in Table 3.8.2-1.

3.8.2.4 Design and Analysis Procedures

Containment penetrations are designed to maintain containment integrity during normal operation of the plant and in the event of a LOCA. Containment penetrations are designed to meet the requirements of Subsection NC and NE of the ASME B&PV Code, Section III, Division 1. Penetrations are designed in accordance with NRC General Design Criteria 53 of 10 CFR 50, Appendix A, and in addition, are designed to meet the following considerations:

a) Ability to withstand the maximum design pressure that can occur due to the postulated rupture of any pipe inside the Containment.

b) Ability to withstand the jet forces associated with the flow from a postulated rupture of the pipe in the penetration and maintain the integrity of the Containment.

c) Ability to accommodate thermal and mechanical stresses encountered in normal operation and other modes of operation and testing.

Penetration sleeves are anchored into the containment wall and designed for thermal, seismic and rupture loads. For penetrations with significant pipe rupture loads, rupture restraints are located so that should there be a failure of one of these pipes, the strain of the liner plate at the penetration boundary is minimized and its integrity is maintained.

3.8.2.4.1 Analysis of MC Components

The design and analysis of the Class MC Components, such as the equipment hatch, personnel and emergency air lock penetrations and Type I mechanical penetration sleeves, are performed in accordance with Article NE 3000 of Subsection NE of the ASME Code, Section III, Division 1 and Regulatory Guide 1.57. In addition, as required by Article CA 1210 (b) of ASME Code Section III, Division 2/ACI 359 Code, the containment penetration sleeves are designed in conjunction with the interactive forces and moments imposed by the cylindrical wall restraints.

In order to determine the interaction between the concrete cylindrical wall and the steel penetration sleeves during all load combinations listed in Table 3.8.2-1, the analysis is performed in the following steps:

a) A three-dimensional general finite-element analysis is performed for the entire concrete containment structure for all load combinations listed in Table 3.8.2-1. The finite-element model incorporates all of the containment penetrations larger than 24 in. in diameter; a finer mesh is provided around all major penetrations in order to obtain reliable output information regarding forces, bending and torsional moments, and shears together with associated displacements. The mechanical and thermal loads introduced into the concrete wall by the piping systems, including the postulated pipe rupture loads, are considered in this analysis. The STARDYNE computer program, described in Appendix 3.8B, is used for this analysis.

b) Preliminary calculations are performed to determine, based on the output from 1 above, the interactive forces between the concrete wall and penetration sleeves, the size and number of the concrete anchorages, size of connecting ring collars and the adequacy of the thicknesses of the penetration sleeves.

c) A three-dimensional local finite-element analysis is performed for the penetration sleeves, incorporating the connecting concrete anchorages, collar rings, liner plate and the surrounding concrete. The local finite-element model includes at least an area that is five times the radius of the penetration from the center of the penetration (beyond this area the effect of the opening is assumed to be negligible).

The finite-element analysis is performed in order to determine the interaction between the penetration sleeve and the cylindrical concrete wall, to determine the forces induced in the concrete anchorages, liner plate and collar ring, and to determine the distribution of stresses in the MC component penetration sleeve and surrounding concrete.

In order to take into consideration the general behavior of the cylindrical concrete wall at the boundary of the finite-element model for the MC components, the output displacements and rotations of the finite-element analysis used in the design of the cylindrical wall are imposed at the corresponding nodal points; the cylindrical concrete wall was subjected to the loads and load combinations listed in Table 3.8.2-1.

Figure 3.8.2-2 shows an isometric view of the local finite-element model of the equipment hatch penetration sleeve, together with the collar ring, liner plate, connecting concrete anchorages and surrounding concrete. The CDC Cybernet-Ansys Computer program, described in Appendix 3.8B, is used for this analysis.

Figure 3.8.2-2 also shows a developed view of the finite-element model for the steel penetration sleeve which is a combination of elastic flat quadrilateral shell elements and elastic flat triangular shell elements. These flat shell elements have both bending and membrane capabilities with both in-plane and normal loads permitted. Each element has six degrees of freedom at each node (three translational and three rotational).

The MC component sleeve is connected to the liner plate directly and is connected to the cylindrical wall by concrete anchorages and surrounding concrete.

The concrete anchorages are simulated as three-dimensional elastic beam elements at each location of the anchorages (a uniaxial element with tension compression, torsion and bending capabilities, with six degrees of freedom at each node). The spring constants for the anchorages used in modeling the beam elements are 800 Kips/in. for tension and 600 Kips/in. for shear. The tension spring constants are used to define the axial forces and the shear spring constants are used to define the shear forces in the beam elements.

The spring constants used for the concrete anchorages are based on the results of tests performed at Lehigh University, Fritz Engineering Laboratory, Bethlehem, Pennsylvania. The concrete anchorages, consisting of 1 in. diameter bent round bars, 16 in. long, with a 4 in. long hook, fabricated from ASME SA105 material, were tested in tension and shear both in the plane of curvature and perpendicular to the plane of curvature. Figures 3.8.2-3 through 3.8.2-5, show the results of the tests. The results of these tests show that the average of the spring constants for the concrete anchorages are 1000 Kips/in. and 800 Kips/in. for tension and shear, respectively. The tests were performed using concrete anchorages embedded in unloaded concrete. Smaller values of spring constants, as mentioned above, were used to take into consideration a decrease in tension and shear capacity of concrete anchorages due to the biaxial tension of the actual reinforced concrete wall which was not considered during the tests. As described in Section 3.8.1.4.4.9, a change in slope of the force-displacement diagrams occurred in the test performed on concrete subjected to biaxial tension and the spring constants of the Nelson studs tested in the biaxial tension region were smaller than the spring constants of the studs tested in the unloaded concrete region.

The surrounding concrete is simulated as compression spar elements (a uniaxial compression-only element with maximum of three translation degrees of freedom at each node; no bending stiffness is included). Since the elements respond in only one direction, the analysis becomes a non-linear problem and requires an iterative solution; a minimum of two iterations are used in the analysis. The spring constants of the concrete used in modeling the compression spar elements are determined using the concept of a semi-infinite plate loaded with a concentrated force in the plane of the plate.

The cylindrical concrete wall and the liner plate are also modeled using a combination of elastic flat quadrilateral shell elements and elastic flat triangular shell elements.

Figure 3.8.2-2 shows the connection between the model of the steel penetration and the cylindrical wall and liner plate. As shown, a rigid beam element is used between the cylindrical wall and the connecting concrete anchorages and the concrete around the steel penetration sleeve. This rigid beam imposes the displacements and rotations of the cylindrical wall into the steel sleeve and concomitantly allows the penetration sleeve to have a different elastic behavior based upon the imposed loading conditions and the different rigidities of the connecting concrete anchorages and surrounding concrete.

The connection between the cylindrical wall and liner plate is modeled by using three dimensional elastic beam elements to simulate the capacity in tension and shear of the liner anchor studs and the shear transferring capacity of the existing concrete between the liner plate and the theoretical center line of the concrete wall. The bearing capacity of the concrete is simulated by using compression spar elements as described above.

Figures 3.8.2-6 and 3.8.2-7 show similar finite-element models used for the analysis of the personnel and emergency air lock penetrations, respectively. In addition, stress reports are prepared by the component manufacturer. These stress reports include the interaction between the concrete containment wall and the penetration sleeves and are reviewed and approved by the Architect Engineer.

The analysis of Type I mechanical penetrations is performed in two parts: the first analysis is a linear analysis and is performed primarily for the portion of the sleeve not backed by concrete; it considers all loads and load combinations listed in Table 3.8.2-1, including the mechanical and thermal load effects of the piping systems during normal operation and postulated accident conditions.

The linear model considers the penetration sleeve fixed in the concrete wall; an ANSYS computer program, described in Appendix 3.8B, is used for the analysis. Axisymmetric models are developed using the 2D axisymmetric element STIF25. Each of these elements can be used for the analysis of structures to which axisymmetric and non-axisymmetric loads are applied. The flued head, penetration sleeve, and portions of the process pipe are included in the model.

Unit loading is applied at the boundary conditions producing unit load stress values for bending, shear, torsion, axial and internal pressure loading conditions. The unit load data is then scaled up to the actual loading and combined using the ANSYS Post Processor "Post 29". The resulting data is then linearized using "Post 1" and these combined stress values are compared to code allowables. The code allowables are presented in Tables 3.8.2-4 and 3.8.2-6.

The models were used for heat transfer analysis to determine the steady state operational temperature gradient. Conversion of the models for the heat transfer analysis was achieved by replacing the STIF25 element with the heat transfer element STIF75. The thermal boundary conditions consist of film coefficients based upon the fluid and air characteristics on the surfaces of the penetrations and the temperatures of the fluid and air.

The second part of the analysis, which takes into consideration the concrete wall penetration sleeve interaction, is a non-linear finite-element analysis.

The non-linear model consists of a three-dimensional model of the penetration, using a cylindrical shell configuration for both the portion of the sleeve backed by concrete and the portion of the sleeve not backed by concrete, up to the interface with the thicker portion of the flued head forging.

Figure 3.8.2-8 shows the main steam and feedwater penetration model used for the non-linear analysis of the sleeve embedded in concrete. The non-linear model considers the penetrations as projecting from the concrete wall up to the interface with the flued head forging in order to take into consideration the flexibility and influence of the piping system. In order to incorporate this influence into the analysis, the output displacements (translations and rotations) from the linear model analysis at this particular location are used as an input for the non-linear model.

As described in the equipment hatch analysis, the liner plate is simulated by using three dimensional elastic beam elements at each node point where the liner is connected to the penetration model. The spring constants of the liner plate used in modeling the elastic beams are listed in Table 3.8.2-2. Radial spring constants are used to define the axial forces in the beam elements and tangential spring constants are used to define the shear forces in the beam elements.

The attachments into the concrete are shown on Figure 3.8.1-10. The attachments are also simulated as three-dimensional elastic beam elements at each location of the attachments. The tension spring constants are used to define the axial forces and the two shear spring constants are used to define the shear forces in the meridional and hoop directions in the beam elements.

The spring constants used for the attachments are based on the results of tests performed at Lehigh University, Fritz Engineering Laboratory, Bethlehem, Pennsylvania (see Reference 3.8.1-70). The attachments, consisting of 1 1/2 x 5 x 15 in. steel plate, ASME SA-155 Class 1 KCF 70 material, welded to a cadweld mechanical connection and a No. 18 reinforcing bar, were tested in tension and shear both in the plane of the plate and perpendicular to the plane of the plate. Figures 3.8.2-9 through 3.8.2-11 show the results of the tests.

The surrounding concrete is simulated as compression spar elements (uniaxial compression-only elements with maximum three translational degrees of freedom at each node point and no bending or torsion considered). The spring constant of the surrounding concrete used in modeling these spar elements is also listed in Table 3.8.2-2. Since the elements respond in only one direction, the analysis becomes a non-linear problem and requires an iterative solution; a minimum of two iterations are used in the analysis.

The collar rings are also modeled as flat quadrilateral plates connected to the penetration sleeve.

Figure 3.8.2-8 shows the finite-element model for the Main Steam and Feedwater Type I penetrations. As shown on this figure, at the connection between the liner plate and the collar ring, at the inside face of the wall, two additional springs are introduced; one compression spar element simulates the concrete and one tension cable element simulates the capacity in tension of the liner plate connecting anchor studs. The bearing spring constants of the concrete and of the anchor studs used in modeling the spar elements are listed in Table 3.8.2-2. At the outside face of the wall compression spar elements simulating only the concrete are considered.

To induce the interaction effects between the penetration sleeve and the surrounding concrete cylindrical wall, imposed displacements are introduced at the ends of the elastic beam elements which simulate the liner plate, the elastic beam elements which simulate the concrete anchorage, and the compression spar elements which simulate the surrounding concrete. These displacements are not equal around the periphery of the penetration. One set of displacements around the periphery has been developed for test pressure and one set of displacements has been developed for the accident conditions associated with the SSE and mechanical loads. The imposed displacements for the test pressure load combination and for the accident associated with the earthquake and mechanical loads due to a postulated pipe break are listed in Table 3.8.2-3.

The thermal differential temperature gradient during accident conditions is considered in the analysis. The temperature of the penetration sleeve is 233 F for the portions backed by concrete and 262 F for the portions not backed by concrete. The temperature of the surrounding concrete is 120 F on the inside face and 90 F on the outside face; an average temperature of 105 F is considered in the analysis.

As shown on Figure 3.8.2-12, a similar finite-element model is used for analysis of Type I mechanical penetrations with diameters less than 18 in.; except that, in this case, no external collar ring is provided at the outside face of the concrete wall, and the imposed displacements are uniform around the perimeter of the penetrations.

To further assure adequacy of structure, the main steam and feedwater region of the containment building was additionally analyzed as described below.

Load combination 6 of Table 3.8.2-1 was used in the structural analysis. The load combination considered the maximum pipe reactions from one main steam pipe ruptured inside the containment, the respective reactions of the intact pipes at their locations, the temperatures and pressures in effect at the time of the ruptured pipe maximum reactions, and the loadings due to safe shutdown earthquake. The structural analysis was performed by the NASTRAN computer program. The stress results of the analysis for the as-built condition of the region are shown in Table 3.8.2-5.

The analysis was nonlinear to account for possible cracking of concrete, and for separation of penetrations end collars from concrete. Two types of nonlinear elements were used, one compression-only spring element, and two concrete cracking elements. The compression-only element was used to simulate the contact surface between the containment concrete and penetration sleeves, and the concrete cracking elements were used to represent the reinforced concrete containment wall. The concrete cracking elements are shown in Figure 3.8.2-14. They consist of a series of flat shell elements with coupled in place and out-of-place stiffnesses which are based on the Kirchoff assumption that the plane normal to the shell surface remains normal after loading. The concrete sections were divided into a sufficient number of layers through the thickness to monitor crack propagation from layer to layer, and from element to element.

The analysis utilized substructuring techniques because the size of the structure made it impractical to model it as an entity. Three basic substructures were used, the penetration area with its six pipes (3 main steam, and 3 feedwater) and surrounding shell, the main steam pipes with their anchors and flued heads, and the feedwater pipes with their anchors and flued heads. For the latter two substructures, to obtain continuity, the portions not in the model were represented by a 6 by 6 stiffness matrix.

The model of the substructure for the penetrations area included the six penetration sleeves, and is shown in Figure 3.8.2-15. The penetration sleeves and end collars were represented by shell elements with contact surfaces represented by compression-only nonlinear springs, and are shown in Figure 3.8.2-17 and 18. The spring constants are based on the elastic modulus of concrete. The shear lugs welded to the sleeve and connected to reinforcing steel were represented by shear-only and axial-only linear springs, and are based on the results of the tests of Reference 3.8.1-70. The containment wall of the substructure was modeled by the nonlinear concrete cracking elements, with the thickness of wall divided into ten layers. As-built reinforcement was used in the model. Figure 3.8.2-19 shows a typical section of concrete and reinforcement.

The models of the substructures for the main steam and feedwater penetrations are similar to each other. Each pipe structure consists of three dimensional solid elements to represent the flued head, and shell elements to represent the pipe. A set of rigid bars at each end of the pipe connects the pipe circumferential grid points to a point at the center of the pipe. The remainder of the pipe run was accounted for by a 6 by 6 matrix at the centerpoint of the pipe, representing a coupled spring element. The individual pipe substructure is connected to the penetration substructure model at the junction of the pipe sleeve and flued head. Figure 3.8.2-16 gives the finite element model of the main steam and feedwater pipes substructure.

The boundary conditions at the four edges of the main steam and feedwater penetrations substructure model are based on the behavior anticipated for the overall structure. The side boundaries were represented by cyclic symmetric conditions which is a conservative boundary assumption for stresses in the local model. Reflective boundary conditions were used for the top and bottom boundaries except for the positive Z-direction. A free boundary was used for this direction to allow for thermal growth of the containment building. The boundary conditions for the pipe substructure were represented by the previously described 6 by 6 matrix representing coupled springs.

Heat transfer analyses were performed for the main steam and feedwater penetration fin assemblies using the HEATING6 computer code, "A Multidimensional Heat Conduction Analysis with the Finite-Difference Formulation," NUREG/CR-0200 Volume 2 Section F10, October 1981. The main steam and feedwater pipelines are assumed to be at their maximum operating temperatures of 557 F and 435 F, respectively. The maximum steady state temperature distribution is calculated for summer conditions with air temperatures of 120 F inside the reactor containment building and 116 F in the steam tunnel.

The fin cooling assemblies are non-safety related and seismically designed.

3.8.2.4.2 Analysis of Mechanical and Electrical Penetration Sleeves

The design and analysis of the containment mechanical and electrical penetration sleeves not backed by concrete are performed in accordance with Article NE-3000 of Subsection NE of the ASME Code, Section III, Division 1, NRC Regulatory Guide 1.57 except that a fatigue analysis is not performed since all of the piping systems passing through the penetrations are Safety Class 2 and do not require a fatigue analysis.

The analysis is similar to the analysis of the Type I mechanical penetrations described in Section 3.8.2.4.1 Figure 3.8.2-13 shows the finite-element model of the Type II and Type III mechanical and electrical penetration sleeves.

3.8.2.5 Structural Acceptance Criteria

The structural acceptance criteria for the containment penetrations that were designed as Class MC components are in accordance with Article NE-3000 of the ASME Code, Section III, Division 1, Regulatory Guide 1.57, and Standard Review Plan 3.8.2.

Table 3.8.2-4 shows the stress limits that were used for structural acceptance of the MC components under various loading conditions.

3.8.2.6 Materials, Quality Assurance and Special Construction Techniques

3.8.2.6.1 Materials

The materials that are utilized for the containment penetrations designated as Class MC components are in accordance with Article NE-2000 of Subsection NE of ASME Code, Section III, Division 1.

The materials for the equipment hatch and personnel and emergency air locks conform to ASME SA-516, Grade 70.

The materials for the containment penetration sleeves that are less than 24 in. in diameter conform to ASME SA-333, Grade 6 seamless, and sleeves larger than 24 in. in diameter conform to ASME SA-155, Class I KCF 70. ASME SA-106, Grade B, Fine Grain, may be substituted for ASME SA-333, Grade 6, provided that it meets the impact test requirements of ASME SA-333, Grade 6.

The materials for flued head fittings conform to ASME SA-182, Grade F304 of F316 alloy steel for stainless steel, or ASME SA-105 for carbon steel. The materials for the collar rings and plates for penetration sleeves conform to ASME SA-516 Grade 70.

When required by Article NE-2300, the penetration sleeve, flued head and load bearing closure plates are impact tested. The test temperature is 0 F.

All process pipes are of identical material as the lines to which they are connected.

The fuel transfer tube expansion joint material conforms to ASME SA-240 Type 304. The weld ends of the bellows are made of either the same base material as the containment penetration sleeve or a material that is ASME Code acceptable to both the expansion bellows and the sleeve. There is no dissimilar metal welding in the field.

Expansion joints which are installed as part of the pressure boundary for a containment penetration are designed in accordance with ASME Section III, Division 1, Articles NC-3624 and NC-3649 requirements.

The expansion joint axial and lateral displacements are determined on the basis of relative thermal growth, seismic displacements, and differential settlements of the adjoining structures.

3.8.2.6.2 Quality Assurance

The overall Quality Assurance Program is in accordance with the Engineering and Construction QA program which was approved by the NRC during the construction permit review. In addition, certain measures as required by the ASME Code for Class MC components are outlined below.

a) Examination and testing applied to the penetrations are in accordance with Articles NE-5000 and NE-6000 of Subsection NE.

b) The Class MC component vendors submit shop procedures to the Architect Engineer. These procedures, as applicable, address welding, nondestructive testing, vacuum box testing and visual examination.

c) Records pertaining to the Class MC components contain three distinct categories: material certifications, welding data and test data. All records are turned over to the owner on completion of the work.

d) All welding procedure qualifications and welder performance qualifications are in accordance with ASME Code, Section IX. The welding design, fabrication, inspection and acceptance conform, as a minimum, to the requirements of ASME Code Section III, Division 1, Subsection NE. The examination of welds for Class MC components is in accordance with Article NE-5000 of the ASME Code, Section III, Division 1.

e) All procedural requirements for nondestructive testing conform, as a minimum, to the requirements of Section V of the ASME Code.

3.8.2.6.3 Special Construction Techniques

No special construction techniques, different from current methods of fabrication, are used for the containment penetrations designated as Class MC components.

Any deviation from or conflict with the requirements of Subsection NE shall be documented on a case-by-case basis. Such deviations shall be evaluated for acceptance and justified on an engineering basis. Applicable documentation and justification will then be made a part of QA records.

3.8.2.7 Testing and In-Service Surveillance Requirements

Testing of Class MC Components is in accordance with Article NE-6000 of Subsection NE of the ASME Code, Section III, Division 1.

The Concrete Containment Structure is subjected to the structural acceptance test as described for the Containment in Section 3.8.1.7.1.

In addition, upon completion of construction and prior to containment testing, the personnel and emergency air locks and equipment hatch are given an operational test which consists of repeated operations until they operate smoothly without binding. Any defects encountered are corrected and the personnel and emergency air locks are retested. The process of testing, correcting defects, and retesting is continued until no defects are detected.

Preoperational and periodic leak tests of the Concrete Containment Structure and testable penetrations are conducted to verify that their leak rate is below the specified design leak rate. These tests are discussed in Section 6.2.6.

3.8.3 CONCRETE AND STRUCTURAL STEEL INTERNAL STRUCTURES OF THE CONCRETE CONTAINMENT

The reinforced Concrete Containment Structure encloses the concrete structures and structural steel components which comprise the Containment Internal Structures. The Containment

Internal Structures provide support for the NSSS equipment during all operational phases and, in the unlikely event of an accident, act to mitigate the consequences of the accident by protecting safety-related equipment and other engineered safety features from the effects induced by the accident.

The concrete internal structures, which consist of the primary and secondary shield walls and other concrete enclosures, form compartments within which the entire Reactor Coolant System (RCS) is located. The main components are the concrete primary shield wall, which encloses the reactor cavity, the semicircular concrete secondary shield walls, which forms the steam generator compartments, the reinforced concrete walls and floors, the fuel storage area, refueling pool and reactor internals laydown areas, the concrete enclosure wall around the pressurizer, the containment steel floors, stairs, and platforms, reactor vessel supports, steam generator supports, and reactor coolant pump supports. The concrete and steel internal structures are supported on a concrete foundation mat 5 ft. thick, resting on the 12-ft. thick concrete containment structure foundation mat. The internal foundation mat is placed on top of the bottom liner plate; no anchorages of the internal structures and internal mat penetrate through the containment liner plate into the external mat. The walls of the internal structures are anchored into the internal foundation mat.

Figures 3.8.3-1 and 3.8.3-2 show the general arrangement of the Containment Internal Structures.

3.8.3.1 Description of the Internal Structures

3.8.3.1.1 Primary Shield Wall

The reactor cavity houses the reactor pressure vessel (RPV) which is located in the center of the Containment.

The reinforced concrete primary shield wall, which surrounds the RPV and encloses the reactor cavity, extends from the top of the foundation mat at Elevation 221.0 ft. to the top of the operating floor at Elevation 286 ft. The thickness of the wall is 9 ft. 3 in. between Elevation 221 ft. and Elevation 248 ft. 9 in., and 4 ft 6 in. between Elevation 248 ft. 9 in. and Elevation 286 ft. The continuation and extension of this wall above the elevation of the RPV head flange forms the sides of the steam generator compartments, fuel storage area, refueling pool, and reactor internals laydown areas.

The RPV bears on six steel base plates which are supported by the primary shield wall immediately beneath the reactor nozzles. The control rod drive mechanism (CRDM) missile shield wall is not supported by the extension of the primary shield wall, but is part of the reactor vessel head lifting rig.

The RPV nozzle penetrations through the reinforced concrete wall are reinforced locally for pipe rupture loadings.

The following are the functions of the primary shield wall:

a) Provide biological shielding during normal operation.

b) Function as a missile shield to prevent any missiles generated within the (RCS) from impinging upon the reactor vessel and to prevent any missiles generated within the reactor cavity from impinging on other components of the NSSS or the containment structure.

c) Provide a support structure for the reactor vessel, and for intermediate platforms.

d) Provide support for pipe whip restraints.

Figure 3.8.3-3 shows the typical reinforcement arrangement in the primary shield wall. Basically, the reinforcement consists of two layers of No. 11 reinforcing bars placed on both inside and outside faces in the vertical and hoop directions. A small quantity of No. 18 reinforcing bars are used in localized areas. Where equipment is supported in some local areas, the wall is heavily reinforced to transfer the loads from the equipment into the concrete wall. The reinforcing bars are spliced using lap splices in the vertical direction and cadweld splices in the horizontal direction. The splices are staggered wherever practical. The vertical bars are embedded into the internal foundation mat.

3.8.3.1.2 Secondary shield walls

The secondary shield walls consist of two half-cylindrical reinforced concrete structures which enclose the steam generators, reactor coolant pumps, and reactor coolant piping on each side of the reactor vessel (see Figures 3.8.3-1 and 3.8.3-2). The secondary shield wall terminates at the primary shield wall and is integrally constructed with it. A divider wall between the primary and secondary shield walls separates the steam generator from the pressurizer. The walls extend from the foundation mat up to the operating floor (Elevation 286 ft.). The thickness of the secondary shield wall is four ft.

Pressure relief areas are provided in the lower region of the secondary shield wall. The barrier wall between each pressure relief area and the Concrete Containment Structure protects the CCS from direct impingement of overpressure or missiles generated within the lower regions of the steam generator compartments.

The following are the functions of the secondary shield wall:

a) Provide biological shielding during normal operations.

b) Function as a missile shield to protect the CCS from missiles generated within the RCS, as well as to protect the RCS from any missiles generated from other equipment within the CCS.

c) Provide a support structure for the operating floor and intermediate platforms.

d) Provide support for pipe whip restraints.

Figure 3.8.3-4 shows the reinforcement arrangement in a typical area of the secondary shield wall. Basically, the reinforcement consists of two layers of No. 11 reinforcing bars on the inside and outside faces and in the vertical and horizontal directions. The horizontal reinforcing bars are spliced using cadweld mechanical splices, and the vertical reinforcing bars are spliced using lap splices. The vertical reinforcing bars are embedded into the internal foundation mat.

3.8.3.1.3 Refueling pool cavity

The refueling pool and reactor internals laydown areas are located within a rectangular enclosure formed by reinforced concrete walls and floors (see Figures 3.8.3-1 and 3.8.3-2). The fuel transfer tube across the Reactor Auxiliary Building system boundary connects the refueling pool of the Containment Building to the fuel transfer canal in the Fuel Handling Building. For the refueling process, the pools are filled with water. A stainless steel liner is provided on the interior faces of these areas to make them watertight. A stainless steel seal ring over the reactor cavity, connected to the RPV flange and pool floor liner, keeps the reactor cavity dry. The walls are up to 5 ft. thick, and the thickness of the bottom slab is 5 ft.

The following are the functions of these areas:

a) Provide biological shielding and a completely watertight compartment in which to carry out the refueling process and transfer of fuel rods from the reactor vessel.

b) Provide a support structure for the operating floor and the intermediate platforms.

Figure 3.8.3-5 shows the arrangement of the reinforcement in the refueling cavity walls.

3.8.3.1.4 Internal foundation mat

The concrete internal structures, consisting of the primary shield wall, secondary shield walls, refueling cavity walls, steam generator pedestals, and reactor coolant pump pedestals, are supported by a common internal foundation mat, five ft. thick, resting on top of the containment external foundation mat. All vertical walls are embedded into the internal mat; no anchorages penetrate through the liner plate and into the external mat. Shear keys are provided at the bottom of the walls in order to allow the transfer of radial and tangential shears from the walls into the internal mat.

Figure 3.8.3-6 shows the internal mat layout.

3.8.3.1.5 Steam Generators and Pressurizer Enclosures

The steam generators are enclosed between the primary and secondary shield walls up to the operating floor at Elevation 286 ft., and within the concrete walls, 4 ft. thick, above the operating floor up to Elevation 304.65 ft. Bumper and snubber type supports are provided for the upper lateral support of the steam generators. The lower lateral supports are provided by steel beams supported by the concrete walls. Figure 3.8.3-7 shows the layout of the steam generator enclosures; the enclosures provide biological shielding during normal operation. At the bottom, the steam generators are supported by vertical steel columns resting on concrete pedestals (see Figure 3.8.3-8).

The pressurizer enclosure consists of reinforced concrete walls, supporting floor, and removable roof; these form a compartment within which the pressurizer is enclosed. An open area is provided at the top for pressure relief in the event of a pressure differential between the enclosure and the surrounding space. The pressurizer enclosure provides biological shielding during normal operation and also serves to contain potential missiles which may be generated as a result of equipment failure within the enclosure.

3.8.3.1.6 Operating Floors and Platforms

The containment floors and platforms are linked by stairs and one service elevator. Except for equipment laydown areas as identified, all floors and stairs are of grating construction to minimize the effects of pressure differentials across their boundaries should a sudden change in pressure occur in the Containment Building.

Structural steel framing is supported by the secondary shield wall and by steel columns at a 64 ft. radius from the centerline of the Containment Building. Figure 3.8.3-16 provides a sketch of the columns. Except for the polar crane brackets, HVAC ducts, piping supports, the dome access ladder, electrical conduits, boxes and fittings, there are no connections to the containment external wall. All horizontal loads are taken back to the secondary shield wall by the horizontal bracing system at each floor.

3.8.3.1.7 Reactor Vessel Support System

The reactor pressure vessel (RPV) is supported and restrained to resist normal operating loads, seismic loads, and loads induced by the postulated pipe ruptures, including a LOCA. The RPV is supported at six points, the three inlet and three outlet nozzles, so that adjacent supports are 50 or 70 degrees apart. Steel pads, which are an integral part of the nozzles, rest on a steel bearing block atop a steel support pedestal as shown on Figure 5.4.14-1. The steel support pedestal is welded to a stiffened base plate at its bottom. The base plate is attached to the reinforced concrete by anchor bolts. The base plate has shear bars on its underside to resist part of the lateral loads, including piping loads.

The transfer of horizontal seismic and postulated accident loads from the RPV and the connecting piping system into the concrete primary shield wall is performed through embedded steel structures as shown on Figure 3.8.3-9. These structures consist of billet plates welded to vertical circular plates, anchored into the concrete wall by using anchor bolts and embedded structural steel assemblies. The gap between the vertical RPV supports and the horizontal RPV supports is shimmed in the cold condition with a predetermined allowance for thermal expansion.

3.8.3.1.8 Steam Generator Support System

The steam generator is supported and restrained to resist normal operating loads, seismic loads, and loads induced by postulated pipe rupture. The support system prevents rupture of the primary coolant pipe due to a postulated rupture in the steam or feedwater pipes.

The steam generator is vertically supported by four steel columns which are pin connected to both the vessel bottom head and to the base plates, as shown on Figure 5.4.14-2. The base plate is anchored to a reinforced concrete pedestal by bolts. The reinforced concrete pedestal is anchored to the internal foundation mat in accordance with the details shown on Figure 3.8.3 8. The lower lateral support consists of horizontal structural beams at the bottom of the steam generator which prevent lateral movement. The upper lateral support consists of snubber and bumper assemblies, located on the steam drum, and connected to the secondary shield wall; the upper lateral support guides the top of the steam generator during expansion and contraction of the RCS.

These supports also provide horizontal restraint for the steam generator during earthquakes and following a LOCA or postulated steam line break. The low friction bearing material provided at the contact surfaces of the supports minimizes resistance to thermal movements.

Figure 3.8.3-10 shows the vertical and upper lateral supports of the steam generator.

3.8.3.1.9 Reactor Coolant Pump Support System

The reactor coolant pumps are supported and restrained to prevent excessive deflection during normal operating, seismic, and postulated pipe rupture conditions. Each pump is supported on three vertical steel columns which are pin connected at the bottom of the pump and at the base plates to produce an articulated support which does not impose significant forces on the concrete during thermal expansion (see Figure 5.4.14-3). The base plates are anchored to the mat by bolts, as shown in Section K of Figure 3.8.3-10. Horizontal support is provided as a restraint for postulated LOCA and seismic loadings; the horizontal supports consist of tie rods embedded into the primary and secondary shield walls.

3.8.3.1.10 Pressurizer Support System

The supports for the pressurizer, as shown on Figure 5.4.14-4, consist of a steel ring plate connected with anchor bolts to the reinforced concrete supporting floor and the upper lateral supports. The upper lateral supports are structural steel struts, which are cantilevered off the concrete shield walls and bear against lugs provided on the pressurizer.

3.8.3.2 Applicable Codes, Standards, and Specification

The Containment Internal Structures and other Seismic Category I Structures conform with applicable requirements of the regulatory guides, codes, and specifications listed below.

<u>General Codes and Standards</u>

OSHA - Occupational Safety and Health Administration, Federal Safety Regulations (1975 listing)

NCSBC - North Carolina State Building Code, 1969 Edition

<u>ACI - American Concrete Institute Standards</u>

211.1-1974 - Recommended Practice for Selecting Proportions for Normal and Heavyweight Concrete

214-1965 - Recommended Practice for Evaluation of Compression Test Results of Field Concrete

301-1972 - Specifications for Structural Concrete for Buildings - Revised 1975

304-1973 - Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete

305-1972 - Recommended Practice for Hot Weather Concreting

306-1966 - Recommended Practice for Cold Weather Concreting

309-1974 - Recommended Practice for Consolidation of Concrete

315-1974 - Manual of Standard Practice for Detailing Reinforced Concrete Structures

318-1971 - Building Code Requirements for Reinforced Concrete

347-1968 - Recommended Practice for Concrete Formwork

349-1976 - Code Requirements for Nuclear Safety-Related Concrete Structures - Appendix C, "Special Provisions for Impulsive and Impactive Effects"

349-1980 - Code Requirements for Nuclear Safety-Related Concrete Structures -Appendix B, "Steel Embedments"

SP-2 - Manual of Concrete Inspection

American Society of Mechanical Engineers (ASME)

"Boiler and Pressure Vessel Code," 1975 Edition

Section II - Material Specifications

Section III, Division I - Nuclear Power Plant Components

Subsection ND "Class 3 Components"

Subsection NE "Class MC Components"

Section IX - Welding and Brazing Qualifications

American Institute of Steel Construction (AISC) - Seventh Edition

Code of Standard Practice for Steel Buildings and Bridges (AISC Code) (9/1/76)

Specification for the Design, Fabrication and Erection of Structural Steel for Buildings (AISC Specification) with supplements through Supplement 3 (6/12/74)

Specification for Structural Joints Using ASTM A 325 or A 490 Bolts (2/4/76)

AISC - A Guide to the Shop Painting of Structural Steel (6/14/72)

American National Standards Institute (ANSI)

N690-1984 Nuclear Facilities: Steel Safety Related Structures for Design, Fabrication, and Erection

American Welding Society (AWS)

D1.1-75 - Structural Welding Code (or Later Revisions)

D12.1-76 - Reinforcing Steel Welding Code

Steel Structure Painting Council (SSPC)

SP-6 - Commercial Blast Cleaning

PA-1 - Shop, Field, and Maintenance Painting

United States Nuclear Regulatory Commission (USNRC)

The following NRC Regulatory Guides, as qualified in Section 1.8, are applicable:

1.10 - Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures.

1.15 - Testing of Reinforcing Bars for Category I Concrete Structures

1.54 - Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants

1.55 - Concrete Placement in Category I Structures

1.60 - Design Response Spectra For Seismic Design of Nuclear Power Plants

1.61 - Damping Values For Seismic Design of Nuclear Power Plants

1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

1.94 - Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants

1.122 - Development of Floor Design Response Spectra For Seismic Design of Floor Supported Equipment or Components

Industry Standards

Industry standards, such as those published by the American Society for Testing and Materials (ASTM) or the American Association of State Highway and Transportation Officials (AASHTO), are used whenever possible to describe materials properties, testing procedures, fabrication, and construction methods.

3.8.3.3 Loads and Loading Combinations

For convenience of reference, all of the design loads and loading combinations specified by the codes, standards, and specifications which govern the design of the Containment Internal Structures and other Seismic Category I structural steel structures are listed below. Some of the loads and loading combinations may not apply to one or more of the structures of the system because of the absence of equipment, piping, or environmental condition which can induce the load.

3.8.3.3.1 Loads

The loads are grouped into the following categories:

a) Normal Loads - Normal loads are those loads encountered during construction, normal plant startup, operation, and shutdown. They include the following:

1) Dead Loads (D) - Dead loads consist of the deadweight of the Containment Internal Structures, the weight of structural steel, any permanent equipment loads, and hydrostatic loads. Equipment loads include the deadweight of the various pieces of equipment, including water or other enclosed fluids, piping, ventilation ducts, and electrical cables and trays supported by the Containment Internal Structures. Hydrostatic loads include the loads exerted by the water in the refueling cavity which is filled only during reactor shutdown and which is computed by assuming the specific weight of refueling water as 62.4 pcf.

For equipment supports, dead load also includes static and dynamic head and fluid flow effects. Specific weights for dead load calculations are given in Section 3.8.1.3.1.a).

2) Live Load (L) - Live Loads are set for the various floor and slabs to assure that the structures are sufficiently strong to support random temporary load conditions during reactor shutdown. These loads are set as follows:

Operating Floor-Concrete:	1000 psf or equipment load in designated laydown area, whichever is greater:
Operating Floor-Steel:	grating floors on structural steel framing: 100 psf or equipment, whichever is greater.
Operating Floor-Steel:	concrete floors on structural steel framing: 200 psf or equipment, whichever is greater.
Other Areas:	100 psf or equipment load, whichever is greater.

The above live loads include movable equipment and other loads which vary in intensity and occurrence.

For equipment supports, the live load includes loads due to vibration and any support movement effects during normal operation and reactor shutdown conditions.

3) Normal Operating Thermal Load (T_o) - The forces caused by the expansion of the Containment Interior Structures due to increased internal ambient temperature during normal operation or shutdown conditions. The temperature of all components of the internal structures (except the primary shield wall) is assumed to uniformly stabilize at the same temperature as the internal ambient temperature

(120 F); the "as constructed" temperature is assumed to be 60 F. Due to neutron capture, the primary shield wall is designed for a maximum temperature of 150 F in the interior of the wall, with the inside and outside faces subjected to normal operating temperatures of 130 F and 120 F, respectively.

4) Pipe or Equipment Anchor Load (operating) (R_o) - The pipe or equipment anchor loads are the loads exerted upon the various structural elements of the Containment Internal Structures by the pipe or equipment restraints for the normal thermal expansion of the various piping systems.

b) Severe Environmental Loads - Severe environmental loads are those loads that could infrequently be encountered during the plant operating life. Included in this category are:

1) Wind Loads (Hu) - Wind loads defined in Section 3.3 are considered for the exposed areas of the other Seismic Category I structural steel structures.

2) Seismic Load (E) - The load generated by the operating basis earthquake (OBE). The simultaneous occurrence of three-directional earthquake motion is considered. Hydrodynamic effects of fluids in containers and lateral soil and groundwater loads under dynamic conditions, which are in excess of the static loads, are included in the seismic loads.

c) Extreme Environmental Loads - Extreme environmental loads are loads which are credible, but highly improbable. These include:

1) Seismic Load (E') - This is the load generated by the safe shutdown earthquake (SSE). The simultaneous occurrence of three-directional earthquake motion is considered. Hydrodynamic effects of fluids in containers and lateral soil and groundwater loads under dynamic conditions, which are in excess of the static loads, are included.

2) Tornado Load (W) - This is the load generated by the design basis tornado and includes tornado missile loadings. Refer to Section 3.3 for details.

d) Abnormal Loads - Abnormal loads are loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. These loads include:

1) Loss-of-coolant accident or Other Postulated Break Pressure Load (P) - A loss-of-coolant accident or other postulated break load is determined by analysis of the pressure transients inside the primary and secondary shield wall during a loss-of-coolant accident, or other postulated break.

2) Accident Thermal Load (T_a) - The accident thermal load represents the force caused by expansion of the Containment Interior Structures due to increased internal ambient temperature under thermal conditions generated by a postulated break, including To. There are no significant effects due to extremely short temperature differentials within the Containment Internal Structures following a LOCA.

3) Pipe or Equipment Anchor Load (accident) (R_a) - The pipe or equipment anchor load is the load exerted upon the various structural elements of the Containment Internal Structures by pipe or equipment restraints under thermal conditions generated by the postulated break, including R_o.

4) Pipe Accident Load (Q) - These are the loads exerted upon the Containment Internal Structures by pipe or equipment as a result of the postulated loss-of-coolant accident or other pipe rupture accidents; they include:

Y_r - Equivalent static load on the structure generated by the reaction of the broken high-energy pipe during the postulated break, 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, 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, as from pipe whipping or small pieces of equipment traveling at high velocities, 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 is assumed with appropriate ductility ratios.

3.8.3.3.2 Load Combinations--Concrete Structures

The following presents load combinations for the Containment Internal Concrete Structures.

For concrete structures, U is the section strength required to resist design loads and is based on methods described in ACI 318-71.

a) Service Load Conditions

1) $U = 1.4D + 1.7L$

2) $U = 1.4D + 1.7L + 1.9E$

If thermal stresses due to T_o and R_o are present, the following combinations are considered:

1) $U = 0.75(1.4D + 1.7L + 1.7T_o + 1.7R_o)$

2) $U = 0.75(1.4D + 1.7L + 1.9E + 1.7T_o + 1.7R_o)$

Where lateral pressures due to liquid are present, and if D or L reduces the effects of these loads in any of the above loading combinations, additional load combinations are considered with the corresponding coefficients taken as 0.9 for D, and zero for L. Both cases of L having its full value or being completely absent are considered.

b) Factored Load Conditions - These conditions represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions.

1) $U = 1.0D + 1.0L + 1.0T_o + 1.0R_o + 1.0E'$

2) $U = 1.0D + 1.0L + 1.0T_a + 1.0R_a + 1.5P$

3) $U = 1.0D + 1.0L + 1.0T_a + 1.0R_a + 1.25P + 1.0 (Y_r + Y_j + Y_m) + 1.25E$

4) $U = 1.0D + 1.0L + 1.0T_a + 1.0R_a + 1.0P + 1.0(Y_r + Y_j + Y_m) + 1.0E'$

In load combinations 2), 3), and 4), the maximum values of P, T_a, R_a, Y_r, Y_j, and Y_m, are applied simultaneously, unless a time history analysis is performed to justify otherwise. For static analysis, appropriate dynamic load factors are used for these loads to reflect their dynamic natures. Combination 3) and 4) are satisfied first without Y_r, Y_j, and Y_m. When considering these concentrated loads, local section strength capabilities may be exceeded provided there is no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent are checked.

3.8.3.3.3 Load Combinations - Steel Structures

The load combinations for steel structures inside the Containment and other Seismic Category I steel structures are as follows:

For structural steel, S is the required section strength based on the elastic 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 steel due to seismic loadings or wind loading is not used. Y denotes the required section strength based on the plastic design methods specified by the AISC Specification, Part 2.

a) Service Load Conditions - The load combinations for the elastic working stress design methods are:

1) $S = D + L$

2) $S = D + L + E$

3) $S = D + L + Hu$

4) $1.5S = D + L + T_o + R_o$

5) $1.5S = D + L + T_o + R_o + E$

6) $1.5S = D + L + T_o + R_o + Hu$

The load combinations for the plastic design methods are:

1) $Y = 1.7 (D +L)$

2) $Y = 1.7 (D + L + E)$

3) $Y = 1.7 (D + L + Hu)$

4) $Y = 1.3 (D + L + T_o + R_o)$

5) $Y = 1.3 (D+ L + T_o + R_o + E)$

6) $Y = 1.3 (D+ L + T_o + R_o + Hu)$

Both cases of L having its full value or being completely absent are considered.

b) Factored Load Conditions - The load combinations for the elastic working stress design methods are:

1) $1.6S = D + L + T_o + R_o + E'$

2) $1.6S = D + L + T_o + R_o + W$

3) $1.6S = D + L + T_a + R_a + P$

**4) $1.6S = D + L + T_a + R_a + P + Y_j + Y_r + Y_m + E$

**5) $1.7S = D + L + T_a + R_a + P + Y_j + Y_r + Y_m + E'$

The load combinations for the plastic design methods are:

1) $0.9Y = D + L + T_o + R_o + E'$

*2) $0.9Y = D + L + T_o + R_o + W$

3) $0.9Y = D + L + T_a + R_a + 1.5P$

4) $0.9Y = D + L + T_a + R_a + 1.25P + Y_j + Y_r + Y_m + 1.25E$

5) $0.9Y = D + L + T_a + R_a + 1.0P + Y_j + Y_r + Y_m + E'$

In the above combinations, thermal loads are neglected when it is shown that they are secondary and self-limiting in nature, and the material is ductile. Both cases of L having its full value or being completely absent are considered. T_a need not be considered to act simultaneously with Y_r and Y_m since accident temperature does not occur until after pipe rupture.

* Snow is not considered to be concurrent with tornado.

** In computing the required section strength, S, the plastic section modulus of steel shapes may be used.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Concrete Internal Structures

The primary and secondary shield walls, the refueling cavity, and the steam generator and pressurizer enclosures constitute the major structural components of the Containment Concrete Internal Structures. The analysis and design of these structures is based on the following procedures.

a) A preliminary sizing and proportioning of the structural members was performed in order to establish the mass-stiffness characteristics to be incorporated in the seismic dynamic analysis and in the finite element structural analysis.

b) Detailed structural analyses were made using the preliminary sized and proportioned members.

   Detailed structural analyses, which incorporate the preselected structural properties as well as the load information, including the dynamic analysis results, were performed by the finite element method of computer analysis using proven, industry-accepted computer programs, such as MRI/STARDYNE, MSC/NASTRAN, or ANSYS, described in Appendix 3.8B.

   The detailed structural analyses furnished stress results, such as bending and torsional moments, normal forces, and shears for each structural element under the action of individual loads and their combinations, as specified in Sections 3.8.3.3.1 and 3.8.3.3.2.

c) A final sizing and proportioning of the structural members was made to satisfy the strength and deformation requirements for the members.

Proportioning or reinforced concrete sections is based on the governing loading conditions in accordance with the strength design method of ACI-318-71. Adequate reinforcing bars are provided to account for tension, and where required, for compression and shear. The reinforcing bars are basically arranged in a rectangular mesh pattern consisting of mutually perpendicular vertical and horizontal bars at each face of the member. A sufficient length of embedment is provided at wall-to-base or wall-to-wall junctures to ensure conditions of fixity or continuity.

Secondary structural components, such as equipment supports and foundations with well-established structural boundaries, are designed separately using conventional methods of structural design.

The Concrete Internal Structures, including the primary shield wall, secondary shield wall, refueling cavity wall, and steam generators and pressurizer walls, are incorporated into a single finite element model with a fixed boundary at the top of the internal mat. Steam generator and reactor coolant pump vertical support pedestals are not included in the model. The STARDYNE computer program, using a thin shell approach, was used for the static analysis of the Concrete Internal Structures. The use of the thin plate element to represent the thick wall was evaluated by calculating outer and inner face fiber stresses by thick and thin cylinder formulas. Comparison of the outer face fiber stresses calculated by the two methods indicates that the thin wall approach that was used is conservative because the outer fiber stresses obtained by thin cylinder calculations are higher than those from the thick cylinder calculations.

The lowest factor of safety that was provided for the reinforcement of the containment internal structures thick walls is 1.4. The wall is the 9 feet 3 inch thick primary shield wall, and is shown in Figure 3.8.3-2.

The Concrete Internal Structures are designed to withstand all imposed loads and load combinations listed in Section 3.8.3.3 within the strength limits specified in Section 3.8.3.5.

Earthquake forces on the Concrete Internal Structures are determined by a dynamic analysis in accordance with the techniques described in Section 3.7. The dynamic loads thus determined are then applied as static loads on the concrete structures, and the static analysis using the finite element model described above is performed.

The pressure within or across a compartment, generated by different postulated breaks, is considered in the analysis.

The containment subcompartments may be subjected to pressure transients caused by the mass and energy releases from postulated pipe ruptures within their boundaries. Analyses have been made to determine the peak pressure that could be produced by a line break discharging into the subcompartments.

Subcompartments which have been analyzed are the reactor cavity and the steam generator-pressurizer combination subcompartment. These analyses represent the original design bases analyses for the subcompartments.

Since LBB has subsequently been approved for application at HNP, the large RCS breaks are eliminated from consideration. Instead, for SGR/PUR, evaluation of postulated breaks in the pressurizer surge and spray lines, RHR lines, and accumulator nozzles were performed to demonstrate that the associated dynamic effects are bounded by the original design bases.

Discussions and figures in Section 3.8.3 reflect the original design basis subcompartment analysis. Section 6.2.1.2a discusses results of SGR/PUR evaluations. The pressure differentials for each containment subcompartment have been calculated using the RELAP 4 computer code (discussed in Section 6.2.1.2).

The analysis techniques, definition of volumes, subcompartment nodalization sensitivity study, models investigated, and postulated pipe breaks considered, together with the results of the analysis including pressure, force, and moment transients, are presented in Section 6.2.1.2.

The primary shield wall is designed for a pressure distribution, spatially varied, obtained as a result of the postulated break of the cold leg pipe. The design pressure distribution (which includes a 40 percent margin over calculated values), spatially varied, for a 150 sq. in. cold leg pipe break is shown on Figure 3.8.3-13.

The design pressure distribution was appropriately rotated around the reactor cavity to account for a break occurring at each hot or cold leg location.

An analysis of the primary shield wall was also performed to confirm that the effects of worst case impact due to reactor pressure vessel (RPV) head drop would be withstood by the wall. Dynamic analysis was performed to obtain the time history of load propagation in the wall. The results were used in static analyses to evaluate bearing at the RPV support base plate, wall capability as a column beneath the base plate, and full wall capability for the loadings induced by the head drop. As-built concrete compressive strength was considered in the analysis.

For the design of the steam generators and pressurizer walls, a break in the compartment housing steam generator Loop 1 has been proven to give the worst pressure transient, since that compartment is the smallest among the three.

All walls within those areas are designed for maximum peak transient pressures multiplied by a dynamic load factor to account for the dynamic nature of the load. The primary and secondary shield walls, and the floors and walls within the subcompartments, are designed for pressures spatially and circumferentially varied, as shown on Figure 3.8.3-14. In this figure the differential pressures shown do not include the dynamic load factors. For Loop 2 and Loop 3 individually, the spatial variations developed for the Loop 1 steam generator subcompartment were applied, or a separate analysis was performed by using the same design methodology developed for Loop 1. The use of the Loop 1 subcompartment pressure variations in the design of Loops 2 and 3 subcompartments has been studied and is conservative.

The walls of the control rod drive mechanism (CRDM) room were designed for the worst conditions generated by the postulated double ended cold leg guillotine break with the reactor coolant pump hatch opened. The pressure distribution on the walls is shown on Figure 3.8.3-15.

The impulse effects due to the pressure within or across a compartment generated by different postulated breaks, as described above, and due to the jet impingement forces, are considered; and the equivalent static loads, including a dynamic load factor to account for the dynamic nature of the loads, are determined by using the analysis procedures described in References 3.8.4-48, 3.8.4-49 and 3.8.4-50. The ductility ratios used in this analysis are in accordance with Reference 3.8.1-3.

As a general rule, elastic behavior satisfying conditions of equilibrium of forces and compatibility of strains is the basis for the static analysis of structures under both the service and factored load conditions. However, local yielding of reinforcing bars due to concentrated loads associated with pipe rupture is permitted provided there is no loss of safety function.

The impactive effects due to pipe rupture loads or internally generated missiles are considered; and the equivalent static loads, including a dynamic load factor to account for the dynamic nature of the loads are determined.

The structural responses are determined by idealization of the actual structure to an equivalent single degree of freedom system and idealization of the impulse load time history to a simple mathematical form. Elasto-plastic behavior of the structural element is assumed, with appropriate ductility ratios as described in Reference 3.8.1-3.

Miscellaneous equipment, compartment slabs, and walls are analyzed by using conventional beam/slab design assumptions and equations. Loadings for these structures consist of dead, live, seismic, pipe rupture, jet impingement, and subcompartment differential pressures, where applicable.

3.8.3.4.2 Miscellaneous Steel Internal Structures

The basic structural components considered in the design and analysis of the steel internal structures, in addition to those described in Section 3.8.3.1.6, are the pipe restraint structures, cable tray supports, H&V duct supports, supports interfacing with NSSS equipment, and miscellaneous platforms. The structures are steel framed; and floors are grating, checkered plate, or concrete, depending upon the requirements for the floor area.

Structural steel framing is analyzed for the effects of the load combinations by conventional procedures for structural analysis and sized for the required section strength, as specified in Section 3.8.3.3. For the floor framing, except where specified or detailed otherwise on the design drawings, all floor beams and girders are considered to be pin connected at their ends.

Compression flanges of the girders are not considered to be laterally supported by the grating floors. The stairs and elevator shaft are seismically analyzed in accordance with Regulatory Guide 1.29, Positions C.2 and C.4.

Miscellaneous platforms not supporting safety-related or Seismic Category I equipment are designed not to fall down on safety-related equipment and are seismically analyzed in accordance with Regulatory Guide 1.29, Positions C.2 and C.4.

The containment crane girders, made of 16 sections which encompass the circumference of the Containment, are set on brackets that conform to the requirements discussed in Section 3.8.1. Each section of girder spans between four brackets and is anchored at one intermediate bracket to take tangential load; it is permitted to slip tangentially at the other intermediate bracket and two end brackets. All brackets have been analyzed and designed to take radial load. Provisions have been made to eliminate major thermal loads from temperature increase and major interaction loads from test pressure. For additional description of the analysis, see Section 3.8.1.4.4.12. The girders have been analyzed for the effects of the load combinations by conventional procedures for structural analysis, and have been sized for the required section strength as specified in Section 3.8.3.3. For crane capacities, see Section 9.1.4.2.2.8.

3.8.3.4.3 Pool Liners

The design of the pool liners for the refueling pool, fuel transfer canal, spent fuel pool, and spent fuel cask pool is based on the following requirements:

a) During Construction - Since the pool liner plates are not used as forms and are attached to embedded plates and structural shapes after the concrete is in place, no construction loads are experienced by the liner plates.

b) During Operation and Shutdown - Inasmuch as the pool liners function as watertight membranes, with the strength of the pool structures furnished by the reinforced concrete walls and pool floors, only changes in temperature which cause differential expansion or contraction of the liners relative to stiffeners buried in concrete are considered.

3.8.3.4.4 Design and Analysis Procedure for the Reactor Coolant System (RCS) Supports

The analysis and design of the RCS supports are described in Section 5.4.14.

3.8.3.5 Structural Acceptance Criteria

The structural acceptance criteria for the Containment Internal Concrete Structures and the internal and other Seismic Category I structural steel structures consists of compliance with the following requirements:

a) Concrete Structures - To assure that the structural integrity of Category I concrete structures is maintained for the service and factored load conditions, the limits of the stress and strain intensity of concrete generally follow the strength design method requirements of ACI 318-71.

   Using the factored loads, the various components have the required load capacity if the stresses in them do not exceed the yield strengths of the materials used. To provide for the possibility that small, adverse variations in dimensions and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in a net under capacity of the component, the load capacities of the individual structural members are reduced by a reduction factor "φ" for the design cases.

   The factors were established for the design on the basis of the function of the component and the effect on its net capacity of the variations enumerated above. These factors are generally in accordance with the ACI 318-71 Code and are tabulated in Table 3.8.3-1.

b) Steel Internal Structures - Structural steel framing is designed for the loading combinations, given in Section 3.8.3.3.3, to exhibit either elastic or plastic behavior in all load carrying elements. To assure that the structural integrity of Seismic Category I steel structures is maintained, limits on the resulting stresses and the required strength capacities as presented in Section 3.8.3.3.3 for service and factored loads are observed.

c) RCS Support Structures - The structural acceptance criteria for the RCS component supports is described in Section 5.4.14.

3.8.3.6 Materials, Quality Assurance, and Special Construction Techniques

3.8.3.6.1 Materials

3.8.3.6.1.1 Concrete Structures

a) Cement - Cement conforms to the requirements of ASTM C-150, "Portland Cement", Type II. Cement is procured and tested by the manufacturer at intervals in accordance with ASTM C-150.

In addition to the tests required of the cement manufacturers, the following tests are performed by CP&L or an organization designated by CP&L once every six months:

ASTM C-114 - Chemical Analysis of Hydraulic Cement

ASTM C-115 - Fineness of Portland Cement by the Turbidimeter, or ASTM C-204 - Fineness of Portland Cement by Air Permeability Apparatus

ASTM C-151 - Autoclave Expansion of Portland Cement

ASTM C-191 - Time of Setting of Hydraulic Cement by Vicat Needle

ASTM C-109 - Compressive Strength of Hydraulic Cement Mortars

ASTM C-190 - Tensile Strength of Hydraulic Cement Mortars

During construction, if cement has been in storage at the site for six months, the following tests are performed by CP&L prior to further use of the cement to check the storage environmental effects on the cement characteristics:

ASTM C-191 - Time of Setting of Hydraulic Cement by Vicat Needle

ASTM C-109 - Compressive Strength of Hydraulic Cement Mortars

Table 3.8.1-5 shows a summary of in-process test results for the cement.

a) Aggregates - Aggregates conform to the requirements of ASTM C-33, "Concrete Aggregates", with the following exceptions:

1) The use of blast-furnace slag is not permitted without the specific written approval of the Engineer in each instance.

2) ASTM C-40 "Organic Impurities in Sands for Concrete", is used; sands that produce a color darker than standard are tested in accordance with ASTM C-87, "Effect of Organic Impurities in Fine Aggregate on Strength of Motar." If the relative strength at seven days, calculated in accordance with Section 10 of ASTM C-87 is less than 95 percent, the sand is rejected. Deviation from this rule is permitted only upon written approval of the Engineer in each instance.

3) Abrasion - The coarse aggregate used for concrete for all structures except the seal mat has a maximum of 40 percent loss in weight when tested by the Los Angeles Machine. The coarse aggregate used for concrete for the seal mat meets the requirements of ASTM C-33.

4) Gradation - In addition to the gradations listed in ASTM C-33, and aggregate designated 78-M (State of North Carolina designation) is used in special areas such as around major penetrations or in reinforcing steel congested areas, with the approval of the engineers. This aggregate meets all other qualifications of ASTM C-33, with the exception of gradation analyses. The results during preliminary concrete mix design have been satisfactory and in accordance with the requirements of ASME Section III, Division 2/ACI-359 Code.

The aggregate is tested by the supplier for gradation and fineness modulus every 500 tons and for specific gravity and absorption every 5000 tons. In addition, aggregate to be used for concreting of Containment Internal Concrete Structures is tested by CP&L or an organization designated by CP&L during concrete production for the requirements and respective frequencies tabulated below:

Requirements	Test Method	Frequency
Gradation	ASTM C-136	Once daily during production (twice daily during production if more than 200 yds. of concrete are placed)
Moisture Content	ASTM C-566	Twice daily during production
Material finer than #200 Sieve	ASTM C-117	Daily during production
Organic Impurities	ASTM C-40	Daily during production
Friable particles	ASTM C-142	Monthly during production
Lightweight Particles	ASTM C-123	Monthly during production
Specific Gravity and Absorption	ASTM C-127 or ASTM C-128	Monthly during production
Los Angeles Abrasion	ASTM C-131 or ASTM C-535	Every 6 months
Potential Reactivity	ASTM C-289	Every 6 months
Soundness	ASTM C-88	Every 6 months
Water Soluble Chlorides	ASTM D-1411	Monthly during production

A sample of in-process test results of the aggregate appears in Tables 3.8.1-6 through 3.8.1-11.

b) Water - Water used in concrete mixing is sampled, tested, and analyzed initially for use in trial mixes and monthly thereafter for use in production concrete by CP&L or an organization designated by CP&L to assure conformance with the following limits and tests:

1) The mixing water, including that contained as free water in the aggregates, does not exceed 250 ppm of chlorides as Cl-as determined by ASTM D-512, "Chloride Ion in Industrial Water and Industrial Waste Water." The water-soluble chloride content of the aggregate is established by the methods described in ASTM D-1411, "Water Soluble Chlorides Present as Admixes in Graded Aggregate Road Mixes."

2) Sulfates 1000 ppm maximum, as determined by ASTM D-516, Method A

3) The total solids content for the mixing water does not exceed 2,000 ppm as measured by American Public Health Association's "Standard Method for Determination of Total Solids."

4) In addition to the above, the water is tested monthly in accordance with the following tests:

Test Method	Requirement
ASTM C-109	Effect on Compressive Strength
ASTM C-191	Setting Time
ASTM C-151	Soundness
APHA 208*	Total Solids
ASTM D-512	Chlorides

* Standard Methods 14th Edition, 1975 American Public Health Association

Table 3.8.1-12 shows a summary of in-process test results for water.

c) Admixtures - Where necessary, admixtures are added to entrain air and increase workability, while reducing the water-cement ratio and retarding the initial set time.

Admixtures are used for all concrete construction in accordance with the following requirements and tested by the supplier at intervals conforming with ASTM C-260 and ASTM C-494.

1) Air Entraining Agent - Air entraining agents conform to ASTM C-260 in proportions such that air-entrainment as specified in ACI-318 is produced as determined by ASTM C-138, C-233, and C-173 or C-231. In order that the proportions may be adjusted to produce the specified percentage of air under varying conditions, the agent is not combined with the cement or other admixture prior to batching.

2) Water Reducing Agents - Water reducing agents used in the concrete conform to ASTM C-494. Final approval of the admixture was contingent upon satisfactory tests with the cement and aggregates used in the work. A set retarding, water reducing agent conforming to ASTM C-494 Type D is used during hot weather in accordance with ACI-305.

Flyash, if used, conforms to ASTM C-618, Class F, and is tested in accordance with ASTM C-311 for every 100 tons of flyash utilized for concrete. Flyash does not exceed 25 percent by weight of cement in the final mix. Concrete produced with flyash meets all of the ASTM requirements specified for standard concrete.

Table 3.8.1-13 shows a sample of in-process test results for preliminary acceptance tests of the admixtures.

d) Cement Grout - Cement, aggregate, water, and admixtures for grout conform to the requirements stipulated above. The proportions of materials are based upon trial mixes using the same type and brand of ingredients as used for construction to meet the specified requirements of consistency, shrinkage, and compressive strength. The tests are performed in accordance with Corps of Engineers' methods of CRD-C-79 and CRD-C-588-76 and ASTM C-109.

e) Concrete - Structural concrete for the Containment Internal Structures is specified to have a minimum design compressive strength of 5000 psi (Class X) 28 days after placing. The concrete mixes yield an air dry weight of at least 140 lb. per cu. ft., at 28 days, in accordance with ASTM C-642.

The design of concrete mixes is in accordance with ACI 211.1-74 "Recommended Practice for Selecting Proportions for Normal and Heavyweight Concrete". The previously specified ingredients are used to obtain material proportions for the specified concrete. During construction, minor modifications of design mixes may be necessitated by variations in aggregate gradation or moisture content.

Concrete construction procedures, including stockpiling, storing, batching, mixing, conveying, depositing, consolidating, curing, and construction joint preparation are in accordance with the provisions of the ACI 318 Code. The requirements of NRC Regulatory Guide 1.55, with the exceptions described in Appendix 3.8A, are also complied with.

f) Reinforcing Steel

1) Reinforcing Bars - All reinforcing bars are new billet steel in accordance with ASTM A-615 Grade 60 (60,000 psi minimum yield strength). Placing and splicing of No. 11 and smaller bars meet the requirements of the ACI 318 Code.

At least one full diameter reinforcing steel sample of each bar size is tested by the reinforcing steel supplier for each 50 tons or fraction thereof of reinforcing bars produced from each heat. No specific method of sample selection is imposed upon the reinforcing steel supplier. These samples are tested by the supplier and accepted by the user based upon ASTM A-370 and ASTM A-615 specifications. All of the requirements of NRC Regulatory Guide 1.15 are complied with, except as noted in Appendix 3.8A and Section 1.8.

All samples are tested for:

(a) Tensile yield strength

(b) Tensile ultimate strength

(c) Elongation in 8 in.

(d) Weight

Inspections are performed as necessary to verify compliance with the specifications.

2) Mechanical Splicing - No. 18 reinforcing bars are spliced with mechanical (Cadweld) splices and No. 11 reinforcing bars may be spliced with mechanical (cadweld) splices even though the majority of No. 11 splices are lap splices. Mechanical splices are in accordance with the requirements of NRC Regulatory Guide 1.10, with the exceptions described in Appendix 3.8A and Section 1.8. The Cadweld Inspection program is also in conformance with Regulatory Guide 1.10.

The average tensile strength of the splices is equal to or greater than the specified ultimate tensile strength of the rebar. The minimum acceptance tensile strength of any splice is 125 percent of the specified minimum yield strength for the particular bar size in the ASTM specification.

All completed splices are visually inspected at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Splices that fail to pass the visual inspections are discarded and replaced, or repaired by welding. Splices that have been discarded are not used for tensile testing.

The splice samples are either production or sister splices for straight bars and straight sister splices for all curved bars. Selected splices are tested in accordance with the following schedule for each position, bar size, grade of bar, and splicing crew, as follows:

Test frequency where only production splices are tested:

1 out of first 10 splices

1 out of next 90 splices

2 out of the next, and each subsequent unit, of 100 splices

Test frequency where combinations of sister and production splices are tested:

1 production splice of the first ten production splices

1 production and three sister splices for the next 90 production splices

1 splice, either a production or sister splice for the next, and subsequent units, of 33 splices. At least one-fourth of the total number of splices tested are production splices.

Straight sister splices are substituted for production samples for splicing sleeves arc welded to structural steel elements.

To be acceptable, sound nonporous filler metal must be visible for the full circumference at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed 1/4 in. from the end of the sleeve due to the packing material. Such indentation is not considered to be a poor splice.

The following reasons constitute cause for visual rejection of splice:

a) Slag in the tap hole where the slag exceeds the thickness of the sleeve wall.

b) Spongy appearance of the filler metal caused by gas blowout.

c) Void areas for each end of splices in any position exceeding the allowable values tabulated below:

Bar Size	Allowable Void Area (*) (Sq. In.)
9	1.02
10	1.03
11	1.53
14	2.15
18	3.00

(*) Allowable void areas for bar sizes smaller than No. 9 are not established.

Joints which do not meet the visual quality acceptance standards are rejected and either completely removed and replaced or repaired by welding.

Repair welding is done by the manual shielded metal arc (SMA) process. The welding electrodes for joining the reinforcing bar to the splice sleeve conform to AWS Specification A5.5 Classification E 8018-B2 1/8 in. or 5/32 in. diameter. Electrodes for joining the splice sleeves to structural steel components conform to AWS A5.1 Classification E 7018.

All rust, scale, oil, grease, dirt, or other foreign substances are removed from the areas to be welded. All degreasing is done by swabbing the weld area with acetone or other approved solvent or cleaner. No residual cleaning compounds are left on the surface prior to welding.

The base material is preheated to 300 F minimum and an interpass temperature of 300 F minimum is maintained during welding. Amperages and voltages are in accordance with the electrode manufacturer's recommendation. All slag, flux, or foreign materials remaining on any bead of welding are removed before laying down the next or successive bead. Stress relieving is not required. After completion of welding, a visual inspection is made for the presence of cracks, surface porosity, slag inclusions, undercut, and inadequate weld size.

For splices in the Containment Internal Structures that fail to meet the tensile test acceptance standards, the following procedures are specified:

a) If any production or sister splice used for testing fails to meet the strength requirements and failure occurs in the bar, the cause of the bar break is investigated. Any necessary corrective action affecting splice samples are implemented prior to continuing the testing frequency.

b) If the running average tensile strength of 15 consecutive samples fails to meet the tensile requirements, splicing is halted. The cause(s) of the failure are investigated, and the necessary corrective action is taken. When splicing is resumed, the splicing frequency is started anew.

3) Welded Splices - Welded splices, if used, comply with Regulatory Guide 1.94.

3.8.3.6.1.2 Structural Steel Structures

Structural steel for the Containment Internal Structures and other Seismic Category I structures is in accordance with the AISC Code of Standard Practice for Steel Buildings and Bridges (AISC Code of Standard Practice), the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings (AISC Buildings Specification), Part I or Part II, according to whether elastic or plastic design methods are used, and the AWS Structural Welding Code (AWS Code) D1.1.

3.8.3.6.1.3 Pool Liner

The requirements for the materials for the refueling pool, fuel transfer canal, spent fuel pool,and spent fuel cask pool liner, and the attachment materials to the pool liner, are in accordance with the following:

Special material testing, examination, and repair for the liner plate, and identification of welding materials are in accordance with the ASME Code Section III, Division 1, Subsection ND.

The various components for the liner plate are as follows:

a) Austenitic Stainless Steel Plate - ASME SA-240 Grade 304 with ASME SFA 5.4 E 308 or ASME SFA 5.9 ER 308 electrodes for joining stainless steel to stainless steel, and ASME SFA 5.4 E 309 or ASME SFA 5.9 ER 309 for joining stainless steel to carbon steel.

b) Austenitic Stainless Steel Forging - ASME SA-403 Grade 304

c) Embedment and Other Structural Steel - ASTM A-36

Fabrication and construction of the pool liner structures is in accordance with the ASME Code Section III, Division 1, Subsection ND and ASME Code Section IX.

d) Fuel Pool Nozzles shall be designed and fabricated in accordance with ASME Code Section III, Division 1, Subsection ND. Fabrication and testing of seal welds between nozzles and fuel pool liner plates shall meet welding requirements of the fuel pool liner.

3.8.3.6.2 Quality Assurance

The overall quality assurance program is in accordance with the Engineering and Construction QA program which was approved by the NRC during the construction permit review. Materials testing, fabrication and construction, and construction testing and examination are in accordance with applicable provisions of ACI specifications. The test methods are frequency of testing for concrete and concrete ingredients conform to the requirements stipulated in Section 3.8.3.6.2.1.

3.8.3.6.2.1 Concrete Structures

The services of an independent laboratory were obtained prior to commencing concrete work. This laboratory or CP&L has determined control mixes to consistencies satisfactory for the work, using the proposed materials, in order to determine the suitable mix proportions necessary to produce concrete conforming to the type and strength requirements specified.

Proportions for concrete mixes are based on laboratory or CP&L trial batches made of materials specifically approved for use and from which individual water/cement ratio curves were developed. Mix proportions were selected to ensure maximum workability and conformance with the concrete compressive strength requirements.

Proportions for the laboratory of CP&L trial batches and the subsequent mix adjustments were in accordance with ACI 211.1-74, "Recommended Practice for Selecting Proportions for Normal and Heavyweight Concrete."

Initially, concrete mix proportions were selected from the appropriate water/cement ratio curves so that the average compression strength exceeded f_c', i.e., 5,000 psi (Class X), and 4,000 psi (Class A), by 1,200 psi. In addition, proportions were selected so that the dry unit weight would not be less than 140 lb./ft.^{3}, and the slump and air content would be four in. and 3 to 6 percent, respectively.

The initial mix proportions were used until sufficient test data (concrete cylinders tested in accordance with ASTM C-39) became available and an over-design strength considerably less than 1,200 psi could be established.

New mix proportions were selected based on the water-cement ratio curves modified by field tests, and the new over-design strength was established so that the requirements of Sub-Subparagraph CC-2232.2(b) of the ASME Section III, Division 2/ACI 359 Code, Winter 1975 addenda are complied with.

Tables 3.8.1-14 through 3.8.1-16 show summaries of the in-process test results for concrete with compression strengths of 5000, 4000, 3000, and 2000 psi.

For concrete used in the Containment Internal Structures, the properties tabulated below are measured - prior to construction - in accordance with the respective specifications and the applicable conditions noted below:

Property	Specification	Age of Sample (Days)	Temperature (F)
1. Slump	ASTM C-143	0	Ambient
2. Compressive Strength	ASTM C-39	3, 7, & 28	As per ASTM C-39
3. Flexural Strength	ASTM C-78	28	As per ASTM C-78
4. Splitting Tensile Strength	ASTM C-496	28	As per ASTM C-496
5. Static Modulus of Elasticity	ASTM C-469	28	As per ASTM C-469
6. Poisson's Ratio	ASTM C-469	28	As per ASTM C-469
7. Coefficient of Thermal Conductivity	CRD-C44	28	As per CRD-C44
8. Coefficient of Thermal Expansion	CRD-C39	28	As per CRD-C39
9. Creep of Concrete in Compression (*)	ASTM C-512	2,7,28,90 days and 1 yr.	As per ASTM C-512
10. Shrinkage Coefficient(*) (length of change of cement mortar and concrete)	ASTM C-157	4,7,14, & 28 days & 8, 16, 32, & 64 weeks	As per ASTM C-157
11. Density (Specific Gravity)	ASTM C-642	28	As per ASTM C-642

* These tests are concurrent with construction.

Concrete slump, temperature, air content, and mechanical properties examinations are performed on a common sample to establish conformance with the provision listed above.

Concrete is sampled at the point of delivery into the forms, or at a centralized location, as approved by CP&L.

The methods used in sampling, making, curing, and testing the concrete samples, either in the field or in the laboratory, are in accordance with the appropriate ASTM Standards and include, but are not necessarily restricted to, the following standards:

ASTM C-172 - Sampling Fresh Concrete

ASTM C-31 - Making and Curing Concrete Compressive and Flexural Test Specimens in the Field.

ASTM C-192 - Making and Curing Concrete Test Specimens in the Laboratory.

ASTM C-39 - Compressive Strength of Cylindrical Concrete Specimens.

ASTM C-567 - Unit Weight of Structural Lightweight Concrete.

ASTM C-138 - Unit Weight, Yield, and Air Content (Gravimetric) of Concrete.

Three-day, seven-day, and 28-day tests are made on 6 x 12 in. cylinders. For each design mix, a correlation between three-day, seven-day, and 28-day strengths is made in the laboratory. Soon after a job starts, a similar correlation is evolved for samples of concrete taken in the field. After the correlation has been established, the results of the seven-day tests may be used as an indicator of the compressive strengths which can be expected at 28 days. If seven-day tests show compressive strengths that are too low, corrective measures are taken at once without waiting for the results of the 28 day tests.

The number of tests cylinders made under various conditions is as follows:

	Min. No of Cylinders	Test Breaks 3-Day	7-Day	28-Day	Extra
1) Until final determination of each design mix for each class of concrete placed in any one day.*					
Each 100 cu. yd. or fraction thereof	14	4	4	4	2
2) For each class of concrete of determined mix placed in any one day					
Each 100 cu. yd. or fraction thereof (or a minimum of one set per day)	4	-	1	2	1

*This is intended to cover only those new design mixes, created by modification of determined design mixes, which have not been proven by lab tests prior to their placement.

The number of cylinders may be reduced to a minimum of four per set if a sufficient number of cylinders (e.g., 100) for a modified mix have proven the mix to be acceptable. Engineering concurrence will be obtained unless lab qualification tests are completed.

The extra cylinders are tested if it is necessary to substantiate 7 or 28 day test results.

The concrete cylinders are tested for compressive strength in accordance with ASTM C 39. The strength level of the concrete is considered satisfactory if:

a) No individual strength test result falls more than 500 psi below the required class strength at 28 days.

b) The averages of all sets of three consecutive strength test results equal or exceed the required class strength at 28 days.

Each 28 day strength test result is the average of two cylinders from the same sample. The variation between the two cylinders must be not more than five percent of their average. A greater variation requires testing of the third (spare) cylinder to determine the average strength. If the third cylinder strength is also more than five percent from the average, the owner determines the reason for such a wide variation in test results and rectifies it.

The coefficient of variation for the tests on each mix, as determined in accordance with ACI 214, must not be greater than 15 percent. A greater variation will require a review of concrete batching, mixing, and transporting facilities and procedures to assure a reduction in this coefficient to the required 15 percent or lower.

The slump tests are performed as follows:

a) One slump test is performed for the first batch placed each day and thereafter for each 50 cubic yards of each class of concrete placed.

b) Slump tests are made for each concrete batch used for test cylinders.

c) Slump tests are made at any time the inspector has reason to suspect that the concrete slumps are not within the allowable tolerances.

The concrete air entrainment content and temperature is taken with each slump test. The concrete unit weight is determined daily during production in addition to slump, air content, and temperature. The batch plant scales are calibrated to the ASTM C-94 standard on a monthly basis. Mixer uniformity tests to the ASTM C-94 standard are performed initially and every six months.

The evaluation of the test results for concrete are in accordance with ACI 214. Sufficient tests are conducted to provide an evaluation of concrete strength.

During concrete operations, inspectors at the batch plant witness the mix proportions of each batch delivered to construction, and periodically sample and test the concrete ingredients. The inspectors ensure that a ticket is provided for each batch, which documents the time loaded, actual proportions of the mix, amount of concrete, and the concrete design strength. The cleanliness of trucks, and the handling and storage of aggregate are checked by the batch plant inspectors. The concrete batch plant complies in all respects, including provisions for storage and precisions of measurements, with ASTM C 94, and the National Ready Mixed Concrete Association (NRMCA) - Certification of Ready Mixed Concrete Production Facilities. Water and ice additions, if necessary, are modified as required by measurement of the moisture content of the aggregate and gradation changes.

Other inspectors at the construction site inspect reinforcing and form placement, make a slump test, make test cylinders, check air content, check concrete temperatures, record weather conditions, and inspect concrete placing and curing. The requirements of Regulatory Guide 1.55, with clarifications described in Appendix 3.8A and Section 1.8, are complied with; the requirements of Regulatory Guides 1.10 and 1.15 with clarifications of Appendix 3.8A and Section 1.8 are also followed.

The following inspections are performed:

a) Visual inspection of fabricated reinforcement is performed to ascertain dimensional conformance with specifications and drawings.

b) Visual inspection of in-place reinforcement is performed by the inspectors to assure dimensional and locational conformance with drawings and specifications.

3.8.3.6.2.2 Structural steel structures

Construction of structural steel structures is as specified in the AISC Code of Standard Practice, the AISC Buildings Specification, and the AWS Structural Welding Code D1.1 Section 9, as well as the applicable component specifications.

3.8.3.6.2.3 Pool liner

Construction of the pool liner structures is in accordance with the ASME Code, Section III, Division 1, Subsection ND.

3.8.3.6.3 Special construction techniques

No unique or untried construction techniques were planned for the fabrication and placement of concrete and erection of concrete reinforcing steel, structural steel, and pool liner.

3.8.3.7 Testing and In-service Surveillance Requirements

There is no planned systematic testing for the Containment Internal Structures or other Seismic Category I steel structures. The structural steel and connections are not required to be subjected to any tests. The framing and connections are generally accessible to visual inspection throughout the operating life of the plant. Seismic Category I steel structures are periodically inspected in accordance with Reg. Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." In addition, accessible elements could be inspected by nondestructive testing procedures, following the occurrence of an abnormal event.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

3.8.4.1 Description of the Structures

The following structures are designed to Seismic Category I requirements:

a) Containment Building and Internal Structures

b) Reactor Auxiliary Building

c) Fuel Handling Building, Unloading Bay

d) Waste Processing Building

e) Diesel Generator Building

f) Tank Building

g) Diesel Fuel Oil Storage Tank Building

h) Main Dam and Spillway

i) Auxiliary Dam and Spillway

j) Auxiliary Separating Dike

k) Seismic Category I Underground Electrical Duct Runs and Manholes

l) Structures for the Emergency Service Water System

m) Masonry Walls in Seismic Category I Structures - Walls are not designated Seismic Category I. Further details are provided in Section 3.8.4.8.

n) Tanks - A list of Seismic Category I Tanks is given in Table 3.8.4-3.

A general layout of Seismic Category I buildings at the plant island is shown on Figure 3.8.4-1.

3.8.4.1.1 Containment Building and Internal Structures

For the description of the concrete Containment and internal structures, see Sections 3.8.1, 3.8.2, and 3.8.3.

3.8.4.1.2 Reactor Auxiliary Building

The Reactor Auxiliary Building (RAB) houses engineered safeguards and supporting systems, switchgear, sampling rooms, and Control Room.

The major piping systems in the Reactor Auxiliary Building are listed as follows:

a) Primary and Demineralized Water

b) Containment Spray

c) RHR and Safety Injection

d) Steam Generator Blowdown

e) Feedwater

f) Main Steam Supply

g) Auxiliary Steam and Condensate Return

h) Service and Essential Services Chilled Water

i) Auxiliary Feedwater

j) Chemical and Volume Control

The Seismic Category I high energy pipe lines and radioactive pipe lines are protected within an enclosed pipe tunnel system and/or supporting systems which are designed to Seismic Category I requirements.

The Reactor Auxiliary Building (RAB-1) consists of two independent structures. One section is designated as RAB 1. The adjacent section is designated as RAB-Common.

The RAB-1 is a reinforced concrete structure, 207 ft. long by 187 ft. wide, varying in height from 69 ft. to 134 ft. from the top of foundation mat to the top of roof. The top of the foundation mat varies from 24 ft. to 70 ft. below finished grade, Elevation 260 ft.

The RAB-Common is a reinforced concrete structure, 120 ft. long by 187 ft. wide by 88 ft. high from the top of mat to the top of roof. The top of the foundation mat is at Elevation 236 ft. except for the pipe tunnel area, which is at Elevation 216 ft.

These buildings are cast-in-place reinforced concrete structures. The floors are supported on beams and girders which are in turn supported on interior columns and/or exterior walls. Where interior shear walls are installed, the beams and girders are supported on the shear walls. All interior shielded walls and partitions, other than shear walls, are either reinforced concrete or concrete block, and are not load bearing. Provisions are made for installation of these walls after the framing, floor systems, and equipment have been installed.

The buildings are supported on separate foundation mats 10 ft. thick which are founded on suitable rock.

A general layout of the Reactor Auxiliary Building is shown on Figures 3.8.4 2 through 3.8.4-6.

3.8.4.1.3 Fuel-handling building

The Fuel-Handling Building (FHB) houses: (1) facilities for storing, moving and handling both new-fuel and spent-fuel, (2) secondary waste equipment, such as evaporators, demineralizers, heaters, condensers and associated pumps, filters, and control panels, and (3) equipment for the Reactor Auxiliary Building, such as recycle evaporators, recycle holdup tanks, and heating, ventilating, and air conditioning ducts and associated pumps, filters, and hydrogen purge unit.

The spent fuel storage facilities consist of three spent fuel pools. The fuel pools are cast-in-place reinforced concrete structures completely lined with stainless steel plate. The dimensions of two of the spent fuel pools are 50 ft. long, 27 ft. wide, and 40 ft. deep. The dimensions of the other fuel pool are 32 ft. long, 20 ft. wide, and 40 ft. deep.

The fuel transfer canal, which interconnects the spent fuel pools, is 300 ft. long, 3 ft. wide, and 26 ft. deep, and lined with stainless steel plate.

Spent fuel is transferred from the Containment Building to the pools via the fuel transfer tube and through the fuel transfer canal. The fuel elements are handled by the spent fuel bridge crane.

The spent fuel is transferred to spent-fuel shipping casks, which are handled by a separate 150 ton capacity crane, which runs on rails supported by exterior wall brackets at Elevation 316 ft.

The new-fuel storage facility consists of one new-fuel pool which is 38 ft. long, 13 ft. wide, and 40 ft. deep (north new-fuel pool).

The new-fuel pool is lined with stainless steel plate. The new-fuel elements are handled by the spent fuel bridge crane and are transferred into the Containment Building via the fuel transfer canal and the fuel transfer tube.

The FHB is 466 ft. long, 50 ft. wide and varies in height from 100 ft. to 120 ft., from the top of foundation mats, at Elevation 216 ft. and Elevation 236 ft. to the top of the roof at Elevation 336 ft. Adjacent sections of the RAB and the Waste Processing building are structurally incorporated into the FHB for building stability. The overall FHB is 534 ft. long and 200 ft. wide.

The FHB structure adjoins the Containment Building on the east side. The FHB is a reinforced concrete Seismic Category I structure, supported on a 10 ft. thick foundation mat founded on suitable rock. All exterior walls, shear walls, interior columns, and floor slabs are cast-in-place reinforced concrete structures. All interior shielding or partition walls, other than shear walls, are either reinforced concrete or concrete block and are not load bearing. The reinforced concrete floors are at Elevation 236 ft., Elevation 261 ft., Elevation 286 ft., Elevation 305 ft., and

Elevation 324 ft., and the roofs are at Elevation 286 ft. and Elevation 336 ft. The floors and roofs are designed to act as horizontal diaphragms to transfer horizontal load to shear walls and to carry vertical loads simultaneously.

The exterior walls are waterproofed on the backfilled faces from the top of the mats up to one foot below grade level. All construction joints in exterior walls in contact with backfill have waterstops.

The northernmost section of the FHB, 98-67 ft. long by 56 ft. wide, is designated as the unloading area. It provides the necessary access facilities for railroad cars on which spent fuel shipping casks will be loaded or for any large vehicle that might require access into the building. The unloading area is a reinforced concrete structure, and the exterior walls below grade are waterproofed.

The sloshing effect due to earthquake on water in the fuel pools was considered in the structural design of the fuel handling building. An analysis of the building in the north-south direction, for example, for SSE including sloshing effect shows that the acceleration, shear and movement were, in general, reduced by approximately 5 percent. Therefore, for conservatism, the sloshing effect is not considered in the dynamic analysis of the building.

A general layout of the FHB is shown on Figures 3.8.4-7 through 3.8.4-14.

3.8.4.1.4 Waste processing building

The Waste Processing Building (WPB) houses the Liquid Waste Processing System, (LWPS) the Gaseous Waste Processing System, (GWPS) and the Solid Waste Processing System (SWPS), together with laboratories and personnel facilities.

The WPB is a reinforced concrete Seismic Category I structure, with cast in place reinforced concrete exterior walls and interior shear walls. All interior shielding or partition walls, other than shear walls, are either reinforced concrete or concrete block, and are not load bearing.

The WPB is 289 ft. long, 191 ft. wide, and 110 ft. high from the top of the foundation mat at Elevation 211 ft. The reinforced concrete floors at Elevation 236 ft., Elevation 261 ft., Elevation 276 ft., and Elevation 291 ft., and the roof at Elevation 321 ft., are designed to act as horizontal diaphragms to transfer horizontal loads to shear walls and to carry vertical loads simultaneously.

The building is supported on a 10 ft. thick reinforced concrete foundation mat, which in turn is founded on suitable rock. The exterior walls below grade are waterproofed on the backfilled faces. All construction joints in exterior walls in contact with backfill except for the 'T' Line wall between Column lines 6 and 12 and '12' Line wall between Column lines 'T' and 'S' are waterproofed with waterstops.

A general layout of the WPB is shown on Figures 3.8.4-15 through 3.8.4-19.

3.8.4.1.5 Diesel-generator building

The Diesel-Generator Building (DGB) houses the stand-by diesel generators, day tanks, silencers, and associated equipment.

The building is located 190 ft. east of the Turbine Building.

The DGB is a Seismic Category I, missile-proofed, reinforced concrete structure. The building is approximately 153 ft. long and 114 ft. wide.

The building is constructed on concrete fill, which is founded on suitable rock. The top of the foundation mat is 3.5 ft. below the grade floor Elevation 261.0 ft.; there is sand fill on top of the foundation mat which provides a space between the foundation mat and grade floor for the electrical main leads and pipe lines. The foundation mat is 6 ft. thick.

The building is cast-in-place concrete with reinforced concrete exterior and interior shear walls, and reinforced concrete floors. The floors are supported on wall beams. Interior walls, other than shear walls, are reinforced concrete walls or concrete masonry (block) walls and are not load bearing walls. All reinforced concrete floor slabs are designed to act as horizontal diaphragms to transfer horizontal forces to shear walls and to carry vertical loads simultaneously.

A general layout of the DGB is shown on Figure 3.8.4-20.

3.8.4.1.6 Tank Building

The Tank Building is located adjacent to the Reactor Auxiliary and Turbine Building. The Tank Building houses the refueling water storage tank, reactor make-up water storage tank, condensate storage tank, and other associated equipment. The Tank Building also houses the waste monitoring tanks, secondary waste sampling tank, their associated pumps, and other facilities.

The Tank Building is a reinforced concrete structure, approximately 142 ft. long by 63 ft. wide, and 83 ft. high. The top of the foundation mat is at Elevation 236 ft. and the top of the roof, which provides for missile protection, is at approximately Elevation 319 ft. The foundation mat is 8 ft. thick and is founded on suitable rock.

The top of the foundation mat is 24 ft. below the finished grade elevation of 260 ft. The Tank Building has cast-in-place reinforced concrete exterior walls, interior shear walls, and reinforced concrete floors, supported on shear walls, beams, and columns. All interior shielding or partition walls, other than shear walls, are either reinforced concrete or concrete block walls and are not load bearing walls. The exterior walls are waterproofed on the backfill face from the top of the mat to one foot below grade level Elevation 260 ft.; the waterproofing membrane terminates in reglets. All construction joints in exterior walls in contact with backfill have waterstops.

The condensate storage tank is protected against tornado missiles by concrete walls and roof.

A general layout of the Tank Building is shown on Figure 3.8.4-21.

3.8.4.1.6.1 Tank Building Seismic Analysis

The overall tank building analysis was run using a time history seismic input forcing function. A response spectra was developed from that analysis for Elevation 261 ft., the elevation of the bottom of the tank. This response spectra was broadened by plus or minus 15 percent as required by Regulatory Guide 1.122. This broadened response spectra resulted in higher accelerations for essentially the entire range of frequencies of the spectra curve, thus increasing the seismic input which in turn increased the accelerations and moments at the common points in detailed tank analysis as compared to overall tank building analysis.

The seismic input for the detailed tank analysis for the fundamental frequency of the tank of 15.67 cycles/sec is 0.61g against 0.39g of the actual unbroadened spectra curve (approximately 56 percent increase).

3.8.4.1.7 Diesel Fuel Oil Storage Tank Building

The Diesel Fuel Oil Storage Tank Building consists of a below grade reinforced concrete structure which provides for two reinforced concrete diesel oil tanks and two pumps. The structure is 94 ft. long, 86 ft. wide, and 24 ft. high (including the foundation mat); the top slab is at Elevation 263 ft. Access to the pumps is provided by two stairwells, each located at each corner of one end of the building. The building is supported on a reinforced concrete foundation mat which is founded on sound rock. The top of the foundation mat is at Elevation 242.25 ft; the mat is 3 ft. 3 in. thick.

The tanks have a capacity of 175,000 gallons each. Each compartment is 66 ft. long, 21 ft. wide, and 18 ft. 6 in. high; the free board is at least 12 in. The inside surfaces of the concrete compartments are lined with carbon steel to prevent leakage. Waterproofing membrane is also provided on the outside face of the exterior walls to prevent groundwater pressure on the steel linings. A drainage system between the concrete walls and steel liner is provided for any leakage through the waterproofing membrane and/or concrete foundation mat.

Each compartment is provided with an access manhole covered by a removable concrete cover and a vent pipe with a flame arrestor.

The pumps are housed in below grade cubicles separated by reinforced concrete walls.

A general layout of the Diesel Fuel Oil Storage Tank Building is shown on Figure 3.8.4-22. For sections and details see Figure 3.8.4-22a.

3.8.4.1.8 Main Dam and Spillway

The Main Dam and Spillway are located on Buckhorn Creek approximately 0.7 miles southeast of its confluence with White Oak Creek. The dam is a Seismic Category I, zone embankment, rockfill structure. For a description of the Main Dam and Spillway and discussion of the analysis of the Main Dam, see Section 2.5.6.

The Main Dam Spillway is located in the right abutment of the dam. It is an uncontrolled spillway with its crest at Elevation 220 ft. The spillway, which is Seismic Category I, consists of a reinforced concrete structure, a lined spillway chute section, and an energy stilling basin. A bridge is constructed over the spillway to provide access to the other side. A railroad bridge crosses the spillway chute. A low level release system is incorporated into the spillway structure. The main dam spillway plans, sections, and details are shown on Figures 3.8.4.34 through 3.8.4-36.

3.8.4.1.9 Auxiliary Dam and Spillway

The Auxiliary Dam and Spillway are located on a tributary of Buckhorn Creek approximately 3.2 miles upstream of the confluence of Buckhorn and White Oak Creeks. The dam is a Seismic Category I, zoned embankment, earth and random rockfill structure. For a description of the Auxiliary Dam and Spillway and discussion of the analysis of the Auxiliary Dam, see Section 2.5.6.

The Auxiliary Dam Spillway, which is Seismic Category I, is located at the right abutment of the dam. The uncontrolled spillway is 170 ft. long at the ogee crest elevation of 252 ft. The spillway chute is unlined except where lining is necessary for erosion control. A concrete apron and training walls are provided downstream of the spillway to prevent scouring of the toe of the dam. A general layout of the Auxiliary Dam Spillway is shown on Figures 3.8.4-37 and 3.8.4-38.

3.8.4.1.10 Auxiliary separating dike

To control and direct the flow of recirculation and to allow for a greater cooling capacity by the reservoir surface area, one separating dike is constructed in the Auxiliary Reservoir. For a description and analysis of the Auxiliary Separating Dike, see Section 2.5.6.

3.8.4.1.11 Seismic Category I underground electrical duct runs and manholes

The underground electrical conduits in the Seismic Category I duct runs in the yard are buried in trenches excavated in the ground below grade. The trenches are backfilled with suitable material to maintain the integrity of the electrical conduits during earthquakes and to provide protection from tornado missiles. The underground electrical conduits include the diesel generator main leads, which connect the diesel generator and turbine-generator in the Turbine Building, and the Seismic Category I electric cable from the intake structure to the Tank Building.

The electrical duct runs are protected against tornado missiles by either sufficient burial below grade or by covering the backfill with reinforced concrete slabs. Reinforced concrete cover slabs are provided at all road and railroad crossings. The ends of electrical duct runs are isolated from the structures and are free to move in any direction. The ends are connected to steel sleeves by elastic boots and stainless steel straps or flexible conduit with threaded fittings.

Seismic Category I manholes are provided in the plant area for routing of underground Seismic Category I electrical power and control cables. The manholes are reinforced concrete cubicles laid out individually or in multiple units and are buried in the ground; the top is six in. above grade elevation in unpaved areas and flush with the paving in paved areas. A sump pit is provided in each manhole cubicle to facilitate checking the presence of leakage in the manhole. The manholes and manhole covers are designed to resist seismic, tornado, and tornado missile loads. The manholes are founded entirely on either rock, existing soil, or compacted random fill.

To permit differential movement between manholes and electrical cables, the cables are not anchored within the manholes. The openings provided in the side walls of the manhole for the cables are covered with steel plates; the steelplates have oversized holes for free movement of electrical conduits. A general layout of Seismic Category I underground electrical duct runs and a typical detail for manholes are shown on Figures 3.8.4-23 and 3.8.4-24.

3.8.4.1.12 Structures for the Emergency Service Water System

The Emergency Service Water System (ESWS) is designed to supply cooling water as discussed in Section 9.2.1. The Seismic Category I structures of the ESWS consist of the Emergency Service Water Intake Channel, Emergency Service Water Screening Structure, Emergency Service Water and Cooling Tower Makeup Intake Structure and Channel, Emergency Service Water Discharge Structure, and the Emergency Service Water Discharge Channel. Retaining walls are provided at the Emergency Service Water Screening Structure. All Concrete structures are reinforced and founded on sound rock.

Cooling water is drawn from either the Auxiliary Reservoir or the Main Reservoir. Water drawn from the Auxiliary Reservoir is carried by a series of steel pipes from the ESW Screening Structure to the Emergency Service Water and Cooling Tower Makeup Intake Structure. Cooling water is discharged into the Auxiliary Reservoir through the Emergency Service Water Discharge Channel.

1. Emergency Service Water Intake Channel - The Emergency Service Water Intake Channel extends from the Auxiliary Reservoir to the Emergency Service Water Screening Structure. The bottom of the channel is at Elevation 238 ft., except at the intake screening structure where it slopes down to Elevation 231 ft. Channel side slopes in rock are approximately four vertical to one horizontal; a 15 ft. wide berm is cut at the interface of soil and rock. Side slopes in soil are one vertical to two horizontal. The channel bottom is 50 ft. wide at all sections except at the intake structure. The lowest water level in the channel is Elevation 246.5 ft. and the maximum velocity in the channel is less than one ft. per sec. at this level. For analysis of the channel slopes, see Section 2.5.6.

2. Emergency Service Water Screening Structure - The ESW Screening Structure, shown on Figures 3.8.4-25 through 3.8.4-27, is located at the eastern end of the ESW Intake Channel. It contains eight bays separated by reinforced concrete walls. Only two bays are used for the ESW system. Each ESW bay, 8 ft. 2 in. wide, is sized for seven ft. wide traveling screens. The maximum velocity through the traveling water screens at normal low water (Elevation 250 ft.) is 0.80 ft. per second. In addition to a traveling screen, each bay contains one coarse screen, one stop log guide, two fine screen guides, and access manholes. Three fire protection water pumps are located on the structure. The top deck is at Elevation 262 ft. and the top of the mat is at Elevation 231 ft. A valve pit containing butterfly valves and expansion joints is located at the rear of the structure. A reinforced concrete enclosure covers the deck to protect the traveling screens and valve pit from tornado missiles. A reinforced concrete skimmer wall, at the front of the intake structure, extends to Elevation 247.5 ft. and prevents ice and floating trash from entering the intake structure.

   Water is drawn from the ESW Intake Channel through the ESW Screening Structure and transported by gravity through steel pipes to the ESW and Cooling Tower Makeup Structure.

3. Emergency Service Water and Cooling Tower Makeup Intake Structure and Channel - The ESW and Cooling Tower Makeup Intake Structure is located at the northern end of the Cooling Tower Makeup Water Intake Channel. The intake structure has fourteen bays. Two bays are used for cooling tower make-up pumps and two bays are

used for ESW system. An isometric view of the incomplete structure is shown in Figure 3.8.4-41. Each ESW bay is 10 ft., 2 in. wide, sized for 8 ft. wide traveling screens. Each ESW bay contains one vertical ESW cooling pump with a design capacity of 20,000 gpm. There are two screen bays which service two cooling tower makeup pumps with a capacity of 26,000 gpm. each; each bay is sized for a 10 ft. wide traveling screen. Each screen bay also contains one coarse screen, one stop log guide, and two fine screen guides. The screen bays, containing ESW pumps, have a concrete dividing wall with an eight by ten ft. butterfly valve. The dividing wall-butterfly valve arrangement permits operation of the ESW pumps from either the Main or Auxiliary Reservoir. Access manholes, ladders, and platforms are provided into the intake pump structure between the course screen and the traveling screens for access to the butterfly valves and pump wells. Screen wash pumps are located on this structure. The top of the deck is at Elevation 262 ft. and the top of the mat is at Elevation 190 ft. Maximum velocity through the traveling screens is 0.80 ft. per second at normal water level (Elevation 220 ft.). A skimmer wall is provided across the front face of the intake to prevent floating trash and ice from entering the structure. The bottom of the skimmer wall is at Elevation 219 ft. The valve pit located at the rear of the structure contains discharge piping, butterfly valves, expansion joints, and strainers. A reinforced concrete enclosure covers the deck to protect all ESWS equipment from tornado missiles.

The Emergency Service Water and Cooling Tower Makeup Intake Channel extends from the Main Reservoir to the ESW and Cooling Tower Makeup Intake Structure and is approximately 2500 ft. long and 45 ft. wide at its invert elevation of 194.0 ft. The walls of the channel have a slope of two horizontal to one vertical in soil, one horizontal to four vertical in rock on the north side of the channel and two horizontal to one vertical in rock on the south side.

4. Emergency Service Water Discharge Channel - The ESW Discharge Channel extends from the ESW Discharge Structure to the Auxiliary Reservoir. The bottom of the channel is at Elevation 240 ft. and it is 50 ft. wide. Channel side slopes in rock are approximately four vertical to one horizontal and in soil they are one vertical to two horizontal. A berm 15 ft. wide is cut at the interface of the soil and rock. The lowest water level in the channel is Elevation 246.5 ft. and the maximum velocity at this level is one ft. per second. The Channel is sized for a maximum flow of 105,000 gpm. For an analysis of the channel slopes, see Section 2.5.6.

5. Emergency Service Water Discharge Structure - The ESW Discharge Structure, shown on Figure 3.8.4-32, is located at the eastern end of the ESW Discharge Channel. It is a reinforced concrete structure which serves as the termination point for the service water discharge piping. The discharge structure has eight bays whereas only two bays are used.

6. Retaining Wall - Reinforced concrete retaining walls, where required, are located at the end of the Cooling Tower Makeup Water Intake Channel and at the ESW Screening structure, shown on Figure 3.8.4-33. The walls are utilized to contain the earth adjacent to the concrete structures.

7. Seismic Category I Underground Pipe Lines - The underground Seismic Category I pipe lines in the yard area are buried in trenches excavated in the ground below grade.

The trenches are backfilled with suitable material to maintain the integrity of the pipe lines during earthquakes. The pipe lines are protected against tornado missiles by either sufficient burial below grade or by covering the backfill with reinforced concrete slabs. Reinforced concrete cover slabs are provided at all road and railroad crossings where six (6) feet of cover does not exist over the top of the conduit.

3.8.4.1.13 Masonry walls

Seismically designed masonry (block) walls only are used in the Containment Building and Diesel Generator Building. All masonry walls in the Reactor Auxiliary Building are seismic except in the hot shop area on Elevation 236 ft., between 'B' and 'H' lines and '43' and '45' line where seismic and non-seismic are utilized. Both seismic and non-seismic are used in the Fuel Handling Building and Waste Processing Building. These walls are utilized for shielding and equipment removal purposes or support of non-safety equipment. Typical details are shown on Figures 3.8.4-39 and 3.8.4-40.

3.8.4.1.14 Tanks

Seismic Category I tanks, as listed in Table 3.8.4-3, are housed within structures or enclosures which are designed to withstand the tornado loadings as further described in Section 3.5.1 and 3.5.2 including Tables 3.5.1-1, 3.5.1-2 and 3.5.2-1 and Fig. 3.5.1-1.

For the Seismic Category I field erected storage tanks (Boric Acid, Boron Recycle Holdup, Refueling Water Storage, Condensate Storage, and Reactor Makeup Water Storage Tank), the maximum responses obtained from the seismic analysis of the tank were increased to include the sloshing effect of the fluid on the tank. The calculation methodology for the sloshing effects was based on "Dynamic Pressure on Fluid Containers" (Chapter 6), Nuclear Reactors and Earthquakes, TID 7024. The field erected storage tanks are cylindrical atmospheric pressure tanks designed, furnished, fabricated, erected and code stamped in accordance with the ASME B & PV Code, Section III, Subsection NC or ND (Refer to Table 3.2.1-1). Allowable stresses are discussed in Section 3.9.3.1.

The Diesel Generator Fuel Oil Storage tanks are below grade reinforced concrete steel lined structures as described in Section 3.8.4.1.7 and are designed and built to the Seismic Category I requirements described in Section 3.8.4.2 to 3.8.4.6. The steel liner is Seismic Category I and is designed and built to the requirements designed in Tables 9.5.4-1 and 3.2.1-1.

The Expansion and Make-Up Tanks are subjected to the input accelerations represented by the floor response spectra, developed at the support elevations of the tanks. The tanks are designed, furnished, fabricated and stamped in accordance with the requirements of ASME B & PV Code, Section III, Subsection ND (refer to Table 3.2.1-1). Allowable stresses are discussed in Section 3.9.3.1. The tanks are of horizontal construction with a capacity of approximately 150 gallons. The Expansion Tanks' design pressure is 0 psig (atmospheric) with water level during plant normal operation at approximately centerline of the tank. The Make-Up Tanks' design pressure is 150 psig and is water-solid during plant normal operation. Due to the tank sizes and tank internal pressures, the liquid sloshing effect is not considered in the analysis of the tanks.

The Spray Additive Tank is subjected to the input accelerations represented by the floor response spectra, developed at the support elevations of the tank. The tank is designed, furnished, fabricated and stamped in accordance with the requirements of ASME B & PV Code, Section III, Subsection ND (refer to Table 3.2.1 1). Allowable stresses are discussed in Section 3.9.3.1. The tank is of horizontal construction with a capacity of approximately 7098 gallons. The tank design pressure is 15 psig. Due to the tank size and tank internal pressure, the liquid sloshing effect is not considered in the analysis of the tank.

The Diesel Generator Fuel Oil Day Tank is subjected to the input accelerations represented by the floor response spectra, developed at the support elevations of the tank. The tank is designed, furnished, fabricated and stamped in accordance with the requirements of ASME B & PV Code, Section III, Subsection ND (refer to Table 3.2.1-1). Allowable stresses are discussed in Section 3.9.3.1. The tank is of vertical construction with a capacity of approximately 3000 gallons. The tank design pressure is atmospheric. The liquid sloshing effect is considered in the analysis of the tank and was based on "Dynamic Pressure on Fluid Containers" (Chapter 6), Nuclear Reactors and Earthquakes, TID 7024. The analysis showed that the maximum sloshing frequency is below 1 Hz. The seismic input at these frequencies is small; therefore the worst case loading was assumed to occur with a completely filled tank.

For the Volume Control Tank, Boron Injection Tank, Boron Injection Surge Tank, Component Cooling System Surge Tank and Accumulator, buckling was considered in the analysis. The analyses were performed in accordance with either ASME Code NC-3300 or Section VIII, Division 2 Rules (see FSAR Tables 3.9.3-2 and 3.9.3-3). Also, Westinghouse has generically evaluated the effects of sloshing in these tanks. Since these tanks are small, sloshing has a negligible effect.

3.8.4.2 Applicable Codes, Standards, and Specifications

The applicable codes, standards, and specifications are given in Section 3.8.3.2.

3.8.4.3 Loads and Loading Combination

3.8.4.3.1 Loads

All reinforced concrete and/or steel Seismic Category I structures are designed for the following loads which are considered in the combinations defined in Section 3.8.4.3.2, and as applicable:

LOAD CATEGORIES	NOTATIONS
a) Normal Loads	
1) Dead Load	D
2) Live Load	L
3) Thermal	To
4) Operating Pipe Anchor Load	Ro
b) Severe Environmental Loads	
1) Operating Basis Earthquake	E
2) Wind	Hu
c) Extreme Environmental Loads	
1) Safe Shutdown Earthquake	E'
2) Tornado	W
d) Abnormal Loads	
1) Pressure Load	Pa
2) Accident Thermal Load	Ta

LOAD CATEGORIES	NOTATIONS
3) Accident Pipe Load	Ra
4) Spent Fuel Cask Drop Impact Load (For Fuel Handling Building and Unloading Bay only)	F
5) Equipment Accident Load	Q

The applications of the above loads are dependent on the specific structure.

The above loads are described as follows:

a) Dead Load (D) - Dead load consists of the weight of the structure, partition walls, and miscellaneous items, such as permanent equipment, crane dead weights, HVAC ducts, electrical cable trays, piping, and roofing. The lateral water pressure due to hydrostatic effects on concrete structures is considered as a dead load.

Groundwater is at Elevation 251 ft. For conservatism, the uplift forces (buoyancy) created by the displacement of groundwater, assumed to be at Elevation 260 ft., are accounted for in condition 5 of Section 3.8.5.5. Groundwater is considered as dead load in the design. Minimum groundwater level is assumed to be at Elevation 204.4 ft. and is accounted for in the design.

Specific weights for dead load calculations are as follows:

1) Concrete: 143 lb/cu ft. maximum and 137 lb/cu ft. minimum;

140 lb/cu ft. is used for design.

2) Reinforcing steel: 489 lb/cu ft.

3) Structural steel: 489 lb/cu ft.

4) Water 62.5 lb/cu ft.

5) HVAC ducts, electrical cable trays, and piping = 50 lb/sq ft. of floor area

6) Roofing = 5 lb/sq ft. of roof area

b) Live Load (L)

Live loads are included to assure that structures are sufficiently strong during normal operation to support random temporary load conditions for maintenance, and to assure structural adequacy for normal or construction loading.

1) All floors and roofs are designed for a 10,000 lb. concentrated load at any one point (2 ft. x 2 ft.) in addition to the designated unit live loads: 30 lb/sq. ft. for roofs, and 100 lb/sq. ft. for all suspended floors and foundation mats. Precipitation loads on roofs are furnished in Subsection 2.3.1.2.8.

2) Earth Pressures (considered as live loads)

a) Soil Properties

(1) Dry unit weight γ_d = 115 pcf

(2) Saturated unit weight γ_s = 130 pcf

(3) Submerged unit weight γ_b = 67.6 pcf

(4) Angle of internal friction φ = 20°

(5) Cohesion C = 400 psf

(6) Coefficient for at rest pressure (See Reference 3.8.4-1) K = 0.7

b) Backfill Material - Normal soil pressure against the structure is considered as "at rest" pressure. The design also considers the loads caused by the pressure of the earth against the structure during an earthquake.

c) Foundation Design - The live load for foundation design is the total reduced live load occurring in the columns and bearing walls immediately above the foundation. The total reductions of live load for foundation design are as follows: (1) no reductions of live load for roofs and floors immediately above foundation mats, and (2) live loads for other floor slabs are reduced in accordance with the Uniform Building Code - 1976.

d) Live Load Present During a Postulated Event - In load combinations affected by a postulated high energy pipe break, both cases of live loads that either have a full value possibly present during a pipe rupture event, or are completely absent, are checked.

e) Movable Equipment Loads are also considered as live load.

c) Wind (Hu) - Wind loading is based on a 179 mph wind, with gust factors included, at 30 ft. above ground level. Distribution of the wind load is made in accordance with ASCE Paper No. 3269 "Wind Forces on Structures," Vol. 126, Part 2, 1961.

The water wave generated from wind is calculated by the methods indicated in "Engineer Technical Letter # 1100-2-8, Dept. of the Army" dated August 1,

1966 (see Section 2.4.3.6). Static and dynamic effects of water waves have been considered in the design of reservoir structures.

d) Tornado (W)

1) Tornado loading for the Seismic Category I structures are based on the following characteristics:

a) W_w - External wind forces resulting from a tornado funnel with a horizontal peripheral tangential velocity of 290 mph and a horizontal translational velocity of 70 mph. Conservatively, this is taken as a 360 mph wind applied uniformly with height and width. The loading distribution around the structure is in accordance with ASCE Paper No. 3269, "Wind Forces on Structures," Vol. 126, Part 2, 1961.

b) W_p - Decrease in atmospheric pressure of 3 psi in 1.5 seconds time at a rate of 2 psi/sec.

c) M - Tornado missile impact force.

2) For determining the tornado load that is used in load combinations for structures, or portions thereof, the most adverse of the combinations below are used, as appropriate:

(i) $W = W_w$

(ii) $W = W_p$

(iii) $W = M$

(iv) $W = W_w + 0.5\ W_p$

(v) $W = W_w + M$

(vi) $W = W_w + 0.5\ W_p + M$

3) The structures are designed to withstand without perforation the impact of high velocity external missiles that might occur during the passage of a tornado. To ascertain the integrity of the structure against tornado missiles, the missile spectra published in Standard Review Plan 3.5.1.4 are used for design (see Section 3.5.1.4).

e) Operating Pipe Anchor Load (Ro) - The actual pipe anchor loads are used to design the structures.

f) Accident Pipe Loads (Ra) - The actual accident pipe loads are used to design the structures.

g) Spent Fuel Cask Drop Impact Load (F) (For Fuel-Handling Building and Unloading Bay only)

The Fuel-Handling Building and unloading bay structure are designed to maintain the integrity of the facilities in the event of a postulated spent fuel cask drop accident within the areas traversed by the spent fuel cask handling crane.

h) Equipment Accident Load (Q) - In addition, pipe or equipment accident loads, Q = Yr + Yj + Ym, are considered in the design, where:

Yr -represents the equivalent static load on a structure generated by the reaction of a broken high-energy pipe during a postulated break; this includes an appropriate dynamic factor to account for the dynamic nature of the load.

Yj -represents a jet impingement equivalent static load on a structure generated by a postulated break; this includes an appropriate dynamic factor to account for the dynamic nature of the load.

Ym -represents a missile impact equivalent static load on a structure generated by or during a postulated break; this includes an appropriate dynamic factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Yr, Yj, and Ym, elasto-plastic behavior is assumed with appropriate ductility ratios, as long as excessive deflections do not result in a loss of function.

i) Earthquake Loads (E, E')

Earthquake loads are computed using the following:

1) Operating Basis Earthquake (E), horizontal ground acceleration 0.075 g.

2) Safe Shutdown Earthquake (E'), horizontal ground acceleration 0.15g.

3) The vertical design response spectra values are 2/3 those of the horizontal design response spectra for frequencies less than 0.25; for frequencies higher than 3.5, they are the same, while the ratio varies between 2/3 and 1 for frequencies between 0.25 and 3.5. For frequencies higher than 33 cps the design response spectra follow the maximum horizontal ground acceleration line.

Static loads and deflections resulting from application of the above accelerations are computed by utilizing a dynamic analysis computer program.

Seismic loads due to hydrodynamic effects on water contained in large containers are computed in accordance with Chapter 6 "Dynamic Pressure on Fluid Containers" of "Nuclear Reactors and Earthquakes," TID 7024.

j) Thermal Load (To) - The load induced by normal gradients across the walls between the building interior and the ambient external environment; the conditions are considered as follows:

1) Summer (For all Seismic Category I buildings, except Containment, unless otherwise noted)

(a) Interior sustained air temperature	104 F
Except:	
-Room temperature up to 19 ft. above operating floor for FHB	89.2 F
-Air temperature* in spent fuel pool pump room of FHB	115.5 F
-Maximum temperature* in some localized spaces below EL. 286' of FHB	116.6 F
-Air temperature in the diesel generator room	120 F
*For design/analysis, however actual temperature is as described in Table 9.4.0-1	
(b) Exterior sustained concrete temperature (air)	90 F
(c) Exterior sustained concrete temperature (soil)	70 F
(d) Spent fuel pool and transfer canal water (for FHB only)	150 F

2) Winter (For all Seismic Category I buildings, except Containment, unless otherwise noted)

(a) Interior sustained air temperature	70 F
(b) Exterior sustained concrete temperature (air)	16 F
(c) Exterior sustained concrete temperature (soil)	45 F
(d) Spent fuel pool and transfer canal water (for FHB only)	150 F

Thermal loads for the foundation mats are based on a constant (summer and winter) temperature of 60 F for the material underlying the mats.

In determining thermal loads, shrinkage effects are taken into account; the shrinkage is considered equivalent to a decrease in temperature (see ACI Publication SP-27).

For all cases, the "as constructed" concrete temperature is assumed to be 60 F. In all cases, the conditions assumed are considered to be of sufficient duration to result in a straight line temperature gradient.

k) Accident Thermal Load (Ta) - The load induced by gradients across the concrete members under thermal conditions generated by a postulated high energy pipe break; this includes the thermal load, To, during normal operating conditions. The thermal conditions generated during a postulated pipe break are used as applicable.

l) Pressure Load (Pa) - The pressure load consists of an equivalent static load within or across a compartment or building generated by a postulated high energy pipe break; this includes an appropriate dynamic factor to account for the dynamic nature of the load.

3.8.4.3.2 Load Combinations

The loads (hereafter referred to as factored loads) utilized to determine the required limiting capacity of any structural element of the reinforced concrete and/or steel Seismic Category I Structures are computed as follows:

Note: C = required load capacity of the structure.

a) Normal Operating

$C = 1.4D + 1.7L$

$C = (0.75)(1.4D + 1.7L + 1.7T_o + 1.7R_o)$

b) Hurricane

$C = 1.4D + 1.7L + 1.7Hu$

$C = (0.75)(1.4D + 1.7L + 1.7Hu + 1.7T_o + 1.7R_o)$

$C = 1.2D + 1.7Hu$

c) Operating Basis Earthquake

$C = 1.4D + 1.7L + 1.9E$

$C = (0.75)(1.4D + 1.7L + 1.9E + 1.7T_o + 1.7R_o)$

$C = 1.2D + 1.9E$

d) Safe Shutdown Earthquake

$C = D + L + T_o + R_o + E'$

e) Tornado

$C = D + L + T_o + R_o + W$

f) Pipe or Equipment Accident

$C = D + L + T_a + R_a + 1.5P_a$

g) Pipe or Equipment Accident Plus Operating Basis Earthquake

$C = D + L + T_a + R_a + 1.25P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25E$

h) Pipe or Equipment Accident Plus Safe Shutdown Earthquake

$C = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0E'$

i) Spent Fuel Cask Drop Accident Plus SSE

$C = D + L + T_o + R_o + E' + F$

In loading combinations (f), (g), and (h), the maximum values of P_a, T_a, R_a, and Q (where Q = Yr + Yj + Ym), including an appropriate dynamic factor, are used.

Load combinations (e), (g), and (h) are satisfied first without missile load in (e) and without Q in (g) and (h). When considering these concentrated loads, local section strength capacities may be exceeded provided there is not a loss of function of any safety related system.

3.8.4.3.3 Load Factors

a) For the factored load cases defined in Section 3.8.4.3.2, the following reduction factors "φ" are used:

1) φ = 0.90 for flexure in concrete

2) φ = 0.85 for axial tension at lapped reinforcing bars

3) φ = 0.85 for diagonal tension, bond, and anchorage in concrete

4) φ = 0.75 for spirally reinforced concrete compression members

5) φ = 0.70 for tied compression members

6) φ = 0.90 for fabricated structural steel

b) Allowable shear of 0.1 $f'c$, is used in designing keys under safe shutdown earthquake (SSE) conditions. For other conditions, the ACI-318-71 code is used.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Assumptions and Boundary Conditions

The basic assumptions and boundary conditions used for design of reinforced concrete Seismic Category I structures are as follows:

a) All Seismic Category I buildings are designed as reinforced concrete box system structures with beam and girder floors spanning between interior columns and interior and exterior cast-in-place concrete bearing walls. The floor slabs are designed to carry vertical loads in addition to acting as horizontal diaphragms to transfer loads to shear walls.

b) All Seismic Category I building walls and columns are doweled to the foundation mats.

There are isolation gaps of sufficient width between all adjacent buildings at all levels to prevent potential pounding of the structures during earthquakes.

3.8.4.4.2 Design and Analysis Procedures

a) All Seismic Category I buildings are analyzed statically, based on the loading combinations described in Section 3.8.4.3. The equivalent static loads which resulted from the application of accelerations or displacements at various levels during the dynamic analysis are utilized.

The stresses of each individual loading condition are combined in accordance with the equations of Section 3.8.4.3, and used for proportioning all components of structures. The design of structural members is in accordance with the ultimate strength design provisions of the ACI-318-71 Code.

Under seismic loading, no plastic analysis is considered. Local yielding of structures is considered permissible due to LOCA or missile forces, provided there is no general failure.

b) For proportioning reinforced concrete structural elements to resist earthquake induced forces, the floor slabs, columns, beams, girders and shear walls of the structures are designed and proportioned in accordance with ACI-318-71. In addition, they also are designed to satisfy the following requirements:

1) To assure a structural failure mode by tensile yielding of reinforcement rather than a compression failure of concrete in slabs, beams, and girders, the maximum reinforcing steel percentage p, or (p p') when compression reinforcement is used, does not exceed 0.5 P_b, where P_b is the steel ratio which would produce a balanced condition for a section under flexure without axial load.

2) To assure a structural failure in the flexure mode rather than a shear failure of concrete in slabs, beams, and girders, the following steps are taken in the design:

a) The ultimate bending capacities of the members were evaluated in accordance with the ACI-318-71 building code.

b) The shear capability of the section is assured to be at least 10 percent in excess of the shear resulting from a uniform load on the member. Therefore, the ultimate moment will occur at the critical section of the member in bending.

c) The positive moment capacity at the face of supports is at least 50 percent of the negative moment capacity at that location.

d) A minimum of 25 percent of the larger amount of negative reinforcing steel required at either end of a beam continues throughout the length of the beam.

e) When pairs of U stirrups are used for web reinforcement, the legs of each U bar extend the full depth of the beam.

3) Column spirals or hoops are anchored inside of the column core.

4) A minimum percentage of web reinforcement is provided in the beams and girders in accordance with Section 11.6 of the ACI-318-71 code, unless the calculation shows that the shear stress plus the effect of torsion is less than one half of the code allowable stress without web reinforcing.

c) Missile Analysis - The exterior portions of the structures are designed to withstand, without perforation, the impact of high velocity external missiles that might occur during the passage of a tornado. The depth of penetration is calculated in accordance with the National Defense Research Committee (NDRC) formula. The exterior portions of the structure are also designed by use of an impactive analysis for the missiles. The missile load is assumed to occur simultaneously with tornado loading. For design criteria and procedures for missile analysis, see Section 3.3 and 3.5.

d) Dynamic Analysis for Seismic Loading - All Seismic Category I buildings are founded upon reinforced concrete mats resting on suitable rock or on concrete fill founded on suitable rock. The structural dynamic analysis of the Seismic Category I buildings is based on use of the response spectra developed for 0.075g for the operating basis earthquake and 0.15g for the safe shutdown earthquake. The procedures and modeling adopted for the dynamic analysis are in accordance with the containment building dynamic analysis for seismic loading using Ebasco's computer program "Dynamic 2037". The vertical dynamic analysis determines the dynamic response of the foundation mat, walls, and floors (where the major equipment is supported). Torsional effects due to asymmetry of the structural components are also investigated in the design.

e) Main Dam and Auxiliary Dam and Separating Dike - For design of dams and dike, see Section 2.5.6.

f) Computer Programs Utilized for Structural and Seismic Analyses - For computer programs utilized for structural and seismic analysis, see Appendix 3.8B.

3.8.4.4.3 Mechanism for Load Transfer

Load transfers from reinforced concrete and/or steel Seismic Category I structures to their foundation mats are discussed in Section 3.8.5.4.1.

3.8.4.5 Structural Acceptance Criteria

The stability of the Main Dam, Auxiliary Dam, and Auxiliary Separating Dike is evaluated by using the results of supplementary field explorations, laboratory testing, and analytical studies. The field explorations, laboratory testing, and analytical studies are described in Section 2.5.6.5.

The structural acceptance criteria for the reinforced concrete and/or steel Seismic Category I structures are given in Section 3.8.3.5.

For all load conditions, the calculated design loads are within the ultimate capacity of the structural members.

3.8.4.6 Materials, Quality Control, and Specific Construction Technique

The basic materials used for construction of the Seismic Category I structures, described in Section 3.8.4.1, are concrete, reinforcing steel, structural steel, backfill, and rock. The material specifications, testing requirements, and quality control measures specified in this section and in Section 3.8.3.6 form a part of the overall Engineering and Construction Quality Assurance Program which was approved by the NRC during the Construction Permit review.

3.8.4.6.1 Concrete construction

Concrete construction as stated in Sections 3.8.3.6.1.1 and 3.8.3.6.2.1 except that structural concrete is specified to have minimums of 4000 psi compressive strength (Class AA concrete) and 137 pcf air dry weight 28 days after placing.

3.8.4.6.2 Reinforcing steel

See Sections 3.8.3.6.1.1(g) and 3.8.3.6.2.1.

3.8.4.6.3 Structural steel

See Sections 3.8.3.6.1.2 and 3.8.3.6.2.2.

3.8.4.6.4 Earth and rock

The fill material properties used for construction of the Main Dam, Auxiliary Dam, Auxiliary Separating Dike, and Seismic Category I channels are described in Section 2.5.6.4.2.

3.8.4.7 Testing and In-Service Surveillance Requirements

There is no planned systematic testing for the Seismic Category I structures after the plant has been placed in operation. The structural steel framing and connections will be generally accessible for visual inspection. Seismic Category I structures are periodically inspected in accordance with Reg. Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear

Power Plants." Performance monitoring of the main and auxiliary dams is discussed in Section 2.5.6.8.

3.8.4.8 Masonry Walls

Seismically designed concrete masonry (block) walls only are used in the Diesel Generator Building, and Containment and Reactor Auxiliary Building which are Seismic Category I buildings. The Fuel Handling Building and Waste Processing Building, which are also Seismic Category I buildings, contain seismically designed and non-seismically designed masonry (block) walls.

Loads and load combinations, as applicable, are shown on Tables 3.8.4-1 and 3.8.4-2.

The following codes are used for the analysis and design of masonry block walls and any associated steel framing:

ACI 531-79	American Concrete Institute "Building Code Requirements for Concrete Masonry Structures"
UBC-79	Uniform Building Code, by International Conference of Building Officials
AISC – Seventh Edition	American Institute of Steel Construction "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings"
ACI 318-71	American Concrete Institute "Building Code Requirements for Reinforced Concrete"

Although the masonry block walls have been designed in accordance with the loads and load combinations indicated in Tables 3.8.4-1 and 3.8.4-2, there are no block walls designated as Seismic Category I. Standard Review Plan Sections 3.7 and 3.8 are complied with to the extent applicable. Standard Review Plan Section 3.5 was not used as no credit is taken for those walls for missile protection.

Dynamic forces in masonry walls due to earthquake were calculated using dynamic analysis. Wall frequencies were calculated and input motion was obtained from the appropriate broadened floor response spectra curves.

No safety related piping or equipment is attached to masonry walls. Non-safety equipment may be attached to hollow block walls. Non-safety equipment attachment loads shall not exceed the combination of 25 pounds per square foot uniform load on 15 percent of the gross area and 200 pounds of concentrated loads located at the midspan of the vertical reinforcing units spaced at not more than four feet on center unless authorized by the engineer. No attachment loads are permitted for seismically designed shielding block walls.

Typical reinforcement details are shown on Figures 3.8.4-39 and 3.8.4-40.

The design criteria utilized in the design of masonry walls located in Seismic Category I structures complies with the NRC Structural Engineering Branch Criteria for Safety Related Masonry Wall Evaluation, dated July 1981. Class I masonry walls have been addressed in

CP&L's June 30, 1982, response to Enclosure 5, Item No. 17, of NRC Acceptance Review of OL Application, dated November 25, 1981.

3.8.4.9 Design Changes to Plant Configuration As a Result of Unit 2, 3 and 4 Cancellation

Changes to seismic Category I structures are being made to accommodate a one unit plant. The changes are described below.

a) Fuel Handling Building - In order to retain the seismic characteristics of the building and to maintain the structural integrity of the building, all the major components of the building, namely, foundation mat, floor slabs and all shear and load bearing walls have been constructed as designed for four units. Only the internal non-load bearing walls and some penetrations and openings in the slabs and walls have been modified in the area which was reserved for Units 2, 3 and 4 equipment. See Figures 3.8.4-7 thru 3.8.4-12.

   Since Units 2, 3 and 4 Reactor Auxiliary Buildings and Containment Buildings have been deleted, the Fuel Handling Building has been isolated from the plant grade fill by a retaining wall on the west side and a series of retaining walls on the east side where required (see Figure 3.8.4-45). The building stability and structural design have been reviewed for additional wind and tornado loads to satisfy the design criteria.

   The retaining walls west and east of the Fuel Handling Building have been seismically designed in accordance with Regulatory Guide 1.29 Positions C.2 and C.4. For details of the retaining wall see Figures 3.8.4-42 and 3.8.4-45.

   The retaining wall west of the Fuel Handling Building has been physically separated from the building by a gap of three (3) feet along the length of the wall and a gap of approximately three (3) inches at the north end. The retaining wall consists of two rows of reinforced concrete pipes erected one over the other and a reinforced concrete wall on top of the pipes. The pipes are filled with concrete and are held back by tie rods and deadmen. Other design and construction features include the following:

   The deadmen for the retaining wall are designed in accordance with the criteria given in Reference 3.8.4-2.

   The tie rods are protected against corrosion by coating the tie rods with Epoxy and also by electrically grounding all reinforcing steel bars and tie rods.

   To avoid erosion of soil near deadmen, the plant grade is protected by turf and no storm drain or other pressure pipes are provided parallel to and within 50 feet in front of the deadmen.

   The soil backfill west of the retaining wall is modified random fill in accordance with Ebasco Specification CAR-SH-CH-8 (see FSAR Appendix 2.51). Possible variations in the properties of the backfill materials, based on laboratory and field test results, were also considered in design (see Sections 2.5.4.5.3.1.3 and 2.5.6.5.5.3).

   The retaining wall is supported mostly on modified random fill with average thickness of 22 feet over the top of the rock. Since the modified random fill under the foundation of the retaining wall was placed ahead of construction of the retaining wall and the area was used as a platform for construction of the FHB, the settlement is expected to be minimal.

Permanent monuments have been installed on the retaining wall and the top line of deadmen for monitoring settlement and lateral movement of the wall (see Section 2.5.6.8). The factors of safety for the tie rods and the deadmen meet the acceptance criteria of Section 3.8.5.5.

There is no safety-related equipment west of "N" line wall, and east of "L" line wall north of Column Line 45. All exterior openings and penetrations in "N" line wall are evaluated for tornado and tornado missile and adequate protection is provided.

b) Containment Building - The Unit 2 containment building mat has been constructed. The mat is used to provide support for the retaining wall east of the Fuel Handling Building. The mat has been stabilized against flotation by backfill which is topped at El. 236 feet by a concrete slab.

c) Reactor Auxiliary Building 2 - The Reactor Auxiliary Building 2 stepped foundation mat and the Emergency Service Water Pipe Tunnel have been constructed. The mat and tunnel have been stabilized against flotation by backfill to various elevations as shown in Figure 3.8.4-45. The backfill is topped by a concrete slab. Where required, exterior and interior walls of Reactor Auxiliary Building 2 have been constructed to retain the plant grade backfill and the stepped backfill on top of the mat inside the building. A seismic Category I Service Water Pipe Tunnel Penthouse has been constructed on top of the pipe tunnel termination at Tank Building 2 to protect and house the piping transition from the tunnel to the tank building. The penthouse is shown in Figure 3.8.4-45.

An access to Reactor Auxiliary Building Common from the Reactor Auxiliary Building 2 area has been provided at El. 236 feet by the construction of an Access Bay (shown by Figure 3.8.4-45).

The Access Bay structure and the retaining walls have been seismically designed in accordance with Regulatory Guide 1.29 Positions C2 and C4. Seismic analysis of the as-built Reactor Auxiliary Building 2 has been performed to obtain seismic response spectra for the as-built structure to verify the design of safety-related piping and system within the building.

d) Tank Building 2 - The building has been constructed to El. 261 feet to protect and house the Emergency Service Water piping. Seismic analysis of the as-built Tank Building 2 has been performed to obtain seismic response spectra for the as-built structure to verify the design of safety-related piping and systems within the building.

e) Waste Processing Building - North-west portion of the Waste Processing Building which was previously isolated by Reactor Auxiliary Building 4 and Tank Building 4 is now subject to plant grade fill. The stability of the building for additional lateral earth pressure and hydrostatic pressure has been reviewed to confirm that it satisfies the design criteria.

All openings in the Waste Processing Building against the plant grade fill have been closed by concrete plugs for the full thickness of the walls.

f) Diesel Fuel Oil Storage Tank Building - The entire building has been constructed as designed except that the two west tanks reserved for Units 3 and 4 have not been lined with steel. These two tanks will not be used for storage of diesel fuel oil. The design of building with these two tanks empty has been reviewed to assure that the design criteria has been satisfied. The dynamic analysis model shown in Figure 3.7.2-16 is also revised.

g) Emergency Service Water Discharge Structure - The structure has been constructed as designed. Only two bays are used for one unit. The pipe penetrations in the east wall for the other bays have been closed off. See Figure 3.8.4-32 for details of the Emergency Service Water Discharge Structure.

h) Emergency Service Water Screening Structure - This structure has been constructed as designed. Only three bays are used for one unit. The pipe penetrations against yard fill in the other bays have been closed off.

See Figure 3.8.4-25 for details of the Emergency Service Water Screening Structure.

i) Emergency Service Water and Cooling Tower Makeup Intake Structure - the eastern half of this intake structure has been constructed as designed for the two units, whereas the other half has been terminated in general at El 223 feet. The incomplete bays have been capped with a reinforced concrete deck. An isometric view of the structure as constructed is shown in Figure 3.8.4-41.

The intake structure as shown in Figures 3.8.4-28 thru 3.8.4-31 has been reanalyzed to determine the seismic response and to develop response spectra curves. The revised dynamic analysis model is shown in Figure 3.7.2-14. The stability analysis and structural design have been checked to satisfy the design criteria.

The designs of safety-related equipment and systems inside the intake structure are reviewed against the new floor response curves and modified as required to satisfy the design criteria.

3.8.5 FOUNDATIONS

3.8.5.1 Description of the Foundations

All Seismic Category I building structural foundations are reinforced concrete mats, founded on suitable rock or concrete fill, as described in the following paragraphs. The foundation rock provides adequate support for the structures under static and dynamic loading conditions. The general layout of the Seismic Category I building foundations at the plant site is shown on Figure 3.8.5-1.

A concrete seal mat, at least 4 in. thick, is founded on suitable rock and provides a level surface for a waterproofing membrane, or for reinforced mat construction where no membrane is required. A four in. thick concrete work slab overlays the waterproofing membrane and underlies the foundation mats.

Generally, sufficient gaps are provided at the foundation mat level between buildings in order to prevent transfer of horizontal forces during earthquakes. However, the sections of the containment building mat and the reactor auxiliary building mat at Elevation 190 ft. are placed against each other in order to prevent movement of the vertical cantilevered leg of the containment building mat.

A concrete seal mat is also provided for all Seismic Category I building foundations.

Exterior foundation walls of Seismic Category I structures, below grade (except the intake and discharge structures) and exposed to groundwater, are waterproofed with a waterproofing membrane. Membranes are protected from damage during backfilling operations by a protective covering. The membranes at the top of exterior walls are terminated in reglets in the wall, 12 in. below plant grade. At the bottom of the wall they either extend 24 in. across the horizontal top surface of the mat or 12 in. below the top of the mat. They are also terminated in reglets in the mat. Laps and splices of all membranes are completely sealed in order to prevent leakage.

Horizontal and vertical construction joints in the exterior foundation walls exposed to groundwater are provided with a single layer of waterstop to grade floor Elevation 261 ft., except for the north-west corner foundation walls of the Waste Processing Building. Separation gaps in the foundation mats between adjacent buildings are waterstopped with a double layer of six in. waterstops.

Nine in. vertical waterstops across the two in. separation gaps at exposed ends of the gaps are provided to seal the leakage of groundwater from outside the building. These vertical waterstops extend from Elevation 262 ft. down to the foundation mat, and connect to the double horizontal six in. waterstops in the mat between adjacent mats to form a U-shape water barrier.

The effects of plant floods on the building foundations are discussed in Section 2.4.10.

The general layout of waterstops and waterproofing membranes for the foundation mats and walls is shown on Figure 3.8.5-2.

The following paragraphs describe the foundations for each Seismic Category I building.

3.8.5.1.1 Containment Building

The foundation mat is a reinforced concrete circular flat slab resting on suitable rock. The thickness of the mat is 12 ft; it has a continuous shear key 2.5 ft. deep by 6 ft. wide along the edge of the mat. There are two depressed areas used as valve chambers. The central portion of the foundation mat is depressed to form the reactor vessel recess. For a description of the interior structure foundation mat, see Section 3.8.3.

The general layout of the containment building foundation mat is shown on Figures 3.8.5-3 and 3.8.5-4.

3.8.5.1.2 Reactor Auxiliary Building

The RAB consist of two independent structures identified as RAB 1 and RAB common.

The RAB foundations are reinforced concrete mats founded on suitable rock. There are three different levels of foundation mats for the RAB 1: Elevation 190 ft., Elevation 216 ft., and Elevation 236 ft. The mats are 10 ft. thick and are placed directly against the sidewalls of the rock excavation.

The RAB-common has a separate foundation from RAB 1. The top of this foundation mat is at Elevation 236 ft., except for the pipe tunnel area, which is at Elevation 216 ft. Sufficient separation gaps are provided between foundation mats.

The general layout of the RAB foundation mat is shown on Figure 3.8.5-5.

3.8.5.1.3 Fuel Handling Building

The foundation for the Fuel Handling Building (FHB) is a 10 ft. thick reinforced concrete mat founded on suitable rock. The foundation is on two levels; the top of the central portion of the mat is at Elevation 236 ft., while the top of both ends of the building mat are at Elevation 216 ft. There are two half-circular sections at each side of the foundation, with shear keys underneath the circular portions. Heavily reinforced concrete walls are provided at the mat level in order to support the fuel pools and superstructure.

The northern-most section of the FHB is designated as the unloading area. It provides access for railroad cars for shipping spent and new fuels. It is a narrow building 95 ft. long, 50 ft. wide, and 105 ft. high. The top of the mat is at Elevation 237 ft. The substructure is filled with sand in order to provide sufficient weight at the lower portion of the building for stability.

The general layout of the FHB foundation mat is shown on Figure 3.8.5-6.

3.8.5.1.4 Waste Processing Building

The foundation for the Waste Processing Building (WPB) is a rectangular reinforced concrete mat, 349 ft. long by 200 ft. wide, with a uniform thickness of 10 ft.

There are two special provisions to satisfy the requirements for building stability to prevent overturning or sliding.

a) The mat is extended at the east, south, and west sides of the building and counterforts are provided in continuation of each structural shear wall.

b) Due to the arrangement of equipment in this building and in adjacent structures, the north side of the building requires additional weight for stability; therefore, an 11 ft. thick reinforced concrete block is added to the underside of the structural foundation mat.

The general layout of the WPB foundation mat is shown on Figure 3.8.5-7.

3.8.5.1.5 Diesel Generator Building

The DGB foundation is a reinforced concrete mat which is supported on concrete fill in order to transfer the foundation loads to suitable rock. The building foundation mat is 155 ft. long by 123 ft. wide by 6 ft. thick. The top of the foundation mat with sand fill on top of it is 3.5 ft. below the grade floor Elevation 261 ft. The electrical main leads and pipe lines are within the sand fill.

The diesel generator pedestals are directly supported on the foundation mat.

The general layout of the DGB foundation mat is shown on Figure 3.8.4-20.

3.8.5.1.6 Tank Building

The tank building foundation is a reinforced concrete mat directly supported on suitable rock. The foundation is a rectangular mat, 142.33 ft. long, 63 ft. wide, and eight ft. thick. The top of the mat is at Elevation 236 ft.

The general layout of the tank building foundation is shown on Figure 3.8.4-21.

3.8.5.1.7 Diesel Fuel Oil Storage Tank Building

The diesel fuel oil storage tank building foundation is a reinforced concrete mat, which is founded on suitable rock. The foundation is 94 ft. long, 86 ft. wide, and 3.25 ft. thick. The foundation supports four reinforced concrete diesel oil compartments and a pump gallery. Only the east compartments are used, and are lined with steel to prevent leakage. The top of the mat is at Elevation 242.25 ft.

The general layout of the diesel fuel oil storage tank building foundation is shown on Figure 3.8.4-22.

3.8.5.1.8 Main Dam and Spillway

For a description of the Main Dam and Spillway, see Section 2.5.6.

3.8.5.1.9 Auxiliary Dam and Spillway

For a description of the Auxiliary Dam and Spillway, see Section 2.5.6.

3.8.5.1.10 Auxiliary Separating Dike

For a description of the Auxiliary Separating Dike, see Section 2.5.6.

3.8.5.1.11 Seismic Category I Underground Electrical Duct Runs and Manholes

The Seismic Category I duct runs and manholes are founded entirely on rock or compacted random fill. The duct run trenches and the areas surrounding the manholes are backfilled with suitable materials.

3.8.5.1.12 Emergency Service Water System

All foundations for the Emergency Service Water System are constructed of reinforced concrete and founded on suitable rock.

The general layout of the emergency service water system structures and foundations is shown on Figures 3.8.4-25 through 3.8.4-33.

3.8.5.2 Applicable Codes, Standards, and Specifications

The pertinent codes, standards, specifications, NRC regulations, and Regulatory Guides governing the design, construction, fabrication, inspection, testing, and material properties of the foundations are referenced in the following sections:

a) Containment Structure - Section 3.8.1.2

b) Containment Internal Structures - Section 3.8.3.2

c) Other Category I Structures - Section 3.8.4.2

3.8.5.3 Loads and Loading Combinations

The loads and loading combinations considered in the design of the foundations are described in the following sections:

a) Containment Structure Section 3.8.1.3

b) Containment Internal Structures Sections 3.8.3.3

c) Other Category I Structures Sections 3.8.4.3

3.8.5.4 Design and Analysis Procedures

The design and analysis procedures used for foundation mats are described in the following sections:

a) Containment Structure -Section 3.8.1.4

b) Containment Internal Structures -Section 3.8.3.4

c) Other Category I Structures -Sections 3.8.4.4 and 3.8.5.4.1

3.8.5.4.1 Design and Analysis Procedures for Seismic Category I Structures

The Seismic Category I foundations are designed and analyzed as a flat slab, restrained by exterior and interior structural shear walls, resting on an elastic foundation.

The foundation parameters for the analysis of the soil-structure interaction are in Sections 2.5.1 through 2.5.6.

Evaluation of the foundation soil pressure associated with dynamic loads involves an iterative solution. The contact soil pressure at the base mat has been considered as an elastic subgrade reaction. The foundation pressures of the base mat on the foundation media are distributed so as to be compatible with the deflection of the substructure and the foundation deformation.

The base mat is analyzed for these forces by using a finite element model of a grid beam on an elastic foundation. The lower section of the shear walls adjoining the base mat is used in the model to account for the stiffness of the superstructure. The base mat is subdivided into a number of fully constrained or continuous beams, depending upon the arrangement of the shear walls and columns. The edges of the panel along the walls are considered to have constrained boundary conditions. Forces due to loads acting on the upper portions of the superstructure not included in the model produce statically equivalent forces and moments at the cut section of the substructure. These equivalent forces are applied as boundary forces on the model.

The interface between the foundation and superstructure satisfies the equilibrium between loads and reactions, compatibility of strains, and boundary conditions.

The loads of the superstructure, equipment, and imposed forces are transferred to the foundation mat through the reactions of the structural system. The further transfer of loads from the foundation mat to the supporting foundation media is achieved by direct bearing, surface friction, and lateral passive resistance.

3.8.5.5 Structural Acceptance Criteria

The structural acceptance criteria relating to stresses, strains, and deformations of foundation mats of buildings are described in the following sections:

a) Containment Structure - Section 3.8.1.5

b) Containment Internal Structures - Section 3.8.3.5

c) Other Category I Structures - Section 3.8.4.5

The load combinations and minimum safety factors for overturning, sliding, and flotation stability are as follows:

Load Combination	Overturning	Sliding	Flotation
1. D + S' + E	1.50	1.50	Not Applicable
2. D + S + Hu	1.50	1.50	Not Applicable
3. D + S' + E'	1.10	1.10	Not Applicable
4. D + S + W	1.10	1.10	Not Applicable
5. D + B'	Not Applicable	Not Applicable	1.10

in which: S' = Soil pressure against the structures during an earthquake (Reference 3.8.5-1).

S = Normal soil pressure against the structures (at rest pressure); (See Section 3.8.4.3.1)

D = Dead load

B' = Buoyancy, (the bouyant force elevation is at 260 ft.)

E = OBE

E' = SSE

W = Tornado

Hu = Wind

NOTE: In load combinations 1 to 4, the hydrostatic buoyant loads, created by the displacement of groundwater, are incorporated in the dead load D. In load combination 5, B' is the buoyant force of the design basis flood.

The calculation of the factors of safety for the building structure is as follows:

$$\text{Factor of Safety Against Overturning} = \frac{\text{Resisting Moment}}{\text{Moments Causing Failure}}$$

$$\text{Sliding} = \frac{\text{Forces Opposing Sliding}}{\text{Forces Causing Sliding}}$$

$$\text{Flotation} = \frac{\text{Force Opposing Flotation}}{\text{Forces Causing Flotation}}$$

The procedure to assure the stability of the Seismic Category I structures against overturning due to the combination of three directional earthquake effects used for conservatism, one horizontal component of $\pm$ 1.0 Fx, one horizontal component of $\pm$ 1.0 Fy, and a vertical component of $\pm$ 0.4 Fz.

The above results are also verified by utilizing the results obtained from the statistically independent components for simultaneous application.

3.8.5.6 Materials Quality Control and Special Construction Techniques

Details of applicable material specifications, quality control provisions, and special construction techniques for Seismic Category I concrete foundations are in Sections 3.8.3.6 and 3.8.4.6. The required 28 day design strength for all building foundation mats is 4000 psi.

Concrete used for the Main Dam and Auxiliary Dam Spillway has a minimum 28-day design strength of 3000 psi.

3.8.5.7 Testing and In-service Surveillance Requirements

The requirements for in-service surveillance of concrete foundations are the same as those for other Seismic Category I structures and are identified in the following sections:

a) Supports located within the Containment -Section 3.8.3.7

b) Other Category I Supports -Section 3.8.4.7

REFERENCES: SECTION 3.8

3.8.1-1 ASME Section III Division 2/ACI 359-75 "Code for Concrete Reactor Vessels and Containments."

3.8.1-2 ACI 318-71 "Building Code Requirements for Reinforced Concrete."

3.8.1-3 ACI 349-75 "Code Requirements for Nuclear Safety Related Concrete Structures" Appendix C "Special Provisions for Impulsive and Impactive Effects."

3.8.1-4 NRC Regulatory Guide 1.10 "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures."

3.8.1-5 NRC Regulatory Guide 1.13 "Spent Fuel Storage Facility Design Basis."

3.8.1-6 NRC Regulatory Guide 1.15 "Testing of Reinforcing Bars for Category I Concrete Structures."

3.8.1-7 NRC Regulatory Guide 1.18 "Structural Acceptance Test for Concrete Primary Reactor Containments."

3.8.1-8 NRC Regulatory Guide 1.19 "Nondestructive Examination of Primary Containment Liner Welds."

3.8.1-9 NRC Regulatory Guide 1.54 "Quality Assurance Requirements for Coatings Applied to Water-Cooled Nuclear Power Plants."

3.8.1-10 NRC Regulatory Guide 1.55 "Concrete Placement in Category I Structures."

3.8.1-11 NRC Regulatory Guide 1.57 "Design-Limits and Loading Combinations for Metal Primary Reactor Containment System Components."

3.8.1-12 NRC Regulatory Guide 1.60 "Design Response Spectra for Seismic Design of Nuclear Power Plants," Rev. 1, Dec. 1973.

3.8.1-13 NRC Regulatory Guide 1.61 "Damping Values for Seismic Design of Nuclear Power Plants," Oct. 1973.

3.8.1-14 NRC Regulatory Guide 1.92 "Combining Model Responses and Spatial Components in Seismic Response Analysis" Rev 1 Feb. 1976.

3.8.1-15 NRC Regulatory Guide 1.63 "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants."

3.8.1-16 NRC Regulatory Guide 1.76 "Design Basis Tornado for Nuclear Power Plants."

3.8.1-17 NRC Regulatory Guide 1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition."

3.8.1-18 NRC 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.

3.8.1-19 NRC Regulatory Guide 1.122 "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components."

3.8.1-20 NRC Branch Technical Position AAB-3-2 "Tornado Design Classification."

3.8.1-21 "Concrete Manual" - Bureau of Reclamation, 8th Edition, 1975, p. 45.

3.8.1-22 NRC Standard Review Plan Sec. 2.3.1 "Regional Climatology."

3.8.1-23 NRC Standard Review Plan Sec. 2.3.2 "Local Meteorology."

3.8.1-24 NRC Standard Review Plan Sec. 3.3.1 "Wind Loading."

3.8.1-25 NRC Standard Review Plan Sec. 3.3.2 "Tornado Loading."

3.8.1-26 NRC Standard Review Plan Sec. 3.5.1.4 "Missiles Generated by Natural Phenomena."

3.8.1-27 NRC Standard Review Plan Sec. 3.5.2. "Structures, Systems and Components to be Protected from Externally Generated Missiles."

3.8.1-28 NRC Standard Review Plan Sec. 3.5.3 "Barrier Design Procedures."

3.8.1-29 NRC Standard Review Plan Sec. 3.7.1 "Seismic Input."

3.8.1-30 NRC Standard Review Plan Sec. 3.7.2 "Seismic System Analysis."

3.8.1-31 NRC Standard Review Plan Sec. 3.7.3 "Seismic Subsystem Analysis."

3.8.1-32 NRC Standard Review Plan Sec. 3.7.4 "Seismic Instrumentation."

3.8.1-33 NRC Standard Review Plan Sec. 3.8.1 "Concrete Containment" (11/14/75).

3.8.1-34 NRC Standard Review Plan Sec. 3.8.2 "Steel Containment."

3.8.1-35 NRC Standard Review Plan Sec. 3.8.3 "Concrete and Steel Internal Structures of Steel or Concrete Containments (11/24/75).

3.8.1-36 NRC Standard Review Plan Sec. 3.8.4 "Other Seismic Category I Structures."

3.8.1-37 NRC Standard Review Plan Sec. 3.8.5 "Foundations."

3.8.1-38 ASCE "Manual of Standard Practices for Design of Nuclear Power Plants."

3.8.1-39 ANSI "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures A58.1-1972."

3.8.1-40 "Wind Forces on Structures", Task Committee on Wind Forces, Transactions, ASCE, Vol. 126, Part 2, Paper No. 3269, 1961.

3.8.1-41 Maher, F. J., "Wind Loads on Dome-Cylinder and Dome-Cone Shapes," Journal of the Structural Division, ASCE, Vol. 92, No. ST5. Proc. Paper 4933, October 1966.

3.8.1-42 Timoshenko S. and Woinowsky-Krieger S., "Theory of Plates and Shells," McGraw Hill 1959.

3.8.1-43 Fugge W., "Stresses in Shells," Springer-Verlag 1960.

3.8.1-44 Billington D. P., "Thin Shell Concrete Structures," McGraw Hill 1965.

3.8.1-45 Hetenyi M., "Beam on Elastic Foundation," University of Michigan Press 1964.

3.8.1-46 Arsham Amirikian, "Design of Protective Structures," Bureau of Yards and Docks, Department of the Navy, Washington, D. C., August 1950.

3.8.1-47 Recht R. F. and Ipson T. W., "Ballistic Perforation Dynamics" Journal of Applied Mechanics, Transactions of ASME, Vol. 30, Series E, No. 3, September, 1963.

3.8.1-48 Biggs J. M.,"Introduction to Structural Dynamics," McGraw Hill 1964.

3.8.1-49 Norris C. H. et. al., "Structural Design for Dynamic Loads" McGraw Hill 1969.

3.8.1-50 Williamson R. A. and Alvy R. R., "Impact Effect of Fragments Striking Structural Elements" Holmes and Narver, Inc. 1973.

3.8.1-51 Suarez M. A., "Impactive Dynamic Analysis" ASCE National Structural Engineering Meeting, Baltimore 1971.

3.8.1-52 Newmark N. M., "An Engineering Approach to Blast-Resistant Design" ASCE Transactions Paper No. 2786.

3.8.1-53 Blume T. A., Newmark, N. M., and Corning L. H., "Design of Multistory Reinforced Concrete Buildings for Earthquake Motions" PCA 1961.

3.8.1-54 Kennedy R. P., "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects" Holmes and Narver, Inc. Anaheim, California.

3.8.1-55 Winter G., "Design of Concrete Structures" McGraw Hill 1964.

3.8.1-56 Wang C. K. and Salmon C. G., "Reinforced Concrete Design" Intext Educational Publishers N.Y.

3.8.1-57 Dunham C. W., "Advanced Reinforced Concrete" McGraw Hill 1964.

3.8.1-58 Peterka J. A. and Cormak J. E., "Adverse Wind Loading Induced by Adjacent Buildings" ASCE J. Structural Div. March 1976.

3.8.1-59 Doan P. L., "Tornado Considerations for Nuclear Power Plants Structures" Nuclear Safety Vol. 11, No. 4, 1970.

3.8.1-60 Dunlop F. A., and Wiedner K., "Nuclear Power Plant Tornado Design Considerations" ASCE J. Power Div. March 1971.

3.8.1-61 Burket F. E., "Effects of Tornado on Buildings" Civ. Eng. Jan. 1963.

3.8.1-62 ACI Publication SP 17 (1973) "Design Handbook in Accordance with the Strength Design Method of ACI 318-71."

3.8.1-63 Jacobsen, L. S. and Ayre R. S., "Engineering Vibrations" McGraw Hill 1958.

3.8.1-64 Harris C. M., Crede C. E. "Shock and Vibration Handbook" McGraw Hill 1961.

3.8.1-65 Langharr H. L., "Energy Methods in Applied Mechanics" John Wiley & Sons 1962.

3.8.1-66 TRW Nelson Division "Embedment Properties of Headed Studs".

3.8.1-67 TRW Nelson Division, "Construction Applications - Bent Concrete Anchors".

3.8.1-68 PCI "Manual on Design of Connections for Precast Prestressed Concrete".

3.8.1-69 AISC "Manual of Steel Construction" 7th Ed. 1970.

3.8.1-70 Fritz Engineering Laboratory: Tests for Concrete Anchors in Concrete Slabs Stressed in Biaxial Tension. Report No. 200.77.477.1.

3.8.1-71 Timoshenko-Goodier, "Theory of Elasticity" McGraw Hill 1951.

3.8.1-72 ASME Boiler & Pressure Vessel Code Sect. III Div. 1, Subsection NA 1975.

3.8.1-73 ASME Boiler & Pressure Vessel Code Sect. III Div. 1, Subsection NE 1975 Class MC components.

3.8.1-74 Savin B. N., "Stress Distribution Around Holes," NASA 1970.

3.8.1-75 Peterson, R. F., "Stress Concentration Design Factors," John Wiley 1966.

3.8.1-76 Brush, Almroth, "Buckling of Bars, Plates, and Shells" McGraw Hill.

3.8.1-77 Doyle, Chu, "Some Structural Considerations in the Design of Nuclear Containment Liners" Nuclear Engineering and Design 1971.

3.8.1-78 Winstead, Burdett, and Armentront, "Liner Anchorage Analysis for Nuclear Containments" ASCE J. Structural Div., Oct. 1975.

3.8.1-79 Bofaut, Carreira, and Walser, "Creep and Shrinkage in Reactor Containment Shells" ASCE J. Structural Div., Oct. 1975.

3.8.1-80 Hardington, Parker, and Spruce, "Liner Design and Development for the Oldbury Vessels" Paper 56, Conference on Prestressed Concrete Pressure Vessels, London, 1967.

3.8.1-81 Young, Tate, "Design of Liner for Reactor Vessels" Paper J57, Conference on Prestressed Concrete Pressure Vessels, London, 1967.

3.8.1-82 Chapman, Carter, "Interaction Between a Pressure Vessel and Its Liner" Paper J58, Conference on Prestressed Concrete Pressure Vessels, London, 1967.

3.8.1-83 Bishop, Horseman, and White, "Linear Design and Construction" Paper J59, Conference on Prestressed Concrete Pressure Vessels, London, 1967.

3.8.1-84 Parker, "Stress Analysis of Liners for Prestressed Concrete Pressure Vessels" Proceedings of the 1st International Conference on Structural Mechanics in Reactor Technology, Paper H6/1, 1971.

3.8.1-85 Yang, "A Matrix Displacement Method on Pre and Post-Buckling Analysis of Liners for Reactor Vessels" Proceedings of the 1st International Conference on Structural Mechanics in Reactor Technology, Paper H6/2, 1971.

3.8.1-86 Salvatori, R., "Failure Angle of Disc" Westinghouse 1971.

3.8.1-87 Bush, S., "Probability of Damage to Nuclear Components Due to Turbine Failure" Nuclear Safety 14, November 1973.

3.8.1-88 "Full-Scale Tornado Missile Impact Tests," EPRI NP-148, April 1976.

3.8.1-89 Russel, C. R., "Reactor Safeguards," Pergammon Press, 1962.

3.8.1-90 ACI 349-80 "Code Requirements for Nuclear Safety-Related Concrete Structures," Appendix B, "Steel Embedments."

3.8.1-91 Calculation 3-D-02-011, "Thermal Analyses for Containment Liner and Steel Structures", Rev. 1.

3.8.4-1 Winterhorn, H. F. and Fang, H., "Foundation and Engineering Handbook," Van Nostrand Reinhold, 1975.

3.8.4-2 Department of Navy, Naval Facilities Engineering Command, Washington, DC, "Design Manual, Soil Mechanics, Foundations, and Earth Structures, NAVFAC DM-7," March 1971.

3.8.5-1 TVA, Technical Report No. 13, "The Kentucky Project," Appendix D, Design of Kentucky Structures Against Earthquakes.

APPENDIX 3.8A COMPLIANCE WITH ASME SECTION III, DIVISION 2/ACI-359 CODE

Carolina Power & Light Company constructs the SHNPP containment structure in accordance with the technical requirements of the ASME Section III, Division 2/ACI 359 Code for concrete reactor vessels and containments, Section III, Division 2, 1975 Edition with Winter 1975 Addenda, specifically as set forth in subsections CA and CC, with the following exceptions:

CA-1000 <u>General Requirement</u>, CA-4000 <u>Quality Assurance</u>, CC-1000 <u>Introduction</u>

Exception

Application will not be made for a Certificate of Authorization. In addition, code stamping, inspection by authorized inspectors, and preparation of C-1, C-2, and N-3 Data Reports will not be done.

Justification

Carolina Power & Light Company considers the above requirements to be non-mandatory based on the effective date of July 1975 for the ASME Section III, Division 2, ACI 359 Code, as defined in paragraph CA1230. Also, due to the completion of design and the procurement and fabrication of some components, such as those listed in a, b, and c below, retroactive code stamping is not possible.

The administrative requirements contained in subsection CA are complied with insofar as they are consistent with the status of contracting, procurement, fabrication and delivery of materials in relationship to the code's effective date, and insofar as delegation of responsibilities to the contractor or fabricator, as implied by the code does not result in dilution of CP&L's responsibility and obligation to directly control key aspects of the construction activities, particularly the quality related activities controlled in accordance with the Corporate QA Program.

The basic division of responsibilities between organizations had been established in accordance with the description contained in Section 1.4 and contractual arrangements had been executed with various vendors and contractors prior to the effective date of the ASME Section III, Division 2/ACI 359 Code. The contracts are as follows:

a) The contract for fabrication and erection of the containment liner was issued October 2, 1972.

   As of April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1.

b) The contract for cement was issued January 31, 1974. The contract for aggregate was issued May 20, 1974.

   An aggregate quarry had been opened, a crushing operation had been set up, and approximately 525,000 tons of aggregate had been delivered to the site before April 29, 1977.

c) The contract for concrete reinforcing steel was issued on October 23, 1973 and approximately 2600 tons of the containment steel had been fabricated and delivered to the site before April 29, 1977.

d) Daniel Construction Company had been contracted to perform certain construction services at CP&L's direction.

The contracts have been subjected to Quality Assurance Controls which conform to 10 CFR 50 and to CP&L's Quality Assurance Program. The Quality Assurance Program criteria meets or exceeds the criteria set forth in Article CA-4000, except for administrative ties to the ASME Code committee and to the Authorized Inspector.

The extent of surveillance of construction activities that result from the QA Program approved by the NRC during the Construction Permit review is at least as penetrating and comprehensive as if administrative ties were established with the Code committee and an Authorized Inspector. The CP&L Corporate QA Program is responsive to the ASME Section III, Division 2 required program (Ref. CA-4220) to a level of detail equivalent to the Code required program, except for references to the Authorized Nuclear Inspector and as further clarified herein.

CA-4225 Training and Personnel Qualifications

Clarification

Carolina Power & Light Company qualifies personnel as stated in the previously submitted position on Regulatory Guide 1.58, which is consistent with CP&L's position on personnel qualification as stated in Section 1.8. Cognizant personnel are adequately trained and qualified for the work performed.

CA-4523 Nondestructive Examination

Exception

Nondestructive examination for the material procured and/or fabricated by April 29, 1977 (see Table 3.8A-1) for the containment liner was performed in accordance with ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda).

Justification

The contract for fabrication and erection of the containment liner was issued October 2, 1972; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971Addenda), as shown in Table 3.8A-1. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code, with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

CC-2112 Special Rules, CC-4100 General Requirements

Exception

Subsection NA was used in lieu of the requirements of Section CA for material for parts and appurtenances already fabricated and/or contracted.

Justification

The contract for fabrication and erection of the containment liner was issued October 2, 1972; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code, with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

CC-2120 Pressure Retaining and Load Bearing Materials

Exception

Carolina Power & Light Company complies with this paragraph, except that the built up structural steel members used to connect reinforcing steel to clear penetrations, and the dome hub plates used to connect reinforcing steel at the apex of the dome, have been fabricated from ASTM-A572 and ASTM-A537 Class 2 steel, respectively. In addition a notch toughness test was performed and the results are in accordance with the requirements of Table I-2.2 of ASME Section III Division 2/ACI 359 Code.

Justification

The exception does not affect the integrity of the structure since the built up steel members are part of the reinforcing system for which sub article CC-2300, Material for Reinforcing Systems, specifies ASTM steels.

CC-2130 Certification of Materials

Exception

For certain materials, CP&L performs the testing or has it performed by an independent testing laboratory, rather than pass this responsibility on to the constructor.

Justification

Certified Material Test Reports (CMTR) are provided in fulfillment of the requirements of Article CC-2000.

Carolina Power & Light Company has elected to assume an active role in the construction effort and retains direct control of some key testing operations, such as the testing of concrete.

Carolina Power & Light Company, in addition, provides certain of the CMTR's for operations performed by CP&L, Daniel Construction Company, or site subcontractors other than the liner erector. The subcontractors are responsible directly to CP&L.

Exception

The material certifications do not contain a statement that gives the manner in which the material was identified, including specific markings.

Justification

The contract for fabrication and erection of the containment liner was issued October 2, 1972; this was prior to the issuance of ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code, with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

The contract for concrete reinforcing steel was issued on October 23, 1973, and approximately 2,600 tons of the containment steel had been fabricated and delivered to the site by April 29, 1977. The purchase orders for the liner and reinforcing steel impose marking of the plates and reinforcing steel by the suppliers. The plates are marked by heat or slab numbers and the reinforcing steel is tagged by heat numbers and bar marks, in accordance with project specifications. This provides adequate traceability for material identification. CP&L includes a statement in the construction report that describes the manner(s) in which materials are identified.

CC-2220 Material for Concrete

Exception

CP&L complies with this paragraph, except for the acceptance limit on flat and elongated particles and testing of aggregate.

Justification

Aggregate for concrete is procured in accordance with ASTM Specification C33. The contract was issued to the aggregate supplier on May 20, 1974, and the project specification does not contain limitations on flat and elongated particles. The quarry had been opened, the aggregate processing plant had been erected and approximately 525,000 tons of aggregate had been delivered to the construction site before April 29, 1977. Since the ASME Section III, Division 2/ACI 359 Code was not issued at the time of quarry development, this was not a factor for consideration.

Testing of aggregate is in accordance with ASME Section III Division 2 revised 1977.

CC-2300 Material for Reinforcing Systems

Exception

The acceptance standards for the reinforcing steel delivered to the site by April 29, 1977, are in accordance with the tests called for in Section 9 of ASTM A615 for a full diameter section, and in accordance with the supplementary requirements section of the same specification, in lieu of the requirements of CC-2331.

The check analysis tolerances for carbon, manganese, phosphorus, and sulfer for the reinforcing steel already delivered and ordered for delivery to the site by April 29, 1977, are in accordance with ASTM A615, in lieu of the requirements of CC-2333.

Justification

The contract for reinforcing steel was issued October 23, 1973. Approximately 2,600 tons of steel had been delivered to the site before April 29, 1977. Since the ASME Section III, Division 2/ACI 359 Code was not issued at the time of the contract, the specific requirements of CC-2331 and CC-2333 were not imposed.

CC-2520 Special Material Testing

Exception

Notch-toughness tests have not been performed on the anchorage bars for the crane brackets.

Justification

This exception will not affect the integrity of the structure based on the low stress levels used in the sizing of these bars, as shown in the following table:

Loading Condition	Actual Stress*
1. Normal Plant Operation	Less than 25%
2. Normal Plant Shutdown (Refueling)	Less than 35%
3. Construction	Less than 50%
4. Normal Plant Operation Plus OBE	Less than 40%
5. Normal Plant Shutdown (Refueling) Plus OBE	Less than 55%
6. Normal Plant Operation Plus SSE	Less than 45% (Less than 30%**)
7. Normal Plant Shutdown (Refueling) Plus SSE	Less than 65% (Less than 45%**)

*Percentage expressed in terms of allowable stress given in AISC Manual for Steel Construction, Part 5. No increase has been considered except as noted by ** below.

**Percentage in parenthesis expressed in terms of permissible increase of allowable stress.

Exception

For material already procured, as noted in Table 3.8A-1, Calibration of Charpy V-notch temperature measuring instruments had been performed in accordance with ASME Section III, Division 1, Winter 1971 Addenda. Therefore, temperature measuring devices were calibrated twice per year in lieu of at least once every three months, as required by CC-2529.

Justification

The contract for fabrication and erection of the containment liner was issued October 2, 1972; this was prior to the issue of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

CC-2600 Welding Material

Exception

Welding and stud material for parts of the containment liner already fabricated, as noted in able 3.8A-1, complies with ASME Section III, Division 1, Winter 1971 Addenda in lieu of CC-2600.

Justification

The contract for fabrication and erection of the containment liner was issued October 2, 1972; this was prior to the issue of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1. CP&L considers this to be equivalent. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code, with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

CC-2700 Material Manufacturer's Quality Assurance Program

Clarification

Carolina Power & Light Company complies with this paragraph, except that the review and audit of the manufacturers were performed by CP&L on material purchased directly by CP&L.

Exception

The concrete material suppliers were qualified to the requirements of Ebasco Specification CAR-SH-CH-6 (concrete), which imposes ASTM Standards for testing.

Justification

The contract for the supply of concrete aggregate was issued on May 20, 1974, and for the cement on January 31, 1974. The aggregate quarry had been opened, the aggregate processing plant had been erected, and 525,000 tons of aggregate had been delivered to the site before April 29, 1977. Since the ASME Section III, Division 2/ACI 359 Code had not been issued, at the time of the contract, the requirements of CC-2700 were not imposed.

CC-4120 Certification of Material and Fabrication or Construction by Component Fabricator or Constructor

Exception

Certification of material and fabrication or construction by component fabricator or constructor.

Justification

Because the design, procurement, and fabrication of certain components occurred prior to the publication and effective date of the ASME Section III, Division 2/ACI 359 Code, CP&L cannot strictly comply with this paragraph. CP&L provides a certified construction report, prepared by the CP&L Resident Engineer, that summarizes and verifies that activities for construction of the containment comply with the construction specifications, design drawings, and requirements of ASME Section III, Division 2/ACI 359 Code, as modified herein.

The contract for fabrication and erection of the containment liner was issued October 2, 1972; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liners to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1.

Concrete materials were contracted for on May 20, 1974 (for aggregate) and January 31, 1974 (for cement). This was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. The quarry had been opened, the aggregate processing plant had been erected, and approximately 525,000 tons of aggregate had been delivered to the site before April 29, 1977.

The contract for reinforcing steel was executed on October 23, 1973; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. Approximately 2,600 tons of steel had been delivered to the site before April 29, 1977.

CC-4240 Curing

Exception

Containment concrete will be maintained above 50F and in a moist condition for at least 7 days after placing. Concrete temperature and moist cure will be maintained during the first 7 days after placing; however, should either the concrete temperature or moist cure be violated during this period, then the cure (temperature or moisture) will be extended one additional day for each day of deficient cure subject to the following conditions:

1. The concrete temperature shall not be allowed to drop below 40F during the initial 48 hours following placement.

2. If the concrete temperature drops below 50F any time during the first three (3) days after placing, the field shall verify that the in-situ strength of the concrete is not less than 0.7 f_c' before discontinuing curing operations.

3. If the concrete temperature drops below 50F or interruption in moist curing occurs any time during the first seven (7) days after placing, the curing period shall be extended at least one full day for each day of deficient (temperature or moist) cure.

Justification

The Engineer's present specifications require a more stringent concrete protection temperature of 50F in lieu of the 40F requirement. CP&L feels that some provisions must be made to extend the cure period of Containment concrete to rectify periodic occurrences of deficient cure. Extension of cure will assure the required design strength of concrete is obtained coupled with the higher protection temperature of 50F.

Research in concrete curing over the years has indicated that once the cure is extended on a concrete specimen having deficient cure, the maximum strength of the specimen can very nearly be retained, especially if the period of deficient cure is of short duration. Also, concrete mixes at the site have a considerable amount of conservatism in their design, allowing the concrete to cure at much higher strengths than specified.

CC-4250 Formwork and Construction Joints

Exception

Carolina Power & Light Company complies with the requirements of CC-4250, except that expanded metal fabric forms are used in selected locations where construction joints pass through dense reinforcing steel or on the dome. Such forms were used only after successful mockup demonstrations and approval by the responsible design engineer. Mortar leakage was held to acceptable limits, as concurred by the design engineer after the successful mockup demonstrations.

Justification

The containment reinforcing steel design was nearly completed by April 29, 1977, and it is not detailed to provide access for heavy wooden forming systems at the indicated construction joints. The containment design was initiated prior to the issuance of the ASME Section III, Division 2/ACI 359 Code.

CC-4330 Splicing of Reinforcing Bars

Clarification

CP&L complies with the requirements of CC-4330 except as noted herein. Responsibility for qualification of splicing procedures and splicers, and maintenance and certification of records is not totally delegated to an installing contractor. CP&L maintains direct control and responsibility for key administrative, testing, inspection, and record-keeping activities.

CP&L has qualified welding procedures and supervised the welder qualification program since CP&L obtained the ASME Certificate of Authorization for the ASME Section III, Division 1, construction activities.

Exception

Straight sister splices are substituted for production samples for bars bent with large radii.

Justification

Straight sister splices are substituted for production samples for curved bars regardless of radii in accordance with paragraph C3 of Regulatory Guide 1.10 which states that curved reinforcing bars do not tensile test accurately. CP&L complies with the position on Regulatory Guide 1.10 (Section 1.8) in lieu of CC-4333.4.5(b). This is considered an acceptable alternative.

CC-4520 Forming, Fitting, and Aligning

Exception

Carolina Power & Light Company complies with the ASME Section III, Division 2/ACI 359 Code, except for the following tolerances which have been incorporated into existing contracts:

a) The difference between the maximum diameter and minimum diameter at a specified elevation does not exceed 0.65 ft., and the radius from the theoretical centerline of the Containment does not have a minus dimension in excess of 2 1/4 in. or a plus dimension in excess of 3 in. These measurements are taken in at least 26 different points at specified elevations, and not more than 10 ft. apart in the vertical direction.

b) A 3/4 in. deviation from a 10-foot straight edge placed in the vertical direction between circumferential seams. Measurements are taken no closer than 12 in. to a welded seam.

c) The maximum deviation from a straight line or from a true circular or spherical form, measured anywhere on the liner in any direction, does not exceed $\pm$ 1/4 in. in a 14 in. span.

d) Elevations are maintained to within 2 in. of the theoretical elevations up to and including the spring line of the dome. Penetration positions are within $\pm$ 1 in. tolerances.

e) Flat-spots or local out-of-roundness does not exceed 2 in. in 15 ft.

f) For the personnel locks, equipment hatch, and valve chambers, the fabricator achieves tolerances in accordance with Articles NE-4000 of the ASME Boiler & Pressure Vessel Code, Section III, Division 1.

Justification

The above tolerances are as comprehensive as the code table of acceptable deviations. Moreover, while not precisely comparable because different bases of measurement are used, the tolerances presently built into the fabrication are as restrictive as the code and, therefore, result in a constructed product of equal quality. Also, the contract for fabrication and erection of the containment liner was issued on October 2, 1972; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code.

CC-4500 Heat Treatment

Exception

The parts of the containment liner noted as fabricated in Table 3.8A-1 comply with the post-weld heat treatment requirements of the ASME Boiler & Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Division 1, Subsection NE, Winter 1971 Addenda; these requirements are equal to or greater than those of the ASME Section III, Division 2/ACI 359 Code.

Justification

The contract for fabrication and erection of the containment liner was issued on October 2, 1972; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code, with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

CC-5000 Construction Testing and Examination, CC-5500 Examination of Liners

Exception

The containment liner was procured to ASME Section III, Division 1, Subsection NE, (Winter 1971 Addenda), and the additional requirements of NRC Regulatory Guide 1.19. This is considered to be an acceptable alternative. Also, see CA-1000 for other exceptions.

Justification

The contract for fabrication and erection of the containment liner was issued on October 2, 1972; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code, with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

CC-5100 Procedures, Qualification, and Evaluations

Exception

The examination of parts and appurtenances that meet the requirements for Class MC, and which are not backed up by concrete for load-carrying purposes, have been fabricated and/or contracted for to meet the requirements of NE-5000. Subsection NA was used in lieu of the requirements of Subsection CA, which did not exist at time of the contract.

Also, existing contracts do not mandate the participation of an Authorized Nuclear Inspector, except for the valve chambers and airlocks which were fabricated and/or contracted for according to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda).

Justification

The contract for fabrication and erection of the containment liner was issued on October 2, 1972; this was prior to the issuance of the ASME Section III, Division 2/ACI 359 Code. By April 29, 1977, the liner fabricator had essentially completed design, purchased material and shop fabricated the bulk of the liner to ASME Section III, Division 1, Subsection NE (Winter 1971 Addenda), as shown in Table 3.8A-1. For further work on the containment liner, including liner to penetration and liner to lock or hatch welds, CP&L complies with the requirements of the ASME Section III, Division 2/ACI 359 Code, with the exception that any other site welding performed on containment penetrations, personnel and escape locks, and the equipment hatch will be in accordance with ASME Section III, Division 1 requirements.

CC-5210 General

Clarification

CC-5210 requires testing to be performed by the constructor. The organizational structure and relationship between the Constructor and CP&L is described in the FSAR; the discussion indicates that certain testing is done by CP&L.

CC-5220 Concrete and Materials

Exception 1:

Carolina Power & Light Company employs a test frequency of once per day by the manufacturer in accordance with ASTM C150, in lieu of once every 1200 tons as required by CC-5221.2.

Justification 1:

The contract for cement was executed on January 31, 1974, which was prior to the requirements of CC-5220 becoming effective. The insignificant increase in safety that could result from renegotiation of the cement contract to include the test frequency required by CC-5221.2 is not commensurate with the costs that would be incurred.

Exception 2:

Delete the requirements for testing of soft particles in accordance with ASTM C 235 as indicated in Table CC-5200-1.

Justification 2:

Both ASTM and ASME/ACI 359 have deleted the requirement for the soft particle test. ASME no longer considers the soft particle test as a valid test. CP&L should not continue expending time to implement these tests.

CC-5320 Examination of Sleeve with Filler Metal Connections

Exception

In lieu of removal and replacement of splices not meeting the stated criteria, repairs are performed as discussed in Section 3.8.1. This repair method is limited to repair of defective cadwelds in applications where a cadweld sleeve is welded directly to embedded structural shapes or in other instances where room does not exist to install two replacement splices.

Justification

As stated in Section 3.8.1.6.1(g2), repairs of defective cadwelds are performed in accordance with qualified procedures and in accordance with the controls outlined in Section 3.8.1.6.1(g2). CP&L considers this adequate to assure the quality of the repairs.

APPENDIX 3.8B COMPUTER PROGRAMS USED IN STRUCTURAL ANALYSES

Abstracts of computer programs used for the design and analyses of the nuclear power plant structures are contained in this appendix. Basically they are industry proven computer programs, such as STARDYNE, NASTRAN, and ANSYS. Some of the computer programs were developed by Ebasco Services, Inc.; the accuracy of these programs has been validated by comparison with results from manual calculations and/or commercially available industry proven computer programs. Other computer programs that are not listed in this Appendix may be used for design and analysis of structures. These computer programs are maintained in accordance with FSAR Section 17.3, "HNP Quality Assurance Program Description."

3.8B.1 DYNAMIC 2037 FREQUENCY RESPONSE AND SPECIAL ANALYSIS OF STRUCTURES

This computer program was developed by Ebasco Services for analysis of the dynamic behavior of structures that are subjected to seismic forces. After a dynamic model of the building structure is established, the program determines dynamic responses in the horizontal or vertical direction.

Responses include natural frequencies, mode shapes, modal damping factors, accelerations, forces, shears, moments, and floor response spectra. In addition, options are available for computing free field ground spectra, displacements, and velocities from a specified earthquake record.

By knowing mass distribution, sectional properties, and the soil structure interaction of a structural system, the stiffness of each member can be calculated and the stiffness matrix for the entire system can be assembled. By using the Jacoby iteration technique, natural periods and associated vibrational mode shapes of the structural system can be determined. Then the program, using multi-degree-of-freedom and modal analysis methods together with either a time-history analysis or response spectrum method, can find the dynamic responses of the structure. To find response spectra, the time history method is employed.

The input may be by punched cards and/or tapes. Input includes geometry, material properties and damping factors of the structures, and data describing the earthquake.

Options are available pertaining to the type of structure and the analysis applicable to the structure. The required input data depends on the specified options. For example, lumped mass cantilevers may have horizontal connecting members between lumped mass points, and earthquake records may be either horizontal, or rocking, or both acting simultaneously. The user can opt to calculate a time history of accelerations and displacements of any mass point during the course of a given earthquake record, or the user can calculate such maximum responses as forces, shears, moments, displacements, and accelerations. The response spectrum method follows the latest R.G. 1.92 "Guidelines for Modal Superposition".

Other program options permit:

a) Parabolic baseline corrections for an uncorrected earthquake acceleration record when velocities and displacements are computed by double integration.

b) Printing out moment and shear time-histories for the base of each cantilever and for the mat foundation.

c) Plotting ground or floor spectra (broadened or unbroadened) for specified damping factors.

d) Including the effect of rotary inertia of the mass point on structural responses.

e) Plotting mode shapes for the first five modes of each cantilever to graphically display structural behavior.

f) Damping factors imposed on the structure system to be either uniform structural or material dependent structural damping.

g) Analyzing the effect of heavy equipment mounted on the floors of the structure by adding "branching mass points" to the vertical dynamic model.

The program is limited to plane structures and the dynamic model should not have more than 15 cantilevers supported by a common foundation mat. Each cantilever is limited to a maximum of 20 mass points for a linear translation case (LTC), and 10 for a rotary-inertia included case (RIC). However, the total number of masspoints for the whole structure is limited to 148 for LTC and 79 for RIC, not counting the base mat.

The user may choose as many as four damping factors and 150 periods or frequencies to calculate a spectrum curve. Output includes natural frequencies, mode shapes, and dynamic responses imposed on a structure by a seismic record.

3.8B.2 NASTRAN

NASTRAN (NASA Structural Analysis) was developed under the sponsorship of the National Aeronautics and Space Administration (NASA) by a committee with representation from eight NASA centers (for specifications), and by the Computer Sciences Corporation and Bell Aerosystems Company (for implementation). NASTRAN is a finite-element computer program for structural analysis that is intended for general use. Structural elements include rods, beams, shear panels, plates, shells of revolution, and scalar and solid polyhedron elements. The range of analysis of the program includes static response to concentrated and distributed loads, to thermal expansion, and to enforced deformation; dynamic response to transient loads, to steady state sinusoidal loads, and to random excitation; and determination of real and complex eigenvalues for use in vibration analysis, dynamic stability analysis, and elastic stability analysis. The program also has a limited capability for the solution of nonlinear problems, including piecewise linear analysis of nonlinear static response and transient analysis of nonlinear dynamic response.

The displacement method is employed throughout the analysis. Structures are modeled with finite elements, including plate elements, shells of revolution elements, shear panels, beams, and rods. Elements are identified by numbers, and are interconnected at a finite number of grid points. The grid points may be defined by a basic or local coordinate system. Each grid point may have six degrees of freedom, representing three displacements and three rotations. After receiving the input data, the first task of the program is to generate the stiffness matrix and the load vector. The next step is the matrix decomposition, which is especially important because of the required computing time, possible error accumulation, and numerical instability. The program takes maximum advantage of matrix sparsity and bandedness. The band width is greatly influenced by the user, who establishes the numbering system for the grid points.

Using the finite element technique, any type of structure can be accurately modeled. Deformation constraints (displacements and rotations) and boundary conditions may be imposed on any grid point. Boundary conditions may be homogeneous or nonhomogeneous. The outputs from the analysis are the displacements and rotations for each grid point and the moments and stresses in each element. Forces in elements may be calculated from the output stresses at the two extreme fibers of each element.

3.8B.3 STARDYNE

The MRI/STARDYNE Structural Analysis System is a fully warranted and supported engineering application package available at CDC-6600 Data Centers.

The MRI/STARDYNE Analysis System consists of a series of compatible digital computer programs designed to analyze linear elastic structural models. The system encompasses the full range of static and dynamic analyses.

The STARDYNE system can be used to evaluate a wide variety of static and dynamic problems:

a) The static capability includes the computation of structural deformations and member loads and stresses caused by a set of thermal, nodal applied loads and/or prescribed displacements.

b) By utilizing the normal mode technique, dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady state harmonic, and random and shock spectra excitation types. Dynamic response results can be presented as structural deformations (displacements, velocities, or accelerations) and/or internal member loads and stresses.

The programmed mathematical operations in the matrix decomposition, the eigenvalue eigenvector extraction, and the error analysis contain state of the art innovations in the field of numerical analysis.

The basic concept of the "Finite Element" method is that every structure may be considered as a "mathematical" assemblage of individual structural components or elements. There must be a finite number of such elements, interconnected at a finite number of nodal points. The behavior of this finite element structural model will closely approximate the behavioral characteristics of the real structure.

The eigenvalues (natural frequencies) and eigenvectors (normal modes) of a structural system are determined by solving the equation

$$w^2[m]\ \{q\}\ -\ [k]\ \{q\}\ =\ 0$$

where:

[m] = the mass matrix

w = the natural frequencies

{q} = the normal modes

[k] = the stiffness matrix

Using the natural frequencies and normal modes with the related mass and stiffness characteristics of the structure, appropriate equations of motion may be evaluated to determine the structural response to dynamic loading.

The general solution procedure consists of stiffness matrix formulation, static analysis, eigenvalue/eigenvector determination, and dynamic response analysis.

The stiffness properties of the individual finite elements are first expressed in a convenient local (element) coordinate system. The element stiffness matrix is then transformed from its local coordinate formulation to a form relating to the global coordinate system. Finally, the individual element stiffness contributing to each nodal point are superimposed to obtain the total assemblage stiffness matrix [k].

During a static analysis, the equation

$$[k] \cdot \{\delta\} = \{P\}$$

where:

[k] = the stiffness matrix

{δ} = the nodal displacement vector

{P} = the applied nodal forces

may be solved to determine the nodal displacements and element internal forces and/or stresses, given a set of applied nodal forces.

In STARDYNE, modal damping may be entered for each computed mode, or for each material number (the number can designate an actual material, or a particular region of the structure, or both). For composite modal damping, the damping of each mode is weighted by the strain energy associated with each material number. In a particular mode, if only one region of the structure has appreciable motion, the modal damping will be equal to the damping value assigned to the material for that region. If the entire structure is in motion, the damping will be the weighted average of all materials (or locations) throughout the structure, depending on the amount of kinetic energy associated with each material (or location).

The STARDYNE modal damping is the same as the one used in the Ebasco in house dynamic code.

The STARDYNE expression for modal damping is:

$$D_j = \frac{\sum_{i=1}^{n} \{\varphi_j\}^T \, b_i \, [k]_i \, \{\varphi_j\}}{\{\varphi_j\}^T \, [K] \, \{\varphi_j\}}$$

where:

n	=	total number of degrees of freedom
b_i	=	equivalent percent of critical damping associated with component i
$\{\varphi_j\}$	=	mode shape vector for mode j
$[k]_i$	=	stiffness associated with component i
$[K]$	=	stiffness matrix for the system

The STARDYNE Analysis System comprises various computer programs such as STAR, DYNRE 1, DYNRE 2, DYNRE 3, DYNRE 4, and DYNRE 5.

The STAR program has two distinct functions; these are static load analysis and eigenvalue/eigenvector extraction. The static analysis and modal extraction phases are based on the "Stiffness Method" or "Displacement Method" and the answers are in the realm of the "Small Displacement Theory".

Transient response to imposed dynamic loadings is treated in DYNRE 1. Input forcing functions may be in the form of forces, initial displacements, initial velocities, and base accelerations. Output nodal forces and/or displacements at selected time points may be processed in STAR for element stresses.

Steady state frequency response to imposed dynamic loadings is computed by DYNRE 2. Input forcing functions may be in the form of distributed forces, base excitations (displacements, velocities, or accelerations) and unit sinusoidal excitations (displacements, velocities, accelerations, or forces) at specific nodes. Displacements at selected angles may be processed in STAR for element stresses.

DYNRE 3 investigates the responses of multi-degree-of-freedom linear elastic structural models subjected to stationary random dynamic loading. DYNRE 3 computes the root mean square (RMS) nodal responses and RMS element stresses, and generates response power spectral density (PSD) curves for selected nodal degrees of freedom. Input forcing power spectrums are defined as shape of spectrum and type of spatial correlation.

DYNRE 4 investigates the responses of multi-degree-of-freedom linear elastic models that are subjected to an arbitrarily oriented foundation shock input. The user may enter SHOCK SPECTRA for any of the directions of motion or call for some ratio of the 1940 El Centro (California) earthquake SPECTRA. Seismic input based upon Regulatory Guide 1.60, as described in Section 3.7.1, can also be employed.

DYNRE 4 computes the ABSOLUTE and/or RMS sum of the nodal responses and/or element stresses. ABSOLUTE responses are computed by adding the motion of the structural component to the motion of the base.

DYNRE 5 computes shock spectrum values from a transient base acceleration time history digitized at equal or unequal time intervals. The user may specify frequencies at which shock spectrum values for displacement, velocity and acceleration will be computed, in turn, for each value of damping entered.

3.8B.4 ANSYS

ANSYS is a large scale general purpose computer program for the solution of several classes of structural analysis problems.

Analytical capabilities include static, dynamic, plastic, creep, and swelling investigations, small and large deflections, steady state and transient heat transfer, and steady state fluid flow.

The matrix displacement method of analysis, based upon finite element idealization is employed throughout the program. The available library contains more than 40 subsystems for static and dynamic analyses, and 10 subsystems for heat transfer analyses. This variety gives the ANSYS program the capability of analyzing frame structures (two dimensional frames, grids, and three dimensional frames), piping systems, two dimensional plane and axisymmetric solids, flat plates, three dimensional solids, axisymmetric and three dimensional shells, and non-linear problems, including interfaces and cables.

Loading on the structure may be forces, displacements, pressures, temperatures, or response spectra. Loadings may be time functions for linear and non-linear dynamic analyses. Loadings for heat transfer analyses include internal heat generation, convection and radiation boundaries, and specified temperatures or heat flows.

The ANSYS program uses the wave front (or "frontal") direct solution method for a system of simultaneous linear equations developed by the matrix displacement method, and gives highly accurate results. The program has the capability of analyzing large structures. There is no practical limit to the number of elements that can be used in a problem. The number of nodes can be in excess of 2500 for three dimensional problems, and 5000 for two dimensional problems. There is no "band width" limitation in the problem definition; however, there is a "wave front" restriction. The "wave front" restriction depends on the amount of core storage available for a given problem. Up to 576 degrees of freedom on the wave front can be handled in a large core. The wave front limitation tends to be restrictive only for analysis of three dimensional solids or if ANSYS is used on a small computer.

ANSYS has the capability of generating substructures (or super-elements). These substructures may be stored in a library file for use in other analyses.

Geometry plotting is available for all elements in the ANSYS library, including isometric, perspective, and sectional views of three dimensional structures. Plotting subroutines are also available for plotting stresses and displacements from two and three dimensional solid or shell analyses, mode shapes from dynamic analyses, distorted geometries from static analyses, transient forces and displacements vs. time curves from transient dynamic analyses, and stress-strain relationships from plastic and creep analyses.

Post-processing routines are available for algebraic modification, differentiation, and integration of calculated results. Root mean square operations may be performed on seismic modal results. Response spectra may be generated from dynamic analysis results. Results from various loading modes may be combined for harmonically loaded axisymmetric structures. Options for multiple coordinate systems in cartesian, cylindrical, or spherical coordinates are available, as well as multiple region generation capabilities to minimize the input data for repeating regions.

3.8B.5 SHELL FINITE ELEMENT ANALYSIS CONSIDERING CRACKING OF CONCRETE EFFECTS

This computer program was developed by Ebasco Services, Inc.

In this program, the analysis of cylinders with axisymmetric loads, such as accident pressure, test pressure, gravity, and temperature loads, is based on the primary membrane theory. In addition, the local bending moment and radial shear in the vicinity of the cylinder-base juncture, are also analyzed by applying the conditions of compatibility at the junctures. The analytical procedures and formulations are based on those contained in References 3.8B-5 through 3.8B-8. The change in sectional properties due to cracking (or cracking) of concrete under accident pressures and test pressures are considered in the analysis.

The following three types of cracks are considered:

a) A membrane crack, which is an axisymmetrical crack or a crack formed around the whole circumference. This crack results from internal pressure loads, or from internal pressure combined with other asymmetric loads.

b) A local membrane crack, which is an asymmetrical local crack constituting only a part of the circumference. This crack results from a seismic load in the normal operating condition.

c) A partial bending crack, which is associated with only a portion of the section (along the thickness). For example, a horizontal crack due to the discontinuity moment at the lower portion of the containment wall under an accident pressure load is considered to be a partial bending crack.

The following considerations are involved in evaluating the effects of concrete cracks:

The containment cases under axisymmetric load are analyzed by the method specified below. The accident pressure is the load that causes a membrane crack in the major portion of the shell and a partial bending crack at each boundary. The membrane stress resultants are not affected by the sectional properties of the shell; however, the boundary discontinuity moments are affected by the sectional properties of the shell. Since this is a material non-linear problem, an iterative process is employed to obtain reliable results.

The containment crack model is shown on Figure 3.8.1-27. The containment analysis used to account for sectional property variations and changes due to concrete cracks is a finite element method which uses the beam on elastic foundation approach to represent the actual cylindrical shell of revolution. The finite element method for determining the stress and displacement fields of a structure is based on the concept that every structure may be regarded as an assemblage of a finite number of discrete elements interconnected at a finite number of nodes. Finiteness of the structural connectivity is the essential feature which separates the analysis from one of continuum mechanics and allows a solution by matrix equations, as described by Zienkiewicz O.C. and Cheung Y.K. in "The Finite Element Method in Structural and Continuum Mechanics", McGraw Hill 1967.

The procedure used to determine the concrete cracking model in the SHELL computer program is as follows:

a) An axisymmetric model was used for the vertical direction. Lengths of members, members' equivalent thickness and initial moment of inertia, temperature, and internal pressure were input into the program. The output furnished moments and shears for each of the elements. Membrane forces were calculated by the shell membrane theory (N_d = PR/2). The moment of inertia of each element was calculated and compared with the value used in the input. Two iterations were performed to obtain convergence for the concrete cracking model.

b) The concrete was considered to be cracked in the circumferential direction, except for the lowest portion of the wall where the foundation mat restrains the wall from deforming.

c) The concrete cracking model used in the SHELL program was verified by the ANSYS computer program. Axisymmetric finite elements were used. Initial equivalent element thickness, moduli of elasticity, temperature and internal pressure were input into the program. The output furnished element moments, shears, and axial forces. Equivalent element thickness and moduli of elasticity were calculated and compared with the values used in the input. Two cycles of iteration were performed.

d) The concrete cracking obtained by the SHELL analysis is less than that by the ANSYS analysis. Since the ANSYS cracking is based on two directional material properties, it is more representative of actual cracking. Use of the SHELL cracking results in more conservative values for moments and shears.

There are three basic phases in a finite element analysis of a structure:

a) Idealization of the original structure into an assemblage of discrete elements.

b) Evaluation of the element stiffness.

c) Analysis of the finite element assemblage.

In short, the finite element analysis consists of the following procedures:

a) The meridional and circumferential membrane force restraints, which are independent of the sectional properties, are first calculated by the classical membrane theory.

$$Nx = \frac{PR}{2} \quad (1)$$

$$N\theta = PR \quad (2)$$

Where:

P = pressure, psf

R = Radius of the containment in ft.

Nx = Meridional membrane force in k/ft.

Nθ = Circumferential membrane force in k/ft.

b) The radial displacements are calculated by the membrane theory, considering a free boundary condition and a completely cracked section.

$$d_i = \frac{P}{K_i} \quad (3)$$

$$K_i = \frac{Et_i}{R^2} \quad (4)$$

where:

d_i = free boundary radial displacement for i^{th} element

E = Young's modulus

t_i = equivalent thickness of the reinforcing steel for i^{th} element

K_i = Shell equivalent modulus of elastic foundation

c) At the vicinity of the boundary, where discontinuity moments and radial shear develop, the axisymmetrical bending theory is used and its closed form solution (Reference 3.8B-8) is employed to construct a flexibility matrix. As shown on Figure 3.8.1-27, that a finite number of elements can be subdivided, each of which may be assigned different sectional properties based on the presumed compression uncracked zone. The equation is written in matrix form:

[f] {F} = {d}

where:

[f] is the flexibility matrix size 2N x 2N

{F} is the generalized forces, including 2N elements {d} is the relative incompatible displacements, which are obtained as described in b above, equation (3).

The flexibility matrix is constructed by the following 4 x 4 element matrice:

$$\begin{bmatrix} a_{11} & a_{12} & a_{13} & a_{14} \\ & a_{22} & a_{23} & a_{24} \\ & & a_{33} & a_{34} \\ & & & a_{44} \end{bmatrix} \begin{bmatrix} S_i \\ M_i \\ S_{i+1} \\ M_{i+1} \end{bmatrix}$$

Where:

a_{11} $= a_{33} = \frac{2\lambda}{\kappa A}$ (sinh λℓ cosh λℓ - sin λℓ cos λℓ)

$A = \sinh^2\lambda\ell - \sin^2 \lambda\ell$

a_{12} $= \frac{-2\lambda^2}{\kappa A}$ ($\sinh^2 \lambda\ell + \sin^2 \lambda\ell$)

a_{13} $= \frac{2\lambda}{\kappa A}$ (sinh λℓ cos λℓ -sin λℓ cosh λℓ)

a_{14} $= \frac{4\lambda^2}{\kappa A}$ (sinh λℓ sin λℓ)

a_{22} $= a_{44}\ \frac{4\lambda^3}{\kappa A}$ (sinh λℓ cosh λℓ + sin λℓ cos λℓ)

a_{23} $= \frac{-4\lambda^2}{\kappa A}$ (sinh λℓ sin λℓ)

a_{24} $= \frac{4\lambda^3}{\kappa A}$ (sinh λℓ cos λℓ + sin λℓ cosh λℓ)

a_{34} $= \frac{2\lambda^2}{\kappa A}$ ($\sinh^2 \lambda\ell$ + sin λℓ)

λ $= \sqrt[4]{\kappa/4\ EI}$

I = Sectional moment of inertia of the corresponding element.

ℓ = length of the shell finite element.

κ $= \frac{Et}{R2}$

d) After the shears and moments are computed the total moments and meridional membrane forces for each specific loading combination are obtained by summing up all the moments and meridional membrane forces due to the individual factored loads.

e) After the total meridional membrane forces and moments at each mode are determined, the compression zone at each node point is computed to check with the presumed compression zone at each node point. If they are sufficiently close, the iterative process is completed and the final stresses are determined. If they are not close, another trial is made.

f) Superposition is not valid in this process; a complete cycle iteration is performed for each load combination.

3.8B.6 EBS/NASTRAN

This computer program was developed by Ebasco Services, Inc. It consists of a series of modules which serve as lead subprograms for computers analysis of various problems. The modules are linked to United Analytics, Inc. UAI/ NASTRAN for overall solution of the problem. The application of the program is to the module which calculates propagation of cracks in concrete sections under loading. The cracked section is represented by a series of flat shell layers (Figure 3.8.2-14) with coupled in plane and out of plane stiffnesses, enabling the monitoring of crack propagation from one layer to the next at each stage of loading, and the determining of the section properties at various extents of cracking.

3.8B.7 ADINA

ADINA (Automatic Dynamic Incremental Nonlinear Analysis) by Adina Engineering, Inc. is a finite element program for static and dynamic displacement analysis of solids, structures, and fluid-structure systems. The program performs linear and nonlinear analysis. The nonlinearities may be due to large displacements, large strains, and nonlinear material behavior. The finite element system response is evaluated using an incremental solution of the equations of equilibrium. In dynamic analysis, implicit time integration (Newmark or Wilson methods) or explicit time integration (central difference method) may be used. A variety of two and three dimensional element types and materials can be treated.

3.8B.8 WHIPRES 2615

This computer program was developed by Ebasco Services, Inc. It is used to perform simplified dynamic analysis to obtain values for pipe restraint and restraint support design. The time history of pipe rupture blowdown forces at the rupture location postulated for an elbow of the piping system is input into the program, and the program computes displacements, velocities and reactions which are applied to design.

3.8B.9 FLUSH

FLUSH is a computer program developed by J. Lysmer, T. Udaka, C. F. Tsai, and M. B. Seed for approximate three dimensional analysis of soil-structure interaction problems. It is based on, and is a further development of computer program LUSH. The program computes maximum shear forces in beam elements, acceleration and velocity response spectra, and plots Fourier amplification functions. Soils and structures can be modeled by plane strain quadrilateral elements. Beam elements may also be used for structures. Multiple nonlinear soil properties for equivalent linear analysis permit the use of different damping values for each element.

3.8B.10 SHAKE (VERSION 2)

SHAKE (Version 2, December 1972) by B. Schnabel, J. Lysmer, and H. B. Seed is a computer program for earthquake response analysis. The program computes the response in a horizontally layered soil-rock system subjected to transient vertically traveling shear waves by Kanai's solution to the wave equation and the Fast Fourier Transform Algorithm. The input motion can be applied to any layer in the system. Systems with elastic base and with variable damping in each layer can be analyzed. Equivalent linear soil properties are used with an iterative procedure to obtain soil properties compatible with the strains developed in each layer.

3.8B.11 ABAQUS

ABAQUS by Hibbitt, Karlsson, and Sorensen, Inc. is a large scale finite element computer program for the solution of static and dynamic linear and nonlinear analysis problems. Simultaneous effects of nonlinearities in materials, geometry, and boundaries can be calculated.

3.8B.12 STRUDL

STRUDL (Structural Design Language) is a computer program developed by Massachusetts Institute of Technology. The program performs various static and dynamic analyses of structural systems. The program also performs individual member selection based on tables of properties and/or standards design criteria.

a) MCAUTO STRUDL

MCAUTO STRUDL is a computer program developed by McDonnell Douglas Automation Company and Multisystems, Inc. The program uses STRUDL.

b) PSDI STRUDL

PSDI STRUDL is a computer program developed by Programs for Structural Design, Inc. The program uses STRUDL.

3.8B.13 BASICPLATE 2476

This computer program was developed by Ebasco Services, Inc. The program performs finite element analysis using plate bending elements. It applies to plate bending problems capable of being mathematically represented by a discretized finite element mesh under plate bending action. The approximate solution technique utilized by the program is based upon the finite element method in engineering science for known elastic homogeneous isotropic media. The element used is a triangular plate bounded by three nodes, each with three permissible degrees of freedom. The translational degree of freedom is normal to the plane of the element. The two rotational degrees of freedom lie in the plane of the element. Basic input consists of the total geometry of the model, the location and types of loading schemes and boundary conditions within the limitations of the program. Output includes the total displacements of each node in each degree of freedom and the internal equilibrating forces of each element, referenced to the centroid of the element.

3.8B.14 NPS BASEPLATE ANALYSIS

This program, by Nuclear Power Services, Inc., calculates anchor bolt loads, plate maximum stress, and load point displacements for flexible baseplates, using a finite element approach. The program performs nonlinear plate bending finite element analysis that incorporates the flexibility of the plate in bending, and plane analysis that assumes the plate is completely rigid in order to determine the shear forces. The analyses are applicable to rectangular baseplates of any size attached by anchor bolts to rigid foundations.

3.8B.15 THPLOT 2524

This computer program was developed by Ebasco Services, Inc. It reads, organizes, and plots earthquake time-histories, acting on input data of selected time stations. The program can be applied to any type of data which can be described against a time frame to obtain histories (such as, for example, of acceleration, overturning moment, and eccentricity about specified axes).

3.8B.16 POSBUKF 2628

This computer program was developed by Ebasco Services, Inc. The program examines elastic post-buckling behavior of a flat plate subjected to thermal and lateral pressure loads. An energy method approach is used for the analysis. A buckled deflected shape is assumed for the plate, and a total potential energy expression for the deflected shape is established. The magnitude of the buckled plate deflections is then determined by minimizing the potential energy. Stresses in the buckled plate are then calculated by utilizing strain-displacement and stress-strain relations. Effects of significant imperfections in the plate are considered separately, since the program assumes that imperfection in the plate is infinitesimal.

3.8B.17 UGEOM 2542

This computer program was developed by Ebasco Services, Inc. It is used in the design of multidirectional U-bar pipe whip restraints. The program calculates the path traversed by the centerline of the pipe during its motion under restraint by the U-bar. Strain in the U-bar is assumed to be constant during each traverse.

3.8B.18 WTM-104

This computer program was developed by Ebasco Services, Inc. It calculates the weight moment of inertia and area moment of inertia with respect to the centroid of the section for different sections of building structures, acting on shapes of rectangular prism, hollow right circular cylinder, sector of hollow right circular cylinder, right wedge (obtuse of oblique), right circular complement, hollow right circular cone, right circular cone frustrum, and spherical cap. The program also calculates lumped weights.

3.8B.19 EAC/EASE

EAC/EASE (Elastic Analysis for Structural Engineering) is a computer program developed by Engineering Analysis Corporation for Control Data Corporation. The program performs static analysis of linear three dimensional structural systems subjected to various loads. The systems may be beams, membranes, or plates, and the loadings may be mechanical, pressure, or thermal, with displacement boundary conditions.

3.8B.20 ANALYSIS OF REINFORCED CONCRETE SECTION

This computer program was developed by Ebasco Services, Inc. for the calculation of stresses and strains in the steel and concrete of reinforced concrete sections with several layers of reinforcement. The program is used for load cases which have combined axial and bending loadings in which the axial loading is non-trivial. Analysis is based on the design equations of CC-3000 of ASME Boiler and Pressure Vessel Code Section III, Division 2, Subsection CC. Input consists of section geometry, location and area of steel in each reinforcement layer, modulus of elasticity of steel and concrete, and the design loading. Output consists of stresses and strains in steel and concrete based on linear variation in stress and strain through the depth of the section.

3.8B.21 CCLU49

This computer program was developed by Ebasco Services, Inc. It is a post processor program which converts STARDYNE finite element (triangular only) stress output (STARDYNE Tape 4, File 2) into element forces and moments to apply to the design of the reinforced concrete structure that was analyzed. Forces and moments are obtained by the use of the equations below. In the equations, F is the element force in the local coordinate system, M is the internal moment of plate theory, S is the element stress, and +Z and -Z identify the element face to which the force or moment applies.

$$\begin{Bmatrix} Fx \\ Fy \\ Fxs \end{Bmatrix} = (T - Tc)/2 \left[\begin{Bmatrix} Sx \\ Sy \\ Sxy \end{Bmatrix}_{+Z} + \begin{Bmatrix} Sx \\ Sy \\ Sxy \end{Bmatrix}_{-Z} \right]$$

$$\begin{Bmatrix} Mx \\ My \\ Mxy \end{Bmatrix} = T^2/12 \left[\begin{Bmatrix} Sx \\ Sy \\ Sxy \end{Bmatrix}_{+Z} - \begin{Bmatrix} Sx \\ Sy \\ Sxy \end{Bmatrix}_{-Z} \right]$$

3.8B.22 CCLU59B

This computer program was developed by Ebasco Services, Inc. It is applied to the Portland Cement Association (PCA) computer program for strength design of reinforced column sections with axial load and biaxial bending. The Ebasco program develops a table from the PCA program analysis of a given column section. The table furnishes values of allowable axial load in combination for various values of bending about either or both principal axes of the given column section.

3.8B.23 CCLU65C

This computer program was developed by Ebasco Services, Inc. It calculates bending moments in a prismatic member with restrained ends which is subjected to a temperature gradient in a transverse direction. The program also prepares a tabulation which furnishes the value of moment along the member with the values of axial load in the members calculated by EAC/EASE due to temperature gradient in the member in its longitudinal direction.

3.8B.24 CCLU67A

This computer program was developed by Ebasco Services, Inc. It is a post processing program for EAC/EASE. Input consists of EASE output files with load data for primary loads at nodal points in beam-column systems. The program uses the data of the files and calculates the value of seismic load (forces and moments) at each nodal point by the square root of the sum of the squares (SRSS) method, acting on the value of load in the three principal directions. The SRSS loads are combined with the primary loads in effect for the node points. The results are stored on tape and used as input for the Portland Cement Association (PCA) computer program for strength design of reinforced column sections with axial load and biaxial bending.

Verification of the computer codes described in Appendix 3.8B is provided as follows:

a) DYNAMIC 2037: "Frequency Response and Special Analysis of Structures." Verification of this program is described in "Verification of Ebasco Code Dynamic 2037" from Ebasco.

b) NASTRAN: This program is verified in "The NASTRAN Demonstration Problem Manual," NASA SP-224(3).

c) STARDYNE: This program is verified in the STARDYNE "Theoretical Manual."

d) ANSYS: This program is verified in "ANSYS Engineering Analysis System Verification Manual" by Swanson Analysis System, Inc.

e) SHELL FINITE ELEMENT ANALYSIS CONSIDERING CRACKING OF CONCRETE EFFECTS. Verification of this program was presented in the Shearon Harris Nuclear Power Plant Preliminary Safety Analysis Report (Section 5.1.1.8.2).

f) EBS/NASTRAN: This program is verified by comparison of the results of the program with known solutions and test results.

g) ADINA: This program is verified in Adina Engineering Report AE 81-1, Automatic Dynamic Incremental Nonlinear Analysis User's Manual.

h) WHIPRES 2615: This program is verified by comparing the results with those of the computer program used for the design of piping systems.

i) FLUSH: The program is documented in Report No. EERC 75-30, Earthquake Engineering Research Center, the University of California, Berkeley, California.

j) SHAKE: The program is justified by the author's comparisons of program results with field observations for a number of cases.

k) ABAQUS: The program is verified in the ABAQUS Example Problems Manual by comparison of program results with the analytical solution of a series of test case example problems for various applications.

l) MCAUTO STRUDL: This program is verified by McDonnell Douglas Automation Company in UO164123, "STRUDL Verification Problems Manual."

m) PSDI STRUDL: This program is verified by UCCEL Corp. in UCCEL Proprietary Quality Assurance Manual Program Verifications and in P-Delta STRUDL Verification Manual.

n) BASICPLATE 2476: The program verification consists of verification for static linear problems and static nonlinear problems in which foundation soils are modeled as compression-only springs. Both linear and nonlinear analysis verifications are obtained by comparing the results of analysis (deflections, moments, and shears for selected points and elements) with corresponding results of computer program ANSYS.

o) NPS BASEPLATE ANALYSIS: The program is verified by the Nuclear Power Services, Inc. through the comparison of solutions obtained from the ANSYS finite element computer program. Some test cases are presented in the NPS BASEPLATE VERIFICATION MANUAL.

p) THPLOT 2524: The program is verified by comparison of program output with known shapes and ordinates of several well-known figures used as input data.

q) POSTBUKF 2628: The program is verified by comparison of program results for representative cases with the results of hand calculations.

r) UGEOM 2540: The program is verified by comparison of program results with the results of hand calculations.

s) WTM-104: Program verification is obtained by comparison of results of program calculations for various shapes with the results of hand calculations.

t) EAC/EASE: Verification of this program is documented in the Engineering Analysis Corporation EAC/EASE Example Problem Manual.

u) ANALYSIS OF REINFORCED CONCRETE SECTION: The program is verified by comparison of program results with the results of hand calculations for representative problems.

v) CCLU49: This program is verified by comparison of program results with the results of hand calculations for representative problems.

w) CCLU59B: The program is verified by comparison of program results with the results of hand calculations for a series of problems.

x) CCLU65C: The program is verified by comparison of program results with those obtained by hand calculations for representative prismatic members.

y) CCLU67A: The program is verified by comparison of program results with those obtained by hand calculations for representative load cases.

REFERENCES: APPENDIX 3.8B

3.8B-1 Ebasco Services, Inc.: Program 2037 User's Manual for Frequency Response and Spectral Analysis of Structures.

3.8B-2 NASTRAN Theoretical Manual.

3.8B-3 STARDYNE Static and Dynamic Structural Analysis System - Theoretical Manual.

3.8B-4 ANSYS Engineering Analysis System - User's Manual.

3.8B-5 Timoshenko S., Woinosky - Krieger S.: "Theory of Plates and Shells" McGraw Hill 1959.

3.8B-6 Flugge W.: "Stresses in Shells" Springer-Verlag 1960.

3.8B-7 Billington P.P.: "Thin Shell Concrete Structures" McGraw Hill 1965.

3.8B-8 Hetenyi M.: "Beam on Elastic Foundation" The University of Michigan Press 1948.

3.8B-9 Zienkiewicz, O. C.: "The Finite Element Method in Engineering Science" McGraw-Hill, 1971.

3.9 MECHANICAL SYSTEMS AND COMPONENTS[1]

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

The following five operating conditions as defined in Section III of the ASME B&PV Code are considered in the design of the Reactor Coolant System, 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 is included in the design specifications.

c) Emergency Conditions (Infrequent Incidents) - Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an allowable stress (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

[1] Further information is contained in the TMI Appendix.

the integrity and operability of the nuclear energy system may be impaired to the extent that consideration requires compliance with safety criteria as may be specified by jurisdictional authorities.

e) Testing Conditions - Testing conditions are those pressure overload tests including hydrostatic test, pneumatic test, and leak test specified. Other types of tests shall be classified under normal, upset, emergency, or faulted conditions.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in 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 enough 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 limiting design transients and the number of cycles of each transient that is normally used for fatigue evaluations are shown in Table 3.9.1-1. In accordance with ASME III, emergency and faulted conditions are not included in fatigue evaluations. The fatigue analyses for the 60-year renewed license are discussed in Chapter 18.

3.9.1.1.1 Normal conditions

The following primary system transients are considered normal conditions:

a) heatup and cooldown at 100°F/hr.,

b) unit loading and unloading at 5 percent of full power/min. between 15 and 100 percent power,

c) step load increase and decrease of 10 percent of full power,

d) large step load decrease with steam dump,

e) steady-state fluctuations,

f) feedwater cycling at hot standby,

g) unit loading and unloading between 0 and 15 percent of full power,

h) boron concentration equalization,

i) refueling,

j) turbine roll test,

k) primary side leak test and,

l) secondary side leak test.

Heatup and Cooldown at 100°F/hr. - The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F/hr. (These operations can take place at lower rates approaching the minimum of 0°F/hr. The expected normal rates are 50°F/hr.).

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/hr. may not be achievable 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/hr.

b) Slower initial heatup rates when using pump energy only.

c) Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry, and gas adjustments.

For design purposes, the ambient temperature of 120°F is assumed. However, the reactor coolant temperature can be as low as 70°F during shutdown period. Between 70°F and 120°F the reactor coolant temperature is assumed to change very slowly without causing any significant thermal transient effects. Additionally, there is sufficient conservatism in the frequency of occurrence of this transient, plus the conservative assumption that ALL of the heatups proceed at a maximum rate of 100°F/hr vs. the maximum plant heatup rate of up to 50°F/hr will account for an initial temperature as low as 70°F.

<u>Unit Loading and Unloading at 5 Percent of Full Power Per Minute</u>

The unit loading and unloading cases are conservatively represented by continuous and uniform ramp power change of 5 percent/min. between 15 percent load and full load. This load swing is the maximum possible consistent with operation under automatic reactor control. The reactor temperature will vary with load as prescribed by the reactor control system.

<u>Step Load Increase and Decrease of 10 Percent of Full Power</u>

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 Rod System is designed to restore plant equilibrium without reactor trip following a 10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent full load, the power range for automatic reactor control. In effect, during load change conditions, the Reactor Control Rod 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 change.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load as determined by first-stage turbine pressure measurement. The pressurizer pressure will also ultimately decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. During the decreasing pressure transient, the saturated water in the pressurizer flashes, 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 initially decreases. Pressurizer level and pressure will initially decrease as a result of the outsurge. Additional charging flow and pressurizer heaters may be actuated. The control system automatically withdraws the control rods to increase core power and temperature. The increasing pressure transient due to pressurizer insurge is reversed by actuation of the pressurizer sprays, and 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.

Large Step Load Decrease With Steam Dump

This transient applies to a step decrease in turbine load from full power, of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator safety valves. However, due to problems with the feedwater system during initial plant start-up, the system was modified such that the capability of accepting a large step load decrease from full power is not possible without a plant trip. Since this was the severest transient for a step load, the results of the design analysis remain bounding. Under the Measurement Uncertainty Recapture-Power Uprate (MUR-PU) the plant steam dump system capacity has been reduced (from 70% to 40%) reflecting a relaxation for the electrical load rejection from 100% to 50% without incurring a rector trip or lifting the MS safety valves.

Steady-State Fluctuations

It is assumed that the reactor coolant temperature and pressure at any point in the system vary around the nominal (steady state) values. These local variations can occur at many frequencies, but for design purposes, two cases should be 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 ± 3.0°F and pressure by ± 25 psi, once during each 2 minute period. The total number of occurrences is limited to 150,000. 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 plant design life does not exceed 3,000,000.

Feedwater Cycling at Hot Standby - This transient can occur when the plant is being maintained at hot standby or no-load conditions. It is assumed that either main feedwater (through the main feedwater nozzle) or auxiliary feedwater (through the auxiliary nozzle) is initiated periodically to maintain the water level in the steam generator. For the auxiliary nozzle, it is conservatively assumed that the water is taken, unheated, from an outside condensate storage tank at 32°F. For the main nozzle, feedwater is assumed to be supplied from a heated source at a lower temperature limit of 100°F.

Feedwater additions required during plant heatup and cooldown operations are also assumed to be covered by the feedwater cycling transient, but with no increase in the total number of cycles.

Unit Loading and Unloading Between 0 Percent and 15 Percent Power

The unit loading and unloading cases between zero and 15% power are represented by continuous and uniform ramp 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% power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot standby condition, with feedwater additions performed as defined in the "Feedwater Cycling" design transient. It is further assumed that the plant will be started using either the auxiliary feedwater followed by a transfer to main feedwater (mode 1) or using the main feedwater only (mode 2). During plant loading, the auxiliary feedwater temperature is assumed to be constant at 32°F and the main feedwater temperature is assumed to increase from 100°F to the 15% power value. During plant unloading the main feedwater temperature is assumed to decrease from 15% power value to 100°F and the auxiliary feedwater temperature is assumed to be constant at 32°F.

In the event of an extended outage where cold ambient temperatures results in feedwater temperatures of ≤ 100°F in portions of the feedwater system piping, the feedwater nozzles may be subjected to increased stresses during the initial plant heat-up, startup, or normal condition transient "unit loading between 0 and 15% power". Under the most extreme conditions (i.e. normal plant transient - Unit Loading Between 0% and 15% Power with the initial feedwater at 40°F and the steam generator secondary side at 557°F), the allowed number of thermal cycles is 60. The remaining number of allowable thermal transients is 120 cycles at a feedwater temperature of ≥ 100°F.

Boron Concentration Equalization

Following any large change in boron concentration in the RCS, the pressurizer spray is initiated in order to equalize concentration between the loops and the pressurizer. This is assumed to occur manually by operation of the pressurizer backup heaters, thus causing a pressure increase which will initiate spray at pressurizer pressure of approximately 2275 psia. The proportional sprays return the pressure to 2250 psia and maintain this pressure by matching the heat input from the backup heaters until the boron concentration is equalized.

Refueling

At the beginning of the refueling operation, the RCS is assumed to have been cooled down to 140°F. 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 outdoors and conservatively assumed to be at 32°F, into the loops by means of the Residual Heat Removal pumps. It should be conservatively assumed that the cold water flows directly into the reactor vessel and that all the fluid in the RCS is replaced with the colder water within 10 minutes.

Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature (no load conditions) and the steam generated is used to perform a turbine roll test. However, the plant cooldown during this test exceeds the 100°F/hr. design rate.

The number of such test cycles is specified at 80 times, to be performed at the beginning of plant operating life prior to irradiation. Since this transient occurs before plant startup, the number of cycles is independent of other operating transients.

Primary Side Leakage Test

Subsequent to each time the primary system is opened a system leakage test is performed according to FSAR Section 5.2.4.7.

During this leakage test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This is accomplished by isolating the steam, feedwater, and blowdown lines.

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, and that the secondary side temperature will be between 120˚F and 250°F. The replacment steam generator is designed for a secondary side leakage test temperature as low as 70°F. In order to not exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized. In addition, the primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements at the existing primary side pressure.

3.9.1.1.2 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 Reactor Coolant System depressurization, (including inadvertent auxiliary spray)*

f) operating basis earthquake.

g) excessive feedwater flow

h) control rod drop,

i) inadvertent safety injection and

j) RCS cold overpressurization

*The RCS depressurization transient (20 occurrences during the life of the plant) is applicable for all components. For the pressurizer only, an additional 10 occurrences of inadvertent auxiliary spray actuation transient is considered. For design purposes it is assumed that no temperature changes in the RCS, with the exception of the pressurizer, occur as a result of initiation of pressurizer auxiliary spray.

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 pressure trip initiated by the Reactor Protection System (RPS). Since redundant means of tripping the reactor are provided as a part of the RPS, transients of this nature are not expected, but are included to ensure a conservative design.

Loss of Power

This transient applies to a situation involving the loss of offsite electrical power to the station, assumed to be operating initially at 100 percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are deenergized and, following coastdown of the reactor coolant pumps, natural circulation in the system decays to some equalibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater, assumed to be at 32oF, from the Auxiliary Feedwater System operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

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.

Reactor Trip From Full Power

A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

The severity of the cooldown transient following a reactor trip depends on the extent of steam generator secondary side cooling. Three basic cooldown cases are considered.

Case A: Reactor Trip with No Cooldown
Case B: Reactor Trip with Cooldown and No SI
Case C: Reactor Trip with Cooldown and SI

Inadvertent Reactor Coolant System Depressurization

a. Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the Reactor Coolant System. These include:

b. Actuation of a single pressurizer safety valve.

c. Inadvertent opening of one pressurizer power operated relief valve, due either to equipment malfunction or operator error.

d. Malfunction of a single pressurizer pressure controller causing one power operated relief valve and two pressurizer spray valves to open.

e. Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error.

f. 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 SIS are actuated when RCS pressure decreases by approximately 1600 psi. 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.

Although inadvertent auxiliary spray actuations are included among the depressurization transient events covered above, the pressurizer safety valve actuation cases selected to represent all the depressurization transients does not involve spray operation. Therefore, for the previous case it is assumed that pressurizer spray is not actuated and that no temperature transients due to flow occur at the pressurizer spray nozzle.

However, should auxiliary spray flow be initiated inadvertently, it could cause severe thermal shock at the pressurizer spray nozzle and on the pressurizer vessel. Therefore, for the purposes of analyzing the spray nozzle and pressurizer vessel an "inadvertent auxiliary spray" transient is defined.

The inadvertent pressurizer auxiliary spray transient will occur if the auxiliary spray valve is opened inadvertently during normal operation of the plant. This will introduce cold water into the pressurizer resulting in a very sharp pressure decrease.

The temperature of the pressurizer auxiliary spray water is dependent upon the performance of the regenerative heat exchanger. The most conservative case is when the letdown steam is shut off and the charging fluid enters the pressurizer unheated. Therefore, for design purposes, the temperature of the spray water is assumed to be 100°F. The spray flow rate is assumed to be 200 gpm. It is furthermore assumed that the pressurizer auxiliary spray will, if actuated, continue for five minutes until it is shut off.

The pressure decreases rapidly to the low pressure reactor trip point. At this pressure the pressurizer low pressure reactor trip is assumed to be actuated; this accentuates the pressure decrease until the pressure decreases to the hot leg saturation pressure. At five minutes, spray is stopped and all the pressurizer heaters return the pressure to 2250 psia. If the pressurizer heaters were not in operation the pressure would remain at the value reached in five minutes.

For design purposes it is assumed that no temperature changes in the Reactor Coolant System, with the exception of the pressurizer, occur as a result of initiation or pressurizer auxiliary spray.

<u>Operating Basis Earthquake</u>

The mechanical stresses resulting from the operating basis earthquake are considered on a component basis. Fatigue analyses, where required by the codes, are performed by the supplier as part of the stress 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.

<u>Excessive Feedwater Flow</u>

An excessive feedwater flow transient is conservatively defined as an umbrella case to cover occurrence of several events of the same general nature. The postulated transient results from inadvertent opening of a feedwater control valve while the plant is at the hot standby or no load condition, with the feedwater, condensate and heater drain systems in operation.

It is assumed that the stem of a feedwater control valve fails and the valve immediately reaches the full open position. In the steam generator directly affected by the malfunctioning valve (failed loop), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of all operating feedwater pumps. Steam flow is assumed to remain at zero and the temperature of the feedwater entering the steam generator is conservatively assumed to be 32°F. A low pressurizer pressure signal actuates the SIS and isolates the main feedwater flow. 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.

Control Rod Drop

This transient occurs if a bank of control rods (worth 1% reactivity) drops into the fully inserted position due to a single component failure. The reactor is tripped on low pressurizer pressure, depending on time in core life and magnitude of the reactivity insertion.

Inadvertent Safety Injection Actuation

A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head centrifugal charging pumps. These pumps deliver the contents of the boron injection tank to the RCS cold legs. The initial portion of this transient is similar to the Reactor Trip from Full Power with no cooldown. Controlled steam dump and feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decrease as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase to the pressurizer power operated relief valve set point and the primary and secondary temperatures to decrease gradually. The transient continues until the operator stops the charging pumps. It is assumed that the plant is then returned to no-load conditions, with pressure and temperature changes controlled within normal limits.

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 the addition of mass (mass input transient) 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. Refer to Sections 5.2.2.11 and 7.6.1.11 for additional information on RCS cold overpressurization during low temperature operation.

3.9.1.1.3 Emergency Conditions

The following primary system transients are considered emergency conditions:

a) Small loss-of-coolant accident
b) Small steam line break
c) Complete loss of flow

Small Loss-of-Coolant Accident

For design transient purposes the small loss-of-coolant accident is defined as a break equivalent to the severance of a 1-inch ID branch connection. (Breaks smaller than 0.375-inch ID 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. It should be assumed that the Safety Injection System is actuated immediately after the break occurs and 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. The following conservative assumptions are made:

a. The reactor is initially in a hot, zero-power condition.

b. The small steam break results in immediate reactor trip and S.I. 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 Reactor Coolant System 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 under-voltage followed by automatic opening of the steam dump system.

3.9.1.1.4 Faulted Conditions

The following primary system transients are considered faulted conditions. Each of the following accidents is evaluated for one occurrence each, except steam generator tube rupture, which is evaluated for six occurrences.

a) reactor coolant pipe break (large loss-of-coolant accident)

b) large steamline break,

c) feedwater line break,

d) reactor coolant pump locked rotor,

e) control rod ejection,

f) steam generator tube rupture, and

g) safe shutdown earthquake.

Reactor Coolant Pipe Break (Large Loss-of-Coolant Accident)

Following a 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 ECCS 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 Steamline Break

This transient is based on the complete severance of the largest steamline.

The following conservative assumptions were made:

a) The reactor is initially in startup conditions.

b) The steamline 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 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 break accident.

Feedwater Line Break

This accident involves a double ended rupture of the main feedwater piping from full power, resulting in the rapid blowdown of one steam generator and the termination of main feedwater flow to the others. The blowdown is completed in approximately 27 seconds. Conditions were conservatively chosen to give the most severe primary side and secondary side transients. The auxiliary feedwater is actuated within 60 seconds and supplies flow to the faulted and intact loops. The operator manually isolates the auxiliary feedwater within 10 minutes after the initiation of the incident.

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 Reactor Coolant System 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 single steam generator tube.

The resultant primary to secondary break flow causes the shell side level to rise in the affected steam generator. The loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or overtemperature ΔT. Plant cooldown following reactor trip leads to a rapid decrease in RCS pressure and pressurizer level. A safety injection (SI) signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The SI signal automatically terminates steam generator blowdown, normal feedwater supply and initiates auxiliary feedwater (AFW) addition via the motor driven AFW pumps. If the steam generator level decreases below the low-low level setpoint in two of the three steam generators or a loss of offsite power occurs, the turbine-driven AFW pump will also be started. Recovery procedures also include isolation of steam flow from, and feedwater flow to the ruptured steam generator. This accident will result in a transient which is no more severe than that associated with a reactor trip from full power. It therefore requires no special treatment insofar as fatigue evaluation is concerned. Six of these occurrences have been postulated for the SHNPP.

Safe Shutdown Earthquake

The mechanical dynamic or static equivalent loads due to the vibratory motion of the safe shutdown earthquake are considered on a component basis as part of the mechanical loading conditions specified in the equipment specifications.

3.9.1.1.5 Test Conditions

The following primary system transients under test conditions are considered:

a) primary side hydrostatic test,

b) secondary side hydrostatic test, and

c) steam generator 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. The hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3107 psig (1.25 times design pressure). In this test, the Reactor Coolant System is pressurized to 3107 psig coincident with steam generator secondary side pressure of 0 psig. These tests are performed prior to plant startup. The number of cycles is independent of other operating transients.

Additional hydrostatic tests as discussed in Section 5.2.4 will be performed to meet the inservice inspection requirements of ASME Section XI. A total of four such tests is expected. The increase in fatigue usage factor caused by these tests is easily covered by the conservative number of primary side leakage tests that are considered for design.

Secondary Side Hydrostatic Test

The secondary side of the steam generator is pressurized at 1.25 design pressure with a minimum water temperature of 120°F coincident with the primary side at 0 psig. The replacement steam generator is designed for a secondary side hydrostatic test temperature as low as 70°F.

These tests may be performed either prior to plant startup or subsequently, following shutdown for major repairs, or both. The number of cycles is therefore independent of other operating transients.

Steam Generator Tube Leakage Test

During the life of the plant it will be necessary to check the steam generator for tube leakage and tube to tube sheet leakage. This is done by visual inspection of the underside (channel head side) of the tube sheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdowns.

For these tests the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure and the primary system remains depressurized. The underside of the tube sheet is examined visually for leaks. If any are observed, the secondary side is then depressurized and repairs made by tube plugging. The secondary side is then repressurized (to a higher pressure) and the underside of the tube sheet is again checked for leaks. This process is repeated until all the leaks are repaired. The maximum (final) secondary side test pressure reached is 840 psig. Both the primary and secondary sides of the steam generator are at temperatures between 70°F and 250°F during these tests.

Test Pressure	Number of Occurrences
200	400
400	200
600	120
840	80

3.9.1.2 Computer Programs Used in Analyses

3.9.1.2.1 NSSS Equipment

The following computer programs have been used by Westinghouse in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment. These are described and verified in Reference 3.9.1-1.

a) WESTDYN-7 - static, dynamic, and fatigue analysis of redundant piping systems,

b) FIXFM3 - time-history response of three-dimensional structures,

c) WESDYN-2 - piping system stress analysis from time-history displacement data,

d) THRUST - (STHRUST in Reference 3.9.1-1) hydraulic loads on loop components from blowdown information,

e) WESAN - reactor coolant loop equipment support structures analysis and evaluation, and

f) WECAN - finite-element structural analysis.

g) STRUDL - steam generator, reactor coolant pump & pressurizer support structures analysis and evaluation

h) MULTIFLEX (Reference 3.9.1-3) - computes thermal-hydraulic-system conditions for loss-of-coolant accidents (LOCA) for the entire Reactor Coolant System.

i) FORCE2 (Reference 3.9.1-3, Appendix B) - computes vertical LOCA forces on reactor vessel internals using thermal-hydraulic-system data supplied by MULTIFLEX.

j) LATFORC (Reference 3.9.1-3, Appendix A) - computes lateral LOCA forces on reactor vessel shell, core barrel, and thermal shield using thermal-hydraulic-system data supplied by MULTIFLEX.

3.9.1.2.2 Balance of Plant Equipment

Computer programs used in the analysis of NSSS vendor supplied Seismic Category I piping and components are described in Section 3.9.1.2.1.

The dynamic and static analyses of other Seismic Category I equipment is analyzed and certified by the equipment manufacturer. Information on seismic qualification is presented in Section 3.9.2.2.

Seismic Category I piping systems within Ebasco scope of supply are analyzed using the following computer programs:

The following is a brief description of each program and the extent of its application.

a) PIPESTRESS 2010 - PIPESTRESS 2010 is a proprietary computer program developed by Ebasco Services, Inc. for linear elastic analysis of three dimensional piping systems including multiple branches and closed loops. It is continuously updated to incorporate additional features and development options. The program constructs a linear finite element model of the piping system using the load-deflection relationships based on the displacement method. Matrix decomposition is used to solve the system of equations for the static problem. The eigenvalue extraction employs matrix decomposition with matrix iteration and purification. Extraction of close eigenvalues and accelerated rate of convergence is accomplished by shifting the origin of the eigenvalues and by coplanar rotation.

The major program features are as follows:

1) The program performs stress calculations in conformance with either:

   a. American National Standards Institute B31.1 Piping Code.

   b. ASME Boiler and Pressure Vessel Code, Section III for Class 1, 2 and 3.

2) Static analysis for loading conditions due to pressure, applied loads, thermal expansion, dad weight, support movement, differential settlement, cold spring and seismic acceleration.

3) Frequency analysis of lumped mass model to compute natural frequencies and mode shapes.

4) Response analysis using single level or multi-level spectra to calculate modal bound solutions for the primary term, and generated support movement cases to calculate bounds for the secondary term. The Left Out Force method may be used to include the effect of the higher, rigid modes.

5) Generalized Response Analysis using the time history of the applied load to directly calculate a modal bound solution. The Left Out Force method may be used to include the effect of the higher, rigid modes.

6) Thermal transient analysis using the finite difference approximation to find thermal gradients in the pipe walls, due to step or ramp temperature changes. The program determines (by an iterative technique) the times during each transient when the various stress terms will be maximized.

7) Combination cases to combined components of forces, moments and deflections from independent loading conditions, using a choice of methods:

   a) Algebraic addition (only for static loads),

   b) Addition to absolute values,

   c) Square root of sum of squares,

   d) Addition in the direction of a specified loading case,

e) Maximum components,

f) Maximum resultants.

8) Restraint Load Combination Case to combine anchor and restraint forces, moments and deflections from independent loading conditions. This case is similar to the above excepting that the computations for piping member component forces, moments, deflections and stresses are not performed.

9) Combined Stress Case to evaluate Equations 9 and 11 of ASME Section III NC/ND-3600 as well as Equations 12 and 14 of ANSI B31.1.

10) Fatigue Analysis is prescribed in Section III of the ASME Code for Class 1 piping. Forces and stresses due to cyclic loads are calculated, and are used to determine the cumulative fatigue damage.

11) Time History Analysis using the program THIST. The Left Out Force method may be used to include the effect of the higher, rigid modes.

The program is limited to linear elastic behavior and small deformations. Program capacity limitations are delineated in the user's manual.

PIPESTRESS 2010 solutions to ASME sample problems have been compared with the solutions to the same sample problems generated by similar, independently written programs in the public domain, namely, ANSYS, PIPESD and ADLPIPE. The comparison shows the PIPESTRESS 2010 results to be substantially identical to results generated by the above programs and by hand calculations. The results were summarized in the Washington Public Power Supply System Nuclear Units No. 3 and 5 PSAR (Docket Nos. STN 50-508 and 509).

In addition, PIPESTRESS 2010 solutions to seven benchmark piping problems were compared with results published in NUREG/CR-1677, Vol. 1, "Piping Benchmark Problems". Results were identical to documented solutions.

b) PLAST 2267 - This program provides information to the piping stress analyst and the designer of pipe whip restraints. The stress analyst requires data on the possible dynamic impingement of some section of pipe on vital (safety related) components. Hence, maximum pipe deflections at the restraints and at other locations are calculated. The possibility that the pipe exceeds its ultimate strain value at some location other than the initial pipe break must also be reviewed. Therefore, the effective plastic strain, as defined later, is furnished for each element. The restraint designer requires data on the maximum reactive force developed by the restraint and the restraint's maximum ultimate strain. This information is included in the output as well.

This program is restricted to small deformations and elasto-plastic materials with bilinear stress strain curves including strain hardening. Although pipe whip dynamic analysis is the present major application of the program, the program may be applied to any elasto-plastic piping frame subjected to dynamic loadings.

The piping system is modeled as a lumped parameter system. The matrix displacement method is used as the means of developing a stiffness matrix and of finding displacements, velocities and accelerations of lumped masses. This method is basic to finding external and internal forces on the system when forces and displacements are specified at its boundaries. The Newmark Beta Method is the numerical integration scheme used to solve the set of second order, ordinary differential equations that represent the equations of motion for the system.

In addition, elements that are known to remain elastic throughout may be so designated.

The length of time the program runs varies with each problem. A cut off point may be designated as occurring at any one or a combination of the following events:

1) 50 percent of the ultimate strain of the restraint is exceeded,

2) The upper bound allowable forces at the restraint is exceeded;

3) Some one of several possible check points (including the restraints) has exceeded its maximum allowable deflection (for interference);

4) Some point in the pipe has exceeded 50 percent of its ultimate strain value;

5) The deflection and/or reaction force at the restraint oscillates about some point below its peak value and subsequent maxima are below the peak value.

The following comments may also be applied in general to this program:

1) The speed of operation depends on the highest natural frequency of the discrete system. From this point of view, short still elements with small mass are not desirable;

2) The solution represents the superposition of all frequency modes in the system at any time. For nonlinear elasto-plastic systems, a mode shape defies definition and so this information is not furnished;

3) Restart features are included so that runs may be made incrementally using the data from an earlier run as input to a subsequent run;

4) Provisions for gaps at pipe whip restraints are in the program;

5) Restraints other than the pipe whip restraints and the pipe anchors (fully fixed points) are to be considered inadequate in restraining pipes from whipping and are conservatively excluded from analysis.

Ebasco Topical Report ETR-1002, "Design Considerations for the Protection from Effects of Pipe Rupture," provides an analytic description of PLAST 2267, including validation cases.

3.9.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Seismic Category I systems or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests as discussed in Subsection 3.9.2.

3.9.1.4 Considerations for the Evaluation of the Faulted Conditions

3.9.1.4.1 Loading Conditions

The structural stress analyses performed on the RCS consider the loadings specified as shown in Table 3.9.1-2. These loads result from thermal expansion, pressure, weight, Operating Basis Earthquake (OBE), Safe Shutdown Earthquake (SSE), system operating transients, LOCA loop hydraulic forces, subcompartment pressurization forces, and reactor vessel loads.

3.9.1.4.2 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.

<u>Pressure</u>

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.

<u>Weight</u>

A dead weight analysis is performed to meet ASME Code requirements by applying a 1.0 g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

<u>Seismic</u>

The input for the reactor coolant loop seismic analysis is in the form of three statistically independent orthogonal time history accelerations. The earthquake accelerations for the horizontal directions are applied to the containment base mat simultaneously with the vertical acceleration.

<u>Loss-of-Coolant Accident</u>

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 3.9.1.1.

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 modulus of elasticity, the coefficient of thermal expansion at the metal temperature, the external movements transmitted to the piping due to thermal growth of the primary equipment and the temperature rise above the ambient temperature define the required input data to perform the flexibility analysis for thermal expansion.

To provide the necessary high degree of integrity for the Reactor Coolant System, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operating conditions.

3.9.1.4.3 Reactor Coolant Loop Analytical Models and Methods

The analytical methods used in obtaining the solution consists of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the time-history integration method for seismic dynamic analysis, and the time-history integration method for loss-of-coolant accident dynamic analysis.

The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffnesses of auxiliary line piping which affects 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-7 computer program, is represented by an ordered set of data which numerically describe the physical system. Figure 3.9.1-1 shows an isometric line schematic of this mathematical model.

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, the coefficient of thermal expansion, the average temperature change from ambient temperature, 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 of 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. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for weight, thermal, and general pressure loading conditions are obtained by using the WESTDYN-7 computer program. The derivation of the hydraulic loads for the loss-of-coolant accident analyses of the loop is covered in Section 3.6.

Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and primary equipment. The containment internal structures and all of the piping loops are included in the coupled building/loop system model. The effect of the equipment motion on the reactor coolant loop/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The steam generator is represented by four discrete masses. The lowest mass is located at the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. The second mass is located midway between the lower and upper support elevations. The third mass is located at the upper support elevation and the fourth at the steam outlet nozzle.

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 is located near the center of gravity of the motor.

The reactor vessel and core internals are represented by four discrete masses.

The component upper and lower lateral supports are inactive during plant heatup, cooldown, and normal plant operating conditions. However, these restraints become active due to the rapid motions of the reactor coolant loop components that occur from dynamic loadings and are represented by stiffness matrices an/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The total response is obtained using the model super position method for time integration of the equations of motion. The results of the analysis are time history forces and displacements. The time history displacement response is then used in computing support loads and in performing the reactor coolant loop piping stress evaluation.

Loss-of-Coolant Accident

Postulated breaks in the RCL, except for Surge, Accumulator, and Residual Heat Removal branch nozzles, have been eliminated from the structural design basis. The mathematical model used in the static analyses is modified for the loss-of-coolant accident analysis 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.

The time-history hydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural lumped-mass node points.

The dynamic structural solution for the full-power loss-of-coolant accident is obtained by using a modified-predictor-corrector-integration technique and normal mode theory.

When elements of the system can only be represented as single acting members (tension or compression members), they are considered as non-linear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple nonlinear elements can be applied at the same node, if necessary.

The time-history solution is performed in program FIXFM3. The input to this program 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 in the system at the postulated break location due to the initial steady-state hydraulic forces, thermal forces, and weight forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response of the system for the static equilibrium position). The time-history displacement solution of all dynamic degrees of freedom is obtained using FIXFM3 and employing 4 percent critical damping.

The loss-of-coolant accident displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the reactor coolant loop. The loss-of-coolant accident analysis of the reactor vessel includes all the forces from internal reactions. Note, for branch line breaks, there are no cavity pressurization loads since the branch lines are outside of the reactor vessel cavity. There are no significant loop mechanical loads for branch line breaks since the loop remains intact. The reactor vessel analysis is described in Subsection 3.9.1.4.6.

The resultant asymmetric external pressure loads on the reactor coolant pump and steam generator resulting from a postulated pipe rupture and pressure buildup in the loop compartments, are applied to the same integrated reactor coolant loop / supports system model used to compute loadings on the components, component supports, and reactor coolant loop piping as discussed above. The response of the entire system is obtained for the various external pressure loading cases from which the internal member forces and piping stresses are calculated. The equipment support loads and piping stresses resulting from the external pressure loading are added to the support loads and piping stresses calculated using the loop LOCA hydraulic forces and RPV motion. The asymmetric subcompartment pressure loads are provided to Westinghouse by Ebasco Services, Inc. The analysis to determine these loads is discussed in Section 6.2

The asymmetric external pressure loads described above were based on the original plant design basis reactor coolant loop breaks. Since the implementation of leak-before-break (LBB), the asymmetric external pressure loads associated with RCS branch lines breaks are bounded by those calculated for the original analysis. Thus, they can be conservatively applied for SGR/Uprating analysis assuming LBB.

The time-history displacements of the FIXFM 3 program are used as input to program WESDYN 2 to determine the internal forces, delections, and stresses at each of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses.

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 computer code into three parts, a uniform, a linear, and 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 3.9.1.1 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat transfer program is used to solve the thermal transient problem. The pipe is represented by at least 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 other 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 3.9.1 2.

The average through-wall temperature, T, is calculated by integrating the temperature distribution across the wall. This integration is performed for all steps so that T_A is determined as a function of time.

$$T_A(t) = \frac{1}{H}\int_o^H T(X,t)dX \quad (1)$$

The range of temperature between the largest and smallest value of T is used in the flexibility analysis to generate the moment loadings caused by the associated temperature changes.

The thermal moment above the midthickness of the wall caused by the temperature distribution through the wall is equal to:

$$M = E_a \int_o^H X - \frac{H}{2}\, T(X,t)dX \quad (2)$$

The equivalent thermal moment produced by the linear thermal gradient about the midwall thickness is equal to:

$$M_L = E_a \frac{\Delta T_1}{12} H^2 \quad (3)$$

Equating M and M_L, the solution for T as a function of time is:

$$\Delta T_1(t) = \frac{12}{H^2} \int_o^H \left[x - \frac{H}{2}\right] T(X,t)dX \quad (4)$$

The maximum nonlinear thermal gradient, T, 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.

$$\Delta T_{21}(t) = |T(0,t) - T_A(t)| - \frac{|\Delta T_1(t)|}{2} \quad (5)$$

Load Set Generation

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 3.9.1-2. 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 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) $\alpha_A T_A$ - $\alpha_B T_B$; (α represents the coefficient of thermal expansion)

d) moment loads due to T_A, and

e) pressure loads.

This procedure produces at least twice as many load sets as transients for each point.

For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors and cumulative usage factors, are calculated. The WESTDYN-7 program is used to perform this analysis in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3650. Since it is impossible to predict the order of occurrence of the transients over the plant life, it is assumed that the transients can occur in any sequence. This is a very conservative assumption.

The incremental usage factor is calculated for the combination of load sets yielding the highest alternating stress intensity range. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles <10^6 are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.1.4.4 Primary Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models. The equipment support structure models are dual-purpose since they are required 1) to quantitatively represent the elastic restraints which the supports impose upon the 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.

For each operating condition, the loads (obtained from the reactor coolant loop analysis) acting on the support structures are appropriately combined. The adequacy of each member of the steam generator supports, reactor coolant pump supports, and piping restraints is verified by solving the ASME III Section NF stress and interaction equations by means of hand calculations or the WESAN (Reference 3.9.1-1) computer program. The adequacy of the RPV support structure is verified using the WECAN computer program and comparing the resultant stresses to the criteria given in ASME III Subsection NF.

3.9.1.4.5 Analyses of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop include the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is Seismic Category I and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3.9.1-2. The equipment is analyzed for (1) the normal loads of weight, pressure, and thermal; (2) mechanical transients of OBE, SSE, and pipe ruptures; and (3) pressure and temperature transients outlined in Subsection 3.9.1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analyses, a set of loads is determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformation is demonstrated between the actual plant loads and the loads used in the analysis of the components. Any deviations where the actual load is larger than the umbrella load is 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 analysis. The response spectrum corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analysis for the steam generator and pressurizer are performed using 2 percent damping for the OBE and 4 percent damping for the SSE. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump internals is performed using the damping for bolted steel structures, that is 4 percent for the OBE and 7 percent for the SSE (2 percent for OBE and 4 percent for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump, since the main flange, motor stand, and motor are all bolted assemblies (See Section 5.4). The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III. These valves are identified in Subsection 3.9.3.2.

3.9.1.4.6 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss-of-coolant accident

3.9.1.4.6.1 Introduction

This section presents the method of computing the reactor pressure vessel response to a postulated loss-of-coolant accident (LOCA). The structural analysis considers simultaneous application of the time-history loads on the reactor vessel resulting from the internal hydraulic pressure transients. The vessel is restrained by reactor vessel support pads and shoes beneath all six of the reactor vessel nozzles, and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

The limiting breaks for Shearon Harris considered forces produced by a rupture of the largest branch line piping, excluding the main (primary) coolant loop from consideration through crediting Leak-Before-Break as discussed in Section 3.6.2.1.1.1 for both cases: a cold leg branch line and a hot leg branch line. The branch lines analyzed for Shearon Harris are the accumulator line, RHR line and the pressurizer surge line. The methods of analysis adopted are related to the type of accident assumed (cold leg break or hot leg break). Note that throughout this description hot or cold leg break refers to where the broken branch lines were attached to the main coolant loop. The size of the postulated break, and its location along the primary loop piping is determined by the size and location of the branch line piping, as a result of crediting Leak-Before-Break exclusion of the main (primary) coolant loop for consideration with respect to LOCA forces.

In summary, three loss-of-coolant accident conditions were analyzed:

a) accumulator line break,

b) pressurizer surge line break, and

c) RHR line break.

3.9.1.4.6.2 Interface Information

Fuel mechanical properties were provided to Westinghouse by Siemens Power Corp. All other input information was developed within Westinghouse. This information includes: reactor internals properties, loop mechanical stiffness, internal hydraulic pressure transients, and reactor support stiffnesses. These inputs allowed formulation of the mathematical models and performance of the analyses, and are described in Sections 3.9.1.4.6.3 through 3.9.1.4.6.6.

3.9.1.4.6.3 Loading Conditions

Following a postulated pipe rupture at a primary system nozzle, the reactor vessel is excited by time-history forces. The internals reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For a cold leg break (an accumulator line break), the depressurization path is through the broken loop inlet nozzle and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, down and around the downcomer annulus and up through the fuel. In the case of a hot leg break (pressurizer surge line break or RHR line break) the wave passes through the RPV outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel. Thus for a hot leg break, the downcomer annulus is depressurized with smaller differences in pressure horizontally across the core barrel than for the cold leg break. For both hot and cold leg breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomenon has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP 8708 (Reference 3.9.1-3).

3.9.1.4.6.4 Reactor Vessel and Internals Modeling

The reactor vessel is restrained by two mechanisms: (1) the three attached reactor coolant loops with the steam generator and reactor coolant pump primary supports and (2) six reactor vessel supports, one beneath each reactor vessel nozzle. The reactor vessel supports are described in Section 5.4.14 and are shown in Figures 5.4.14-1 and 3.8.3-9. The support shoe provides restraint in the horizontal directions and for downward reactor vessel motion.

The reactor vessel model consists of a 3-dimensional finite element representation of the reactor vessel, supports, upper and lower internals and fuel. Non-linear gap effects at critical supports are taken into account.

3.9.1.4.6.5 Analytical Methods

The time history effects of the internals loads are applied to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown load and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

3.9.1.4.6.6 Results of the Analysis

As described, the reactor vessel and internals were analyzed for three postulated break locations. Table 3.9.1-4 summarizes the displacements and rotations of and about a point representing the intersection of the centerline of the nozzle attached to the leg in which the break was postulated to occur and the vertical centerline of the reactor vessel. Table 3.9.1-4 is historical information. Maximum displacements are no longer used. A detailed time history analysis is performed for equipment qualification.

The maximum loads induced in the vessel supports due to the postulated pipe break are given in Table 3.9.1-5. These loads are per vessel support and are applied at the vessel nozzle pad. It is conservatively assumed that the maximum horizontal and vertical loads occur simultaneously and on the same support, even though the time-history results show that these loads occur neither simultaneously nor on the same support. The largest vertical loads are produced on the support opposite the broken loop nozzle. The largest horizontal loads are produced on the supports which are the most perpendicular to the broken loop nozzle horizontal centerline.

3.9.1.4.7 Stress Criteria for ASME Code Class 1 Components and Component Supports

All ASME Code 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 were used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code with supplementary options outlined below:

a) Elastic system analysis and component inelastic analysis - This is an acceptable method of evaluation for faulted conditions if the rules of F-1323.1(a) are met for component supports, within the scope of Subsection NF and if primary stress limits for components are taken as greater of 0.70 S_u or S_y + 1/3 (S_u -S_y) for membrane stress and greater of 0.70 S_{ut} or S_y + 1/3 (S_{ut} -S_y) for membrane-plus-bending stress, where material properties are taken at appropriate temperature.

b) Elastic/inelastic system analysis and component/test load method - The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

ASME Code Class 1 component supports are analyzed for the design, normal, upset and emergency conditions in accordance with ASME III Subsection NF. The analysis and test methods and associated allowable limits used in the evaluation of faulted conditions are those defined in ASME III Appendix F. Although the component supports were designed and procured prior to the issuance of ASME III Subsection NF, the allowable stress criteria of NF 3000 (and Appendix F, by reference) have been applied in the analysis of the supports.

Loading combinations and allowable stresses for ASME Class 1 components and component supports are given in Tables 3.9.1-2 and 3.9.1-3. For faulted condition evaluations, the effects of the safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) are combined using the square root of the sum of the squares (SRSS) method. Justification for this method of load combination is contained in References 3.9.1-4 and 3.9.1-5.

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1 Vibration, Thermal Expansion, and Dynamic Effects Testing on Piping

3.9.2.1.1 Reactor Coolant System

A piping vibrational and dynamic effects testing program is conducted for the reactor coolant loop/supports systems during preoperational testing of the SHNPP. The purpose of these tests (summarized in paragraph 14.2.12.1.12) is to confirm that the system has been adequately designed and supported for vibration as required by Section III of the ASME Code, Paragraph NB-3622.3. The tests include reactor coolant pump starts and trips. If vibrations are observed which, from visual examination, appear to be excessive, either: 1) an instrumented test program will be conducted and the system reanalyzed to demonstrate that the observed levels do not cause ASME Code stress and fatigue limits to be exceeded; 2) the cause of the vibration will be eliminated; or 3) the support system will be modified to reduce the vibrations. Particular attention is provided at those locations where the vibrations are expected to be the largest for the particular transient being studied.

During initial heatup of the Reactor Coolant System, restraints, supports, and hangers will be observed for proper operation. Adequate clearances between piping and building structure will be verified.

The layout, size, and other design considerations of the reactor coolant loop and surge line piping used in the SHNPP are very similar to those employed in Westinghouse plants now in operation. The reactor coolant loop and surge line piping are adequately designed and supported to minimize vibration and allow thermal expansion. In addition, vibration levels of the reactor coolant pump, which is the only mechanical component that could cause vibration of the reactor coolant loop and surge line piping, are measured as discussed in Section 5.4.1.

3.9.2.1.2 Other Plant Piping Systems

A piping vibration and thermal expansion test program is conducted for the following systems or portions of systems as described in Regulatory Guide 1.68 (see Sections 14.2.12.1.12 and 14.2.12.1.44):

a) ASME Code Class 1, 2, and 3 systems.

b) All other high-energy piping systems inside Seismic Category I structures.

c) High-energy portions of systems in which failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level.

d) Seismic Category I portions of moderate-energy piping systems located outside Containment.

This test program satisfies the requirements of Paragraphs NC-3622.3 and ND 3622.3 of the ASME B&PV Code, Section III.

3.9.2.1.3 Test Program

The program includes a list of systems, or portions of systems, to be tested, and the flow modes and transients to which these systems are subjected. A list of selected locations for visual observation of vibration is included. Thermal expansion observation and measurement locations are provided, along with the applicable acceptance criteria.

The test program is prepared by the piping design organization or another qualified organization and approved by the SHNPP Superintendent-Start-up in accordance with the plant Start-up Manual. The program consists of the following types of tests:

a) Steady-State Piping Vibration Test

During the steady-state vibration test, qualified individuals familiar with the subject piping systems observe the lines during the normal mode of system operation. If, in their judgment, excessive vibration occurs, either: 1) an instrumented test program will be conducted and the system reanalyzed to demonstrate that the observed levels do not cause ASME Code stress and fatigue limits to be exceeded; 2) the cause of the vibration is eliminated; or 3) the support system will be modified to reduce the vibrations. Particular attention will be provided at those locations where the vibrations are expected to be the largest.

b) Piping Dynamic Response Tests

During the dynamic response tests, piping systems are subjected to routine transients, such as pump starts and trips (including reactor coolant pump), valve closure (including pressure-relieving and turbine trip valve), and control valve modulation, and are visually observed during such transients by qualified individuals. If evidence of excessive piping motion is observed, the lines will be reviewed by the design organization to determine the necessary corrective action.

c) Piping Thermal Expansion Tests

During the thermal expansion tests, pipe deflections are measured or observed at various locations based on the location of snubbers, hangers, and expected large displacements. One complete thermal cycle, i.e., cold position to hot position to cold position, is monitored. The objective of the thermal expansion test is to verify that the piping system is free to expand thermally (i.e., piping does not bind or lock at spring hangers and snubbers, nor interfere with structures or other piping), and that piping displacements do not exceed design stress limits. Thermal expansion tests are not conducted for systems with normal fluid temperatures less than 250 F.

d) Snubber Operability Tests

On lines subject to testing, pre-service examination will be made on all snubbers. This examination will be made after snubber installation but not more than six months prior to initial system pre-operational testing, and will as a minimum verify the following:

1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

3) Snubbers are not seized, frozen or jammed.

4) Adequate swing clearance is provided to allow snubber movement.

5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, and cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, reexamination of items 1, 4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements will be repaired or replaced and re-examined in accordance with the above criteria.

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250F will be verified as follows:

a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, observe the snubber expected thermal movement.

b) For those systems which do not attain operating temperature, visually confirm or calculate that the snubber will accommodate the projected thermal movement.

c) Observe the snubber swing clearance at specified heatup and cooldown intervals. Any discrepancies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

Where practical, the above tests are performed during the preoperational testing phase. Those tests that cannot be performed as part of the preoperational testing phase due to required plant conditions will be performed as part of the start-up and power escalation phase.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

The operability of Seismic Category I mechanical equipment is demonstrated as described below if the equipment is determined to be active, i.e., mechanical operation is relied on to perform a safety function. The operability of active ASME Code Class 2 and 3 pumps, active

ASME Code Class 1, 2, and 3 valves, and their respective drives, operators, and vital auxiliary equipment is shown by satisfying the criteria given in Section 3.9.3.2. Other active mechanical equipment is shown to be operable by 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 3.9.3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10 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 tests showing that the equipment retains its structural integrity under the simulated test environment.

A list of Seismic Category I equipment for NSSS and balance of plant and methods of qualification used are provided in Table 3.9.2-1 and Appendix 3.9A, respectively.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The program used to establish the integrity of reactor internals has evolved from extensive design analysis, model testing, and post-hot-functional inspection. Additionally, full-size reactors have been instrumented to measure dynamic behavior including a SHNPP size plant; and the measurements have been compared with predicted values.

The reactor instrumentation program was instituted as part of a basic philosophy of instrumenting the internals of the "first-of-a-kind" of the current nuclear steam supply system designs for power plants. These data provide added assurance of the adequacy of the internal design and assist in the development of increased capability for the prediction of the dynamic behavior of pressurized water reactor (PWR) internals. The "first-of-a-kind" plants that have been instrumented are R. E. Ginna (two loops), H. B. Robinson No. 2 (three loops), Indian Point Unit II (four loops), and Trojan (neutron panels and 17 x 17 style internals).

The H. B. Robinson No. 2 reactor has been established as the prototype for the Westinghouse three-loop plant internals verification program. Subsequent three loop plants are similar in design. Experience with other reactors indicates that plants of similar designs behave in a similar manner. For these reasons, the instrumentation program conducted on the H. B. Robinson Unit 2 internals will qualify the reactor internals at SHNPP.

The only significant differences between the SHNPP's internals and the H. B. Robinson No. 2's internals are the replacement of the annular thermal shield with neutron shield panels and the substitution of 17 x 17 fuel assemblies for 15 x 15 assemblies.

The replacement of the thermal shield with segmented neutron shield panels results in a reduction of the flow-induced vibrations of the reactor core structures. This conclusion was confirmed in tests with a 1/24 scale model (Reference 3.9.2-1). The flow test was first conducted on a model with a thermal shield and then on a model with neutron shield panels. The results indicated that the vibration levels of the internals were low and levels on the neutron shield panel were negligible. Reference 3.9.2-2 justifies in more detail the comparison of the relative effects of replacing the annular thermal shield with neutron shielding panels.

There is no change in the configuration of the reactor internals core support structures from the 15 x 15 fuel assembly configuration due to the incorporation of the 17 x 17 fuel assembly. The mechanical properties of the 17 x 17 fuel assembly, such as fuel assembly weight and beam stiffness, are virtually identical to the 15 x 15 fuel assembly; therefore, their input to the reactor internals core support structures is the same; and the response of the total reactor internals core support structural model will not change.

The remainder of the core structure design has not been changed, and consequently remains identical to the prototype which has been tested and proven to be well within design expectations and limits.

The Portland General Electric Company Trojan plant internals were instrumented for strain measurements on the core barrel, and on the 17 x 17 guide tube subject to highest cross flow. The Trojan plant was the lead plant featuring neutron panels and 17 x 17 style internals. The data obtained in this program provides verification of Westinghouse analysis and scale model predictions of 17 x 17 and neutron panel behavior in a full size plant and is applicable to the Shearon Harris Nuclear Power Plant.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

The Three Loop Internals Assurance Program conducted on H. B. Robinson No. 2, supplemented by the Trojan data on neutron panels and 17 x 17, jointly satisfy the intent of Regulatory Guide 1.20 (see Section 1.8).

The core support structures receive the normal before and after hot functional inspection by Westinghouse to satisfy regulatory position C.3.1 of Regulatory Guide 1.20. This inspection includes the points on revised Figure 3.9.2-1, summarized as follows:

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 internals components that are similar to those which were examined on the prototype H. B. Robinson No. 2 design.

   The inside of the vessel is 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.

e) Lower Internals - A particularly close inspection was made on the following items or areas using a 5X or 10X magnifying glass, where applicable. The locations of these areas are shown on Figure 3.9.2-1.

1) Upper barrel to flange girth weld.

2) Upper barrel to lower barrel girth weld.

3) Upper core plate aligning pin. Examine bearing surfaces for any shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.

4) Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

5) Baffle assembly locking devices. Check for lockweld integrity.

6) Lower barrel to core support girth weld.

7) Neutron shield panel screw locking devices and dowel pin cover plate welds. Examine the interface surfaces for evidence of tightness and 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 all the joints for tightness and soundness of the locking devices.

11) Secondary core support assembling welds.

12) Lower radial support keys and inserts (examine for any shadow marks, burnishing, buffing, or scoring. Check the integrity of the lockwelds). These members supply the radial and torsion constraint of the internals at the bottom relative to the reactor vessel while permitting axial growth between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing, or shadowing marks that would indicate pressure loading and relative motion between the two parts. Some scoring of engaging surfaces is also possible and acceptable.

13) Gaps at baffle joints. (Check for gaps between baffle and top former, and at baffle to baffle joints).

f) Upper Internals

A particularly close inspection was made on the following items or areas, using a magnifying glass of 5X or 10X magnification, where necessary. The locations of these areas are shown on Figure 3.9.2-1.

1) Thermocouple conduits, clamps, and couplings.

2) Guide tube, support column, and thermocouple column assembly locking devices.

3) Support column and conduit assembly clamp welds.

4) Upper core plate alignment inserts. Examine for any shadow marks, burnishing, buffing, or scoring. Check the locking devices for integrity of lockwelds.

5) Thermocouple conduit gusset and clamp welds.

6) Thermocouple end plugs. (Check for tightness.)

7) Guide tube enclosure welds, tube-transition plate welds and card welds.

Acceptance standards were the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals were subjected to a total operating time at greater than normal full flow conditions (three pumps operating) of at least 240 hours. This provides cyclic loading of approximately 107 cycles on the main structural elements of the internals. In addition, there was some operating time with only one and two pumps operating.

Since there were no signs of abnormal wear, no harmful vibrations were detected, and no apparent structural changes took place, the three-loop core support structures are considered to be structurally adequate and sound for operations.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Condition

The following events are considered in the faulted conditions category:

a) Loads produced by a pipe rupture of the reactor coolant loop for cold leg, cross-over leg, and hot leg breaks;

b) Response due to a safe shutdown earthquake (SSE); and

c) Combination of LOCA and SSE.

The analysis methods for reactor internals are discussed in Section 3.9.3, except for seismic analysis. Seismic analysis of the reactor internals is described in Section 3.7.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The dynamic behavior of reactor components has been studied, using experimental data obtained from operating reactors along with results of model test and static and dynamic tests in the fabricator's shops and at the plant site. Extensive instrumentation programs to measure vibration of reactor internals (including prototype units of various reactors) have been carried out during preoperational flow tests.

From scale model tests, information on stresses, displacements, flow distribution and fluctuating differential pressures is obtained. Studies have been performed (Reference 3.9.2-1) to verify the validity and to determine the prediction accuracy of models for determining reactor internals vibration due to flow excitation.

Vibration of structural parts during preoperational tests is measured using displacement gauges, and/or accelerometers, and/or strain transducers. The signals are recorded with magnetic tape recorders. Onsite-offsite signal analysis is done using both hybrid real-time and digital techniques to determine the approximate frequency and phase content. In some structural components, the spectral content of the signals include nearly discrete frequency or very narrow-band frequency, usually due to excitation by the reactor coolant pumps and other components that reflect the response of the structure at a natural frequency to broad band, mechanically or flow induced excitation. Damper factors are also obtained from wave analyses.

In general, the determination of internals responses proceeds as follows. Frequencies and spring constants are obtained analytically and these values are confirmed with test results. Theoretical and experimental studies have provided information on the added apparent mass of the water, which has the effect of decreasing the natural frequency of the component. Damping coefficients are established experimentally and forcing functions are characterized from previous studies including those discussed above. Once these factors are established, the response can be computed analytically.

In addition, the responses of important reactor structures are measured during preoperational reactor tests and the frequencies and mode shapes of the structures are obtained. Once all of the dynamic parameters are obtained as explained above, the forcing functions can be estimated. When combined, these studies provide indications of the internals behavior during reactor operation.

Pre- and post-hot functional inspection results, in the case of plants similar to prototypes, serve to confirm predictions that the internals are well behaved. Any gross motion or undue wear would be evident following the application of approximately 10^7 cycles of vibration expected during the test period.

3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

ASME Code Class 1, 2, and 3 system components are designed in accordance with the rules and methods specified in the ASME code. The design stress limits of the ASME code (including code cases) are selected to insure the pressure retaining integrity of safety class equipment. Code cases where utilized by the A/E have been approved by Regulatory Guide 1.84, "Code Case Acceptability - ASME III, Design and Fabrication," and 1.85, "Code Case Acceptability ASME III, Materials." Code cases utilized by the NSSS vendor are discussed in FSAR section 5.2.1.

Stress limits for A/E-supplied Class 2 and 3 components are described in section 3.9.3.1.2.2.

ASME code Class 2 and 3 components are designed for the concurrent loadings produced by pressure, deadweight, temperature distribution, the vibratory motion of the safe shutdown earthquake (SSE), and the dynamic system loadings associated with the appropriate plant faulted condition.

In addition to the 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 seismic events and pipe rupture. Two types of seismic loadings are considered: operational basis earthquake (OBE) and safe shutdown earthquake (SSE).

For the OBE loading condition, the Nuclear Steam Supply System is designed to be capable of continued safe operation. Therefore, for this loading condition critical structures and equipment needed for this purpose are required to operate within design limits. The seismic design for the SSE is intended to provide a margin in design that assures capability to shutdown and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that required critical structures and components do not lose their capability to perform their safety function. This is referred to as the "no loss of function" criteria and the loading condition as the "safe shutdown earthquake" loading condition.

Not all critical components have the same functional requirements for safety. For example, the Containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practices, generally elastic behavior of this structure under the "safe shutdown earthquake" loading condition must be ensured. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the later where the principal requirement is that they retain their contents and allow fluid flow.

The design loading combinations for specific plant operating conditions are listed in Tables 3.9.3-1 and 3.9.3-7 through 3.9.3-11. The specific criteria that provide the bases for design of a particular component are given in the specific sections that describe the corresponding fluid systems. The design pressure, temperature, and other design transients that are considered in the design of each mechanical component are also listed. The loads are combined for each component to insure that the severest combination is specified for non-faulted systems. The loads that may be imposed by a faulted system on a non-faulted system, i.e., fluid jet impingement and pipe whip impingement are considered separately. These effects are accounted for on a case by case basis and are described in Section 3.6.

It should be noted that for emergency and faulted conditions, the fundamental design criterion is that the integrity of the pressure boundary be maintained for non-faulted system piping, vessels and inactive components, and that non-faulted system active components maintain minimum required performance capability.

The design rules and associated design stress limits applied in the design of ASME code Class 2 and 3 components are in accordance with the ASME code, Section III, Subsections NC and ND, respectively. In those areas of design where the applicable rules of Subsection NC and ND are not explicit, the rules are supplemented as described herein, and in Tables 3.9.3-2 through 3.9.3-6 and Tables 3.9.3-8 through 3.9.3-11.

The full spectrum of plant process conditions is divided into four categories in accordance with their anticipated frequency of occurrence. The four categories--normal, upset, emergency, and faulted--are considered in the derivation of design loading combinations for those systems and components necessary to meet the design requirements for a specific plant process condition.

The operating condition categories are defined as follows:

a) Normal Condition - Except as noted in items b through d below, any condition in the course of system startup, operation in the design power range, hot standby, or system shutdown.

b) Upset Condition - Any deviations from normal condition anticipated to occur often enough that design should include a capability to withstand the conditions without operation impairment. The Upset Condition includes those transients caused by a fault in a system component requiring its isolation from the system, transients due to a loss of load or power, and any system upset not resulting in a forced outage.

The Upset Conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status.

c) Emergency Condition - Any deviations from Normal Condition which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence, but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not exceed 25 for the life of the plant.

d) Faulted Condition - 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 where considerations of public health and safety are involved.

Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities. Among the Faulted Conditions may be a specified earthquake for which safe shutdown is required.

The design criteria for faulted load conditions applied to engineered safety features outside the reactor coolant pressure boundary are intended to demonstrate component functional integrity. The design requires rigidity of active pumps and valves, and limits deflections to allowable clearances.

These requirements are demonstrated by dynamic analyses for each type of pump and valve used. These functional design requirements apply only to active components whose operability is relied upon to perform a safety function (as well as reactor shutdown) during the transient or event considered in the respective operating condition category.

Testing and/or analysis is representative of the combinations of seismic and operating stresses in order to assure component operability. Those components, once installed, are tested preoperationally as indicated in Chapter 14, and are also subject to periodic inspections and tests during the course of normal plant maintenance.

3.9.3.1.1 Loading Combinations, Design Transients and Stress Limits for ASME Code Class 1 Components and Supports

The ASME Code Class components are constructed in accordance with the ASME Code Section III. Table 3.9.1-3 lists stress criteria for ASME Class I components.

For faulted condition analysis of Class 1 piping systems, Appendix F of the ASME Code, Section III, permits the use of Equation (9) of subsection NB-3652 with a stress limit of 3.0 S_m. This criterion is presently being used by Westinghouse for Class 1 piping evaluation. In spite of the fact that Appendix F states that these limits are intended only to assure structural integrity, it has been proven by analysis as well as by testing per WCAP-9990, Structural Analysis of Reactor Coolant Loop for SHNPP that these limits do, in fact, provide sufficient assurance that the piping will not experience gross distortion such that the function of the system would be impaired. The basis for this position is as follows.

The primary stresses, as calculated by Equation (9) of NB 3652, result basically from two sources: the internal pressure in the piping during the event under consideration and bending moments imposed on the piping system. A postulated condition of gross distortion in the piping system, however, is only dependent upon the bending moment. Thus, it is important to investigate the maximum bending moment which would be permitted to exist in the piping system (via Equation [9] ≤ 3.0 S_m), under conditions encountered during a faulted condition event.

The maximum allowable bending stress in a piping system can be determined from Equation (9) by subtracting the pressure stress (calculated from the faulted condition pressure and temperature at the particular location of interest) from the maximum total allowable stress 3.0 S_m. For essential Class 1 piping systems of the Westinghouse NSSS, the maximum stress due to an applied bending moment (using Equation [9] ≤ 3.0 S_m) would be approximately 2.6 times the yield stress (S_y), taking into account the appropriate pressure stress. However, in over 95 percent of the essential NSSS Class 1 piping, the maximum bending stress would be less than 2.25 times the yield stress (S_y).

Finite element analyses of those dimensionally modelled elbows have been conducted. The elbows analyzed ranged in diameter from 1 1/2 in. to 14 in. with thickness to radius ratios ranging from 0.19 to 0.34. Large deformation considerations were complimented in the analysis in order to account for the effect of the ovalization of the pipe cross section on the moment carrying capacity of the elbow. Stress strain curves the elasto-plastic strain hardening properties are used to describe the behavior of the stainless steel material of the elbows. Pure bending moment is applied to the elbow via monotonically increasing rotations of the unconstrained ends of the elbow.

The moment rotation performance curves of the elbows are obtained and the percentage ovalization and change in flow area are calculated for all the loading stages. Prior to failure, the elbows experienced ovalization up to 45 percent and decrease in flow area as high as 35 percent. However, such values are never approached if the requirements of ASME code are met (i.e., equation (9) < 3.0 S_m). The highest ovalization produced by the moment given by Equation (9) < 3.0 S_m proved to be less than 2.00 percent and the maximum area change was less than 0.5 percent. For conservation, it was assumed that the entire allowable stress was due to pure bending which would produce the greatest distortion (Analysis Reference: NLS-84-389, dated 4-6-84).

As stated earlier, it is the bending moment that is of primary concern in the consideration of piping system collapse, i.e., the tendency to form a plastic hinge. Tests have been performed by Westinghouse on ten inch stainless steel pipe similar to that utilized in the Westinghouse NSSS. These tests indicated that the ratio of the plastic hinge moment to the bending moment that causes initial yielding in the pipe is approximately 3.2. Therefore, the ratio between the plastic hinge moment to the moment given by Equation (9) < 3.0 S_m is no less than (3.2/2.6) and for most of the RCS piping the ratio equals (3.2/2.25). Thus, the use of Equation (9) ≤ 3.0 S_m will limit bending moments in the piping system to values less than that which would be required to form a plastic hinge.

Also from the tests, it was observed that, at the point of initial plastic hinge formation, the angle of rotation of the pipe reached a maximum of less than 20 degrees. Even under this loading condition, there was no discernable distortion of the cross section of the pipe. Since the moment in the pipe will be limited to less than the plastic hinge moment, the rotation of the pipe would also be much less than observed in the tests. Therefore, a comparison of the moment necessary to form a plastic hinge to the maximum moment that would be permitted by Equation (9) demonstrates that Equation (9) sufficiently limits the bending moment in the pipe such that a plastic hinge will not be formed. Therefore, the pipe rotation will be limited such that gross distortion of the cross section will not occur.

In addition, Equation (9) applies stress indices based on pipe size and bend radius, to the stresses calculated for curved pipe (elbows), which, in effect, limit the stresses, and thus the applied moments in the pipe to less than those allowed for straight pipe. Using a procedure similar to that described above (determination of the maximum bending moment allowed by equation [9] $\leq 3.0\ S_m$), the maximum moment in any elbow of the essential lines of the Westinghouse RCS piping will be limited to a value which would produce a stress of approximately 1.5 S_y in an equivalent straight pipe. This is significantly lower than the moment allowed in a straight pipe by Equation (9) which resulted in a maximum stress of 2.6 S_y and which was shown to be less than the plastic hinge moment.

The above discussion demonstrates that when Equation (9) is used in conjunction with a stress limit of 3.0 S_m, the moments allowed by Equation (9), both in a straight pipe and in elbows, are sufficiently low such that the functional capability of the system is not impaired. (Test Reference: WCAP-7503, Supplement 1).

A detailed discussion of ASME Code Class 1 components is provided in Section 5.4.

A discussion of core support structures is provided in Section 3.9.5. These structures were designed and fabricated prior to application of Section NG of the ASME Code.

3.9.3.1.2 Loading combinations, design transient and stress limits for ASME Code Class 2 and 3 components and supports

3.9.3.1.2.1 NSSS - ASME Code Class 2 and 3 components

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9.3-1.

Limits for each of the loading combinations are component oriented and are presented in Tables 3.9.3-2 and 3.9.3-3 for tanks, Table 3.9.3-4 for inactive pumps, Table 3.9.3-5 for active pumps and Table 3.9.3-6 for valves. Inactive components are those whose operability is 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. Active pumps and valves are discussed in Section 3.9.3.2. Design of Class 2 and 3 supports is discussed in Section 3.9.3.4.

3.9.3.1.2.2 Non-NSSS Supplied ASME Code Class 2 and 3 Components

Design pressure, temperature and other loading conditions that provide the basis for design of fluid systems' Code Class 2 and 3 components are presented in the sections which describe the systems.

The design loading combinations for ASME Code Class 2 and 3 components are given in Table 3.9.3-7.

Stress limits for each of the loading combinations are component-related and are presented in Tables 3.9.3-8, 3.9.3-9, 3.9.3-10, and 3.9.3-11 for pumps and valves, tanks, pressure vessels, and piping, respectively. Stress limits for component supports are discussed in Section 3.9.3.4. Refer to Section 1.8 for a complete discussion of compliance with Regulatory Guide 1.48.

The austenitic stainless steel Class 2 and 3 piping systems were reviewed to identify the portions of piping required to deliver flow under faulted plant conditions. The highest stressed elbows which are representative of those portions of the systems were evaluated using the methodology and acceptance criteria of NEDO-21985. All elbows met the criteria. The current design basis is therefore sufficiently conservative to assure that functional capability is maintained under all plant conditions per the acceptance criteria of NUREG-0800, SRP 3.9.3, Appendix A.

For a description of the analysis performed for piping, refer to Section 3.7.3. Mathematical models used to analyze other equipment are discussed in Appendix 3.9A.

3.9.3.2 Operability Assurance Program

3.9.3.2.1 Introduction

The Pump and Valve Operability Assurance Program ensures the operability of safety related active pumps and valves as required by 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 1, 2, 4, 14, 30, and the quality assurance requirements of Appendix B of 10 CFR Part 50. The discussion of the Pump and Valve Operability Assurance Program (PVORT) included in this section is a historical description of the program that was compiled for operating license review. The HNP IST Program Plan (previously ISI-203) provides the basis for current and future operability assessments of safety related components. Pump and valve operability information is readily available from the following sources, which includes but is not limited to, seismic qualification reports, vendor drawings, vendor manuals, records management system and equipment databases. The site and corporate procedures which control the design change process ensures compliance with the aforementioned requirements for future design changes. The PVORT document and associated tables in Section 3.9.3 will not be updated. Operability is defined as the ability to perform all required safety functions over the entire range of service conditions, which may include normal, upset, emergency and faulted plant conditions.

Active pumps and valves are those components that require a mechanical motion in performing a safety function. Safety-related equipment is defined as equipment whose failure could prevent satisfactory performance of one or more of the following functions:

a) Emergency Reactor Shutdown

b) Containment Isolation

c) Reactor Core Cooling

d) Containment and Reactor Heat Removal

e) Prevent Significant Radioactivity Release

These safety functions are accomplished in either of the following ways:

Automatically - An electrical, pneumatic, or hydraulic source causes operation of the equipment.

Manually - The equipment must change state in order to allow a system safety function to be performed, e.g., administrative procedures which require that a valve be manually operated in order for the system to transfer to an alternate operating mode which performs a safety function. Manual valves used solely to isolate equipment for maintenance are not considered to be active.

Safety-related pumps and valves are classified as ASME Code Class 1, 2, and 3. Active pumps and valves are designed and tested in accordance with Section III of the ASME Boiler and Pressure Vessel Code. Permissible piping loads are restricted and testing and analysis are performed to confirm operability.

The list of safety-related systems is included in Section 3.2 (Table 3.2.1 1). Tables 3.9.3-12, 3.9.3-13, 3.9.3-14, and 3.9.3-15 contain a list of safety-related active valves and pumps. These tables were compiled for operating license review and are not updated. A list of safety related active pumps and valves is maintained through corporate and site procedures.

3.9.3.2.2 Methodology

The operability of active pumps and valves is assured by design and safety review during the construction and operational phases of the following items: (1) specification of safely conservative performance requirements and service conditions, (2) manufacturer's tests and/or analyses in accordance with ASME, IEEE and other codes and standards as modified by NRC guidance, if any, and (3) start-up and in-service testing as required for the safe operation of the plant. The equipment is also verified operational during its installed life by a surveillance and maintenance program. The loading combinations and design transients for safety-related active pumps and valves are described in Section 3.9.3.1.

The methods are utilized in the demonstration of operability are described below:

a) Each component is designed to be capable of performing its safety function(s) during and following design bases events (i.e., LOCA, MSLB, HELB, etc.). The design specification includes the applicable loading conditions and requires the manufacturer to perform analyses and/or tests to demonstrate operability.

b) Analyses and tests are used to demonstrate the operability of each component under applicable seismic and dynamic loadings. Seismic and dynamic qualification of mechanical and electrical equipment is described in detail in Section 3.10. Testing is used as the method for seismic qualification if the component is mechanically or structurally complex, such that its performance cannot be adequately predicted by analysis.

c) Each component is reviewed and inspected to assure compliance of critical parameters with specifications and drawings. This confirms that the design bases, environmental conditions, and functional requirements, which are listed in specifications and/or design drawings, are met (e.g., confirming that specified materials and processes are used, that wall thicknesses meet code requirements, and that fits and finishes meet the specification requirements). These parameters include design conditions and performance requirements, such as pressure, temperature, radiation, chemical spray, submergence, flow, pump head, available NPSH, pump speed, stall current, valve opening and closing times and maximum differential pressure.

d) Testing is performed to verify adequacy of as-built components. These tests (i.e., hydrostatic test, leakage rate test, etc.) confirm the operability of the pressure retaining capability, leakage characteristics, and structural integrity of each active pump and valve.

e) Pre-operational, start-up and periodic in-service testing in conjunction with a surveillance and maintenance program demonstrate operability readiness throughout the life of the plant.

3.9.3.2.2.1 Operability tests and analyses of active pumps (non-NSSS scope)

Factory tests for each pump verify that workmanship and materials are free from defects and that design and construction are satisfactory. Hydrostatic tests of pressure-retaining parts of each pump are performed at the factory. The manufacturer tests each pump over its full operating range and supplies performance curves indicating head, efficiency and power requirements at various capacities. In addition, tests demonstrating that performance meets the requirements of the equipment specification are run in the field.

The manufacturer is required to demonstrate the equipment's ability to perform its required function during and after the time it is subjected to the forces resulting from seismic conditions in combination with all other applicable loadings. The manufacturer provides documentation explaining his methods of seismic analysis or testing and the results for each piece of equipment supplied, based on criteria provided to manufacturers for seismic qualification of seismic Category I equipment. Detailed information on seismic qualification of active pumps is provided in Section 3.10. Pump motors which are safety-related, seismic Category I, are seismically qualified in accordance with IEEE 344-1975. Pump motors which are safety-related, seismic Category I, and are located in a harsh environment satisfy the requirements of 10 CFR 50.49.

The environmental qualification program for safety-related electrical and mechanical equipment is described in Section 3.11.

3.9.3.2.2.2 Operability Testing and Analysis of Active Valves (Non-NSSS Scope)

Safety-related valves are hydrostatically tested and tested for seat tightness. The manufacturer is required to demonstrate the equipment's ability to perform its required function during and after the time it is subjected to the forces resulting from seismic conditions in combination with all other applicable loadings. The manufacturer provides documentation explaining his methods of seismic analysis or testing and the results for the equipment supplied, based on criteria provided to manufacturers for seismic qualification of seismic Category I equipment. Detailed information on seismic qualification of active valves is provided in Section 3.10.

Electric operators and other electrical appurtenances required for safety operation are seismically qualified in accordance with IEEE 344-1971/1975 and satisfy the requirements in 10 CFR 50.49 for harsh environments.

The environmental qualification program for electrical and mechanical equipment is described in Section 3.11.

3.9.3.2.2.3 Operability Tests and Analysis of Active Pumps (NSSS Scope)

Active pumps are qualified for operability by first undergoing rigid tests prior to and after installation in the plant. The in-shop tests include: (1) hydrostatic tests of pressure retaining parts to 150 percent of the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature; (2) seal leakage tests; and (3) performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump parameters. Also monitored during these operating tests are bearing temperatures 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 testing, hot functional tests, and the required periodic inservice inspection and operational tests. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant.

In addition to these tests, active pumps will be qualified for operability by assuring that they will start up, continue operating, and not be damaged during the faulted conditions. The pump manufacturer will be required to show by analysis, correlated by tests, prototype tests, or existing documented data, that the pump will perform its safety function when subjected to loads imposed by the maximum seismic accelerations and the maximum faulted nozzle loads. It is required that testing or dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. This frequency is sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A pump with lowest natural frequency above 33 Hz is considered essentially rigid. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 2.1g in two orthogonal horizontal directions and 2.1g in the vertical direction acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. If rubbing or impact is predicted, prototype tests or existing documented data is used to demonstrate that the pump will not be damaged or cease to perform its design function; the effect of rubbing or impacting on pump operation is evaluated by comparison of the contacting surfaces to similar surfaces of pumps that have been or will be tested. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration.

The changes in operating rotor clearances caused by casing distortions due to these nozzle loads are considered. The maximum seismic nozzle loads combined with the loads imposed by the seismic accelerations are also considered in analysis of the pump supports and the calculated misalignment is shown to be less than that misalignment which could cause pump misoperation. The stresses in the supports are below those which ensure that support distortion is short duration (equal to the duration of the seismic event) and support elasticity is maintained.

Performing these analyses with the conservative loads stated assures that critical parts of the pump will not be damaged during the short duration of the faulted condition and that the reliability of the pump during post-faulted condition operation will not be impaired by the seismic event.

If a natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis and adjusted accelerations are determined using the same conservatisms contained in the accelerations used for "rigid" structures (2.1g orthogonal horizontal and vertical). The static analysis is then performed using the adjusted accelerations; the stress limits stated in Table 3.9.3-5 must still be satisfied.

To verify analytical techniques and provide data for correlation to analytical results, full assembly operability testing was performed on a Charging/Safety Injection Pump. The assembly consisted of an 11 stage centrifugal pump, speed increaser gear and a 600 HP induction motor mounted on a common baseplate typical of normal plant installation. Of all Westinghouse supplied NSSS active pump assemblies, this one was chosen as being most representative of the various design features of active pumps. The assembly was mounted on a shaker table such that triaxial seismic input could be simulated. A flow loop connected to the pump permitted full pump operation while special fixtures were fabricated to apply nozzle loads to the suction and discharge nozzles. Instrumentation including accelerometers, strain gauges, strain bolts, proximity probes and thermocouples were used to monitor the complete assembly during testing.

In general, the testing consisted of a preseismic resonance search, a preseismic pump head-flow characterization, five OBE's, four SSE's, pump head-flow characterization between seismic runs, a post-test resonance search and a post-test pump head-flow characterization. The pump was started prior to and during seismic testing without difficulty. As a result of the testing, no pump damage was visually observed or measured and the hydraulic characteristics remained within specific tolerances. It was concluded that the test pump assembly remained operational during and after a design basis seismic event.

The specific pump attributes (e.g., weight, RPM, gear ratio, full load current) of both the test unit and the pumps employed at Shearon Harris are compared in a Pump and Valve Operability report which includes a summary report for the testing performed on the charging pump assembly.

To complete the seismic qualification procedures, the pump motor is qualified for operation during the maximum seismic event. Any auxiliary equipment identified as vital to the operation of the pump or the pump motor and which is not proven adequate for operation by the pump or motor qualification will also be separately qualified by meeting the requirements of IEEE 344-1975, with the additional requirements and justifications outlined in Section 3.10.

The program described above gives the required assurance that the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings and therefore, will perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.2.4 Operability Testing and Analysis of Active Valves (NSSS Scope)

Safety-related active valves must also perform their safety-related function in times of an accident. Tests and analyses are conducted to provide assurance that these valves will operate during a seismic event.

The safety-related valves will be subjected to a series of stringent tests prior to service and during plant life. Prior to installation, the following tests are performed: shell hydrostatic tests to ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and operational tests to verify that the valves will open and close within the specified time limits when subjected to the design differential pressure. Cold hydro-tests, hot functional qualification tests, periodic inservice inspections, and period inservice operations are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant.

Active valves are designed in accordance with ASME B&PV Code Section III. To demonstrate structural integrity, an analysis of the valve extended structure is performed with static equivalent seismic SSE loads applied at the center of gravity of the extended structure. Class 1 valves will be designed/analyzed according to the rules of the ASME Code, Section III, NB-3500.

In addition to the preservice and inservice testing described above, full assembly valves, representative of each design type undergo testing to verify operability during simulated plant faulted conditions. Westinghouse, working in conjunction with the valve manufacturer, evaluates the various valve attributes (e.g., material composition, weight, wall thickness, size) and selects valves that are the most susceptible to seismic induced loads for testing. This permits extrapolation of demonstrated operational performance to other valves within the design family. The pump and valve operability report prepared for Shearon Harris identifies the tested valves and each plant specific valve for which a given tested valve is employed for qualification purposes. A comparative analysis is performed with conclusions drawn on acceptability. The test procedures are described below.

The valve assembly is mounted in a manner that conservatively represents typical valve installations. The valve assembly includes the operator and appurtenances normally attached to the valve assembly in service. The faulted condition nozzle loads are accounted for in either of two ways: (1) loads equivalent to the faulted condition nozzle loads are simultaneously applied to the valve (through its mounting) during the test or (2) by analysis, the nozzle loads are shown to not affect the operability of the valve. Operability of the valve during a faulted condition is demonstrated by satisfying the following criteria:

a) Active valves are designed to have the lowest natural frequency greater than 33 Hz.

b) The complete valve assembly extended structure is statically deflected by an amount equal to the deflection caused by the faulted condition accelerations. This is done by applying the appropriate loads representing these accelerations at the center of gravity of the extended structure in the direction that yields the greatest deflection. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

c) The valve is cycled while in the deflected position, and cycle times are recorded. This data is compared to similar data taken in the undeflected condition to evaluate the significance of any change.

d) Motor operators, and other appurtenances necessary for operation, are qualified with additional requirements and justifications as supplied in Section 3.10.

The accelerations which are used for the static valve qualification shall be equivalent, as justified by analysis, to 4.0g acting in two orthogonal horizontal directions and 4.0g vertical simultaneously. The piping designer must limit accelerations to these levels.

If the lowest natural frequency of the valve is less than 33 Hz, a dynamic analysis will be performed to determine the equivalent accelerations to be applied during the static test. The analysis will account for the amplification of the input acceleration by considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations will be determined using the same conservatisms contained in the 4.0g orthogonal horizontal/vertical accelerations used for "rigid" valves. The adjusted acceleration will then be used in the static analysis, and the valve operability will be assured by the methods outlined.

Valves that are safety-related but can be classified as not having an extended structure, such as check valves and safety valves, are considered separately. Check valves are characteristically simple in design, and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact, and there are no extended structures or masses whose motion could cause distortions that could restrict operation of the valves. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve since the valve disc is typically designed to be isolated from the body wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any body distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard design or analyses methods, the ability of the valves to operate is assured by the design features. The valves will also undergo the following: (1) stress analysis of critical parts which may affect operability, including the faulted condition loads, (2) in-shop hydrostatic test, (3) in-shop seat leakage test, and (4) periodic insitu valve exercising and inspection to assure functional ability of the valve.

Pressurizer safety valves will be qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves: (1) stress and deformation analyses of critical items that might affect operability for faulted condition loads, (2) in-shop hydrostatic and seat leakage tests, and (3) periodic insitu valve inspection. In addition, a static load equivalent to that applied by the faulted condition is applied at the top of the bonnet, and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capability during a seismic event.

Using these methods, all safety-related valves on the NSSS systems will be qualified for operability during a faulted event. The methods outlined above conservatively simulate the seismic event and assure that the active valves will perform their safety-related function. Alternate valve operability testing, such as dynamic vibration testing will be allowed if it is shown to adequately assure the faulted condition functional ability of the valve.

3.9.3.2.3 Implementation

The Operability Assurance Program has been implemented during the course of SHNPP design and construction phase by performance of the following activities:

a) Establish environmental, seismic and dynamic loading conditions applicable during all plant normal and accident conditions.

b) Specify the applicable codes, test and analysis procedures and equipment performance requirements.

c) Review and evaluate manufacturer's plans, drawings, test/analysis programs and results with respect to demonstrating operability.

d) Audit manufacturer's facilities and equipment to verify that they meet the Quality Assurance requirements of 10 CFR 50, Appendix B.

Operability of active pumps and valves is determined based on inclusion of the following aspects:

a) Verification that appropriate manufacturers' tests and/or analyses have been conducted, reviewed, and approved, and the equipment conforms to the design requirements.

b) Significant aging mechanisms are addressed. For example, the review identifies the usable life of the mechanical equipment's non-metallic components based on normal and accident temperature and radiation conditions. Refer to Section 3.11.

c) When qualification testing is performed, test sequences are identified.

d) Margins are identified. The service conditions and conditions used during tests and analyses are addressed.

e) The results of tests and/or analyses are reported for each component.

f) Operability analyses are supported by test documents. The documentation identifies qualification results of previous similar equipment, or justifies sealing from models, where applicable.

g) The equipment is usually tested and/or analyzed as an assembly. However, there are tests/analyses of separate components for some equipment provided operability of the assembly is adequately indicated.

h) The manufacturers recommended maintenance items are identified and considered in the maintenance program. In addition, the usable life of the equipment's non-metallic materials is determined based on environmental conditions.

i) Safety-related pumps and valves are included in the SHNPP Initial Test and Inservice Inspection Programs as described in FSAR Sections 14.2 and 3.9.6, respectively.

j) Adequate and easily retrievable documentation is maintained containing operability information and review results.

k) Documentation contains sufficient references to trace the history of the qualified equipment from the design and purchase specifications through to its operability demonstration, installation, maintenance, and in-service tests.

A review was performed for each active pump and valve. The main purpose of this review was to confirm the adequacy of the equipment structural and operational integrity under normal, accident and post-accident conditions. The review involved three major areas of the specific equipment application in the plant; namely, general component information, equipment function, and equipment qualification. The reference material consulted to complete the review includes the following:

a) Design Specifications

b) Design Drawings from Vendor

c) Manufacturers Test and Analysis Reports

d) Instruction Manuals and Catalogs

e) Valve List and Line List

f) System Flow Diagrams

g) Equipment Specifications

h) FSAR

i) Design Calculations

j) Vendor Correspondence

k) Plant Operations Manuals

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

3.9.3.3.1 Non-NSSS Supplied Pressure Relief Devices

Pressure-relieving devices installed for the protection of ASME Class 2 and 3 components are designed and installed in accordance with the requirements of ASME Section III.

The steady state load due to reaction force from the opening and subsequent venting of a safety or relief valve includes consideration of both momentum and pressure effects and is computed by the formula:

$$F = \frac{W}{g_c} V_e + (P_e)A$$

where:

F	=	reaction force (lb. force)
W	=	mass flowrate (relieving capacity stamped on the valve x 1.11) (lbm/sec.)
g_c	=	dimensional constant, (32.2 lb. - mass ft./lbf-sec.^{2})
V_e	=	exit velocity, (ft./sec.)
P_e	=	static gage pressure at exit (lbf/in.^{2})
A	=	exit flow area, (in.^{2})

Dynamic effects associated with sudden application of the relief valve force are accounted for by a dynamic load factor which is determined as follows:

a) A value of 2 as permitted by Regulatory Guide 1.67 (see Section 1.8), or

b) Time history dynamic analysis of the system may be performed using system dynamic characteristics and expressing relief valve force as a function of time reaching its steady state valve based upon the valve opening time.

For Class 2 and 3 piping, the moment due to F (including the dynamic load factor) is included in the MB term of Equation 9 of Section NC-3652.2 of ASME Section III to calculate stresses at the header-valve inlet nozzle junction.

Fabrication and installation of the valve inlet nozzle to the header are in full compliance with the applicable provisions of ASME Section III for branch connections. Stresses in these pipes, including the effects of valve discharge thrust, are maintained within code limits.

For any pipe run having more than one safety/relief valve, the most severe combination of relief valves discharging simultaneously, including all valves on one side of the system, is considered in determining pipe stresses.

For closed systems where the fluid is discharging from a safety-relieving device to another vessel or chamber, the dynamic interaction forces of the effects of this loading are included in the MB term of equation 9 of Section NC-3652 of ASME Section III.

The fluid induced forcing functions were obtained from valve manufacturer and calculated using one-dimensional equations for the conservation of mass, momentum, and energy. These forcing functions are applied at locations along the piping system where change in fluid flow direction occurs. A dynamic load factor of 2 is used for static analysis if dynamic analysis is not performed.

Pressure relieving devices have been constructed, located, and installed so that they are readily accessible for inspection and repair and so that they cannot be readily rendered inoperative. Safety or relief valves have been set to relieve at a pressure not exceeding the design pressure of the vessel at the design temperature.

For a description of the computer programs used in the analyses, refer to Section 3.9.1.2.

3.9.3.3.2 NSSS Supplied Pressurizer Safety and Relief Devices

The pressurizer safety and relief valve discharge piping systems provide overpressure protection for the RCS. The three spring-loaded safety valves, located on top of the pressurizer, are designed to prevent system pressure from exceeding design pressure by more than 10%. The three power-operated relief valves, also located on top of the pressurizer, are designed to prevent system pressure from exceeding the normal operating pressure by more than 100 psi. They are also utilized for reactor vessel low temperature overpressure protection and in the mitigation of an SGTR event. A water seal is maintained upstream of each valve to minimize leakage. Condensate accumulation on the inlet side of each valve prevents any leakage of hydrogen gas or steam through the valves. The valve outlet side is sloped to prevent the formulation of additional water pockets.

The pressurizer safety valves, manufactured by Crosby, are self-actuated spring loaded valves with backpressure compensation. The power-operated relief valves, manufactured by Copes Vulcan, are air operated globe valves, capable of automatic operation via high pressure signal or remote manual operation. The safety valves and relief valves are located in the pressurizer cubicle and are supported by the attached piping.

When the pressurizer pressure reaches the set pressure (2500 psia for safety valve and 2350 psia for relief valve) and the valve opens, the high pressure steam in the pressurizer forces the water in the water loop seal through the valve and down the piping system to the pressurizer relief tank. For each pressurizer safety and relief piping system, an analytical hydraulic model is developed to represent the condition described above. The piping from the pressurizer nozzle to the relief tank nozzle is modeled as a series of control volumes and flow paths. The model development and analysis of the pressurizer power operated relief valve and safety valve discharge piping is described in a report submitted by Ebasco in response to Safety Evaluation Report Confirmatory Item No. 6 in accordance with Item II.D.1 of NUREG-0737. All three safety valves are assumed to open simultaneously while the relief valves remain closed. Similarly, the relief valves open simultaneously while the safety valves are closed. A description of the model development and the results of the analysis is contained in a report which was submitted under separate cover (Reference 3.9.3-1), and evaluated by the NRC. (Reference 3.9.3-2.)

Fluid acceleration inside the pipe generates reaction forces on all segments of the line which are bounded at either end by an elbow or bend. Reaction forces resulting from fluid pressure and momentum variations are calculated by Ebasco in the above mentioned report. These forces are expressed in terms of the fluid properties for the transient hydraulic analysis. Unbalanced forces are calculated for each straight segment of pipe from the pressurizer to the relief tank. The time histories of these forces are used for the subsequent structural analysis of the Class 1 portion of the pressurizer safety and relief lines performed by Westinghouse.

The mathematical model used by Westinghouse in the seismic analysis of the safety and relief lines is modified for the valve thrust analysis to represent the safety and relief valve discharge. The time history hydraulic forces are applied to the piping system lump mass points. The dynamic solution for the valve thrust is obtained by using a modified predictor-corrector-integration technique and normal mode theory.

The time history solution is performed in subprogram FIXFM. The input to this subprogram consists of the natural frequencies and normal modes, and applied forces. The natural frequencies and normal modes for the modified pressurizer safety and relief line dynamic model are determined with the WESTDYN program. The support loads are computed by multiplying the support stiffness matrix and the displacement vector at each support point. The time history displacements of the FIXFM subprogram are used as input to the WESTDYN2 subprogram to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation, the displacements are treated as imposed deflections on the pressurizer safety and relief line masses. The results of this solution are included in the design and faulted load combinations as well as the fatigue evaluation.

The loading combinations considered by Westinghouse in the analysis of the Class 1 portion of the pressurizer safety and relief system piping are given in Table 3.9.1-2a. These load combinations are consistent with the final recommendations of the piping subcommittee of the EPRI PWR PSARV performance test program.

The computer programs used in the pressurizer safety and relief line analysis are discussed in Section 3.9.1.2. For additional discussion of valve operability and overpressure protection during Inadvertent Operation of ECCS and a Main Feedline Break, See Section 5.2.2.2.

3.9.3.4 Components Supports

See Section 3.9.1 for ASME Code Class 1 component supports which are supplied by the NSSS Vendor. Class 2 and 3 supports have been designed as follows:

3.9.3.4.1 NSSS - Supplied Component Supports

a) Linear Supports for Tanks and Heat Exchangers

    1) Normal - The allowable stresses of A.I.S.C.-69 Part 1 are employed for normal condition allowables.

2) Upset - Stress limits for upset conditions are 33 percent higher than those specified for normal conditions. This is consistent with Paragraph 1.5.6 of A.I.S.C.-69 Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

3) Emergency - Not applicable.

4) Faulted - Stress limits for faulted condition are the same as for the upset condition.

b) Plate and Shell Supports for Tanks and Heat Exchangers

1) Normal - Normal condition limits are those specified in ASME Sec. VIII, Division 1 or AISC-69 Part 1.

2) Upset - Stress limits for upset condition are 33 percent higher than those specified for normal conditions. This is consistent with Paragraph 1.5.6 of AISC Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

3) Emergency - Not applicable.

4) Faulted - Stress limits for faulted condition are the same as for the upset condition.

c) Plate and Shell Supports for Pumps - The stress limits used for ASME Code Class 2 and 3 plate and shell component supports are identical to those used for the supported component. These allowable stresses are such that the design requirements for the components and system structural integrity are maintained.

3.9.3.4.2 Non-NSSS supplied component supports

Loading combinations and design transients applicable to the design of component supports are discussed in Section 3.9.3.1.

The corresponding stress limits applied to the design of component supports are as specified in Table 3.9.3-7a. The supports for active components are tested and/or analyzed as discussed in Section 3.9.2 and this Section.

Mechanical snubbers are constructed in accordance with the requirements of ASME III, Subsection NF, and comply with the rules of Subsection NF for the materials, design, fabrication, examination and testing of component supports. Code stamping of the snubbers is not required. The applicable code is the 1977 Edition, with Addenda through Summer 1978, for snubbers purchased after February 1980.

Hydraulic snubber assemblies are constructed in accordance with the requirements of ASME III, Subsection NF, 1980 Edition, including Addenda through Winter 1981 and comply with the rules of Subsection NF for materials, design, fabrication, examination, and testing of component supports. ASME III, Subsection NF, 1998 Edition, including Addenda through 2000 is also applicable for hydraulic snubber assemblies. Additionally, hydraulic snubbers are constructed to the requirements of ASME Code Cases in accordance with Regulatory Guide 1.84 and 1.85. However, no ASME Code Stamp is required. The hydraulic snubbers are not required to function during any steady state condition. NOTE: For individual snubber parts of hydraulic snubbers, Section III, Subsection NF, 1980 Edition, including Addenda through Winter 1981 is applicable.

3.9.4 CONTROL ROD DRIVE SYSTEM (CRDS)

3.9.4.1 Descriptive Information of CRDS

Control rod drive mechanisms (CRDM) are located on the dome of the reactor vessel. They are coupled to rod cluster control assemblies which have absorber material over the entire length of the control rods. The control rod drive mechanism is shown in Figure 3.9.4-1 and schematically in Figure 3.9.4-2.

The primary function of the CRDM is to insert or withdraw rod cluster control assemblies within the core to control average core temperature and to shutdown 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 five separate subassemblies. They are the pressure vessel, seismic sleeve assembly, 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 latch housing is the lower portion of the pressure vessel and contains the latch assembly. The rod travel housing is the upper portion of the pressure vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the reactor core.

b) The seismic sleeve assembly includes the seismic sleeve, pilot cap, and socket-head set screws. The seismic sleeve assembly mounts on top of the rod travel housing assembly and is anchored by the set screws.

c) 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.

d) 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 which incorporate a 1/16 in. lift action in their operation.

   The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8 in. steps by the lift coil. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8 in. step.

e) The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

   The drive rod has 5/8 in. grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and 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 in./min. Withdrawal of the rod cluster control assembly is accomplished by magnetic forces while insertion is by gravity.

The mechanism internals are designed to operate in 650 F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650 F and 2500 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 3.9.4-2 withdraws and inserts a rod cluster control assembly as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation, the stationary gripper coil of the drive mechanism holds the rod cluster control assembly in a static position until a stepping sequence is initiated, at which time the movable gripper coil and lift coil is energized sequentially.

Rod Cluster Control Assembly Withdrawal

The rod cluster control assembly is withdrawn by repetition of the following sequence of events (refer to Figure 3.9.4-2).

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 in. 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 in. 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 in. gap between the movable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 in.).

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 in. The 1/16 in. 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 in. 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 in. for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute and the drive rod assembly (which has a 5/8 in. groove pitch) is raised 72 grooves per minute. The rod cluster control assembly is thus withdrawn at a rate up to 45 in. per minute.

Rod Cluster Control Assembly Insertion

The sequence for rod cluster control assembly insertion is similar to that for control rod withdrawal, except the timing of lift coil (c) ON and OFF is changed to permit lowering the control assembly.

a) Lift Coil (C) - ON - The 5/8 in. 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 in. 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 rod cluster control assembly, causes the stationary gripper latches and plunger to move downward 1/16 in. until the load of the drive rod assembly and attached rod cluster control assembly is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

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 rod cluster control drop down 5/8 in.

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 in. The 1/16 in. 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 rod cluster control assembly withdrawal, up to 72 times per minute which gives an insertion rate of 45 in. per minute.

Holding and Tripping of the Control Rods

During most of the plant operating time, the control rod drive mechanisms hold the rod cluster control assemblies withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached rod cluster control assemblies hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the rod cluster control assembly plus the stationary gripper return swing is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by weight acting upon the latches. After the rod cluster control assembly is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9.4.2 Applicable CRDS Design Specification

The design of the reactivity control components takes into consideration temperature effects, thermal clearances, stress on structural membranes resulting from normal and accident conditions, and material compatibility.

The latch housing and the rod travel housing are the pressure containing components of the CRDM. They are designed in accordance with the requirements of the ASME Code Section III for Class 1 vessel appurtenance. The latch assembly, the drive rod assembly and the coil stack assembly are non-pressure containing components, classified as ANS non-nuclear safety. The design and testing of these components is discussed in the following sections.

3.9.4.3 Design Loads, Stress Limits, Allowable Deformation

1. Pressure Containing Components

The CRDM pressure containing components are designed to withstand loads originating from normal, upset, emergency and faulted conditions and to confirm the ability to trip when subjected to the seismic disturbance.

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 are listed in Table 3.9.1-1

j) Pump overspeed

k) Seismic loads (Operating Basis Earthquake and Safe Shutdown Earthquake)

l) Blowdown forces (due to cold and hot leg break)

The main objective of the analysis is to satisfy allowable stress limits, identified in the ASME Code Section III, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials.

2. Non-Pressure Retaining Components

The latch assembly and the drive rod assembly are designed for a minimum operating life, without refurbishment or replacement, of two and a half million (2.5 x 10^6) steps corresponding to approximately five thousand five hundred (5500) full travel excursions, and four hundred (400 reactor trips.

For faulted conditions, the design of the latch assembly and the drive rod assembly confirm the ability to trip the control rod since it is not possible, for any postulated latch assembly malfunction, to cause engagement of gripper latches with the drive rod. All postulated failures of the drive rod assembly, either by fracture or uncoupling, lead to a reduction in reactivity. If the drive rod fractures at any elevation, the portion remaining coupled to the control rod falls in the core, guided by the absorber rodlets.

3. Results of Dimensional and Tolerance Analysis

With respect to the control rod drive mechanism 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 control rod drive mechanism in these four critical areas. These clearances have been proved by lift 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 are 0.011 in. At the maximum design temperature of 650 F minimum clearance is 0.0045 in. and at the maximum expected operating temperatures of 550 F is 0.0057 in.

Latch Arm - Drive Rod Clearances

The control rod drive mechanism incorporates a load transfer action. The movable or stationary gripper latch is not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9.4-3 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9.4-4 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearance of the coil stack assembly over the latch housing was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70 F, the inside diameter of the coil stack is 7.308/7.298 in. The outside diameter of the latch housing is 7.260/7.270 in.

Thermal expansion of the mechanism due to operating temperature of the control rod drive mechanism results in minimum inside diameter of the coil stack being 7.440 in. at 222°F and the maximum latch housing diameter being 7.426 in. at 650°F.

Under the extreme tolerance conditions listed above it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

To confirm the above coil stack thermal clearances, four coil stack assemblies were removed from four hot control rod drive mechanisms mounted on 11.035 in. centers on a 550°F test loop, allowed to cool, and then replaced without incident.

Coil Fit in Core Housing

CRDM and coil housing clearances are selected to that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot control rod drive mechanism.

3.9.4.4 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 (Reference 3.9.4-1).

To confirm the mechanical adequacy of the fuel assembly, the control rod drive mechanism, and full length rod cluster control assembly, functional test programs have been conducted on a full scale 12 ft. control rod. The 12 ft. prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test the control rod drive mechanism was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive-line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the full length control rod drive mechanisms meets the design requirement of 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry. This trip time requirement will be confirmed for each control rod drive mechanism prior to initial reactor operation and at periodic intervals after initial reactor operation as required by the Technical Specifications.

There are no significant differences between the prototype control rod drive mechanisms and the production units. Design materials, tolerances and fabrication techniques (see Section 4.2.3) are the same.

These tests have been reported in Reference 3.9.4-1.

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 (refer to the Technical Specifications). Latch assembly inspection is recommended after 2.5 x 10^6 steps have been accumulated on a single control rod drive mechanism.

If a rod cluster control assembly cannot be inserted, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one rod cluster control assembly can be tolerated. More than one inoperable rod cluster control assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable rod cluster control assemblies has been limited to one as discussed in the Technical Specifications.

In order to demonstrate proper operation of the control rod drive mechanism and to ensure acceptable core power distributions during rod cluster control assembly, partial-movement checks are performed on the rod cluster control assemblies (refer to 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 beginning of decay of stationary gripper coil voltage to dashpot entry.

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 in. axial movement Load Transfer 0.047 in. nominal axial movement
Cold (ambient) performance test at design load	Operating Speed 45 in./min.
5 full travel excursions	Trip Delay Free fall of drive rod to begin within 150 msec

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

3.9.5.1 Design Arrangements

The reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct reactor coolant flow past the fuel elements, direct reactor coolant flow to the pressure vessel head, provide gamma and neutron shielding, and guides for the incore instrumentation.

The reactor 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 reactor coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and through the vessel outlet nozzles. A small portion of the reactor 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 3.9.5-1. 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 reactor 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 16 in. thick core support casting which forms the bottom of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate reactor coolant distribution is obtained through the use of the lower core plate and core support forging.

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 in. wide by 148 in. long by 2.8 in. thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guides by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in Reference 3.9.5-1.

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 forging and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, reactor coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an inconel clevis block is welded to the vessel inner diameter. Another inconel insert block is bolted to each of these blocks and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam supported at the top and the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. 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.

<u>Upper Core Support Assembly</u>

The upper core support assembly, shown in Figures 3.9.5-2 and 3.9.5-3 consists of the upper support plate assembly, and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the upper 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 upper support plate and are restrained by pins in the upper core plate for proper orientation and support.

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 lower 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 is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the upper support plate assembly and then the reactor vessel head. Transverse loads from reactor coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support plate and upper core plate. The upper 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 7.7.1-9 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 in. and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 ft. 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 ft. above the seal line is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. These are the only conditions which affect the incore instrumentation support structure.

3.9.5.2 Design Loading Conditions

The following list of loading conditions provides the basis for the design of the reactor internals:

Fuel and reactor weight, fuel and core component spring forces including spring preloading, differential pressure and coolant flow forces, temperature gradients, vibratory loads including operating basis earthquake seismic loads, thermal transient listed in Table 3.9.1-1, control rod trip, loop out of service loads, loss of load and pump overspeed, loss-of-coolant accident, loads due to steam break and loss of flow and loads due to safe shutdown earthquake.

The combination of these loads results in the various loading categories listed in the following paragraph.

3.9.5.3 Design Loading Categories

The following paragraph categorizes the design loading combinations into the various design loading categories adopted by the ASME B&PV code (Normal, Upset, Emergency, and Faulted).

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 operating basis earthquake seismic loads.

f) Normal and upset operational thermal transients listed in Table 3.9.1 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 loss-of-coolant accident.

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 loss-of-coolant accident.

b) Safe Shutdown Earthquake.

The design loadings are combined to fit into either the normal, upset, emergency, or faulted condition similar to those defined in the ASME Code Section III. These combinations follow the intent of the "Hopper diagrams" as indicated by Figures NG-3221.1, NG-3224.1, and by Appendix F of the ASME Code Section III Rules for Evaluating Faulted Conditions.

3.9.5.4 Design Bases

Design Bases Description

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3.9.1-1.

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 the loss-of-coolant accident plus the safe shutdown earthquake condition, the defection criteria of critical internal structures are the limiting values given in Table 3.9.5-1. The corresponding no loss of function limits are included in Table 3.9.5-1 for comparison purposes with the allowed criteria.

The criteria for the core drop accident is based upon analyses which 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 in. An additional displacement of approximately 3/4 in. 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 in. which is insufficient to permit the tips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless 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.

The design bases for the mechanical design of the reactor vessel internals components are as follows:

a) The reactor internals in conjunction with the fuel assemblies shall direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the head transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

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 have been 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 operating basis earthquake, safe shutdown earthquake and pipe ruptures and meet the requirement of item (e) below.

e) The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

f) Following the design basis accident, the plant shall be 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 3.9.5-1. 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 3.9.5-1.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9.2.

The basis for the design stress and deflection criteria is identified below:

Allowable Stresses

For normal operating conditions the intent of Section III, Subsection NG of the ASME Nuclear Power Plant Components Code is used as a basis for evaluating acceptability of calculated stress. Both static and alternating stress intensities are considered.

The allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design basis accident used for the SHNPP reactor internals is based on the 1973 draft of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions. Stress categories and limits are indicated in Figure NG-3221-1, Normal and Upset Conditions, and NG-3224-1, Emergency Conditions. Rules for evaluating faulted conditions are provided in Appendix F of the ASME Code, Section III.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

Inservice testing of pumps and valves at SHNPP is controlled by HNP IST Program Plan. This IST Program Plan establishes testing requirements to assess the operational readiness of certain ASME Code Class 1, 2, and 3 (see Section 3.2.2) pumps and valves that are required to:

1. Shut down the reactor to the safe shutdown condition, or
2. Maintain the reactor in the safe shutdown condition, or
3. Mitigate the consequences of an accident.

The IST Program Plan is conducted in accordance with the OM Code of the ASME Code for Operation and Maintenance of Nuclear Power Plants required by 10 CFR 50.55a(f) and listed in the plant Technical Specifications.

The IST Program Plan is submitted for review to the NRC.

3.9.6.1 Inservice Testing of Pumps

The preservice pump test program was conducted in accordance with Subsection IWP of ASME Section XI, 1980 Edition, including the addenda through the Winter, 1981. New reference values are established per the OM Code requirements in Section 3.9.6.

The IST Program Plan lists the pumps subject to testing and indicates the test parameters to be measured.

3.9.6.2 Inservice Testing of Valves

The preservice valve test program was conducted in accordance with Subsection IWV of the ASME Section XI, 1980 Edition, including the addenda through the Winter, 1981. New reference values are established per the OM Code requirements in Section 3.9.6.

The IST Program Plan lists the valves subject to testing and indicates the test parameters to be measured.

3.9.6.3 Relief Requests

Requests for relief from the OM Code of the ASME Code for Operation and Maintenance of Nuclear Power Plants requirements are submitted in the Pump and Valve Test Program. Information provided describes the specific area of relief requested, explains why compliance with the OM Code is impractical, and describes alternative test procedures.

A continuing program of radiation surveys during the refueling programs will be performed to ensure that any possible future problem areas are detected at an early stage. Should additional experience in the maintenance and inspection of operating plants indicate that areas exist where access will be either limited or impossible, requests for relief from the OM Code requirements will be made.

REFERENCES: SECTION 3.9

3.9.1-1 "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, Provision 1, May 1977.

3.9.1-2 "Sample Analysis of a Class 1 Nuclear Piping System," prepared by ASME Working Group on Piping, ASME Publication, 1972.

3.9.1-3 "Takeuchi, I., et al., "Multiflex - A Fortran-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP 8708-P-A/WCAP-8709A (September 1977).

3.9.1-4 Witt, F. J., Bamford, W. H., Esselman, T. C., "Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," WCAP-9283, March 1978.

3.9.1-5 Bogard, W. T., Esselman, T. C., "Combination of Safe Shutdown Earthquake and Loss-of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants, WCAP-9279, March 1978.

3.9.1.6 "Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program," WCAP-8929, June 1977.

3.9.2-1 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.

3.9.2-2 Bloyd, C. N., Ciaramitaro W., Singleton, N. R., "Designs Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan Power Plant," WCAP-8780 (Non-Proprietary), May 1976.

3.9.3-1 "Analysis of Pressurizer Power Operated Relief Valve and Safety Valve Discharge Piping," dated April 1984 (transmitted via NLS 84 263, dated June 28, 1984 and

Supplemental Reports NLS 86 197 dated June 1986, NLS-86-250 dated July 1986, and NLS-87-167 dated September 1987.)

3.9.3-2 NRC letter, R. A. Becker to L. W. Eury, May 31, 1989, "Evaluation of CP&L's Shearon Harris plant specific submittals in response to NUREG 0737, TMI Action Plan Requirement, Item II.D.1."

3.9.3-3 Westinghouse, CN-CSE-99-83, Determination of Primary Equipment Support Footprint Loads due to SG Replacement and Upgrading, Rev. 3.

3.9.3-4 Westinghouse, CN-CSE-99-87, Primary Equipment Support Qualification due to SG Replacement and Upgrading, Rev. 0.

3.9.3-5 Westinghouse, Letter CQL-92-523, ET-NSL-OPL-11-92-545, 11/20/92, Justification for Continued Operation for Steam Generator Snubber Operability.

3.9.4-1 Cooper, F. W. Jr., "17 x 17 Drive Line Components Tests - Phase 1B 11, 111, D-Loop-Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (non-Proprietary), December, 1974.

3.9.5-1 Kraus, S., "Neutron Shielding Pads," WCAP-7870, May, 1972.

3.9.5-2 Fabic, S., "Computer Program WHAM for Calculation of Pressure Velocity, and Force Transient in Liquid Filled Piping Networks," Kaiser Engineers Report No. 67-49-R, 1967.

3.9.5-3 Bohm, G. J. and LaFaille, J. P., "Reactor Internals Response Under a Blowdown Accident," First Intl. Conf. on Structural Mechanics in Reactor Technology, Berlin, September 20-24, 1971.

APPENDIX 3.9A SEISMIC CONSIDERATIONS FOR EQUIPMENT SPECIFICATIONS

For Seismic Category I equipment and supports, the vendor must demonstrate the equipment's ability to perform its required function during and after the time that it is subjected to the forces resulting from the seismic conditions. This can be accomplished in various ways. Two methods commonly used are to:

a) Predict the equipment's performance by mathematical analysis, or

b) Test the equipment under simulated seismic conditions.

If properly justified, other methods may be used. The documentation provided for the equipment must clearly justify the choice of analysis method.

All Seismic Category I, safety-related equipment and the qualification programs are summarized in a seismic and dynamic qualification and status list (SQRT Master List). The SQRT Master List is submitted to the NRC under separate cover. The format of this list is shown in Figure 3.10.1-1.

Mathematical Analysis Method

This method should be used for equipment which can be modeled to predict its response.

The analysis method should consist of the following:

a) Model the equipment as multidegree of freedom discrete lumped mass system interconnected with mass free springs, and with sufficient mass points to ensure adequate representation.

b) Determine the natural frequencies and mode shapes of the equipment as it will be mounted in service.

c) The following damping factors shall be used depending on the type of equipment.

	Percent of Critical Damping	
	OBE	SSE
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

d) Perform a static or dynamic analysis as required in the following paragraphs. In the analysis, the two major horizontal directions (E-W and N S) shall be considered acting simultaneously with the vertical direction in the most conservative manner.

1) If the equipment, including its supports, is capable of being lumped into a single mass and the natural period certified is less than 0.03 seconds, it may be analyzed statically. (The seismic loads for this method of analysis are obtained from the floor response spectra. Static coefficients are furnished to the Seller.) In this analysis all seismic loads may be assumed to act at the center of gravity of the equipment.

2) If the equipment model is a multidegree of freedom mode, and the natural periods of the equipment including its supports are less than 0.03 seconds, it may be analyzed statically. In this static analysis, the seismic forces on each component of the equipment are obtained by concentrating its mass at its center of gravity.

3) If the equipment cannot be analyzed under criteria in 1) or 2) above, than a dynamic analysis must be performed using the Response Spectra Modal Analysis Technique (floor response spectra are furnished).

In the Response Spectra Modal Analysis Technique, the peak frequencies indicated could vary ±15 percent. The square root of the sum of the squares method should normally be used to combine the modal response when the Response Spectra Modal Analysis Technique is employed. In those cases, however, where modal frequencies are closely spaced, the responses of the closely spaced modes should be combined by the sum of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the square root of the sum of the squares method in accordance with Regulatory Guide 1.92.

4) For the analysis in all cases (1, 2 and 3 above) the stresses shall be calculated as follows:

   (a) The unit stresses from the safe shutdown earthquake loads shall be added directly to the unit stresses from other applicable loadings (pressure, deadweight, etc.) for the faulted plant condition.

   (b) The unit stresses from the OBE shall be added directly to the unit stresses from other applicable loading for the upset and emergency plant conditions.

   (c) An analysis should be performed for five OBE events, each containing ten cycles of maximum stress, as determined from the required acceleration spectrum at the mounting surface. The analysis must show that OBE events, followed by one SSE event, will not result in fatigue failure of equipment to perform its Class IE function.

5) Allowable stresses for ASME Section III components are as specified in Section 3.9.3.

For non-ASME equipment, the unit stresses from safe shutdown and operating basis earthquakes shall be added to stresses from other applicable loadings, as in paragraph 4) above. The allowable stress shall not be increased due to addition of the OBE load. Due to addition of the SSE load, the allowable stress may be increased to a value of 90 percent of the yield stress of materials, provided no undue deflection takes place. The analysis in all cases should include evaluation of the effects of the calculated stresses on mechanical strength, alignment, electrical performance and noninterruption of function as related to the functional requirements of the equipment during a SSE.

Testing Method

a) This is the preferred method of qualifying equipment for seismic conditions.

b) Seismic tests must be performed by subjecting equipment to vibratory motion which conservatively simulates that to be seen at the equipment mounting during an SSE and OBE. The equipment to be tested should be mounted on the vibration generation in a manner that simulates the intended service mounting. The vibratory motion shall be applied to the three major perpendicular axes. Each of two major horizontal directions (E-W and N-S) shall be tested separately but simultaneously, with the vertical direction to give the most severe combination. The equipment being tested must demonstrate its ability to perform its intended function and sufficient monitoring equipment should be used to evaluate performance before, during, and following the test.

   Seller shall submit his detailed testing procedures for approval. Actual testing shall be in accordance with IEEE-344-1975, Section 6 entitled "Testing" or other approved criteria.

Documentation and Proposal Data

Seller shall furnish documentation for seismic design as follows:

Method for seismic qualification he intends to perform on the equipment at the time of submitting proposal.

Prior to shipment of equipment, seller shall submit his detailed seismic test and/or analysis report. If the natural frequency of the equipment was mathematically determined and found to be less than 33 cycles per second, Seller must provide the modeling method used, as well as the mass points locations, spring constants and section modulus to facilitate total system seismic analysis by Purchaser.

In any case, Seller must supply support loadings (static and dynamic at the equipment bases) and anchor bolt sizes and locations to meet the seismic considerations specified herein. If the equipment was tested and the natural frequencies are less than 33 cycles per second, Seller must provide a mathematical model with mass points locations, spring constants, and section modulus of the equipment to be used in Purchaser's piping seismic analysis.

c) At the time of shipment of equipment from shop, Seller shall provide a certificate of compliance with the specified seismic requirements.

Acceptance Criteria with Respect to Seismic Design

The acceptance of Seller's equipment by Purchaser with respect to seismic design will be based on the proven capability of the equipment to perform its safety function during and subsequent to a SSE. Acceptance will be determined by an evaluation of Seller's tests, and/or calculations to verify that the equipment stress levels and/or deformations are within the Manufacturer's acceptance limits for equipment operation.

APPENDIX 3.9B DESCRIPTION OF SEISMIC ANALYSIS AND SUMMARY OF RESULTS FOR NON-NSSS SUPPLIED ACTIVE PUMPS AND THEIR

This information was provided for operating license review and is not updated. Qualification information is readily available from seismic qualification reports, vendor drawings, records management system and equipment database.

3.9B.1 CONTAINMENT SPRAY PUMPS

3.9B.1.1 Description

The containment spray pumps are Ingersoll Rand Model 8 x 23 WDF with Westinghouse AC motor frame 500 BP 39 model. This pump is vertical, single stage, diffusor type pump with an operating speed of 1780 rpm. The pumps and motors have been shown by calculations to be adequate for the specified purpose when subjected to the operating and seismic conditions specified.

The seismic loads specified are as follows:

Horizontal: 0.2 g SSE and 0.11 g OBE

Vertical: 0.15 g

In static analysis, the seismic forces on the pump assembly were obtained by concentrating masses at the centers of gravity of the parts under analysis and multiplying by the appropriate seismic accelerations. These forces were considered to act simultaneously in the two horizontal and one vertical directions.

3.9B.1.2 Summary of Results

A summary of the stresses, deflections and loads is given below. The actual values are given and compared to the ASME Code allowable.

Structural Integrity

Component	Calculated, psi		Allowable, psi
	OBE	SSE	
Casing Foot Attachment	3,031	5,541	37,200
Casing Discharge Nozzle Attachment	9,994	19,671	37,200
Casing Suction Nozzle Attachment	5,847	8,586	37,200
Main Flange Bolting		15,560	59,000
Foot	9,102	18,001	27,390
Foot Weld	9,368	18,550	27,390
Anchor Bolting - Tension	8,901	18,523	50,000
-Shear	3,576	7,183	33,330
Support Head		110	12,600
Motor Attachment Bolting		395	50,000

Operability

Description	Calculated	Allowable
Rotor/Stator Deflection (Motor Air Gap)	.0002	.055 in.
Impeller/Ring Deflection	.0080	.015
Shaft/Cover Deflection at Mechanical Seal	.0021	.010

A computer analysis for the frequencies was performed using the ANSYS computer program. The analysis was included in the development of a multi degree of freedom mathematical model to determine natural frequency and model to determine natural frequency and mode shapes. The first five calculated frequencies are as follows:

1	49.1 Hz
2	49.5
3	121.5
4	122.8
5	203.5

3.9B.2 CHILLED WATER PUMPS

3.9B.2.1 Description

The chilled water pumps are Goulds Pumps, Inc. Model 3405M Size 6 x 8 - 17DV pumps with a motor. The seismic analysis is directed toward proving the structural integrity and functional capability of the pump.

A dynamic model of the pump is developed and a computer frequency analysis of the pump is made. The lowest frequency of the pump/motor/bedplate system is shown to be above 35 Hertz and thus the system can be treated statically.

The nozzle loads, seismic loads, and normal loads are imposed upon the computer model and a stress and deflection analysis of the entire assembly is made. The resulting stresses are compared to the allowables given in the specification and governing codes. The deflections are compared to operating clearances or other limiting criteria.

The nozzle discontinuity stresses are calculated by the method of the ASME Code, ND-3652, where the pump casing/discharge nuzzle intersection is treated as an equivalent tee In a conservative manner. The suction nozzle is treated as a curved elbow.

The discharge and suction flanges are treated by the method of the ASME Code, ND-3647, which is for the normal loads and external forces and moments caused by weight and thermal. No known accepted method exists to treat the flanges for the external forces and moments due to seismic. Thus, in this report they are treated the same as the deadweight and thermal which is believed to be conservative.

Loading Criteria

a) Seismic Loadings

The seismic loads applied exceed those given in the specification and are:

OBE	0.5 g Vertical
	0.5 g Lateral X and Z
SSE	1 g Vertical
	1 g Lateral X and Z

The seismic loads are applied separately in two horizontal and the vertical directions. The two horizontals are then combined directly by hand with the vertical loads and the maximum nozzle plus normal operating loads. This is more conservative than the square root of the sum of the squares method.

A computer analysis for the loading conditions is made. The analysis is made for the following load cases:

1) 1g Lateral X

2) 1g Lateral Z

3) 1g Vertical

4) Faulted Nozzle loads + Impeller Loads in Y Direction

5) Faulted Nozzle loads + Impeller Loads in X Direction

b) Nozzle Loads

The nozzle loads are tabulated below for this pump.

	Faulted Case	
	Suction	Discharge
FR, Resultant Force, lb.	4,200	2,790
MR, Resultant Mom, in. lb.	126,075	63,750

The forces and moments are transferred to the pump impeller centerline for the static analysis in the worst possible manner. The emergency, upset, and normal loads are 90 Percent, 80 percent and 60 Percent respectively of the faulted loads.

c) Internal Pressure Loading

The internal pressure design conditions are 150 psig at 150 F.

d) Shaft Torsional Loading

The motor horsepower is 100, at 1775 rpm. Thus, the maximum torque is:

$$T = \frac{63000(100)}{1775} = 3{,}549\ in.-lb.$$

3.9B.2.2 Summary of Results

A summary, of the actual faulted values are compared to the normal allowables below, except where noted otherwise.

Components	Actual	Allowable
Motor Hold Down Bolt Stress - Shear, psi	2,037	10,000
- Tensile, psi	3,680	20,000

Components	Actual	Allowable
Pump Hold Down Bolt Stress - Shear, psi	12,637	16,427
- Tensile, psi	40,140	53,300
Anchor Bolt Stress - Shear, psi	8,267	10,000
- Tensile, psi	13,115	14,773
Shaft Stress, psi	17,451	17,500
Components	Actual	Allowable
Frame, Stress, psi	10,986	21,600
Thrust Retainer Bolt Stress - Tensile, psi	1,786	20,000
Pump Bearing Cap Bolt Stress - Shear, psi	1,030	10,000
- Tensile, psi	1,222	20,000
Pump Pedestal Stress, psi	4,241	21,600
Nozzle Stress - Discharge, psi	18,420	21,000
-Suction, psi	5,970	21,000
Nozzle Flange Stress - Discharge, psi	20,145	21,000
-Suction, psi	26,153	33,600
Pedestal Weld Stress, psi	10,492	10,800
Pump Bearing Loads - Inboard, lbs.	342	6,338
- Outboard, lbs.	1,846	9,849
Flexible Coupling Misalignment, Radians	.00117	.017
Impeller Key Stress - Shear, psi	3,462	10,500
Impeller Relative Deflections, in.	.004	.009

Motors Seismic Analysis:

Motors must be adequate for operation during and after both safe shutdown earthquake and one half safe shutdown earthquake conditions. The equipment shall be designed to withstand the combined effect of all normal operating loads acting simultaneously with the horizontal and vertical direction earthquakes. Horizontal and vertical seismic loads shall be combined by the root of the sum of the squares method and the results added directly to the normal operating loads.

Seismic Loads for the 1/2 SSE are:

Horizontal (1)	1.580 G
Horizontal (2)	1.740 G
Vertical	1.760 G

Seismic Loads for the SSE are:

Horizontal (1)	1.920 G
Horizontal (2)	1.900 G
Vertical	2.000 G

Maximum allowable stress levels for the 1/2 SSE are 17143 psi and for the SSE are 36000 psi.

3.9B.3 AUXILIARY FEEDWATER PUMPS

3.9B.3.1 Description

The auxiliary feedwater pumps are Ingersoll-Rand Model 4x9NH-7 stage and 3HMTA-9 stage pumps. The drivers for the above models are Westinghouse Model 500X12 and Terry Turbine Co. Turbine Model GS-2, respectively. The analyses show that the pumps, motors and turbine are structurally adequate to withstand the specified seismic conditions.

The seismic loads specified are:

	Horizontal		Vertical
OBE:	0.12g (x),	0.13g (z).	0.4g (y)
SSE:	0.22g (x),	0.22g (z),	0.35g (y)
Dead Weight:	-	1g (y),	-

Natural frequencies (Hz) for the motor driven pumps (3HMTA-9) are as follows:

MODE	FREQUENCY (CYCLES/TIME)	
1	55.6	Motor Shaft
2	55.9	Motor Shaft
3	75.6	First Critical Speed of Pump
4	80.4	
5	85.7	

Natural frequencies (Hz) for the turbine driver pumps (4x9NH 7) are as follows:

MODE	FREQUENCY (CYCLES/TIME)	
1	38.9	(First Critical Speed of Pump)
2	39.7	
3	46.1	
4	47.6	
5	49.8	

3.9B.3.2 Summary of Results

Loads

PD	=	Design Pressure
PO	=	Operating Pressure
DW	=	Dead Weight
EL	=	Ext. Pipe Load
OBE	=	Operating Basis Earthquake
SSE	=	Safe Shutdown Earthquake

Motor Driven Pump

Component or Part	Combined Stress Designation	Loads	Calculated Value	Allowable Value
Fan. Bolt @Node #9	Tension +Shear	DW+ELF+SSE	U.F. = .360	U.F. < 1.0
	Bearing		11380 psi	<54130 psi
Weld @ Top PL to Pedestal	Tension + Shear	DW+ELF+SSE	2510 psi	<36000 psi
Weld @ Pedestal to Bedplate	Tension + Bending	DW+ELF+SSE	12121 psi	<36000 psi
Motor Bolt @Node #49	Tension	DW+ELF+SSE	963 psi	<87500 psi
Motor Shear Pin@ Node # 54	Shear	DW+ELF+SSE	4620 psi	<36170 psi
	Bearing		1650 psi	<73500 psi
	Lug Tear Out		770 lb.	<44660 lb.
Pump Flg.	Long.		22993 psi	<33600 psi
Suction	Rad.	ELF+PD	6431 psi	<33600 psi
	Tan.		10597 psi	<33600 psi
Pump M Tag Bolt	Tension	DW+ELF+SSE	12880 psi	<87500 psi
Shear Pin	Shear	DW+ELF+SSE	10320 psi	<36170 psi
Pump Foot	Lug Tear Out	DW+ELF+SSE	4035 lb.	<34575 lb.
	Bearing	DW+ELF+SSE	4570 psi	<43380 psi
	Shear	DW+ELF+SSE	255 psi	<14000 psi
	Tension	DW+ELF+SSE	10780 psi	<28000 psi
Shaft	Principal Stress	DW+ELF+SSE	5034 psi	<7500 psi
Radial & Thrust Bearings	Life Calculations	DW*SSE+ Impeller thrust	34 Years	-
Entire Assembly	Dynamic	DW	75 cps	>33 cps
Casing	Direct	PD	11866 psi	<14000 psi
	Direct Bending	PD	15196 psi	<21000 psi
Casing Flange	Shear	PD	7149 psi	<14000 psi
	Normal	PD	12749 psi	<21000 psi
Flange Bolts	Direct	PD	22395 psi	<25000 psi
	Direct + Bending	PD	45206 psi	<50000 psi
Stuffing Box bolts	Tension	PD	12225 psi	<25000 psi
Discharge	Axial		8550 psi	<33600 psi
Nozzle @ Pump	Hoop	ELF +	4505 psi	<28000 psi
Casing Crotch	Shear	PD	720 psi	<28000 psi
Section	Torsion		1800 psi	<28000 psi
Casing @ Discharge	Principal	ELF +		
Nozzle Interface	Stress	PD	15185 psi	<28000 psi

<u>Loads</u>

PD	=	Design Pressure
PO	=	Operating Pressure
DW	=	Dead Weight
EL	=	Ext. Pipe Load
OBE	=	Operating Basis Earthquake
SSE	=	Safe Shutdown Earthquake

Turbine Pump Driver

Component or Part	Combined Stress Designation	Loads	Calculated Value	Allowable Value
Entire Assembly	Dynamic	DW	39 cps	>33 cps
Casing	Direct	PD	11946 psi	<14000 psi
	Direct + Bending	PD	13010 psi	<21000 psi
Casing Flange	Shear	PD	5023 psi	<14000 psi
	Normal	PD	6619 psi	<21000 psi
Flange Bolts	Direct (Prelim)	PD	20220 psi	<25000 psi

Turbine Pump Driver

Component or Part	Combined Stress Designation	Loads	Calculated Value	Allowable Value
	Direct (Result)	PD	46509 psi	<50000 psi
	Direct + Bending (Result)	PD	55936 psi	<75000 psi
Stuffing Box Bolts	Tension	PD	15282 psi	<25000 psi
Discharge/Suction	Axial	ELF +	16569 psi	<33600 psi
Nozzles @ Pump	Hoop	PD	.5630 psi	<28000 psi
Casing Crotch	Shear	PD	1100 psi	<16800 psi
Section	Torsion & Shear		4734 psi	<16600 psi
Casing @ Discharge	Principal	ELF +		
Nozzle Interface	Stress	PD	20102 psi	<28000 psi
Pump Flange	long		41238 psi	<50400 psi
Suction	Rad	ELF + PD	11634 psi	<33600 psi
	Tan		19171 psi	<33600 psi
Pump Mtg.	Tension	DW+ELF+SSE	25230 psi	<87500 psi
Bolt Shear Pin	Shear	DW+ELF+SSE	20290 psi	<16170 psi
Pump Foot	Lug Tear Out	DW+DLF+SSE	7935 lb.	<34575 lb.
	Bearing	DW+DLF+SSE	8990 psi	<43380 psi
	Shear	DW+DLF+SSE	395 psi	<14000 psi
	Tension	DW+DLF+SEE	11716 psi	<28000 psi
Shaft	Torsion	DW+Shaft Torque	7486 psi	<7500 psi
Radial Bearings	Life	DW+SSE	34 Years	-
	Calculations	+ Shaft Torque		
Thrust Bearings	Life	DW+SSE	10.8 years	-
	Calculations	+ Shaft Torque		
Fan. Bolt	Tension + Shear	DW+ELF+SSE	U.F. = 644	U.F.F1.0
@ Node #35	Bearing	DW+ELF+SSE	17020 psi	<54130 psi
Weld @ Top PL	Tension +	DW+ELF+SSE	4620 psi	<36000 psi
to Pedestal	Shear			
Weld @ Pedestal to Bedplate	Tension + Shear	DW+ELF+SSE	27810 psi	<36000 psi
Turbine Bolt	Tension	DW+ELF+SSE	4505 psi	<87500 psi
@ Node # 62				
Turbine Shear	Shear	DW+ELF+SSE	10780 psi	<16170 psi
Pin R Node #54	Bearing	DW+ELF+SSE	1990 psi	<88700 psi
Lug Tear Out	DW+ELF+SSE	2954 lb.	<142550 lb.	
Oil Cooler	Tension +	DW+ELF+SSE	By Inspection	
Bolts	Shear		-the Loads are Acceptable	

The pump, motor, turbine, shaft and bedplate assembly were modeled for a 3-D finite element computer analysis using the ANSYS computer program. STIF4 3-D elements were used to simulate the bean members and STIF21 mass elements were used to stimulate the weights of the structure.

3.9B.4 DIESEL OIL TRANSFER PUMPS

3.9B.4.1 Description

The diesel oil transfer pumps are Golds Model 3196 ST Size 1 x 1/2 - 6 pumps with Westinghouse motor frame 213 T. The seismic analysis is directed toward proving both the structural integrity and functional capability of the pump.

A dynamic model is developed and a computer frequency analysis is made to obtain the frequencies of the assembly, as required by the specification.

The nozzle loads and seismic loads are imposed on the computer model of the assembly and the resulting stresses and deflections are calculated. The stresses are then compared to the allowables given in the specification.

The nozzles are analyzed for the maximum nozzle loads. The equivalent pressure caused by the nozzle bending moments and axial loads is calculated in accordance with the ASME Code and Code Case 1677 and imposed on the flanges.

This pump casing is of complex geometry and has been well verified for normal operation by service experience and hydrostatic tests. The seismic and nozzle loads imposed negligible stress in the casing except at nozzle penetrations and the frame adapter flange which have been analyzed.

The SSE case was analyzed and the resulting stresses, deflections, loads, etc. were found to be less than the 1/2 SSE allowables. Thus, only the SSE + Normal operating case is given, except where noted.

<u>Seismic Loading</u>

The lowest natural frequency of the pump system, including bedplate is 48 cps. The following loads exceed those given in the specifications.

	SSE	1/2 SSE
Horizontal X and Z	1.0g	.5g
Vertical	1.0g	.5g

The loads were applied to the center of mass of each individual pump component. The loads for three directions are applied, as required by the specification. Loads for X, Y, and Z are added directly.

A computer analysis for the loading condition is made. The analysis is made for the following load cases:

1) 1.0 g Lateral X Seismic

2) 1.0 g Lateral Z Seismic

3) 1 g Vertical Seismic

4) Maximum Nozzle + Impeller Loads

Load Case 3 is ratioed by the proper factor to obtain net upward or downward load, as appropriate.

3.9B.4.2 Summary of Results

A summary of the stresses, deflections, and loads are given below. The SSE + Normal values are given and compared to the Normal allowable values, except as noted:

Components	Actual	Allowable
Motor Hold Down Bolt Stress - Shear, psi	5,436	10,000
- Tensile, psi	8,789	19,303
Pump Hold Down Bolt Stress - Shear, psi	7,361	12,320
- Tensile, psi	15,450	38,222
Anchor Bolt Stress – Shear, psi	5,408	10,000
– Tensile, psi	7,850	19,347
Shaft Stress, psi	6,198	17,500
Frame, Stress, psi	8,974	21,600
Thrust Retainer Bolt Stress, psi	2,381	20,000
Pump Frame Bolt Stress - Shear, psi	7,882	10,000
- Tensile, psi	13,936	15,930
Frame Adapter Bolt Stress - Tensile, psi	13,118	14,575
Frame Adapter Flange Stress, psi	16,394	26,250
Maximum Nozzle Stress - Discharge psi	24,701	25,875
- Suction psi	17,725	25,875
Nozzle Flange Stress - Discharge psi	27,742	31,050*
- Suction psi	13,631	25,875
Pump Bearing Loads, lbs. - Inboard	300	4,443
- Outboard	1,566	8,855
Flexible Coupling Misalignment, Radians	.007	.017
Impeller Connection Stress - Shear psi	1,239	8,750
Impeller Connection Stress - Tensile psi	1,644	17,500
Impeller Relative Deflection, in.	.003	.025

* 1/2 SSE allowable used here.

3.9B.4.3 Conclusions

Gould's Seismic Analysis Report ME-669 including Addenda Number I meets all requirements of specification CAR-SH-M-14 Revision 3 dated September 26, 1979 Pius Attachment No. 5 Revision 9 dated November 2, 1977; the ASME Code, Section 111, Class 1, 1971 Edition including Summer 1973 Addendum: the ASME, Code, Case 1677; and accepted good practice in design analysis.

The analysis shows that the pump is structurally adequate to withstand all loading conditions and will perform its intended function during Normal, Normal + OBE, and Normal + SSE loads, and will not experience fatigue failure during the cyclical events described in Attachment 5, Revision 9.

3.9B.5 EMERGENCY SCREEN WASH PUMPS

3.9B.5.1 Description

Emergency screen wash pumps are Crane-Deming Model 3067 Size A10 pumps with Reliance Electric motors. The seismic analysis is directed toward proving the structural integrity and functional capability of the pump.

A dynamic model is developed and a computer frequency analysis is made to obtain the frequencies of the assembly, as required by the specification. The assembly is shown to be rigid.

The nozzle loads and seismic loads are imposed on the computer model of the assembly and the resulting stresses and deflections are calculated. The stresses are then compared to the allowables given in the specification. The deflections are compared to operating clearances or other limiting criteria.

The nozzles are analyzed for the maximum nozzle loads. The equivalent pressure caused by the nozzle bending moments and axial loads is calculated in accordance with the ASME Code and imposed on the flanges.

This pump casing is of complex geometry and has been well verified for normal operation by service experience and hydrostatic tests. The seismic and nozzle loads imposed negligible stress in the casing except at nozzle penetrations and the frame adapter flange which have been analyzed.

Since the maximum actual stresses, deflections, and loads are less than the upset allowables, only the Faulted case calculations are given.

<u>Loading Criteria</u>

a) Seismic Loading

The lowest natural frequency of the pump system, including the bedplate, is 37 cps. The following loads exceed those given in the specifications.

	SSE	1/2 SSE
Horizontal X and Z	1.0g	0.5g
Vertical	1.0g	0.5g

The loads were applied to the center of mass of each individual pump component. The loads for three directions were applied, as required by the specification. Loads for X, Y, and Z are added directly.

A computer analysis for the loading condition is made. The analysis is made for the following load cases:

1) 1.0 g lateral X Seismic

2) 1.0 g Lateral Z Seismic

3) 1.0 g Vertical Seismic

4) Maximum Nozzle + Impeller Loads in X direction

5) Maximum Nozzle + Impeller Loads in Z Direction

Load case 3 is ratioed by the proper factor to obtain net upward or downward load, as appropriate.

The joint and member input data is the same for the frequency analysis and is not repeated.

b) Nozzle Loads

The maximum nozzle loads for this pump are given below.

	2 In. Discharge	3 In. Suction
^{F}axial, lbs	300	500
^{F}shear, lb.s (Resultant)	300	500
^{M}bending, in.-lbs. (Resultant)	900	1500
^{M}torsional in.-lbs.	900	1500

The forces and moments are applied to the pump centerline for the static analysis.

c) Internal Pressure Loading

The internal pressure design conditions are 150 psig at 100F.

d) Shaft Torsional Loading

The motor horsepower is 15 at 3500 rpm. Thus, the torque is:

$$T = \frac{63000(15)}{3500} = 270\ in.-lbs.$$

e) Other Pump Normal Loads

The pump impeller is subjected to a 105 pound radial and a 250 pound axial load during normal operation.

3.9B.5.2 Summary of Results

A summary of the stresses, deflections, and loads are given here. The actual values are given and compared to the allowable values. The Faulted actual values are compared to the Upset allowables

Components	Actual	Allowable
Motor Hold Down Bolt Stress-Shear, psi	2,880	11,986
-Tensile, psi	7,469	29,000
Pump Hold Down Bolt Stress-Shear, psi	5,981	9,300
-Tensile, psi	15,587	22,250

Components	Actual	Allowable
Anchor Bolt Stress - Shear, psi	4,168	11,986
- Tensile, psi	7,832	29,000
Shaft Stress, psi	8,362	17,500
Frame, Stress, psi	11,839	21,600
Thrust Retainer Bolt Stress, psi	1,255	29,000
Upper Pump Frame Bolt Stress - Shear, psi	3,919	25,833
- Tensile, psi	54,233	62,500
Lower Pump Frame Bolt Stress - Shear, psi	4,241	9,300
- Tensile, psi	12,441	22,520
Frame Adapter Bolt Stress - Tensile, psi	28,143	30,800
Frame Adapter Flange Stress psi	8,485	23,100
Maximum Nozzle Stress - Discharge, psi	6,067	23,100
- Suction	5,382	23,100
Nozzle Flange Stress - Discharge, psi	9,801	23,100
-Suction, psi	6,676	23,100
Pump Bearing Loads, Lbs. - Inboard	506	15,951
- Outboard	2,490	1,8079
Flexible Coupling Misalignment, Radians	.00636	.017
Impeller Connection Stress - Tensile psi	433	17,500
-Shear psi	1,776	8,750
Impeller Relative Deflection, in.	.00676	.015
Adapter/Frame Bolting Stress Tensile psi	3,489	22,520

Reliance Electric Motor Seismic Analysis

Nameplate Rating:

HP Output:	15	Type:	PB
rpm:	3515	Frame Number:	254T
Cycles:	60	Enclosure:	TEFC XT
Volts:	460	Remarks:	Horizontal, Foot, Mounted

This seismic analysis is based on the following data:

1) Half coupling: 5.5 lb.

2) Half coupling center of gravity location 81 in. inboard from end of shaft.

3) The Dynamic-Rigid analysis procedure, as specified in IEEE 344-1975, was used based on the lowest natural frequency of the motor not being less than 33 Hz. Experience, based on analysis and tests, shows this fundamental mode of vibration to be associated with the rotor mass, supported by the shaft and stationary supporting structure. The critical speed of the rotor mass system is calculated by Program 704 and is substantially above 1980 rpm (33 Hz x 60 sec./min). The Dynamic-Rigid analysis procedure is therefore valid.

3.9B.6 CHILLED WATER CONDENSER RECIRCULATING PUMPS

3.9B.6.1 Description

Chilled water condenser recirculating pumps are Goulds Model Pumps, Inc. Model 3196 MT Size 4 x 6 - 10 pumps with Westinghouse motors. The seismic analysis is directed toward proving the structural integrity and functional capability of the pump.

A dynamic model is developed and a computer frequency analysis is made to obtain the frequencies of the assembly, as required by the specification. The assembly is shown to be rigid.

The nozzle loads and seismic loads are imposed on the computer model of the assembly and the resulting stresses and deflections are calculated. The stresses are then compared to the allowables given in the specifications. The deflections are compared to operating clearances or other limiting criteria.

The nozzles are analyzed for the faulted nozzle loads. The equivalent pressure caused by the nozzle bending moments and axial loads is calculated in accordance with the ASME Code including ASME Code Case 1677 and imposed on the flanges.

This pump casing is of complex geometry and has been well verified for normal operation by service experience and hydrostatic tests. The seismic and nozzle loads impose negligible stress in the casing except at nozzle penetrations and the frame adapter flange which have been analyzed.

Since the Faulted actual stresses, deflections, and loads are less than the Normal allowables, only the Faulted case calculations are given.

Loading Criteria

a) Seismic Loading

The lowest natural frequency of the pump system, including bedplate, is 41 cps. The following loads exceed those given in the specification.

	SSE Seismic	OBE Seismic
Horizontal X and Z	1.0 g	.5 g
Vertical	1.0 g	.5 g

The loads were applied to the center of mass of each individual pump component. The loads for three directions are applied as required by the specification. Loads for X, Y, and Z are added directly for conservatism.

A computer analysis for the loading condition is made. The analysis is made for the following load cases:

1) 1.0 g Lateral X Seismic
2) 1.0 g Lateral Z Seismic

3) 1.0 Vertical Seismic

4) Maximum Nozzle + Impeller Loads in Y Direction

5) Maximum Nozzle + Impeller Loads in Z Direction

Load Case 3 is ratioed by the proper factor to obtain net upward or downward load, as appropriate.

The joint and member input data is the same as for the frequency analysis and is not repeated.

b) Nozzle Loads

The nozzle loads for this pump as given in the specification are tabulated below.

Nozzle Loads	Suction			Discharge		
	Faulted	Emerg.	Upset/Normal	Faulted	Emerg.	Upset/Normal
Axial Force, Lb.	843	723	602	594	406	424
Resultant Shear Force, lb.	506	434	361	356	243	254
Result. Bending Mom., in.-lbs.	14,858	12,736	10,613	7,417	6,357	5,298
Torsional Mom., in.-lbs.	17,830	15,283	12,736	8,900	7,629	6,357

The forces and moments are applied to the pump impeller centerline for the static analysis.

c) Internal Pressure Loading

The internal pressure design conditions 150 psig at 105F.

d) Shaft Torsional Loading

The motor horsepower is 20 at 1755 rpm. Thus, the torque is:

$$T = \frac{63000(20)}{1755} = 718\ in.-lbs.$$

3.9B.6.2 Summary of Results

A summary of stresses, deflections, and loads are given below. The actual values are given and compared to the allowable values. The Faulted actual values are compared to the Normal allowables, except where noted.

Components	Actual	Allowable
Motor Mold Down Bolt Stress - Shear, psi	2,872	10,000
- Tensile, psi	7,476	20,000
Pump Hold Down Bolt Stress - Shear*, psi	20,188	28,350
- Tensile*, psi	48,426	67,725
Anchor Bolt Stress - Shear, psi	15,336	16,800
- Tensile, psi	13,692	25,000
Shaft Stress, psi	9,993	17,500
Frame Stress, psi	14,944	21,600
Thrust Retainer Bolt Stress, psi	1,560	20,000
Upper Pump Frame Bolt Stress - Shear, psi	18,269	18,900
- Tensile*, psi	3,960	45,150

Components	Actual	Allowable
Lower Pump Frame Bolt Stress - Shear, psi	2,908	10,000
- Tensile, psi	24,063	30,000
Frame Adapter Bolt Stress - Tensile, psi	14,290	25,000
Frame Adapter Flange Stress, psi	20,564	21,000
Maximum Nozzle Stress - Discharge psi	7,463	21,000
- Suction psi	4,522	21,000
Nozzle Flange Stress - Discharge psi	12,645	21,000
- Suction psi	16,089	21,000
Pump Bearing Loads, lbs. - Inboard	861	9,103
- Outboard	3,115	15,037
Flexible Coupling Misalignment, Radians	.006	.015
Impeller Connection Stress - Shear psi	4,724	8,730
- Tensile psi	972	17,500
Impeller Relative Deflection, in.	.006	.012
Adapter/Frame Bolting Stress - Tensile psi	5,702	25,000
By pass Piping Stress, psi	9,056	15,040

*This is Emergency allowable, see details for Upset case.

Motors Seismic Analysis:

Motors must be adequate for operation during and after both safe shutdown earthquake and one-half safe shutdown earthquake conditions. The equipment shall be designed to withstand the combined effect of all normal operating loads acting simultaneously with the horizontal and vertical direction earthquakes. Horizontal and vertical seismic loads shall be combined by the root of the sum of the squares method and the results added directly to the normal operating loads.

Seismic Loads for the 1/2 SSE are:

Horizontal (1)	1.580 g
Horizontal (2)	1.740 g
Vertical	1.760 g

Seismic Loads for the SSE are:

Horizontal (1)	1.920 g
Horizontal (2)	1.900 g
Vertical	2.000 g

Maximum allowable stress levels for the 1/2 SSE are 17143 psi and for the SSE are 36000 psi.

3.9B.7 EMERGENCY SERVICE WATER BOOSTER PUMPS

3.9B.7.1 Description

The emergency service water booster pumps are Goulds Pumps, Inc. Model 3405L size 12 x 14-12 pumps with Siemens-Allis motors. The seismic analysis is directed toward proving the structural integrity and functional capability of the pump.

A dynamic model of the pump is developed and a computer frequency analysis is made. The lowest frequency of the pump is shown to be 41.5 cycles per second and thus it can be treated statically.

The nozzle loads, seismic loads, and normal loads are imposed on the computer model and a stress and deflection analysis of the entire assembly is made. The stresses are compared to the allowables given in the specification. The deflections are compared to operating clearances or other limiting criteria.

The nozzle discontinuity stresses are calculated by the method of the ASME Code, ND-3652, where the pump casing/discharge nozzle intersection is treated as an equivalent tee in a conservative manner. The suction nozzle is treated as a curved elbow.

The discharge and suction flanges are treated by the method of the ASME Code, ND-3647, which is for external forces and moments caused by weight and thermal. No known accepted method exists to treat the flanges for the external forces and moments due to seismic. Thus, in this report the seismic loads were treated in a manner believed to be in accordance with good practice in seismic design. ASME Code Case 1677 was used in the analysis of the flanges.

Loading Criteria

a) Seismic Loadings

The seismic loadings applied exceed those given by the specification and are:

OBE	.5 g Vertical .5 g Lateral X and Z
DBE	1 g Vertical 1 g Lateral X and Z

The seismic loads are applied separately in two horizontal and the vertical directions. The two horizontals are then combined directly by hand with the vertical loads and the maximum nozzle plus normal operating loads.

A computer analysis for the loading conditions is made. The analysis is made for the following load cases:

1) Maximum Nozzle Loads + Impeller loads
2) 1 g Vertical
3) 1 g Lateral X Seismic
4) 1 g lateral Z Seismic

b) Nozzle Loads

The maximum nozzle loads are tabulated below for this pump, which has a 14 in. suction and a 12 in. discharge. These nozzle loads per Goulds Pumps, Inc. and are:

	12 in. Discharge	14 in. Discharge

Resultant Force, lbs.	2,000	2,000
Components F_x, F_y, F_z, lb.	1,155	1,155
Resultant Moment, In.-lbs.	28,920	28,920
Components M_x, M_y, M_z, in.-lbs.	16,697	16,697

The forces and moments are transferred to the pump impeller centerline for the static analysis.

c) Internal Pressure Loading

The internal pressure design conditions are 225 psig at 140 F.

d) Shaft Torsional Loading

The motor horsepower is 200 at 1780 rpm. Thus, the maximum torque is:

$$T = \frac{63000(200)}{1780} = 7{,}079\ in. = lbs.$$

3.9B.7.2 Summary of Results

A summary of the actual Normal + DBE + Max. nozzle loads are compared to the normal allowables below.

Components	Actual	Allowable
Motor Hold Down Bolt Stress - Shear, psi	3,525	10,000
- Tensile, psi	5,820	20,000
Pump hold Down Bolt Stress - Shear * psi	6,210	12,320
- Tensile * psi	22,259	40,000
Anchor Bolt Stress, Shear, psi	9,310	12,500
- Tensile, psi	22,026	25,000
Shaft Stress, psi	17,593	26,250
Frame Stress, psi	9,018	21,600
Thrust Retainer Bolt Stress, psi	1,684	20,000
Pump Bearing Bolt Stress - Shear, psi	1,059	10,000
- Tensile, psi	222	20,000
Pump Pedestal Stress - Shear, psi	3,514	21,600
Nozzle Stress - Discharge psi	3,447	21,000
- Suction psi	2,363	21,000
Nozzle Flange Stress - Discharge psi	24,601	25,200*
- Suction psi	25,006	25,200*
Pedestal Weld Stress, psi	3,640	10,800
Pump Bearing Loads, lbs. - Inboard	723	10,688
- Outboard	1,585	10,688
Flexible Coupling Misalignment, Radians	.00106	.017
Impeller Key Stress - Shear, psi	2,484	30,000
Impeller Contact Stress, psi	357	3,000

* 1/2 SSE allowable used here.

<u>Motor Seismic Analysis</u>

Summary of Calculations

The motor fundamental natural frequency has been determined to be: 5160 cpm or 86.0 Hz.

Proof of adequacy applying the safe shutdown earthquake accelerations will prove seismic withstand capability for the operating basis earthquake as well.

Horizontal (North-South)	=	0.30 “G”
Horizontal (East-West)	=	0.30 “G”
Vertical	=	0.40 “G”

	STRESSES	
	ACTUAL	ALLOWABLE
1. Anchorage System		
1.1. Normal Operation Loading		
Tensile	612	20000
Shear (Per Bolt)	-	10000
1.2. Seismic Leading		
Horizontal "G" Induced		
Tensile (Max.)	366	20000
Shear (Per Bolt)	336	10000
1.2. Vertical "G" Induced		
Tensile (Per Bolt)	448	20000
Shear		10000
1.3 Combined Loading		
Tensile (Max.)	900	20000
Shear (Max.)	576	10000

2. Rotor	
Maximum Deflection at Core	0.0017 in.
Maximum Deflection at End of Shaft Extension	0.0017 in.
Lateral Natural Frequency	9167 cpm
Maximum Bending Stress	1369 psi
Maximum Shear Stress	123 psi
3. Bearings	
Maximum Bearing Loads: Front End Radial	630 lb.
Rear (Extension) End	646 lb.
Front End Thrust	-
Rear End Thrust	152 lb.
4. Conduit Box	
Maximum Tension in Attachment Bolts	23171 psi
Maximum Shear in Attachment Bolts	3255 psi

3.9B.7.3 Conclusions

This analysis is prepared in accordance with the following standards and specifications:

IEEE-344-1975, IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations.

EBASCO Services Incorporated Specification 214-70, Motors for Station Auxiliary Service Furnished with Driven Equipment Rated Up to 460V and 250 HP (Excluding Valve Motors).

The attached calculations verify that the subject motor is capable of continuous operation under normal operating loads acting simultaneously with two horizontal components and one vertical component of the specified seismic event. No malfunction or loss of function is indicated.

3.9B.8 EMERGENCY SERVICE WATER PUMPS

3.9B.8.1 Description

The Emergency Service Water pumps are Ingersoll-Dresser Model 35LKX-2 with General Electric 8-pole 1300 HP AC Motor, Model 5K6356 x 621A. This pump is vertical, double stage, diffuser type pump with an operating speed of 885 rpm.

The seismic loads specified are as follows:

a) Horizontal: 0.4 g SSE and 0.2 g OBE

b) Vertical: 0.30 g SSE and 0.15 g OBE

The analysis is directed towards verifying both the structural integrity and 26 functional capability of the pump. The natural frequencies of the pump are determined by developing a lumped mass model of the pump and motor assemblies. The model is prepared for analysis by computer. The lowest frequency is found to be 18 hertz.

Once the frequencies are determined, the lateral seismic loads are obtained from the specification furnished for these pumps and a modal analysis is performed.

A static analysis of the pump and motor assemblies is made for the vertical case using the same model as used for the frequency analysis.

The computer code ICES-STRUDL is used for the analysis.

3.9B.8.2 Deleted by Amendment No. 49

3.9B.8.3 Summary of Results

A summary of the stresses, deflections and load is given below. The faulted actual values are compared to the normal allowables given herein.

Components		Actual	Allowable
Maximum Column Stress, PSI (SA-106 Gr. B)		13,678	15,000
Maximum Column Flange Stress, PSI (SA-105)		16,368	26,250
Bolt Stress, PSI (SA-193 B7)		30,330	37,500
Maximum Pump Casing Flange Stress, PSI (SA-216 WCB)		19,150	21,000
Bolt Stress, PSI	(SA-516)	25,108	26,250
	(SA-193 B7)	27,505	37,500
Nozzle Stress, PSI (SA-234 WPB)		16,216	26,250
Anchor Bolt Stress, PSI – Tensile		5,304	By Others
– Shear		643	By Others
Motor Hold Down Bolt Stress, PSI – Tensile (SA-193 B7)		2,415	37,500

Components	Actual	Allowable
– Shear	781	12,500
Sole Plate Stress, PSI (SA-36)	23,931	26,100
Motor Support Stress, PSI (SA-36)	2,642	21,750
Shaft Stress, PSI (A-322 Tp. 4140)	11,150	17,500
(A-276 Tp. 410)	14,330	14,400
Discharge Nozzle Flange Stress, In-lbs.	3,315,135	5,527,550
Pump Casing Stress, PSI (SA-216 WCB)	4,167	14,000
Impeller Clearance, Inches	.001	.012
Seismic Support Stress, PSI (SA-36)	15,958	21,750
Motor Support Plate Stress, PSI (SA-36)	10,642	21,750
Shaft Deflections, Inches	.052	.11
Seismic Restraint Bolting Stress, PSI – Tensile (SA-307)	18,121	20,000
– Shear	916	12,500
Motor Support Bolt Stress, PSI – Tensile (SA-193 B7)	6,089	37,500
– Shear	1,886	10,000
Motor Accelerations, G's – DBE	.59	--
– OBE	.43	
Seismic Restraint Pin Stress, PSI	40,798	51,660

APPENDIX 3.9C DESCRIPTION OF SEISMIC ANALYSIS AND SUMMARY OF RESULTS FOR NON-NSSS SUPPLIED ACTIVE VALVES

This information was provided for operating license review and is not updated. Qualification information is readily available from seismic qualification reports, vendor drawings, records management system and equipment database.

3.9C.1 BORG WARNER CORPORATION

3.9C.1.1 Introduction

The seismic and operability analysis report of the following valves has been performed for the designs detailed in Nuclear Valve Division of Borg Warner Corporation drawing numbers, as shown. Design specifications are in Ebasco Specification, High Pressure Alloy and Carbon Steel Valves (900 lb ANSI and higher 2 1/2 in. and larger), Project Id. No. CAR-SH-M-32F, Rev. 5 Dated 2/1/79 and P.O. #NY-435193. The valve assembly is built to the criteria of ASME, B&PV Code, Section III, Class 2 or 3 Nuclear Valves 1974 Edition.

Valve Size (In.)	Pressure Rating (lb.)	Material	Valve Type	Operator	N.V.D. Assembly DWG. No.	Valve Tag Number
16	921	Carbon Steel	Gate	Hydraulic	4350BB5-1	2FW-V265AB
						2FW-V275AB
						2FW-V285AB
16	921	Carbon Steel	Tilting Disc Check	-	465QBB1-1	2FW-V235N
						2FW-V245N
						2FW-V255N

In accordance with the design requirements, the valve(s) and appurtenance(s) shall be qualified by the procedures and guidelines of the Ebasco Specification No. CAR-SH-M-32F, Addendum F, seismic considerations for Mechanical Equipment. Basically, three modes of operation are considered: Upset Condition, Emergency Condition and Faulted Condition. For Faulted Mode (safe shutdown earthquake), a seismic load factor (SLF) of 3.0 g shall be applied in each of two orthogonal horizontal directions in combination with a SLF of 2.0 g in the vertical direction, all action simultaneously. The Upset Condition (operational basic earthquake) is similar to Faulted Condition, except that the SLF values shall be taken as 1/2 of the respective values of the safe shutdown earthquake.

The method of qualification will be based upon a static analysis of a rigid system, and will consist of performing a static structural analysis of the equipment under equivalent static forces conservatively representing the actual dynamic loadings. Seismic forces on each component of the equipment are obtained by concentrating its mass at its center of gravity and multiplying by the appropriate SLF. Rigid systems are defined as systems which have no natural frequency less than 33 cycles per second.

All the values used within this analysis are the actual dimensions taken from the detail prints. In all cases, the values are greater than respective d_m and t_m values required by ASME Boiler and Pressure Vessel Code, Table NB 3542-1.

Standard engineering practice shall be used to determine the maximum stress conditions in all portions of equipment. It shall be demonstrated that the maximum stresses meet the acceptance criteria for the selected valve assembly materials defined in Table 3.9.3-8.

NVD has used the 100 F pressure rating of the applicable valve pressure class along with faulted mode loading when calculating stress levels. As a conservative approach, upset allowable stress limits are compared to faulted loading, thus substantiating the design under all faulted and upset conditions.

The ASME Class 2 valve design criteria is based upon the rules of ASME, B&PV Code, NB 3200 and standard engineering practices. The allowable stress limits of 1.1S and 1.655 shall be taken for the primary membrane (P_m) and (local) primary membrane plus primary bending [(P_m or P_1) + P_B)] stress categories, respectively, for pressure boundary components. The allowable stress values S for Class 2.3 components are cited in Tables I-7-1 through I-7-3 of the ASME B&PV Code. For non-pressure boundary components, the stress limits are taken as $0.6s_y$ (AISC allowable working stress limit) for upset mode and 0.9 s_y for emergency and faulted modes.

An idealized structure system shall be modeled to simulate the vibratory mode of the valve assembly. The calculated minimum natural frequency of vibration shall be examined to satisfy the specification limit of 33 cycles per second.

3.9C.1.2 Summary

For 16 in. gate valve with hydraulic operator:

Description of Valve Section	Material Specification	Allowable Upset Mode (KSI)	Calculated Faulted Mode (KSI)
Body, Main Run	SA216 GR WCB	28.87	14.68
Body, Neck	SA216 GR WCB	28.87	15.32

Body, Thd. Rif.	SA216 GR WCB	28.87	14.85
Yoke, Legs	SA216 GR WCB	32.40	10.32
Yoke, Flange	SA216 GR WCB	32.40	14.69
Clamp, Yoke	SA216 GR WCB	32.40	14.16
Bolt, Clamp	A564 TY630	103.5	19.46

Description of Valve Section	Required Minimum Frequency (Cycle/Sec.)	Calculated Natural Frequency (Cycle/Sec.)
Valve Assembly	33.00	47.88
Valve Body, Tran.	33.00	707.05
Valve Body, Tors.	33.00	68.88

For 16 in. Tilting Disc Check Valve:

Description of Valve Section	Material Specification	Allowable Upset Mode (KSI)	Calculated Faulted Mode (KSI)
Body, Main Run	SA216 GR WCB	28.87	11.67
Body, Neck	SA216 GR WCB	28.87	13.94
Body Bonnet JT	SA216 GR WCB	28.87	9.79
Disk	SA216 GR WCB	28.87	2.86

Description of Valve Section	Required Minimum Frequency (Cycle/Sec.)	Calculated Natural Frequency (Cycle/Sec.)
Valve Body, Tran.	33.00	9558.85
Valve Body, Tors.	33.00	1475.20

3.9C.1.3 Conclusion

The above valves have been evaluated and qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, Subsection NC, 1974 and the requirements in EBASCO Specification, High Pressure Alloy and Carbon Steel Valves (900 lb ANSI and Higher, 2-1/2 in. and Larger), Ebasco project. Identification No. CAR-SH-M-32F, Rev. 5, Dated 2/1/79 and P.O. #NY-435193.

In accordance with the design specifications, it has been demonstrated that the valve assembly satisfies the design criteria for stresses and deformations. A series of simple mechanical systems has been modeled to simulate the vibratory modes of the valve assembly. With conservative quantities for mass and inertia, the minimum natural frequency of vibration satisfies the specification limit of 33 cycles per second, which classifies the component as a rigid system.

Pneumatic hydraulic operator, NVD P/N 38991 mounted on the 16 in. Gate Valve and qualified in accordance to IEEE 323-1974, IEEE 344-1975 and IEEE 344-1975 and IEEE 344-1975 and IEEE 382-Jan. and Oct. 1977 draft. The pneumatic hydraulic operator was manufactured by Borg-Warner Nuclear Division. Borg-Warner qualification report 1736 consisted of Wyle Laboratories Test Report No. 57530 and AETL Report No. MJO-5488-7968 "Modal Analysis of a Nuclear Valve Operator."

3.9C.2 JAMESBURY CORPORATION

3.9C.2.1 Introduction

In accordance with the Jamesbury Corporation Purchase Order No. N0778 John Henry Associates Inc., has performed a seismic qualification for the Wafer-Shere Valves as shown on the following table.

Size	Valve Model	Actuator Model	Pressure Rating (lb.)	Valve Tag No.
6"	8226MA	SMB000/2 H0BC	150	3SW B74SA, 75SA, 76SB, 77SB
8"	8226EA	SMB005/H1BC	150	2SW B45SA, 46SA, B47SA, 48SA, B49SA, 50SB, B51SB, 52SB, 3SW B70SB, 71SA, B72SB, 73SB.
30"	8229MT	SMB00/7 1/2 H2BC	150	3SW B1SA, 2SB

The stress analysis was performed using finite element computer models, hand calculations and results of previously performed analyses of similar valves and actuators.

In the STARDYNE finite element model, the linkage bracket shaft, and disc were modeled in considerable structural detail. Where the valve body's flexibility was considered important in the stress calculations, the body, or a portion of it, was included in the model.

The valves were analyzed for the following static equivalent loads in g's:

	1/2 SSE	SSE
Horizontal	1.5	3.0
Vertical	1.0	2.0

The seismic loads were applied in the horizontal and vertical directions simultaneously. The shaft torque and design pressure were applied during the seismic loading. Computations to determine the stresses developed in the bolts of the bracket flanges were also performed, and the displacement of disc center was determined.

3.9C.2.2 Summary and Conclusion

The stresses developed under the combined loadings specified were all within the appropriate allowable values, given in Ebasco Design Specification and ASME Section III, 1977 Edition. The displacements calculated will not bind any parts of the system, cause any leakage at the stuffing box, nor render any of the valves inoperative. Thus, it is assured that the seismic event of the magnitude indicated in the design specification will not adversely affect the function or subsequent operation of the valves. The results of the natural frequency and stresses are shown in Table A below.

Valve motor operators were manufactured by Limitorque. These operators are representative of the prototype units that were successfully seismically tested in accordance with Aero Nav Laboratories, Inc. Report No. 5771, 5772, 5773, 5774, 5770, and 5 6167 5. These reports show the operators comply with the intent of IEEE Standard 344 1975 requirements. SMB model motor operator's natural frequencies much in excess of 33 hertz and qualified up to levels of 6g acceleration in any of the three axis.

TABLE A

a) F: Lowest Natural Frequency, cps
αM+B: Body Neck Membrane + Bending Stress, psi

Valve Size	F	αm+B	Allow 1.25
6”	41.6	9800	21000
8”	34.7	14260	21000
30”	44.5	2406	21000

b) αM+B: Shaft Membrane + Bending Stress psi
αv: Shaft Shear Stress, psi

Valve Size	αm+B	Allow .9 αy	αv	Allow .6αy
6”	24547	112500	13545	75000
8”	43184	112500	20603	75000
30”	55997	112500	8550	75000

c) αM+B: Wafer Support Membrane + Bending Stress, psi
αl: Wafer Plate Stress, psi

Valve Size	αm+B	αl,2	Allow 1.25
6”	5750	4933	21000
8”	9431	9937	21000
30”	10439	9794	21000

d) Δ: Center of Wafer Stress displacement, effectively equals ΔZ

Valve Size	Δ
6”	1.0×10^{-3}
8”	2.3×1.0^{-3}
30”	$1.0 \times 10\text{-}^{2}$

e) αm+B: Bracket membrane + Bending Stress, psi
αm: Bolt tensile stress, psi

Valve Size	αm+B	Allow .9 αy	αv	Allow .6αy
6”	8054	24300	9549	82800
8”	20135	24300	23873	82800
30”	9302	24300	16719	82800

3.9C.3 PACIFIC VALVES

3.9C.3.1 Introduction

Pacific Seismic Reports FA-5475 and FA-5476 are intended to define a maximum envelope in terms of pressure, temperature, operator size and seismic loads for which the subject valves (shown on the below table) are qualified.

The technique used to determine the adequacy of the design is an equivalent static analysis. The variables used for the analysis in this report were selected with great care in order to produce results which are both universal and conservative.

The analysis employs classical strength of materials equations for the stress computations at all sections except for body bonnet flange stresses, which are investigated with the modern flange design method of the ASME Boiler and pressure vessel code, and the body run section, which is evaluated by the ratio of section properties per the code.

Size	Valve Model	Pressure Rating (lb.)	Valve Tag. No.
2 1/2"	150-7-WE-X (Hand Operated Gate Valve)	150	3CH-V875B, 130SA 3CX-V112SB
3"	180-7-WE-X (Swing Check Valve)	150	3SC-V28SA,-V33SB
4"	58809-7-WE-X (Tilting Disc Check Valve)	900	3AF-V1SA,-2SB, -V8SA,-17SA, -V21SA,-34SB, -V37SB, -31SB
6"	58809-7-WE-X (Tilting Disc Check Valve)	900	3AF-V35AB, 3MS-V99SA,-100SB

3.9C.3.2 Summary and Conclusion

The allowable stress level at a given section of the valve is a function of the materials of construction, temperature, and acceptance criteria.

The temperature of the material has an effect on the allowable stress. This analysis assumes that the wetted pressure boundary parts are at 650 F. All other parts are assumed to be uninsulated and in partial contact with the wetted parts at a temperature of 300 F.

The ASME Section III Code requirement concerning allowable stresses is applicable only to the pressure retaining materials of the body, bonnet, and body-to-bonnet flange bolting. Allowable stresses of Table I-7.0 of Appendix I of ASME Section III are used as the acceptance criteria of the pressure retaining parts. Non-pressure retaining parts are compared to 60 percent of the material yield strength for acceptable criteria.

Table B is a compilation of the allowable tensile stresses for each type of section. The shear allowable is 0.5 of the tensile allowable.

TABLE B

PART TYPE	Qualified ASME SPEC	Materials GR/Type	Max. Qualified Temperature	Acceptance Criterion	Value in KSI
Pressure Retaining Casting	SA216 SA217 SA217 SA217 SA351	WCB WC6 C5 CA15 CF3M/CF8M	650 F	Sa	14.8
Pressure Retaining Bolting	SA564 SA193	630 B7	300 F	Sa	25
Non-Pressure Retaining Bolting	A354 A193 A453	BD B7 660	300 F	.6Sy	63
Non-Pressure Retaining Parts	A513 A216	WCB	300 F	.6Sy	19.1

Fundamental Frequency:

Natural frequencies and mode shapes were calculated using a finite element computer analysis. The computer program and model are capable of transmitting forces in all 6 degrees of freedom.

Valve Size	F (Hz)
2 1./2"	200
3"	907
4"	1000
6"	830

Seismic Acceleration: (g values)

Valve Size	X-Axis	Y-Axis	Z-Axis	Resultant
2 1/2"	31.36	31.36	31.36	54.90
3"	32	32	32	56.01
4"	32	32	32	56.01
6"	32	32	32	56.01

Component Stress Level:

a) For 2 1/2" Valve;

Component	Calculated Stress (psi)	Allowable Stress (psi)
Body Neck: Tensile	2999	14,800
: Shear	319	7,400
Body-Bonnet Fasteners	19,670	25,000

b) For 3: Valve;

Component	Calculated Stress (psi)	Allowable Stress (psi)
Body Neck: Tensile	743	14,800
: Shear	114	7,400
Body-Bonnet Fasteners	13,287	25,000

c) For 4" Valve;

Component	Calculated Stress (psi)	Allowable Stress (psi)

Body Neck: Tensile	6,103	14,800
: Shear	221	7,400

d) For 6" Valve;

Component	Calculated Stress (psi)	Allowable Stress (psi)
Body Neck: Tensile	3,387	14,800
: Shear	97	7,400
Body-Bonnet Fasteners		

3.9C.4 YARWAY CORPORATION

3.9C.4.1 Introduction

The following assumptions were considered for analysis:

1) Model valve for frequency analysis as a massless cantilever beam with mass of motor operator, yoke, valve body and valve stem lumped at center of gravity motor actuator.

2) Assume valve is rigidly supported at inlet and outlet due to pipe supports located in immediate vicinity on both sides of valve.

3) Valve is modeled as a single D.O.F. system and can be analyzed by the equivalent static analysis procedure.

4) Motor actuator is assumed to have been analyzed separately.

Yarway Corp. Seismic Analysis Report No. 958540 is for the following valves;

Valve Size	Model	Actuator	Valve Tag Number
2", Globe, 1500#	5515B-F316M	Motor-Limitorque SMB-00-10	3CT-V85SA 3CT-V88SB

3.9C.4.2 Summary of Results

The natural frequencies calculated for the valve are summarized below.

COMPONENT	FREQUENCY(HZ)
Valve Incl. Actuator	34.4
Valve Stem	109

The seismic stress calculations led to the following results for the design conditions.

COMPONENT	MAXIMUM STRESS (psi)
Yoke	11,454
Yoke/Electric Actuator	38,081
Body	8,635
Yoke/Body Connection	13,806
Stem	6,347

A summary of support (piping) reactions for the design seismic loads considered as follows:

Maximum Values of:

Axial Force (lb.)	515
Shear (lb.)	6,152
Bending Moment (in. lb.)	10,436
Torsional Moment (in. lb.)	10,610

It is shown in the Table that the stresses on the previous table combined with operating stresses are less than allowable values.

STRESS RESULTS AND ALLOWABLE STRESS VALUES
2" 5515B-F316M Welbond Valve

Component		Stresses (psi) Loading Seismic	Pressure	Actuator	Total	Allowable
Yoke	Membrane	171	105	1,676	1,952	18,260
	Combined	11,454	105	1,676	13,235	27,390
Yoke/Electric Actuator Connecting Bolts		38,081	-	17,608	55,689	102,000
Body	Hoop Membrane	-	404	-	404	18,370
	Longitudinal Membrane	202	246	1,262	1,708	18,370
	Combined	8,693	348	4,465	13,506	27,555
Yoke/Body Connection	Membrane	298	101	2,922	3,321	18,370
	Combined	13,816	101	2,992	16,839	27,555
Stem	Membrane	27.4	712	20,544	21,284	33,340
	Combined	6,347	712	20,544	27,603	48,510

3.9C.4.3 Conclusions

The calculated natural frequency of the valve (being above 33 Hz) shows that it is valid to consider the valve as a rigid system, and used the equivalent static analysis procedure. Seismic design loads are as follows:

a) Horizontal 3 g's

b) Vertical 2 g's

On the basis of the methods used and results obtained in this report, it may be concluded that the 2 in. Yarway welbond motorized valve has been analyzed conservatively and is acceptable.

Motor operators were manufactured by Limitorque. Subject operators were successfully seismically tested in accordance with Aero Nav. Laboratories, Inc. Report No. 5770 through 5773. These reports show that the operators comply with the intent of IEEE Standard 344-1975 requirements. SBM Model motor operators natural frequencies much in excess of 33 Hz and qualified up to levels of 6 g acceleration in any of the three axis.

3.9C.5 ANCHOR/DARLING VALVE COMPANY

3.9C.5.1 Introduction

The objective of the seismic report is to demonstrate that in consideration of the design basis earthquake, the combined stresses do not exceed 1.5 times ASME Code allowable primary stresses for the material used.

To obtain loads in the seismic analysis of a valve, acceleration values in units of "g's" are multiplied by the total mass of the extended parts resulting in an inertia force. The inertia force is applied at the center of gravity of the extended parts as a static loading case such that the forces, moments, and stresses can be calculated.

The valve assembly is analyzed assuming that the body is an anchored rigid mass and that the seismic load plus operating pressure plus dead weight plus operational loads are acting upon the valve simultaneously.

The yoke legs, yoke clamp, and upper body areas are analyzed to prove that the valve is designed within allowable stress.

When the term super-structure of the valve is used, it includes all parts mounted on the valve above the body and bonnet.

Allowable stresses are taken at atmospheric temperature except for the body which is taken at design temperature.

The seismic qualification for 2 1/2 in. and larger CS and SS active valves which are supplied from A/D as follows:

Valve Size/Type	Actuator	ANSI Pressure Rating (lb.)	Anchor/Darling SR No.	Valve Tag No.
3"-Gate	Limitorque-Motor SMB 000	150	S.0 E-9074, Rev.A	2MD-V36Sa,77SA
3"-Check	-	150	S.0 E-5796-6, Rev.A	2IA-V33SN
3"-Gate	Air Piston	150	S.0 E-5796-7, Rev.3	2IA-V34SN
4"-Gate	Air Piston	150	S.0 E-5796, Rev.A	3SW-V237SA, 238SB, 3SW-V266SA, 267SB
4" Check	-	150	S.0 E-5796-8 Rev.A	2FP-V48SN
4" Gate	Limitorque-Motor SMB 00	900	S.0 E-5796-I	2AF-V10SB, 19SB, 23SB, 2AF-V116SA, 117SA, 118SA
6"-Check	-	150	S.O. E-9074, Rev.A	2FP-V46SN
6"-Check	-	900	S.O. E-9074	2AF-VI53SAB, 154SAB
6" Gate	Limitorque Motor SMB 00	900	S.O. E-5796, Rev.C	2MS-V8SB,95A
8"-Gate	Limitorque Motor	300	S.O E-5796-17 Rev.B	2CT-V21SA

Valve Size/Type	Actuator	ANSI Pressure Rating (lb.)	Anchor/Darling SR No.	Valve Tag No.
	SMB 000			
8"-Check	-	300	S.0 E-9074-23	2CT-V27SA,51SB
12"-Gate	Limitorque Motor SMB 000	150	S.0 E-5796-21, Rev.B	2CT-V2SA, 3SB,6SA,7SB
12" Check	-	150	S.0 E-5796-20, Rev.A	2CT-V4SB,-V55B

3.9C.5.2 Summary of Results

Seismic Loads:

Horizontal Acceleration (G2): 4.24 g

Vertical Acceleration (G1): 2.0 g

Stresses: Comparison of calculated vs. allowable stresses.

a) 6 in. Gate w/Motor Operator;

Body Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	11,461	21,750
2) Radial Flange	5,452	21,750
3) Tangential Flange	3,996	21,750
(Gasket Seating)		
1) Longitudinal Hub	11,800	25,620
2) Radial Flange	5,674	25,620
3) Tangential Flange	4,160	25,620

Bonnet Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	10,593	21,750
2) Radial Flange	5,483	21,750
3) Tangential Flange	6,194	21,750
(Gasket Seating)		
1) Longitudinal Hub	10,606	25,620
2) Radial Flange	5,473	25,620
3) Tangential Flange	6,206	25,620
Yoke Leg Bolts	14,388	42,500
Yoke Bending Stress	9,220	26,250
Motor Yoke Bolting		
Tensile	6,104	25,000
Shear	1,093	15,000

b) 4 in. Check Valve:

Body Stresses (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	5,405	26,250
2) Radial Flange	4,504	26,250
3) Tangential Flange	1,804	26,250

Stresses (Gasket Seating)

1) Longitudinal Hub	10,274	26,250
2) Radial Flange	9,767	26,250
3) Tangential Flange	3,915	26,250

Bonnet (calculated thickness vs. actual thickness)

Calculated (in.)	Actual (in.)
.614	1.03

c) 12 in. Check Valve:

Body Stresses (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	3,197	21,750
2) Radial Flange	1,278	21,750
3) Tangential Flange	1,032	21,750
Stresses (Gasket Seating)		
1) Longitudinal Hub	4,323	25,620
2) Radial Flange	2,064	25,620
3) Tangential Flange	1,670	25,620

Bonnet (calculated thickness vs. actual thickness)

Calculated (in.)	Actual (in.)
1.05	2.05

d) 3 in. Gate W/Air Piston Operator:

Body Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	12,573	26,250
2) Radial Flange	15,380	26,250
3) Tangential Flange	6,277	26,250
Cylinder Stress (tensile)	5,156	90,000
Operator Yoke Bolting:		
Tensile	1,522	42,500
Shear	458	25,500
(gasket seating)		
1) Longitudinal Hub	14,104	26,250
2) Radial Flange	17,253	26,250
3) Tangential Flange	7,042	26,250
Bonnet Flange (operating cond.)		
1) Longitudinal Hub	23,086	26,250
2) Radial Flange	8,487	26,250
3) Tangential Flange	10,519	26,250
(gasket seating)		
1) Longitudinal Hub	25,177	26,250
2) Radial Flange	9,256	26,250
3) Tangential Flange	11,472	26,250
Yoke Leg Bolts	39,393	42,500

Yoke Bending Stress	18,558	26,250

e) 12 in. Gate w/Motor Operator

Body Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	8,312	21,750
2) Radial Flange	5,665	21,750
3) Tangential Flange	3,899	21,750
(gasket seating)		
1) Longitudinal Hub	8,951	25,620
2) Radial Flange	6,228	25,620
3) Tangential Flange	4,284	25,620

Bonnet Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	12,860	21,750
2) Radial Flange	6,079	21,750
3) Tangential Flange	3,554	21,750
(Gasket Seating)		
1) Longitudinal Hub	13,476	25,620
2) Radial Flange	6,420	25,620
3) Tangential Flange	3,554	25,620
Yoke Leg Bolts	6,505	42,500
Yoke Bending Stress	4,817	26,250
Motor Yoke Bolting:		
Tensile	17,280	25,000
Shear	2,109	15,000

f) 8 in. Gate w/Motor Operator:

Body Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	10,695	21,750
2) Radial Flange	4,983	21,750
3) Tangential Flange	4,317	21,750
(gasket seating)		
1) Longitudinal Hub	11,730	25,620
2) Radial Flange	5,652	25,620
3) Tangential Flange	4,896	25,620

Bonnet Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	15,678	21,750
2) Radial Flange	5,095	21,750
3) Tangential Flange	4,883	21,750
(Gasket Seating)		
1) Longitudinal Hub	16,701	25,620
2) Radial Flange	5,505	25,620
3) Tangential Flange	5,278	25,620

Yoke Leg Bolts	9,396	42,500
Yoke Bending Stress	13,327	26,250
Motor Yoke Bolting:		
Tensile	8,475	25,000
Shear	1,438	15,000

g) 3 in. Check Valve:

Body Stresses (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	2,704	26,250
2) Radial Flange	2,329	26,250
3) Tangential Flange	936	26,250
Stresses (Gasket Seating)		
1) Longitudinal Hub	13,183	26,250
2) Radial Flange	13,900	26,250
3) Tangential Flange	5,585	26,250

g) 3 in. Check Valve:

Bonnet (calculated thickness vs. actual thickness)

Calculated (in.)	Actual (in.)
.40	1.03

h) 4 in. Gate w/Motor Operator:

	Calculated (psi)	Allowable (psi)
Yoke Clamp Stress	2,539	32,400
Clamp Bolt Stress	8,595	94,500
Yoke Leg, Tensile Stress	5,494	32,400
Shear Stress	1,439	18,000
Yoke Bending Stress – Sect. A- A	4,873	26,250
Sect. B-B	10,024	26,250
Operator Bolting, Tensile Stress	10,296	25,000
Shear Stress	1,122	15,000

i) 4 in. Gate w/Air Piston Operator:

Body Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	12,237	26,250
2) Radial Flange	10,076	26,250
3) Tangential Flange	4,002	26,250
(gasket seating)		
1) Longitudinal Hub	12,831	26,250
2) Radial Flange	10,732	26,250
3) Tangential Flange	4,263	26,250

Bonnet Flange (operating cond.)	Calculated (psi)	Allowable (psi)
1) Longitudinal Hub	7,037	26,250
2) Radial Flange	8,678	26,250
3) Tangential Flange	9,786	26,250

(Gasket Seating)

1) Longitudinal Hub	7,276	26,250
2) Radial Flange	9,207	26,250
3) Tangential Flange	10,383	16,250
Yoke Leg Bolts	15,792	42,500
Yoke Bending Stress	4,469	26,250
Cylinder Stresses:	14,470	90,000
	6,303	90,000
Operator Yoke Bolting: Tensile	5,565	42,5200
Shear	3,089	25,500

Summary

The seismic analysis of the main steam power operated relief valve is performed and reported. The stress values of the critical valve components, because of the seismic loading, are determined and are summarized on the following Table C. As can be seen, the stresses are within allowable limits.

The functional operability analysis shows that stem binding will not occur due to seismic deflection.

TABLE C

CALCULATED MAXIMUM STRESS UNDER LOADS

NO.	SELECTED VALVE COMPONENT	OPERATING	OBE OPERATING & 50 PERCENT SEISMIC	SSE OPERATING & 100 PERCENT SEISMIC	MATERIAL
*1	Yoke Mounting Bolts	7,712 psi	15,754 psi	23,795 psi	ASTM A 193 GR B 7
*2	Yoke Leg	953	3,068	5,182	ASTM A 36 STL

MAXIMUM ALLOWABLE STRESS UNDER LOADS

NO.	SELECTED VALVE COMPONENT	OPERATING	OBE OPERATING & 50 PERCENT SEISMIC	SSE OPERATING & 100 PERCENT SEISMIC
*1	Yoke Mounting Bolts	25,000 psi	30,000 psi	30,000 psi
*2	Yoke Leg	14,500 psi	17,400 psi	17,400
3	Natural Frequency	52 Hz	MINIMUM REQUIRED	33 Hz

NOTES:
*Allowable stress for structural steel are from 1974 ASME Code, subsection NA, for Class 2 components.

3.9C.5.3 Conclusion

The analysis has demonstrated that the valve assembly satisfies the seismic design requirements prescribed in Ebasco Project Specification No. CAR-SH-M32A and complies with the applicable sections of the ASME BPVC 1971 including Summer 1972 Addenda for Class 2 and 3 valves.

The fundamental natural frequency of the valve assembly is higher than 30 cycles and the natural period is less than 0.033 sec. The valve is therefore classified as rigid.

The analysis was performed considering the worst case using the safe shutdown earthquake parameters. Seismic accelerations, horizontal and vertical were assumed to be acting simultaneously and the vertical acceleration pin assembly. Computed stresses satisfy the allowable stress limits.

The design is assured to be functional during and after the specified seismic event.

Motor operators were manufactured by Limitorque. Subject operators were successfully seismically tested in accordance with Aero Nav. Laboratories, Inc., Report No. 5770 through 5773. These reports show that the operators comply with the intent of IEEE Standards 344-1975 requirements. SMB-00 model motor operators natural frequencies much in excess of 33 Hz and qualified up to levels of 6g acceleration in any of the three axis.

3.9C.6 ANDERSON, GREENWOOD & COMPANY

3.9C.6.1 Introduction

The purpose of the design report is to provide verification of valve suitability to the requirements specified in Ebasco Specification CAR-SH-BE- 27 and the ASME Code, Section III, 1977 Edition, and to formalize design notes and calculations. Section I provides the necessary stress analysis and natural frequency calculations to satisfy the requirements of References 1 and 2. The remaining Sections provide backup information and detailed design calculations.

Seismic Accelerations greater than those specified in Reference I were used for the analysis.

Horizontal Seismic	=	4.0G
Vertical Seismic	=	3.0G

In some cases a higher value for pressure or temperature may have been used for anlaysis.

All parts are analyzed for maximum loading conditions including SSE. For pressure retaining parts, the resulting stress levels are compared with allowable stress levels at design temperature.

For non-pressure retaining parts, the resulting stress levels are compared to material yield stress at design temperature. Shear allowables are taken as .60 times these values. Since code allowables have a safety factor incorporated, all pressure retaining parts shall have an F_a >1.00. Non-pressure retaining parts shall have an $F_a > 1.50$.

$$F_a = \frac{\text{Allowable Stress @ Design Temperature}}{\textit{Calculated Stress}}$$

Functional operability of these valves is assured in that a detailed stress analysis using "faulted" loading conditions has been performed on all components and stress levels were shown to be well below allowable stresses for "normal" conditions. (= .625 x yield stress) indicating that no permanent set will occur. The minor deformations that occur due to the loadings indicated will not affect the safety related functions of these simple swing check valves.

3.9C.6.2 Summary

For the 6 in. HVAC check valves:

DESCRIPTION OF VALVE SECTION	MATERIAL SPECIFICATION	ALLOWABLE UPSET MODE (SHEAR STRESS)	CALCULATED FAULTED MODE (SHEAR STRESS)
Disc Assembly	SA-240-347	18,300 psi at 150 F	5846 psi
Arm Assembly	SA-269-304	18,540 psi @ 150 F	2946 psi
Pivot Bracket Bolts	Bronze B438 Grade 1 Type 2	6,600 psi	350 psi
Shaft	SA-276-304	18,540	1928 psi

DESCRIPTION OF VALVE SECTION	REQUIRED MINIMUM FREQUENCY (Hz)	CALCULATED MINIMUM FREQUENCY(Hz)
Valve Assembly	33.00	169

3.9C.6.3 Conclusion:

The above values have been evaluated and qualified in accordance with the requirements of Pargraph NA3350 of the ASME Boiler and Pressure Vessel Code Section III; Nuclear power plant components and the requirement of Ebasco Project Identification CAR-SH-BE- 27, R.7.

In accordance with the Design Report from Anderson, Greenwood Co., it has been demonstrated that the valve assembly satisfies the design criteria for stresses. A simple mechanical system has been modeled to simulate the vibration modes of the valve assembly. The natural frequency of the valve assembly classifies the component as a rigid system.

3.9C.7 CROSBY VALVE AND GAGE COMPANY

3.9C.7.1 Introduction

The purpose of Test Report No: 3833 is to determine that the following valves meet the seismic requirements of the design specification Ebasco Seismic Specification No. CAR-SH-M 55, Revision #9.

Size	Valve Model	Valve Tag No.	
6 x R x 10	HA-75-FN	2MS-RISA	2MS-R9SA
		2MS-R2SB	2MS-R10SA

2MS-R3SA	2MS-R11SB
2MS-R4SA	2MS-R12SA
2MS-R5SB	2MS-R13SA
2MS-R6SA	2MS-R14SB
2MS-R7SA	2MS-R15SA
2MS-R8SB	

Operating stresses are at a maximum when the valve is open and discharging fluid at maximum accumulated pressure. The forces generated during discharge include a force caused by compression of the valve spring as well as a force due to the maximum of the discharging fluid and a force due to the static pressure developed at the elbow outlet.

The forces shown in this report are those resulting from the valve discharging freely to atmosphere. When valves discharge to a closed system, the entire configuration should be analyzed by the piping designer to determine the forces transmitted by and to the valve.

The natural frequency of the valve is calculated to demonstrate that a rigid body approach is appropriate. Seismic stresses are then combined with operating stresses and compared to code allowables. Stresses are determined in the minimum cross section in the valve body.

3.9C.7.2 Seismic Loads

The magnitude of the applied load was required to be equal to the combined weight of all the valve superstructure parts times a seismic coefficient of 5.0. The following parts, comprising the valve superstructure, were weighed:

Spindle	Bearing Adapter	Forked Lever
Bonnet	Adjusting Bolt	Forked Lever Pin
Bonnet Stud Nuts (6)	Adjusting Bolt Nut	Forked Lever Pin Cotter
Spring	Spindle Nut	Lever
Spring Washers (2)	Spindle Nut Cotter	Lever Pin
Bearing	Cap	Lever Pin Cotters (2)
Bearing Pin	Cap Set Screws (4)	

These parts were found to have a combined weight of 756 lbs. To add a margin of conservatism the weight was assumed to be 800 lbs.

Static Coefficient Seismic Load = 5.0 x Superstructure Weight

= 5.0 x 800

Static Coefficient Seismic Load = 4000 lbs.

The specified seismic accelerations are:

$g_h = 3.0$

$g_v = 2.0$

Therefore the Seismic loads are:

$F_{sx} = Wg_h$

$F_{sx} = 1880 \times 3.0$

$\underline{F_{sx} = 5640 \text{ lbs.}}$

$F_{sz} = \underline{F_{sx} = 5640 \text{ lbs.}}$

$F_{sy} = Wg_v$

$F_{sy} = 1880 \times 2.0$

$F_{sy} = 3760$ lbs.

Operating Loads

Forces generated by valve operation are negligible.

Set Points

The following set point variation obtained in seismic qualification testing.

	Set Point Pressure (psig)			Maximum Variation From Base Value Average Percent	
	Average	Maximum	Minimum	Positive	Negative
Base Value Cycles (No Load)	1114.67	1119	1112	.388	.002
Cycles With Seismic Load	1111.33	1113	1110	---	.004
Cycles With No Load	1108.67	1112	1105	---	.009
Cycles With Both Loads	1107.33	1112	1103	NA*	NA*
Cycles With No Loads	1114.33	1116	1112	.119	.002
Overall Qualification Test Values	1111.27	1119	1103	.388	.010

*Since the discharge force can only exist after valve opening, set pressure variation with the simulated discharge force applied is not meaningful.

3.9C.7.3 Summary

The natural frequency of the valve was calculated and found to be greater than 33 Hertz permitting static analysis.

Stresses due to operating loads and seismic effects were determined at the minimum cross section in the valve body.

Stresses were found to be less than code allowables, demonstrating conformance with the seismic requirements of Ebasco Seismic Specification CAR-SH-M-55, Rev. # 9.

3.9C.7.4 Conclusion

For the section analyzed, the calculated stresses were compared to the allowable stress from Appendix I of Section III of the ASME Code.

The maximum primary membrane stress is less than the allowable stress, the sum of primary membrane stress plus bending stress is less than 1.5 times the allowable stress and the maximum shear stress is less than 0.6 times the allowable stress, demonstrating the structural adequacy of the design.

3.9C.8 CONTROL COMPONENTS, INC.

3.9C.8.1 Introduction

The seismic analysis is based on requirements set forth in EBASCO Specification CAR-SH-M-56. Seismic accelerations have been applied independently to each axis and assumed to act simultaneously in horizontal 3 g's (X and Z axis) and vertical (Y axis) direction, 2 g's.

Operational loads are combined with seismic loads and total load has been used in order to calculate maximum stresses.

The following subject valve is analyzed for stress, under the seismic conditions specified. The sections selected for analysis are those judged to have the peaks in the stress profile of the valve. This analysis is directed to that part of the assembly beyond the pressure boundary, and to the effect of the valve top works on the yoke-to-bonnet joint. The valve assembly is treated as a free body. Only those sections considered to be significant are analyzed. These are critical sections of the bonnet-yoke interface which includes yoke legs and yoke mounting bolts.

In addition, a functional operability analysis is performed to insure that stem binding will not occur during a seismic event.

Each analysis is as intensive as is required for conservative evaluation of stresses.

Valve Size	Valve Model	ANSI Rating	Valve Tag No.
8 x 10	OXG9-X8-X8BW-10BW	900 lb.	2MS-PI8SA 2MS-PI9SB 2MS-P20SA

Seismic analysis has been performed by using WANG Computer 2200 A/B. The language used in the computer program is called BASIC.

Details of mathematical models and derivations of stress formulas are presented in the form of an appendix for each component so that an individual can verify the computer output results with long hand calculations.

All appendices were compiled together in one program in order to avoid duplication of input data. Values of all input data required to run this program are presented with output results.

3.9C.8.2 Seismic Loads

Seismic Forces: (SSE)

Vertical: 2.0 g's in Y axis $\pm$ g gravity

Horizontal: 3.0 g's in X & Z axis

Forces are assumed to act simultaneously.

OBE = Operating basis earthquake = 50 percent of SSE

SSE = Safe shutdown earthquake

3.9C.8.3 Actuator

The valve is equipped with an electro hydraulic piston type actuator.

The actuator is supplied with a pressurized accumulator in order to obtain desired flow condition through this valve in case actuator operating pressure fails.

ACTUATOR PART NUMBER	252780105
FAILURE POSITION	Closed
P = MAXIMUM OPERATING PRESSURE	3500 psig
A = ACTUATOR PISTON AREA	4.91 in.^{2}
a = ACTUATOR STEM AREA	1.0 in.^{2}
STROKE	10 in.

3.9C.9 ROCKWELL INTERNATIONAL

3.9C.9.1 Introduction

To prove by calculations that the maximum unit stresses, including seismic loadings and weights of parts do not exceed the maximum allowable stress limits, thereby meeting the requirements of the ASME B & PVC Section III, Division I for Class 3 valves. Seismic loading shall be 3 g's in the horizontal direction and 2 g's vertically to comply with the requirements of EBASCO Design Specification CAR-SH-M34R & 36R.

	Valve Size	Valve Model	ANSI Rating	Valve Tag No.
A.	2 in.	3674T3-Check	1500	3AF-V27SA 3AF-V28SB 3AF-V29SAB

	Valve Size	Valve Model	ANSI Rating	Valve Tag No.
B.	2 in.	838YT3	1500	3FO-V23SA 3FO-V24SB 3CT-V86SAB
C.	2 in.	3674F316T2	1500	2CT-VI3SA 2CT-V35SB

3.9C.9.2 Assumptions

There are several basic assumptions that will be made. These are:

1) That all factors and estimations will be made such that the most conservative results are obtained.

2) That the check valve is only supported on one end.

3) That weights of parts, whether measured or calculated, are nominal.

4) That the valve is in either a horizontal or vertical line.

3.9C.9.3 Conclusion

The SIZE 2 Figures 3674T3, 838YT3 and 3674F316T2 valves meets the requirement that the unit stresses in the valve parts do not exceed 1.5 times the maximum allowable stress limits and the thread shear stress does not exceed .6 times the maximum allowable stress limits when the unit stresses and shear stresses due to internal pressure and operational loading are combined with a seismic loading of 2g vertical and a horizontal resultant of 3g. The general membrane stress does not exceed one times the maximum allowable stress limit.

That the check valve cantilevered at one end has a natural frequency greater than 33 Hz.

3.9C.10 TRW MISSION

3.9C.10.1 Introduction

The purpose of this analysis is to demonstrate the seismic acceptability of various sizes TRW Mission Duo Chek check valves. This acceptability will be in accordance with specific seismic requirements as set forth by Ebasco Services Inc., Agent Specification CAR-SH-M-70, ASME Section III Class 2 & 3.

	Valve Size	Valve Model	ANSI Rating	Valve Tag No.
A.	6 in.	K15SPF-Ul2	150 lb.	3CE-V4ISA 3CE-V42SB
B.	8 in.	K15SPF-Ul2	150 lb.	3CE-V43SAB 3SW-V800SA 3SW-V821SB
C.	14 in.	15SEF-U01	150 lb.	3SW-V41SB
D.	36 in.	G15SPF-TO4	150 lb.	3SW-V369SN

The requirements are as follows:

The seismic qualifications of Class 3 equipment shall demonstrate an equipment's ability to perform its required function during and after the time that it is subjected to the forces resulting from a seismic disturbance.

All valves shall be capable of withstanding the simultaneous application of the following loads:

a) All normal operating loads including pressure.

b) A horizontally applied inertial load equivalent to a seismic acceleration of 3.0 g.

c) A vertically applied inertial load equivalent to a seismic acceleration of 2.0 g.

All seismic loads may be assumed to act at the center of gravity of the equipment.

The following assumptions were made in the analysis:

1) The seismic analysis of the TRW Mission Duo Chek will be based on the response of the valve body to the determined load stresses.

2) The Duo Chek is a wafer-type check valve and does not have any external operators or control devices of any kind.

3) The analysis will assume the maximum acceleration of 3.0 g's in the two major horizontal directions, acting simultaneously with a vertical seismic acceleration of 2.0 g's.

4) The valve body may be treated as a right circular cylindrical section of constant inside and outside diameter. The error introduced by neglecting the varying cross section and the presence of the valve rib will be conservative inasmuch as these effects, if considered, would increase the stiffness rate.

5) With regard to vibration theory, our model will behave as a single degree of freedom system. However, the model will be oriented in such a way as to give us the most conservative answer, or highest natural period of vibration.

6) The nature of the induced stress in our model will be of pure compression and bending. The stresses will be added to give us the resultant stress due to the seismic load.

The following steps to be observed;

1) The natural period of vibration, T, of the system to be determined.

2) Using Newton's second law of motion, the maximum inertia force applied to our system to be computed.

3) The maximum compressive stress to be determined by dividing the inertia force by the projected cross sectional area of our model normal to the direction of the force. Bending stresses to also be determined.

4) The seismic loads to be added directly to the stresses from other applicable loads.

5) The total load stresses to be compared to the allowable stresses defined by "ASME Boiler and Pressure Vessel Code," Section III, 1974.

Natural Frequency:

The natural period of vibration and natural frequency for all the valves have been tabulated below:

Size	Figure Number	T (sec.)	F (Hertz)
6 in.	K15 SPF-UI2	0.000120	8313
8 in.	K15 SPF-UI2	0.000155	6442
14 in.	15SEF-U01	0.000217	4602

Stresses:

The seismic results and stresses results for the subject valves are tabulated below:

Size	Figure Number	S_h	S_v	H	S_{rh2}	SR_{V2}
6 in.	K15 SPF-UI2	5.585	4.894	1680	2225	2225
8 in.	K15 SPF-UI2	7.176	6.279	1966	2379	2379
14 in.	15SEF-U01	9.483	8.127	3990	4215	4215

Size	Figure Number	Material	ASME Allowable Stress (psi)
6 in.	K15 SPF-U12	ASME SA-216 GR. WCB	17500
8 in.	K15 SPF U12	ASME SA-216 GR. WCB	17500
14 in.	15SEF-U01	ASME SA-216 GR. WCB	17500

3.9C.10.2 Conclusion

1) In each analysis, the natural period of vibration of the model was found to be well below the figure required by the specifications.

2) The stresses created by the seismic stimuli are relatively small and therefore had very little effect on the valve.

3.9C.11 MASONEILAN

3.9C.11.1 Introduction

The seismic qualification program for Masoneilan control valves is divided into three major phases. Each of the three phases, modal testing, seismic analysis, and side load testing provide information with respect to seismic qualification and when combined demonstrate that the subject equipment meets the seismic requirements section of Ebasco specification CAR-SH-M-66M.

Valve Size	Valve Model	ANSI Rating	Valve Tag No.
3"	40000 Series	900	3AF F1SA-1
			3AF F2SA-1
			3AF F3SA-1

The following table compares the frequencies determined by modal testing with those determined by analysis:

Direction	Natural Frequency by Analysis	Natural Frequency by Test
Parallel with	15.57 Hz	19.14 Hz
Pipe Vertical	> 33 Hz	> 33 Hz
Perpendicular to Pipe	16.30 Hz	19.14 Hz

3.9C.11.2 Stress Analysis Results

Item	Stress (psi)	Allowable (psi)	Safety Factor
Valve Body			See Note 1
Bonnet - SH	14649	25500	1.74
SR	8349	25500	3.05
ST	14465	22500	1.76
Body Bonnet Bolts	----	----	See Note 2
Clamp Nut	11990	26400	2.20
Threads	2024	17600	8.70
Actuator	3377	15000	4.44

Note 1: Valve body is shown to be adequate by meeting the pressure temperature rating of ANSI B16.34 and the area and section modulus ratio requirements of ASME III, Paragraph ND-3521.

Note 2: Body to Bonnet Bolting is shown to be adequate demonstrating that the actual bolt area is greater than the required bolt area determined by analysis in accordance with ASME III, Appendix XI.

3.9C.11.3 Conclusion

Results of the seismic analysis indicated that the modal acceleration experienced by the actuator using the supplied response spectra is 1.5g. Actual testing was performed utilizing a 4.0g equivalent static load. The purpose of applying the increased loading was to provide added confidence and assurance that the equipment could withstand the postulated seismic environment. The subject valves are seismically qualified and meet the requirements of ASME Section III and Ebasco Specification CAR-SH-M66M.

3.9C.12 TARGET ROCK

3.9C.12.1 Introduction

The purpose of this Design Report is to provide verification that Target Rock Corporation (TRC) solenoid operated globe valve Model 79Q (Type Y and Type T) complies with the requirements of the Ebasco Specification CAR-SH-M-73A, including the design requirements of the ASME Code Class 3 valves, and the dynamic analysis method of IEEE-344-1975.

Valve Size	Valve Model	Valve Tag No.
3/8”	79Q-006	2SP V23SA-1
	79Q-006	2SP V11SB-I
	79Q-006	2SP V12SA-1
	79Q-006	2SP V111SB-1
	79Q-005	2SP V113SB-1
	79Q-005	2SP V114SB-1
	79Q-005	2SB- V115SB-1
	79Q-006	2SP V2SA-1
	79Q-006	2SP V1SB-1
	79Q-006	2SP V21SN-1
	79Q-006	2SP V22SN-1
	79Q-009	2SP V90SB-1
	79Q-009	2SP V91SB-1
	79Q-005	2SP V116SA-1
	79Q-005	2SB- V120SA-1
	79Q-005	2SP V121SA-1
	79Q-005	2SP V86SB-1
	79Q-005	2SP V85SB-1
	79Q-005	2SP V81SB-1
	79Q-005	2SP V80SB-1
	79Q-005	2SP V122SA-1
2"	79Q-005	2SW V652SB-1
	79Q-005	2SW V649SA-1
1"	79Q-008	3CX V2281SB-
	79Q-008	3CX V2280SA
	79Q-008	3CX V2283SB-
	79Q-008	3CX V2282SA
	79Q-018	3SA-V301SA
	79Q-018	3SA-V362SB-
	79Q-018	3SA-V366SB-
	79Q-008	3SW V808SA
	79Q-008	3SW V869SB-

3.9C.12.2 Summary and Conclusion

The report contains the stress analysis of the valve body and bonnet. Included there in are connecting pipe strength comparison data, minimum wall thickness, combined seismic and pressure stresses, natural frequency for extended parts, and operability analysis. The analyses performed demonstrate that the TRC valves, Model 79Q (Type Y and Type T) comply with the design requirements of Ebasco Specification CAR-SH-M73A, therefore, the valves are acceptable.

The following tables are indicated results of the stress analysis data. Table 3.9C.12-1 for Type-Y solenoid valves and Table 3.9C.12-2 for Type-T solenoid valves.

3.9C.13 BIF

3.9C.13.1 Introduction

The purpose of the BIF Analysis Report is to determine that identical size valves, material and construction with different electrical actuations are qualified to meet the seismic requirements of Ebasco Design Specification CAR-SH-BE-35.

The calculated natural frequency for each subject valve was determined to be greater than 33 Hz.

Size	Actuation Model	Valve Tag No.
20"	SMC 04 5/HIBC	3AV-B1SA
		3AV-B2SA
		3AV-B4SB
		3AV-B5SB
6"	SMC 04 2/HOBC	3AV-B3SB
		3AV-B6SA
16"	SMB00 10/H2BC	3CZ-B1SA
		3CZ-B2SB
12"	SMB00 5/HOBC	3CZ-B3SA
		3CZ-B12SB

In accordance with Ebasco Specification which specified that the equipment shall be seismically qualified for 3.0g, in both OBE and SSE Conditions, BIF Report indicates that the seismic load factors used are as follows:

OBE: Horizontal = 3g Vertical = 4g (1g due to self weight)

SSE: Horizontal = 3g Vertical = 4g (1g due to self weight)

BIF addressed operability during SSE by maintaining stress intensities that are below design allowables.

3.9C.13.2 Summary and Conclusion

The stresses developed under the combined loading were analyzed for critical parts of the equipment such as support brackets, bolts, bracket plates, welds, disc and shaft. The subject parts were all within the appropriate allowable value as per ASME Section III 1977 Edition through winter 1978.

The valve motor operators were manufactured by Limitorque. These operators are representative of the prototype units that were successfully seismically tested in accordance with Acro Nav Laboratories, Report Nos. 5770, 5771, 5772, 5773, 5774, and 5 6167 5. These reports show the operators comply with the intent of IEEE Standard 344-1975 requirements.

3.9C.14 ITT/HAMMEL DAHL

3.9C.14.1 Introduction

The purpose of the report is to summarize the seismic qualification program that was performed to demonstrate the structural adequacy of the Seismic Class 1 control valve assemblies to the design criteria of the ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components, and to seismic frequency criteria identified in the Ebasco Specification CAR-SH-M66H.

The qualification of the valve assembly was performed using classical strength of material theory along with finite element analysis (SAP IV).

Where the valve assembly, supported only by the valve nozzle, has a fundamental natural frequency less than 33 Hz, an actuator support is required. Therefore, the valve assemblies analyzed include the effects of an external support to the actuator.

ITT/Hammel has performed a seismic qualification for the various size control valves as shown in Tables 3.9C.14-1 through 3.9C.14-20.

3.9C.14.2 Summary Conclusion

The results of the resonant search test indicate that the structural natural frequency of the valve assembly is greater than 33 Hz, therefore, it is classified as rigid. The seismic loading of the valve assembly in a vertical upright position was determined using the same model that was used to determine the natural frequency.

The worst case force and moment distribution due to dead weight, a 3 g static load acting in one horizontal direction, a 3 g static load acting in the other horizontal direction, and a 2 g static seismic load acting in the vertical direction simultaneously for the faulted condition was determined in Load Case 1. Load Case 2 was determined as above for the upset and emergency condition except only one-half of the SSE load was used.

3.9C.15 ALLIS CHALMERS

3.9C.15.1 Introduction

The purpose of this report is to demonstrate that this 8'x10' rectangular butterfly valve, supplied for the Emergency Service Water Intake Structure will withstand indicated service conditions without loss of function. The seismic analysis was accomplished by static seismic analysis using a lumped mass system to represent the component weight to the section analyzed. As per Ebasco Specification CAR-SH-M78, Floor Spectra.

Size	Operator	Tag Number
8'x10'	Limitorque	3SW B3SA-1

H6BC/SMB3-80 3SW B4SB-1

3.9C.15.2 Summary and Conclusion

The calculation for the valve and mounting of the operator are based on the general practice of the valve industry and AWWA Specification C 504 for butterfly valves, or the requirements of ASME Section III and ANSI B16.34 where applicable. The valve disc shafts, body, mounting bracket, pins, bolts, and bracket welding have all been reviewed.

The design stresses are within allowable stresses given in ASME Code Section IV. The valve, the mounted operator and bolting meet code requirements, where applicable.

Calculations shown are for valve with 30 psi design pressure and 125°F max temperature.

Body features exceed requirements, weight and center of gravity for the mounting bracket and operator are listed in the calculation for the bolting and for the critical section in the mounting bracket.

The valves motor operator was manufactured by Limitorque. The operator is representative of the prototype unit that were successfully seismically tested in accordance with Aero Mav Laboratories, Inc. Report No. 5770, 5771, 5772, 5773, 5774, and 5 6167 5. These reports show the operators comply with the intent of IEEE Standard 344-1975 requirements.

The results of the static analysis provide the natural frequency and stresses as shown in Table 3.9C.15-1.

APPENDIX 3.9D INSERVICE PUMP AND VALVE TESTING PROGRAM

IS REPLACED BY HNP-IST-003, HNP IST PROGRAM PLAN - 3RD INTERVAL

3.10 SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC QUALIFICATION CRITERIA

3.10.1.1 Equipment Supplied by the NSSS Vendor

This section presents information that demonstrates safety related instrumentation and electrical equipment requiring seismic qualification is capable of performing designated safety related functions in the event of an earthquake. The information presented includes identification of safety related electrical equipment requiring seismic qualification that is within the scope of the Westinghouse Nuclear Steam Supply System (NSSS), qualification criteria employed for each item of equipment, definition of the applicable seismic environment, and documentation of the qualification process employed to demonstrate the required seismic capability.

3.10.1.1.1 Qualification standards

The methods of meeting the general requirements for seismic qualification of safety related instrumentation and electrical equipment as described by General Design Criteria (GDC) 1, 2, and 23 are described in Section 3.1. The general methods of implementing the requirements of

Appendix B to 10 CFR Part 50 are in accordance with the Engineering and Construction QA Program approved by the NRC during the Construction Permit review.

The qualification and documentation procedures used for equipment and supports which were purchased prior to March 1, 1977, are in compliance with IEEE-344-1971 and Standard Review Plan 3.10 (Revision 1), Section II.1.a or the Supplemental Qualification Program (Reference 3.10.2-2). The qualification and documentation procedures for equipment and supports purchased on or after March 1, 1977, are in compliance with IEEE-344-1975. A historical list of all safety related instrumentation and electrical equipment requiring seismic qualifications within Westinghouse NSSS scope is provided in Table 3.10.1-1. The list of safety related equipment in Table 3.10.1-1 is historical and is not updated. Safety related equipment requiring seismic qualification is maintained in the equipment database. Qualification information is readily available from the seismic qualification reports, vendor drawings, record management system, and equipment database.

3.10.1.2 Equipment Supplied by Other than NSSS Vendor

The safety related electrical (includes instrumentation and control) and mechanical equipment and their supports which are not in the NSSS scope, have been qualified by testing and/or analysis to Seismic Category I requirements to verify their ability to withstand the effects of earthquakes and other applicable accident-related loadings (i.e., dynamic loadings).

In addition, the qualification and documentation procedures for Seismic Category I electrical equipment and their supports have been prepared utilizing the guidance of IEEE-344-1975 and Regulatory Guide 1.100. Such equipment and supports which are Class 1E are qualified in accordance with IEEE-323, as discussed in Section 3.11. IEEE-344 is considered ancillary to IEEE-323 and any exceptions taken by equipment vendors to age testing requirements have been evaluated and accepted when the vendors provided acceptable justification for the exception. A historical list of safety related electrical and mechanical equipment requiring seismic qualification within the non-NSSS scope is provided in Table 3.10.1-2. The list of safety related equipment in Table 3.10.1-2 is historical and is not updated.

Safety related equipment requiring seismic qualification is maintained in the equipment database. Qualification information is readily available from the seismic qualification reports, vendor drawings, record management system, and equipment database.

For Class 1E electric equipment located in a harsh environment, the aging and test sequence aspects of seismic qualification are based on the requirements of IEEE-323 and are addressed in Section 3.11. Also addressed in Section 3.11 are the environmental effects on non-metallic subcomponents of safety related mechanical equipment located in a harsh environment.

For equipment located in mild environment as defined in 10 CFR 50.49, consideration of aging aspects of environmental qualification are, as a minimum, assured by SHNPP compliance with the general quality and surveillance requirements applicable to electric equipment in accordance with other regulations such as 10 CFR 50, Appendix B.

Purchase specifications for mechanical and electrical equipment require vendors to establish testing and analysis procedures in order to substantiate the required performance of the equipment. The vendors have accomplished the seismic qualification either by analyzing the equipment mathematically, by an actual test with simulated seismic conditions, or by a combination of both. The choice of method used for seismic qualifications has been based on practicality, size, shape, and complexity of the equipment as well as the reliability of the conclusions. Both methods are fully described in IEEE-344 and/or Standard Review Plan 3.10. The purchase specifications define the options available to the vendor under these standards and require compliance with them. The vendor must demonstrate that the equipment is functionally operable and/or maintains its structural integrity as applicable, when subjected to the response spectra/accelerations and other loads specified in the purchase specifications.

Combinations of seismic loads and the relevant dynamic and static loads are utilized in the qualification program. The load combinations for mechanical equipment are summarized in Section 3.9. Electrical equipment is subjected to seismic, operating and dynamic loads caused by the postulated transients as applicable. Equipment is acceptable only if it can sustain the stresses and distortions induced by these loads and still function as required.

The results of seismic qualification analysis and/or tests performed on equipment have been furnished by the vendors in report form. In some limited cases where detailed vendor information, including proprietary data, is maintained at the vendor's facility, certifications to the applicable SHNPP seismic requirements have been furnished.

Documentation prepared for the operating license review is described in Section 3.10.4.

The seismic qualification of Seismic Category I equipment supports is described in Sections 3.7.3 and 3.10.3.

3.10.2 METHODS AND PROCEDURES FOR SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.10.2.1 Equipment Supplied by the NSSS Vendor

Seismic qualification of safety related electrical equipment is demonstrated by either type testing, analysis or a combination of these methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including: practicability, complexity of equipment, economics, availability of previous seismic qualification to earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the individual equipment qualification reference.

3.10.2.1.1 Equipment Qualified in Compliance with IEEE-344-1971 and Standard Review Plan 3.10 (Revision 1), Section II.1.a

a) Type Test - From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE-344-1971 to seismically qualify equipment. Much of this early testing was reported in WCAP-7817 and WCAP-7821 as referenced in Table 3.10.1-1. The input form selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported in Reference 3.10.2-1. Further, this input has been justified with respect to the methods of IEEE-344-1975 and documented in Reference 3.10.2-3. In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (Reference 3.10.2-2). This retesting was performed at the request of the NRC staff on agreed selected items of equipment employing multi frequency, multi-axis test inputs (Reference 3.10.2-4) 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.

b) Analysis - The structural integrity of safety related motors is demonstrated by a static seismic analysis in accordance with IEEE-344-1971, with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, gravitational and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas; 1) maximum rotor deflection, 2) maximum shaft stresses, 3) maximum bearing load and shaft slop 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 the qualification reference.

3.10.2.1.2 Equipment Qualified in Compliance with IEEE-344-1975

a) Type Test - The original single-axis sine beat testing and the additional retesting completed under the Supplemental Test Program have been the subject of generic review by the NRC staff. Both test programs are described later in this section. 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 NRC staff, the Supplemental Qualification Program (Reference 3.10.2-2), no additional qualification testing is required to demonstrate acceptability to IEEE-344-1975, provided that:

    1) The Westinghouse aging evaluation program for aging effects on complex electronic equipment located outside Containment demonstrates there are not deleterious aging phenomena. In the event that the aging evaluation program identifies materials that are marginal, either the materials will be replaced or the projected qualified life will be adjusted.

    2) Any changes made to the equipment due to a) above or due to design modifications do not significantly affect the seismic characteristics of the equipment.

    3) The previously employed test inputs can be shown to be conservative with respect to applicable plant specific response spectra. The equipment that requires no additional testing is identified in Reference 3.10.2-5, Table 7.1 and the test results in the applicable EODP's of Reference 3.10.2-6.

For equipment tests after July, 1974 (i.e., new designs, equipment not previously qualified, or previously qualified equipment that does not meet 1), 2), and 3), above), seismic qualification by test is performed in accordance with IEEE-344-1975. Where testing is utilized, multi-frequency multi-axis inputs are developed by the general procedures outlined in Reference 3.10.2-4. The test results contained in the individual EODP's of Reference 3.10.2-6 demonstrate that the measured test response spectrum envelopes the applicable required response spectrum (RRS) defined for generic testing as specified in Section 1 of the EODP (Reference 3.10.2-6). Qualification for plant specific use is established by verification that the generic RRS specified by Westinghouse envelopes the applicable plant specific response spectrum. Alternative test methods, such as single frequency, single axis inputs, are used in selected cases as permitted by IEEE-344-1975 and Regulatory Guide 1.100.

b) Analysis - The structural integrity of safety related motors (Reference 3.10.2-6, Table 3.10.1-1 EQDP-AE-2 and 3) is demonstrated by a static seismic analysis in accordance with IEEE-344-1975, with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, gravitational and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas; 1) maximum rotor deflection, 2) maximum shaft stresses, 3) maximum bearing load and shaft slope at the bearings, 4) maximums tresses 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 EQDP's (Reference 3.10.2-6).

3.10.2.2 Equipment Supplied by Other than NSSS Vendor

The purchase specifications for safety related equipment contain seismic input data for the safe shutdown earthquake (SSE) and operating basis earthquake (OBE). This input data consist of floor response spectra (for the appropriate damping values) and/or appropriate "G" values for the various levels of the building (taking into consideration the location of the equipment on the floor) or other mounting locations such as pipes, ducts, etc. Each set of curves consists of two horizontal and one vertical design response spectra curves at the floor elevation of the equipment mounting location.

A description of the seismic analysis of subsystems (i.e., piping, ducts, cable trays, etc.) is given in Section 3.7.3.

The equipment supplier's seismic qualification program demonstrates the ability of the equipment to perform its required function during and after the time it is subjected to the forces resulting from the application of five consecutive OBE's followed by one SSE, with a proper combination of other applicable concurrent loads.

Depending upon the practicability of the method for the type, size, shape, and complexity of the equipment and the reliability of the conclusion, the equipment supplier uses testing, analysis, or a combination of testing and analysis as a method of qualification, as follows:

a) Testing - Testing has been the preferred method of qualification. It is performed by subjecting the equipment to vibratory motions, which conservatively simulate the OBE and SSE responses at the equipment mounting locations. The SSE test is preceded by five events of the OBE. The test input motions are such that the resulting response spectra envelope the design floor response spectra.

Thermal and radiation aging is performed prior to seismic testing for equipment qualified in accordance with IEEE-323-1974 unless it could be justified by the supplier that the equipment would not approach the end-of-life condition during the installed life, when subjected to the specified service conditions.

Instrumentation and electrical equipment are tested in the operational mode and their operability is verified before, during, and after the testing. Test methods described in Section 6.6 of IEEE-344-1975 are utilized to perform the required qualification testing. The test input motion has generally been of the random type.

b) Analysis - Analysis without testing is accepted when the equipment functional operability can be assured by its structural integrity alone. The procedures described in Sections 5.2 through 5.4 of IEEE-344-1975 are utilized. Component fatigue is checked for the effect of five OBEs and one SSE when the analysis method of qualification is used.

c) Combination of Testing and Analysis - When the equipment cannot be qualified by testing or analysis alone because of its size and complexity, a combined testing and analysis method is utilized. Methods described in Section 7 of IEEE-344-1975 are used for qualification.

3.10.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS FOR MECHANICAL AND ELECTRICAL EQUIPMENT AND INSTRUMENTATION

3.10.3.1 Equipment Supplied by the NSSS Vendor

Where supports for the electrical equipment and instrumentation are within the Westinghouse NSSS scope of supply, the seismic qualification tests and/or analyses are conducted, including the supplied supports. The equipment qualification references identify the equipment mounting employed for qualification purposes which establish interface requirements for the equipment to ensure subsequent in-plant installation does not prejudice the qualification established by Westinghouse.

3.10.3.2 Equipment Supplied by Other than the NSSS Vendor

For non-NSSS equipment, analysis and/or testing was performed for supports of safety related mechanical and electrical equipment to verify their structural capability to withstand a postulated seismic event.

Generally, a three dimensional model of a given support was used to perform a seismic response analysis. The effect of three components of the earthquake on a support was considered simultaneously and the results (stress, displacement, and deformation at the location of interest) were evaluated by the square root of the sum of the squares method (SRSS). Detailed descriptions of the analysis methods utilized are provided in Section 3.7.3.

In the qualification of supports by tests, generally a random seismic input motion was used. During the test either a dummy load or the actual equipment was mounted on the support in order to closely simulate the dynamic behavior of the support.

The supports, where practicable, were rigidly designed such that there was no dynamic amplification from the supports and therefore the floor response spectra were directly utilized as seismic input criteria to seismically qualify the devices or components which are mounted on the support. Where rigid supports were not feasible, amplification of floor response due to support flexibility was accounted for in the qualification of supported equipment. A detailed description is provided in Section 3.7.3.

3.10.4 OPERATING LICENSE REVIEW

3.10.4.1 NSSS Equipment

The individual qualification references listed in Table 3.10.1-1 provide the results of tests and analyses performed to verify the criteria established in Section 3.10.1.1, employing the qualification methods described in Sections 3.10.1.2 and 3.10.1.3.

3.10.4.2 Non-NSSS Equipment

The Seismic Category I equipment is designed and qualified to perform its safety related function during and after the SSE. The documentation of the methods and results of the tests and analyses for the non-NSSS electrical and mechanical equipment demonstrating proper implementation of the criteria established during the Construction Permit (CP) review and verifying that all applicable loads have been properly defined and accounted for consists of the following:

a) SQRT Master List - A list of safety related equipment and components for which seismic qualification by test and/or analysis was required. The format of the tabulation was essentially the same as suggested in Reference 3.10.1-1. A sample of the SQRT Master List format is given in Figure 3.10.1-1.

b) Seismic Qualification Reports - Reports that were submitted by the vendors include the results of seismic tests and/or analyses to verify and document that the requirements of applicable codes and standards, as outlined in the purchase specifications, were met and that the equipment is capable of performing its intended design functions under the specified seismic and dynamic loads. Test reports include a description of the test facility, test procedures, test results and conclusions. Analysis reports include a description of the analysis.

   Reports were approved and signed, documenting that the equipment has met its performance requirements when subjected to the specified seismic accelerations and that it met the requirements of IEEE-344.

c) Seismic and Dynamic Qualification Summaries - Seismic and dynamic qualification summary information was provided for mechanical and electrical equipment selected from the NRC from the SQRT master list. The format of the seismic and dynamic qualification summaries followed the suggested format in Reference 3.10.1-1.

The seismic qualification documentation described in a, b and c, above has been processed as QA records in accordance with Section 17.3 "HNP Quality Assurance Program Description." In addition, supplementary documentation (i.e., specifications, pertinent drawings, etc.) has also been processed as QA records in accordance with Section 17.3 and is readily available for audit.

Seismic qualification of safety related mechanical and electrical equipment as described in Section 3.10 is maintained through corporate and site procedures. Seismic qualification reports are updated and maintained current as equipment is replaced, further tested or otherwise further qualified. The seismic qualification reports are retrievable through the site record management system.

The SQRT master list and seismic and dynamic qualification summaries were compiled for operating license review and are not updated. Distinction between NSSS vendor scope and non-NSSS vendor scope is listed in FSAR Tables 3.10.1-1 and 3.10.1-2. Qualification information is readily available from the seismic qualification reports, vendor drawings, record management system, and equipment database.

REFERENCES: SECTION 3.10

3.10.1-1 Letter dated February 2, 1983 from G. W. Knighton (NRC) to Mr. E. E. Utley (CP&L), "Request for Additional Information for Seismic and Dynamic Qualification Review for Shearon Harris 1 and 2."

3.10.2-1 Morrone, A., "Seismic Vibration Testing With Sine Beats," WCAP-7558, October 1971.

3.10.2-2 NS-CE-692, Letter dated July 10, 1975 from C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC).

3.10.2-3 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.

3.10.2-4 Jarecki, S. J., "General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary), September 1975 and WCAP-8695 (Non-Proprietary), August 1975.

3.10.2-5 Butterworth, G. and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 2, February 1979.

3.10.2-6 "Equipment Qualification Data Packages," Supplement 1 to WCAP 8587, November 1978.

3.11 ENVIRONMENTAL DESIGN OF ELECTRIC AND MECHANICAL EQUIPMENT

3.11.0 GENERAL

Equipment that is relied on to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria 1, 2, 4, and 23 of Appendix "A" and Sections III and XI of Appendix "B" to 10 CFR 50, is applicable to equipment located inside and outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability have been set forth in 10 CFR 50.49.

The purpose of this section is to provide information on the environmental conditions and design bases for which safety related electrical and mechanical equipment is designed to ensure compliance with the above. In addition, this section describes the applicants' environmental qualification program and methodology for compliance with NUREG-0588 Category II guidelines and therefore 10 CFR 50.49.

This section consists of a written description, tables, figures, appendices, and data references describing the equipment qualification for safety-related Class IE components used in the plant. Descriptions of these tables, figures, appendices, and references are as follows (Tables 3.11.0-1, 3.11.0-2, and 3.11.0-3 were compiled for operating license review and are not updated):

Table 3.11.0-1 - This table lists the NSSS supplied safety-related equipment with the applicable qualification reference indicated.

Table 3.11.0-2 - This table lists the Ebasco supplied safety-related equipment.

Table 3.11.0-3 - This table lists the CP&L site supplied safety-related equipment.

Table 3.11.1-1 - This table defines the location codes used in the Master List.

Figure 3.11.1-1 - This figure provides the format and legend for the SHNPP "Master List."

Figure 3.11.1-2 - This figure provides a legend for the SHNPP Component Evaluation Sheet.

Appendix 3.11A - This appendix contains the NUREG-0588 Comparison.

Appendix 3.11B - This appendix addresses the Containment and balance of plant area's Zone Maps for temperature, radiation, pressure, and humidity.

Appendix 3.11C - This appendix contains supplemental analyses and their results used to demonstrate the thermal response of safety-related equipment located inside Containment, and the subsequent ability to survive and operate during and after the design basis accident.

Appendix 3.11E - This appendix contains supplemental analysis and results used to demonstrate the thermal response of safety-related equipment located inside the main steam tunnel and the subsequent ability to survive and operate during and after a main steam line break (MSLB).

WCAP-8587, Supplement No. 1 - This qualification reference indicates the individual qualification details for each particular type of equipment, meeting IEEE-323-1974, supplied by the NSSS Vendor, Westinghouse. This WCAP and supplement are not contained in the FSAR and are generic reference documents for all NSSS supplied IE equipment meeting IEEE-323-1974.

WCAP-7410-L, WCAP-7744 and the Westinghouse Environmental Supplemental Qualification Testing Program (see Westinghouse Letter NS-CE-692, C. Eicheldinger to D. B. Vassallo, July 10, 1975, and NRC Letter from D. B. Vassallo to C. Eicheldinger, November 19, 1975) - This qualification reference indicates the qualification details for equipment supplied by the NSSS Vendor, Westinghouse which meets IEEE-323-1971.

These WCAPs and the Supplemental Program are not contained in the FSAR and are generic reference documents for all NSSS supplied IE equipment meeting IEEE-323-1971.

The design environmental criteria for safety-related electrical and mechanical equipment are based on equipment location. Radiation Environment for qualification of electrical and mechanical equipment is based on radiation doses calculated using source terms and methodology discussed in NUREG-0588, NUREG-0588 Rev. 1, and Section II-B.2 of NUREG-0737. As far as practical, equipment for these systems is located outside the Containment Building or other areas where high radioactivity levels or adverse environmental conditions could exist under normal, test, or accident conditions.

Safety-related equipment is capable of performing its intended functions under the following specified environmental conditions:

a) All safety-related components are capable of meeting their rated performance specifications under the environmental service conditions expected as a result of normal operating requirements, including the range of expected minimum and maximum environmental conditions.

b) All safety-related equipment is capable of completing its functions under the environmental service conditions related to the design basis accident. The environmental service conditions related to a design basis accident are specified to include: normal operating conditions existing before the event, conditions generated by the event, and conditions which exist subsequent to the event for such time as is required for the protective actions to be carried to completion.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

3.11.1.1 Equipment Identification

The methodology to determine which equipment important to safety is to be environmentally qualified is based on the IE Bulletin 79-01B approach of reviewing plant systems which perform safety functions. The equipment within such systems, which are necessary for the performance of the safety function, are identified and qualified environmentally to demonstrate acceptable performance throughout its installed life.

Plant safety related systems are identified in FSAR Table 3.2.1-1. The specific equipment, within safety related systems, which is environmentally qualified, is identified on separate master lists submitted to the NRC. All equipment defined in the scope of 10 CFR 50.49 is included in the Shearon Harris EQ Program.

3.11.1.2 Environmental Conditions

Normal and accident environmental conditions are explicitly identified in various FSAR sections. FSAR Section 3.11B addresses environmental conditions used for qualification purposes.

SHNPP has in place an area temperature monitoring program to ensure that normal operating temperature limits are not exceeded and safety-related equipment is not subjected to temperatures in excess of their environmental qualification temperatures. Area temperature monitoring is implemented in accordance with site procedure PLP-114 "Relocated Technical Specifications and Design Basis Requirements".

3.11.2 QUALIFICATION TESTS AND ANALYSIS

Environmental qualification testing and/or analysis based on tests are performed on safety related equipment located in a harsh environment. The results are evaluated for compliance with the Category II NUREG-0588 guidelines.

Nuclear Steam Supply System (NSSS) Class 1E equipment is qualified under the Westinghouse environmental qualification program as stated in Westinghouse Topical Report WCAP 8587. This report describes the basic methodology on which the Westinghouse qualification program is based and includes qualification methods used for harsh environment Class 1E equipment.

The NRC has reviewed and accepted the generic qualification methodology described in Westinghouse Topical Report 8587. The applicants review the report to verify applicability to Shearon Harris.

Specifically, all reviews consider but are not limited to the following:

a) Assurance that the test report is applicable to SHNPP. This is accomplished by assuring that the project name, purchase order and equipment specification as a minimum are identified on or traceable to the report.

b) A comparison of the test sample is made to assure that the equipment tested is identical to or representative of the purchased equipment.

c) The aging (radiation, humidity, temperature, electro-mechanical cycling, etc., as required) simulation is evaluated to determine if the test equipment has been placed in a condition which simulates its expected end of qualified life condition prior to design basis accident testing. Process temperatures, when applicable, are addressed.

d) The design basis accident environmental test conditions (temperature, pressure, chemical spray, etc.) are evaluated to determine if they envelop the Shearon Harris expected environmental conditions in the unlikely event of a design basis accident.

e) Anomalies observed during qualification testing are evaluated.

In addition, other items such as test sequence, margin, and interfaces are also addressed during the environmental qualification report review process.

Compliance with the various NRC Regulatory Guides and General Design Criteria is described in FSAR Sections 1.8 and 3.1, respectively.

3.11.3 QUALIFICATION TEST RESULTS

A summary of the harsh environment qualification test results for each type of qualified safety related equipment is provided in the Environmental Qualification Document Package (EQDP) for each equipment. Documentation packages are prepared for equipment groups by type and manufacturer (e.g., all Target Rock Solenoid Operator harsh environment qualification documents are contained in a single documentation package).

Typical documents which are addressed in the environmental qualification documentation packages are:

a) Equipment List,

b) Qualification Analysis,

c) Qualification Document Assessment,

d) Equipment Aging Information,

e) Operating Experience Data,

f) Maintenance Requirements, and,

g) Qualification Test Reports.

The various documentation packages are permanently stored and maintained at the Shearon Harris Nuclear Power Plant.

3.11.4 LOSS OF VENTILATION

3.11.4.1 Equipment Qualification

Plant areas containing safety-related equipment and their support systems are provided with temperature controlled environment during normal and worst DBA conditions if required. The maximum environmental parameters for different plant areas are shown in Appendix 3.11B for both normal and post-accident plant operations. The HVAC systems serving the spaces containing the safety-related equipment for accident operation, and requiring a temperature controlled environment, are also safety related and Class 1E qualified in their design. Qualification details of the safety-related HVAC equipment and components are referenced in the "Master List" and presented in the "Component Evaluation Sheets".

3.11.4.2 Air Conditioning Systems

During normal plant operation, both safety and non-safety air conditioning equipment provide the design environment in different plant areas. During accident conditions, only safety-related air conditioning systems provide filtering, cooling, and recirculation of air to maintain the proper environment, where required, inside spaces which house the safety-related equipment and components. Cooling coils in air conditioning systems use either Service Water or Essential Services Chilled Water both of which are safety-related systems.

The Seismic Category I, Safety Classes 2 and 3 Air Conditioning Systems, are powered from Class 1E electrical power supplies and are provided as described in Section 9.4. They are designed such that the single failure of an active component during and after a design basis accident does not result in complete loss of ventilation and cooling of the area requiring a temperature controlled environment, or affect the ability of the safety-related systems served by the air conditioning equipment to fulfill their safety functions. Should the air conditioning unit in one of the rooms containing a Seismic Category I, Safety Class 2 or 3 system become inoperative during normal or accident operation, redundant equipment is still available to mitigate the consequences of a design basis accident. The temperature and humidity inside the Control Room envelope are controlled at all times to assure a proper environment for personnel, equipment, instruments and controls located within.

3.11.4.3 Ventilation Systems

Plant ventilation systems provided include both safety and nonsafety systems. Cooling is accomplished in some areas by using 100% outside air.

Two redundant Safety Class 3, Seismic Category I fan coolers are provided where required in the Reactor Auxiliary Building for the spaces containing safety-related equipment. The system design assures that proper ambient temperature is maintained at all times. It is not considered credible that simultaneous loss of the two redundant units could occur.

Humidity is not controlled during accident conditions in most areas, except in the Control Room, and 100 percent humidity is assumed in these areas unless otherwise indicated.

3.11.4.4 Design Basis Temperatures

The maximum temperatures considered in the sizing of ventilation and cooling systems serving safety-related systems were determined considering the following factors:

a) Maximum outdoor design temperatures for the geographical area of the plant (both wet-bulb and dry-bulb readings) per ASHRAE standards.

b) Maximum internal piping thermal loads, if applicable, for the particular space or room, using maximum operating temperatures of the pipe contents and design lengths of those pipes for each mode of operation.

c) Maximum internal electrical load from lighting, electrical cables, trays and equipment.

d) Maximum heat transfer from miscellaneous equipment surfaces.

e) Maximum heat transfer from the surfaces of open pools and tanks, using the maximum operating temperature of the contents.

f) Maximum heat transfer through floor and ceiling or roof from and to the adjoining spaces.

3.11.4.5 Temperature Conditions Inside Containment and Main Steam Tunnel During/After a Design Basis Accident

The temperature conditions inside the Containment or Main Steam Tunnel resulting from a design basis accident are a function of time until steady state conditions are established and are discussed in Section 6.2.1 and Section 3.6A respectively. For the purposes of equipment qualification, these conditions are separated into different time periods as shown in the following figures:

3.11.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

3.11.5.1 Chemical Environment

Safety Related Systems are designed to perform their safety-related functions in the temperature, pressure, and humidity conditions discussed in Section 3.11.1 and in Section 6.2. In addition, components of ESF systems inside the Containment are designed to perform their safety-related functions in a long-term contact with boric acid and sodium hydroxide solutions, recirculated through the Safety Injection System (SIS) and Containment Spray System (CSS).

The pH time history of the water both in the containment spray and in the containment sump, as well as the boron concentration in the Reactor Coolant System, is discussed in Section 6.5.2.

The containment atmosphere is maintained below 4 volume percent hydrogen consistent with the recommendations of Regulatory Guide 1.7. The extent to which this and other recommendations of Regulatory Guide 1.7 are followed are discussed in FSAR Section 6.2.5.

The CVCS, SIS, and CSS are designed for both the maximum and long-term boric acid concentration of 2400-2600 ppm at a pH of 7.0 to 11.0. (This is the most severe caustic spray environment resulting from the addition of 30% weight sodium hydroxide.)

3.11.5.2 Radiation Environment

Safety related systems and components are designed to perform their safety related functions after the normal operational exposure plus one accident exposure. The normal operational exposure is based on the design source terms presented in Section 11.1 and Section 12.2.1. Post-accident system and component radiation exposures are dependent on equipment location. Source terms and other accident parameters are presented in Section 12.2.1 and in Chapter 15. For safety related systems, normal operational exposure and post-accident radiation exposures are listed in Appendix 3.11B.

The degree to which the recommendations of Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-coolant accident for Pressurized Water Reactors," has been used in determining the source terms used in evaluating radiation exposure is detailed in Section 1.8.

The design radiation exposures are based on gamma and beta radiation. The effects of beta radiation are effectively attenuated by small amounts of shielding, such as conduits for cable and casings for equipment. Organic materials which are located inside the Containment are identified in Section 6.1.2.

3.11.6 PRESSURE ENVIRONMENT

For the purpose of equipment qualification, the range of normal operating pressure inside containment is -1" Wg. to +52.7" Wg. (1.6 psig).

Other plant areas have a normal operating pressure range of approximately -1/4" Wg. to + 1/4" Wg.

Design Basis Accident pressure conditions inside containment are discussed in Section 6.2.1 and the main steam tunnel accident pressure conditions are discussed in Section 3.6A. For the purpose of equipment qualification, the accident pressure conditions have been separated into different time periods as shown in the following figures:

Figure 3.11.6-1 Pressure Profile Inside Containment (Combines LOCA/MSLB)

Figure 3.11.6-2 Pressure Profile Inside Containment (MSLB)

Figure 3.11.6-3 Pressure Profile Inside Main Steam Tunnel (MSLB)

In all other plant areas, during a design basis accident inside containment or the main steam tunnel, the pressure remains at the initial atmospheric condition.

3.11.7 ENVIRONMENTAL QUALIFICATION OF MECHANICAL EQUIPMENT

Safety-related mechanical equipment is environmentally qualified in accordance with the requirements of General Design Criteria 1 and 4 of 10 CFR 50, Appendix A, and Sections III and XVII of 10 CFR 50, Appendix B. The components have been designed, procured, fabricated, tested and documented in accordance with the appropriate quality groups of Regulatory Guide 1.26 and the corresponding safety class indicated in Section 3.2. In addition, for those safety-related components which form a pressure boundary, the above features are in accordance with ASME Section III.

Materials not traceable to the above codes used for packing,"O" rings, diaphragms and other normal maintenance items, have been specified and selected for their specific environmental conditions. In addition, the non-metallic subcomponents of safety-related mechanical equipment located in harsh environmental areas are evaluated with respect to their capabilities under the normal and accident environmental service conditions. In conjunction with programs for surveillance, maintenance and periodic testing, the above features assure the continued ability of mechanical equipment to perform its safety functions.

APPENDIX 3.11A NUREG-0588 COMPARISON

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
1. ESTABLISHMENT OF THE QUALIFICATION PARAMETERS FOR DESIGN BASIS EVENTS	
1.1 Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)	

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(1) The time-dependent temperature and pressure, established for the design of the containment structure and found acceptable by the staff, may be used for environmental qualification of equipment.	1.1 (1) Time dependent temperature and pressure LOCA profiles are used. Refer to figures in FSAR Sections 3.11.4 and 3.11.6 and Appendix 3.11B.
(2) Acceptable methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are summarized below. Acceptable methods for calculating mass and energy release rates are summarized in Appendix A.	(2) Mass and energy release rates are consistent with those summarized in NUREG 0588 Appendix A. Refer to FSAR Section 6.2.1.3 for details.
Pressurized Water Reactors (PWRs)	
Dry Containment - Calculate LOCA containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Standard Review Plan (SRP) Section 6.2.1.1.A NUREG-75/087. The assumption of partial revaporization will be allowed. Other assumptions that reduce the temperature response of the containment will be evaluated on a case-by-case basis.	GOTHIC is used in calculating the post-LOCA containment environment. Refer to FSAR Section 6.2.1.1.3.2
Ice Condenser Containment - Calculate LOCA containment environment using LOTIC or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.B, NUREG-75/087.	SHNPP does not have an ice condenser containment; therefore, this is not applicable.
Boiling Water Reactors (BWRs)	
Mark I, II, and III Containment - Calculate LOCA environment using methods of GESSAR Appendix 3B or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.C, NUREG-75/087.	SHNPP is a PWR; therefore, this is not applicable.
(3) In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser types of plants, the generic envelope shown in Appendix C may be used for qualification testing.	(3) SHNPP is a dry containment PWR; therefore, this is not applicable.
(4) The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4) Plant-specific containment temperature and pressure profiles are used. Refer to figures in FSAR Sections 3.11.4 and 3.11.6 and Appendix 3.11B.
1.2 Temperature and Pressure Conditions Inside Containment – Main Steam Line Break (MSLB)	

<u>CATEGORY II</u> Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(1) Where qualification has not been completed, the environmental parameters used for equipment qualification should be calculated using a plant-specific model based on the staff-approved assumptions discussed in Item 1 of Appendix B.	1.2 (1) A plant-specific analysis consistent with the requirements of NUREG 0588, utilizing CONTEMPT-LT 28 as noted in FSAR Section 6.2.1.1.3.3, has been used to determine the temperature and pressure conditions inside containment for a MSLB.
(2) Other models that are acceptable for calculating containment parameters are listed in Section 1.1(2).	(2) See 1.2 (1) above.
(3) In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser plants, the generic envelope shown in Appendix C may be used.	(3) SHNPP is a dry containment PWR; therefore, this is not applicable. See 1.1 (1) above.
(4) The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4) Plant-specific containment temperature and pressure design profiles are used. Refer to 1.1 (1) above.
(5) Where qualification has been completed but only LOCA conditions were considered, then it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable: (a) Calculate the peak temperature from an MSLB using a model based on the staff's approved assumptions discussed in Item 1 of Appendix B. (b) Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in Item 2 of Appendix B. (c) If the calculated surface temperature exceeds the qualification temperature, the staff requires that (i) additional justification be provided to demonstrate that the equipment can maintain its required functional operability if its surface temperature reaches the calculated value or (ii) requalification testing be performed with appropriate margins, or (iii) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature.	(5) In general, combined MSLB/LOCA profiles are utilized for time-dependent temperatures and pressures (Refer to FSAR Figures 3.11.4-1 and 3.11.6-1, respectively) regardless of less stringent qualification requirements; however, in those cases where the test condition profile does not envelope the applicable Shearon Harris profile, the following technique is used: - Additional justification (e.g., component thermal lag analysis using either CONTEMPT-LT28 or GOTHIC Version 6.1b [Refer to FSAR Appendix 3.11C]) is provided to demonstrate that the equipment can maintain its required functional operability or - requalification testing is performed with appropriate margins, or - qualified physical protection may be provided to assure that the equipment experiences only the conditions for which it is qualified.

<u>1.3 Effects of Chemical Spray</u>

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971		Shearon Harris Nuclear Power Plant Program
The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, Item (e) for caustic spray solution guidelines	1.3	The most severe containment spray environment (boron concentration and pH level) is used for environmental qualification. The actual (calculated) spray environment bounds any postulated single failure.
1.4 Radiation Conditions Inside and Outside Containment		
The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following which that equipment must remain functional. It should be assumed that the DBA related environmental conditions occur at the end of the equipment qualified life. The sample calculations in Appendix D and the following positions provide an acceptable approach for establishing radiation limits for qualification. Additional radiation margins identified in Section 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used.	1.4	For qualification purposes, reductions in air dose due to spray washout and plateout are used in calculating the post-accident radiation environments. Therefore, radiation doses used in qualification are maximum total integrated dose calculated over the equipment qualified life, plus that associated with the most severe design basis accident.
(1) The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For all other non-LOCA design basis accident conditions, a source term involving an instantaneous release from the fuel to the atmosphere of 10 percent of the noble gases (except Kr-85 for which a release of 30 percent should be assumed) and 10 percent of the iodines is acceptable.		(1) The source term used in all cases in determining the radiation environment is that 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products are released instantaneously from the fuel to the containment atmosphere.
(2) The calculation of the radiation environment associated with design basis accidents should take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures.		(2) Time-dependent transport of released fission products within various regions of containment and auxiliary structures is assumed in the calculation of the radiation environment associated with design basis accidents.

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(3) The initial distribution of activity within the containment should be based on a mechanistically rational assumption. Hence, for compartmented containments, such as in a BWR, a large portion of the source should be assumed to be initially contained in the drywell. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate.	(3) The initial distribution of activity within the containment is based on a mechanistically rational assumption as described in FSAR Section 12.2. Since the internal structures of the containment were designed to provide vertical compartments around each of the steam generators and the reactor vessel and since the Containment Spray and/or the containment ventilation and filtration systems provide mixing for the containment atmosphere, a determination was made to assume a uniform distribution of activity throughout the containment.
(4) Effects of ESF systems, such as containment sprays and containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 (NUREG-75/087) and the related sections referenced in the Appendices to that section.	(4) Credit for the removal of airborne activity by ESF systems has been taken. In addition, the distribution of activity is taken into account as described in (3) above and by (5) below.
(5) Natural deposition (i.e., plate-out) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50 percent instantaneous plate-out of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow of washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment.	(5) The SHNPP model assumes removal by plate-out using a mechanistic model considering elemental, particulate, and organic fractions of halogens as well as the particulate fractions of solid fission products. The SHNPP model also assumes mechanistic spray removal and dilution of the 50 percent halogen core inventory and 1 percent solid fission products inventory source terms with the combined volumes of the Reactor Coolant, Accumulators, and the Refueling Water Storage Tank. The resulting sump activity as a function of time is given on FSAR Table 12.2.1-26.
(6) For unshielded equipment located in the containment, the gamma dose and dose rate should be equal to the dose and dose rate at the centerpoint of the containment plus the contribution from location dependent sources such as the sump water and plate-out, unless it can be shown by analyses that location and shielding of the equipment reduces the dose and dose rate.	(6) The gamma dose and dose rate used in qualification for equipment located inside containment is calculated for various zones utilizing distance and shielding credits. Refer to FSAR Appendix 3.11B for applicable doses in various zones.
(7) For unshielded equipment, the beta doses at the surface of the equipment should be the sum of the airborne and plate-out sources. The airborne beta dose should be taken as the beta dose calculated for a point at the containment center.	(7) For unshielded equipment, the beta dose is calculated at the most conservative location for all appropriate contributors of beta doses including airborne, and suspended sources.

<u>CATEGORY II</u> Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(8) Shielded components need be qualified only to the gamma radiation levels required, provided an analysis or test shows that the sensitive portions of the component or equipment are not exposed to beta radiation or that the effects of beta radiation heating and ionization have no deleterious effects on component performance.	(8) Components are qualified, by exposure to gamma radiation only, to the total (numerical) integrated dose required. The total dose includes gamma and beta radiation and appropriate shielding credits with adequate justification.
(9) Cables arranged in cable trays in the containment should be assumed to be exposed to half the beta radiation dose calculated for a point at the center of the containment plus the gamma ray dose calculated in accordance with Section 1.4(6). This reduction in beta dose is allowed because of the localized shielding by other cables plus the cable tray itself.	(9) See 1.4 (8) above. In addition, the beta dose at the equipment may be reduced by equipment covering material (i.e., cable jackets, boxes, etc.) and thickness as permitted by Section 4.1.2 of I&E Bulletin 79-01B. In these cases, justification is provided.
(10) Paints and coatings should be assumed to be exposed to both beta and gamma rays in assessing their resistance to radiation. Plate-out activity should be assumed to remain on the equipment surface unless the effects of the removal mechanisms, such as spray wash-off or steam condensate flow, can be justified and quantified by analysis or experiment.	(10) Paints and coatings are assumed to be exposed to both beta and gamma rays in assessing their resistance to radiation. Plate-out activity is assumed to remain on the equipment.
(11) Components of the emergency core cooling system (ECCS) located outside containment (e.g., pumps, valves, seals and electrical equipment) should be qualified to withstand the radiation equivalent to that penetrating the containment, plus the exposure from the sump fluid using assumptions consistent with the requirements stated in Appendix K to 10 CFR Part 50.	(11) Components of the Residual Heat Removal System and the Containment Spray System located outside containment are qualified to withstand the radiation equivalent to that penetrating the containment plus the exposure from the sump fluid. See 1.4 (5) above.
(12) Equipment that may be exposed to radiation doses below 104 rads should not be considered to be exempt from radiation qualification, unless analysis supported by test data is provided to verify that these levels will not degrade the operability of the equipment below acceptable values.	(12) Equipment exposed to radiation doses at any level are not considered to be exempt from radiation qualification, unless analysis supported by test data and/or operating experience is provided to verify that these levels will not degrade the operability of the equipment below acceptable values. Otherwise equipment is qualified to their required doses. See 1.4 (8) and (9) above.
(13) The staff will accept a given component to be qualified provided it can be shown that the component has been qualified to integrated beta and gamma doses which are equal to or higher than those levels resulting from an analysis similar in nature and scope to that included in Appendix D (which uses the source term given in Item (1) above), and that the component incorporates appropriate factors pertinent to the plant design and operating characteristics, as given in these general guidelines.	(13) The applicants' environmental qualification program complies with the guidelines previously described in Item 1.1 (1) through (12).

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(14) When a conservative analysis has not been provided by the applicant for staff review, the staff will use the radiation environment guidelines contained in Appendix D, suitably corrected for the differences in reactor power level, type, containment size, and other appropriate factors.	(14) A conservative analysis has been provided by the applicant in the FSAR sections referenced above.
1.5 Environmental Conditions for Outside Containment	
(1) Equipment located outside containment that could be subjected to high-energy pipe breaks should be qualified to the conditions resulting from the accident for the duration required. The techniques to calculate the environmental parameters described in Sections 1.1 through 1.4 (Category II) above should be applied.	1.5 (1) Equipment located outside containment is qualified to operate following a high-energy pipe break as described in FSAR Section 3.6 and Appendix 3.6A. In some cases, additional justification with component thermal lag analysis (using COMPARE Mod.1A, COMPRESS Mod.1, GOTHIC Version 3.4d or GOTHIC Version 6.1b (Refer to FSAR Appendix 3.11E)) is provided to demonstrate that equipment can maintain its functional operability. Only that equipment necessary to mitigate or monitor the consequences of the postulated HELB accident is qualified to the respective HELB conditions.
(2) Equipment located in general plant areas outside containment where equipment is not subjected to a design basis accident environment should be qualified to the normal and abnormal range of environmental conditions postulated to occur at the equipment location.	(2) Equipment located in general plant areas outside Containment are qualified for the maximum normal and abnormal range of environmental conditions postulated in the equipment area. Refer to the figures in FSAR Appendix 3.11B for applicable environmental parameters in these general plant areas.
(3) Equipment not served by Class 1E environmental support systems, or served by Class 1E support systems that may be secured during plant operation or shutdown, should be qualified to the limiting environmental conditions that are postulated for the location, assuming a loss of the environmental support system; or, there may be designs where a loss of the environmental support system may expose some equipment to environments that exceed the qualified limits. For these designs, appropriate monitoring devices should be provided to alert the operator that abnormal conditions exist and to permit an assessment of the conditions that occurred in order to determine if corrective action, such as replacing any affected equipment, is warranted.	(3) Equipment served by Class 1E environmental support systems that may be secured during plant operation or shutdown will be qualified for the limiting Anticipated Operation Occurrence (AOO) environmental conditions assuming loss of the environmental support system, but such conditions are considered to be within the AOO temperature envelope of mild environment.
2. QUALIFICATION METHODS	
2.1 Selection of Methods	

<u>CATEGORY II</u> Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(1) Qualification methods should conform to the requirements defined in IEEE Std. 323 1971.	2.1 (1) Qualification methods conform to the guidelines of IEEE Std. 323-1971; however, much of the equipment has been upgraded to meet NRC Regulatory Guide 1.89 Revision 0 and its adopted standard IEEE Std. 323-1974 as described in FSAR Section 1.8. Refer to FSAR Tables 3.11.0-1 and 3.11.0-2 for qualified equipment which has been upgraded.
(2) The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached.	(2) In general equipment located in a harsh environment is qualified for the time required by type test in an accident test environment. Supplementary review and analysis is necessary to demonstrate that the test environmental conditions exceed or are equivalent to the applicable Shearon Harris conditions. Functional operability is required during qualification testing.
(3) The environmental qualification of equipment exposed to DBA environments should conform to the following positions. The bases should be provided for the time interval required for operability of this equipment. The operability and failure criteria should be specified and the safety margins defined.	(3) The environment qualification of equipment located in a harsh environment conforms to the following:
(a) Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.	(a) Equipment that must function in order to mitigate or monitor any accident is qualified as stated in 2.1 (2) above, to demonstrate operability for the time required.
(b) Any equipment (safety-related or non-safety-related) that need not function in order to mitigate any accident, but that must not fail in a manner detrimental to plant safety should be qualified by test to demonstrate its capability to withstand any accident environment for the time during which it must not fail.	(b) Non-safety related equipment in this category has been upgraded to Class 1E status. Safety-related equipment is qualified as described in 2.1 (2) above.
(c) Equipment that need not function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety need only be qualified for its non-accident service environment. Although actual type testing is preferred, other methods when justified may be found acceptable. The bases should be provided for concluding that such equipment is not required to function in order to mitigate any accident, and that its failure in any mode in any accident environment is not detrimental to plant safety.	(c) This equipment is qualified for a mild environment as described in 10 CFR 50.49. The applicant complies with this requirement with respect to safety-related equipment. (NUREG-0588 is only applicable to safety-related equipment.)

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(4) For environmental qualification of equipment subject to events other than a DBA, which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history, or any applicable combination thereof, coupled with partial type test data may be found acceptable, subject to the applicability and detail of information provided.	(4) When the environment from such an event (e.g., loss of offsite power) is enveloped by the environment from anticipated operational occurrences rather than significant design basis event changes, the area is defined as a mild environment area; therefore, the equipment is qualified under mild environmental conditions.
2.2 Qualification by Test	
(1) The failure criteria should be established prior to testing.	2.2 (1) In lieu of failure criteria, the Applicant has insured that the qualifications by test include an acceptance criteria. Completed testing which did not include a specific acceptance criteria are analyzed or verified acceptable for their application.
(2) Test results should demonstrate that the equipment can perform its required function for all service conditions postulated (with margin) during its installed life.	(2) Refer to Section 3 for details on margin.
(3) The items described in Section 5.2 of IEEE Std. 323-1971 supplemented by items (4) through (12) below constitute acceptable guidelines for establishing test procedures.	(3) SHNPP utilizes these guidelines for establishing test procedures. In addition, equipment upgraded to the 1974 standard utilizes the guidelines of Section 6.3 of IEEE Std. 323-1974 as applicable supplemented by items (4) through (12) below.
(4) When establishing the simulated environmental profile for qualifying equipment located inside containment, it is preferred that a single profile be used that envelops the environmental conditions resulting from any design basis event during any mode of plant operation (e.g., a profile that envelops the conditions produced by the main steamline break and loss-of-coolant accidents).	(4) SHNPP utilizes a simulated combined MSLB/LOCA environmental profile for equipment inside containment as shown on FSAR Figures 3.11.4-1 and 3.11.6-1. The preferred method of qualification is to assure that this profile is enveloped by the environmental test profile to which the equipment is qualified.
(5) Equipment should be located above flood level or protected against submergence by locating the equipment in qualified watertight enclosures. Where equipment is located in watertight enclosures, qualification by test or analysis should be used to demonstrate the adequacy of such protection. Where equipment could be submerged, it should be identified and demonstrated to be qualified by test for the duration required.	(5) In general, equipment is located above the maximum flood level. Equipment required to be located below the maximum flood level is qualified to operate in a submerged condition or justification is provided to demonstrate that the equipment can perform its safety function for the duration required before being submerged and subsequent failure will not affect the accomplishment of safety function.

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(6) The temperature to which equipment is qualified, when exposed to the simulated accident environment, should be defined by thermocouple reading on or as close as practical to the surface of the component being qualified. If there were no thermocouples located near the equipment during the tests, heat transfer analysis should be used to determine the temperature at the component. (Acceptable heat transfer analysis methods are provided in Appendix B.)	(6) The temperature to which equipment is qualified is monitored throughout the test to assure that it was exposed to the bulk temperature equivalent to or more severe than that temperature assumed in the bounding envelope derived from the accident analysis. In some cases, this monitoring is based on using the steam tables and the measured steam pressure to obtain the saturated steam temperature.
(7) Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability.	(7) Equipment performance characteristics are monitored before, during, and after testing. The degree of equipment monitoring (i.e., periodic or continuous) is based on equipment function, failure modes, and practicality of testing.
(8) Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the onsite spray systems actuate.	(8) During simulated event testing, a caustic spray is used. Spray system actuation is delayed so as to simulate the required conditions as closely as possible.
(9) The operability status of equipment should be monitored continuously during testing. For long-term testing, however, monitoring at discrete intervals should be justified if used.	(9) See 2.2 (7) above.
(10) Expected extremes in power supply voltage range and frequency should be applied during simulated event environmental testing.	(10) During simulated event environmental test application of voltage/frequency extremes may not be feasible. Post test is the point at which extremes of voltage/frequency are considered. Voltage/frequency tolerance is typically enveloped by industry standards which is the design constraint for the design of the power distribution system as described in FSAR Section 8. Design optimization is verified for voltage, frequency, etc. This ensures the adequacy of equipment and distribution system.
(11) Dust environments should be addressed when establishing qualification service conditions.	(11) Equipment susceptibility to dust is considered in the plant maintenance procedures or by the use of protective covers.
(12) Cobalt-60 is an acceptable gamma radiation source for environmental qualification.	(12) Cobalt-60 of an equivalent source is used.
<u>2.3 Test Sequence</u>	
(1) Justification of the adequacy of the test sequence selected should be provided.	2.3 (1) Justification for the test sequence is provided. In addition, the test environmental conditions are reviewed to assure that they simulate as close as practicable the postulated environment.

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(2) The test should simulate as closely as practicable the postulated environment.	(2) Environmental service conditions expected to occur are enveloped by the test simulation environment and/or by supplementing analysis and review.
(3) The test procedures should conform to the guidelines described in Section 5 of IEEE Std. 323-1971.	(3) See 2.2 (3) above.
(4) The staff considers that, for vital electrical equipment such as penetrations, connectors, cables, valves and motors, and transmitters located inside containment or exposed to hostile steam environments outside containment, separate effects testing for the most part is not an acceptable qualification method. The testing of such equipment should be conducted in a manner that subjects the same piece of equipment to radiation and the hostile steam environment sequentially.	(4) In general, equipment which must perform a safety function in a harsh environment is qualified by subjecting "sample" equipment to the test conditions. Where this is impractical (e.g., due to size limitations) justification is provided for separate effects testing. Sequential testing is the standard method of test with exceptions documented and justified.
2.4 Other Qualification Methods	
Qualification by analysis or operating experience implemented, as described in IEEE Std. 323-1971 and other ancillary standards, may be found acceptable. The adequacy of these methods will be evaluated on the basis of the quality and detail of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified. It is required that when these methods are employed some partial type tests on vital components of the equipment be provided in support of these methods.	2.4 In general, supplementary review and analysis is used to evaluate test data to demonstrate qualification. Testing is generally employed to qualify the equipment.
3. MARGINS	
(1) Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any margins (conservatism) applied during the derivation of the specified plant parameters.	3. (1) The applicant has utilized the NRC staff acceptable approach of demonstrating that the temperature, pressure, and radiation conditions are derived using the NUREG-0588 methodology which is sufficiently conservative such that margin need account only for inaccuracies in the test equipment. See Resolution of Comment 70 in NUREG-0588, Rev. 1
(2) The margins provided in the design will be evaluated on a case-by-case basis. Factors that should be considered in quantifying margins are (a) the environmental stress levels induced during testing, (b) the duration of the stress, (c) the number of items tested and the number of tests performed in the hostile environment, (d) the performance characteristics of the equipment while subjected to the environmental stresses, and (e) the specified function of the equipment.	(2) See 3 (1) above.

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
(3) When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for uncertainties such as production errors and errors associated with defining satisfactory performance (e.g., when only a small number of units are tested).	(3) Appendix C is appliable to BWR and ice condenser containments. SHNPP is a dry containment PWR; therefore, qualification to Appendix C is not applicable.
(4) Some equipment may be required by the design to only perform its safety function within a short time period into the event (i.e., within seconds or minutes), and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be required to perform a safety function but must not fail within a short time period into the event, and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories is required to remain functional in the accident environment for a period of at least one hour in excess of the time assumed in the accident analysis. For all other equipment (e.g., post-accident monitoring, recombiners, etc.), the 10 percent time margin identified in Section 6.3.1.5 of IEEE Std. 323-1974 may be used.	(4) Equipment procured for short-term operation has been reviewed to assure that it is qualified for the time required to operate with additional margin.
4. AGING	
(1) Qualification programs that are committed to conform to the requirements of IEEE Std. 382-1972 (for valve operators) and IEEE Std. 334-1971 (for motors) should consider the effects of aging. For this equipment, aging effects, regardless of its location in the plant, should be considered and included in the qualification program.	4. (1) The effects of aging are considered for the qualification programs that are committed to conform to the requirements of IEEE Std. 382-1972 (for valve operators) and IEEE Std. 334-1971 (for motors).
(2) For other equipment, the qualification programs should address aging only to the extent that equipment that is composed, in part, of materials susceptible to aging effects should be identified, and a schedule for periodically replacing the equipment and/or materials should be established. During individual case reviews, the staff will require that the effects of aging be accounted for on selected equipment if operating experience or testing indicates that the equipment may exhibit deleterious aging mechanisms.	(2) Aging effects on all Class 1E equipment located in a harsh environment are considered. Specific maintenance/surveillance requirements are referenced in the Equipment Qualification Documentation Package. Where it has been determined that the qualified life of equipment or subcomponents is less than required, an applicable replacement interval will be so noted. This does not preclude development of surveillance and maintenance activities to support a possible extension of qualified life. This information will be incorporated into the applicant's maintenance/surveillance program.

THE FOLLOWING CATEGORY I PORTIONS OF SECTION 4 ARE APPLICABLE FOR THE QUALIFICATION PROGRAMS THAT ARE COMMITTED TO CONFORM TO THE REQUIREMENTS OF IEEE STD. 382-1972 (FOR VALVE OPERATORS) AND IEEE STD. 334-1971 (FOR MOTORS).

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
4.1 Aging effects on all equipment, regardless of its location in the plant, should be considered and included in the qualification program.	4.1 The effects of aging are considered for all safety related equipment located in a harsh environment.
4.2 The degrading influence discussed in Sections 6.3.3, 6.3.4, and 6.3.5 of IEEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of equipment should be considered and included as part of the aging programs.	4.2 The degrading influences discussed in Sections 6.3.3, 6.3.4, and 6.3.5 of IEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of equipment are considered and included as part of the Equipment Qualification Program.
4.3 Synergistic effects should be considered in the accelerated aging programs. Investigation should be performed to assure that no known synergistic effects have been identified on materials that are included in the equipment being qualified. Where synergistic effects have been identified, they should be accounted for in the qualification programs. Refer to NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), "Qualification Testing Evaluation Quarterly Reports," for additional information.	4.3 Synergistic effects are considered and are a part of SHNPP's ongoing Environmental Qualification Program.
4.4 The Arrhenius methodology is considered an acceptable method of addressing accelerated aging. Other aging methods that can be supported by type tests will be evaluated on a case-by-case basis.	4.4 In general, Arrhenius methodology and other aging methods (when used) are supported by type tests and supplementary analysis.
4.5 Known material phase changes and reactions should be defined to insure that no known changes occur within the extrapolation limits.	4.5 Known material phase changes are evaluated if necessary, during qualification to insure that no known changes occur within the limits of qualification.
4.6 The aging acceleration rate used during qualification testing and the basis upon which the rate was established should be described and justified.	4.6 The aging acceleration rate used during qualification testing and the basis for the rate is described and identified in the Equipment Qualification Documentation Package.
4.7 Periodic surveillance testing under normal service conditions is not considered an acceptable method for ongoing qualification, unless the plant design includes provisions for subjecting the equipment to the limiting service environment conditions (specified in Section 3(7) of IEEE Std. 279-1971) during such testing.	4.7 In general, Class 1E equipment located in a harsh environment is qualified by testing. Periodic surveillance testing is not used as a method of qualification.
4.8 Effects of relative humidity need not be considered in the aging of electrical cable insulation.	4.8 SHNPP complies with this recommendation
4.9 The qualified life of the equipment (and/or component as applicable) and the basis for its selection should be defined.	4.9 The qualified life of the equipment and the basis for its selection is included in the specific Equipment Qualification Documentation Package.

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
4.10 Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and in other methods that may be reasonably assumed, coupled with good engineering judgment.	4.10 Qualified life is established as described.
END OF APPLICABLE CATEGORY I PORTIONS OF SECTION 4	
5. QUALIFICATION DOCUMENTATION	
(1) The staff endorses the requirements stated in IEEE Std. 323-1974 that, "The qualification documentation shall verify that each type of electrical equipment is qualified for its application and meets its specified performance requirements. The basis of qualification shall be explained to show the relationship of all facets of proof needed to support adequacy of the complete equipment. Data used to demonstrate the qualification of the equipment shall be pertinent to the application and organized in an auditable form."	5. (1) The main purpose of the qualification documentation is to provide auditable evidence that each type of equipment is qualified for its application and meets its specified performance requirements. Section 3.11 of the SHNPP FSAR provides information on the type of documentation generated as evidence of qualification.
(2) The guidelines for documentation in IEEE Std. 323-1971 when fully implemented are acceptable. The documentation should include sufficient information to address the required information identified in Appendix E. A certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program.	(2) Refer to Section 3.11 of the SHNPP FSAR for a description of the documentation generated to demonstrate qualification.

APPENDIX 3.11B ENVIRONMENTAL PARAMETER ZONE MAPS

There is no text to include in this appendix section. All figures have been removed.

APPENDIX 3.11C SUPPLEMENTAL ENVIRONMENTAL QUALIFICATION OF INSIDE CONTAINMENT SAFETY RELATED ELECTRICAL EQUIPMENT

The maximum predicted atmospheric temperature in containment exceeds the maximum equipment qualification temperature for some safety related equipment. In these cases, the equipment is further analyzed to demonstrate that the maximum equipment temperature lags the maximum atmospheric temperature and remains below the maximum equipment qualification temperature.

All analyses are performed in accordance with the requirements of NUREG-0588. Table 3.11C-1 summarizes the mass and energies, the computer code, and the type of equipment used in each analysis and the maximum analyzed and qualified temperature for each type of equipment analyzed. Analyses have been performed using both CONTEMPT-LT Mod 28 and GOTHIC Version 6.1b. The containment atmosphere pressure and temperature response predicted by GOTHIC was benchmarked against the containment response predicted by CONTEMPT for both the 102 Percent, 1.4 ft^2 Main Steam Line Break (MSLB) assuming failure of the Main Feed

Isolation Valve (MFIV) and the 102 Percent, Double Ended Hot Leg Break (DEHLB). The benchmarks for both cases indicate that the GOTHIC response agrees well with the CONTEMPT response. In both cases, the GOTHIC response is conservative with respect to the CONTEMPT response except during the super-heated release phase of the LOCA. The differences that occur during this period are negligible.

The analyses are performed using design inputs like those used to compute the containment environment temperature in FSAR section 6.2. All analyses were performed using mass and energies (M&Es) predicted for the plant configuration subsequent to Steam Generator Replacement and Power Uprate (SGR/PUR) except as outlined by note 2 on Table 3.11C-1. All CONTEMPT analyses were performed using M&Es from the 102 percent power, 1.4 ft2 MSLB (assuming single failure of the MFIV) case since this case produces the maximum containment atmosphere temperature. The GOTHIC analyses listed in Table 3.11C-1 were performed using the M&Es from the 102 percent power, 1.4 ft^2 MSLB (assuming single failure rate of the MFIV) case and the 102 percent power DEHLB case. The GOTHIC analyses indicate that the equipment temperature predicted for the 102 percent power MSLB case bounds the 102 percent DEHLB case.

The temperature responses for selected equipment are provided in Figures 3.11C-1 through 3.11C-3. Per Table 3.11C-1, the maximum analyzed equipment temperature remains less than the maximum qualified temperature for all equipment.

The Rosemount transmitters installed inside containment were subjected to greater than 350°F during qualification testing. For the MSLB conditions shown in FSAR Figure 3.11.4-3, the transmitter surface temperature will be lower than the resultant 344.6°F cable surface temperature, given the greater mass, heat capacity, etc. of the metal transmitter housing. Therefore, these postulated MSLB conditions are not considered to exceed the qualification time/temperature profile of these transmitters.

Other electrical safety-related equipment is qualified to inside containment DBA [MSLB and LOCA] requirements as documented by their applicable Equipment Qualification Documentation Package (as described within Section 3.11.3).

APPENDIX 3.11D APENDIX 3.1D WAS SUPERSEDED BY APPENDIX 3.11A. SHNPP POSITION ON ENVIRONMENTAL QUALIFICATIONS

DELETED BY AMENDMENT NO. 16

APPENDIX 3.11E SUPPLEMENTAL ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT INSIDE THE MAIN STEAM TUNNEL

The maximum predicted atmospheric temperature in the main steam tunnel exceeds the maximum equipment qualification temperature for some safety related equipment. In these cases, the equipment is further analyzed to demonstrate that the maximum equipment temperature lags the maximum atmospheric temperature and remains below the maximum equipment qualification temperature.

All analyses are performed in accordance with the requirements of NUREG-0588. Table 3.11-E-2 summarizes the mass and energies, the computer code, and the type of equipment used in each analysis and the maximum analyzed and qualified temperature for each type of equipment analyzed. A spectrum of analyses has been performed using the COMPARE Mod 1A, COMPRESS Mod 1, GOTHIC Version 3.4d, and GOTHIC Version 6.1b computer codes. In all cases, the main steam tunnel is modeled using multiple volumes. All equipment is conservatively located in the volume that produces the maximum atmospheric temperature. The main steam tunnel temperature response predicted by GOTHIC was benchmarked against the containment response predicted by COMPRESS for the 70 percent power, 0.5 ft2 main steam line break (MSLB). The GOTHIC response agrees well with and is conservative with respect to the COMPRESS response. (Refer to FSAR section 3.6A.3-2 for the steam tunnel temperature response using COMPRESS.)

COMPARE analyses [3.11E-3] were performed using mass and energies (M&Es) predicted prior to Steam Generator Replacement and Power Uprate (SGR/PUR). The NAMCO limit switch was evaluated using four different cases assuming an initial temperature of 116 °F and a reactor power of 102 percent with and without auxiliary feedwater (AFW) isolation. The four cases were- 1) 1.4 ft^2 break, 2) 0.86 ft^2 break, 3) 0.5 ft^2 break and 4) 0.1 ft^2 break. The analyses that credited AFW isolation produced acceptable results whereas those that did not credit auxiliary feedwater isolation produced unacceptable results. Crediting AFW isolation is consistent with the M&E analysis. The M&E analysis credits AFW isolation so as to maximize the superheat of the steam discharged from the break. The COMPARE analyses indicated that the 0.5 ft^2 break with AFW isolation was the most limiting. The 102 percent power and 0.5ft^2 break case was the run for the NAMCO limit switch and the ASCO solenoid valve. The maximum equipment temperatures predicted for these components were then used to envelope the maximum temperatures for other components that were not specifically modeled. The solenoid housing and the insulated valve body of the ASCO solenoid valves were modeled separately from the NAMCO limit switches to account for internal heat generated by the energized coil. These analyses indicated that the 0.5 ft^2 break with AFW isolation was the most limiting. The COMPARE analyses are hereafter referred to as the original design bases thermal lag analyses. All subsequent analyses evolved from and were effectively benchmarked against the original design bases thermal lag analyses.

The same cases analyzed using COMPARE were then repeated using GOTHIC Version 3.4d at an initial temperature of 125°F. These analyses credited AFW isolation. The GOTHIC Version 3.4d tunnel response was benchmarked against the COMPARE response. The response agreed favorably with the COMPARE response. Additionally, a sensitivity study was performed using GOTHIC Version 3.4d and initial temperatures of 116, 122 and 125 °F to determine the impact of increasing the initial temperature on the maximum equipment temperature. The results of the sensitivity study indicated that the increase in the maximum equipment temperature is typically less than the increase in the initial temperature.

The same cases were later analyzed with COMPRESS using SGR/PUR M&Es, an initial temperature of 122°F, and crediting AFW isolation. The maximum steam tunnel atmospheric temperature is associated with a 1.4 ft^2 MSLB at 102% power and is depicted in Figure 3.11.4-4. The design inputs used in COMPRESS thermal lag analysis are like those used to predict the maximum tunnel atmospheric temperature subsequent to SGR/PUR in FSAR section 3.6A.3.2 except that the limiting break size for component temperatures is 0.5 ft^2. In all cases, the tunnel atmospheric temperature response for the 0.5 ft^2 MSLB based on the GOTHIC Version 3.4d analyses performed using M&Es prior to SGR/PUR bound those produced by the COMPRESS analyses using M&Es subsequent to SGR/PUR. A comparison of the atmospheric temperature response for the 0.5 ft^2 MSLB break at different power levels before and after SGR/PUR using

COMPRESS and GOTHIC Version 3.4d respectively are given in Figure 3.11E-5. Consequently, all of the equipment temperatures reported in Table 3.11E-2 are based on the GOTHIC Version 3.4d analyses performed using the pre-SGR/PUR configuration except for the GOTHIC Version 6.1b analysis of the Okonite 90 mil single conductor inside conduit.

The GOTHIC Version 6.1b analysis (or the Okonite 90 Mil single conductor inside conduit) was performed using M&Es perdicted for SGR/PUR. Four different cases were analyzed. These cases were- 1) 102 percent power, 1.4 ft^2 break 2) 102 percent power, 0.5 ft^2 break, 3) 70 percent power, 1.4 ft^2 break, and 4) 70 percent power, 0.5 ft2 break. The results of the GOTHIC analysis indicate that the 70 percent power, 0.5 ft^2 break was most limiting. These results are consistent with the COMPARE results generated using pre SGR/PUR mass and energies.

In all of the cases outlined in Table 3.11E-2, the maximum analyzed equipment temperature remains less than the maximum qualified temperature.

The qualification of equipment in the tunnel involved adding heat sinks to the original design basis thermal lag analyses COMPARE deck to represent the safety related components. The heat sink surface temperatures were then compared to the manufacturer's equipment qualification temperatures to determine whether or not the equipment surface temperature falls within the qualification envelope.

The Mass and Energy release used for the original design basis thermal lag analysis is the generic MSLB blowdown data supplied by the Westinghouse Owners' Group (Reference 3.11E-1). These tables were generated using conservative assumptions which resulted in early tube bundle uncovery. For selected cases, this data was adjusted to take credit for the instrumentation at the Harris Plant which would isolate the auxiliary feedwater to the affected generator following main steam line isolation. Because of the termination of the auxiliary feedwater into the generator, tube bundle uncovery and end of blowdown will both occur earlier. In order to account for the more rapid increase in the fluid enthalpy expected as a result of the loss of the cooler auxiliary feedwater flow, the enthalpy of the blowdown was ramped to the maximum value of 1,290 Btu/lbm over 10 seconds. This sudden rise in enthalpy is included whenever both auxiliary feedwater isolation and tube bundle uncovery have occurred.

Immediately following the start of the blowdown, heat transfer to the equipment in the steam tunnel is dominated by steam condensing on the equipment surface. This heat transfer mechanism is very efficient, and the surface temperature of the component rapidly rises to the saturation temperature of the steam. When the surface temperature rises above the saturation temperature or the saturation temperature falls due to a change in room pressure, the heat transfer becomes characterized by a forced convection heat transfer mechanism. Due to the high room temperatures and low pressures in the steam tunnel following an MSLB, this transition occurs very early in the event. The forced convection heat transfer coefficient in a flowing fluid is based on a correlation having the following form:

$$h = \frac{k(Nu)}{D}$$

where:

$$Nu = C(Re)^n$$

Re = $\frac{VDp}{u}$

C,n = Emperical constants dependent on geometry and Reynolds number provided in Table 3.11E-1

h = heat transfer coefficient, Btu/hr-ft F

D = component outside diameter, ft.

k = thermal conductivity of fluid, Btu/hr-ft F

V = fluid velocity, ft/sec.

p = fluid density, lbm/ft^3

u = fluid viscosity, lbm/ft-sec.

Therefore, the heat transfer coefficient for a component in a fluid with a specific velocity and a given set of fluid properties is dependent upon the component size and geometry.

The actual components evaluated were the ASCO solenoid valve and the NAMCO limit switch. Since each of the safety related components in the steam tunnel can be modeled using a similar geometry, it was not necessary to include an additional heat sink for every piece of equipment. These particular components combine the smallest dimensions (maximizing the heat transfer coefficient) with the thinnest housing wall thicknesses to give the highest surface temperatures of any equipment in the tunnel. The ASCO solenoid valve has a housing thickness of 0.09 inches and is energized during normal operation. The body of the valve has a 0.125 inch thickness and the smallest overall dimensions. This component was qualified in an energized condition at an ambient temperature of 346°F. The NAMCO limit switch has a casing of 0.125 inches and a qualification temperature of 340°F. All of the remaining equipment is bounded by the NAMCO limit switch. Heat sinks added to model any other safety-related equipment would show lower surface temperatures.

Both heat sinks were modeled using a cylindrical geometry and given an initial surface temperature of 222°F. This initial temperature corresponds to the saturation temperature of pure steam at 18 psia, which conservatively represents the effects of condensing heat transfer with an infinite heat transfer coefficient. Following the pressure spike at 0.25 seconds, the saturation temperature falls and convection heat transfer begins. Heat transfer coefficients for each break size were calculated using the correlation given above. The velocity used in calculating the Reynolds number is based upon the blowdown rate, minimum fluid density, and the area of the first subcompartment junction in the COMPARE model. The rated flow of the fans in the steam tunnel is approximately 3% of the initial flow from the 1.4 ft.^{2} break, and is not expected to contribute significantly to the average velocity in the room.

The spectrum of break sizes provided in Reference 3.11E-1 was considered in the original design basis thermal lag analysis in addition to the adjustment for auxiliary feedwater isolation following main steam line isolation. Isolation of auxiliary feedwater to the affected steam generator has opposite effects, depending upon the size of the break. For the large breaks where steam line isolation occurs before tube bundle uncovery, isolation of auxiliary feedwater reduces the maximum component temperature by terminating the blowdown at an earlier time in the event. The 1.4 $ft.^2$ break was the only case in which auxiliary feedwater isolation was necessary to prevent the component surface temperatures from exceeding the EQ limit. For the smaller breaks where steam line isolation occurs after tube bundle uncovery, the termination of auxiliary feedwater results in increased component surface temperatures because of the more rapid rise in the enthalpy of the blowdown fluid. As with the larger breaks, the blowdown is terminated earlier; however, this effect is minimized with the smaller breaks because of the lower flow rate out of the break. The break size found to result in the highest component temperatures was the 0.5 $ft.^2$ break. Although larger breaks give higher atmosphere temperatures, higher component surface temperatures at the time of tube bundle uncovery, combined with the length of time that the atmosphere temperature remains above the EQ limit makes the 0.5 $ft.^2$ break the most limiting case.

The heat transfer coefficients used in the analysis of the NAMCO limit switch and ASCO solenoid valve are shown in Figures 3.11E-1 and E.11E-2, respectively. The surface temperatures of these components for the limiting 0.5 ft^2 break are shown in Figures 3.11E-3 and 3.11E-4. A maximum surface temperature of 315.7°F was calculated for the NAMCO limit switch. The ASCO solenoid valve required insulation to reduce its maximum surface temperature from 357.1°F to 247.7°F in order to remain below the qualification temperature.

The original design basis thermal lag analysis covered a spectrum of break sizes with and without steam line isolation and continuous auxiliary feedwater flow to the affected steam generator. The results of this analysis demonstrate that for a main steam line break in which the tube bundle is uncovered and superheated steam is released, the maximum surface temperature of the safety-related equipment in the SHNPP steam tunnel will not exceed the temperature reached during the manufacturers' equipment qualification program.

REFERENCES: APPENDIX 3.11E

3.11E-1 Westinghouse Owners' Group Letter WOG-84-235, dated September 11, 1984, HELB Superheated Mass/Energy Release Outside Containment, Guidelines for Evaluation

3.11E-2 Kreith, Frank, Principles of Heat Transfer, 3rd Edition, Harper & Row Publishers, 1973

3.11E-3 NLS-86-310, Letter from CP&L's S.R. Zimmerman to NRC's H.R. Denton, "High Energy Line Breaks Outside Containment" dated September 19, 1986

3.12 FUKUSHIMA RELATED REQUIRED ACTIONS

3.12.1 INTRODUCTION

On March 11, 2011, an earthquake-induced tsunami caused Beyond-Design Basis (BDB) flooding at the Fukushima Dai-ichi Nuclear Power Station in Japan. The flooding caused by the tsunami rendered the emergency power supplies and distribution systems inoperable resulting in an extended loss of alternating current (AC) power (ELAP) in five of the six units on the site. The ELAP led to the loss of core cooling as well as spent fuel pool cooling capabilities and a significant challenge to containment. All direct current (DC) power was lost early in the event on Units 1 & 2 and after some period of time at the other units. Units 1, 2, and 3 were affected to such an extent that core damage occurred and radioactive material was released to the surrounding environment.

The U.S. Nuclear Regulatory Commission (NRC) assembled a special task force, the Near-Term Task Force (NTTF) in order to advise the Commission on actions the U.S. Nuclear Industry should undertake in order to preclude a release of radioactive material in response to a natural disaster such as that seen at Fukushima Dai-ichi. NTTF members created NRC Report, “Recommendations of Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” referred to as the “90-day Report,” which contained a large number of recommendations for improving safety at U.S. nuclear power sites.

Subsequently, the NRC issued Order EA-12-049, “Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events” (Reference 3.12.1-1), Order EA-12-051, “Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation” (Reference 3.12.1-2) to implement strategies for Beyond-Design-Basis External Events (BDBEE), and reliable spent fuel pool instrumentation, respectively.

3.12.1.1 Order EA-12-049

NRC Order EA-12-049 was effective immediately and directed Shearon Harris Nuclear Power Plant to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling in the event of a beyond-design-basis external event.

The Nuclear Energy Institute (NEI), working with the nuclear industry, developed guidelines for nuclear stations to implement the strategies specified in NRC Order EA-12-049. These guidelines were published in the NEI 12-06 document entitled, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide” (Reference 3.12.1-3). This guideline was endorsed by the NRC in final interim staff guidance (ISG) document JLD-ISG-2012-01, Revision 0, “Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,” dated August 29, 2012 (Reference 3.12.1-4).

The NEI 12-06 FLEX implementation guide adopts a three-phase approach for coping with a BDB event.

Phase 1 – The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling capabilities.

Phase 2 – The transition phase requires providing sufficient portable onsite equipment to maintain or restore these functions until resources can be brought from off site.

Phase 3 – The final phase requires obtaining sufficient offsite resources to sustain these functions indefinitely.

This three-phase approach was utilized to develop the FLEX strategies for Shearon Harris Nuclear Power Plant. Strategies, equipment details, storage locations, periodic maintenance, and programmatic controls for mitigating beyond-design-basis external events are contained in an overall program document for flexible response to extended loss of all AC power (FLEX).

Program changes are controlled in accordance with NEI 12-06, Section 11.8, as endorsed by the NRC.

3.12.1.2 Order EA-12-051

NRC Order EA-12-051 was effective immediately and directed Shearon Harris Nuclear Power Plant to provide a reliable means of remotely monitoring wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. The technical scope of the NRC Order EA-12-051 specifies that Shearon Harris Nuclear Power Plant provide:

- Primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool
- Display in an area accessible following a severe event
- Independent electrical power to each instrument channel and provide an alternate remote power connection capability

The Nuclear Energy Institute (NEI), working with the nuclear industry, developed guidelines for nuclear stations to implement the instrumentation specified in NRC Order EA-12-051. These guidelines were published in the NEI 12-02, Revision 1 document entitled, “Industry Guidance for Compliance with NRC Order EA-12-051, ‘To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation’,” (Reference 3.12.1-5). This guideline was endorsed by the NRC in final interim staff guidance (ISG) document JLD-ISG-2012-03, Revision 0, “Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation,” dated August 29, 2012 (Reference 3.12.1-6).

The three critical levels to be monitored in the Shearon Harris Nuclear Power Plant spent fuel pools in which reliable indication of the water level capable of supporting identification of pool water level conditions by trained personnel are:

1. Level that is adequate to support operation of the normal fuel pool cooling system
2. Level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck
3. Level where fuel remains covered and actions to implement make-up water addition should no longer be deferred

These three critical levels are monitored at Shearon Harris Nuclear Power Plant using installed Wave-Guided Radar Wide Range Spent Fuel Pool Level Instrumentation. The FLEX program document, as mentioned above, describes Spent Fuel Pool Instrumentation program requirements including procedures, testing and calibration, and quality assurance.

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Auxiliary Dam and Spillway	NA	-	-	I	A	See Note (30)
Auxiliary Separating Dike	NA	-	-	I	A	See Note (30)
Main Dam and Spillway	NA	-	-	I	A	See Note (30)
Emergency Service Water System Structures & Channels	NA	-	-	I	A	See Note (30)
Auxiliary Reservoir Channel	NA	-	-	I	A	See Note (30)
Containment Building and Containment Liner	2	ASME III Division 2/ACI 359	-	I	A	See Note (30) and FSAR 3.8
Reactor Auxiliary Building	NA	-	-	I	A	See Note (30)
Fuel handling Building and Unloading Bay	NA	-	-	I	A	See Note (30)
Waste Processing Building	NA	-	-	I	A	See Note (30)
Tank Building	NA	-	-	I	A	See Note (30)
Diesel Generator Building	NA	-	-	I	A	See Note (30)
Turbine Building	NA	-	-	See Note (19)	A	See Note (30)
Containment Air Locks, Equipment Hatch and Valve Chamber	2	ASME III	MC	I	A	See Note (29) and FSAR 3.8
Diesel Fuel Oil Storage Tanks and Tank Building	NA	-	-	I	A	See Note (30)
Containment Internal Structures	NA	-	-	I	A	
Containment Crane Supports	NA	-	-	I	A	
Site Grade Elevations	NA	-	-	-	A	See Note (39)
Cooling Tower	NNS	-	-	-	E	
Electrical Manholes for Emergency Power and Control Cables	NA	-	-	I	A	See Note (30)
Old Reactor Vessel Head Storage Facility	NNS	-	-	-	E	
Old Steam Generator Storage Facility	NNS	-	-	-	E	
Systems and Components Reactor Coolant System						
Reactor Vessel	1	ASME III	1	I	A	
Steam Generator (Tube Side)	1	ASME III	1	I	A	See Note (4)
(Shell Side)	2	ASME III	1	I	A	
Pressurizer	1	ASME III	1	I	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Reactor Coolant Hot and Cold Leg Piping, Fittings, and Fabrication	1	ASME III	1	I	A	
Surge Pipe, Spray Pipe Fittings, and Fabrication	1	ASME III	1	I	A	See Note (5)
Crossover Leg Piping, Fittings, and Fabrication	1	ASME III	1	I	A	1
Pressurizer Safety Valves	1	ASME III	1	I	A	
Pressurizer Power Operated Relief Valves and Block Valves	1	ASME III	1	I	A	
Valves of Safety Class 1 to Safety Class 2 Interface	1	ASME III	1	I	A	
Pressurizer Relief Tank	NNS	ASME VIII	-	-	E	
Reactor Coolant Thermowell	1	ASME III	1	I	A	
Auxiliary Reactor Coolant Piping (Drains, etc.)	2	ASME III	2	I	A	
Pressurizer Relief Valve Discharge Lines ()Between Pressurizer Nozzle and Relief Valve Only)	1	ASME III	1	I	A	
Reactor Coolant Loop Drain Lines (RC Loop to Second Normally Closed Valve)	1	ASME III	1	I	A	
Reactor Coolant Pump						
a) RCP Casing	1	ASME III	1	I	A	
b) Main Flange	1	ASME III	1	I	A	
c) Thermal Barrier	1	ASME III	1	I	A	
d) #1 Seal Housing	1	ASME III	1	I	A	
e) #2 Seal Housing	2	ASME III	1	I	A	See Note (4)
f) Pressure-Retaining Bolting	1	ASME III	1	I	A	
RCP Motor		NEMA MG1	-	I	A	
a) Motor Rotor	NNS	-	-	I	A	
b) Motor Shaft	NNS	-	-	I	A	
c) Shaft Coupling	NNS	-	-	I	A	
d) Flywheel	NNS	-	-	I	A	
e) Bearing (Motor Upper Thrust)	NNS	-	-	I	A	See Note (6)
f) Motor Bolting	NNS	-	-	I	A	
g) Motor Stand	NNS	-	-	I	A	
h) Motor Frame	NNS	-	-	I	A	
i) Upper Oil Reservoir (UOR)	NNS	-	-	I	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
j) UOR Oil Cooler	3	ASME III	3	I	A	
k) Lower Oil Reservoir (LOR)	NNS	-	-	I	A	
l) LOR Cooling Coil	3	ASME III	3	I	A	
m) Lube Oil Piping	NNS	-	-	I	A	
Instrumentation	IE	-	-	I	A	See Note (15)
Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Reactor Vessel or Core Related						
Reactor Vessel Shoes and Shims	1	ASME III	1	I	A	See Note (20)
Irradiation Sample Holder	2	-	-	I	A	
Irradiation Samples	NNS	-	-	-	E	
Control Rod Drive Mechanism (CRDM) Dummy Can Assemblies	NNS	-	-	-	E	
CRDM Assemblies (except housing)	NNS	-	-	-	E	
CRDM Housing	1	ASME III	1	I	A	
Control Rod Drive Controls (in part)	IE	-	-	I	A	
Reactor Vessel Internals	2	See Note 37	-	I	A	
Primary Source Rods	NNS	-	-	-	E	
Neutron Detector Positioning Device	2	-	-	I	A	
Reactor Vessel Insulation Shell	NNS	-	-	-	E	
Reactor Coolant Pipe Insulation	NNS	-	-	-	E	
Rod Cluster Controls Full Length Assembly	2	-	-	I	A	
Burnable Poison Rod Assemblies	NNS	-	-	-	E	
Prim. & Sec. Source Gen. Assy.	NNS	-	-	-	E	
Integrated Head (IH) Package						
IH Cable Assemblies	NNS	-	-	-	E	
Radial Arm Stud Tensioner Hoist Assembly	NNS	-	-	-	E	
IH Cooling Fans	NNS	-	-	-	E	
IH Cooling Shrouds	NNS	-	-	-	E	
IH Cable Tray	NNS	-	-	-	E	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
IH Shroud Cooling Fan Duct	NNS	-	-	-	E	
IH Lift Rig	NNS	-	-	-	E	
IH Lift Rods	1	-	-	I	A	
IH Missile Shield	1	-	-	I	A	
IH Reactor Vessel Stud Support Collars	NNS	-	-	-	E	
IH Lift Rig Oper. Support Stand	NNS	-	-	-	E	
Incore Instrumentation						
Seal Table Assembly	1	-	-	I	A	See Note (22)
In Core Monitoring System (in part)	IE	-	-	I	A	
Flux Thimble Tubing	2	ASME III	2	I	A	
Flux Thimble Fittings	2	ASME III	2	I	A	
Thimble Guide Tubing	1	ASME III	1	I	A	
Reactor Coolant System Equipment Supports						
Hydraulic Shock Suppressors	1	ASME III	1	I	A	See Note (9)
Crossover Leg Restraint	1	ASME III	1	I	A	See Note (9)
Steam Generator Manway Cover Support	NNS	-	-	-	E	
Reactor Coolant Pump Supports	1	ASME III	1	I	A	See Note (9)
Pressurizer Support Ring	1	ASME III	1	I	A	See Note (9)
Reactor Vessel Supports	1	ASME III	1	I	A	See Note (9)
Control Rod Drive Mechanism Seismic Support Tie Rods	1	ASME III	1	I	A	See Note (9)
Reactor Coolant Pump Forging Type A	1	ASME III	1	I	A	See Note (9)
Steam Generator Forging Type A	1	ASME III	1	I	A	See Note (9)
Chemical & Volume Control System						
Regenerative HX	2	ASME III	2	I	A	
Letdown HX (tube side)	2	ASME III	2	I	A	
(shell side)	3	ASME III	3	I	A	
Mixed Bed Demineralizer	3	ASME III	3	See Note (7)	A	
Cation Bed Demineralizer	3	ASME III	3	See Note (7)	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class[1]	Code	Code Class	Seismic Category[2]	Quality Class[23]	Remarks
Reactor Coolant Filter	2	ASME III	2	I	A	
Volume Control Tank	2	ASME III	2	I	A	
Charging (High Head Safety Injection) Pumps	2	ASME III	2	I	A	
Charging Pump Motors	IE	-	-	I	A	
Seal Water Injection	2	ASME III	2	I	A	
Seal Water Return Filter	2	ASME III	2	I	A	
Boric Acid Blender	3	ASME III	3	I	A	
Letdown Orifices	2	ASME III	2	I	A	
Excess Letdown HX (tube side)	2	ASME III	2	I	A	
(shell side)	2	ASME III	2	I	A	
Seal Water HX (tube side)	2	ASME III	2	I	A	
(shell side)	3	ASME III	2	I	A	
Chemical Mixing Tank	NNS	ASME VIII	-	-	E	
Chemical Mixing Tank Orifice	NNS	-	-	-	E	
Boric Acid Tanks	3	ASME III	3	I	A	
Boric Acid Filter	3	ASME III	3	I	A	
Boric Acid Transfer Pump	3	ASME III	3	I	A	
Boric Acid Transfer Pump Motors	IE	-	-	I	A	
Boric Acid Batching Tank	NNS	ASME VIII	-	-	E	
Reactor Coolant Pump (RCP) Standpipe	NNS	ASME VIII	-	-	E	
RCP Standpipe Orifice	NNS	-	-	-	E	
RCP Seal Bypass Orifice	1	ASME III	1	I	A	
System Piping Valves						
a) Part of RCPB	1	ASME III	1	I	A	
b) Required for reactor coolant letdown and makeup	2	ASME III	2	I	A	
c) Required for providing boric acid for the letdown and makeup loop	3	ASME III	3	I	A	
d) Normally or automatically Isolated from parts of system covered by a, b, or c	NNS	ANSI B31.1	-	-	E	
Instrumentation	IE	-	-	I	A	See Note (15)

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Boron Thermal Regeneration Subsystem						
Moderating HX (tube side)	3	ASME III	3	See Note (7)	A	
(shell side)	3	ASME III	3	See Note (7)	A	
Letdown Chiller HX (tube side)	3	ASME III	3	See Note (7)	A	
(shell side)	NNS	ASME VIII	-	-	E	
Letdown Reheat HX (tube side)	2	ASME III	2	I	A	
(shell side)	3	ASME III	3	See Note (7)	A	
Thermal Regeneration Demineralizer	3	ASME III	3	See Note (7)	A	
Chiller Pump	NNS	-	-	-	E	
Chiller Surge Tank	NNS	ASME VIII	-	-	E	
Chiller Unit	NNS	-	-	-	E	
a) Evaporator	NNS	ASME VIII	-	-	E	
b) Condenser	NNS	ASME VIII	-	-	E	
c) Compressor	NNS	-	-	-	E	
System Piping and Valves						
a) Note normally or automatically isolated from safety class components except sample lines	3	ASME III	3	I	A	
b) Other	NNS	ANSI B31.1	-	-	E	
Boron Recycle System						
Recycle Holdup Tank	3	ASME III	3	I	A	
Recycle Monitor Tank	NNS	AWWA D-100	-	-	E	
Recycle Monitor Tank Pump Casing NNS	NNS	ANSI B73.1	-	-	E	
Recycle Evap. Feed Demineralizer	NNS	ASME III	See Note (50)	See Note (18)	C	
Recycle Evap. Feed Filter	NNS	ASME III	See Note (50)	See Note (18)	C	
Recycle Evap. Condensate Demineralizer	NNS	ASME VIII	-	-	E	
Recycle Evap. Reagent Tank	NNS	ASME VIII	-	-	E	
Recycle Holdup Tank Vent Ejector	NNS	ASME III	See Note (50)	See Note (18)	C	
Recycle Evap. Condensate Filter	NNS	ASME VIII	-	-	E	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Recycle Evap. Concentrate Filter	NNS	ASME VIII	-	-	E	
Recycle Evaporator Package						
a) Feed Preheater						
1) Feed Side	NNS	ASME III	See Note (50)	See Note (18)	C	
2) Steam Side	NNS	ASME VIII	-	-	E	
b) Gas Stripper	NNS	ASME III	See Note (50)	See Note (18)	C	
c) Submerged Tube Evap.						
1) Feed Side	NNS	ASME III	See Note (50)	See Note (18)	C	
2) Steam Side	NNS	ASME VIII	-	-	E	
d) Evaporator Condenser						
1) Distillate Side	NNS	ASME III	See Note (50)	See Note (18)	C	
2) Cooling Water Side	3	ASME III	3	I	A	
e) Distillate Cooler						
1) Distillate Water Side	NNS	ASME III	See Note (50)	See Note (18)	C	
2) Cooling Water Side	3	ASME III	3	I	A	
f) Absorption Tower	NNS	ASME III	See Note (50)	See Note (18)	C	
g) Vent Condenser						
1) Gas Side	NNS	ASME III	See Note (50)	See Note (18)	C	
2) Cooling Water Side	3	ASME III	3	I	A	
h) Distillate Pump	NNS	ASME III	See Note (50)	See Note (18)	C	
i) Concentrate Pump	NNS	ASME III	See Note (50)	See Note (18)	C	
j) Piping and Valves						
1) Feed	NNS	ASME III	See Note (50)	See Note (18)	C	
2) Distillate	NNS	ASME III	See Note (50)	See Note (18)	C	
3) Concentrate	NNS	ASME III	See Note (50)	See Note (18)	C	
4) Cooling	3	ASME III	3	I	A	
5) Steam	NNS	ANSI B31.1	-	-	E	
Steam Piping and Valves						
a) Not normally or automatically isolated from safety class components	3	ASME III	3	See Note (7)	A	
b) Other	NNS	ANSI B31.1	-	-	E	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Safety Injection System						
Accumulators	2	ASME III	2	I	A	
Boron Injection Tank (BIT)	2	ASME III	2	I	A	
Hydro Test Pump	NNS	-	-	-	E	
System Piping and Valves						
a) Part of RCPB	1	ASME III	1	I	A	
b) Required for initial injection or long-term recirculation of sump water for emergency core cooling	2	ASME III	2	I	A	
c) Normally or automatically isolated from parts of system by a & b	NNS	ANSI B31.1	-	-	E	
Instrumentation (in part)	IE	-	-	I	A	See Note (15)
Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Residual Heat Removal System						
Residual Heat Removal (Low Head Safety Injection) Pumps	2	ASME III	2	I	A	
RHR Pump Motors	IE	-	-	I	A	
Residual Heat Exchanger(tube side)	2	ASME III	2	I	A	
(shell side)	3	ASME III	3	I	A	
System Piping and Valves						
a)Part of RCPB	1	ASME III	1	I	A	
b)Required for residual heat removal	2	ASME III	2	I	A	
c)Normally or automatically isolated from parts of system covered by a or b	NNS	ANSI B31.1	-	-	E	
d)Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Instrumentation (in part)	IE	-	-	I	A	See Note (15)
Containment Spray System						
Refueling Water Storage Tank	2	ASME III	2	I	A	
Spray Additive Tank	3	ASME III	3	I	A	
Containment Spray Pumps	2	ASME III	2	I	A	
Containment Spray Pump Motors	IE	-	-	I	A	
Eductors	2	ASME III	2	I	A	
Containment Spray Nozzles	2	ASME III	2	I	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class[(1)]	Code	Code Class	Seismic Category[(2)]	Quality Class[(23)]	Remarks
System Piping and Valves						
a) Required for initial injection or long-term recirculation of sump water	2	ASME III	2	I	A	
b) Required for spray additive	3	ASME III	3	I	A	
c) Normally or automatically isolated from parts of sys. covered by a or b	NNS	ANSI B31.1	-	-	E	
d) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Instrumentation	IE	-	-	I	A	See Note (15)
Component Cooling System						
Component Cooling Pumps	3	ASME III	3	I	A	
Component Cooling Pump Motors	IE	-	-	I	A	
Heat Exchangers (tube side)	3	ASME III	3	I	A	
(shell side)	3	ASME III	3	I	A	
Surge Tank	3	ASME III	3	I	A	
Chemical Addition Tank	NNS	-	-	-	E	
System Piping and Valves						
a) Required for performance of Safety Class 2 or 3 components and which are in service during some normal mode of plant operation or are testable	2 or 3	ASME III	2 or 3	I	A	
b) Normally or automatically isolated from parts of system covered by a) above	NNS	ANSI B31.1	-	-	E	
c) Piping and valves required for performance of safety functions of SC2 components and which are not in service during any normal mode of plant operation and are not testable	2	ASME III	2	I	A	
d) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Reactor Coolant Drain Tank Heat Exchanger (shell side)	2	ASME III	2	I	A	
Instrumentation	IE	-	-	I	A	See Note (15)
Containment Penetration Pressurization System						
System Piping and Valves Connected to Penetrations	2	ASME III	2	I	A	
Instrumentation	NNS	-	-	-	E	
Waste Processing Building (WPB) Cooling System						
WPB Cooling Pumps	NNS	-	-	-	E	
Heat Exchanger (tube and shell side)	NNS	ASME VIII	-	-	E	
Piping and Valves	NNS	ANSI B31.1	-	-	E	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Screen Wash System						
Emergency Service Water Screen Wash Pumps	3	ASME III	3	I	A	
Emergency Service Water Screen Wash Pump Motors	IE	-	-	I	A	
System Piping and Valves from Emergency Service Water Pumps to Traveling Screens	3	ASME III	3	I	A	
Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Service Water System						
Normal Service Water Pumps	NNS	-	-	-	E	
Emergency Service Water Self-Cleaning Strainers	3	ASME III	3	I	A	
Emergency Service Water Pumps	3	ASME III	3	I	A	
Emergency Service Water Pump Motors	IE	-	-	I	A	
Service Water Booster Pumps	3	ASME III	3	I	A	
Normal Service Water Self-Cleaning Strainer	NNS	-	-	-	E	
Emergency Service Water Booster Pump Motors	IE	-	-	I	A	
Traveling Water Screens	NA	-	-	I	A	
System Piping and Valves						
a) Required for operation of Containment Fans Coolers (inside containment)	2	ASME III	2	I	A	
b) Required for performance of other safety functions	3	ASME III	3	I	A	
c) Normally or automatically Isolated from parts of system covered by a) and b) above	NNS	ANSI B31.1	-	-	E	
d) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Instrumentation	IE	-	-	I	A	See Note (15)
Sampling System						
Sample Heat Exchanger (tube side)	NNS	-	-	-	E	
(shell side)	NNS	-	-	-	E	
Sample Vessel	NNS	-	-	-	E	
Gross Failed Fuel Detector (GFFD)	NNS	-	-	-	E	
GFFD Sampler Cooler	NNS	-	-	-	E	
System Piping and Valves						
a) Part of RCPB	2	ASME III	2	I	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class[1]	Code	Code Class	Seismic Category[2]	Quality Class[23]	Remarks
b) Normally or automatically isolated from a) above	NNS	ANSI B31.1	-	-	E	
c) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Compressed Air System						
Compressor, Aftercoolers, Receivers, Air Dryers	NNS	-	-	-	E	
Piping and Valves for Accumulator 1A-SA and 1C-SB	3	ASME III	3	I	A	See Note (42)
Accumulators						
a) Required for performance of function of Safety Class 1, 2, or 3 valves	3	ASME III	3	I	A	
b) Other	NNS	-	-	-	E	
c) Required for performance of functions of Safety Class I valves	3	ASME VIII	-	I	A	See Note (41)
Containment Isolation Systems						
a) Part of RCPB	1	ASME III	1	I	A	
b) From first isolation valve inside containment penetration weld) to outermost isolation valve	2	ASME III	2	I	A	
c) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Instrumentation	IE	-	-	I	A	See Note (15)
Fuel Pool Cooling and Cleanup System						
Fuel Pool Heat Exchanger (tube side)	3	ASME III	3	I	A	
(shell side)	3	ASME III	3	I	A	
Fuel Pool Cooling Pumps	3	ASME III	3	I	A	
Fuel Pool Cooling Pump Motors	IE	-	-	I	A	
Fuel Pool Demineralizer Filter	NNS	ASME VIII	-	-	E	
Fuel Pool Demineralizer	NNS	ASME VIII	-	-	E	
Fuel Pool Refueling Water Purification Filter	NNS	ASME VIII	-	-	E	
Fuel Pool Strainers	3	ASME III	3	I	A	
Fuel Pool Skimmer Filters	NNS	ASME VIII	-	-	E	
Fuel Pool Skimmer Pumps	NNS	-	-	-	E	
Fuel Pool and Refueling Water Purification Pump	NNS	-	-	-	E	
Fuel Pool Skimmers	NNS	-	-	-	E	
Fuel Pool Liner	NNS	-	-	I	A	See Note (21)

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Fuel Pool Nozzles	NNS	-	-	I	A	See Note (21) and (21A)
System Piping and Valves						
a) Required for cooling and makeup to the fuel pools	3	ASME III	3	I	A	
b) Makeup from RWST	NNS	ANSI B31.1	-	-	E	See Note (47)
c) Required for fuel pool cleanup and normally isolated from a)	NNS	ANSI B31.1	-	-	E	
Instrumentation	IE	-	-	I	A	
Fuel Handling System						
Manipulator Crane	NNS	-	-	-	B	
Reactor Vessel Internals Lifting Device	NNS	-	-	-	E	
Rod Cluster Control Changing Fixture	NNS	-	-	-	E	
Reactor Vessel Stud Tensioner	NNS	-	-	-	E	
Rack Insert Handling Tool	NNS	-	-	-	E	
Spent Fuel Handling Tool	3	-	-	I	A	See Note (10)
Fuel Transfer System						
a) Fuel Transfer Tube and Flange	2	ASME III	2	I	A	See Note (11)
b) Remainder of System	NNS	-	-	-	E	
New Fuel Elevator	NNS	-	-	-	E	
New Fuel Racks	3	-	-	I	A	
Portable Underwater Lights	NNS	-	-	-	E	
Load Cell	NNS	-	-	-	E	
Stud Hole Plug Handling Fixture	NNS	-	-	-	E	
Stud Hold Plugs	NNS	-	-	-	E	
Rod Cluster Control Thimble Plug Tool	NNS	-	-	-	E	
Spent Fuel Bridge Crane	NNS	-	-	I	A	(MECH. ONLY)
Source Installation Guide	NNS	-	-	-	E	
Crane Scales	NNS	-	-	-	E	
Irradiation Tube End Plug Seat Jack	NNS	-	-	-	E	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Burnable Poison Rod Handling Tool	NNS	-	-	-	E	
Irradiation Sample Handling Tool	NNS	-	-	-	E	
Burnable Poison Assembly Rack Inserts	NNS	-	-	-	E	
Neutron Absorbing Rack Inserts (Metamic)	3	-	-	I	A	
Trash Basket	NNS	-	-	-	E	
Trash Basket Lifting Tool	NNS	-	-	-	E	
Control Rod Drive Shaft Handling Fixture	NNS	-	-	-	E	
Control Rod Drive Shaft Unlatching Tool Full Length	NNS	-	-	-	E	
New Fuel Elevator Winch	NNS	-	-	-	E	
New Fuel Assembly Handling Fixture	NNS	-	-	-	E	
New Rod Cluster Control Handling Fixture	NNS	-	-	-	E	
Lower Internals Storage Stand	NNS	-	-	-	E	
Upper Internals Storage Stand	NNS	-	-	-	E	
Load Cell Linkage	NNS	-	-	-	E	
Spent Fuel Storage Racks	3	-	-	I	A	
Refueling Cavity Seal Ring	3	-	-	I	A	See Note (45)
Instrumentation	IE	-	-	I	A	See Note (15)
Liquid Waste Processing System	NNS	See Note (25)	See Note (25)	-	-	See Note (25)
Reactor Coolant Drain Tank	NNS	ASME VIII	-	-	E	
Reactor Coolant Drain Tank Pump	NNS	ANSI B73.1	-	-	E	
Reactor Coolant Drain Tank Heat Exchanger (shell side)	2	ASME III	2-	I-	A	
(tube side)	NNS	ASME VIII			E	
System Piping and Valves						
a) Not normally or automatically isolated from SC-3 components	3	ASME III	3	I	A	
b) Other	NNS	B31.1	-	-	C	
Gaseous Waste Processing System						
Gas Compressor	NNS	-	-	-	C	
Gas Decay Tank	3	ASME III	3	I	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Hydrogen Recombiner (Catalytic)	NNS	-	-	-	C	
System Piping and Valves						
a) Not normally or automatically isolated from SC-3 component	3	ASME III	3	I	A	
b) Other	NNS	B31.1	-	-	C	
Solid Waste Processing System	NNS	See Note (26)	See Note (26)	-	-	See Note (26) See Note (27)
Containment Cooling System						
Containment Fan Coolers						
a) Fans and Casings	2	-	-	I	A	
b) Supply Fan Motor	IE	-	-	I	A	
c) Cooling Coils	2	ASME III	2	I	A	
d) Ductwork and dampers up to concrete airshafts	2	-		I	A	
e) Ductwork and dampers downstream of concrete airshafts	NNS	-	-	-	B	See Note (18)
Containment Fan Coil Units	NNS	-	-	-	E	
Assoc. Ductwork	NNS	-	-	-	B	See Note (18)
Instrumentation	IE	-	-	I	A	See Note (15)
Containment Ventilation System						
Airborne Radioactivity Removal System	NNS	-	-	-	E	
CRDM Cooling Systems	NNS	-	-	-	E	
Containment Combustible Gas Control System						
Deleted row in Amendment 62						
Hydrogen Monitoring System (0-10% range capability)						
a) System Piping, Valves and Sample Lines						
1) Containment Isolation	2	ASME III	2	I	A	
2) Normally or automatically isolated from a) above	NNS	ASME III	See Note (51)	-	B	See Note (18)
3) Sample Lines inside Containment	NNS	ASME III	See Note (51)	-	B	See Note (18)
4) Sample Lines in RAB	NNS	ASME III	See Note (51)	-	B	See Note (53)
b) Hydrogen Analyzer Cabinet	NNS	-	-		B	See Note (18)

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
c) Remote Control Panel	NNS	-	-		B	See Note (18)
d) Remote Sample Dilution Panel	NNS	-	-	-	E	
Containment Vacuum Relief (except blind flanges and valves for leak testing)	2	ASME III	2	I	A	
Up to & Including Isolation Valves	2	ASME III	2	I	A	
Other Duct Inside Containment	NNS	-	-	-	B	See Note (18)
Instrumentation	IE	-	-	I	A	
Primary Shield Cooling System	3	-	-	I	A	
Instrumentation	NNS	-	-	-	E	
Reactor Supports Cooling System	3	-	-	I	A	
Instrumentation	NNS	-	-	-	E	
Reactor Auxiliary Building (RAB) Ventilation System						
RAB Normal Ventilation System						
a) Tornado and isolation dampers	3	-	-	I	A	
b) All other components	NNS	-	-	-	B	See Note (18)
RAB Steam Tunnel Ventilation	3	-	-	I	A	
RAB Emergency Exhaust System	3	-	-	I	A	
RAB ESF Equipment Cooling Systems	3	-	-	I	A	
RAB ESF Battery Rooms Exhaust Fans	3	-	-	I	A	
RAB Computer and Communications Room	NNS	-	-	-	E	
a) HVAC System Tornado Protection Dampers & Ductwork	3	-	-	I	A	
b) Other Ductwork	-	-	-	-	E	
RAB Switchgear Room Ventilation	3	-	-	I	A	
a) Smoke Purge Isolation Valves	3	-	-	I	A	
b) Smoke Purge Makeup Isolation Dampers	3	-	-	I	A	
RAB Electric Equipment Protection Rooms Ventilation System Including	3	-	-	I	A	
a) HV Equipment Room Exhaust	3	-	-	I	A	
b) Smoke Purge Isolation Valves and Dampers	3	-	-	I	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class[1]	Code	Code Class	Seismic Category[2]	Quality Class[23]	Remarks
Instrumentation	IE	-	-	I	A	See Note (15)
Waste Processing Building Ventilation Systems	NNS	-	-	-	E	
MCC and Instrument Rack Area Local Cooler	3	-	-	I	A	
Control Room HVAC Systems						
Normal Supply and Exhaust Subsystem						
a) Supply Fans & Casings	3	-	-	I	A	
b) Cooling Coils	3	ASME III	3	I	A	
c) Electric Heating Coils	IE	-	-	I	A	
d) Ducts and Dampers	3	-	-	I	A	
e) Valves for Outside Air Intakes & Exhausts	3	-	-	I	A	
f) Radiation Detectors	IE	-	-	I	A	
g) Smoke Detectors	NNS	-	-	-	D	
h) Offices and Kitchen Area Electric Heating Coils	NNS	-	-	-	B	See Note (18)
i) Exhaust Fans and Dampers	NNS	-	-	-	E	
Purge Makeup and Exhaust						
a)Boundary Isolation Valves	3	-	-	I	A	See Note (14)
b)Fans	NNS	-	-	-	E	See Note (18)
Ducts	3	-	-	I	A	
Control Room Emergency Filtration System	3	-	-	I	A	
Instrumentation	IE	-	-	I	A	See Note (15)
Fuel Handling Building HVAC Systems						
Air Conditioning System for the Operating Floor						
a) Air Handling Unit	NNS	-	-	-	E	
b) Exhaust Fans	NNS	-	-	-	E	
c) Ductwork and Dampers						
1) Isolation Dampers & Ductwork	3	-	-	I	A	
2) Duct in Unload Area	3	-	-	I	A	
3) Other	NNS	-	-	-	B	See Note (18)

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Emergency Exhaust System for the Operating Floor	3	-	-	I	A	
Air Cleaning Units	3	-	-	I	A	
Ductwork and Dampers						
1) Isolation Dampers	3	-	-	I	A	
Spent Fuel Pump Room Ventilation System	3	-	-	I	A	
Instrumentation	IE	-	-	I	A	See Note (15)
Fuel Oil Transfer Pump House Ventilation System	3	-	-	I	A	
Diesel Generator Building Ventilation System	3	-	-	I	A	
a) DGB - Electric Room Ventilation	3	-	-	I	A	
b) DGB - F.O. Day Tank and Silencer Room Ventilation	3	-	-	I	A	
c) DGB - Diesel Generator Room Ventilation	3	-	-	I	A	
Emergency Service Water Intake Structure Ventilation System	3	-	-	I	A	
a) Pump Room	3	-	-	I	A	
b) MCC Room	3	-	-	I	A	
Essential Services Chilled Water System						
Chilled Water Pumps	3	ASME III	3	I	A	
Water Chillers						See Note (43)
a) Refrigerant Compressors	3	ANSI B9.1	-	I	A	
b) Water Cooler (Tube Side)	3	ASME III	3	I	A	
(shell side)	3	ASME III	3	I	A	
Refrigerant Piping and Valves	3	ANSI B9.1	-	I	A	
Refrigerant Transfer System	NNS	ANSI B9.1	-	-	B	See Notes (18) and (43)
Lubricating System Oil Pump, Piping, & Valves	3	ANSI B9.1	-	I	A	
Condenser Water Recirculation Pumps	3	ASME III	3	I	A	
Expansion Tanks	3	ASME III	3	I	A	
Chilled Water Piping and Valves						
a) Required to provide chilled water to safety-related air handling units	3	ASME III	3	I	A	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
b) Required only for RAB NNS Ventilation Systems and automatically isolated from components included in (a) above	NNS	-	-	-	B	See Note (18)
c) Operators for Safety-Related Active Valves	IE	-	-	I	A	
Instrumentation	IE	-	-	I	A	See Note (15)
Nonessential Services Chilled Water System	NNS	-	-	-	E	
Cont. Atmos. Purge & Makeup System (Normal & Pre-Entry)						
Containment Isolation Valves and Piping	2	ASME III	2	I	A	
Ductwork Inside Containment						
a) Normal Purge from Isolation Valve to Floor EL 236.00’	3	-	-	I	A	
b) Normal Purge from Floor EL 236.00’ to S-61 & S-62 ARRS))	NNS	-	-	-	E	
c) Normal Makeup, Pre-entry Makeup, & Pre-entry Purge	NNS	-	-	I	B	See Note (18)
Ductwork Outside Containment						
a) Normal & Pre-entry Purge from Isolation Valves thru RAB Floor EL 261.00’	3	-	-	I	A	
b) Normal & Pre-entry Makeup from Isolation Valves thru RAB Floor EL 286.00’	3	-	-	I	A	
c) Other	NNS	-	-	I	B	See Note (18)
d) Other	NNS	-	-	-	E	
Instrumentation (isolation valves only)	IE	-	-	I	A	See Note (15)
Operators for Safety-Related Active Valves	IE	-	-	I	A	(See Note 31)
Containment Hydrogen Purge Exhaust and Makeup System						
Ductwork Inside Containment up to the Isolation Valve	NNS	-	-	-	B	See Note (18)
Containment Isolation Valves and Piping	2	ASME III	2	I	A	
From Isolation Valves Outside Containment to floor penetration at RAB Elevation 261 ft.	3	-	-	I	A	
Other	NNS	-	-	-	B	See Note (18)
Operators for Safety-Related Active Valves	IE	-	-	I	A	(See Note 31)
RAB Outside Air Intakes for Containment Vacuum Relief and Purge Systems	2	-	-	I	A	
Instrumentation (isolation valves only)	IE	-	-	I	A	See Note (15)
Turbine Building HVAC Systems	NNS	-	-	-	E	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Standby Diesel Generator System						
Diesel Generator	IE	IEEE 387	-	I	A	See Note (14)
Instrumentation (in part)	IE	-	-	I	A	See Note (15)
DG Fuel Oil Storage and Transfer System						
Fuel oil storage tank	-	-	-	I	A	See Note (34)
Fuel oil storage tank liner	-	ASME VIII	-	I	A	See Note (34)
Fuel oil transfer pumps and associated piping, valves and strainers	3	ASME III	3	I	A	
Fuel oil day tanks	3	ASME III	3	I	A	
Fuel oil unloading pumps and associated piping, valves, and strainers	NNS	-	-	-	E	
Instrumentation (in part)	IE	-	-	I	A	See Note (15)
Operators for Safety-Related Activated Valves	IE	-	-	I	A	See Note (31)
Diesel Generator Cooling Water System						
Jacket water heat exchanger, piping and valves	3	ASME III	3	I	A	See Note (35)
Diesel Generator Air Starting System						
Diesel generator air receivers and associated piping and valves essential for emergency operation	3	ASME III	3	I	A	
Air compressor and after cooler	NNS	-	-	-	B	
Piping and tubing associated with starting air system	NNS	-	-	-	B	
Dryer	NNS	-	-	-	B	
Diesel Generator Lubrication System (Excluding Engine Mounted Piping and Components)	3	ASME III	3	I	A	See Note (51)
Diesel Generator Air Intake and Exhaust System						
a) Intake Piping	3	ASME III	3	I	A	
b) Exhaust Piping	3	ANSI B31.1	-	I	A	
c) Exhaust Silencer	3	-	-	I	A	
d) Instrumentation	IE	-	-	I	A	See Note (15)
Alternate Seal Injection System (ASI)						
Dedicated Shutdown Diesel Generator System						

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Fire Protection System	NNS	See Note (17)	-	See Note (18)	D	
Nitrogen Supply System	NNS	-	-	-	E	
Hydrogen Supply System	NNS	-	-	-	E	
Fuel Transfer Canal Liner	NNS	-		I	A	See Note (21)
Shipping Cask Pool Liner	NNS	-	-	I	A	See Note (21)
Reactor Cavity Liner	NNS	-	-	I	A	
Reactor Auxiliary Building Decontamination Liner	NNS	-	-	-	E	
FHB Cask Washdown Area Liner	NA	-	-	I	A	
Main Steam System						
System Piping and Valves						
a) From the steam generator up to and including the MSIV; all branch connections from this section up to and including the first normally closed or automatic closure shutoff valve (this includes safety valves and MS PORVs)	2	ASME III	2	I	A	
b) From the MSIV up to and including the last seismic restraint in the Turbine Building	NNS	See Note (16)	See Note (16)	I	A	
c) Downstream of last seismic restraint in Turbine Building	NNS	ANSI B31.1	-	-	E	
d) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
e) Turbine Gland Sealing System	NNS	ANSI B31.1	-	-	E	
f) From the AFW Terry Turbine Steam Admission Valves to the atmospheric vents	3	ASME III	3	I	A	
Instrumentation	IE	-	-	I	A	See Note (15)
Steam Generator Blowdown System						
System Piping and Valves						
a) From steam generator to and including outboard containment isolation valves	2	ASME III	2	I	A	
b) From containment isolation valves to RAB wall	3	ASME III	3	I	A	
Other	NNS	ANSI B31.1	-	-	C/E	Ref. 2165-S 2120
Condensate and Feedwater System						
Condensate and Feedwater Pumps	NNS	-	-	-	E	
Condenser Evacuation System	NNS	ANSI B31.1	-	-	E	
System Piping Valves						

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class[1]	Code	Code Class	Seismic Category[2]	Quality Class[23]	Remarks
a) Feedwater piping from the steam generator back to and including the MFIV check valve; all branch connections from this section up to and including the first normally closed shutoff valve	2	ASME III	2	I	A	
b) MFW Control valves and bypass control valves	3	ASME III	3	I	A	See Note (4)
c) From the MFIV check valve back to the last seismic restraint in the Turbine Building	NNS	See Note (16)	See Note (16)	I	A	
d) Upstream of last seismic restraint in Turbine Building	NNS	ANSI B31.1	-	-	E	
e) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Instrumentation						
f) Nitrogen supply to the MFIV actuators from the remote control panels	2	ANSI B31.1	-	I	A	See Note (48)
g) Nitrogen supply to the MFIV actuators from the accumulator tanks to the remote control panels, including the remote control panels.	2	ANSI B31.1	-	I	A	See Note (48)
h) Nitrogen supply to the MFIV actuators from the accumulator check valves to the tanks	2	ANSI B31.1	-	I	A	See Note (48)
i) Nitrogen supply to the MFIV actuators from the plant nitrogen system (beginning at hanger FW H 00878) to accumulator tank check valves	NNS	ANSI B31.1	-	I	A	See Note (48)
j) Accumulator tanks	2	ASME III	2	I	A	
<u>Auxiliary Feedwater System</u>						
AFW Pumps (Motor & Turbine Driven)	3	ASME III	3	I	A	
AFW Pump Motors	IE	-	-	I	A	
Condensate Storage Tank	3	ASME III	3	I	A	
AFW Pump Turbine Driver	3	ASME III	-	I	A	See Note (28)
System Piping and Valves						
a) From steam generator up to and including the containment isolation valves	2	ASME III	2	I	A	
b) Other	3	ASME III	3	I	A	
c) Operators for Safety-Related Active Valves	IE	-	-	I	A	See Note (31)
Instrumentation	IE	-	-	I	A	See Note (15)
<u>Condenser Circulating Water Systems</u>	NNS	-	-	-	E	
<u>Demineralized Water Storage System</u>						
Demineralized Water Storage Tank	NNS	-	-	-	E	
Reactor Makeup Water Storage Tank	3	ASME III	3	I	A	
Instrumentation (in part)	IE	-	-	I	A	See Note (15)

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
Reactor Makeup Water Pump, Pipes/Valves	3	ASME III	3	I	A	
Reactor Makeup Water Pump Motors	NNS	-	-	-	E	
Radiation Monitoring System						
Safety Area Monitors	IE	-	-	I	A	See Note (15)
Safety Effluent Monitors	IE	-	-	I	A	See Note (15)
Safety Process Monitors	IE	-	-	I	A	See Note (15)
Safety Airborne Radiation Monitors	IE	-	-	I	A	See Note (15)
Accident Monitors (RG 1.97) (High Range Area	IE	-	-	I	A	See Note (15) and See Note (33)
Post-Accident Monitoring Instrumentation NUREG 0737						
Containment Pressure Indication	IE	-	-	I	A	See Note (15) and See Note (32)
Containment Water Level Indication (Wide range and narrow range)	IE	-		I	A	See Note (15) and See Note (32)
Plant Vent Stack Radiation Monitoring	IE	-	-	I	A	See Note (15) and See Note (33)
Waste Processing Building Exhaust Stack Radiation Monitoring	NNS	-	-	-	E	See Note (33)
Main Steam Line Radiation Monitoring	IE	-	-	I	A	See Note (15) and See Note (33)
Condenser Vacuum Pump Radiation Monitoring	NNS	-	-	-	E	See Note (33)
High Range Containment Radiation Monitoring	IE	-	-	I	A	See Note (15)
Electrical Systems and Components						
ESF 6.9 kV Bus	IE	-	-	I	A	
ESF 6.9 kV Switchgear	IE	-	-	I	A	
ESF 480V Switchgear and Transformers	IE	-	-	I	A	
ESF 480V Motor Control Centers, including 120V Transformers and Power Panels	IE	-	-	I	A	
480-208/120V Transformers (normal/emergency) (lighting, except Main Control Room)	NNS	-	-	-	E	

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks
480-208/120V Transformers (normal/emergency lighting, Main Control Room only)	NNS	-	-	-	B	See Note (44)
ESF 120V Uninterruptible AC System	IE	-	-	I	A	
ESF Station Batteries, Battery Racks, and Chargers	IE	-	-	I	A	
Safety-related Motors	IE	-	-	I	A	
Safety-related power, control and instrument cables and associated cable splices and connectors as applicable	IE	-	-	-	A	
Safety-related raceway	-	-	-	I	A	
Safety-related terminal blocks	IE	-	-	I	A	
Nonsafety-related raceways and supports containing Nonsafety cables whose failure may damage other safety-related items						
Cable Trays	NNS			I	A	See Note (40)
Conduits	NNS				E	See Note (40)
Supports	NNS			I	A	See Note (40)
Containment Electrical Penetrations 120 Volt D.C. System	2	-	-	I	A	
Primary and backup protective devices associated with class IE containment electrical penetrations and 6.9 kV nonclass IE system	IE	-	-	I	A	
See Calc 8S44-P-101 for a listing of Station Blackout Components					B	
Reactor Protection System (in part)	IE	-	-	I	A	
Out of Core Neutron Monitoring System	IE	-	-	I	A	See Note (49)
ESF Protection System (T, P, or S signals)	IE	-	-	I	A	
Reactor Coolant System Vents						
Piping from Reactor Vessel up to and including flow restrictor	1	ASME III	1	I	A	
Piping and valves up to and including second isolation valve	2	ASME III	2	I	A	
All other piping	NNS	ANSI B31.1	-	-	E	
Instrumentation	IE	-	-	I	A	See Note (15)
Inadequate Core Cooling System (in part)	IE	-	-	I	A	See Note (15)
Associated piping and valves	2	ASME III	2	I	A	

Notes to Table 3.2.1-1

(1) ANSI N18.2 - 1973 and ANSI N18.2a - 1975 Safety Classes 1, 2, 3, and NNS as defined in Section 3.2.2. Class IE is defined in IEEE 308.

(2) Seismic design as defined in FSAR Section 3.2.1.
I = Seismic Design meets Seismic Category I requirements

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks

- = Seismic Category I requirements are not applicable

*(3) Quality Assurance Requirements (Design and Construction Phase)

B = Equipment meets the QA requirements of 10 CFR 50, Appendix B.
For items that do not have a safety class associated with it but are only Seismic Category I, only the pertinent requirements of 10 CFR 50, Appendix B apply.

- = The QA requirements of 10 CFR 50, Appendix B are not mandatory.

a = Quality Assurance provisions will be in accordance with ETSB 11-1. Vendor QA programs based on ASME Section VIII are acceptable to comply with ETSB 11-1 QA requirements.

b = The quality assurance program for fire protection, which was approved by the NRC during the construction permit review, has been followed. However, for components of the fire protection system designed, specified, procured, manufactured or fabricated, or installed prior to the institution of the Fire Protection Quality Assurance Program (February 18, 1977), the program has been followed to the extent practicable.

(4) Represents code class upgrading. As permitted by paragraph NA-2134 of the ASME Code, Section III, this component is upgraded from the minimum required code class to a higher code class.

(5) For safety classifications of other piping and associated valves in the reactor coolant system and other auxiliary systems, refer to the engineering flow diagrams in Chapters 5, 6, and 9. Code classes are those required by the safety class.

(6) Applies only to bolting involved with coastdown function.

(7) This component is Safety Class 3 under the definition of 2.2.3(1), (3), or (4) of ANSI N18.2-1973 and qualifies for no special seismic design by meeting the four conditions listed in FSAR Section 3.2.1.2. Portions of systems in which this component is located that perform the same safety function likewise qualify for no special seismic design.

(8) Classified on the basis that flow restriction is provided in the piping.

(9) The Reactor coolant system supports are not stamped. They are also not part of the reactor coolant pressure boundary and, therefore, are not included in any code class.

(10) Failure could cause releases of radioactivity.

(11) Portions of containment boundary.

(12) Deleted

(13) Deleted

(14) There is no specific code which classifies the diesel generator as Safety Class 3. IEEE 323 is the governing design code.

(15) Instrumentation - Instrumentation required to actuate, maintain operation of, or detect failure of equipment needed to safe shutdown, isolate, and maintain the reactor in a safe condition and prevent uncontrolled release of radioactivity from the plant is Class IE/Seismic Category I. Instrumentation designated as Class IE/Seismic Category I includes as Seismic Category I all sensing lines, instrument valves, and instrument racks. Instrument racks containing Class IE equipment are also considered Class IE. Systems noted as Class IE may also contain non-Class IE equipment. Refer to Chapter 7 for specific identification of Class IE equipment.
Instrument-sensing lines that are connected to ASME Classes 1 or 2 process piping or vessels and that are used to actuate or monitor safety-related systems are not less than ASME Class 2, Seismic Category I from their connections at the process piping or vessel to the sensing instrument. Similarly, instrument-sensing lines connected to ASME Class 3 systems are not less than ASME Class 3, Seismic Category I.
The instrument itself is specifically exempted from the ASME program (per NCA-1130, paragraph C of the ASME Boiler and Pressure Vessel Code) and should not be hydrostatically tested. This also applies to the associated capillary on applicable instruments.

(16) This piping is stress analyzed to the Class 3 requirements of ASME B&PV Code, Section III, designed and fabricated in accordance with ANSI Power Piping Code B31.1, with a 10 CFR 50 Appendix B program applied.

(17) Piping which serves hose stations and standpipes required to protect shutdown equipment is designed to ANSI B31.1 requirements and is seismically qualified. For

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks

NFPA 805 applications, Reference NFPA 805 Chapter 3, Section 3.6.4 and LAR Table B-1 as accepted in Amendment No. 133 to Renewed Facility Operating License No. NPF-63 (dated June 28, 2010).

(18) Those portions of this system whose failure may have an adverse effect on a nearby safety-related component are seismically supported and seismically designed and are subject to the appropriate QA requirements.

(19) The reinforced concrete mat and walls of the Unit 1 Turbine Building between Column Line 42 (approximate) and 43 (approximate) are designed and constructed to Seismic Category I requirements due to the presence of the diesel generator service water pipe tunnel and Class 1 electrical cable area above the pipe tunnel (see Figure 1.2.2-60). This area is designed and constructed to withstand the collapse of the Turbine Building concurrent with an SSE.

(20) Provides mechanical support for Safety Class 1 component.

(21) Will be designed and fabricated to the applicable portions of ASME III, although it is not classified as ANS Safety Classes 1, 2, or 3.

(21A) Fuel pool nozzles will be considered from the fuel pool liner to the first shop girth weld.

(22) Provides support to the Safety Class 1 pressure boundary conduit.

(23) Quality Classification

A - Safety-related

B - Nonsafety Seismic, or falls under Regulatory Guide 1.97 (Instrumentation for Light-Water-Cooled Nuclear Power Plants to assess plant and environs conditions during and following an accident) or falls under Regulatory Guide 1.155 (Station Blackout). See Calc 8S44-P 101 for a listing of Station Blackout Components.

C - Radwaste

D - Fire Protection

E - Nonsafety, Nonseismically Designed

*(24) Quality Assurance Requirements (Operations Phase)

Q - QA requirements will meet 10 CFR 50, Appendix B criteria.

R - QA requirements will be ETSB 11-1 QA requirements as a minimum. Optionally "Q" requirements may be imposed.

F - QA requirements will meet fire protection QA requirements as a minimum. Optionally "Q" requirements may be imposed.

\- QA requirements of 10 CFR 50, Appendix B are not mandatory.

(25) The code and code class for individual components in the liquid waste processing system can be found on Table 11.2.1-7.

(26) The code and code class for individual components in the solid waste processing system can be found on Table 11.4.2-4.

(27) The ETSB 11-1 QA applies to components listed in Table 11.4.2-4 except those listed as manufacturer's standard.

(28) ASME III code applies to oil cooler and trip/throttle valve only.

(29) Not stamped.

(30) Structure is designed to withstand design wind/tornado loadings and missile impacts.

(31) Valve operators for active valves, as listed in Tables 3.9.3-13 and 3.9.3-14, are motor, solenoid, or electrohydraulic and qualified to meet IEEEs 344 and 323 standards.

(32) Containment water level and containment pressure indication are redundant and Class 1E up to the isolation device at the SPDS cabinet.

(33) Radiation Monitor is not redundant.

(34) The fuel oil storage tanks are reinforced concrete tanks with steel liners. Although these tanks are not classified as Safety Class 3, they are designed and constructed commensurate with their intended safety function.

(35) See FSAR Figure 9.5.5-1 for ASME III/non-ASME III boundary.

(36) This table is an overview of all the structures, components, and systems. Refer to the Component Q List (2165 S 2241) for further details.

(37) The reactor internals were fabricated prior to implementation of subsection NG of the ASME Code. However, the reactor internals were designed and fabricated

TABLE 3.2.1-1 CLASSIFICAITON OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Safety Class(1)	Code	Code Class	Seismic Category(2)	Quality Class(23)	Remarks

consistent with ASME Code requirements.

(38) Containment sump screens are seismically designed.

(39) Modifications to site grade elevations will be subject to the pertinent provisions of the Quality Assurance Program to assure that site flooding analyses remain valid.

(40) Modifications or maintenance affecting the structural integrity of these items will be subject to the pertinent provisions of the Quality Assurance Program.

(41) Designed and fabricated in accordance with Section VIII of ASME B&PV Code 1971 Edition through Summer 1973 Addenda. Accumulator tanks 1A-SA and 1C-SB (for PORV's 1RC-P527SA-1 and 1RC-P529SB-1) meet the design criteria of ASME Code Section III, Class 3. The vessel stamping remains Section VIII U Stamp.

(42) The 1" check valves 3SI-V396SA-1, 3SI-V400SB-1, 3SI-V669SA-1, and 3SI-V671SB-1 are not ASME Section III valves.

(43) The operating components of the chiller are quality Classification A. Specification CAR-SH-BE-05 lists those components that are ASME Section III. The refrigerant piping and tubing within the operating pressure boundary are quality Classification A. The refrigerant transfer system is quality Classification B. The break between these quality classes must occur at Valves A and C on the transfer system. The tubing from the 14-inch suction elbow up to and including Valves A and C on the transfer system and the tubing up to and including the relief valve marked Part Number 34 on Drawing 1364-19167 are quality Class A. The piping that runs from the refrigerant transfer system storage tank to the chiller is quality Classification A from the chiller to the first valve from the storage tank. The valve is marked Number 66 and Letter E on Drawing 1364-19166.

(44) Refer to FSAR Section 9.5.3 for more information on N/E AC lighting requirements.

(45) It is designed and fabricated to the applicable portions of ASME III, although it is not code stamped by an Authorized Nuclear Inspector.

(46) Deleted

(47) See FSAR Figures 6.2.2-1, 9.1.3-1, and 9.1.3-2 for ASME Section III/non-ASME boundaries.

(48) The components associated with the MFIV control panel and the components mounted on the accumulator panel are safety-related and seismically mounted.

(49) While structural integrity of all NIS console components is assured, only the NIS Power Range channels have been qualified to Seismic Category I. Both ex-core NI channels of the NFMS (PAM) have been qualified to Seismic Category I.

(50) Designed and fabricated to the applicable portions of ASME III, but downgraded from ANSI Safety Class 3 to NNS.

(51) As allowed by EC 276441, Diesel Generator Lube Oil Heater may be designed and fabricated in accordance with either ASME Section III Class 3 (1977 edition – Summer 77 Addenda), or ASME Section VIII Div. 1 (2007 edition – Summer 2009 Addenda). Safety Class and Quality Class must remain 3 and ‘A’ respectively (Section VIII heaters must be commercially dedicated for use in a safety-related application).

Reference Letters: EB-FC-587, EB-FC-613, CO-00916, EB-C-06787, HO-84122

* These notes for historical purposes only.

TABLE 3.3.0-1

LISTING OF TORNADO AND WIND CRITERIA FOR SEISMIC CATEGORY I STRUCTURES

	Wind - Tornado CRITERION
Containment Structure	Ta
Containment Internal Structures	Tb
Containment Crane Supports	Tb
Reactor Auxiliary Building	Ta
Fuel Handling Building	Ta
New and Spent Fuel Storage Pools	Tb
New and Spent Fuel Storage Racks	Tb
Waste Processing Building	Ta
Emergency Service Water and Cooling Tower Makeup Water Intake Structure which includes the Screening Structure and Pumping Chambers	Ta
Main Dam and Spillway	Ta
Auxiliary Dam and Spillway	Ta
Auxiliary Reservoir Separating Dike	Ta
Emergency Service Water Screening Structure and Associated Intake Channel	Ta
Emergency Service Water Discharge Structure and Associated Discharge Channel	Ta
Diesel Generator Building	Ta
Condensate Storage Tank	Tb
Refueling Water Storage Tank	*
Diesel Fuel Oil Storage Tank Building	Ta
Electrical Manholes for Underground Seismic Category I Cables of the Auxiliary and Emergency Power Systems	Ta

NOTES TO TABLE 3.3.0-1 (Cont'd)

Ta - Design of structures to withstand such wind or tornado loading and/or missiles

Tb - Protected from such wind or tornado loading and/or missiles by structures designed to withstand them

* The Refueling Water Storage Tank is not required for plant shut-down following a tornado and, therefore, it is not designed for tornado effects

TABLE 3.3.1-1

EFFECTIVE VELOCITY PRESSURES

(V_{30} = 179 mph)

Z (ft)	q_F (psf)	qp (psf)	qM (psf)
Less than 30	85	121	81
30	106	121	81
50	121	136	95
100	140	155	115
150	153	170	129
200	163	182	140
250	170	191	150
300	178	197	157

TABLE 3.3.1-2

PRESSURE COEFFICIENTS SELECTION GUIDE

				RECTANGULAR BUILDINGS					
				EXTERNAL PRESSURE COEFFICIENTS					
				ROOFS				WALLS	
			INTERNAL PRESSURE COEFFICIENTS Cpl	GABLED		ARCH			
Cp or CD	SPHERES Cf	STACKS Cf		Local Cpl	GLOBAL Cp	Local Cpl	GLOBAL Cp	Local Cpl	GLOBAL Cp
Table 4(f) Ref 3.3.1-2 or Figure 6 Ref 3.3.1-3	Figure 6 Ref 3.3.1-3	Table 15 and Section 6.7 in Ref 3.3.1-1 or Table 4(f) in Ref 3.3.1-2	Table 11 and Section 6.5.4 in Ref 3.3.1-1 or Tables 4(a) and 4(b) in Ref 3.3.1-2	Table 10 and Section 6.5.3.2.4 in Ref 3.3.1-1	Wind Parallel to Ridge: Section 6.5.3.2.1, Ref 3.3.1-1 Wind Perpendicular to Ridge: Section 6.5.3.2.3 and Table 9, Ref 3.3.1-1 or Fig 7 in Ref 3.3.1-2	Table 10 and Section 6.5.3.2.4 in Ref 3.3.1-1	Wind Parallel to Axis: Section 6.5.3.2.1, Ref 3.3.1-1 Wind Perpendicular to Axis: Table 8, Ref 3.3.1-1 or Table 4(e) in Ref 3.3.1-2	Section 6.5.3.1 in Ref 3.3.1-1	Table 7 in Ref 3.3.1-1 or Table 4(a) in Ref 3.3.1-2

TABLE 3.4.1-1

SEISMIC CATEGORY I STRUCTURES,
SYSTEMS AND COMPONENTS REQUIRING
FLOOD PROTECTION

SYSTEM OR COMPONENT		LOCATION*
	CONDITION A**	
1.	Main Dam and Spillway	Main Reservoir
2.	Auxiliary Dam and Spillway	Auxiliary Reservoir
3.	Auxiliary Reservoir Separating Dike	Auxiliary Reservoir
4.	Auxiliary Reservoir Channel	Auxiliary Reservoir
5.	Emergency Service Water Intake Channel	Auxiliary Reservoir and Plant Site
6.	Emergency Service Water Discharge Channel	Auxiliary Reservoir and Plant Site
7.	Emergency Service Water Screening Structure****	Plant Site
8.	Emergency Service Water Discharge Structure****	Plant Site
9.	Emergency Service Water and Cooling Tower Make-up Water Intake Structure****	Plant Site
	CONDITION B***	
1.	Electrical Manholes for Cables of the Auxiliary and Emergency Power System	Plant Site
2.	Cables of the Auxiliary and Emergency Power System	Plant Site
3.	Diesel Fuel Oil Storage Tank Bldg	Plant Site
4.	Diesel Generator Building	Plant Site
5.	Tank Building	Plant Site
6.	Waste Processing Building*****	Plant Site
7.	Fuel Handling Unloading Area Bldg*****	Plant Site
8.	Fuel Handling Building	Plant Site
9.	Reactor Auxiliary Bldg	Plant Site
10.	Containment Building	Plant Site
11.	Part of Turbine Building	Plant Site

* Plant site is shown on Figure 3.4.1-1.

** Condition A: Designed to withstand the effects of the design basis flood level or flood condition.

*** Condition B: Positioned to preclude effects of the design basis flood level or flood condition.

**** These structures are designed for Condition A for those portions exposed to flood, wave, and wind conditions. However, the top decks are located above all calculated water levels meeting Condition B criteria.

***** These structures are designed to preclude effects of the design basis flood level and to withstand flood conditions.

TABLE 3.5.1-1 STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
1) Chemical & Volume Control System					
a) Charging Pump	RAB	EL. 236'	1.2.2-23	A,B, and AB Pumps Installed Separated by walls	Located below grade. Floor above is 2' thick at the H.P. and 1'9" thick at the L.P., 4,000 psi concrete with 0.5% reinforced steel.
b) Demineralizers	RAB	EL. 261'	1.2.2-27	None	Floor above is 2' thick at the H.P. and 1'-9" thick at the L.P., 4000 psi concrete with 0.5% reinforcing steel; side wall 4' thick, 4000 psi concrete with 0.5% reinforcing steel.
c) Letdown, Heat Moderating, and Letdown Chiller Heat Exchangers	RAB	EL. 236'	1.2.2-23	None	Same as 1a above
d) Boric Acid Tank	RAB	EL. 261'	1.2.2-27	None	Same as 1b above
e) Boric Acid Transfer Pumps	RAB	EL. 261'	1.2.2-27	None	Same as 1b above

TABLE 3.5.1-1 STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
f) Volume Control Tank	RAB	EL. 261'	1.2.2-27	None	Same as 1b above
2) Component Cooling Water System					
a) CCW Heat Exchanger	RAB	EL. 236'	1.2.2-23	A & B Trains, widely separated	Same as 1a above
b) CCW Pumps	RAB	EL. 236'	1.2.2-23	A & B Trains, widely separated	Same as 1a above
c) CCW Surge Tank	RAB	EL. 305'	1.2.2-35	Tank has baffle, separating A & B Trains	Roof is 2' thick 4000 psi concrete with 0.5% reinforcing steel; side wall is 3' thick, 4000 psi concrete with 0.7% reinforcing steel; wall facing Turbine Bldg. is 4' thick, 4000 psi concrete with 0.5% reinforcing steel.
3) Service Water System					
a) Emergency Service Water Pumps	Emergency Service Water	EL. 262'	9.2.1-1 9.2.1-2	A & B Trains, separated by walls	Located inside concrete intake structure approximately 1200' from the south end of the Turbine Building

TABLE 3.5.1-1 STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
b) Service Water Booster Pumps	RAB	EL. 236'	1.2.2-23	A & B Trains, widely separated	Same as 1a above
4) Fuel Pool Cooling System					
a) Fuel Pool Heat Exchangers	FHB	EL. 236'	1.2.2-56	A & B Trains	Two floors are above at 261' and 286' each 2' thick at the H.P. and 1'-9" thick at the L.P., 4000 psi concrete with approx. 1% reinforcing steel.
b) Fuel Pool Cooling Pumps	FHB	EL. 236'	1.2.2-56	A & B Trains	Same as 4a above
c) New & Spent Fuel Pools	FHB	EL. 286'	1.2.2-55	Not Applicable	Walls above EL 286' are 3' thick 4000 psi concrete with 0.5% reinforcing steel; roof at EL 336' is 2' thick 4000 psi concrete with 0.5% reinforcing steel.
5) Condensate Storage Tank	Storage Tank Building	EL. 261'	1.2.2-84	None	Roof 2'-6" thick, walls 3'-0" thick 4000 psi concrete with approx. 0.5% reinforcing steel.
6) Waste Gas Decay Tanks	WPB	EL. 236'	1.2.2-48	Not applicable	Roof at El 261' 4'-0" thick, 4000 psi concrete with 0.7% reinforcing steel.

TABLE 3.5.1-1 STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
7) Diesel Oil Storage Tanks & Fuel Oil Transfer Pumps	In Yard	Below grade	1.2.2-2	Tanks separated into four compartments, but only the two east compartments are used; A & B transfer pumps	Concrete roof 2'-0" thick, 4000 psi concrete with approx. 0.5% reinforcing steel.
8) Above ground Diesel Fuel piping (3FO2-42SA-1)	In Yard	EL. 261'	1.2.2-2	Separated from "B" Diesel Fuel piping by internal and external walls	Inside 4000 psi reinforced concrete barrier with a minimum concrete thickness of 2' and approx. 0.5% reinforcing steel, barrier extends 5' below grade.
9) Auxiliary Feedwater System	RAB	EL. 236'	1.2.2-23	A, B, and AB pumps separated by internal walls	Same as 1a above
10) Main Steam & Safety Valves	M.S. & FW pipe tunnel outside containment RAB	EL. 263'	1.2.2-27	15 Code Safety Valves	Missile Barrier above - 2' thick 4000 psi concrete with 0.55% reinforcing steel.

TABLE 3.5.1-1 STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
11) Main Control Room	RAB	EL. 305'	1.2.2-35	Not Applicable	Roof is 2' thick, 4000 psi concrete with 0.5% reinforcing steel; wall facing Turbine Bldg is 3' thick, 4000 psi concrete with 0.7% reinforcing steel; double wall at 2" gap is 3' thick (each wall), 4000 psi concrete with 0.7% reinforcing steel.
12) Emergency Diesel Generators	In Yard	EL. 261'	1.2.2-2	A & B D-G's for the unit separated by walls	Located in 4000 psi reinforced concrete building with concrete thickness of 2' (minimum) and approx. 0.5% reinforcing steel.
13) Emergency Electrical Equipment & Battery Rooms	RAB	EL. 286'	1.2.2-31	A & B Train Equipment	2' thick roof with 4000 psi concrete 0.55% reinforcing steel and 3' thick wall with 4000 psi concrete, 0.7% reinforcing steel.
14) Switchgear Room Air Conditioning Units	RAB	EL. 286'	1.2.2-31	A & B Trains Separated	Same as 12)
15) H & V Cooling Unit for Auxiliary Feed Pump, CCW Pump & Charging Area	RAB	EL. 236'	1.2.2-23	A & B Trains Separated	Same as 1a)

TABLE 3.5.1-1 <u>STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY</u>

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
16) Control Room II and V Equipment Room	RAB	EL. 305'	1.2.2-35	A & B Trains Separated	Same as 12)
17) Charging Pump Compartment Coolers	RAB	EL. 236'	1.2.2-23	A & B Trains Equipment	Same as 1a)
18) Containment Fan Coolers	RCB	EL. 286' EL. 286'	1.2.2-3 1.2.2-7	A & B Trains Equipment	Same as 19) below
19) Pressurizer Air-Operated Relief Valves	RCB	In pressurizer relief piping		None	Same as 19) below
20) Containment	RCB		1.2.2-2	Not Applicable	RCB walls 4.5' thick, 5000 psi concrete with 4% reinforcing steel; dome is 2.5' thick, 4000 psi concrete with 4% reinforcing steel.
21) Reactor Coolant Sample Heat Exchanger	RAB	EL. 236'	1.2.2-23	None	Same as 1a)
22) Essential Service Chilled Water System					
a) HVAC Chillers	RAB	El 261'	1.2.2-27	A & B Trains widely separated	Same as 1b above

TABLE 3.5.1-1 STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
b) Chilled Water Pumps	RAB	El 261'	1.2.2-27	A & B Trains widely separated	Same as 1b above
c) Condenser Water Pumps	RAB	El 261'	1.2.2-27	A & B Trains widely separated	Same as 1b above
d) Closed Expansion Tanks	RAB	El 261'	1.2.2-27	A & B Trains widely separated	Same as 1b above
e) Air-Handling Units	FHB	El 261'	1.2.2-55	A & B Trains	Floor above is 2' thick at the HP and 1' -9" at the LP, 4000 psi concrete with approx. 1% reinforced steel.
	Sec. Waste Treat Area	El 236'	1.2.2-48	A & B Trains	Floor above is 2' thick at the HP and 1' -9" at the LP, 4000 psi concrete with approx. 1% reinforced steel.
	RAB	Various	1.2.2-31 thru 1.2.2-44	A & B Trains	Same as 1a, 1b, 2c, 12 above for respective elevations. El 190' and El 216' are below grade.

TABLE 3.5.1-1 STRUCTURES, SYSTEMS, AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN OR WHOSE DAMAGE BY MISSILES COULD RESULT IN SIGNIFICANT RELEASE OF RADIOACTIVITY

Structure, System Component	Building	Location	General Arrangement Fig. No.	Redundancy, Separation	Barrier Materials & Thickness
23) Containment Equipment Hatch	RCB	El 298'	1.2.2-2, 1.2.2-27, 1.2.2-11		Concrete walls around containment hatch above E1 286’ are min 3’ thick 4000 psi concrete with -2% reinforcing steel for g line wall and -4% for H line wall. Removable missile shield wall above E1 286’ is 2’ -6$_{11}$ thick 4000 psi concrete with -1% reinforcing steel: removable missile shield roof at E1 317’ is 2’ -6 1/2$_{11}$ thick 4000 psi concrete with -2% reinforcing steel

TABLE 3.5.1-2 OUTDOOR SAFETY-RELATED STRUCTURES, SYSTEMS, AND COMPONENTS PROTECTED AGAINST TORNADO GENERATED MISSILES

STRUCTURE, SYSTEM COMPONENT	MISSILE PROTECTION AFFORDED
1) Containment Buildings	Exterior walls and domes are designed in accordance with Section 3.5.1.
2) Reactor Auxiliary Building	Exterior walls and roofs are designed in accordance with Section 3.5.1.
3) Waste Processing Building	Exterior walls up to EL 261 and roofs at El 261 are designed in accordance with Section 3.5.1.
4) Fuel Handling Building	Exterior walls and roofs are designed in accordance with Section 3.5.1.
5) Emergency Service Water and Cooling Tower Makeup Water Intake Structure	Exterior walls and roofs are designed in accordance with Section 3.5.1.
6) Emergency Service Water Screening Structure	Exterior walls and roofs are designed in accordance with Section 3.5.1.
7) Condensate Storage Tank Enclosure	Exterior walls and roofs are designed in accordance with Section 3.5.1.
8) Reactor Make-up Water Storage Tank Enclosure(1)	Exterior walls are designed in accordance with Section 3.5.1.
9) Diesel Fuel Oil Storage Tank Structure	Exterior walls and roofs are designed in accordance with Section 3.5.1.
10) Diesel Generator Building	Exterior walls and roofs are designed in accordance with Section 3.5.1.
11) Seismic Category I Electrical Manholes	Exterior walls and roofs are designed in accordance with Section 3.5.1.
12) Refueling Water Storage Tank Enclosure	Exterior walls are designed in accordance with Section 3.5.1.
13) All HVAC Air Intakes and Exhausts for Safety-Related Systems	Are provided with missile protective concrete wall barriers
14) Diesel Generator Combustion Air Intake and Exhaust	Are provided with missile protective concrete wall barriers

TABLE 3.5.1-2 OUTDOOR SAFETY-RELATED STRUCTURES, SYSTEMS, AND COMPONENTS PROTECTED AGAINST TORNADO GENERATED MISSILES

STRUCTURE, SYSTEM COMPONENT	MISSILE PROTECTION AFFORDED
15) Underground Class IE Cabling	Are separated to the greatest extent possible according to the criteria set forth in Section 3.5.1; provided with an adequate depth of earth cover and/or a concrete slab covering, and in areas where redundant lines must cross paths, a concrete slab is placed between the redundant lines.
16) Underground Safety-Related Piping	Are separated to the greatest extent possible according to the criteria set forth in Section 3.5.1; are provided with an adequate depth of earth cover and/or a concrete slab covering, and in areas where redundant lines must cross paths, a concrete slab is placed between the redundant lines.
17) Containment Equipment Hatch	Exterior walls and roofs are designed in accordance with Section 3.5.1.
18) Above Ground Exterior Piping	Are provided with missile protective concrete barriers. Barriers encase the piping and extend from exterior concrete walls designed in accordance with Section 3.5.1 to below grade where an adequate depth of earth cover is provided.

Note: (1) The refueling water storage tank and the reactor make-up water storage enclosures are not provided with a roof. In the unlikely event that a postulated tornado missile strike causes complete loss of a tank and its contents, ample time is available to bring the plant to a shutdown, as required by Technical Specifications.

Note: (2) As discussed in Section 3.5.1.4, TMRE is an alternative methodology for determining whether protection from tornado-generated missiles is required. Table 3.5.1-2a lists conditions where the TMRE methodology has demonstrated that tornado missile protection is not required.

TABLE 3.5.1-2a SAFETY-RELATED STRUCTURES, SYSTEMS, AND COMPONENTS THAT DO NOT REQUIRE PROTECTION FROM TORNADO GENERATED MISSILES BASED ON TORNADO MISSILE RISK EVALUATOR METHODOLOGY

System	Component
Auxiliary Feedwater	Turbine Driven Auxiliary Feedwater Pump exhaust pipe (3MS16-185SAB-1) is exposed to potential tornado missiles.
6.9 kV Standby AC Power, Emergency Diesel Generators (EDG)	“A” train Diesel Fuel Oil supply line (3FO2-42SA-1) to the Day Tank in the Diesel Generator Building is exposed to potential tornado missiles through Security door 1FP-D1133.
	Electrical conduits (17179Q SA and 16255V SA) in the Diesel Generator Building, EL 261’ common corridor, are exposed to potential tornado missiles through multiple openings in the corridor exterior wall (two HVAC vent openings with manbarriers installed, Unit 1 Security door 1FP-D1133, and Unit 2 Security door 1FP-D1134).
	Inverted neck vent lines 7LO6-34-1, 7LO6-36-1, 7EA8-11-1, and 7EA8-13-1 located on the Diesel Generator Building roof are exposed to potential tornado missiles.
	Conduits 17199K SA in the “A” EDG room and 17196X SB in the “B” EDG room are exposed to potential tornado missiles through the east exterior wall air intake louvers for each room.
	“A” & “B” train EDG Fuel Oil Return lines are exposed to potential tornado missiles through the east exterior wall air intake louvers.
Emergency Service Water (ESW)	“A” train electrical conduits in the ESW Intake Screening Structure are exposed to potential tornado missiles through Security door 1FP-D1336. Affected conduits are 17049M SA, 12295C SA, 12219M SA, 12312D SA, 16091H SA, 16091ESA, 16091D SA, 16091C SA, 16091G SA, 16091A SA, 16149J SA, 12296J SA, and 12219J SA.
	Conduits 12293E-SA and 13292C-SA, “A” Traveling Screen Wash supply line, “A” Traveling Screen motor, and Cabinet Y21-C7-ESF-A inside the “A” ESW pump room are exposed to potential tornado missiles through penetration seals E2264, P4042, and E2266 in the “A” ESW Intake Structure east exterior wall.
	The “A” or “B” ESW Traveling Screens are exposed to potential tornado missiles through a steel checkered plate covering the coarse screen and stop log guides.
	The “A” or “B” ESW Traveling Screens are exposed to potential tornado missiles above the water if the Main Reservoir is at the lowest level allowed by Technical Specifications.
	Submerged SSCs: • Auxiliary Reservoir Traveling Screen Train A • Auxiliary Reservoir Traveling Screen Train B • ESW Pump Train A • ESW Pump Train B • Main Reservoir 1SW-3 and 1SW-4 • Aux Reservoir ESW 1SW-1 and 1SW-2

System	Component
Battery Room Ventilation	The EL 305’ HVAC exhaust plenums for both the 1A-SA and 1B-SB Battery Rooms within the Main Steam Penthouse on the RAB EL 305’ roof are susceptible to missiles. The EL 305’ HVAC exhaust duct and motor operator for “A” and “B” RAB SWGR RM (1AV-11:002 and 1AV-13:002) are exposed to potential tornado missiles.
Main Steam	1MS-81, 1MS-83, and 1MS-85 are exposed to potential tornado missiles through two Main Steam pipe openings in the Main Steam Penthouse on the RAB EL 305’ roof in the east penthouse wall.
	The Main Steam Safety Relief Valve vent pipes/stacks and the Main Steam Power Operated Relief Valve vent pipes/stacks on the RAB EL 305’ roof are exposed to potential tornado missiles.
Main Steam Tunnel Ventilation	Electrical conduits 112751R SA and 12751A SA and supply fan S64 S3 SA located in the Main Steam EL 305’ HVAC “A” Train Supply Air Intake pillbox are exposed to potential tornado missiles through Security door 1FP-D0515 and through louvered HVAC air intakes with steel manbarriers.
	Electrical conduits 12753A SB & 12753J SB and supply fan S65 S3 SB located in the Main Steam EL 305’ HVAC B Train Supply Air Intake pillbox are exposed to potential tornado missiles through Security door 1FP-D0516 and through louvered HVAC air intakes with steel manbarriers.
Essential Services Chilled Water	The “A” and “B” train Essential Services Chilled Water System Expansion Tanks 1CH-E085 and 1CH-E086 and the 2” connecting pipe at RAB EL 324’ are exposed to potential missile pipe through outside air intake openings in the Reactor Auxiliary Building EL 324’ exterior walls. There are 3 large openings through 41 line wall and 3 through 45 line wall.
Reactor Auxiliary Building	The Reactor Auxiliary Building roof HVAC Exhaust Stack 1AV-ERABVS is exposed to potential tornado missiles.
	Reactor Auxiliary Building EL 305’ Outdoor Ambient Air Pressure Sensing Instrument Tubing PDT-1AV-4834-A1SAHP-T378 (A Train) and PDT1AV4834B1SBHP-T379 (B Train) are exposed to potential tornado missiles.
ESW Intake Structure Ventilation	Outdoor air temperature elements TE-01EV-6589ASA, TE-01EV-6589BSB, TE-01EV-6591ASA, and TE-01EV-6591BSB are vulnerable to potential tornado missiles.
	Electrical conduit 17072F-SB inside the “B” ESW electrical room is exposed to potential tornado missiles through Security door 1FP-D1173 opening.
Fuel Handling Building Ventilation	Outdoor Ambient Air Pressure Sensing Instrument Tubing PDT-*1FL-5027ASA-T368 and PDT-*1FL-5027BSB-T370 are exposed to potential tornado missiles.
Diesel Generator Building Ventilation	Outdoor air temperature elements TE-6902A-SA and TE-6902B-SB are vulnerable to potential tornado missiles.

TABLE 3.5.1-3

CHARACTERISTICS OF TORNADO-GENERATED MISSILE SPECTRUM

	Missile	Weight In Lbs.	Impact Area Sq. Ft.	Impact Velocity Ft./Sec.
1)	Wood Plank 4 in. x 12 in. x 12 ft. long	200	.333	422
2)	Steel Pipe 3 in. diameter x 10 ft. long schedule 40	78	.0155	211
3)	Steel Rod 1 in. diameter x 3 ft. long	8	.00545	317
4)	Steel Pipe 6 in. diameter x 15 ft. long, schedule 40	285	.0388	211
5)	Steel Pipe 12 in. diameter x 15 ft. long, schedule 40	743	.1014	211
6)	Utility Pole 13½ in. diameter x 35 ft. long	1490	.994	211
7)	Automobile	4000	20	106

These missiles are considered to be capable of striking in all directions. Missiles 1, 2, 3, 4 and 5 are considered at all elevations and Missiles 6 and 7 for elevations up to 30 feet above the highest grade level within 1/2 mile of the facility structures.

TABLE 3.5.1-4

SUMMARY OF CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

Calculation Data

Typical Examples of Postulated Missiles	Missile Weight (lbs.)	Impact Velocity (ft./sec)	Kinetic Energy (ft.-lb.)	Penetration (in.)	Assumptions
1. Mechanism Top Cap and Drive Rod Assembly Impacting on same Missile Shield Spot (See note below)	133	150	46,757	0.80	Drive shaft further pushes the plug into shield.
2. Drive Rod Assembly Latched to Mechanism	1,500	12.1	1,490	0.057	-------

Note: The control rod drive mechanisms (CRDMs) were replaced with the reactor vessel closure head during RFO-22. The replacement CRDM integrated rod travel housing (IRTH) is a one-piece design that eliminates the top cap; therefore, the results in this table are conservative.

TABLE 3.5.1-5

VALVE - MISSILE CHARACTERISTICS

Missile Description	Weight (lb.)	Flow Discharge Area (in.^{2})	Thrust Area (in. ^{2})	Weight To Impact Area (in. ^{2})	Impact Area Ration (psi)	Velocity (fps)
Safety Valve Bonnet (3" x 6" or 6" x 6")	350	2.86	80	24	14.60	110
3-Inch Motor-Operated Isolation Valve Bonnet (plus motor and stem)	400	5.5	113	28	14.1	135
3-Inch Air-Operated Relief Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
4-Inch Air Operated Spray Valve	200	9.3	50	50	4	190

TABLE 3.5.1-6

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}	9.0 in.^{2}
Missile Weight	11.0 lb.	15.2 lb.
Area of Impact	3.14 in.^{2}	3.14 in.^{2}
[Missile Weight] / Impact Area	3.5 psi	4.84 psi
Velocity	20 ft./sec.	120 ft./sec.

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

Characteristics	"without well"	"with well"
Flow Discharge Area	0.11 in.^{2}	0.60 in.^{2}
Thrust Area	3.14 in.^{2}	3.14 in.^{2}
Missile Weight	4.0 lb.	6.1 lb.
Area of Impact	3.14 in.^{2}	3.14 in.^{2}
[Missile Weight] / Impact Area	1.27 psi	1.94 psi
Velocity	75 ft./sec.	120 ft./sec.

TABLE 3.5.1-7

CHARACTERISTICS OF OTHER MISSILES
POSTULATED WITHIN CONTAINMENT

	Reactor Coolant Pump Temperature Element	Instrument Well of Pressurizer	Pressurizer Heaters
Weight	0.25 lb.	5.5 lb.	15 lb.
Discharge Area	0.50 in.^{2}	0.442 in.^{2}	0.80 in.^{2}
Thrust Area	0.50 in.^{2}	1.35 in.^{2}	2.4 in.^{2}
Impact Area	0.50 in.^{2}	1.35 in.^{2}	2.4 in.^{2}
$\frac{\text{Missile Weight}}{\text{Impact Area}}$	0.5 psi	4.1 psi	6.25 psi
Velocity	260 ft./sec.	100 ft./sec.	55 ft./sec.

TABLE 3.5.1-17

HIGH ENERGY SYSTEM INSTRUMENT WELL CREDIBLE
MISSILE GENERATION OUTSIDE CONTAINMENT

System	Instrument Wells
Main Steam System	TE-MS-0430 installed on top of piping, ejection assumed 15° max. from vertical, does not impact on safety related equipment.
Main Feedwater	None Located in RAB
Chemical and Volume Control (Letdown)	TW - 7242 TI - 7242 I-TE-143/I-TW-143 I-TE-144/I-TW-144 are located in Letdown heat exchanger compartment. Letdown heat exchanger is not required for safe shutdown. Compartment prevents the communication of credible missiles with adjoining areas.
Steam Generator Blowdown	None located in RAB
Auxiliary Feedwater	None located in RAB

TABLE 3.5.2-1 <u>BARRIERS DESIGNED FOR MISSILES</u>

Structure	Protection Afforded	Missile Type
Reactor vessel primary shield wall	The reactor vessel is protected from missiles originating outside the primary shield wall	Internal missiles resulting from pressurized components or rotating equipment
Steam generator/ secondary shield wall	The Containment and equipment located between the SG/secondary shield wall and the Containment are protected from missiles generated within the secondary shield wall	Internal missiles resulting from pressurized components or rotating equipment
Control rod drive mechanism barrier	The missile barrier prevents the ejection into the Containment of the worst postulated missile from the head area	Internal missiles resulting from pressurized components
Reactor Containment Building	The Containment is designed to prevent external missiles from damaging the liner	External and internal missiles
Reactor Auxiliary Building exterior walls, roof and removable hatch covers	Equipment located within the RAB is protected from external missiles	External missiles
Fuel-Handling Building exterior walls, roof and removable hatch covers	Equipment located within the FHB is protected from external missiles	External missiles
Control Room exterior walls and roof	Equipment located within the Control Room is protected from external missiles	External missiles
Waste Processing Building exterior walls, roof and removable hatch covers	Equipment located within the WPB is protected from external missiles	External missiles
ESWS Screening, Intake, and Discharge Structures walls, roof, and removable hatch covers	Equipment located within the structures is protected from internal and external missiles	Internal and external missiles

TABLE 3.5.2-1 BARRIERS DESIGNED FOR MISSILES

Structure	Protection Afforded	Missile Type
Diesel Generator Building exterior walls and roof	Equipment located within the DGB is protected from external missiles	External missiles
Condensate Storage Tank Building walls and roof	The CSTB is protected from external missiles	External missiles
Diesel Fuel Oil Storage Tank Building walls, roof, and removable hatch covers	The DFOSTB is protected from external missiles	External missiles
Reactor Auxiliary Building walls and roof	The containment equipment hatch is protected from external missiles by concrete missile blocks.	External missiles
Local Concrete Barriers and Steel Doors		
Reactor Auxiliary Building	HVAC valves at Elevation 286 and HVAC fans at Elevation 305 are protected from external missiles	External missiles
	Doors in external walls at Elevation 261 ft. and 305 ft. are protected by concrete barriers	External missiles
	Steel doors protect from external missiles	External missiles
Fuel Handling Building	Door in stair enclosure at Elevation 286 ft. is protected by concrete barriers.	External missiles
Diesel Fuel Oil Storage Tank Building	HVAC intake and exhaust and oil overflow at Elevation 263 ft. is protected by concrete barriers	External missiles
Turbine Building	Electrical conduits and service water piping in pipe tunnel is protected by concrete slab at Elevation 261 ft.	External missiles

TABLE 3.5.3-1

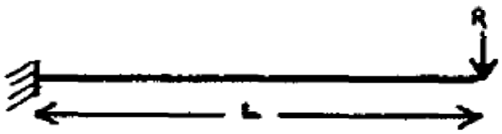
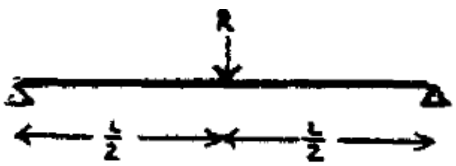
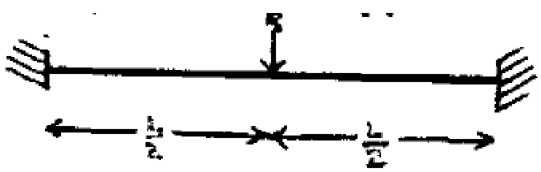
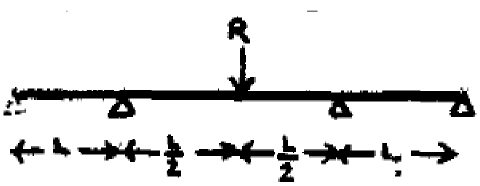
DUCTILITY FACTORS

1)	Tension-reinforced concrete beams and slabs (flexure controls design)	$\mu = \frac{0.05;}{p}$	$\mu \leq 12.5$; $p=\frac{A_s}{bd}$
2)	Doubly reinforced concrete beams and slabs (flexure controls design)	$\mu = \frac{0.05;}{p-p'}$	$\mu \leq 15.0$; $p' = \frac{A_s'}{bd}$
			A_s = area of tension reinforcement
			A_s' = area of compression reinforcement
			b = width of section
			d = depth of section to reinforcement
			p = percentage tensile reinforcement
			p' = percentage compression reinforcement
3)	Concrete beams and slab in region requiring shear reinforcement		
	a) Shear carried by concrete and stirrups	$\mu = 1.3$	ϵ_u = uniform ultimate strain of material
	b) Shear carried completely by stirrups	$\mu = 3.0$	ϵ_y = strain at yeild of material
4)	Concrete columns and walls (compression members)	$\mu = 1.3$	
5)	Structural steel tension member	$\mu = 5.0 \frac{\epsilon_u}{\epsilon_y}$	
6)	Structural steel flexural members		
	a) Open sections (I, WF, T) Members proportioned to preclude lateral and local plastic buckling	$\mu \leq 20$	
	b) Closed sections (pipe box)	$\mu \leq 25.0$	
	c) Members where shear governs design	$\mu \leq 6.0$	
7.	Structural steel columns	$\mu \leq 1.0$	1/r > 60
		$\mu \leq 1.0$	30 ≤ 1/r < 60
		$\mu \leq 6.0$	1/r < 30
			l =effective length of column r =radius of gyration (see AISC 69 Specifications)
8.	Energy absorbers		

Various types of energy absorbers are available. Their ductility or load deflection characteristics are as defined by the supplier.

TABLE 3.5.3-2

RESISTANCE/YIELD DISPLACEMENT VALUES FOR BEAMS

Description	Resistance	Yield
1) Cantilever 	$R = \frac{M_u}{L}$	$x_y = \frac{RL^3}{3EI}$
2) Simply Supported 	$R = \frac{4M_u}{L}$	$x_y = \frac{RL^3}{48EI}$
3) Fixed Supports 	$R = \frac{M_u}{L}$	$x_y = \frac{RL^3}{192EI}$
4) Multi-Span 	$R = \frac{M_u}{L}$	$x_y = \frac{0.011RL^3}{EI}$

R = Yield resistance

x_y = Yield displacement

Mu = Ultimate moment capacity

L = Span

E = Modulus of Elasticity

I = Moment of Inertia

TABLE 3.5.3-3

RESISTANCE/YIELD DISPLACEMENT VALUES FOR RECTANGULAR SLABS*

Yield Displacement at Center

$$x_y = \frac{\alpha R a^2}{EI} (1-\mu^2)$$

where:

x_y = yield displacement

R = Yield resistance

a = Short side of slab

b = Long side of slab

μ = Poisson's ratio

E = Modulus of elasticity

I = Moment of inertia per unit width

M_u = Ultimate moment capacity per unit width

α = Dimensionless numerical factor with fixed values as listed below

(1) Simply supported on all four sides with load at center

$$R = 2\pi M_u$$

b/a	1.0	1.1	1.2	1.4	1.6	1.8	2.0	3.0	∞
α	.01160	.01265	.01353	.01484	.01570	.01620	.01651	.01690	.01695

(2) Fixed supports on all four sides with load at center

$$R = 4\pi M_u$$

b/a	1.0	1.2	1.4	1.6	1.8	2.0	∞
α	.00560	.00647	.00691	.00712	.00720	.00722	.00725

Note: * Source: Timoshenko, S and Woinowsky-Krieger, S

"Theory of Plates and Shells" McGraw-Hill (1959).

TABLE 3.6.2-1

POSTULATED BRANCH LINE BREAK LOCATIONS FOR THE LOCA ANALYSIS OF THE PRIMARY COOLANT LOOP

Location of Postulated Rupture	Type	Break Opening Area*
1. Surge Branch Nozzle	Guillotine	Effective Cross-Sectional Flow Area of the Surge Pipe
2. Accumulator Branch Nozzle	Guillotine	Effective Cross-Sectional Flow Area of the Accumulator Pipe
3. Residual Heat Removal Branch Nozzle	Guillotine	Effective Cross-Sectional Flow Area of the RHR Pipe

* Less Break opening break area will be used if justified by analysis, experiments, or considerations of physical restraints such as concrete walls or structural steel.

TABLE 3.6A-1

PIPE WHIP RESTRAINTS
MAIN STEAM PIPING INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-MS-1-R-1H		X	X	Hard	2MS-32-1SA-1 Loop 1	R-HMS-1-1	C	3.6A-1 & 3.6A-29
R-MS-1-R-3H	X,A	X	X	Hard & CR	2MS-32-1SA-1 Loop 1	R-HMS-1-1	C	3.6A-1 & 3.6A-29
R-MS-1-R-4H	X,A	X		Hard & CR	2MS-32-1SA-1 Loop 1	R-HMS-1-1	C	3.6A-1 & 3.6A-29
R-MS-1-R-5HB		X		Hard & CR	2MS-32-1SA-1 Loop 1	R-HMS-1-5	C	3.6A-1 & 3.6A-29
R-MS-1-R-7S		X		Soft	2MS-32-1SA-1 Loop 1	R-HMS-1-5	C	3.6A-1 & 3.6A-29
R-MS-1-R-9H		X	X	Hard	2MS-32-1SA-1 Loop 1	R-HMS-1-5	C	3.6A-1 & 3.6A-29
R-MS-2-R-10H		X	X	Hard	2MS-32-2SA-1 Loop-2	R-HMS-2-1	C	3.6A-1 & 3.6A-30
R-MS-2-R-12H	X,A	X		Hard	2MS-32-2SA-1 Loop-2	R-HMS-2-1	C	3.6A-1 & 3.6A-30
R-MS-2-R-13H	X,N	X,E		Hard & CR	2MS-32-2SA-1 Loop-2	R-HMS-2-5	C	3.6A-1 & 3.6A-30
R-MS-2-R-14H		X	X	Hard	2MS-32-2SA-1 Loop-2	R-HMS-2-1	C	3.6A-1 & 3.6A-30
R-MS-3-R-15H		X	X	Hard	2MS-32-3SA-1 Loop-3	R-HMS-3-1	C	3.6A-1 & 3.6A-31
R-MS-3-R-17H	X,A	X		Hard & CR	2MS-32-3SA-1 Loop-3	R-HMS-3-1	C	3.6A-1 & 3.6A-31
R-MS-3-R-19HB		X		Hard & CR	2MS-32-3SA-1 Loop-3	R-HMS-3-5	C	3.6A-1 & 3.6A-31
R-MS-3-R-23H		X	X	Hard	2MS-32-3SA-1 Loop-3	R-HMS-3-5	C	3.6A-1 & 3.6A-31
R-MS-3-R-25		X		Hard	2MS-32-3SA-1 Loop-3	R-HMS-3-5	C	3.6A-1 & 3.6A-31
R-MS-1-R-26	X,B			Hard	2MS-32-1SA-1 Loop-1	R-HMS-1-5	C	3.6A-1 & 3.6A-29
R-MS-1-R-27		X		Hard	2MS-32-1SA-1 Loop-1	R-HMS-1-5	C	3.6A-1 & 3.6A-29

*A - Parallel To Axis of Pipe Above
B - Parallel To Axis of Pipe Below
N - North; S-South; E-East; W-West
X - Means Supported in Both Directions, Except Where Otherwise Noted
**CR - Crushable Material Type Restraint
***C - Circumferential Break

TABLE 3.6A-2

PIPE WHIP RESTRAINTS
FEEDWATER PIPING - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-FW-15-R-1CMB****		X,S,W		Soft	2FW16-68SN-1 Loop-2	R-HFW-68-1	C	3.6A-7 & 3.6A-30
R-FW-15-R-2CM****	X,B	X		Hard & CR	2FW16-68SN-1 Loop-2	R-HFW-68-1	C	3.6A-7 & 3.6A-30
R-FW-15-R-4CM****		X,E	X,+y	Hard & CR	2FW16-68SN-1 Loop-2	R-HFW-68-5	C	3.6A-5 & 3.6A-30
R-FW-15-R-5S		X,W		Soft	2FW16-68SN-1 Loop-2	R-HFW-68-5	C	3.6A-7 & 3.6A-30
R-FW-15-R-6H		X,S	X,-y	Hard	2FW16-68SN-1 Loop-2	R-HFW-68-5	C	3.6A-5 & 3.6A-30
R-FW-15-R-7H		X	X	Hard	2FW16-68SN-1 Loop-2	R-HFW-68-5	C	3.6A-5 & 3.6A-30
R-FW-17-R-7CM****		X,N-W		Soft	2FW16-69SN-1 Loop-3	R-HFW-69-1	C	3.6A-7 & 3.6A-31
R-FW-17-R-9‡****	X,B	X		Hard	2FW16-69SN-1 Loop-3	R-HFW-69-2 R-HFW-69-3	C	3.6A-7 & 3.6A-31
R-FW-17-R-16CMS		X		Soft & CR	2FW16-69SN-1 Loop-3	R-HFW-69-5	C	3.6A-5 & 3.6A-31
R-FW-17-R-17H		X	X	Hard	2FW16-69SN-1 Loop-3	R-HFW-69-5	C	3.6A-5 & 3.6A-31
R-FW-13-R-19CM****		X,S-W		Soft	2FW16-67SN-1 Loop-1	R-HFW-67-1	C	3.6A-7 & 3.6A-29
R-FW-13-R-2BCM****	A			CR	2FW16-67SN-1 Loop-1	R-HFW-67-1	C	3.6A-5 & 3.6A-29
R-FW-13-R-26CM		X,N-W		CR	2FW16-67SN-1 Loop-1	R-HFW-67-5	C	3.6A-5 & 3.6A-29
R-FW-13-R-27H		X	X	Hard	2FW16-67SN-1 Loop-1	R-HFW-67-5	C	3.6A-5 & 3.6A-29
R-FW-15-R-29CM	A			CR	2FW16-68SN-1 Loop-2	R-HFW-68-1	C	3.6A-5 & 3.6A-30
R-FW-17-R-30CM	A			CR	2F16-69SN-1 Loop-3	R-HFW-69-1	C	3.6A-5 & 3.6A-31

*A - Parallel to Axis of Pipe Above
B - Parallel to Axis of Pipe Below
N - North; S-South; E-East; W-West
X - Means Supported in Both Directions, Except Where Otherwise Noted
+y - Above The Restraint
-y - Below The Restraint
**CR - Crushable Material Type Restraint
***C - Circumferential Break
**** Component no longer functions as a pipe whip restraint, component remains inside containment.
‡NOTE: The seat plate and crushable material sub-assembly for this restraint has been deleted. The supporting steel shall remain and be tagged R FW-17-R-9.

TABLE 3.6A-3

PIPE WHIP RESTRAINTS
AUXILIARY FEEDWATER - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-AF-59-R-1****	X,A E-S			Hard	2AF6-59SAB-1 Loop-1	R-HAF-93-1	C	3.6A-8
R-AF-59-R-8		X N-W		CR	2AF6-59SAB-1 Loop-1	R-HAF-59-6	C	3.6A-8
R-AF-7-R-9****	X,A N-E			Hard	2AF6-59SAB-1 Loop-2	R-HAF-92-1	C	3.6A-8
R-AF-7-R-17		X,W		CR	2AF6-7SAB-1 Loop-2	R-HAF-7-6	C	3.6A-8
R-AF-60-R-18****	X,A W			Hard	2AF6-60SAB-1 Loop-3	R-HAF-91-1	C	3.6A-8
R-AF-60-R-26		X S-W		CR	2AF6-60SAB-1 Loop-3	R-HAF-60-6	C	3.6A-8

*A - Parallel to Axis of Pipe Above
B - Parallel to Axis of Pipe Below
N - North; S-South; E-East; W-West
X - Means Supported in Both Directions, Except Where Otherwise Noted
+y - Above The Restraint
-y - Below The Restraint
**CR - Crushable Material Type Restraint
***C - Circumferential Break
**** - Component no longer functions as a pipe whip restraint. However, component remains inside containment.

TABLE 3.6A-4

PIPE WHIP RESTRAINTS
STEAM GENERATOR BLOWDOWN - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-BD-3-R-1			X,-y	Soft	2BD4-3SN-1	R-HBD-3-1	C	3.6A-24
R-BD-3-R-3	X,B N-W			CR	2BD4-3SN-1	R-HBD-3-1	C	3.6A-24
R-BD-11-R-8			X,-y	Soft	2BD4-11SN-1	R-HBD-11-1	C	3.6A-24
R-BD-11-R-10	X,B S-W			CR	2BD4-11SN-1	R-HBD-11-1	C	3.6A-24
R-BD-7-R-15		X,S	X,-y	Hard	2BD2-7SN-1	R-HBD-7-1	C	3.6A-24
R-BD-7-R-16		X,S	X,-y	Hard	2BD2-7SN-1	R-HBD-7-1	C	3.6A-24
R-BD-7-R-17		X,W		Hard	2BD2-7SN-1	R-HBD-7-1	C	3.6A-24

*A - Parallel to Axis of Pipe Above
B - Parallel to Axis of Pipe Below
N - North; S-South; E-East; W-West
X - Means Supported in Both Directions, Except Where Otherwise Noted
+y - Above The Restraint
-y - Below The Restraint
**CR - Crushable Material Type Restraint
***C - Circumferential Break

TABLE 3.6A-5 PIPE WHIP RESTRAINTS CHEMICAL AND VOLUME CONTROL SYSTEM - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-CS-83-R-1			X,-y	CR	2CS3-83SN-1	R-HCS-83-1	C	3.6A-10
R-CS-83-R-2		X N-E		Hard	2CS3-83SN-1	R-HCS-84-1	C	3.6A-10
R-CS-83-R-5		X S-W		CR	2CS3-83SN-1	R-HCS-83-4	C	3.6A-10
R-CS-83-R-6		X,E		Soft	2CS3-83SN-1	R-HCS-83-4	C	3.6A-9
R-CS-83-R-9		X N-W		CR	2CS3-83SN-1	R-HCS-83-4	C	3.6A-9
R-CS-83-R-12	X,A N-E			CR	2CS3-83-SN-1	R-HRC-45-4 R-HRC-118-2 R-HRC-118-1 R-HRC-118-3	C	3.6A-9
R-CS-114-R-13		X	N,W	Hard	2CS3-114-SN-1	R-HCS-95-1	C	3.6A-10
R-CS-114-R-15			X	Hard	2CS3-114-SN-1	R-HCS-114-1	C	3.6A-10
R-RC-45-R-1	X,A S-E			CR	1RC3-45SN-1	R-HRC-45-3 R-HRC-45-4 R-HRC-118-1 R-HRC-118-2 R-HRC-118-3	C	3.6A-9
R-RC-45-R-2			X,+y	CR	1RC3-45SN-1	R-HRC-45-1	C	3.6A-9
R-CS-85-R-21		X E-S		Hard	2CS3-85-SN-1	R-HCS-85-4	C	3.6A-10
R-RC-25-R-47			B,+y	Hard	1RC4-25-SN-1	R-HRC-25-1	C	
R-RC-28-R-3	X,A S-N			CR	1RC3-28SN-1	R-HRC-28-4 R-HRC-117-1 R-HRC-117-2 R-HCS-28-1 R-HRC-117-3	C	3.6A-9
R-RC-28-R-4			X,+y	CR	1RC3-28SN-1	R-HRC-28-1	C	3.6A-9
R-CS-87-R-22			X,-y	CR	2CS3-87-SN-1	R-HCS-87-1	C	3.6A-10
R-CS-96-R-30		X S,W		CR	2CS3-96-SN-1	R-HCS-92-1 R-HCS-140-1 R-HCS-91-1 R-HCS-139-1	C	3.6A-10

TABLE 3.6A-5 PIPE WHIP RESTRAINTS CHEMICAL AND VOLUME CONTROL SYSTEM - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-CS-96-R-31		X S-W		Soft	2CS3-96SN-1	R-HCS-138-1 R-HCS-91-1 R-HCS-134-1 R-HCS-90-1	C	3.6A-10
R-CS-88-R-32		X N-E		Soft	2CS3-88-SN-1	R-HCS-140-1 R-HCS-139-1 R-HCS-92-1 R-HCS-91-1	C	3.6A-10
R-CS-88-R-33		X N-E		Soft	2CS3-88-NS-1	R-HCS-139-1 R-HCS-90-1 R-HCS-91-1 R-HCS-138-1	C	3.6A-10
R-CS-88-R-34			X,+y	Hard	2CS3-88-SN-1	R-HCS-88-1	C	3.6A-10

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

+y - Above The Restraint

-y - Below The Restraint

**CR - Crushable Material Type Restraint

***C - Circumferential Break

TABLE 3.6A-6

PIPE WHIP RESTRAINTS
PRESSURIZER SURGE LINE INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-RC-35*-R-21			X,+y	Soft	1RC14-35SN-1	R-HRC-35-1	C	3.6A-23
R-RC-35*-R-22		X S-E		Soft	1RC14-35SN-1	R-HRC-35-4	C	3.6A-23
R-RC-35*-R-25		X N-W		Soft	1RC14-35SN-1	R-HRC-35-4	C	3.6A-23
R-RC-35*-R-26		X N-W		Soft	1RC14-35SN-1	R-HRC-35-6	C	3.6A-23

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

+y - Above The Restraint

-y - Below The Restraint

**CR - Crushable Material Type Restraint

***C - Circumferential Break

TABLE 3.6A-7

PIPE WHIP RESTRAINTS
PRESSURIZER SPRAY LINE - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-RC-25-R-47			X,+y	CR	1RC4-25SN-1	1R-HRC-25-1	C	3.6A-14

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

+y - Above The Restraint

-y - Below The Restraint

**CR - Crushable Material Type Restraint

***C - Circumferential Break

TABLE 3.6A-8

PIPE WHIP RESTRAINTS
SAFETY INJECTION SYSTEM - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-SI-162-R-25		X N-W		CR	1S112-162-SB-1	R-HRC-46-4	C	3.6A-20
R-RC-26-R-9		X		Soft & CR	1RC12-26SA-1	R-HRC-26-6	C	3.6A-20
R-RC-46-R-11		X		CR	1RC12-46SB-1	R-HRC-46-4 R-HRC-46-6	C	3.6A-20
R-RC-27-R-28			X,+y	CR	1RC6-27SA-1	R-HRC-27-3 R-HRC-27-4	C	3.6A-20
R-RC-47-R-30			X,+y	CR	1RC6-47SB-1	R-HRC-47-3 R-HRC-47-4	C	3.6A-20
R-RC-47-R-31	X,A S-E			Hard	1RC6-47SB-1	R-HRC-47-1	C	3.6A-20

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

+y - Above The Restraint

-y - Below The Restraint

**CR - Crushable Material Type Restraint

***C - Circumferential Break

TABLE 3.6A-9

PIPE WHIP RESTRAINTS
RESIDUAL HEAT REMOVAL SYSTEM - INSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
R-RC-12-R-16		X,S	X,-y	CR	1RC12-12SA-1	R-HRC-12-1	C	3.6A-21 3.6A-22
R-RC-51-R-18		X,S	X,-y	CR	1RC12-51SB-1	R-HRC-51-1	C	3.6A-21 3.6A-22

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

+y - Above The Restraint

-y - Below The Restraint

**CR - Crushable Material Type Restraint

***C - Circumferential Break

TABLE 3.6A-10

PIPE WHIP RESTRAINTS
MAIN STEAM PIPING - OUTSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
A-MS-8-RA		X,N	X	Hard	5-MS44-8-1	A-HMS-8-1 A-HMS-8-2	C	3.6A-1
A-MS-8-RB		X,S	X,+y	Hard	5-MS44-8-1	A-HMS-8-1 A-HMS-8-2	C	3.6A-1 3.6A-32.1
A-MS-8-RC		X		Soft	5-MS44-8-1	A-HMS-8-1 A-HMS-8-2	C	3.6A-1 3.6A-32.1
A-MS-8-RD		X		Soft	5-MS44-8-1	A-HMS-8-1 A-HMS-8-2	C	3.6A-2 3.6A-32.1
A-MS-8-RE		X		Soft	5-MS44-8-1	A-HMS-8-1 A-HMS-8-2	C	3.6A-2 3.6A-32.1
A-MS-8-RF		X		CR	5-MS44-8-1	A-HMS-8-1 A-HMS-8-2	C	3.6A-2 3.6A-32.1
A-MS-9-RA		X,N	X	Hard	5-MS44-9-1	A-HMS-9-1 A-HMS-9-2	C	3.6A-1 3.6A-32.1
A-MS-9-RB		X,S	X	Hard	5-MS44-9-1	A-HMS-9-1 A-HMS-9-2	C	3.6A-1 3.6A-32.1
A-MS-9-RC		X		Soft	5-MS44-9-1	A-HMS-9-1 A-HMS-9-2	C	3.6A-1 3.6A-32.1
A-MS-9-RD		X		Soft	5-MS44-9-1	A-HMS-9-1 A-HMS-9-2	C	3.6A-2 3.6A-32.1
A-MS-9-RE		X		Soft	5-MS44-9-1	A-HMS-9-1 A-HMS-9-2	C	3.6A-2 3.6A-32.1
A-MS-9-RF		X		CR	5-MS44-9-1	A-HMS-9-1 A-HMS-9-2	C	3.6A-2 3.6A-32.1

*A - Parallel to Axis of Pipe Above
B - Parallel to Axis of Pipe Below
N - North; S-South; E-East; W-West
X - Means Supported in Both Directions, Except Where Otherwise Noted
+y - Above The Restraint
-y - Below The Restraint
**CR - Crushable Material Type Restraint
***C - Circumferential Break

TABLE 3.6A-11 PIPE WHIP RESTRAINTS FEEDWATER - OUTSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
A-FW-12-R-1		X		Hard	AFW16-12-1	A-HFW-12-1 A-HFW-12-2 A-HFW-12-3 A-HFW-12-4	C	3.6A-6 3.6A-33
A-FW-12-R-2		X		Hard	AFW16-12-1	A-HFW-12-1 A-HFW-12-2 A-HFW-12-3 A-HFW-12-4	C	3.6A-6 3.6A-33
A-FW-12-R-3		X		Hard	AFW16-12-1	A-HFW-12-1 A-HFW-12-2 A-HFW-12-3 A-HFW-12-4	C	3.6A-5 3.6A-33
A-FW-12-R-5		X		Hard	AFW16-12-1	A-HFW-12-1	C	3.6A-7 3.6A-33.1
A-FW-14-R-6		X		Hard	AFW16-14-1	A-HFW-14-1	C	3.6A-7 3.6A-33.1
A-FW-16-R-7		X		Hard	AFW16-16-1	A-HFW-16-1	C	3.6A-7 3.6A-33.1
A-FW-14-R-8		X		Hard	AFW16-14-1	A-HFW-14-1 A-HFW-14-2 A-HFW-14-3 A-HFW-14-4	C	3.6A-6 3.6A-33
A-FW-14-R-9		X		Hard	AFW16-14-1	A-HFW-14-1 A-HFW-14-2 A-HFW-14-3 A-HFW-14-4	C	3.6A-6 3.6A-33
A-FW-14-R-10		X		Hard	AFW16-14-1	A-HFW-14-1 A-HFW-14-2 A-HFW-14-3 A-HFW-14-4	C	3.6A-5 3.6A-33
A-FW-16-R-12		X		Hard	AFW16-16-1	A-HFW-16-1 A-HFW-16-2 A-HFW-16-3 A-HFW-16-4	C	3.6A-6 3.6A-33

TABLE 3.6A-11 PIPE WHIP RESTRAINTS FEEDWATER - OUTSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
A-FW-16-R-13		X		Hard	AFW16-16-1	A-HFW-16-1 A-HFW-16-2 A-HFW-16-3 A-HFW-16-4	C	3.6A-6 3.6A-33
A-FW-16-R-14		X		Hard	AFW16-16-1	A-HFW-16-1 A-HFW-16-2 A-HFW-16-3 A-HFW-16-4	C	3.6A-5 3.6A-33

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

**CR - Crushable Material Type Restraint

***C - Circumferential Break

TABLE 3.6A-12

PIPE WHIP RESTRAINTS
AUXILIARY FEEDWATER - OUTSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
A-AF-1-R-7		X,N		Hard	3AF4-1SA-1	A-HAF-1-1	C	3.6A-5

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

+y - Above The Restraint

-y - Below The Restraint

**CR - Crushable Material Type Restraint

***C - Circumferential Break

TABLE 3.6A-13

PIPE WHIP RESTRAINTS
STEAM GENERATOR BLOWDOWN - OUTSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
A-BD-3-R-1	X,A S-E			CR	2BD4-3SN-1	Note 1	C	3.6A-25
A-BD-13-R-10*		X,W		Soft	3BD4-13SN-1	A-HBD-13-4	C	3.6A-25
A-BD-7-R-11	X,B N-E			Hard	2BD4-7SN-1	Note 1	C	3.6A-25
A-BD-14-R-18		X,W		CR	3BD4-14SN-1	A-HBD-14-3	C	3.6A-25
A-BD-14-R-19		X,S		Soft	3BD4-14SN-1	A-HBD-14-3	C	3.6A-25
A-BD-15-R-27		X,S		Hard	3BD4-15SN-1	A-HBD-15-4	C	3.6A-25

*A - Parallel to Axis of Pipe Above

B - Parallel to Axis of Pipe Below

N - North; S-South; E-East; W-West

X - Means Supported in Both Directions, Except Where Otherwise Noted

+y - Above The Restraint

-y - Below The Restraint

**CR - Crushable Material Type Restraint

***C - Circumferential Break

Note 1: Pipe breaks A-HBD-3-1 and A-HBD-7-1 have been exempted by analysis. However, pipe whip restraints remain in place.

TABLE 3.6A-14

PIPE WHIP RESTRAINTS
CHEMICAL AND VOLUME CONTROL SYSTEM - OUTSIDE CONTAINMENT

Pipe Whip Restraint Identification No.	Protection Direction* Axial	Lat	Vert	Type of Restraint**	Line Designation	Pipe Break No.	Type of Break***	Figure No.
A-CS-300-R-51		X,E		Soft	2CS4-300-SA-1	A-HCS-300-3	C	3.6A-13 3.6A-14
A-CS-300-R-53		X,N		CR	2CS4-300-SA-1	A-HCS-300-3	C	3.6A-13 3.6A-14
A-CS-315-R-81	X,A S-E			Soft	2CS4-300-SA-1	R-HCS-95-1	C	3.6A-13 3.6A-14
A-CS-300-R-112		X,S		Soft	2CS2-300-SN-1	A-HCS-300-3	C	3.6A-13 3.6A-14
A-SI-84-R-2		X,S		Soft	2S14-84-SA-1	A-HSI-84-4	C	3.6A-13
A-SI-51-R-20		X,S		Hard	2S13-51SA-1	A-HSI-50-1	C	3.6A-18
A-SI-51-R-22	X,A S-E			Soft	2S13-51-SA-1	A-HSI-51-2	C	3.6A-18
A-SI-50-R-30		A		Soft	2S13-50-SA-1	A-HSE-50-4	C	3.6A-18
A-CS-300-R-51		X,E		Soft	2CS4-300-SA-1	A-HCS-300-3	C	3.6A-13 3.6A-14
A-CS-300-R-53		X,N		CR	2CS4-300-SA-1	A-HCS-300-3	C	3.6A-13
A-CS-315-R-81		X,A S-E		Soft	2CS3-315-SA-1	R-HCS-95-1	C	3.6A-13 3.6A-14

*A - Parallel to Axis of Pipe Above
B - Parallel to Axis of Pipe Below
N - North; S-South; E-East; W-West
X - Means Supported in Both Directions, Except Where Otherwise Noted
+y - Above The Restraint
-y - Below The Restraint
**CR - Crushable Material Type Restraint
***C - Circumferential Break

TABLE 3.6A-15 MAIN STEAM BREAK EXCLUSION REGION PIPE STRESSES VS. ALLOWABLE MAIN STEAM LINE NO: 2MS34-235SA-1, 5MS32-4-1

MODE POINT	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) .8(1.2S_h + S_a)	MODE POINT	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) . 8 (1.25 S_h + S_a)
1	14666	37800	1106	8263	37800
2	7017	37800	1107	8563	37800
3	7946	37800	12	9308	37800
3001	7942	37800	13	11000	37800
3002	7939	37800	14	12195	37800
3003	7934	37800	1415	11717	37800
4	12939	37800	1416	11678	37800
657	3151	37800	15	11028	37800
5	13579	37800	1600	10972	37800
663	3157	37800	16	15336	37800
6	14507	37800	17	14595	37800
669	3157	37800	1718	11288	37800
7	15678	37800	18	14225	37800
675	3160	37800	19	12597	37800
6701	8490	37800	20	11014	37800
8	16391	37800	21	13495	37800
681	3172	37800	22	16582	37800
11	8767	37800			
9	8776	37800			
10	9292	37800			
1105	8144	37800			
59	14539	37800	49	9533	37800
58	7021	37800	4705	8365	37800
57	7952	37800	4706	8490	37800
5603	7949	37800	4707	9037	37800
5602	7945	37800	47	9976	37800
5601	7941	37800	46	11000	37800
56	12925	37800	45	12052	37800
687	3151	37800	4445	11210	37800
55	13540	37800	4446	11166	37800
693	3155	37800	44	10316	37800
54	14315	37800	4300	10002	37800
699	3159	37800	43	12686	37800
53	15020	37800	42	12190	37800
705	3168	37800	4142	10117	37800
5354	8068	37800	41	11993	37800
52	15854	37800	40	11128	37800
711	3175	37800	39	10317	37800

TABLE 3.6A-15 MAIN STEAM BREAK EXCLUSION REGION PIPE STRESSES VS. ALLOWABLE MAIN STEAM LINE NO: 2MS34-235SA-1, 5MS32-4-1

MODE POINT	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) .8(1.2S_h + S_a)	MODE POINT	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) . 8 (1.25 S_h + S_a)
48	8976	37800	38	12437	37800
50	8987	37800	30	14799	37800
81	14627	37800	6097	8143	37800
80	6997	37800	69	8703	37800
79	7928	37800	68	10200	37800
7803	7925	37800	67	11710	37800
7802	7921	37800	6667	11255	37800
7801	7916	37800	6668	10972	37800
78	12943	37800	66	10745	37800
717	3157	37800	6502	10812	37800
77	13583	37800	65	14950	37800
723	3160	37800	64	14264	37800
76	14511	37800	6364	11171	37800
729	3163	37800	63	13930	37800
75	15673	37800	62	12947	37800
735	3162	37800	61	10904	37800
7576	8443	37800	60	13255	37800
74	16399	37800	33	16250	37800
741	3167	37800			
70	8569	37800			
72	8579	37800			
71	9077	37800			
6905	7948	37800			
6906	8067	37800			
28	16157	37800	3205	6141	37800
29	16653	37800	32	17452	37800
84	14617	37800	128	13673	37800

NOTE: Combined Stress + Pr + Wt + Th + OBE + (SH or RV)
Where: Pr = Pressure stress
Wt = Weight stress
Th = Maximum thermal stress
OBE = OBE Seismic stress
SH = Steam Hammer
RV = Relief valve

TABLE 3.6A-16

PIPE STRESSES VS. ALLOWABLE
PIPE RUPTURE CONDITION

MAIN STEAM PIPING

LINE NO.	NODE POINT NO.	WEIGHT	PRESSURE	OBE	REL. SEIS.	RUPTURE	COMBINED STRESS PSI	1.8 SH PSI
2MS-34-235-SA-1	1	1235	3815	458	4151	5049	11602	31500
2MS-34-235-SA-1	4	509	3815	511	148	6761	11106	31500
2MS-34-235-SA-1	5	701	3815	512	1063	8216	12817	31500
2MS-34-235-SA-1	6	818	3815	515	2210	9535	14434	31500
2MS-34-235-SA-1	7	862	3815	518	3359	10614	15816	31500
2MS-34-235-SA-1	8	806	3815	523	3867	10493	15816	31500
2MS-34-235-SA-1	12	241	3815	2485	123	11475	15798	31500
2MS-34-236-SB-1	59	1236	3815	523	4151	5175	11706	31500
2MS-34-236-SB-1	56	509	3815	510	148	8326	12661	31500
2MS-34-236-SB-1	55	701	3815	511	1063	9881	14467	31500
2MS-34-236-SB-1	54	819	3815	514	2210	11113	15976	31500
2MS-34-236-SB-1	53	863	3815	518	3359	12126	17271	31500
2MS-34-236-SB-1	52	806	3815	524	3867	11354	15854	31500
2MS-34-236-SB-1	47	241	3815	3170	123	10942	15449	31500
2SM-34-237-SA-1	81	1236	3815	415	4151	5069	11601	31500
2MS-34-237-SA-1	78	504	3815	516	148	6761	11106	31500
2SM-34-237-SA-1	77	701	3815	518	1063	8216	12817	31500
2SM-34-237-SA-1	76	819	3815	521	2210	9535	14436	31500
2MS-34-237-SA-1	75	862	3815	524	3359	10614	15822	31500
2SM-34-237-SA-1	74	806	3815	528	3887	10493	15816	31500
2SM-34-237-SA-1	69	240	3815	1876	123	11475	15683	31500

WT = Weight stress (psi)
PRESS = Pressure stress (psi)
OBE = OBE Seismic stresses (psi)
REL. SEIS. = OBE Relative seismic stresses due to anchor displacement (psi)
RUPTURE = Maximum stresses due to postulated pipe breaks
COMBINED STRESS = WT + PRESS + $[(\text{OBE})^2 + (\text{REL. SEIS})^2 + (\text{RUPTURED})^2]^{1/2}$

TABLE 3.6A-17.1 PIPE STRESS VS. ALLOWABLE OPERATING CONDITION FEEDWATER LINE NO. 2FW16-13SN-1 AND 4FW16-12-1

NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+ Sa)	NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+ Sa)	NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+ Sa)
2710	7217	32400	2805	8833	32400	292	9858	32400
2715	7217	32400	96231	8917	32400	96222	8740	32400
2715	7217	32400	96231	8917	32400	96222	8740	32400
271	7217	32400	9623	9038	32400	96221	8772	32400
271	7217	32400	9623	9038	32400	96221	8772	32400
2711	7369	32400	281	8374	32400	9622	8348	32400
2711	7369	32400	281	8374	32400	9622	8348	32400
272	10593	32400	282	8489	32400	29201	6124	32400
272	10593	32400	282	8489	32400	29201	6124	32400
27201	11709	32400	2825	8956	32400	2920	7359	32400
27201	11709	32400	2825	8956	32400	2920	7359	32400
2721	12326	32400	2826	9211	32400	2922	9021	32400
2721	14539	32400	2826	9211	32400	2922	9021	32400
2720	13486	32400	283	9097	32400	29211	9539	32400
2720	11557	32400	283	9097	32400	29211	9539	32400
2722	9986	32400	284	8304	32400	2921	10096	32400
2722	9986	32400	284	8304	32400	2921	10096	32400
2723	9512	32400	285	8882	32400	293	11809	32400
2723	10467	32400	285	8882	32400	293	11809	32400
274	13377	32400	286	9379	32400	294	10176	32400
274	11514	32400	286	9379	32400	2942	4374	32400
2751	19287	32400	28701	10387	32400	2941	5196	32400
2751	19287	32400	28701	9302	32400	2941	5196	32400
27501	18954	32400	287	8470	32400	294	8295	32400
27501	18954	32400	287	8470	32400	294	11404	32400
2750	18689	32400	2875	8584	32400	295	13465	32400
2750	18689	32400	2875	8584	32400	295	13465	32400
27601	16823	32400	2879	9013	32400	2961	10454	32400
27601	16823	32400	2879	9013	32400	2961	10454	32400
2760	15640	32400	9882	10121	32400	2962	9655	32400
2760	15640	32400	9882	10121	32400	2962	9655	32400
27610	14072	32400	2883	8571	32400	2963	10454	32400
27610	14072	32400	2883	8571	32400	2963	10454	32400
2761	15197	32400	98910	9352	32400	2970	9655	32400
2761	15197	32400	98910	10442	32400	2970	9655	32400
27801	10719	32400	9891	9173	32400	297	10453	32400
27801	10719	32400	9891	9173	32400	2991	9152	32400
2781	8306	32400	2891	8837	32400	2991	9152	32400
2781	8306	32400	2891	8837	32400	29901	7905	32400
278	11572	32400	290	10004	32400	29901	7905	32400
278	11572	32400	291	7675	32400	2992	15575	32400
279	9952	32400	2911	4374	32400	2992	15575	32400
280	8802	32400	291	4502	32400	2993	18291	32400
280	8802	32400	291	7646	32400	2993	2316	32400
2805	8833	32400	292	9858	32400	21 BEG	1197	32400

TABLE 3.6A-17.2

PIPE STRESS VS. ALLOWABLE OPERATING CONDITION FEEDWATER

LINE NO. 2FW16-17SN-1 AND 4FW16-14-1

NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+Sa)	NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+Sa)	NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+Sa)
4420	12939	33000	451	8834	33000	4794	7745	33000
442	12939	33000	451	8834	33000	4619	7069	33000
442	12939	33000	452	8949	33000	4619	7069	33000
1442	12939	33000	452	8949	33000	9602	6744	33000
1442	12939	33000	4525	8997	33000	9602	6744	33000
2442	12939	33000	4525	8997	33000	4620	8046	33000
2442	12939	33000	4526	9148	33000	4620	8046	33000
1244	8237	33000	4526	9148	33000	9601	6911	33000
1244	8237	33000	453	9201	33000	9601	6911	33000
2244	9223	33000	453	9201	33000	4628	7440	33000
2244	9223	33000	454	8540	33000	4628	7440	33000
444	10177	33000	454	8540	33000	4618	8084	33000
444	10177	33000	455	8893	33000	4618	8084	33000
4444	10036	33000	455	8893	33000	463	8696	33000
4444	11269	33000	456	9122	33000	463	8696	33000
445	11791	33000	456	9122	33000	464	7965	33000
445	10395	33000	457	9004	33000	464	7598	33000
1044	10887	33000	457	9004	33000	4641	5016	33000
1044	10887	33000	4575	9082	33000	4641	5016	33000
4451	12879	33000	4575	9082	33000	4642	4374	33000
4451	12879	33000	1459	9291	33000	464	9126	33000
4450	13013	33000	1459	8305	33000	465	10946	33000
4450	13013	33000	4590	8440	33000	465	10946	33000
446	11694	33000	4590	8440	33000	466	8986	33000
446	11694	33000	4585	9132	33000	466	8986	33000
4460	11230	33000	4585	9132	33000	4661	8292	33000
4460	11230	33000	9975	10022	33000	4661	8292	33000
4461	10271	33000	9975	10022	33000	4662	8290	33000
4461	10271	33000	4574	9346	33000	4662	8290	33000
1446	8556	33000	4574	10437	33000	3663	8287	33000
1446	8556	33000	9591	9584	33000	3663	8287	33000
9603	9832	33000	9591	9584	33000	467	8285	33000
9603	9832	33000	4591	9031	33000	467	8285	33000
448	9775	33000	4591	9031	33000	4691	8056	33000
448	9775	33000	5460	9124	33000	4691	8056	33000
449	8982	33000	5460	9124	33000	1469	7375	33000
449	8982	33000	460	10774	33000	1469	7375	33000
1044	8936	33000	460	10774	33000	4692	8809	33000
1044	8936	33000	461	10501	33000	4693	9809	33000
450	8455	33000	4611	7217	33000	4693	1025	33000
450	8455	33000	461	7630	33000	24	868	33000
4505	7359	33000	462	9574	33000			
9604	7217	33000	462	9574	33000			
4505	9000	33000	4794	7745	33000			

TABLE 3.6A-17.3

PIPE STRESS VS. ALLOWABLE OPERATING CONDITION FEEDWATER

LINE NO. 2FW16-19SN-1 AND 4FW16-16-1

NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+Sa)	NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+Sa)	NODE	COMBINED STRESS (PSI)	ALLOWABLE STRESS (PSI) 0.8 (1.2Sh+Sa)
367	10199	33000	378	9167	33000	386	4449	33000
3680	9675	33000	3780	8468	33000	3861	4374	33000
3680	9645	33000	3780	8468	33000	386	6724	33000
3675	15685	33000	3785	9897	33000	387	8144	33000
3675	15685	33000	3785	9897	33000	387	8144	33000
368	15822	33000	379	9156	33000	8899	6953	33000
368	15822	33000	379	9156	33000	8899	6953	33000
9368	15822	33000	380	8680	33000	8877	7102	33000
9368	15822	33000	380	8680	33000	8877	7102	33000
3685	8767	33000	9502	8227	33000	3877	6865	33000
3685	8767	33000	9502	8227	33000	3877	6865	33000
369	9583	33000	3800	8077	33000	9612	6918	33000
369	10580	33000	3800	8077	33000	9612	6918	33000
3691	10266	33000	3805	8391	33000	3870	6625	33000
3691	9344	33000	3805	8391	33000	3870	6625	33000
3692	8725	33000	3806	8571	33000	9611	6788	33000
3692	8725	33000	3806	8571	33000	9611	6788	33000
3693	8534	33000	3381	8471	33000	3872	7382	33000
3693	9073	33000	3381	8471	33000	3872	7382	33000
370	10843	33000	3382	8173	33000	3875	7869	33000
370	9760	33000	3382	8173	33000	3875	7869	33000
3705	11310	33000	3383	8326	33000	3871	8524	33000
3705	11310	33000	3383	8326	33000	3871	8524	33000
371	14811	33000	3384	8469	33000	388	9959	33000
371	14811	33000	3384	8469	33000	388	9959	33000
3711	14927	33000	382	8546	33000	389	9110	33000
3711	14927	33000	382	8546	33000	389	8329	33000
3715	15067	33000	3825	8640	33000	3888	4993	33000
3715	15067	33000	3825	7713	33000	3888	4993	33000
3718	14830	33000	3829	7727	33000	3887	4374	33000
3718	14830	33000	3829	7727	33000	389	10477	33000
372	13319	33000	3840	7835	33000	390	12383	33000
372	13319	33000	3840	7835	33000	390	12383	33000
3720	12619	33000	3839	8111	33000	392	9811	33000
3720	12619	33000	3839	8111	33000	392	9811	33000
3725	11886	33000	3826	8289	33000	393	9811	33000
3725	11886	33000	3826	8289	33000	393	9811	33000
3721	10612	33000	3925	8349	33000	3931	9811	33000
3721	10612	33000	3925	8349	33000	3931	9811	33000
3722	8310	33000	3824	8563	33000	3950	9811	33000
3722	8310	33000	3824	8563	33000	3950	9811	33000
374	10839	33000	9825	9073	33000	3932	9811	33000
374	10839	33000	9825	9073	33000	3932	9811	33000
3740	10871	33000	3823	7907	33000	3991	10907	33000
3740	10871	33000	3823	7907	33000	3991	10907	33000
9613	10906	33000	9800	8624	33000	8888	8040	33000
9613	10906	33000	9800	9642	33000	8888	8040	33000
375	10480	33000	9841	8622	33000	3992	9507	33000
375	10480	33000	9841	8622	33000	3992	9507	33000
376	9504	33000	3841	8612	33000	33993	11178	33000
376	9504	33000	3841	8612	33000	33993	4380	33000
3760	9462	33000	5385	8395	33000	3993	4374	33000
3760	9462	33000	5385	8395	33000	33993	1213	33000
377	9542	33000	385	9390	33000	17	959	33000
377	9542	33000	385	6548	33000			
378	9167	33000	386	6748	33000			

TABLE 3.6A-18

PIPE STRESSES VS. ALLOWABLES
PIPE RUPTURE CONDITION
FEEDWATER PIPING
LOOP #1

LINE NO.	NODE POINT NO.	WEIGHT	PRESSURE	OBE	REL. SEIS.	RUPTURE	COMBINED STRESS PSI	1.8 SH PSI
2FW16-13SN-1	290	438	7217	1514	106	16810	24533	27000
2FW16-13SN-1	292	1429	7217	3170	249	17210	26147	27000
2FW16-13SN-1	2929	1124	7217	2718	367	11160	19833	27000
2FW16-13SN-1	2920	657	7217	1787	345	9547	17593	27000
2FW16-13SN-1	2921	1798	7217	3386	262	8300	17983	27000
2FW16-13SN-1	293	845	7217	5575	438	12700	21939	27000
2FW16-13SN-1	295	2614	7217	5154	646	14400	25139	27000
2FW16-13SN-1	2961	1998	7217	3038	523	16221	25726	27000
2FW16-13SN-1	2992	2909	7217	5357	1918	17060	28110	27000
2FW16-19SN-1	385	649	7217	843	126	13090	20984	27000
2FW16-19SN-1	387	1391	7217	1746	233	10700	19452	27000
2FW16-19SN-1	3877	684	7217	1318	430	8280	16296	27000
2FW16-19SN-1	3870	809	7217	716	640	9030	17107	27000
2FW16-19SN-1	3871	1868	7217	1636	498	9000	18246	27000
2FW16-19SN-1	388	894	4375	3415	902	10890	16717	27000
2FW16-19SN-1	390	2551	4375	4062	883	14900	22395	27000
2FW16-19SN-1	392	1954	4375	2475	611	19970	26461	27000
2FW16-19SN-1	393	1954	4375	2475	611	23450	29917	27000
2FW16-17SN-1	460	483	7217	1271	153	14920	22675	27000
2FW16-17SN-1	462	1304	7217	2198	286	14620	23308	27000
2FW16-17SN-1	4619	629	7217	1194	472	9750	17680	27000
2FW16-17SN-1	4620	988	7217	1078	1215	9700	18040	27000
2FW16-17SN-1	4618	1807	7217	1163	418	12120	21207	27000
2FW16-17SN-1	463	719	7217	2231	604	12800	20943	27000
2FW16-17SN-1	465	2427	4375	2582	498	16810	23816	27000
2FW16-17SN-1	466	1852	4375	1591	450	19460	25757	27000
2FW16-17SN-1	4662	1681	4375	1439	308	24020	30121	27000

WT = Weight stress (psi)
PRESS = Pressure stress (psi)
OBE = OBE Seismic stresses (psi)
REL. SEIS. = OBE Relative seismic stresses due to anchor displacement (psi)
RUPTURE = Maximum stresses due to postulated pipe breaks
COMBINED STRESS = WT + PRESS + $[(OBE)^2 + (REL.\ SEIS)^2 + (RUPTURED)^2]^{1/2}$

TABLE 3.6A-19

SUMMARY OF DESIGN BASIS BREAKS

MAIN STEAM AND FEEDWATER PIPING - INSIDE CONTAINMENT
LOOP #1

MAIN STEAM:

LINE NO.	BREAK NO.	NODE	LOCATION TYPE	REASON FOR SELECTION	FIGURE NO.
2MS-32-1SA-1	R-HMS-1-1	1	Anchor at Nozzle	Terminal Point	3.6A-29
2MS-32-1SA-1	R-HMS-1-5	23	Anchor at Containment	Terminal Point	3.6A-29

FEEDWATER:

2FW-16-67SN-1	R-HFW-67-1	301	Anchor at Nozzle	Terminal Point	3.6A-29
2FW-16-67SN-1	R-HFW-67-5	322	Anchor at Containment	Terminal Point	3.6A-29

TABLE 3.6A-20

SUMMARY OF DESIGN BASIS BREAKS

MAIN STEAM AND FEEDWATER PIPING - INSIDE CONTAINMENT
LOOP #2

MAIN STEAM:

LINE NO.	BREAK NO.	NODE	LOCATION TYPE	REASON FOR SELECTION	FIGURE NO.
2MS-32-2SB	R-HMS-2-1	1	Anchor at Nozzle	Terminal Point	3.6A-30
2MS-32-2SB	R-HMS-2-5	13	Anchor at Containment	Terminal Point	3.6A-30

FEEDWATER:

2FW-16-68-SN	R-HMS-68-1	301	Anchor at Nozzle	Terminal Point	3.6A-30
2FW-16-68-SN	R-HMS-68-5	318	Anchor at Containment	Terminal Point	3.6A-30
2FW-16-68-SN	R-HFW-68-6	313	SR Elbow	High Relative Intermediate Stress Point	3.6A-30
2FW-16-68-SN	R-HFW-68-7	3133	ButtWeld	High Relative Intermediate Stress Point	3.6A-30

TABLE 3.6A-21

SUMMARY OF DESIGN BASIS BREAKS

MAIN STEAM AND FEEDWATER PIPING - INSIDE CONTAINMENT LOOP #3

MAIN STEAM:

LINE NO.	BREAK NO.	NODE	LOCATION TYPE	REASON FOR SELECTION	FIGURE NO.
2MS-32-3SA	R-HMS-3-1	1	Anchor at Nozzle	Terminal Point	3.6A-31
2MS-32-3SA	R-HMS-3-5	23	Anchor at Containment	Terminal Point	3.6A-31

FEEDWATER:

2FW-16-69-SN	R-HFW-69-1	301	Anchor at Nozzle	Terminal Point	3.6A-31
2FW-16-69-SN	R-HFW-69-5	325	Anchor at Nozzle	Terminal Point	3.6A-31

TABLE 3.7.1-1

DAMPING FACTORS USED IN SEISMIC ANALYSIS

	Percent of Critical Damping	
	Operating Basis Earthquake (E) 0.075 g ground acceleration	Safe Shutdown Earthquake (E') 0.15 g ground acceleration
Equipment and large-diameter piping system, pipe diameter greater than 12 in.	2	3*
Small-diameter piping systems, pipe diameter equal to or less than 12 in.	1	2
Welded steel plate assemblies and welded steel frame structures	2	4
Bolted or riveted steel frame structures	4	7
Reinforced concrete frames, buildings, and containment and internal structure	4	7
Rock	2	5

*For the reactor coolant loop, a damping factor of four is used as discussed in Section 3.7.1.3 and Table 3.7.2-16.

TABLE 3.7.1-2

FOUNDATION DESIGN PARAMETER

Foundation Material	Compressional Wave Velocity (ft. per sec.)		Density (lbs. per cu ft.)	Poisson's Ratio	Young's Modulus or Modulus of Deformation (lbs. per sq. in.)	
	Field	Lab			Static	Dynamic
Residual Soil	1200-2000	-	130	0.44	-	3×10^3
Weathered and Fractured Rock	5000-7150	4500	160	0.37	-	2×10^5
Sound Bedrock	10,000-13,000	7000-10,000	160	0.35	2×10^6	2×10^6

TABLE 3.7.2-1

STABILITY ANALYSIS - OVERTURNING MOMENTS EVALUATION

	Method of Analysis			
Seismic Category I Structure	Three Statistically Independent Excitations	Modified	Seismic Category I Structure	Three Statistically Independent Excitations
Containment	X	X	X	2D
Reactor Auxiliary Building-1	Not Applicable	X		2D
Fuel Handling Building	Not Applicable	X		2D
Waste Processing Building	Not Applicable	X		2D
Condensate Storage Tank Building	Not Applicable	X		3D
Diesel Generator Building	Not Applicable	X		2D
Diesel Fuel Oil Storage Tank	Not Applicable	Not Applicable	Not Applicable	2D
ESWS Intake Structure	X	X		3D
ESWS Discharge Structure	Not Applicable	X		2D
ESWS Intake and Screening Structure	Not Applicable	X		2D

TABLE 3.7.2-2

CONTAINMENT BUILDING-NATURAL FREQUENCIES, EIGENVALUES, AND PARTICIPATION FACTORS

	NORTH-SOUTH (SSE AND OBE)			EAST-WEST (SSE AND OBE)			VERTICAL (SSE AND OBE)		
MODE	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR
1	4.083	658	40.9868	4.084	658	40.8377	11.411	5140	54.0378
2	8.781	3044	36.1767	9.853	3833	40.1458	15.603	9612	0.4118
3	12.792	6460	19.8504	12.944	6615	14.3868	19.738	15381	36.7004
4	20.157	16040	-28.4568	20.718	16946	-25.4882	24.561	23815	2.5608
5	24.132	22990	12.6458	24.756	24194	16.8659	24.933	24543	-4.9561
6	29.673	34759	2.3360	29.882	35251	-1.6055	26.948	28670	-1.2271
7	31.781	39873	11.1870	35.131	48724	-7.8430	36.745	53302	9.9025
8	46.099	83895	-2.0337	46.211	84305	-2.4989	45.546	81895	8.0874
9	50.073	98985	-2.4035	52.543	108988	1.1764	59.914	141714	-1.5240
10	52.836	110207	-0.1013	53.037	111049	-0.9990	78.030	240368	-1.7930
11	55.880	123273	0.8026	58.370	134506	-0.3119	100.721	400494	-0.6167
12	58.750	136261	-0.3268	65.421	168964	-0.5230	116.704	537689	-0.5264
13	69.589	191181	-0.6532	69.828	192495	-0.7867	133.884	707645	0.0939
14	78.135	241019	0.3122	90.757	325176	0.1705	150.823	898042	-0.2301
15	97.285	373642	-0.0019	97.285	373641	-0.0016	180.300	1282360	0.0450
16	98.436	382530	-0.0403	98.436	382534	-0.0417	213.398	1797790	-0.00135

TABLE 3.7.2-3

REACTOR AUXILIARY BUILDING - 1
NATURAL FREQUENCIES, EIGENVALUES, AND PARTICIPATION FACTORS

	NORTH-SOUTH (SSE AND OBE)			EAST-WEST (SSE AND OBE)			VERTICAL (SSE AND OBE)		
MODE	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR
1	7.577	2,266	53.4510	5.800	1,328	51.2133	11.839	5533	71.5054
2	16.215	10,380	26.3272	13.655	7,361	-32.7981	14.207	7969	-1.0759
3	20.550	16,671	31.9091	18.078	12,901	24.3123	17.139	11596	-3.8889
4	29.189	33,635	-23.5734	26.630	27,997	23.3123	17.455	12029	-6.1171
5	35.486	49,712	-9.0755	33.021	43,046	19.5382	17.834	12556	-4.7728
6	38.626	58,900	8.3289	35.592	50,012	2.5608	21.037	17471	-4.7878
7	41.972	69,547	11.8641	37.492	55,493	-3.0125	28.021	30998	4.7313
8	44.185	77,074	13.2569	45.045	80,104	19.9147	30.520	36774	18.2629
9	52.109	107,198	-5.9954	52.358	108,225	0.3369	54.806	118582	-7.7226
10	66.485	174,504	-0.3501	62.723	155,315	-0.3206	72.083	205127	5.3035
11	83.957	278,275	0.0588	81.261	260,687	0.1070	87.629	303148	-1.3855
12	99.078	387,540	0.0310	93.538	345,408	0.0262	101.551	407122	-0.4040
13	104.475	430,930	-0.0184	98.794	385,318	0.0233	105.003	435276	-2.7689
14	107.583	456,930	0.0435	128.788	654,806	0.0835			
15	549.153	11,905,500	0.6436	451.198	8,037,020	0.9412			
16	660.881	17,242,700	0.4735	470.640	8,744,560	0.9457			
17	13,567,300	7,266,860,000	-0.000000058	6264.370	1,549,230,000	-0.0000012			
18	17,212,900	11,696,900,000	0.0000000013	9732.000	3,739,070,000	0.00000000032			

TABLE 3.7.2-4

REACTOR AUXILIARY BUILDING - COMMON
NATURAL FREQUENCIES, EIGENVALUES, AND PARTICIPATION FACTORS

	NORTH-SOUTH (SSE AND OBE)			EAST-WEST (SSE AND OBE)			VERTICAL (SSE AND OBE)		
MODE	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR
1	8.029	2545	50.7609	7.605	2,283	49.7179	14.355	8135	62.0463
2	19.664	15,265	35.0884	19.092	14,390	36.1529	20.272	16224	-2.0336
3	26.679	28,100	23.0042	26.551	27,830	23.2826	21.960	19037	-4.1024
4	35.753	50,463	3.6867	34.522	47,049	5.5674	22.272	19582	-4.2849
5	40.989	66,329	-1.1672	38.110	57,337	0.0049	24.479	23656	-4.9605
6	50.176	99,390	-0.1030	48.369	92,362	-0.1959	32.189	40905	21.2191
							54.878	117597	-3.7974
							77.252	235605	0.7718
							109.451	472934	-0.0258

TABLE 3.7.2-5

FUEL HANDLING BUILDING NATURAL FREQUENCIES, EIGENVALUES, AND PARTICIPATION FACTORS

	NORTH-SOUTH (SSE AND OBE)			EAST-WEST (SSE AND OBE)			VERTICAL (SSE AND OBE)		
MODE	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR
1	9.2635	3,388	104.539	6.43916	1,637.	88.8958	11.4713	5195.	118.12
2	18.8263	13,992	-46.2726	10.1432	4,062.	17.8522	15.4129	9378.	-3.4714
3	30.2536	36,134	34.0476	11.0706	4,838.	39.5877	20.0326	15843.	-0.5289
4	39.5612	61,787	0.4670	13.7359	7,449.	49.6235	25.8449	26370.	-0.3396
5	45.0815	80,234	12.5362	16.0477	10,167.	0.0230	30.1831	35966.	-19.1977
6	50.5371	100,828	7.0862	18.4954	13,505.	-7.6642	49.9724	98587.	9.3590
7	55.8847	123,295	1.2763	23.1051	21,075.	41.3221	80.6494	256781.	-3.5720
8	80.0782	253,156	0.000747	28.8824	32,933.	8.0182	93.4117	344479.	2.0440
9	144.8790	828,647	0.0000923	30.4568	36,621.	8.6460	102.8540	417637.	0.4763
10	1702.4100	114,417,000	-0.001797	32.1129	40,712.	-7.0264	148.4290	869751.	-0.0009495
11	1726.6300	117,695,000	0.19135	34.7523	47,679.	2.3222	268.4150	2,844,290.	0.00000826
12	3283.8600	425,724,000	-0.01613	35.3185	49,245.	-6.5805	3162.73	394,896,000.	-0.0001186
13				40.7284	65,487.	10.4229	3199.50	404,134,000.	-0.00000123
14				44.0595	76,637.	6.1062	6216.18	1,525,480,000.	-0.0000102
15				48.7658	93,884.	0.03055			
16				48.9772	94,699.	-0.26415			
17				49.4139	96,396.	-4.8907			
18				75.3655	224,236.	-0.12234			
19				91.7368	332,236.	0.0001108			
20				925.8590	33,841,500.	-0.00376			
21				972.9350	37,370,400.	0.0000000876			
22				1120.2000	49,539,400.	0.000000742			
23				1562.3500	96,363,800.	0.21178			

TABLE 3.7.2-6

WASTE PROCESSING BUILDING - NATURAL FREQUENCIES, EIGENVALUES, AND PARTICIPATION FACTORS

	NORTH-SOUTH (SSE AND OBE)			EAST-WEST (SSE AND OBE)			VERTICAL (SSE AND OBE)		
MODE	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR	FREQUENCY	EIGEN VALUE	PARTICIPATION FACTOR
1	7.316	2,113	100.00	7.391	2157	95.5815	11.439	5166	117.089
2	15.123	9,029	60.6718	15.450	9423	61.5651	16.230	10399	-3.926
3	22.861	20,633	26.0213	22.426	19854	36.8996	23.668	22115	-1.662
4	31.986	40,392	-5.1869	26.544	27816	-0.09501	26.750	28250	-25.352
5	33.931	45,451	-2.7116	30.337	36333	-9.7439	44.750	79059	-6.705
6	40.258	63,981	1.1551	37.238	54743	3.4782	65.150	167568	-1.500
7	41.302	67,344	-1.5425	49.093	95149	-0.007224	82.559	269082	0.530
8	69.154	188,799	-0.01524	61.883	151182	-0.02307	91.629	331452	0.1073
9							96.174	365151	-0.00812
10							136.496	735524	-0.00456

TABLE 3.7.2-7

CONTAINMENT BUILDING-MAXIMUM STRUCTURAL RESPONSES
EAST-WEST DIRECTION

	SSE				OBE			
	(ft)	(g)	(kip)	(kip ft)				
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.032207	0.712797	2369.	0.	0.019810	0.451369	1500.	0.
2	0.031195	0.678370	4025.	18955.	0.019181	0.429071	2548.	12003.
3	0.028168	0.566653	6030.	107510.	0.017197	0.365792	3808.	68053.
4	0.024994	0.460116	10362.	234141.	0.015314	0.315666	6758.	148018.
5	0.017808	0.372348	14825.	741885.	0.010827	0.243449	9646.	473878.
6	0.010246	0.317865	17605.	1419006.	0.006288	0.218181	11119.	940970.
7	0.007606	0.290936	18685.	1696939.	0.004662	0.194257	11931.	1125902.
8	0.006007	0.267188	19775.	1902470.	0.003666	0.173981	12733.	1245118.
9	0.003279	0.212583	20803.	2297974.	0.001944	0.128290	13381.	1466229.
10	0.005694	0.520765	2400.	0.	0.003221	0.293865	1354.	0.
11	0.005015	0.466928	6940.	39955.	0.002834	0.262290	3905.	22546.
12	0.003840	0.364779	10581.	213457.	0.002167	0.202240	5923.	120162.
13	0.003167	0.304574	12549.	340424.	0.001787	0.165893	7000.	191238.
14	0.002409	0.239416	14432.	503565.	0.001361	0.128053	8008.	282244.
15	0.001443	0.184793			0.000813	0.099443		
16	0.000020	0.000681	39907.*	3552413.**	0.000012	0.000398	22375.*	2073286.**

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-7 (continued)

NORTH-SOUTH DIRECTION

	SSE				OBE			
	(ft)	(q)	(kip)	(kip ft)				
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.032661	0.722956	2403.	0.	0.020038	0.496667	1651.	0.
2	0.031617	0.688080	4083.	19225.	0.019399	0.471134	2801.	13207.
3	0.028481	0.575207	6118.	109044.	0.017486	0.387805	4173.	74828.
4	0.025186	0.488902	10520.	237518.	0.015469	0.322647	7057.	162462.
5	0.017771	0.377166	15439.	753017.	0.010899	0.247558	9908.	508242.
6	0.010060	0.324227	17848.	1471935.	0.006269	0.216869	11466.	954052.
7	0.007417	0.296949	18758.	1784278.	0.004623	0.191146	12112.	1138201.
8	0.005824	0.272766	19668.	1990619.	0.003624	0.170334	12885.	1257388.
9	0.003115	0.216655	20554.	2383989.	0.001906	0.124867	13516.	1498405.
10	0.006758	0.571935	2635.	0.	0.004606	0.355828	1640.	0.
11	0.005879	0.501668	7514.	43881.	0.003982	0.307565	4630.	27300.
12	0.004242	0.363425	11141.	231723.	0.002821	0.212763	6754.	143061.
13	0.003261	0.284379	12988.	365411.	0.002148	0.159946	7789.	224106.
14	0.002305	0.224208	14678.	534252.	0.001477	0.123243	8674.	325358.
15	0.001313	0.171857			0.000784	0.091144		
16	0.000019	0.000569	36000.*	3427676.	0.000012	0.000356	21391.**	2058984.**

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-7 (continued)

VERTICAL DIRECTION

	SSE		OBE	
Mass No.	Displacements (ft)	Accelerations (g)	Displacements (ft)	Accelerations (g)
1	0.003227	0.495946	0.001860	0.289468
2	0.002771	0.427925	0.001597	0.248296
3	0.002020	0.323349	0.001163	0.185986
4	0.001694	0.282065	0.000975	0.161518
5	0.001476	0.254907	0.000849	0.145392
6	0.001056	0.211166	0.000606	0.120350
7	0.004207	0.684343	0.002456	0.405967
8	0.001014	0.262754	0.000592	0.146855
9	0.001000	0.258435	0.000583	0.144085
10	0.000947	0.242088	0.000550	0.133827
11	0.000899	0.228337	0.000521	0.125477
12	0.000837	0.211476	0.000483	0.115562
13	0.001404	0.409741	0.000794	0.245092
14	0.001459	0.446301	0.000853	0.276223
15	0.002114	0.410040	0.001182	0.244569
16	0.000758	0.191594	0.000435	0.106381

TABLE 3.7.2-8

REACTOR AUXILIARY BUILDING - 1 - MAXIMUM STRUCTURAL RESPONSES
NORTH-SOUTH DIRECTION

	OBE				SSE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.005319	0.369488	1742.	1894.	0.008349	0.577568	2723.	2837.
2	0.004770	0.341262	9796.	180562.	0.007483	0.535359	15358	279220.
3	0.003838	0.280259	15599.	455248.	0.006041	0.444554	24562.	731617.
4	0.002529	0.200714	20985.	918108.	0.004006	0.324617	33325.	1461629.
5	0.001077	0.120855	25621.	1480309.	0.001727	0.210313	41079.	2340171.
6	0.000670	0.101523	7672.	1657412.	0.001076	0.185475	12585.	2620916.
7	0.000499	0.094420	10260.	1813779.	0.000801	0.174570	17477.	2866429.
8	0.000277	0.084369	3435.	1878413.	0.000448	0.164081	6926.	2953348.
9	0.000120	0.080310			0.000205	0.159957		
10	0.000004	0.000378	3834.*	1956111.**	0.000006	0.000530	8151.*	3065550.**

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-8 (continued)

EAST-WEST DIRECTION

	OBE				SSE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.007824	0.323245	1524.	10961.	0.013083	0.538722	2540.	16841.
2	0.006973	0.283754	8221.	159183.	0.011671	0.474949	13749.	246622.
3	0.005471	0.226417	12758.	402673.	0.009174	0.370472	21420.	660416.
4	0.003540	0.161675	17020.	836936.	0.005960	0.291357	28225.	1382102.
5	0.001572	0.112226	20708.	1388491.	0.002670	0.211505	35444.	2305821.
6	0.000704	0.091779	2889.	1586161.	0.001195	0.176197	5735.	2645733.
7	0.000433	0.088453	4887.	1655736.	0.000765	0.172946	10512.	2761313.
8	0.000168	0.080881	2368.	1680337.	0.000317	0.160047	5686.	2808856.
9	0.000066	0.078151			0.000129	0.156131		
10	0.000005	0.000241	3357.*	1636830.**	0.000009	0.000380	7736.*	2723366.**

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-8 (continued)

VERTICAL DIRECTION

	OBE		SSE	
Mass No.	Displacements	Accelerations	Displacements	Accelerations
1	0.001488	0.240256	0.002448	0.395598
2	0.001461	0.236527	0.002404	0.389896
3	0.001351	0.222212	0.002225	0.367148
4	0.001182	0.200176	0.001950	0.332263
5	0.000963	0.171518	0.001591	0.287197
6	0.000774	0.146877	0.001279	0.248743
7	0.002113	0.357283	0.003474	0.591035
8	0.002744	0.480563	0.004419	0.766330
9	0.002564	0.438854	0.004154	0.709510
10	0.002143	0.373392	0.003458	0.593792
11	0.002689	0.445696	0.003949	0.632330
12	0.000932	0.198295	0.001543	0.296819
13	0.000557	0.119862	0.000922	0.204986

TABLE 3.7.2-9

REACTOR AUXILIARY BUILDING - COMMON - MAXIMUM STRUCTURAL RESPONSES

NORTH-SOUTH DIRECTION

	SSE				OBE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.008785	0.617023	7294.	0.	0.005606	0.388436	4592.	0.
2	0.007644	0.531131	13367.	138583.	0.004870	0.337692	8467.	87242.
3	0.006070	0.432619	22028.	392553.	0.003854	0.275215	13977.	247750.
4	0.003608	0.313684	28965.	939271.	0.002273	0.190481	18362.	597169.
5	0.000989	0.193512			0.000635	0.108233		
6	0.000010	0.000642	39326.*	2054957.	0.000006	0.000397	24897.*	1312451.**

EAST-WEST DIRECTION

	SSE				OBE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.006751	0.523377	6187.	0.	0.003962	0.304902	3604.	0.
2	0.006021	0.459591	11488.	117550.	0.003548	0.266826	6682.	68481.
3	0.004927	0.383085	19049.	335826.	0.002922	0.224132	11037.	195441.
4	0.003192	0.304721	25702.	812041.	0.001915	0.176997	15290.	470108.
5	0.000920	0.192757			0.000550	0.103197		
6	0.000004	0.00369	37802.*	1617574.**	0.000002	0.000218	22462.	943337.**

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-9 (continued)

VERTICAL DIRECTION

	SSE		OBE	
Mass No.	Displacements	Accelerations	Displacements	Accelerations
1	0.001573	0.373984	0.000941	0.226160
2	0.001496	0.356691	0.000894	0.215340
3	0.001355	0.326277	0.000808	0.196274
4	0.001087	0.273252	0.000646	0.162538
5	0.002387	0.598327	0.001462	0.381317
6	0.002429	0.560427	0.001463	0.337349
7	0.002186	0.510726	0.001316	0.309371
8	0.001987	0.456755	0.001206	0.281505
9	0.000706	0.207712	0.000418	0.119549

TABLE 3.7.2-10

FUEL HANDLING BUILDING-MAXIMUM STRUCTURAL RESPONSES

NORTH-SOUTH DIRECTION

	SSE				OBE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.005658	0.553071	13885.	0.	0.003139	0.311442	7819.	0.
2	0.005420	0.509025	22794.	166625.	0.003011	0.285452	12815.	93829.
3	0.005172	0.470333	26415.	348980.	0.002878	0.265782	14835.	196349.
4	0.004777	0.453132	34378.	639545.	0.002665	0.251232	19288.	359536.
5	0.003865	0.408754	63274.	1292724.	0.002170	0.232437	35204.	724969.
6	0.003193	0.366850	99166.	2802218.	0.001797	0.209730	55503.	1545262.
7	0.002462	0.315460	99197.	4285151.	0.001387	0.179684	55520.	2374995.
8	0.001977	0.281336	133034.	5277125.	0.001114	0.159720	74655.	2930199.
9	0.001391	0.238591	79950.	6607462.	0.000783	0.133057	44564.	3668276.
10	0.001214	0.225593	62104.	7004355.	0.000683	0.125054	34409.	3891095.
11	0.001070	0.216517			0.000602	0.118676		
12	0.000001	0.000140	67163.*	8206667.**	0.000001	0.000078	36867.*	4577709.**

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-10 (continued)

REACTOR AUXILIARY BUILDING - COMMON - MAXIMUM STRUCTURAL RESPONSES

EAST-WEST DIRECTION

	SSE				OBE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.009199	0.562832	2276.	0.	0.005679	0.377817	1410.	0.
2	0.007592	0.461563	3108.	45515.	0.004678	0.324995	1868.	28192.
3	0.004971	0.339186	9042.	138763.	0.003068	0.215043	5515.	83968.
4	0.003829	0.284147	14255.	363052.	0.002347	0.176049	9011.	218548.
5	0.003038	0.247566	14268.	573212.	0.001860	0.148878	9018.	348199.
6	0.002527	0.230324	18036.	713419.	0.001549	0.132038	11251.	435555.
7	0.013281	0.617575	10480.	0.	0.008510	0.405729	6516.	0.
8	0.012474	0.585328	19406.	125764.	0.008007	0.380960	12429.	78194.
9	0.011564	0.547029	22437.	278038	0.007433	0.354235	14391.	177272.
10	0.010165	0.489515	30853.	524843.	0.006549	0.313422	19794.	335223.
11	0.007039	0.381453	44102.	1111045.	0.004560	0.237494	27917.	711317.
12	0.004801	0.312139	58746.	2206506.	0.003054	0.187861	37509.	1409240.
13	0.003175	0.252142	58767.	3079188.	0.001997	0.149108	37523.	1964672.
14	0.002125	0.216097	75718.	3666856.	0.001316	0.123897	47385.	2335097.
15	0.001439	0.195829	25145.	4388950.	0.000885	0.107693	15579.	2800767.
16	0.010403	0.551266	3016.	0.	0.006473	0.351681	1828.	0.
17	0.008527	0.453546	4501.	60326.	0.005376	0.290011	2778.	36569.
18	0.004515	0.324350	9389.	195349.	0.002920	0.207780	6062.	119898.
19	0.003318	0.281870	15057.	429679.	0.002126	0.172809	9851.	262803.
20	0.002646	0.253870	15069.	642687.	0.001684	0.150809	9859.	408291.
21	0.002215	0.235731	19190.	784780.	0.001402	0.136632	12349.	505353.
22	0.001155	0.18845			0.000711	0.101611		
23	0.000005	0.000263	77892.*	7449002.**	0.000003	0.000154	47538.*	4708806.**

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-10 (continued)

VERTICAL DIRECTION

	SSE		OBE	
Mass No.	Displacements	Accelerations	Displacements	Accelerations
1	0.002768	0.421014	0.001610	0.244807
2	0.002719	0.415095	0.001581	0.241331
3	0.002663	0.408427	0.001548	0.237408
4	0.002573	0.397611	0.001496	0.231040
5	0.002358	0.371717	0.001370	0.215739
6	0.002191	0.351544	0.001272	0.203696
7	0.002011	0.329969	0.001167	0.190845
8	0.001891	0.315581	0.001097	0.182501
9	0.001740	0.298404	0.001010	0.171949
10	0.001664	0.289816	0.000966	0.166671
11	0.004981	0.775433	0.002989	0.473876
12	0.002691	0.414821	0.001566	0.242107
13	0.002753	0.430793	0.001599	0.251060
14	0.001589	0.281226	0.000922	0.161393

TABLE 3.7.2-11

WASTE PROCESSING BUILDING - MAXIMUM STRUCTURAL RESPONSES

NORTH-SOUTH DIRECTION

	SSE				OBE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.011500	0.872136	567.	0.	0.007333	0.534639	348.	0.
2	0.010932	0.790830	25489.	6803.	0.006961	0.485484	15647.	4170.
3	0.009177	0.562908	45127.	771476.	0.005818	0.351623	27809.	473582.
4	0.008090	0.519376	54450.	1448388.	0.005116	0.325829	33426.	890716.
5	0.006787	0.478891	80397.	2259842.	0.004277	0.297337	50948.	1392107.
6	0.004825	0.387860	113270.	4139157.	0.003034	0.239090	71034.	2539434.
7	0.002420	0.252656			0.001512	0.149718		
8	0.000008	0.000580	168351.*	8426765.**	0.000005	0.000346	104609.*	5363196.**

EAST-WEST DIRECTION

	SSE				OBE			
Mass No.	Displacements	Accelerations	Shears	Moments	Displacements	Accelerations	Shears	Moment
1	0.010637	0.810647	527.	0.	0.006745	0.509267	311.	0.
2	0.010304	0.760825	24504.	6323.	0.006541	0.477836	15390.	3972.
3	0.008842	0.556516	44263.	741430.	0.005621	0.344549	27279.	465659.
4	0.007814	0.518028	53346.	1404682.	0.004960	0.326546	32953.	874839.
5	0.006389	0.476570	79125.	2204869.	0.004046	0.298569	50045.	1363923.
6	0.004299	0.372318	110062.	4004144.	0.002715	0.229693	69476.	2479548.
7	0.001806	0.237091			0.001134	0.133328		
8	0.000004	0.000255	127159.*	8150330.**	0.000002	0.000159	78602.*	5182845.**

TABLE 3.7.2-11 (continued)

VERTICAL DIRECTION

	SSE		OBE	
Mass No.	Displacements	Accelerations	Displacements	Accelerations
1	0.003060	0.468433	0.001770	0.273839
2	0.003018	0.462498	0.001746	0.270144
3	0.002809	0.432973	0.001623	0.251609
4	0.002655	0.411613	0.001533	0.238276
5	0.002442	0.382388	0.001409	0.220154
6	0.002139	0.343307	0.001233	0.197564
7	0.005473	0.870744	0.003229	0.520604
8	0.003119	0.471665	0.001804	0.270343
9	0.002173	0.347173	0.001252	0.199893
10	0.001704	0.291935	0.000981	0.166150

* Shear at mass center of the foundation mat

** Moment at rotation center

TABLE 3.7.2-12

COMPARISON OF NATURAL FREQUENCIES AND PARTICIPATION FACTORS FOR THE CONTAINMENT BUILDING USING TWO (2) DIMENSIONAL TORSIONAL MODELS

TWO DIMENSIONAL TORSIONAL MODEL

	NORTH-SOUTH MODEL		EAST-WEST MODEL	
MODE	Frequency (cps)	Participation Factors	Frequency (cps)	Participation Factors
1	4.014	1.635880	4.014	1.631330
2	8.795	1.545989	8.996	0.000303
3	8.988	0.014794	9.937	1.629149
4	12.755	-0.775433	12.891	-0.568476
5	16.650	-0.054222	16.791	-0.002345
6	20.411	0.726631	20.929	0.729744
7	24.207	-0.338834	24.701	-0.420241
8	24.816	0.001374	24.996	-0.005374
9	30.163	-0.132010	30.371	0.035216
10	31.596	0.301862	34.741	0.086477
11	34.239	0.004389	34.890	0.195279
12	41.276	0.000001	41.416	-0.000002
13	44.282	-0.088853	44.405	-0.103127
14	48.820	-0.040841	51.317	0.000462
15	51.221	0.001035	52.373	-0.000720
16	54.013	-0.024129	58.145	-0.044756
17	57.863	-0.042085	63.818	0.001225
18	63.820	-0.001122	64.484	-0.000273
19	64.274	-0.000037	65.141	0.015431
20	66.325	-0.040646	66.809	-0.052799

NOTE:

The natural frequencies and participation factors for the two dimensional dynamic models are presented in Table 3.7.2-2.

For 3-D Torsional Model See Figure 3.7.2-9.

For 2-D Torsional Model See Figure 3.7.2-10.

For 2-D Dynamic Model See Figure 3.7.2-1.

TABLE 3.7.2-12 (continued)

COMPARISON OF NATURAL FREQUENCIES AND PARTICIPATION FACTORS FOR THE CONTAINMENT BUILDING USING THREE (3) DIMENSIONAL TORSIONAL MODELS

THREE DIMENSIONAL TORSIONAL MODELS

		PARTICIPATION FACTORS		
MODE	Frequency (cps)	N-S	E-W	VERT.
1	4.0139	1.635879	0.001568	0.000447
2	4.0144	-0.001566	1.631318	-0.003427
3	8.7951	1.545936	0.001052	-0.001621
4	8.9877	0.014756	-0.001739	-0.000021
5	9.9332	0.000135	1.628012	0.003039
6	12.7549	-0.775498	0.000883	-0.000441
7	12.8905	-0.000718	-0.569920	0.001542
8	16.6444	-0.054113	0.006511	0.000004
9	20.4096	0.726502	0.008549	0.000139
10	20.9201	-0.000794	0.729781	0.003026
11	24.2065	-0.338662	-0.002464	-0.000593
12	24.6862	0.000682	-0.396949	-0.001807
13	24.8231	0.002151	-0.027703	-0.000104
14	30.1623	-0.132295	-0.000192	0.000826
15	30.3696	-0.000043	0.034413	0.003513
16	31.5940	0.301724	0.000118	-0.001355
17	34.2303	0.004276	-0.007247	-0.000657
18	34.8270	0.003301	0.273403	0.030904
19	41.2759	0.000001	0.000014	0.000023
20	44.2819	-0.088793	0.001971	0.000316

TABLE 3.7.2-13

COMPARISON OF MAXIMUM STRUCTURAL RESPONSES* FOR CONTAINMENT BUILDING USING TWO AND THREE DIMENSIONAL MODELS

	DISPLACEMENTS							
MASS POINTS	TRANSLATION NS (X) (ft)			TRANSLATION EW (Y) (ft)			ROTATION (z) (X 10^{-5})	
	3-D** Torsional Model	2-D** Torsional Model	2-D** Dynamic Model	3-D** Torsional Model	2-D** Torsional Model	2-D** Dynamic Model	3-D** Torsional Model	2-D** Torsional Model
External Structure								
1	0.03341	0.0334	0.0327	0.00000	0.0332	0.0322	0.1314	0.1315
2	0.03233	0.0323	0.0316	0.00000	0.0321	0.0312	0.1313	0.1314
3	0.02913	0.0291	0.0285	0.00000	0.0290	0.0282	0.1308	0.1309
4	0.02623	0.0262	0.0252	0.00000	0.0261	0.0250	0.1302	0.1302
5	0.01866	0.0187	0.0178	0.00000	0.0187	0.0178	0.1169	0.1169
6	0.01073	0.0107	0.0101	0.00000	0.0108	0.0102	0.0892	0.0892
7	0.00799	0.0080	0.0074	0.00000	0.0081	0.0076	0.0804	0.0804
8	0.00634	0.0063	0.0058	0.00000	0.0065	0.0060	0.0756	0.0756
9	0.00352	0.0035	0.0031	0.00000	0.0036	0.0033	0.0668	0.0668
Internal Structure								
10	0.00683	0.0068	0.0068	0.00001	0.0054	0.0057	0.2855	0.2855
12	0.00592	0.0059	0.0059	0.00000	0.0047	0.0050	0.3415	0.3415
15	0.00424	0.0042	0.0042	0.00000	0.0036	0.0038	0.2783	0.2782
18	0.00322	0.0032	0.0032	0.00000	0.0029	0.0032	0.1956	0.1966
19	0.00220	0.0022	0.0023	0.00000	0.0022	0.0024	0.1075	0.1075
Mat								
25	0.00118	0.0012	0.0013	0.00000	0.0013	0.0014	0.0571	0.0571

* For SSE Horizontal Synthetic Excitation Based on R.G. 1.60 (Not statistically independent)

** Excitation applied in N-S direction

*** Excitation applied in E-W direction

TABLE 3.7.2-13 (continued)

ACCELERATIONS

MASS POINTS	TRANSLATION NS (X) (ft./sec.)			TRANSLATION EW (y) (ft./sec.)		ROTATION (z) ($x10^{-3}$)	
	3-D** Torsional Model	2-D** Torsional Model	2-D** Dynamic Model	2-D** Torsional Model	2-D** Dynamic Model	3-D** Torsional Model	2-D** Torsional Model
External Structure							
1	0.7170	0.7170	0.7230	0.6989	0.7128	0.1734	0.1735
2	0.6817	0.6817	0.6881	0.6640	0.6784	0.1731	0.1732
3	0.5684	0.5684	0.5752	0.5524	0.5667	0.1715	0.1716
4	0.4900	0.4900	0.4889	0.4764	0.4601	0.1693	0.1674
5	0.3960	0.3960	0.3772	0.3924	0.3723	0.1281	0.1282
6	0.3278	0.3278	0.3242	0.3197	0.3179	0.0839	0.0839
7	0.3037	0.3037	0.2970	0.2950	0.2909	0.0784	0.0785
8	0.2817	0.2817	0.2728	0.2734	0.2672	0.0810	0.0811
9	0.2294	0.2294	0.2167	0.2233	0.2126	0.0840	0.0840
Internal Structure							
10	0.5813	0.5813	0.5719	0.5338	0.5208	0.7121	0.7123
12	0.5061	0.5062	0.5017	0.4762	0.4669	0.6799	0.7001
15	0.3604	0.3605	0.3634	0.3706	0.3648	0.5601	0.5604
18	0.2861	0.2861	0.2844	0.3101	0.3046	0.4140	0.4142
19	0.2317	0.2317	0.2242	0.2412	0.2394	0.2511	0.2513
Mat							
25	0.1781	0.1781	0.1729	0.1846	0.1848	0.0817	0.0818

TABLE 3.7.2-13 (continued)

MASS POINT	SHEAR (k) N-S (x) 3-D Torsional Model** Direction x	y	E-W (y) 2-D** Torsional Model	2-D** Dynamic Model	2-D** Torsional Model	2-D** Dynamic Model
External Structure						
1	2334	1	2334	2403	2271	2369
2	3963	1	3963	4083	3857	4025
3	5925	2	5925	6118	5777	6030
4	10482	1	10482	10520	10161	10462
5	15265	2	15264	15439	15201	14825
6	17731	2	17730	17848	17782	17605
7	18717	2	18717	18758	18798	18685
8	19738	2	19738	19668	19881	19775
9	20720	2	20720	20554	20916	20803
Internal Structure						
10	2660	7	2660	2635	2453	2400
12	7497	4	7496	7514	7047	6940
15	10998	3	10998	11141	10720	10581
18	12820	4	12821	12988	12700	12549
19	14518	7	14678	14678	14625	14432
Mat						
25						

TABLE 3.7.2-13 (continued)

BENDING (x 10^3 K.FT) MOMENTS

	N-S (x)			E-W (y)			Torque (x 10^3 K.ft)
	3-D Torsional Model** Direction		2-D** Torsional Model	2-D** Dynamic Model	2-D** Torsional Model	2-D** Dynamic Model	2-D** Torsional Model
	x	y					
External Structure							
1	0.01	18.67	18.67	19.23	18.17	18.96	0.085
2	0.04	105.85	105.85	109.04	103.03	107.51	0.223
3	0.07	230.27	230.28	237.52	224.36	234.14	0.422
4	0.14	742.09	742.09	753.02	722.25	741.89	6.661
5	0.08	1446.26	1446.23	1471.94	1405.99	1419.01	15.145
6	0.07	1755.20	1755.16	1784.28	1716.63	1696.94	17.595
7	0.06	1960.02	1959.99	1990.62	1923.41	1902.47	18.301
8	0.06	2352.66	2352.62	2383.99	2321.02	2297.97	18.744
9	0.08	2736.32	2736.29	2671.40	2710.06	2589.22	18.608
Internal Structure							
10	0.12	44.29	44.29	43.88	40.84	39.96	24.133
11	0.20	231.72	231.70	231.72	217.02	213.46	19.355
12	0.19	363.70	363.67	365.41	345.44	340.42	47.464
18	0.17	530.37	530.34	534.25	510.55	503.57	47.249
19	0.15	748.14	748.13	827.81	729.92	792.21	53.840
Mat							
25							

NOTE: For 3-D Torsional Dynamic Model See Figure 3.7.2-9.

For 2-D Torsional Dynamic Model See Figure 3.7.2-10.

For 2-D Dynamic Model See Figure 3.7.2-1.

TABLE 3.7.2-14

TANK BUILDING
NATURAL FREQUENCIES, EIGENVALUES, AND PARTICIPATION FACTORS

			PARTICIPATION FACTOR (SSE AND OBE)		
MODE	FREQUENCY	EIGENVALUE	NORTH-SOUTH	EAST-WEST	VERTICAL
1	5.089	1022	1.8899	0.0048	0.0039
2	5.372	1139	-0.0059	1.8243	-0.0250
3	5.651	1261	-0.0626	0.0471	-0.0003
4	5.824	1339	0.0045	-0.1670	0.0761
5	7.757	2375	-0.0044	-0.0392	1.9077
6	8.380	2772	0.0566	0.3893	0.9201
7	9.313	3424	1.1204	-0.0277	-0.0527
8	11.323	5062	0.0132	1.3150	-0.1151
9	12.620	6288	0.7505	0.0078	0.0372
10	12.682	6350	0.0718	0.1055	0.1070
11	12.996	6667	-0.2449	-0.0241	-0.0143
12	13.551	7249	0.1520	0.0343	0.0891
13	13.562	7261	-0.0542	0.1408	0.0122
14	13.914	7643	0.0027	-0.2304	-0.0221
15	14.482	8280	0.0197	-0.0043	-0.0113
16	14.940	8811	-0.0064	0.0007	0.0023
17	15.126	9033	-0.0744	0.0173	-0.9859
18	16.771	11103	0.2836	-0.0286	-0.0231
19	18.059	12876	0.0507	0.7021	0.0294
20	20.667	16863	0.7557	-0.0655	-0.0355
21	21.859	18863	0.2641	-0.0462	1.9363
22	23.304	21440	-0.0226	0.5749	0.0612
23	23.677	22132	-0.2072	-0.1032	0.2026
24	24.283	23279	-0.2595	-0.0102	-0.0042
25	24.803	24297	0.0026	-0.1032	0.0234
26	24.944	24564	-0.0307	-0.0061	-0.0015
27	25.087	24846	0.0009	-0.1997	-0.0153
28	27.565	29998	-0.0848	-0.3994	0.0427
29	28.135	31249	-0.2187	0.1922	-0.0095
30	29.282	33850	-0.0251	0.0601	-1.3475
31	30.499	36722	-0.0094	0.0001	0.0002
32	32.198	40998	0.0007	-0.0009	-0.0001
33	32.493	41680	0.0023	-0.0007	-0.0001
34	33.987	45601	0.0086	-0.5262	0.0091
35	34.938	48188	0.3051	0.0590	-0.0015
36	37.237	54742	-0.6639	-0.0258	0.0137

TABLE 3.7.2-15

TANK BUILDING - MAXIMUM STRUCTURAL RESPONSES

EAST-WEST DIRECTION

SSE

	DISPLACEMENTS			ACCELERATIONS				
Mass No.	N-S	E-W	VERT	N-S	E-W	VERT	SHEAR*	MOMENT*
1	0.000065	0.004573	0.001733	0.007532	0.357336	0.132571	993	0.
2	0.000051	0.003022	0.001701	0.007404	0.253241	0.120138	1127	31657
3	0.000040	0.003021	0.000852	0.006427	0.253315	0.080050	1127	15829
5	0.000158	0.016997	0.003085	0.010688	0.612094	0.377125	547	0.
6	0.000094	0.009619	0.002339	0.008848	0.363254	0.201251	1229	12034
7	0.000159	0.015833	0.002519	0.011611	0.591456	0.279259	609	0.
8	0.000094	0.009466	0.002055	0.005503	0.353784	0.151019	1414	12180
9	0.000039	0.003432	0.000080	0.005999	0.493359	0.008190	187	0.
14	0.000048	0.002005	0.001621	0.009886	0.222126	0.109287	1455	42759
15	0.000041	0.002186	0.001508	0.007214	0.227609	0.099481	1708	47530
16	0.000017	0.001202	0.000072	0.004956	0.194353	0.004172	261	3740
17	0.000017	0.001201	0.000066	0.004928	0.194561	0.004067	7656	82036
21	0.000003	0.000321	0.000015	0.001159	0.158250	0.002117	1905	404060
22	0.000003	0.000310	0.000015	0.001159	0.157993	0.002117	10512	0.

OBE

	DISPLACEMENTS			ACCELERATIONS				
Mass No.	N-S	E-W	VERT	N-S	E-W	VERT	SHEAR*	MOMENT*
1	0.000053	0.002653	0.001060	0.005869	9.204630	0.082719	569	0.
2	0.000040	0.001733	0.001031	0.005086	0.142210	0.074833	639	18140
3	0.000032	0.001733	0.000560	0.004409	0.142313	0.051206	639	9070
5	0.000175	0.011769	0.001975	0.009960	0.392538	0.216907	351	0.
6	0.000099	0.006396	0.001440	0.007405	0.241503	0.131805	805	7722
7	0.000146	0.009924	0.001510	0.009287	0.354098	0.154480	365	0.
8	0.000089	0.006103	0.001272	0.004943	0.229564	0.082601	887	7300
9	0.000029	0.001949	0.000049	0.004391	0.272498	0.005009	104	0.
14	0.000039	0.001124	0.000981	0.006725	0.123224	0.068135	930	27847
15	0.000031	0.001230	0.000967	0.004936	0.127406	0.062814	1051	29475
16	0.000013	0.00649	0.000042	0.003748	0.101873	0.002811	143	2080
17	0.000013	0.000650	0.000040	0.003728	0.102103	0.002708	4440	109492
21	0.000003	0.000169	0.000010	0.000364	0.077652	0.001331	934	238252
22	0.000003	0.000163	0.000010	0.000860	0.077493	0.001331	5840	0.

*Shear & Moments are calculated for predominant direction only. (East-West)

TABLE 3.7.2-15 (Cont'd)

TANK BUILDING - MAXIMUM STRUCTURAL RESPONSES

NORTH-SOUTH DIRECTION

SSE

Mass No.	DISPLACEMENTS N-S	DISPLACEMENTS E-W	DISPLACEMENTS VERT	ACCELERATIONS N-S	ACCELERATIONS E-W	ACCELERATIONS VERT	SHEAR*	MOMENT*
1	0.008090	0.000086	0.000101	0.449143	0.009254	0.015250	1248	0.
2	0.006029	0.000040	0.000094	0.356803	0.005861	0.013983	2276	39786
3	0.005691	0.000041	0.000222	0.328043	0.004712	0.017499	687	0.
5	0.021661	0.000108	0.000375	0.694084	0.010366	0.027442	620	0.
6	0.012672	0.000057	0.000333	0.396279	0.006786	0.014685	1364	13640
7	0.020191	0.000091	0.000117	0.623927	0.006577	0.019569	643	0.
8	0.013273	0.000052	0.000079	0.425216	0.004872	0.008175	1610	12860
9	0.003699	0.000097	0.000235	0.316991	0.015233	0.018314	121	0.
14	0.004594	0.000052	0.000254	0.302878	0.010787	0.012227	1672	47740
15	0.004705	0.000024	0.000060	0.258871	0.006293	0.009242	1944	53110
16	0.001978	0.000035	0.000224	0.205117	0.008871	0.009277	199	2420
17	0.001976	0.000013	0.000024	0.204874	0.003794	0.003227	8861	217131
21	0.000337	0.000004	0.000004	0.155918	0.001241	0.000894	1876	474100
22	0.000301	0.000004	0.000004	0.155691	0.001218	0.000894	11674	0.

OBE

Mass No.	DISPLACEMENTS N-S	DISPLACEMENTS E-W	DISPLACEMENTS VERT	ACCELERATIONS N-S	ACCELERATIONS E-W	ACCELERATIONS VERT	SHEAR*	MOMENT*
1	0.004716	0.000069	0.000072	0.269124	0.007240	0.009995	748	0.
2	0.003498	0.000034	0.000068	0.201716	0.004325	0.009180	1329	23846
3	0.003347	0.000034	0.000134	0.191615	0.004086	0.009661	401	0.
5	0.013578	0.000073	0.000255	0.458411	0.007060	0.019025	409	0.
6	0.007727	0.000039	0.000222	0.240321	0.004298	0.010614	860	8998
7	0.012541	0.000076	0.000071	0.410859	0.005911	0.013150	423	0.
8	0.008042	0.000040	0.000055	0.259347	0.003241	0.006008	1013	8460
9	0.002060	0.000065	0.000136	0.180283	0.011218	0.014837	68	0.
14	0.002653	0.000084	0.000163	0.160805	0.006580	0.008883	1024	30498
15	0.002686	0.000018	0.000036	0.144818	0.004284	0.006132	1200	33785
16	0.001127	0.000022	0.000128	0.110492	0.005431	0.005619	110	1360
17	0.001126	0.000010	0.000014	0.110344	0.002870	0.001986	5185	132739
21	0.000190	0.000003	0.000003	0.073851	0.000920	0.000570	888	283104
22	0.000169	0.000003	0.000003	0.073306	0.000904	0.000570	6514	0.

*Shear & Moments are calculated for predominant direction only (North-South)

TABLE 3.7.2-15 (Cont'd)

TANK BUILDING - MAXIMUM STRUCTURAL RESPONSES

VERTICAL DIRECTION

SSE

	DISPLACEMENTS			ACCELERATIONS		
Mass No.	N-S	E-W	VERT	N-S	E-W	VERT
1	0.000123	0.000833	0.003548	0.015605	0.075405	0.339773
2	0.000084	0.000389	0.003406	0.012310	0.038287	0.327840
3	0.000075	0.000390	0.003576	0.008631	0.038517	0.338004
5	0.000165	0.000862	0.007295	0.017288	0.067380	0.616142
6	0.000097	0.000391	0.005699	0.013694	0.026389	0.451700
7	0.000126	0.001192	0.009294	0.015450	0.066736	0.709201
8	0.000080	0.000698	0.007615	0.010208	0.040428	0.549512
9	0.000070	0.000290	0.001031	0.009990	0.032049	0.284540
14	0.000071	0.000117	0.003332	0.011779	0.015025	0.320203
15	0.000054	0.000193	0.004395	0.008004	0.018712	0.371616
16	0.000029	0.000107	0.000766	0.004857	0.011027	0.170389
17	0.000029	0.000108	0.000763	0.004843	0.011362	0.170408
21	0.000005	0.000021	0.000251	0.001073	0.002723	0.137300
22	0.000005	0.000020	0.000251	0.001068	0.002890	0.137300

OBE

	DISPLACEMENTS			ACCELERATIONS		
Mass No.	N-S	E-W	VERT	N-S	E-W	VERT
1	0.000096	0.000748	0.001971	0.011205	0.065437	0.194227
2	0.000060	0.000345	0.001887	0.007572	0.030002	0.186093
3	0.000052	0.000346	0.002088	0.006066	0.030144	0.195106
5	0.000115	0.000695	0.003820	0.013198	0.052195	0.326886
6	0.000059	0.000290	0.003156	0.009314	0.021144	0.235226
7	0.000097	0.000849	0.005529	0.012011	0.056775	0.417704
8	0.000053	0.000493	0.004507	0.007279	0.029784	0.324487
9	0.000051	0.000204	0.000565	0.006977	0.024669	0.144470
14	0.000043	0.000084	0.001845	0.006988	0.009904	0.180678
15	0.000030	0.000120	0.002609	0.004470	0.011489	0.217377
16	0.000018	0.000070	0.000414	0.003046	0.007300	0.085952
17	0.000018	0.000071	0.000412	0.003034	0.007457	0.085742
21	0.00003	0.000013	0.000132	0.000709	0.001762	0.064594
22	0.000003	0.000013	0.000132	0.000715	0.001772	0.064594

TABLE 3.7.2-16

DAMPING VALUES USED FOR SEISMIC SYSTEMS ANALYSIS FOR WESTINGHOUSE SUPPLIED EQUIPMENT

Item	Damping (Percent of Critical) Upset Conditions (OBE)	Damping (Percent of Critical) 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
Reinforced Concrete Structures	4	7

*Applicable to 12 inch or larger diameter piping.

TABLE 3.7.2.-17

ACTUAL IN-PLACE CONCRETE STRENGTH

BUILDING	AVERAGE CONC. CYL. STRENGTH f'c (PSI)
REACTOR CONT. BLDG.	5627
F.H. BLDG.	5347
R.A. BLDG.	5890
R.A. BLDG.	5730
TANK BLDG.	5713
D.G. BLDG.	5739
ESW SCREEN STRUCT.	5633
ESW INTAKE STRUCT.	5633
DIESEL FUEL OIL STORAGE TANK BUILDING	5564

TABLE 3.7.2-18

FACTOR OF SAFETY-OVERTURNING

BUILDING	DBE	OBE
Fuel Handling Building	1.60	1.87
Reactor Auxiliary Building	1.22	2.17
Reactor Auxiliary Building	1.57	2.74
Tank Building	1.66	2.66
Waste Processing Building	2.73	3.50
Diesel Generator Building	4.80	more than 4.80
Screen Structure Building	1.56	1.90
Containment Building	1.66	1.99
Intake Structure Building	1.30	1.96

The above factors of safety were calculated according to FSAR Section 3.8.5.5.

All numbers shown above are taken from design calculation books of respective buildings.

TABLE 3.7.3-1

MAXIMUM STRESS COMPARISON

Sample Problem	Static Point	Static Stress (psi)	Dynamic Point	Dynamic Stress (psi)
No. 1				
Fig. 3.7.3-2	50	16047	50	8290
No. 2				
Fig. 3.7.3-3	-	-	4	7588
Fig. 3.7.3-4	28	9235	-	-
No. 3				
Fig. 3.7.3-5	2	8528	2	6039

TABLE 3.7.3-2

CALCULATION OF RESONANT FREQUENCIES AND RESPONSE OF A THREE DIMENSIONAL PIPING SYSTEM DUE TO EARTHQUAKE

TYPE OF ANALYSIS: Calculation of resonant frequencies, and response of a three dimensional piping system, due to earthquake.

PROBLEM: A three dimensional piping system which is described on the following pages:

The problem presents PIPESTRESS 2010 calculated displacements, rotations, forces, and moments. The piping system (Mathematical Model) used by PIPESTRESS 2010 is shown in Figure 3.7.3-12. PIPESTRESS 2010 was run by Ebasco engineers. Tables 3.7.3-2A, 3.7.3-2B, and 3.7.3-2C summarize various results.

INTRODUCTION: The piping system shown in Figure 3.7.3-12 consists of various sections of pipe of different cross-sections. Points 101 and 12 are rigid anchors. Point 8 is free to move in X and φX directions. Point 11 is restrained in Y direction. It is desired to determine displacements, rotations, forces, and moments resulting from an earthquake in the three spacial directions of excitation.

PROBLEM DATA: The locations and lengths of components are shown in Figure 3.7.3 12. The following table shows different sizes of pipe and the corresponding node points.

NODE POINTS From	NODE POINTS To	PIPE O.D. (in.)	PIPE WALL THK (in.)	WEIGHT Lb/Ft Length	MATERIAL
101	1				
1	2				
2	3				
4	51	10.75	0.5	65.52	SA106 GRB
51	5				
5	53				
53	54				
5	52				
52	6				
6	7				
7	8	8.625	0.406	35.52	SA106 GRB
54	9				
9	10				
10	11				
11	12				
3	4	10.75	2.0	250.19	SA106 GRB

(Valve)

Points 101 and 12 are prevented from having any displacements. Point 8 is restrained in Y and Z directions (Linear as well as rotational coordinate directions).

The restraint at 8 has linear stiffness at 1.0×10^6 and 1.0×10^7 lb/in. (in Y & Z respectively) and rotational stiffness of 1.2×10^{10} and 1.2×10^9 in-lb/rd (in Y & Z respectively).

TABLE 3.7.3-2 (Cont'd)

An earthquake is input using the floor response spectrum method. The following table shows the spectrum.

Period (Sec)	Acceleration 'g'	Period (Sec)	Acceleration 'g'	Period (Sec)	Acceleration 'g'
.033	0.8	.071	2.15	.204	1.6
.054	0.8	.083	0.85	.238	7.5
.057	0.9	.111	0.0	.239	5.8
.059	0.9	.179	1.05	.278	5.8
.065	1.25	.189	1.45	.313	1.45
.067	1.20	.196	1.60	.333	1.25

The earthquake is applied in the three spatial directions of excitation.

MATHEMATICAL MODEL:

The mathematical model is shown in Figure 3.7.3-12. PIPESTRESS 2010 uses the lumped mass method. The masses are at the nodes shown in Figure 3.7.3-12. Each node has 3 degrees of freedom, except for the fully restrained nodes (anchors).

The following table shows node point coordinates:

NODE POINT COORDINATES

Point No.	Coordinates (in ft.) X	Y	Z
1	0.0	8.75	0.0
2	-1.25	10.0	0.0
3	-10.0	10.0	0.0
4	-11.08333	10.0	0.0
5	-16.6667	10.0	0.0
6	-16.6667	19.00	0.0
7	-17.6667	20.00	0.0
8	-36.6667	20.00	0.0
9	-19.83333	10.00	0.0
10	-20.83333	10.00	1.00004
11	-20.83333	10.00	10.0
12	-20.83333	10.00	20.00
51	-15.91670	10.00	0.00
52	-16.66670	10.75	0.00
53	-17.41670	10.00	0.00
54	-18.16670	10.00	0.00
101	0.0	0.0	0.00

A complete dynamic analysis (comprising all possible translational modes of vibration - in this particular case, 45 modes of vibration since 15 mass points are used) is performed.

RESULTS: Tables 3.7.3-2A, 3.7.3-2B, 3.7.3-2C, and 3.7.3-2D list results of PIPESTRESS 2010.

TABLE 3.7.3-2A

PERIODS (SEC) OF VIBRATION

MODE	
1	.19151
2	.15200
3	.13045
4	.09661
5	.08007
6	.04457
7	.03659
8	.0322
9	.01702
10	.01431
11	.01424
12	.00804
13	.00789
14	.00755
15	.00674
16	.00586
17	.00520
18	.00497
19	.00409
20	.00402
21	.00366
22	.00302
23	.00286
24	.00274
25	.00247
26	.00226
27	.00217
28	.00215
29	.00186
30	.00185
31	.00169
32	.00159
33	.00157
34	.00142
35	.00103
36	.00103
37	.00072
38	.00068
39	.00052
40	.00045
41	.00043
42	.00038
43	.00033
44	.00025
45	.00018

TABLE 3.7.3-2B

DISPLACEMENTS

NODE POINT NUMBER	X. (in)	Y (in)	Z (in)	Øx	Øy	Øz
1	.162	.000	.102	.00162	.00017	.00230
3	.197	.259	.117	.00324	.00048	.00192
5	.197	.397	.057	.00400	.00107	.00173
9	.197	.451	.014	.00406	.00116	.00139
11	.074	.027	.000	.00185	.00100	.00053
6	.448	.397	.564	.00519	.00248	.00241
7	.464	.393	.591	.00493	.00320	.00142
8	.464	.001	.000	.00493	.00001	.00007

TABLE 3.7.3-2C

REACTIONS AT RESTRAINT NODES

NODE POINT NUMBER	F_x (lbs.)	F_y (lbs.)	F_z (lbs.)	M_x (Ft-lb)	M_y (Ft-lb)	M_z (Ft-lb)
101	2480	903	1099	11870	629	21334
12	277	1401	851	3826	2969	698
8		708	624		8210	7013
11		2681				

TABLE 3.7.3-2D

INTERNAL FORCE & MOMENTS

NODE POINT NUMBER	F_x (lbs.)	F_y (lbs.)	F_z (lbs.)	M_x (Ft-lb)	M_y (Ft-lb)	M_z (Ft-lb)
1	2285	878	779	4619	629	3715
3	1798	670	217	4344	4446	4608
5	1526	726	488	4344	3970	7156
9	172	957	744	3529	950	1554
11	165	1265	775	10183	932	698
6	1362	372	575	273	2328	4888
7	1105	366	273	0	2500	5584

TABLE 3.7.4-1

SEISMIC MONITORING SYSTEM COMPONENTS

Quantity	Type	Tag No.	Building and Elevation	Instr. Location Shown on Drawing
Triaxial Time History Accelerograph System (T/A)				
1	T/A	SE*1SM-5200A	Containment Building Elevation +221 ft MSL	CAR2166G449 (B-9)
1	T/A	SE*1SM-5200B	Containment Building Elevation +286 ft MSL	CAR2166G453 (E-6)
1	T/A	SE*1SM-5200C	DG FOST Building Elevation +242-3 ft MSL	CAR2166G413 (F-4)
Digital Recording, Playback System (D/TR), (D/PB)				
3	D/TR	SR*1SM-5200A1 SR*1SM-5200A2 SR*1SM-5200A3	Control Room RAB Elevation (Seismic Monitoring Panel) +305 ft MSL	--
1	D/PB	1SM-5202	Control Room RAB Elevation (Seismic Monitoring Panel) +305 ft MSL	--
Triaxial Peak Accelerograph System (P/A)				
1	P/A	SR*1SM-5202A	SG 1A-SN Pedastal Containment Building Elevation +238 ft MSL	G451 (H-6)
1	P/A	SR*1SM-5202B	Containment Building Reactor Piping Loop 2 (Line No. 1RC3-45SN-1)	G452 (I-9)
1	P/A	SR*1SM-5202C	RAB Building Elevation +236 ft MSL	G436503 (H-16)
Triaxial Response Spectrum Recorders (Passive, Active) and Annunciator (TR/SR, ATR/SR, and TR/SA)				
1	TR/SR	SR*1SM-5203A	SG 1B-SN Pedastal Containment Building Elevation (N/S Axis) +238 ft MSL	-G450 (E-9)
1	TR/SR	SR*1SM-5203B	SG 1B-SN Pedastal Containment Building Elevation (E/W Axis) +238 ft MSL	-G450 (F-10)
1	TR/SR	SR*1SM-5203C	SG 1B-SN Pedastal Containment Building Elevation (Vert Axis) +238 ft MSL	-G450 (F-10)
1	A TR/SR	SE*1SM-5200A	Containment Building Elevation (N/S Axis) +221 ft MSL	-G449 (B-10)
1	A TR/SR	SE*1SM-5200B	Containment Building Elevation (E/W Axis) +221 ft MSL	-G449 (B-10)
1	A TR/SR	SE*1SM-5200C	Containment Building Elevation (Vert Axis) +221 ft MSL	-G449 (B-10)
1	TR/SR	SR*1SM-5206A	RAB Elevation (N/S Axis) +216 ft MSL	-G436501 (K-10)
1	TR/SR	SR*1SM-5206B	RAB Elevation (E/W Axis) +216 ft MSL	-G436501 (K-10)
1	TR/SR	SR*1SM-5206C	RAB Elevation (Vert Axis) +216 ft MSL	-G436501 (K-10)
1	TR/SR	SR*1SM-5207A	DG FOST Building (N/S Axis) Elevation +242-3 ft MSL	-G413 (F-3)
1	TR/SR	SR*1SM-5207B	DG FOST Building (E/W Axis) Elevation +242-3 ft MSL	-G413 (F-3)
1	TR/SR	SR*1SM-5207C	DG FOST Building (Vert Axis) +242-3 ft MSL	-G413 (F-3)
1	TR/SA	SA*1SM-5205	Control Room RAB Elevation (Seismic Monitoring Panel) +325 ft MSL	--
Triaxial Seismic Switch				
1	S/S	SS*1SM-5201	Containment Building Elevation +221 ft MSL	-G449 (B-9)
Triaxial Time-History Accelerograph Control Unit				
1			Control Room RAB Elevation (Seismic Monitoring Panel) +305 ft MSL	--

TABLE 3.8.1-1

DESIGN, PROCUREMENT, FABRICATION AND ERECTION STATUS OF CONTAINMENT COMPONENTS, PARTS AND APPURTENANCES

			Procurement, Fabrication, Erection		Ebasco		Report	
		Design	Prior to 4/29/1977	After 4/29/1977	Construction Specification	Stamp Required ASME Sect III Div. 2/ACI 359	Data Report Div. 2	Stress Report Div. 1
Components	Reinforced Concrete Mat	ASME Sect III Div.2/ACI 359	NA	ASME Sect III Div 2/ACI 359	CAR-SH-CH-6	NA	Yes	
	Reinforced Concrete Wall	ASME Sect III Div.2/ACI 359	NA	ASME Sect III Div 2/ACI 359	CAR-SH-CH-6	NA	Yes	
	Reinforced Concrete Dome	ASME Sect III Div.2/ACI 359	NA	ASME Sect III Div 2/ACI 359	CAR-SH-CH-6	NA	Yes	
Parts	Steel Liner	ASME Sect III Div.2/ACI 359	*	**	CAR-SH-AS-1	***	Yes	
	Anchor Studs	ASME Sect III Div.2/ACI 359	*	**	CAR-SH-AS-1	NA	Yes	
	Crane Supports & Brackets	AISC 1970	AISC 1970	**	CAR-SH-AS-1	NA	Yes	
	Equipment Hatch	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***		Yes
	Personnel Air Lock	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***		Yes
	Emergency Air Lock	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***		Yes
	Valve Chamber	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***		Yes
	Type I Penetration Sleeves	ASME Sect III Div.1 Subsect NE	NA	ASME Sect III Subsect NE	CAR-SH-M-54	***		Yes
	Type II Penetration Sleeves	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***	Yes	
	Type III Penetration Sleeves	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***	Yes	
	Electrical Penetrations	ASME Sect III Div.1 Subsect NE	NA	ASME Sect III Div.I Subsect NE	CAR-SH-E-28	***		Yes
	Fuel Transfer Tube Penetration Sleeve	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***	Yes	Yes
	Sump Recircul. RHR Sleeve (Sleeve Nos. 47 & 48)	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***	Yes	
	Sump Recircul. Cont Spray Sleeve (Sleeve Nos. 49 & 50)	ASME Sect III Div.1 Subsect NE	*	**	CAR-SH-AS-1	***	Yes	
Attachments to Liner	Spray Piping, HVAC Pads	AISC 1970	AISC 1970	**	CAR-SH-AS-1			
	Test Channels and Angles	AISC 1970	AISC 1970	**	CAR-SH-AS-1			
Materials	Concrete	NA	ACI 318-71	ASME Sect III Div.2/ACI 359	CAR-SH-CH-6	Produced and certified in accordance with CC 2000 with exceptions listed in Appendix 3.8A		
	Reinforcing Steel	NA	ACI 318-71	ASME Sect III Div 2/ASI 359	CAR-SH-CH-7A			
Concrete Embedments	NA	Manuf.Recomm.	ASME Sect III Div 2/ACI 359	CAR-SH-AS-7 and CAR-SH-CH-16				
Measuring Devices (strain, stress, etc.)	NA	Manuf.Recomm.	Manuf.Recomm.					
Waterproofing Membrane	NA	Manuf.Recomm.	Manuf.Recomm.	CAR-SH-CH-12				
Water Stops	NA	Manuf.Recomm.	Manuf.Recomm.	CAR-SH-CH-13				
Mechanical Splices	NA	Manuf.Recomm.	ASME Sect III Div.2/ACI 359	CAR-SH-CH-15				

* Prior to April 29, 1977 for all these items the procurement, fabrication and erection were performed in accordance with ASME Code Sect III Div.1, Subsection NE, Winter 1971 Addendum.

** After April 29, 1977 for all these items the procurement, shipping, erection, shop painting, testing and inspection are performed in accordance with ASME Code Sect. III Div.2/ACI 359, Winter 75 Addendum and Associated Sections of the ASME Code Sect. III Div.1, Winter 75 Addendum.

*** No Stamp; Acceptance based on the Structural Integrity Test. Materials, fabrication and construction, testing and examination in accordance with the Engineering and Construction QA program which was approved by the NRC during the construction permit review.

For status of procurement, fabrication and erection of parts as of April 29, 1977 see Table 3.8A-1 in Appendix 3.8A.

TABLE 3.8.1-2

CONTAINMENT STRUCTURE LOAD COMBINATIONS AND LOAD FACTORS

a) Service Load Combinations

1) Test Pressure
   C = 1.0 (D + L + Pt + Tt)
2) Construction
   C = 1.0 (D + L + To + Hu)
3) Normal Operating
   C = 1.0 (D + L + To + Ro + Pv)
4) Operating Basis Earthquake
   C = 1.0 (D + L + To + Ro + E + Pv)
5) Hurricane
   C = 1.0 (D + L + To + Ro + Hu + Pv)

b) Factored Load Combinations

6) Operating Basis Earthquake
   C = 1.0D + 1.3L + 1.0(To + Ro) + 1.5E + 1.0Pv
7) Hurricane
   C = 1.0D + 1.3L + 1.0(To + Ro) + 1.5Hu + 1.0Pv
8) Safe Shutdown Earthquake
   C = 1.0(D + L + To + Ro + E' + Pv)
9) Tornado
   C = 1.0(D + L + To + Ro + W + Pv)
10) Loss of Coolant Accident
    a) C = 1.0(D + L) + 1.5P + 1.0(Ta + Ra)
    b) C = 1.0(D + L) + 1.0P + 1.0Ta + 1.25Ra
11) Loss of Coolant Accident with OBE
    C = 1.0(D + L) + 1.25P + 1.0(Ta + Ra) + 1.25E
12) Loss of Coolant Accident with Hurricane
    C = 1.0(D + L) + 1.25P + 1.0(Ta + Ra) + 1.25Hu
13) Operating Basis Earthquake, Hurricane, and Flooding
    C = 1.0(D + L + To + E + Hu + Hq)
14) Loss of Coolant Accident with SSE
    C = 1.0(D + L + P + Ta + Ra + E' + Rr)

In all combinations, the live load, L, is considered either with full value or completely absent.

In load combinations 10 through 14, the maximum values of P, Ta, Ra, and Rr, including an appropriate load factor to account for the dynamic nature of the load, are used or a time history is performed.

Load combinations 9, 10a, 10b, and 14 are first satisfied without the impulsive loads (P, Rrr, Rrj) or the impactive loads (Wm and Rrm); yield strain and displacement may be exceeded, providing that the energy absorption capability or the resistance function of the structure, limited by one-third or two-thirds of the ductility at failure, are not exceeded when considering the impulse or impact loads, respectively.

In all factored load combinations used for the analysis of the liner, all load factors are taken equal to 1.0.

TABLE 3.8.1-3

STRESS AND STRAIN ALLOWABLES FOR LINER AND LINER ANCHORS

LINER PLATE ALLOWABLES

Stress/Strain Allowables*

Load Combination	Membrane	Membrane Plus Bending
Construction	$f_{st}=f_{sc}=2/3\ f_{py}$	$f_{st}=f_{sc}=2/3\ f_{py}$
Service	$E_{st}=E_{sc}=0.002$ in/in	$E_{st}=E_{sc}=0.004$ in/in
Factored	$E_{sc}=0.005$ in/in $E_{st}=0.003$ in/in	$E_{sc}=0.014$ in/in $E_{st}=0.010$ in/in

*The types of strains limited by this table are strains induced by deformation or constraint.

LINER ANCHOR ALLOWABLES

Force/Displacement Allowables

Load Combinations in Table 3.8.1-2	Mechanical Loads**	Displacement Limited Loads***
1 through 9	Lesser of Fa= 0.67Fy Fa= 0.33Fu	$\delta a = 0.25\ \delta u$
10 through 14	Lesser of Fa= 0.9Fy Fa= 0.5Fu	$\delta a = 0.50\ \delta u$

**Mechanical loads are those which are not self-limiting or self-relieving with load application.

***Displacement limited loads are those resulting from constraint of the structure or constraint of adjacent material and are self-limiting or self-relieving.

Legend:

f_{st}, f_{sc}	=	allowable liner plate tensile or compressive stress, respectively
f_{py}	=	specified tensile yield strength of liner
E_{st}, E_{sc}	=	allowable liner plate tensile or compressive strain, respectively
Fa	=	allowable liner anchor force capacity
Fy	=	liner anchor yield force capacity
Fu	=	liner anchor ultimate force capacity
δa	=	allowable displacement for liner anchors
δu	=	ultimate displacement capacity for liner anchors

TABLE 3.8.1-4

CONTAINMENT STRUCTURE STRENGTH REDUCTION FACTORS

Item and Stress Category		Service Load Combinations Φ Note (2)	Factored Load Combinations Φ Note (2)
Concrete Compressive Stress:			
Primary Loads:	Membrane	0.30 or 0.40*	0.60
	Membrane plus bending:	0.45	0.75
Primary Plus Secondary:			
	Membrane	0.45	0.75
	Membrane plus bending (Note 1)	0.60	0.85
Concrete Tensile Stress:			
Reinforcing Steel Tensile Stress		0.50 or 0.60**	0.90****
Reinforcing Steel Compressive Stress		0.50 or 0.66**	0.90****

*Applicable only to load combinations which include either Hu or E loads.

**For load combinations in which temporary pressure loads or temperature effects loads are combined with other loads.

***The others may exceed 0.5 f_y for compatibility with the concrete but this stress will not be used for load resistance.

****The tensile strain may exceed yield when the effects of thermal gradients through the concrete section are included.

*****The strains may exceed yield when acting in conjunction with the concrete if the concrete requires strains larger than the reinforcing yield to develop its capacity.

NOTE:

(1) The maximum allowable primary-plus-secondary membrane and bending compressive stress of 0.85 f'c corresponds to a limiting strain of 0.002 in./in. as required by ASME Section III, Division 2/ ACI 359 Code.

(2) Φ is the strength reduction factor.

TABLE 3.8.1-5

SUMMARY OF ACCEPTANCE TEST RESULTS
CEMENT

Compound/Property	Range Max.	Min.	Avg.
Autoclave expansion (%)	+0.02	-0.02	0.00
Initial set (hr.)	3:54	1:53	2:43
Final set (hr.)	5:26	2:45	3:54
3-day strength (psi)	2930	1975	2453
7-day strength (psi)	4390	3290	3878
Air content of mortar (%)	9.0	7.6	8.3
Blaine (min.)	4040	3819	3864
SiO_2 (%)	22.8	21.3	21.8
Al_2O_3 (%)	4.35	3.37	4.05
Fe_2O_3 (%)	4.15	3.47	3.81
MgO (%)	2.02	1.28	1.65
SO_3 (%)	2.72	2.35	2.59
Loss on ignition (%)	1.64	0.90	1.23
Insol. residue (%)	0.48	0.27	0.38

TABLE 3.8.1-6

SUMMARY OF ACCEPTANCE TEST RESULTS
SIEVE ANALYSIS AND FINENESS MODULUS
FINE AGGREGATE (SAND)

	Cumulative Percent Passing		
Sieve Size	Coarsest	Finest	Result
3/8	100	100	100
No. 4	95	100	100
No. 8	80	100	95
No. 16	50	85	70
No. 30	25	60	37
No. 50	5	30	10
No. 100	0	10	3.3
No. 200	0	3	1.3
F.M.			2.85

TABLE 3.8.1-7

SUMMARY OF ACCEPTANCE TEST RESULTS
FINE AGGREGATE (SAND)

Property	Specification Max.	Min.	Test Result
Friable particles (%)	1.00	0.0	0.32
Lightweight particles (%)	1.00	0.0	0.02
Absorption (%)	N/A	N/A	0.56
Reduction in alkalinity mm./l	N/A	N/A	Innocuous
Dissolved silica mm./l	N/A	N/A	Innocuous
$NaSO_4$ soundness (%)	1.0	0.0	4.4

TABLE 3.8.1-8

SUMMARY OF ACCEPTANCE TEST RESULTS
SIEVE ANALYSIS
COARSE AGGREGATES
(1-1/2" GRAVEL)

		Cumulative Percent Passing
Sieve Size	Range	Result
2"	100	100
1-1/2"	90-100	96
1"	20-50	35
3/4"	0-15	6
3/8"	0-5	1.5

TABLE 3.8.1-9

SUMMARY OF ACCEPTANCE TEST RESULTS
SIEVE ANALYSIS
COARSE AGGREGATES
(3/4" GRAVEL)

		Cumulative Percent Passing
Sieve Size	Range	Result
1"	100	100
3/4"	90-100	98
3/8"	20-55	35
No. 4	0-10	8
No. 8	0-5	3.5

TABLE 3.8.1-10

SUMMARY OF ACCEPTANCE TEST RESULTS
AGGREGATE NO. 4 (1-1/2 IN. GRAVEL) AND NO. 67 (3/4 IN. GRAVEL)[1]

	Specification		
Property	Max.	Min.	Result
Friable particles (%)	0.50	0.00	0.00
Lightweight particles (%)	1.00	0.00	0.00
Absorption (%)	N/A	N/A	0.90
Specific gravity (SSD)	N/A	N/A	2.65
L. A. abrasion (%), 40%	N/A	N/A	22.8
Reduction in alkalinity	N/A	N/A	Innocuous
Dissolved silica	N/A	N/A	Innocuous
$NaSO_4$ soundness (%)	12	0	2.7

[1] 50% - 50% blend of each aggregate

TABLE 3.8.1-12

SUMMARY OF ACCEPTANCE TEST RESULTS
WATER

Property	Variance[1] Range Max.	Min.	Avg.	No. of Tests
Initial time of set, vicat (min)	9	1	5.5	20
Final time of set, vicat (min)	21	0	11.6	20
Autoclave expansion	+0.02	-0.04	-0.01	24
7-day compressive strength (%)	16.8	0.2	3.6	20

	Range Max.	Min.	Avg.	No. of Tests
Chlorides (ppm)	243	0.2	43.9	40
Solids (ppm)	825	18	265.9	40
Sulfates (ppm)	55.6	0.8	6.3	40

[1] Comparison of test water with control water

TABLE 3.8.1-13

SUMMARY OF ACCEPTANCE TEST RESULTS
ADMIXTURES

	Water Reducing						Air Entrainment		
	Range			Range			Range		
Property	Max.	Min.	Avg.	Max.	Min.	Avg.	Max.	Min.	Avg.
Solids (%)	42.8	41.7	42.2	43.3	40.6	42.2	27.3	27.2	27.3
Specific gravity	1.193	1.190	1.192	1.214	1.192	1.200	1.083	1.081	1.082
pH	7.6	7.2	7.4	7.4	6.4	6.8	12.9	12.8	12.9
Chloride(ppm)		3			6			2	

TABLE 3.8.1-14

SUMMARY OF ACCEPTANCE TEST RESULTS
CONCRETE (4000 PSI)

	MIX ID			
TEST RESULT	M-72	M-80	M-87	M-97
Slump (inches)	4	6	9 3/4	6
Compressive Strength @ 28 days (PSI)	6300	6950	7555	6035
Flexural Strength @ 28 days (PSI)	743	785	850	665
Splitting Tensile Strength @ 28 days (PSI)	620	510	580	525
Static Modulus of Elasticity @ 28 days	4.8×10^6	4.9×10^6	8.8×10^6	4.1×10^6
Poisson's Ratio	0.27	0.29	0.17	0.21
Coefficient of Thermal Conductivity (BTU/ft-hr-°F)	1.06	1.11	1.35	1.12
Coefficient of Thermal Expansion (in-in/°F)	5.6×10^{-6}	5.4×10^{-6}	6.5×10^{-6}	5.5×10^{-6}
Unit Weight (PCF)	143.7	142.9	137.5	138.7

TABLE 3.8.1-15

SUMMARY OF ACCEPTANCE TEST RESULTS
CONCRETE (5000 PSI)

	MIX ID			
TEST RESULT	M-44	M-56	M-57	M-81
Slump (inches)	3 3/4	4	4	6
Compressive Strength @ 28 days (PSI)	5270	5620	5450	6085
Flexural Strength @ 28 days (PSI)	652	640	675	770
Splitting Tensile Strength @ 28 days (PSI)	525	595	575	490
Static Modulus of Elasticity @ 28 days	4.0×10^{6}	5.3×10^{6}	5.1×10^{6}	4.5×10^{6}
Poisson's Ratio	0.22	0.21	0.30	0.26
Coefficient of Thermal Conductivity (BTU/ft-hr-°F)	0.97	1.32	1.08	1.07
Coefficient of Thermal Expansion (in-in/°F)	4.2×10^{-6}	5.6×10^{-6}	5.0×10^{-6}	6.5×10^{-6}
Unit Weight (PCF)	140.7	141.8	141.1	141.9

TABLE 3.8.1-16

SUMMARY OF ACCEPTANCE TEST RESULTS
CONCRETE (3000 PSI)

	MIX ID	
TEST RESULT	M-54	M-55
Slump	4	4
Compressive Strength @ 28 days (PSI)	4590	4220
Flexural Strength @ 28 days (PSI)	N/A	N/A
Splitting Tensile Strength @ 28 days (PSI)	495	505
Static Modulus of Elasticity @ 28 days	5.1×10^6	5.3×10^6
Poisson's Ratio	0.22	0.17
Coefficient of Thermal Conductivity (BTU/ft-hr-°F)	0.97	1.20
Coefficient of Thermal Expansion (in-in/°F)	5.6×10^{-6}	5.5×10^{-6}
Unit Weight (PCF)	140.7	143.5

TABLE 3.8.2-1

CONTAINMENT PENETRATIONS LOAD COMBINATIONS

A. For Class MC Components: Equipment Hatch, Personnel and Emergency Air Locks and Type I Sleeves

(Refer to Section 3.8.2.3 for load case descriptions)

1) $D + L + P_t + T_t$
2) $D + L + T_o + R_o$
3) $D + L + T_o + R_o + E$
4) $D + L + T_a + R_a + P_a + E$
5) $D + L + T_a + R_a + P_a + E'$
6) $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$

B. For Class 2 (NC) Components: Flued-head

1) Normal - $D + T_{ML}$
2) Upset - $D + T_{ML} + [(OBE + RV + WIND)^2 + (SH)^2]^{1/2}$
3) Emergency - $D + T_{ML} + [(DBE + RV + WIND)^2 + (SH)^2]^{1/2}$
4) Faulted 1 - $D + T_{ML} + [(DBE)^2 + (SH)^2\ (R)^2]^{1/2}$
5) Faulted 2 - $D + T_L + [(DBE) + RV + WIND)^2 + (SH)^2]^{1/2}$

D	-	Deadload
TML	-	Thermal loads: maximum of thermal loads for start-up and normal operations and shutdown conditions
OBE	-	Loads generated by the operating basis earthquake
DBE	-	Loads generated by the design bases earthquake (SSE)
RV	-	Relief valve discharge loads
SH	-	Steamhammer loads
WIND	-	Wind loads
TL	-	Thermal loads due to post-accident temperature/pressure causing containment dilation.

TABLE 3.8.2-2

CONTAINMENT MS&FW PENETRATIONS - SPRING CONSTANTS USED IN FINITE - ELEMENT ANALYSIS

Penetration Sleeve Location Outside Diameter (in.)		Sleeve Wall Thick. (in.)	Collar	Concr. Wall Thick Inch	Spring Constants (1) For Liner Plate (Kip/In^2)		Spring Constants (2) For Concrete Attachments (Kip/In^2)			Spring Constant (3) for Liner Anchor Studs (Kip/In^2)	Spring Constant for Concrete	
					Radial	Tangential	Tension	Hoop Shear	Merid. Shear		Radial (4) ($K/In^2/In$)	Bearing (5) ($K/In^2/In$)
M.S. 56	Lateral	2.75	Figure 3.8.1-17	78	210	208	5000	2000	300	217	108	740
	Central	2.75	Figure 3.8.1-17	78	200	195	5000	2000	300	217	105	740
F.W. 30	Lateral	2.00	Figure 3.8.1-17	78	377	376	5000	2000	300	214	206	811
	Central	2.00	Figure 3.8.1-17	78	373	369	5000	2000	300	214	206	811

1) Spring constants for the liner plate are defined per linear inch around the outside perimeter of collar ring.

2) Spring constants given are for each attachment. For location of concrete attachments see Figure 3.8.1-10.

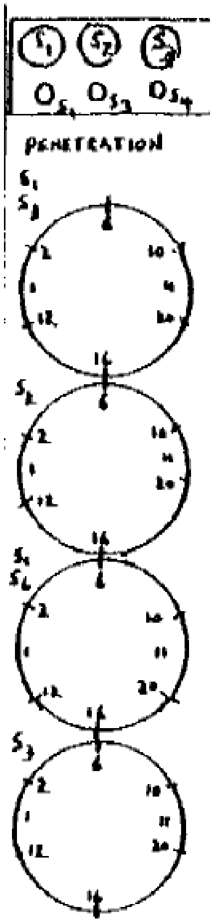
3) The spring constant for the liner anchor studs is defined per linear inch around the outside perimeter of the collar ring and is used in modeling the cable element on the collar ring.

4) Spring constants for the concrete are defined per square inch of supporting area and are used in modeling the radial spar element that simulates the surrounding concrete.

5) The spring constant for bearing is defined per linear inch around the outside perimeter of the collar ring and is used in modeling the bearing spar element on the collar ring.

TABLE 3.8.2-3

CONTAINMENT HS&FW PENETRATIONS - IMPOSED RADIAL DISPLACEMENTS



	Radial Displacement at Node																			
Load Condition	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
Test Pressure Condition	-2478	-2204	-1411	-518	226	616	305	-412	-1363	-2174	-2524	-2121	-1287	-370	308	661	293	-460	-1488	-2251
Accident Condition	-2792	-2529	-1562	-228	967	1587	1270	321	-1002	-2299	-2976	-2213	-1067	363	1265	1609	1007	-187	-1574	-2660
Test Pressure Condition	-1797	-1550	-912	-192	381	687	407	-203	-1000	-1633	-1894	-1533	-886	-126	460	726	431	-164	-1033	-1652
Accident Condition	-3671	-3343	-2149	-556	804	1525	1190	49	-1472	-2878	-3611	-2928	-1484	171	1294	1556	823	-551	-2178	-3320
Test Pressure Condition	-1281	-1036	-566	-119	190	256	-44	-502	-975	-1301	-1335	-1236	-919	-491	-68	210	168	-153	-592	-1067
Accident Condition	-1434	-1352	-903	-267	340	698	600	127	-523	-1192	-1538	-1105	-493	111	595	683	347	-291	-949	-1433
Test Pressure Condition	-1093	-853	-437	-67	175	214	-62	-454	-855	-1139	-1150	-1080	-838	-477	-105	166	149	-116	-484	-913
Accident Condition	-1863	-1692	-1110	-365	308	638	449	-141	-872	-1633	-1904	-1544	-865	-158	442	625	308	-385	-1120	-1759

1) Tabular data to be multiplied by $(10)^{-6}$.

2) Radial displacements are in feet.

TABLE 3.8.2-4

CONTAINMENT PENETRATIONS SLEEVE STRESS LIMITS

Combination No.		Gen. Memb. P_m	Local Memb. P_L	Bend + Local Memb. $P_B + P_L$	Primary & Secondary Stresses	Peak Stresses	Buckling Note (3)
(1)		$.9S_y$	$1.25S_y$	$1.25S_y$	$3S_m$	N/A	N/A
(2) & (3)		S_m	$1.5S_m$	$1.5S_m$	$3S_m$	N/A	N/A
(4)		S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	N/A
(5)	Not integral and continuous	S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	N/A
	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	N/A
(6)	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 $1.8S_m$ or $1.5S_y$	N/A	N/A	N/A
	Integral and continuous	85% of Stress Intensity Limits of Appendix F			N/A	N/A	N/A

Notes: (1) Thermal stresses need not be considered in computing Pm, PL, and PB.

(2) Thermal effects are considered in:

(a) Specifying stress intensity limits as a function of temperature.

(b) Analyzing effects of cyclic operation (NB-3222.4).

(3) Since the penetration sleeve diameter to thickness ratio is moderate, compressive stresses do not govern the penetration design.

Pm - General primary - membrane stress Sy - Yield strength of material

PL - Local primary - membrane stress Sm - Stress intensity

PB - Primary membrane stress

TABLE 3.8.2-5

ANALYSIS RESULTS FOR AS-BUILT CONDITION OF THE MAIN STEAM & FEEDWATER REGION OF THE CONTAINMENT

MATERIAL	ITEM	MAXIMUM STRESS (KSI)	ALLOWABLE STRESS (KSI)
Concrete Bearing at Collar	Compression	1.52	2.25
Concrete Element	Compression	1.43	2.25
Rebar	Tension	21.41	40.0
	Compression	10.48	40.0
Anchorage Assembly	Tension	16.67	34.88
	Compression	10.25	34.88
	Shear (Circumferential)	14.21	33.13
	Shear (Longitudinal)	9.91	36.44

TABLE 3.8.2-6

CONTAINMENT PENETRATION FLUED HEAD STRESS LIMITS

Combination	Gen. Memb. P_m	(Memb. or Local) + Bending (P_m or P_L) + P_B
Normal Level A	1.0 S_h	1.5 S_h
Upset Level B	1.1 S_h	1.65 S_h
Emergency Level C	1.5 S_h	1.8 S_h
Faulted Level D	2.0 S_h	2.4 S_h

NOTES:

(1) Thermal loadings are considered as mechanical loads on penetration assembly due to attached piping.

P_m General primary membrane stress

P_L Local primary membrane stress

P_B Primary membrane stress

S_h Allowable stress at design temperature

TABLE 3.8.3-1

CAPACITY REDUCTION FACTORS

φ = 0.90 for concrete in flexure.

φ = 0.85 for diagonal tension, bond, and anchorage in concrete.

φ = 0.75 for spirally reinforced concrete compression members.

φ = 0.70 for tied compression members.

φ = 0.90 for fabricated structural steel.

φ = 0.90 for mild reinforcing steel in tension (excluding splices).

φ = 0.85 for mild reinforcing steel in tension at lapped splices.

φ = 0.90 for mild reinforcing steel in tension with mechanical splices.

TABLE 3.8.4-1

LOAD COMBINATION TABLE FOR SEISMICALLY DESIGNED CONCRETE MASONRY WALL

Load Category	D	L	T_o	R_o	W	P_a	W_t	E	R_a	E'	T_a	Value for Allowable Stresses[1]
Normal	X	X	X	X								S
Severe Environmental (1)	X	X	X	X				X				S
Severe Environmental (2)	X	X	X	X	X							S
Extreme Environmental (1)	X	X	X	X						X		U
Extreme Environmental (2)	X	X	X	X			X					U
Abnormal	X	X				X			X		X	U
Abnormal/Severe Environmental	X	X				X		X	X		X	U
Abnormal/Extreme Environmental	X	X				X			X	X	X	U

[1] Values for S and U are specified in Table 3.8.4-2.

Table 3.8.4-2

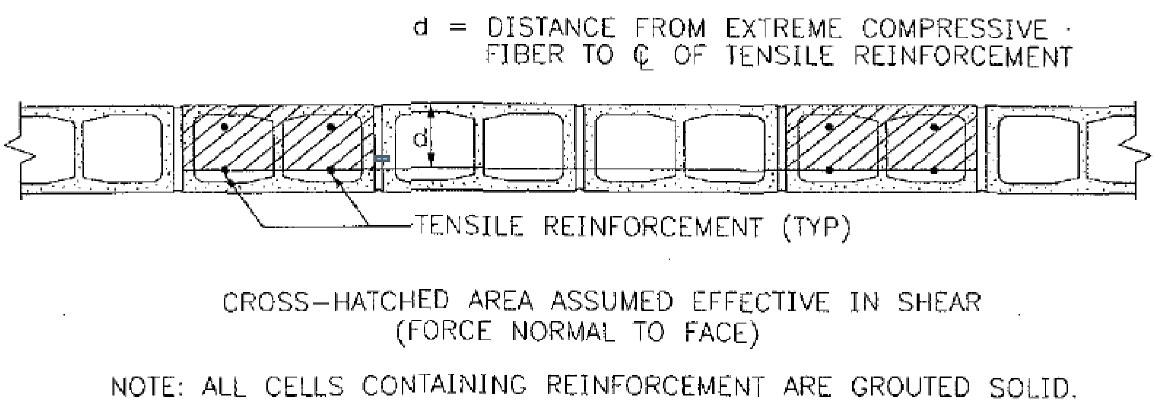
ALLOWABLE STRESSES IN REINFORCED MASONRY

Description	S Allowable (psi)[5]	S Maximum (psi)	U Allowable (psi)[5]	U Maximum (psi)
Compressive				
Axial[1]	$0.22f'_m$	1000	$0.55f'_m$	2000
Flexural	$0.33f'_m$	1200	$0.83f'_m$	2400
Bearing				
On full area	$0.25f'_m$	900	$0.62f'_m$	1800
On one-third area or less	$0.375f'_m$	1200	$0.95f'_m$	2400
Shear				
Flexural members[2]	$1.1\sqrt{f'_m}$	50	$1.43\sqrt{f'_m}$	75
Shear Walls[3,4]				
Masonry Takes Shear				
M/Vd >1	$0.9\sqrt{f'_m}$	34	$1.5\sqrt{f'_m}$	56
M/Vd =0	$2.0\sqrt{f'_m}$	74	$3.4\sqrt{f'_m}$	
Reinforcement Takes Shear				
M/Vd >1	$1.5\sqrt{f'_m}$	75	$2.5\sqrt{f'_m}$	125
M/Vd =0	$2.0\sqrt{f'_m}$	120	$3.4\sqrt{f'_m}$	180
Reinforcement				
Bond[6]				
Plain Bars		60		80
Deformed Bars		140		186
Tension				
Grade 40		20,000		$0.9f_y$
Grade 60		24,000		$0.9f_y$
Joint Wire		$0.5f_y$ but not to exceed 30,000		$0.9f_y$
Compression		$0.4f_y$ but not to exceed 24,000		$0.9f_y$
Shear Reinforcement and/or Bolts		$0.5f_y$		$0.75f_y$

Table 3.8.4-2 (cont'd)

Notes to Table 3.8.4-2:

1) These values should be multiplied by $(1-(h/40t)^3)$ if the wall has a significant vertical load.
2) This stress should be evaluated using the effective area shown in figure below:



3) Net bedded area shall be used with these stresses.
4) M is the maximum bending moment occurring simultaneously with the shear load V at the section under consideration. d is the depth from compression face of wall to centroid of tensile reinforcement. For M/Vd values between 0 and 1 interpolate between the values given for 0 and 1.
5) f'_m = specified compressive strength of masonry of net cross-sectional area, (= 1,150 psi).
   f_y = specified yield strength of reinforcement (= 60,000 psi).
6) Alternately, use development length $1_d = 0.0015\ d_b f_s$, where d_b = nominal diameter of bar or wire in inches, f_s calculated stress in reinforcement in psi. 1_d shall not be less than 12".

TABLE 3.8.4-3

SEISMIC CATEGORY I TANKS AND THEIR LOCATION

Tank*	Building	Location	General Arrangement Fig. No.
Volume Control Tank	RAB	See Table 3.5.1-1	
Boric Acid Tank	RAB	See Table 3.5.1-1	
Boron Recycle Holdup Tank	RAB	El 236.00'	Fig. 1.2.2-23
Boron Injection Tank	RAB	El 216.00'	Fig. 1.2.2-19
Boron Injection Surge Tank	RAB	El 236.00'	Fig. 1.2.2-23
Refueling Water Storage Tank	Tank Bldg	El 261.00'	Fig. 1.2.2-84
Spray Additive tank	RAB	El 216.00'	Fig. 1.2.2-19
Component Cooling System Surge Tank	RAB	See Table 3.5.1-1	
Reactor Coolant Drain Tank	CB	El 236.00'	Fig. 1.2.2-3
Expansion Tank (Essential Services)	RAB	El 324.00'	Fig. 1.2.2-35
Diesel Generator Fuel Oil Storage Tank	Yard	See Table 3.5.1-1	
Diesel Generator Fuel Oil Day Tank	DGB	El 280.00'	Fig. 1.2.2-86
Condensate Storage Tank	Tank Bldg	See Table 3.5.1-1	
Reactor Makeup Water Storage Tank	Tank Bldg	El 261.00'	Fig. 1.2.2-84
Accumulator	CB	El 261.00'	Fig. 1.2.2-7
Nitrogen Accumulator	CB	El 295.00' El 302.00’	Fig 1.2.2-15
Make-up Tank (Essential Services)	RAB	El 261.00'	Fig. 1.2.2-27

* From Table 3.2.1-1

TABLE 3.8A-1

PROCUREMENT, FABRICATION, AND ERECTION
STATUS OF THE CONTAINMENT LINER***

PART OR ATTACHMENT	PROCUREMENT***	FABRICATION***	ERECTION***
Dome Liner 1/2 in.	YES	YES	NO
Containment Wall Liner 3/8 in.	YES	YES	NO
Sump Pit Liner Assembly 1/4 in.	YES	YES	NO
Shear Key Liner & Emb. Studs 1/4 in	YES	YES	NO
Bottom Liner & Emb. 1/4 in.	YES	YES	NO
Crane Plate Assembly Including			
Backup Bars	YES	YES	NO
Brackets	YES	YES	NO
Transfer Tube Sleeve	YES	YES	NO
Type II Penetration Pipe Sleeve	YES	YES	NO
H & V Sleeve	YES	YES	NO
Equipment Hatch	YES	NO	NO
Elec. Pen Sleeve	YES	YES	NO
Personnel Lock & Anchorage	YES	NO	NO
Escape Lock & Anchorage	YES	NO	NO
Valve Chamber & Anchorage	YES	YES	NO
Leak Chase	YES	YES	NO
Type I Penetration Anchorage System	NO	NO	NO

*** Status as of April 29, 1977.

TABLE 3.9.1-1

SUMMARY OF LIMITING REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Normal Conditions	Occurrences
1. Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr.).	200 (each)
2. Unit loading and unloading at 5 percent of full power per minute between 15 and 100 percent power	18,300 (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	150,000 (Initial) 3,000,000 (Random)
6. Feedwater cycling at hot standby	2,000
7. Unit loading and unloading between 0 and 15 percent power	500 (each)
8. Boron concentration equalization	26,400
9. Refueling	80
10. Turbine roll test	80
11. Primary side leakage test	200
12. Secondary side leakage test	80

Upset Conditions	Occurrences
1. Loss of load, without immediate reactor trip	200
2. Loss of power (with natural circulation in the Reactor Coolant System)	40
3. Partial loss of flow (loss of one pump)	80
4. Reactor trip from full power	
Case A - No Cooldown	230
Case B - Cooldown, No SI	160
Case C - Cooldown with SI	10
5a. Inadvertent RCS depressurization	20
5b. Inadvertent Auxiliary Spray Cooling	10
6. Operating Basis Earthquake (5 earthquakes of 10 cycles each)	50 cycles
7. Excessive Feedwater Flow	30
8. Control Rod Drop	80
9. Inadvertent Safety Injection	60
10. RCS Cold Overpressurization	10

Emergency Conditions*	Occurrences
1. Small Loss-Of-Coolant	5
2. Small Steam Line Break	5
3. Complete Loss of Flow	5

Faulted Conditions*	Occurrences
1. Reactor coolant (primary system) pipe break (large loss-of-coolant accident)	1
2. Large steam line break	1
3. Feedwater line break	1
4. Steam generator tube rupture	6
5. Reactor coolant pump locked rotor	1
6. Control rod ejection	1
7. Safe Shutdown Earthquake	1

Test Conditions	Occurrences
1. Primary side hydrostatic test	10
2. Secondary side hydrostatic test	10
3. Steam generator tube leakage test**	800

*In accordance with the ASME Code, Section III, emergency and faulted conditions are not included in fatigue evaluation.

TABLE 3.9.1-1 (Continued)

**Westinghouse Analysis of Record (AOR) (i.e. WCAP-14778) listed two more transients, which are Loop Out of Service and Inadvertent Startup of an Inactive Loop. The Harris Nuclear Power plant is not licensed to operate with N-1 loops. As such, these two transients are not applicable to HNP. Therefore, it is not necessary to add these transients into the FSAR. However, it has been recommended that if the current component specific engineering specifications include these transients, these transients will continue to be considered in the structural analysis of that specific component.

TABLE 3.9.1-2

LOADING COMBINATIONS FOR ASME CLASS 1 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

Plant Classification	Design/Service Level	Loading Combination
Design		Design pressure, design temperature, deadweight
Normal	Service level A	Normal condition transients, deadweight
Upset	Service level B	Upset condition transients, deadweight, OBE
Emergency	Service level C	Emergency condition transients, deadweight
Faulted	Service level D	Faulted condition transients, deadweight, SSE, pipe rupture loads

TABLE 3.9.1-2a

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING AND SUPPORTS

ASME Class 1 Portion (Analyzed by Westinghouse)

CONDITION CLASSIFICATION	LOAD COMBINATION(2)	ASME III NB-3650 EQUATION 9 STRESS LIMITS
Design	Design Pressure, Weight	1.5 Sm(5)
Normal	Normal Condition Transients, Weight	1.5 Sm
Upset	Upset Condition Transients, Weight, OBE(1), VT-R(3)	1.8 Sm/1.5 Sy(6)(7)
Emergency	Emergency Condition Transients, Weight, VT-S(3)	2.25 Sm/1.8 Sy(7)
Faulted	Faulted Condition Transients, Weight, SSE(1), VT-M(3), MS/FWPB or DBPB(4)	3.0 Sm
Faulted	Faulted Condition Transients, Weight, SSE(1), VT-M(3), LOCA(4)	3.0 Sm

NOTES:

(1) The OBE and SSE loadings include the effects of seismic anchor motions.

(2) Dynamic loads are combined by SRSS.

(3) Valve thrust loads (VT) are loads resulting from the rapid acceleration or deceleration of a water mass, non-condensable gases, or both. These loads are defined as follows:

VT-R: Loadings due to relief valve discharge

VT-S: Loadings due to safety valve discharge

VT-M: Maximum of VT-R and VT-S

(4) Pipe rupture loadings are defined as follows:

MS/FWPB: Main steam or feedwater pipe break

DBPB: Design basis pipe break

LOCA: Loss of coolant accident

(5) Sm: Allowable design stress intensity value at temperature

(6) Sy: Yield strength value at temperature

(7) The allowable stress is the lesser of the two values given.

TABLE 3.9.1-3

STRESS CRITERIA FOR ASME B&PV CODE, SECTION III
CLASS 1 COMPONENTS[a]

Design/Service Level	Vessels/Tanks	Piping	Pumps	Valves	Component Supports [1]
Design and service level A	ASME B&PV Code, Section III NB 3221, 3222	ASME B&PV Code, Section III NB 3652, 3653	ASME B&PV Code, Section III NB 3221, 3222	ASME B&PV Code, Section III NB 3520, 3525	ASME B&PV Code, Section III, Subsection NF NF 3221, NF 3222, NF 3231.1(a)
Service level B (UPSET)	ASME B&PV Code, Section III NB 3223	ASME B&PV Code, Section III NB 3654	ASME B&PV Code, Section III NB 3223	ASME B&PV Code, Section III NB 3525	ASME B&PV Code, Section III, Subsection NF NF 3223, 3231.1(a)
Service level C (Emergency)	ASME B&PV Code, Section III NB 3224	ASME B&PV Code, Section III NB 3655	ASME B&PV Code, Section III NB 3224	ASME B&PV Code, Section III NB 3526	ASME B&PV Code, Section III NF 3224, 3231(b)
Service level D (Faulted)	ASME B&PV Code, Section III see paragraph 3.9.1.4 NB 3225	ASME B&PV Code, Section III see paragraph 3.9.1.4 NB 3656	ASME B&PV Code, Section III (No active class 1 pump used) NB 3225	(b)	ASME B&PV Code, Section III, Subsection NF see paragraph 3.9.1 NF 3225, 3231.1(c)

P_e, P_m, P_b, Q_t, C_p, S_n and S_m as defined by ASME B&PV Code, Section III

[a] A test of the components may be performed in lieu of analysis.

b. CLASS I VALVE SERVICE LEVEL D CRITERIA

ACTIVE	INACTIVE
Calculate P_m from Subsection NB3545.1 with Internal Pressure $P_s = 1.25\ P_s$ $P_m \leq 1.5S_m$	Calculate P_m from Subsection NB3545.1 with Internal Pressure $P_s = 1.50\ P_s$ $P_m \leq 2.45S_m$ or $0.75S_u$
Calculate S_n from Subsection 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) $S_n \leq 3S_m$	Calculate S_n from Subsection NB3545.2 with $C_p = 1.5$ $P_s = 1.50P_s$ $Q_t^2 = 0$ P_ed= 1.3X value of P_ed from equations of 3545.2(b)(1) $S_n \leq 3S_m$

[1] Subsection NF is used for stress criteria only. See FSAR Subsection 3.9.1.4.7

TABLE 3.9.1-4

MAXIMUM REACTOR VESSEL DISPLACEMENTS AT REACTOR VESSEL CENTERLINE

Break	Maximum Vertical Displacement (inches)	Maximum Horizontal Displacement (inches)	Maximum Rotation (Radians)
144 in.^{2}	0.034	0.069	0.00016
RPV Inlet Break	-0.033	-0.0	-0.00028
144 in.^{2}	0.008	0.080	0.00004
RPV Outlet Break	-0.013	-0.0	-0.00019
Double Ended			
Pump Outlet	0.009	0.055	0.00022
Break	-0.021	-0.006	-0.00028

(Maximum displacements are no longer used. A detailed time history analysis is performed for equipment qualification). (Historical Information)

TABLE 3.9.1-5

MAXIMUM REACTOR VESSEL SUPPORT LOADS
FOR POSTULATED PIPE RUPTURE CONDITIONS

Maximum Vertical Pad Support Load	Maximum Horizontal Pad Support Load
762 kips	400 kips

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Reactor Coolant System	
Reactor Vessel	1
CRDM Housing	1
Steam Generator (tube side)	1
(shell side)	1
Pressurizer	1
Reactor Coolant Hot and Cold Leg Piping, Fittings and Fabrication	1
Surge Pipe, Fittings and Fabrication	1
Crossover Leg Piping, Fittings and Fabrication	3
Pressurizer Safety Valves	1
Power Operated Relief Valves	1
Valves of Safety Class 1 to Safety Class 2 Interface	1
Pressurizer Relief Tank	1
Reactor Coolant Thermowell	1
Reactor Coolant Pump	
RCP Casing	1
Main Flange	1
Thermal Barrier	1
Thermal Barrier Heat Exchanger	1
#1 Seal Housing	1
#2 Seal Housing	1
Pressure Retaining Bolting	1
RCP Motor	1
Motor Rotor	1
Motor Shaft	1
Shaft Coupling	1
Spool Piece	1
Flywheel	
Bearing (Motor Upper Thrust)	1
Motor Bolting(2)	1
Motor Stand	1
Motor Frame	1
Upper Oil Reservoir (UOR) UOR Oil Cooler	1

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Lower Oil Reservoir (LOR) LOR Cooling Coil	1
<u>Reactor Coolant System (Cont.)</u>	
Lube-Oil Piping	1
<u>Chemical & Volume Control System</u>	
Regenerative HX	1
Letdown HX (tube side)	1
(shell side)	1
Mixed Bed Demineralizer	1
Cation Bed Demineralizer	1
Volume Control Tank	1
Centrifugal Charging Pump	2
Boric Acid Blender	1
Letdown Orifices	1
Excess Letdown HX (tube side)	1
(shell side)	1
Seal Water HX (tube side)	1
(shell side)	1
Boric Acid Transfer Pump	2
Boric Acid Batching Tank	5
Chemical Mixing Tank	5
Boron Meter	5
Reactor Coolant Pump (RCP) Standpipe	5
RCP Standpipe Orifice	5
RCP Seal Bypass Orifice	2
<u>Boron Thermal Regeneration Subsystem</u>	
Moderating HX (tube side)	1
(shell side)	1
Letdown Chiller HX (tube side)	1
(shell side)	5
Letdown Reheat HX (tube side)	1
(shell side)	1
Thermal Regeneration Demineralizer	1
Chiller Pump	
Chiller Surge Tank	5

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Boron Thermal Regeneration Subsystem (Cont.)	
Chiller Unit	5
Evaporator	5
Condenser	5
Compressor	5
Emergency Core Cooling System	
Accumulators	1
Boron Injection Tank	1
Hydro Test Pump	5
Residual Heat Removal System	
Residual Heat Removal Pump	2
Residual Heat Exchanger (tube side)	1
(shell side)	1
Boron Recycle System	
Recycle Evap. Feed pump	2
Recycle Evap. Feed Demineralizer	2
Recycle Evap. Feed Filter	2
Recycle Evap. Concentrate Filter	5
Reactor Evap. Condensate Demineralizer	5
Reactor Makeup Water Pump	5
Recycle Evap. Reagent Tank	5
Recycle Holdup Tank Vent Ejector	2
Recycle Evaporator Package	
1. Feed Preheater	
a. Feed Side	2
b. Steam Side	5
2. Gas Stripper	2
3. Submerged Tube Evap	
a. Feed Side	2
b. Steam Side	5
4. Evaporator Condenser	
a. Distillate Side	2
b. Cooling Water Side	2
5. Distillate Cooler	
a. Distillate Water Side	2
b. Cooling Water Side	2

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Boron Recycle System (Cont.)	
6. Absorption Tower	2
7. Vent Condenser	
a. Gas Side	2
b. Cooling Water Side	2
8. Distillate Pump	2
9. Concentrate Pump	2
10. Piping	
a. Feed	2
b. Distillate	2
c. Concentrate	2
11. Valves	
a. Feed	2
b. Distillate	2
c. Concentrate	2
d. Cooling	2
e. Steam	5
Main Steam	
Main Steam Isolation Valves	2
Liquid Waste Processing System	
Reactor Coolant Drain Tank	5
Reactor Coolant Drain Tank Pump	5
Reactor Coolant Drain Tank HX (tube side)	5
(shell side)	2
Waste Evaporator Feed Pump	5
Waste Evaporator Package	
1. Feed Preheater	
a. Feed Side	5
b. Steam Side	5
2. Gas Stripper	5
3. Submerged Tube Evaporator	
a. Feed side	5
b. Steam Side	5
4. Evaporator Condenser	
a. Distillate Side	5
b. Cooling Water Side	5

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Liquid Waste Processing System (Cont.)	
5. Distillate Cooler	
a. Distillate Water Side	5
b. Cooling Water Side	5
6. Absorption Tower	5
7. Vent condenser	
a. Gas Side	5
b. Cooling Water Side	5
8. Distillate Pump	5
9. Concentrate Pump	5
10. Piping	
a. Feed	5
b. Distillate	5
c. Concentrate	5
11. Valves	
a. Feed	5
b. Distillate	5
c. Concentrate	5
d. Cooling	5
e. Steam	5
Waste Evaporator Condensate Demineralizer	5
Waste Evaporator Condensate Tank	5
Waste Evaporator Condensate Tank Pump	5
Chemical Drain Tank	5
Chemical Drain Tank Pump	5
Spent Resin Sluice Pump	5
Laundry & Hot Shower Tank Pump	5
Floor Drain Tank Pump	5
Waste Monitor Tank Pump	5
Waste Monitor Tank Demineralizer	5
Waste Evaporator Reagent Tank	5
Gaseous Waste Processing System	
Gas Compressor	5
Gas decay Tank([3])	5
Hydrogen Recombiner (Catalytic)	5

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Gas Decay Tank Drain Pump	2
Fuel Handling System	
Manipulator Crane(10)	5
Reactor Vessel Internals Lifting Device	2 or 5(4)
Rod Cluster Control Changing Fixture	5
Reactor Vessel Stud Tensioner	5
Spent Fuel Handling Tool(5)	1
Fuel Transfer System Fuel Transfer Tube & Flange(6)	1
Conveyor System & Controls(7)	1
Remainder of System	
New Fuel Elevator	5
New Fuel Racks	3
Portable Underwater Lights	5
Load Cell	5
Stud Hole Plug Handling Fixture	5
Stud Hole Plugs	5
Upper Internals Storage Stand	5
Rod Cluster Control Thimble Plug Tool	5
Spent Fuel Bridge Crane	1
Fuel Handling System (Cont.)	
Source Installation Guide	5
Crane Scales	5
Irradiation Tube End Plug Seat Jack	5
Burnable Poison Rod Handling Tool	5
Irradiation Sample Handling Tool	5
Burnable Poison Assembly Rack Inserts	5
Control Rod Drive Shaft Handling Fixture	5
Control Rod Drive Shaft Unlatching Tool, Full Length	5
New Fuel Elevator Winch	5
New Fuel Assembly Handling Fixture	5
New Rod Cluster Control Handling Fixture	5
Lower Internals Storage Stand	5
Upper Internals Storage Stand	5
Load Cell Linkage	5
Spent Fuel Storage Racks	3

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Reactor Vessel or Core Related	
Reactor Vessel Shoes and Shims[8]	1
Irradiation Sample Holder	2
Irradiation Samples	5
Control Rod Drive Mechanism (CRDM) Dummy Can Assemblies	5
CRDM Assemblies	5
CRDM Seismic Support Tie Rod Assemblies	2
Reactor Vessel Internals	2
Primary Source Rods	5
Neutron Detector Positioning Device	2
Reactor Vessel Insulation - Shell	5
Reactor Coolant Pipe Insulation	5
Rod Cluster Controls Full Length Assembly	1
Burnable Poison Rod Assemblies	5
Primary & Secondary Source General Assembly	5
Component Cooling Water System	
Component Cooling Water Pump	3
Component Cooling Water Heat Exchanger	3
Component Cooling Water Surge Tank	3
Integrated Head (IH) Package	
IH Cable Assemblies	5
Radial Arm Stud Tensioner Hoist Assembly	5
IH Cooling Fans	5
IH Cooling Shroud	5
IH Cable Tray	5
IH Shroud Cooling Fan Duct	5
IH Lift Rig	5
IH Lift Rods	1
IH Missile Shield	1
IH Reactor Vessel Stud Support Collars	5
IH Lifting Rig Operatory Support Stand	5
Sampling System	
Gross Failed Fuel Detector (GFFD)	5
GFFD Sample Cooler	2

Table 3.9.2-1: METHOD OF QUALIFICATION FOR NSSS EQUIPMENT

Component	Seismic Design (1)
Incore Instrumentation	
Seal Table Assembly[9]	1
Flux Thimble Tubing	1
Flux Thimble Fittings	1
Thimble Guide Tubing	1
Reactor Coolant System Equipment Supports	
Hydraulic Shock Supressors	1
Crossover Leg Restraint	1
Steam Generator Manway Cover Support	5
Reactor Coolant Pump Supports	1
Pressurizer Support Ring	1
Reactor Vessel Supports	1
Control Rod Drive Mechanism Seismic Support Tie Rods	1
Reactor Coolant Pump Forging Type A	1
Steam Generator Forging Type A	1

Notes: Table 3.9.2-1

1. Numbers in this column identify the method used for seismic qualification as follows. Information as to seismic qualification methods is given in Section 3.7.
    1. Individual analysis dynamic method.
    2. Generic analysis dynamic method.
    3. Static method using dynamic load factors.
    4. Testing method.
    5. Non-seismic design.
2. Applies only to bolting involved with coastdown function.
3. Supports for the gas decay tanks must be designed to withstand the operating basis earthquake.
4. Portions that furnish support to control rod drive mechanisms are seismically designed, the rest is non-seismically design.
5. Failure could cause releases of radioactivity.
6. Portions of containment boundary.
7. Protects fuel from damage during transportation.
8. Provides mechanical support for the reactor vessel.
9. Provides support to the Safety Class 1 pressure boundary conduit.
10. The manipulator crane will not collapse, derail or cause a fuel assembly to become disengaged from the gripper, as a consequence of an SSE.

TABLE 3.9.3-1

Design Loading Combinations for ASME Code
Class 2 and 3 NSSS Supplied Components

Condition Classification	Loading Combination
Design and Normal	Design pressure Design temperature* Dead weight, nozzle loads**
Upset	Upset condition pressure, Upset condition metal temperature,* deadweight, OBE, nozzle loads**
Emergency	Emergency condition pressure, emergency condition metal temperature,* SSE, deadweight, nozzle loads**
Faulted	Faulted condition pressure, faulted condition metal temperature,* deadweight, SSE, nozzle loads**

*Temperature is used to determine allowable stress only.

**Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration and include the thermal effects of attached piping.

TABLE 3.9.3-2

Stress Criteria for ASME Code
Class 2* and 3 NSSS Supplied Tanks

Condition	Stress Limits
Design and Normal	$\sigma m < 1.0$ S $(\sigma m$ or $\sigma L) + \sigma b < 1.5$ S
Upset	$\sigma m < 1.1$ S $(\sigma m$ or $\sigma L) + \sigma b < 1.65$ S
Emergency	$\sigma m < 1.5$ S $(\sigma m$ or $\sigma L) + \sigma b < 1.80$ S
Faulted	$\sigma m < 2.0$ S $(\sigma m$ or $\sigma L) + \sigma b < 2.4$ S

*Applies for tanks designed in accordance with the ASME Code Section III, NC-3300.

TABLE 3.9.3-3

STRESS CRITERIA FOR CLASS 2 NSSS SUPPLIED TANKS*

Condition	Stress Limits
Design and Normal	Pm $<$ 1.0 Sm P_L $<$ 1.5 Sm (Pm or P_L + Pb $<$ 1.5 Sm
Upset	Pm $<$ 1.1 Sm P_L $<$ 1.65 Sm (Pm or P_L + Pb $<$ 1.65 Sm
Emergency	Pm $<$ 1.2 Sm P_L $<$ 1.8 Sm (Pm or P_L) + Pb $<$ 1.8 Sm
Faulted	Pm $<$ 2.0 Sm (Pm or P_L) + Pb $<$ 3.0 Sm

*Applies for tanks designed in accordance with the ASME Code, Section VIII, Division 2 rules.

TABLE 3.9.3-4

Stress Criteria for ASME Code Class 2 and 3 NSSS Supplied Inactive Pumps

Condition	Stress Limits
Design and Normal	$\sigma m<1.0$ S (σm or σL) + $\sigma b<1.5$ S
Upset	$\sigma m<1.1$ S (σm or σL) + $\sigma b<1.65$ S
Emergency	$\sigma m<1.5$ S (σm or σL) + $\sigma b<1.80$ S
Faulted	$\sigma m<2.0$ S (σm or σL) + $\sigma b<2.4$ S

TABLE 3.9.3-5

Stress Criteria for ASME Code Class 2 and 3
NSSS Supplied Active Pumps

Condition	Stress Limits
Design, Normal and Upset	$\sigma m<1.0$ S $\sigma m + \sigma b<1.5$ S
Emergency	$\sigma m<1.2$ S $\sigma m + \sigma b<1.65$ S
Faulted	$\sigma m<1.2$ S $\sigma m + \sigma b<1.8$ S

TABLE 3.9.3-6

STRESS CRITERIA FOR ASME CODE CLASS 2 AND 3
NSSS SUPPLIED ACTIVE AND INACTIVE VALVES*

Condition	Stress Limits	P_{max}
Design and Normal	Valve bodies conform to ASME Section III	1.0
Upset	$\sigma m<1.1$ S (σm or σL) + $\sigma b<1.65$ S	1.1
Emergency	$\sigma m<1.5$ S (σm or σL) + $\sigma b<1.80$ S	1.2
Faulted	$\sigma m<2.0$ S (σm or σL) + $\sigma b<2.4$ S	1.5

*Valve nozzle (piping load) stress analysis is not required when both 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 for valve body material is equal to or greater than the code allowable stress, of 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 the requirement, an analysis in accordance with the design procedure for Class 1 valves is an acceptable alternate method.

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.

TABLE 3.9.3-7

DESIGN LOADING COMBINATIONS FOR NON-NSSS SUPPLIED ASME CODE CLASS 2 AND 3 PIPING

Condition Classification	Loading Combination
Design + Normal	Internal pressure, weight, sustained loads, thermal expansion
Upset	Internal pressure, weight, sustained loads, upset occasional loads (2), OBE (1), thermal expansion
Emergency	Internal pressure, weight, sustained loads, SSE, emergency occasional loads (2)
Faulted	Internal pressure, weight, sustained loads, SSE, pipe rupture and/or jet impingement effects where applicable

Notes:

(1) OBE has been considered for all Seismic Category I components.

(2) Includes temporary loads such as relief valve thrusts depending upon the specific plant process condition.

TABLE 3.9.3-7a

DESIGN STRESS LIMITS AND LOADING COMBINATIONS
FOR NON-NSSS SUPPLIED ASME
CODE CLASS 1, 2, AND 3 COMPONENT SUPPORTS

Condition Classification	Loading Combination	Service Limits(4)(5)(6)	
		Flexure	Shears
Design + Normal	Internal pressure + weight+ sustained loads + thermal expansion.	S(3)	S
Upset	Internal pressure + weight + sustained loads + upset occasional loads(2)+ OBE + thermal expansion.	S	S
Emergency	Internal pressure + weight + sustained loads + SSE + emergency occasional loads.	1.33S(1)	1.3S
Faulted	Internal pressure + weight+ sustained loads + SSE + pipe rupture and/or jet impingement effects where applicable.	1.5S	1.3S

Notes:

(1) This limit may be increased to 1.5S for non-essential systems.

(2) Includes temporary loads such as relief valve thrusts depending upon the specific plant process condition.

(3) Service limit, 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."

(4) Deformation criteria is 1/16 in. for rigid frame supports including anchors and 1/32 in. for frames supporting snubbers. These limits apply to deflection in the direction of restraint at the point of load, under upset conditions.

(5) Compressive stress in compression members is limited to 2/3 critical buckling stress.

(6) Material yield stress is not exceeded.

TABLE 3.9.3-8

STRESS CRITERIA FOR NON-NSSS SUPPLIED ASME CODE CLASS 2 AND 3 PUMPS AND VALVES

Condition	Stress Limits	P_{max}
Design and Normal	Valve bodies conform to the requirements of ASME Section III.	1.0
Upset	$\sigma m<1.1\ S$ $\sigma m + \sigma b <1.65\ S$	1.1
Emergency	$\sigma m<1.5\ S$ $\sigma m + \sigma b <1.80\ S$	1.2
Faulted	$\sigma m<2.0\ S$ $\sigma m + \sigma b <2.4\ S$	1.5

Note: (1) Stress limits are applicable to the pressure-retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable. Design requirements listed in this table are not applicable to valve discs, stems, set rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

σm = General Membrane Stress

σb = Bending Stress

S = Allowable Stress

(2) If the crotch region geometry is such that section modulus and metal area is at least 110 percent of connected piping, the integrity of the valve is assured by the integrity of the connected pipe which is designed and analyzed in accordance with ASME NC/ND-3600.

(3) The maximum pressure shall not exceed the tabulated factors listed under P_{max} times the design pressure or times the rated pressure at the applicable service temperature. If these pressure limits are met, the stress limits in this table are considered to be satisfied.

TABLE 3.9.3-9

STRESS CRITERIA FOR NON-NSSS SUPPLIED ASME CODE CLASS 2 AND 3 TANKS

Condition	Stress Limits
Design and Normal	Stresses induced by loading combination shall be within normal working stresses
Upset	Stresses induced by loading combination shall be within normal working stresses
Faulted (including SSE)	1.5 times the normal working stresses but plastic strains or failure shall not occur. The integrity of the tank shall be maintained and proper functioning assured
Faulted (including Tornado) (1)(2)	1.5 times the normal working stresses but plastic strains or failure shall not occur. The integrity of the tank shall be maintained and proper functioning assured.

Notes: (1) Applies to the reactor makeup water storage tank only.

(2) Tornado (W_t) equals any of the following for the worst condition for the tank roof, shell, anchorage, and eave. The tank design considers the tank as being at least 90 percent full of product or empty.

$W_t = W_w$

$W_t = W_p$

$W_t = W_w + 1/2\ W_p$

W_w and W_p are described in Section 3.8.4.3

TABLE 3.9.3-10

STRESS CRITERIA FOR NON-NSSS SUPPLIED ASME CODE CLASS 2 AND 3 PRESSURE VESSELS

Component	Condition	Stress Limits	S Allowable σm	S Allowable $(\sigma m \text{ or } \sigma_L) + \sigma_b$
Pressure Vessel (ASME III, Div. 1)	Design/Normal	ASME III, NC/ND-3300	1.0S	1.5S
	Upset	ASME III, NC/ND-3300	1.1S	1.65S
	Emergency	ASME III, NC/ND-3300	1.5S	1.8S
	Faulted	ASME III, NC/ND-3300	2.0S	2.4S
Pressure Vessel (ASME VIII, Div.2)		Not Applicable (See Section 1.8, R-G. 1.48)		

TABLE 3.9.3-11

STRESS CRITERIA FOR NON-NSSS SUPPLIED ASME CODE CLASS 2 AND 3 PIPING

Condition	Equation*	S Allowable
Design/Normal	(8)	$1.0S_h$
	(10)	S_A
	(11)	$S_h + S_A$
Upset	(9)	$1.2S_h$
	(10)	S_A
	(11)	$S_h + S_A$
Emergency	(9)	$1.8S_h$
Faulted	(9)	$2.4S_h$

*Equations from ASME - III, Subsection NC/ND-3650, 1971 Edition through Summer 1973 Agenda. For design loading combinations, refer to Table 3.9.3 7.

TABLE 3.9.3-12

NSSS Supplied Active Pumps

Pump	Item Number	System	ANS Safety Class	Normal Mode	Active Mode	Basis/Function
Centrifugal charging pumps 1, 2, and 3	APCH	CVCS	2	On/Off	On	ECCS operation; safe shutdown
Boric Acid transfer pumps 1 and 2	APBA	CVCS	3	On/Off	On	Safe shutdown
Residual heat removal pumps 1 and 2	APRH	RHRS	2	Off	On	ECCS operation
Component cooling pumps 1, 2, and 3	APCC	CCS	3	On/Off	On	ECCS operation; safe shutdown

NOTE: This is historical information. See Section 3.9.3.2.1.

TABLE 3.9.3-13 NSSS SUPPLIED ACTIVE CLASS 1, 2, AND 3 VALVES

Ebasco Tag Number	Westinghouse Tag Number	System	Location	Environmental Qualification	Type	Operator	Manufacturer	Safety Class	Valve Design Rating	System Operating Conditions	Size	Function
1RC-R528SN 529 530	1-8010A,B,C	RC	RCB	-	Relief	Self-actuated	Crosby	1	1500	2485 psig 589 F	6	Pressurized Safety
2RC-V525SB	1-8046	RC	RCB	-	Check	ΔP	Westinghouse	2	150	3 psig 120 F	3	Containment Isolation
2RC-D525SB	1-8028	RC	RAB	(6)	Diaphragm	Air	Grinnell	2	150	3 psig 120 F	3	Containment Isolation
2RC-D528SA-1	1-8047	RC	RCB	-	Diaphragm	Air	Grinnell	2	150	3 psig 120 F	1	Containment Isolation
2RC-D529SB-1	1-8033	RC	RCB	-	Diaphragm	Air	Grinnell	2	150	3 psig 120 F	1	Containment Isolation
2CS-V516SA	1-8112	CS	RCB	(1)	Globe	Motor	Velan	2	1500	150 psig 200 F	2	Containment Isolation
2CS-V517SB	1-8100	CS	RAB	(6)	Globe	Motor	Velan	2	1500	150 psig 200 F	2	Containment Isolation
1CS-V505SN	1-8378	CS	RCB	-	Check	ΔP	Westinghouse	1	1500	2435 psig 560 F	3	RCS Press. Bound.Isol.
1CS-V506SN	1-8379	CS	RCB	-	Check	ΔP	Westinghouse	1	1500	2435 psig 560 F	3	RCS Press. Bound.Isol.
1RC-P527SA-1 1RC-P529SB-1	1-PCV-445A 1-PCV-444B	RC	RCB	(2),(7)	Globe	Air	Copes Vulcan	1	1500	2485 psig 650 F	3	Mitigate SGTR
1CS-V504SN	1-8347	CS	RCB	-	Check	ΔP	Westinghouse	1	1500	2435 psig 560 F		RCS Press. Bound.Isol.
1CS-V507SN	1-8346	CS	RCB	-	Check	ΔP	Westinghouse	1	1500	2435 psig 560 F	3	RCS Press. Bound.Isol.
1CS-L500SN	1-LCV-460	CS	RCB	(2)	Globe	Air	Copes Vulcan	1	1500	2485 psig 589 F	3	RCS Press. Bound.Isol.
1CS-L501SN	1-LCV-459	CS	RCB	(2)	Globe	Air	Copes Vulcan	1	1500	2485 psig 589 F	3	RCS Press. Bound.Isol.
2CS-V515SN	1-8381	CS	RCB	-	Check	ΔP	Westinghouse	2	1500	2430 psig 130 F	3	Containment Isolation
2CS-V610SA	1-8107	CS	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2670 psig 170 F	3	Containment Isolation

TABLE 3.9.3-13 NSSS SUPPLIED ACTIVE CLASS 1, 2, AND 3 VALVES

Ebasco Tag Number	Westinghouse Tag Number	System	Location	Environmental Qualification	Type	Operator	Manufacturer	Safety Class	Valve Design Rating	System Operating Conditions	Size	Function
2CS-V518SB	1-8152	CS	RAB	(6)	Globe	Air	Copes Vulcan	2	600	600 psig 382 F	3	Containment Isolation
2CS-V609SB	1-8108	CS	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2670 psig 170 F	3	ECCS Operation
1CS-V510SB-1	1-8153	CS	RCB	-	Globe	Air	Copes Vulcan	1	1500	2485 psig 589 F	1	ESF
1CS-V509SB-1	1-8154	CS	RCB	-	Globe	Air	Copes Vulcan	1	1500	2485 psig 589 F	1	ESF
2CS-V585SA	1-8106	CS	RAB	(6)	Gate	Motor	Westinghouse	2	1500	1350 psig 130 F	3	ECCS Operation
2CS-V612SA	1-8926	CS	RAB	-	Check	ΔP	Westinghouse	2	150	220 psig 200 F	8	ECCS Operation
2CS-V605SA V606SB	1-8133A,B	CS	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2670 psig 170 F	4	ECCS Operation
2CS-V603SA V604SB	1-8132A,B	CS	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2670 psig 170 F	4	ECCS Operation
2CS-V594SA V596SB V595SAB	1-8481A,B,C	CS	RAB	-	Check	ΔP	Westinghouse	2	1500	2670 psig 170 F	3	ECCS Operation
2CS-V600SB V602SB V601SAB	1-8109A,B,C	CS	RAB	(6)	Globe	Motor	Velan	2	1500	2350 psig 130 F	2	ECCS Operation
2CS-L520SA-1 L521SB-1	1-LCV-115C,E	CS	RAB	(6)	Gate	Motor	Westinghouse	2	150	75 psig 250 F	4	ECCS Operation
2CS-V583SN	1-8440	CS	RAB	-	Check	ΔP	Westinghouse	2	150	15 psig 115 F	4	ECCS Operation
2CS-L523SA-1 L522SB-1	1-LCV-115B,D	CS	RAB	(6)	Gate	Motor	Westinghouse	2	150	15 psig 115 F	8	ECCS Operation
2CS-V586SB	1-8104	CS	RAB	(6)	Globe	Motor	Velan	2	1500	15 psig 150 F	2	Safe Shutdown
2CS-V778SA	1-8453A	CS	RAB		Check	ΔP	Westinghouse	2	150	15 psig 115 F	8	ECCS Operation
2CS-V779SB	1-8453B	CS	RAB		Check	ΔP	Westinghouse	2	150	15 psig 115 F	8	ECCS Operation

TABLE 3.9.3-13 NSSS SUPPLIED ACTIVE CLASS 1, 2, AND 3 VALVES

Ebasco Tag Number	Westinghouse Tag Number	System	Location	Environmental Qualification	Type	Operator	Manufacturer	Safety Class	Valve Design Rating	System Operating Conditions	Size	Function
2CS-V511SA V512SA V513SA	1-8149A,B,C	CS	RCB	(2)	Globe	Air	Copes Vulcan	2	1500	600 psig 382 F	2	Containment Isolation
1RH-V503SA	1-8701A	RH	RCB	(1)	Gate	Motor	Westinghouse	1	1500	2235 psig 620 F	12	Containment Isolation
1RH-V502SB	1-8702A	RH	RCB	(1)	Gate	Motor	Westinghouse	1	1500	2235 psig 620 F	12	Containment Isolation
1RH-V501SA	1-8701B	RH	RCB	(1)	Gate	Motor	Westinghouse	1	1500	2235 psig 620 F	12	Containment Isolation
1RH-V500SB	1-8702B	RH	RCB	(1)	Gate	Motor	Westinghouse	1	1500	2235 psig 620 F	12	Containment Isolation
2CS-V522SB-1	1-8102A	CS	RAB	-	Globe	Motor	Velan	2	1500	2340 psig 130 F	1 1/2	Containment Isolation
2CS-V523SB-1	1-8102B	CS	RAB	-	Globe	Motor	Velan	2	1500	2340 psig 130 F	1 1/2	Containment Isolation
2CS-V524SB-1	1-8102C	CS	RAB	-	Globe	Motor	Velan	2	1500	2340 psig 130 F	1 1/2	Containment Isolation
2RH-V509SA V508SB	1-8716A,B	RH	RAB	-	Check	ΔP	Westinghouse	2	300	600 psig 350 F	10	ECCS Operation
2RH-V506SB	1-8706B	RH	RAB	-	Gate	Motor	Westinghouse	2	300	600 psig 350 F	8	Accident Mitigation
2RH-V507SA	1-8706A	RH	RAB	-	Gate	Motor	Westinghouse	2	300	600 psig 350 F	8	Accident Mitigation
1SI-V507SA V508SB V509SA	1-8998A,B,C	SI	RCB	-	Check	ΔP	Westinghouse	1	1500	2485 psig 350 F	6	ECCS Operation
2SI-V506SA V505SB	1-8801A,B	SI	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2635 psig 300 F	3	ECCS Operation
2SI-V502SA	1-8885	SI	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2635 psig 300 F	3	ECCS Operation
2SI-V501SB	1-8886	SI	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2635 psig 300 F	3	ECCS Operation
2SI-V500SA	1-8884	SI	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2635 psig 300 F	3	ECCS Operation
2SI-V554SB	1-8860	SI	RCB	-	Diaphragm	Air	Copes Vulcan	2	1500	660 psig 120 F	1	Containment Isolation

TABLE 3.9.3-13 NSSS SUPPLIED ACTIVE CLASS 1, 2, AND 3 VALVES

Ebasco Tag Number	Westinghouse Tag Number	System	Location	Environmental Qualification	Type	Operator	Manufacturer	Safety Class	Valve Design Rating	System Operating Conditions	Size	Function
2SI-V555SA	1-8871	SI	RCB	-	Diaphragm	Air	Copes Vulcan	2	1500	2485 psig AMB F	3/4	Containment Isolation
2SI-V530SB	1-8880	SI	RCB	-	Diaphragm	Air	Copes Vulcan	2	600	700 psig 120 F	1	Containment Isolation
1SI-V512SA-1 V513SB-1 V514SA-1	1-8993A,B,C	SI	RCB	-	Check	DP	Westinghouse	1	1500	2485 psig 350 F	6	ECCS Operation
1SI-V544SA-1 V545SB-1 V546SA-1	1-8956A,B,C	SI	RCB	-	Check	DP	Westinghouse	1	1500	2485 psig AMB	12	ECCS Operation
1SI-V547SA-1 V548SB-1 V549SA-1	1-8948A,B,C	SI	RCB	-	Check	DP	Westinghouse	1	1500	2485 psig AMB	12	ECCS Operation
1SI-V584SA-1 V585SB-1 V586SA-1	1-8973A,B,C	SI	RCB	-	Check	ΔP	Westinghouse	1	1500	2485 psig 359 F	6	ECCS Operation
2RH-F512SB F513SA	FCV-602A&B	RH	RAB	-	Flo-Ctrl.	Motor	Velan	2	600	600 psig 350 F	3	Normal Operation
2SI-V581SA V580SB	1-8974A,B	SI	RCB	-	Check	ΔP	Westinghouse	2	1500	2485 psig 350 F	10	Containment Isolation
2SI-V579SA V578SB	1-8888A,B	SI	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2485 psig 350 F	10	ECCS Operation
2SI-V577SA V576SB	1-8887A,B	SI	RAB	(6)	Gate	Motor	Westinghouse	2	600	535 psig 350 F	10	ECCS Operation
1SI-V510SA V511SB	1-8988A,B	SI	RCB	-	Check	ΔP	Westinghouse	1	1500	2485 psig 350 F	6	Containment Isolation
2SI-V587SA	1-8889	SI	RAB	(6)	Gate	Motor	Westinghouse	2	1500	2485 psig 350 F	10	ECCS Operation
2SI-V571SA V570SB	1-8811A,B	SI	RCB	(6)	Gate	Motor	Westinghouse	2	300	400 psig 350 F	14	Containment Isolation
2SI-V573SA V572SB	1-8812A,B	SI	RAB	(6)	Gate	Motor	Westinghouse	2	300	400 psig 350 F	14	ECCS Operation
2SI-V575SA V574SB	1-8809A,B	SI	RAB	(6)	Gate	Motor	Westinghouse	2	300	400 psig 350 F	14	ECCS Operation

TABLE 3.9.3-13 NSSS SUPPLIED ACTIVE CLASS 1, 2, AND 3 VALVES

Ebasco Tag Number	Westinghouse Tag Number	System	Location	Environmental Qualification	Type	Operator	Manufacturer	Safety Class	Valve Design Rating	System Operating Conditions	Size	Function
2SI-V591SA V590SB	1-8958A,B	SI	RAB	-	Check	ΔP	Westinghouse	2	300	400 psig 350 F	14	ECCS Operation
2CS-V757SA V759SB	8490A,B	CS	RAB	-	Globe	Motor	Velan	2	1500	2350 psig 130 F	2	ECCS Operation
2CS-V758SB V760SA	8489A,B	CS	RAB	-	Globe	Motor	Velan	2	1500	2350 psig 130 F	2	ECCS Operation
2SI-V550SB	1-8961	SI	RAB	-	Globe	Air	Copes Vulcan	2	1500	2485 psig AMB F	3/4	Containment Isolation
3CC-B5SA	1-9370	CC	RAB	(6)	Butterfly	Motor	Contintal	3	150	108 psig 112 F	18	ECCS Operation
3CC-B6SB	1-9371	CC	RAB	(6)	Butterfly	Motor	Contintal	3	150	108 psig 112 F	18	ECCS Operation
3CC-B19SA	1-9384	CC	RAB	(6)	Butterfly	Motor	Contintal	3	150	108 psig 105 F	18	ECCS Operation
3CC-B20SB	1-9385	CC	RAB	(6)	Butterfly	Motor	Contintal	3	150	108 psig 105 F	18	ECCS Operation
2CC-V169SA V170SB	1-9480A,B	CC	RAB	(6)	Gate	Motor	Velan	2	150	108 psig 105 F	6	Isolation
2CC-V171SN	1-9500	CC	RAB	-	Check	ΔP	Velan	2	150	108 psig 105 F	6	Containment Isolation
2CC-V172SB	1-9485	CC	RAB	(6)	Gate	Motor	Velan	2	150	108 psig 105 F	6	Containment Isolation
2CC-V182SB	1-9486	CC	RAB	(6)	Gate	Motor	Velan	2	150	108 psig 125 F	6	Containment Isolation
2CC-V173SN	1-9504	CC	RCB	-	Check	ΔP	Velan	2	150	108 psig 105 F	6	Isolation
2CC-V183SB	1-9482	CC	RAB	(6)	Gate	Motor	Velan	2	150	108 psig 115 F	6	Containment Isolation
2CC-V184SA	1-9481	CC	RCB	(1)	Gate	Motor	Velan	2	150	108 psig 115 F	6	Containment Isolation
2CC-V190SB	1-9484	CC	RAB	(6)	Gate	Motor	Velan	2	1500	108 psig 122 F	4	Containment Isolation
3CC-V162SA-1 V164SB-1 V163SAB-1	1-9390A,B,C	CC	RAB	-	Check	ΔP	Velan	2	150	108 psig 116 F	18	ECCS Operation

TABLE 3.9.3-13 NSSS SUPPLIED ACTIVE CLASS 1, 2, AND 3 VALVES

Ebasco Tag Number	Westinghouse Tag Number	System	Location	Environmental Qualification	Type	Operator	Manufacturer	Safety Class	Valve Design Rating	System Operating Conditions	Size	Function
2WL-D650SB-1	1-7136	WL	RAB	(6)	Diaphragm	Air	Grinnel	2	150	125 psig 130 F	3	Containment Isolation
2WL-L600SA-1	1-LCV-1003	WL	RCB	(2)	Globe	Air	Copes Vulcan	2	150	125 psig 130 F	3	Containment Isolation
2MS-V1SAB	CQL-MSVASV-1	MS	RAB	(5)	Globe	Piston	Rockwell	2	900	1091 psig 557 F	34	M.S.I.V.
2MS-V2SAB	CQL-MSVASV-1	MS	RAB	(5)	Globe	Piston	Rockwell	2	900	1091 psig 557 F	34	M.S.I.V.
2MS-V3SAB	CQL-MSVASV-1	MS	RAB	(5)	Globe	Piston	Rockwell	2	900	1091 psia 557 F	34	M.S.I.V.
2CC-F2SN	1-FCV-685	CC	RAB	-	Gate	Motor	Velan	2	1500	108 psig 122 F	4	ESF
3CC-L1SA-1	1-LCV-670	CC	RAB	-	Diaphragm	Air	Grinnel	3	150	108 psig 122 F	3/4	ESF
3CC-L2SA-1	1-LCV-676	CC	RAB	-	Diaphragm	Air	Grinnel	3	150	108 psig 122 F	3/4	ESF
2CC-V191SA	1-9483	CC	RCB	(1)	Gate	Motor	Velan	2	1500	108 psig 122 F	4	Containment Isolation
3CC-V165SA	1-9431A	CC	RAB	(6)	Gate	Motor	Velan	3	150	108 psig 145 F	12	ECCS Operation
3CC-V167SB	1-9431B	CC	RAB	(6)	Gate	Motor	Velan	3	150	108 psig 145 F	12	ECCS Operation
2WG-D590SA-1	1-7126	WG	RCB	-	Diaphragm	Air	Grinnel	2	150	2 psig 100 F	3/4	Containment Isolation
2WG-D291SB-1	1-7150	WG	RCB	-	Diaphragm	Air	Grinnel	2	150	2 psig 100 F	3/4	Containment Isolation
3CC-D547SA-1	-	CC	RAB	-	Diaphragm	Air	ITT-Grinnel	3	150	108 psig 105 F	4	ECCS Operation
3CC-D548SB-1	-	CC	RAB	-	Diaphragm	Air	ITT-Grinnel	3	150	108 psig 105 F	4	ECCS Operation
2CC-R5SN	1-9513	CC	RCB		Relief	Self Actuated	Crosby	2	150	108 psig 145 F	3/4	Containment Isolation
2CC-R6SN	1-9512	CC	RCB		Relief	Self Actuated	Crosby	2	150	108 psig 145 F	3/4	Containment Isolation

NOTE: This is historical information. See Section 3.9.3.2.1.

Notes: TABLE 3.9.3-13

System Designations

RC - Reactor Coolant System

CS - Chemical and Volume Control System

RH - Residual Heat Removal System

SI - Safety Injection System

CC - Component Cooling Water System

WL - Liquid Waste Processing System

MS - Main Steam System

Building Designations

RCB - Reactor Containment Building

RAB - Reactor Auxiliary Building

Environmental Conditions

(1) The operators on these motor operated valves are environmentally qualified to the program outlined in Westinghouse letter NS-CE-692, (C. Eicheldinger to V. B. Vassallo) dated 7/10/75.
(2) The qualification of the operators on these air operated valves is accomplished via failure mode and effects analysis which has demonstrated that these valves fail in a safe position. This program is outlined in Westinghouse letter NS-CE-755, dated 8/15/75.
(3) These valves inside RAB are qualified to meet temperature ranges stated in Section 9.4 for indoor application.
(4) These valves inside RCB are qualified to environmental conditions stated in Section 3.11.
(5) Although located in enclosed structures, these valves inside the intake structure, Diesel Generator Building and Fuel Oil Storage Tank Building may be exposed to, and are qualified to, the outdoor temperature range in Section 9.4 for outdoor application. Heat tracing is provided when necessary for the process application.
(6) These valves are located outside RAB, and qualified to the outdoor temperature range stated in Section 9.4 for outdoor applications.
(7) These valves inside RCB are qualified to environmental conditions expected for a Steam Generator Tube Rupture event.

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3AF-F1SA	AF	RAB	(3)	Flow Control Globe	Electro/ Hydraulic	Masoneilan	3	900	1600 psig @ 125 F	3	ESF Operation
3AF-F2SA	AF	RAB	(3)	Flow Control Globe	Electro/ Hydraulic	Masoneilan	3	900	1600 psig @ 125 F	3	ESF Operation
3AF-F3SA	AF	RAB	(3)	Flow Control Globe	Electro/ Hydraulic	Masoneilan	3	900	1600 psig @ 125 F	3	ESF Operation
3AF-F4SB	AF	RAB	(3)	Flow Control Globe	Electro/ Hydraulic	ITT/Hammel Dahl	3	900	1600 psig @ 125 F	3	ESF Operation
3AF-F5SB	AF	RAB	(3)	Flow Control Globe	Electro/ Hydraulic	ITT/Hammel Dahl	3	900	1600 psig @ 125 F	3	ESF Operation
3AF-F6SB	AF	RAB	(3)	Flow Control Globe	Electro/ Hydraulic	ITT/Hammel Dahl	3	900	1600 psig @ 125 F	3	ESF Operation
3AF-P1SA	AF	RAB	(3)	Press. Control Globe	Electro/ Hydraulic	ITT/Hammel Dahl	3	900	1600 psig @ 125 F	4	ESF Operation
3AF-P2SB	AF	RAB	(3)	Press. Control Globe	Electro/ Hydraulic	ITT/Hammel Dahl	3	900	1600 psig @ 125 F	4	ESF Operation
3AF-V1SA	AF	RAB	(3)	Check	ΔP	Pacific	3	900	1600 psig @ 125 F	4	ESF Operation
3AF-V2SB	AF	RAB	(3)	Check	ΔP	Pacific	3	900	1600 psig @ 125 F	4	ESF Operation
3AF-V3SAB	AF	RAB	(3)	Check	DP	Pacific	3	900	1600 psig @ 125 F	6	ESF Operation
3AF-V8SA	AF	RAB	(3)	Check	DP	Pacific	3	900	1600 psig @ 450 F	4	ESF Operation
2AF-V10SAB	AF	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1600 psig @ 450 F	4	Containment Isolation
2AF-V19SB	AF	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1600 psig @ 450 F	4	Containment Isolation
3AF-V17SA	AF	RAB	(3)	Check	DP	Pacific	3	900	1600 psig @ 450 F	4	ESF Operation
3AF-V21SA	AF	RAB	(3)	Check	DP	Pacific	3	900	1600 psig @ 450 F	4	ESF Operation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2AF-V23SB	AF	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1600 psig @ 450 F	4	Containment Isolation
2AF-V166-SAB-1	AF	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1300 psig @ 450 F	1	Containment Isolation
2AF-V162-SAB-1	AF	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1300 psig @ 450 F	1	Containment Isolation
2AF-V163-SAB-1	AF	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1300 psig @ 450 F	1	Containment Isolation
2AF-V165-SAB	AF	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1300 psig @ 450 F	1	Containment Isolation
2AF-V167-SAB	AF	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1300 psig @ 450 F	1	Containment Isolation
2AF-V164-SAB	AF	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1300 psig @ 450 F	1	Containment Isolation
3AF-V221SA-1	AF	RAB	(3)	Check	WP	Anchor-Darling	3	1500	1700 psig @ 450°F	4	ESF Operation
3AF-V222SA-1	AF	RAB	(3)	Check	WP	Anchor-Darling	3	1500	1700 psig @ 450°F	4	ESF Operation
3AF-V223SA-1	AF	RAB	(3)	Check	WP	Anchor-Darling	3	1500	1700 psig @ 450°F	4	ESF Operation
3AF-V224SB-1	AF	RAB	(3)	Check	WP	Anchor-Darling	3	1500	1600 psig @ 450°F	4	ESF Operation
3AF-V225SB-1	AF	RAB	(3)	Check	WP	Anchor-Darling	3	1500	1600 psig @ 450°F	4	ESF Operation
3AF-V226SB-1	AF	RAB	(3)	Check	WP	Anchor-Darling	3	1500	1600 psig @ 450°F	4	ESF Operation
3AF-V34SB	AF	RAB	(3)	Check	ΔP	Pacific	3	900	1600 psig @ 450 F	4	ESF Operation
3AF-V37SB	AF	RAB	(3)	Check	ΔP	Pacific	3	900	1600 psig @ 450 F	4	ESF Operation
2AF-V116SA	AF	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1600 psig @ 450 F	4	Containment Isolation
2AF-V117SA	AF	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1600 psig @ 450 F	4	Containment Isolation
2AF-V118SA	AF	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1600 psig @ 450 F	4	Containment Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3AF-V31SB	AF	RAB	(3)	Check	ΔP	Pacific	3	900	1600 psig @ 450 F	4	ESF Operation
2AF-V153SAB	AF	RCB	(4)	Check	ΔP	Anchor-Darling	2	900	1185 psig @ 600 F	6	ESF Operation
2AF-V154SAB	AF	RCB	(4)	Check	ΔP	Anchor-Darling	2	900	1185 psig @ 600 F	6	ESF Operation
2AF-V155SAB	AF	RCB	(4)	Check	ΔP	Anchor-Darling	2	900	1185 psig @ 600 F	6	ESF Operation
3CE-V41SA	CE	RAB	(3)	Check	ΔP	TRW-Mission	3	150	150 psig @ 140 F	6	ESF Operation
3CE-V42SB	CE	RAB	(3)	Check	ΔP	TRW-Mission	3	150	150 psig @ 140 F	6	ESF Operation
3CE-V43SAB	CE	RAB	(3)	Check	ΔP	TRW-Mission	3	150	150 psig @ 140 F	8	ESF Operation
2CT-V2SA	CT	RAB	(3)	Gate	Motor	Anchor-Darling	2	150	150 psig @ 300 F	12	ECCS Operation
2CT-V3SB	CT	RAB	(3)	Gate	Motor	Anchor-Darling	2	150	150 psig @ 300 F	12	ECCS Operation
2CT-V4SA	CT	RAB	(3)	Check	ΔP	Anchor-Darling	2	150	150 psig @ 300 F	12	ECCS Operation
2CT-V5SB	CT	RAB	(3)	Check	ΔP	Anchor-Darling	2	150	150 psig @ 300 F	12	ECCS Operation
2CT-V6SA	CT	RAB	(3)	Gate	Motor	Anchor-Darling	2	150	45 psig @ 300 F	12	Containment Isolation
2CT-V7SB	CT	RAB	(3)	Gate	Motor	Anchor-Darling	2	150	45 psig @ 300 F	12	Containment Isolation
2CT-V13SA	CT	RAB	(3)	Check	ΔP	Rockwell	2	1500	50 psig @ 200 F	2	ECCS Operation
2CT-V21SA	CT	RAB	(3)	Gate	Motor	Anchor-Darling	2	300	300 psig @ 300 F	8	ECCS Operation
2CT-V27SA	CT	RCB	(4)	Check	ΔP	Anchor-Darling	2	300	300 psig @ 300 F	8	ECCS Operation
2CT-V35SB	CT	RAB	(3)	Check	ΔP	Rockwell	2	1500	50 psig @ 200 F	2	ECCS Operation
2CT-V43SB	CT	RAB	(3)	Gate	Motor	Anchor-Darling	2	300	300 psig @ 300 F	8	ECCS Operation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2CT-V51SB	CT	RCB	(4)	Check	ΔP	Anchor-Darling	2	300	300 psig @ 300 F	8	ECCS Operation
3CT-V85SA	CT	RAB	(3)	Globe	Motor	Yarway	3	1500	15 psig @ 200 F	2	ECCS Operation
3CT-R1SAB	CT	RAB	(3)	Safety	S-A	Crosby	3	150	15 psig @ 200 F	1x1 ¼	Protect ECCs
3CT-X3SAB-1	CT	RAB	(3)	Vacuum Breaker	-	Anderson-	3	150	15 psig @ 200 F	2	ECCS Operation
3CT-X4SAB-1	CT	RAB	(3)	Vacuum Breaker	-	Anderson-	3	150	15 psig @ 200 F	2	ECCS Operation
3CT-V88SB	CT	RAB	(3)	Globe	Motor	Yarway	3	1500	15 psig @ 200 F	2	ECCS Operation
3FO-V23SA	FO	FOST	(5)	Check	ΔP	Rockwell	3	Note 1	100 psig @ 125 F	2	ESF Operation
3FO-V258SA	FO	FOST	(5)	Globe	Hand	Yarway	3	1500	100 psig @ 125 F	2	ESF Operation
3FO-V24SB	FO	FOST	(5)	Check	ΔP	Rockwell	3	Note 1	100 psig @ 125 F	2	ESF Operation
3FO-V259SB	FO	FOST	(5)	Globe	Hand	Yarway	3	1500	100 psig @ 125 F	2	ESF Operation
2BD-V11SA	BD	RAB	(3)	Globe	Air-Piston	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	4	Containment Isolation
2BD-V15SA	BD	RAB	(3)	Globe	Air-Piston	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	4	Containment Isolation
2BD-P6SB-1	BD	RCB	(4)	Globe	Air Piston	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	4	Isolation
2BD-P7SB-1	BD	RCB	(4)	Globe	Piston	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	4	Isolation
2BD-P8SB-1	BD	RCB	(4)	Globe	Piston	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	4	Isolation
21A-V33SN	IA	RCB	(4)	Check	ΔP	Anchor-Darling	2	150	125 psig @ 125 F	3	Containment Isolation
21A-V192SA	IA	RAB	(3)	Globe	Diaphragm	Copes-Vulcan	2	600	125 psig @ 125 F	3	Containment Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2MS-P18SA	MS	RAB	(3)	Press. Control Globe	Electro-Hyd	Control Comps	2	900	1185 psig @ 600 F	8x10	ESF Operation
2MS-P19SB	MS	RAB	(3)	Press. Control Globe	Elect-Hyd	Control Comps	2	900	1185 psig @ 600 F	8x10	ESF Operation
2MS-P20SA	MS	RAB	(3)	Press. Control Globe	Elect-Hyd.	Control Comps	2	900	1185 psig @ 600 F	8x10	ESF Operation
2MS-R1SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R2SB	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R3SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R4SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R5SB	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R6SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R7SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R8SB	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R9SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R10SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R11SB	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R12SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R13SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2MS-R14SB	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-R15SA	MS	RAB	(3)	Relief Safety	S-A	Crosby	2	900	1185 psig @ 600 F	6x10	ESF Operation
2MS-V8SA	MS	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1185 psig @ 600 F	6	ESF Operation
2MS-F1SAB	MS	RAB	(3)	Flow Control	Diaphragm	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	3	Containment Isolation
2MS-F2SAB	MS	RAB	(3)	Flow Control	Diaphragm	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	3	Containment Isolation
2MS-F3SAB	MS	RAB	(3)	Flow Control	Diaphragm	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	3	Containment Isolation
2MS-V59SAB	MS	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	2	Containment Isolation
2MS-V60SAB	MS	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	2	Containment Isolation
2MS-V61SAB	MS	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	900	1185 psig @ 600 F	2	Containment Isolation
2MS-V9SB	MS	RAB	(3)	Gate	Motor	Anchor-Darling	2	900	1185 psig @ 600 F	6	ESF Operation
3MS-V99SA	MS	RAB	(3)	Check	ΔP	Pacific	2	900	1185 psig @ 600 F	6	ESF Operation
3MS-V100SB	MS	RAB	(3)	Check	ΔP	Pacific	3	900	1185 psig @ 600 F	6	ESF Operation
2MS-V122SAB	MS	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1185 psig @ 600 F	1	Containment Isolation
2MS-V124SAB	MS	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1185 psig @ 600 F	1	Containment Isolation
2MS-V126SAB	MS	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1185 psig @ 600 F	1	Containment Isolation
2MS-V36SA	MD	RCB	(4)	Gate	Motor	Anchor-Darling	3	150	60 psig @ 125 F	3	Containment Isolation
2MD-V77SB	MD	RAB	(3)	Gate	Motor	Anchor-Darling	2	150	60 psig @ 125 F	3	Containment Isolation
3SC-V48SA	SC	SCRN STR	(5)	Globe	EH	ITT/Hammel Dahl	3	150	200 psig @ 95 F	3	ESF Operation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3SC-V15SA	SC	ESWIS	(5)	Globe	EH	ITT/Hammel Dahl	3	150	200 psig @ 95 F	3	ESF Operation
3SC-V26SA	SC	SCRN STR	(5)	Globe	EH	ITT/Hammel Dahl	3	150	200 psig @ 95 F	3	ESF Operation
3SC-V30SB	SC	ESWIS	(5)	Globe	EH	ITT/Hammel Dahl	3	150	200 psig @ 95 F	3	ESF Operation
3SC-V31SB	SC	SCRN STR	(5)	Globe	EH	ITT/Hammel Dahl	3	150	200 psig @ 95 F	3	ESF Operation
2SP-V23SA-1	SP	RAB	(3)	Globe	Solenoid	Target Rock	2	2500	2485 psig @ 650 F	3/8	Containment Isolation
2SP-V11SB-1	SP	RAB	(4)	Globe	Solenoid	Target Rock	2	2500	2485 psig @ 650 F	3/8	Containment Isolation
2SP-V12SA-1	SP	RAB	(3)	Globe	Solenoid	Target Rock	2	2500	2485 psig @ 650 F	3/8	Containment Isolation
2SP-V111SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	2500	2485 psig @ 650 F	3/8	Containment Isolation
2SP-V113SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	700 psig @ 300 F	3/8	Containment Isolation
2SP-V114SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	700 psig @ 300 F	3/8	Containment Isolation
2SP-V115SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	700 psig @ 300 F	3/8	Containment Isolation
2SP-V116SA-1	SP	RAB	(3)	Globe	Solenoid	Target Rock	2	1500	700 psig @ 300 F	3/8	Containment Isolation
2SP-V2SA-1	SP	RAB	(3)	Globe	Solenoid	Target Rock	2	2500	2485 psig @ 680 F	3/8	Containment Isolation
2SP-V1SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	2500	2485 psig @ 680 F	3/8	Containment Isolation
2SP-V90SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/4	Isolation
2SP-V91SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/4	Isolation
2SP-V120SA-1	SP	RAB	(3)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/8	Containment Isolation
2SP-V121SA-1	SP	RAB	(3)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/8	Containment Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2SP-V75SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/4	Isolation
2SP-V85SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/4	Isolation
2SP-V81SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/4	Isolation
2SP-V80SB-1	SP	RCB	(4)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/4	Isolation
2SP-V122SA-1	SP	RAB	(3)	Globe	Solenoid	Target Rock	2	1500	1185 psig @ 600 F	3/8	Containment Isolation
3SW-B5SA	SW	RAB	(3)	Butterfly	Motor	Anchor/Darling	3	150	150 psig @ 140°F	30	ESF Operation
3SW-B6SB	SW	RAB	(3)	Butterfly	Motor	Anchor/Darling	3	150	150 psig @ 140°F	30	ESF Operation
3SW-B8SA	SW	RAB	(3)	Butterfly	Motor	Anchor/Darling	3	150	150 psig @ 140°F	36	ESF Operation
3SW-B13SB	SW	RAB	(3)	Butterfly	Motor	Anchor/Darling	3	150	150 psig @ 140°F	30	ESF Operation
3SW-B14SB	SW	RAB	(3)	Butterfly	Motor	Anchor/Darling	3	150	150 psig @ 140°F	30	ESF Operation
3SW-B15SA	SW	RAB	(3)	Butterfly	Motor	Anchor/Darling	3	150	150 psig @ 140°F	30	ESF Operation
3SW-B16SB	SW	RAB	(3)	Butterfly	Motor	Anchor/Darling	3	150	150 psig @ 140°F	30	ESF Operation
3SW-B64SA	SW	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	225 psig @ 195 F	14	ESF Operation
3SW-B65SB	SW	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	225 psig @ 195 F	14	ESF Operation
3SW-B70SA	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	8	ESF Operation
3SW-B71SA	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	8	ESF Operation
3SW-B72SB	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	8	ESF Operation
3SW-B73SB	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	8	ESF Operation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3SW-B75SA	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	6	ESF Operation
3SW-B77SB	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	6	ESF Operation
3SW-B74SA	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	6	ESF Operation
3SW-B76SB	SW	RAB	(3)	Butterfly	Motor	Jamesbury	3	150	150 psig @ 140 F	6	ESF Operation
2SW-V142SN	SW	RCB	(4)	Check	ΔP	Anchor-Darling	2	150	225 psig @ 140 F	12	Containment Isolation
2SW-B88SAB	SW	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	2	150	225 psig @ 140 F	12	Containment Isolation
2SW-B89SA	SW	RCB	(4)	Butterfly	Diaphragm	ITT/Hammel Dahl	2	150	225 psig @ 140 F	12	Containment Isolation
2SW-B90SB	SW	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	2	150	225 psig @ 140 F	12	Containment Isolation
3SW-V39SA	SW	RAB	(3)	Check	ΔP	Atwood & Morrill	3	150	225 psig @ 140 F	14	ESF Operation
3SW-V41SB	SW	RAB	(3)	Check	ΔP	Atwood & Morrill	3	150	225 psig @ 140 F	14	ESF Operation
3SW-V540SA	SW	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @ 140 F	1 1/2	ECCS Operation
3SW-V543SB	SW	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @ 140 F	1 1/2	ECCS Operation
3SW-V542SA	SW	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @ 140 F	1 1/2	ECCS Operation
3SW-V544SB	SW	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @ 140 F	1 1/2	ECCS Operation
3SW-V367SA	SW	RAB	(3)	Check	ΔP	Atwood & Morrill	3	150	150 psig @ 140 F	30	ESF Operation
3SW-V368SB	SW	RAB	(3)	Check	ΔP	Atwood & Morrill	3	150	150 psig @ 140 F	30	ESF Operation
2FW-V26SAB	FW	RAB	(3)	Gate	Piston	Hiller	2	921	1860 psig @ 450 F	16	Containment Isolation
2FW-V27SAB	FW	RAB	(3)	Gate	Piston	Hiller	2	921	1860 psig @ 450 F	16	Containment Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2FW-V28SAB	FW	RAB	(3)	Gate	Piston	Borg Warner Hiller	2	921	1860 psig @ 450 F	16	Containment Isolation
2FW-V23SN	FW	RAB	(3)	Check	ΔP	Borg Warner	2	921	1860 psig @ 450 F	16	Isolation
2FW-V24SN	FW	RAB	(3)	Check	ΔP	Borg Warner	2	921	1860 psig @ 450 F	16	Isolation
2FW-V25SN	FW	RAB	(3)	Check	ΔP	Borg Warner	2	921	1860 psig @ 450 F	16	Isolation
2FW-V93SAB-1	FW	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1300 psig @ 450 F	1	Containment Isolation
2FW-V94SAB-1	FW	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1300 psig @ 450 F	1	Containment Isolation
2FW-V91SAB-1	FW	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1300 psig @ 450 F	1	Containment Isolation
2FW-V92SAB-1	FW	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1300 psig @ 450 F	1	Containment Isolation
2FW-V89SAB-1	FW	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1300 psig @ 450 F	1	Containment Isolation
2FW-V90SAB-1	FW	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	2	1500	1300 psig @ 450 F	1	Containment Isolation
2FP-B1SA	FP	RAB	(3)	Butterfly	Diaphragm	Jamesburry	2	150	175 psig @ 125 F	6	Containment Isolation
3CH-B1SB	ESCWS Supply	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CH-B3SA	ESCWS Supply	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CX-B1SB	ESCWS Return	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CX-B4SA	ESCWS Return	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3SW-V821SB	SW	RAB	(3)	Wafer Check	ΔP	TRW-Mission	3	150	150 psig @ 140 F	8	Prevent Backflow
3CX-V243SB	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2 1/2	Temperature Control
3CX-W17SA	ESCWS Return	RAB	(3)	Three-Way Globe	Electro-Hydraulic	ITT/Hammel Dahl	3	150	150 psig @ 125 F	1 1/2	Temperature Control

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3CX-W24SB	ESCWS Return	RAB	(3)	Three-Way	Electro-Hydraulic	ITT/Hammel Dahl	3	150	150 psig @ 125 F	1 1/2	Temperature Control
3AV-B1SA	Emer. Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 124 F	20	Open-Close
3AV-B2SA	Emer. Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 124 F	20	Open-Close
3AV-B3SB	Emer. Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 124 F	6	Open-Close
3AV-B4SB	Emer. Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 124 F	20	Open-Close
3AV-B5SB	Emer. Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 124 F	20	Open-Close
3AV-B6SA	Emer. Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 124 F	6	Open-Close
3CZ-B1SA	CR O.A. Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	3 psig @ 95 F	16	Isolation
3CZ-B2SB	CR O.A. Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	3 psig @ 95 F	16	Isolation
3CZ-B3SA	CR Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 75 F	12	Isolation
3CZ-B4SB	CR Exh. Sys.	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 75 F	12	Isolation
3CZ-B5SA	Equip. Protect Rm OA Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 95 F	12	Isolation
3CZ-B6SB	Equip. Protect Rm OA Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 95 F	12	Isolation
3CZ-B7SA	HVAC Equip. Rm Exh. System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 90 F	12	Isolation
3CZ-B8SA	HVAC Equip. Rm Exh. System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 90 F	12	Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3CZ-B9SA	CR Emer. O.A. Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	3 psig @ 95 F	12	Isolation
3CZ-B10SB	CR Emer. O.A. Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	3 psig @ 95 F	12	Isolation
3CZ-B11SA	CR Emer. O.A. Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	3 psig @ 95 F	12	Isolation
3CZ-B12SB	CR Emer. O.A. Intake	RAB	(3)	Butterfly	Motor	BIF	3	150	3 psig @ 95 F	12	Isolation
3CX-V63SB	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2 1/2	Temperature Control
3CX-V83SA	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2 1/2	Temperature Control
3CZ-B19SA	CR Emer. Filtration System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 62 F	20	Open-Close
3CZ-B20SB	CR Emer. Filtration System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 62 F	20	Open-Close
3CZ-B21SA	CR Emer. Filtration System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 62 F	20	Open-Close
3CZ-B22SB	CR Emer. Filtration System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 62 F	20	Open-Close
3CZ-B23SA	CR Emer. Filtration System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 62 F	20	Open-Close
3CZ-B24SB	CR Emer. Filtration System	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 62 F	20	Open-Close
3CZ-B25SA	CR Air Handling Unit	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 80 F	36	Open-Close
3CZ-B26SB	CR Air Handling Unit	RAB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 80 F	36	Open-Close

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3FV-B2SA	FHB Emer. Exh. Sys.	FHB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 120 F	24	Open-Close
3FV-B4SB	FHB Emer. Exh. Sys.	FHB	(3)	Butterfly	Motor	BIF	3	150	1 psig @ 120 F	24	Open-Close
3AV-V3SA	RAB Emer. Exh. Sys.	RAB	(3)	Check Valve	ΔP	Anderson/Greenwood Co.	3	150	150 psig @ 150 F	6	Bleed air for charcoal decay heat cooling
3AV-V4SB	RAB Emer. Exh. Sys.	RAB	(3)	Check Valve	ΔP	Anderson/Greenwood Co.	3	150	150 psig @ 150 F	6	Bleed air for charcoal decay heat cooling
3CZ-V1SA	CR Emer. Filtration System	RAB	(3)	Check Valve	ΔP	Anderson/Greenwood Co.	3	150	150 psig @ 150 F	6	Outside Air Intake
3CZ-V2SB	CR Emer. Filtration System	RAB	(3)	Check Valve	ΔP	Anderson/Greenwood Co.	3	150	150 psig @ 150 F	6	Outside Air Intake
1CS-V711SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary
1CS-V70SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary
2CS-V129SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	220 psig @ 200 F	2	Safe Shutdown
3CS-V222SN	CS	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @ 250 F	2	Safe Shutdown
3CS-V223SN	CS	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @250 F	2	Safe Shutdown
1SI-V39SA V45SB V51SA	SI	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	Containment Isolation
1SI-V63SA V69SB V75SA	SI	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	ContainmentIsolation
2SI-V530SB	SI	RAB	-	Globe	Pneumatic	Copes-Vulcan	2	1500	800 psig @ 200 F	1	Containment Isolation
ISI-V84SA V90SB V96SA	SI	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary, Containment Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
ISI-V17SA V23SB V29SA	SI	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary, Containment Isolation
2CB-B1SA	CB Containment Vacuum Relief	RAB	(3)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	24	Open-Close
2CB-B2SB	CB Containment Vacuum Relief	RAB	(3)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	24	Open-Close
2CP-B1SA	CB Normal Containment Purge Make-up	RCB	(4)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	8	Containment Isolation
2CP-B2SB	CB Normal Normal Containment Purge Make-up	RAB	(3)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	8	Containment Isolation
2CP-B3SA	CB Containment Pre-Entry Purge Make-up	RCB	(3)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	42	Containment Isolation
2CP-B4SA	CB Containment Pre-Entry Purge Make-up	RAB	(3)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	42	Containment Isolation
2CP-B5SA	CB Normal Containment Purge	RCB	(4)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	8	Containment Isolation
2CP-B6SB	CB Normal Containment Purge	RAB	(3)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	8	Containment Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2CP-B7SA	CB Containment Pre-Entry Purge	RCB	(4)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	42	Containment Isolation
2CP-B8SB	CB Containment Pre-Entry Purge	RAB	(3)	Butterfly	Pneumatic	BIF	2	150	45 psig @ 366 F	42	Containment Isolation
2CB-V1SA	Vacuum Relief System	RCB	(4)	HVAC Check Valves	None	Anderson/Greenwood Co.	2	150	150 psig @ 366 F	24	Vacuum Relief
2CB-V2SB	Vacuum Relief System	RCB	(4)	HVAC Check Valves	None	Anderson/ Greenwood Co.	2	150	150 psig @ 366 F	24	Vacuum Relief
1CS-V22SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	1 1/2	Safe Shutdown
1CS-V23SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	1 1/2	Safe Shutdown
1CS-V24SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	1 1/2	Safe Shutdown
2CS-V25SB	CS	RCB	(4)	Check	ΔP	Rockwell	2	1500	2800 psig @ 200 F	1 1/2	Containment Isolation
2CS-V26SB	CS	RCB	(4)	Check	ΔP	Rockwell	2	1500	2800 psig @ 200 F	1 1/2	Containment Isolation
2CS-V27SB	CS	RCB	(4)	Check	ΔP	Rockwell	2	1500	2800 psig @ 200 F	1 1/2	Containment Isolation
1CS-V34SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	1 1/2	Safe Shutdown
1CS-V35SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	1 1/2	Safe Shutdown
1CS-V36SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	1 1/2	Safe Shutdown
2CS-V67SB	CS	RCB	(4)	Check	ΔP	Rockwell	2	1500	150 psig @ 500 F	3/4	Containment Isolation
2SI-V188SA	SI	RCB	(4)	Check	ΔP	Rockwell	2	1500	700 psig @ 300 F	1	Containment Isolation

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2SI-V150SB	SI	RCB	(4)	Check	ΔP	Rockwell	2	1500	2735 psig @ 300 F	1	Containment Isolation
2CC-V51SN	CC	RCB	(4)	Check	ΔP	Rockwell	2	Note 1	150 psig @ 200 F	3/4	Containment Isolation
2CC-V50SN	CC	RCB	(4)	Check	ΔP	Rockwell	2	1500	2485 psig @ 650 F	3/4	Containment Isolation
3CC-V64SN	CC	RCB	(4)	Check	ΔP	Rockwell	3	1500	2485 psig @ 650 F	2	RCS pressure boundary isol.
3CC-V65SN	CC	RCB	(4)	Check	ΔP	Rockwell	3	1500	2485 psig @ 650 F	2	RCS pressure boundary isol.
3CC-V284SN	CC	RCB	(4)	Check	ΔP	Rockwell	3	1500	2485 psig @ 650 F	2	RCS pressure boundary isol.
3CC-V209SN	CC	RAB	(3)	Check	ΔP	Rockwell	3	Note 1	150 psig @ 200 F	3/4	CCWS pressure boundary isol.
3CC-V210SN	CC	RAB	(3)	Check	ΔP	Rockwell	3	Note 1	150 psig @ 200 F	3/4	CCWS pressure boundary isol.
3SW-V868SA-1	SW	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	150 psig @ 140 F	1	ESF Operation
3SW-V869SB-1	SW	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	150 psig @ 140 F	1	ESF Operation
3SW-V870SA-1	SW	RAB	(3)	Check	ΔP	Rockwell	3	Note 1	150 psig @ 140 F	1	ESF Operation
3SW-V871SB-1	SW	RAB	(3)	Check	ΔP	Rockwell	3	Note 1	150 psig @ 140 F	1	ESF Operation
2CS-V136SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	2800 psig @ 200 F	2	ESF Operation (Note 2)
2CS-V137SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	2800 psig @ 200 F	2	ESF Operation (Note 2)
2CS-V138SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	2800 psig @ 200 F	2	ESF Operation (Note 2)
2SP-V448SA-1	SP	RCB	(5)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Det. Rad. Monitor
2SP-V449SB-1	SP	RCB	(3)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Det. Rad. Monitor
2SP-V450SA-1	SP	RCB	(5)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Det. Rad. Monitor

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
2SP-V451SB-1	SP	RCB	(3)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Det. Rad. Monitor
3CH-B2SA-1	ESCWS Supply	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CH-B4SB-1	ESCWS Supply	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CX-B2SA-1	ESCWS Return	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CX-B3SB-1	ESCWS Return	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CX-R1SA-1	ESCWS Return	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	1	Pressure relief
3CX-R2SA-1	ESCWS Return	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	1	Pressure relief
3CX-R3SB-1	ESCWS Return	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	1	Pressure relief
3CX-R4SB-1	ESCWS Return	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	1	Pressure relief
3CX-V121SA-1	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	1	Temperature control
3CX-V122SA	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	600	150 psig @ 125 F	1.5	Temperature control
3CX-V123SA	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	600	150 psig @ 125 F	1.5	Temperature control
3CX-V244SB	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	600	150 psig @ 125 F	1	Temperature control
3CX-V245SB	ESCWS Return	RAB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	600	150 psig @ 125 F	1.5	Temperature control
3CX-V247SB	ESCWS Return	WPB	(3)	Globe	Diaphragm	ITT/Hammel Dahl	3	600	150 psig @ 125 F	1	Temperature control
3CX-W1SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature control
3CX-W2SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature control

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3CX-W3SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature control
3CX-W4SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature control
3CX-W5SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature control
3CX-W7SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	3	Temperature control
3CX-W8SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W9SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	3	Temperature Control
3CX-W10SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	3	Temperature Control
3CX-W12SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W13SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W14SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	3	Temperature Control
3CX-W15SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W16SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	3	Temperature Control
3CX-W18SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	1.5	Temperature Control
3CX-W19SA	ESCWS Return	FHB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W20SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W21SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W22SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3CX-W23SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	3	Temperature Control
3CX-W25SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	1.5	Temperature Control
3CX-W26SB	ESCWS Return	FHB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W27SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W29SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	2.5	Temperature Control
3CX-W32SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	1500	150 psig @ 125 F	1.5	Temperature Control
3CX-W33SB	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	1500	150 psig @ 125 F	1.5	Temperature Control
3CX-W34SA	ESCWS Return	RAB	(3)	Three-Way Globe	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	1.5	Temperature Control
3CX-V2281SB-1	FP	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3CX-V2280SA-1	FP	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3CX-V2283SB-1	FP	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3CX-V2282SA-1	FP	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3SA-V301SA-1	SA	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3SA-V302SB-1	SA	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3SA-V306SB-1	SA	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3SA-V307SA-1	SA	RAB	(3)	Globe	Solenoid	Target-Rock	3	150	125 psig @ 125 F	1	ESF
3SW-B300SA-1	SW	RAB	(3)	Butterfly	Electro-Hydraulic	ITT/Hammel Dahl	3	150	150 psig @ 140 F	10	Temp. Control Modulating

TABLE 3.9.3-14 NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3SW-B303SB-1	SW	RAB	(3)	Butterfly	Electro-Hydraulic	ITT/Hammel Dahl	3	150	150 psig @ 140 F	10	Temp. Control Modulating
3CX-R6SA-1	CSCWS Return	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	3/4	Pressure Relief
3CX-R5SB-1	CSCWS Return	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	3/4	Pressure Relief
3CH-R2SB-1	CSCWS Supply	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	3/4	Pressure Relief
3CH-R1SA-1	CSCWS Supply	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 125 F	3/4	Pressure Relief
3SW-R16SA-1	SW	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 140 F	3/4	Pressure Relief
3SW-R17SB-1	SW	RAB	(3)	Relief	S-A	Crosby	3	150	150 psig @ 140 F	3/4	Pressure Relief
3SW-V800SA-1	SW	RAB	(3)	Water Check	ΔP	TWR/Mission	3	150	150 psig @ 140 F	8	Prevent Backflow
3SI-R525SA-1	SI	RCB	(4)	Relief	S-A	Crosby	3	600	800 psig @ 300 F	1	Pressure Relief
3SI-4526SB-1	SI	RCB	(4)	Relief	S-A	Crosby	3	600	800 psig @ 300 F	1	Pressure Relief

EXPLANATION AND LEGEND FOR TABLE 3.9.3-14

CAT	Ebasco piping category corresponding to ASME Code Class 1, 2, and 3 (also corresponds to ANSI N18.2/N18.2a Safety Class 1, 2, and 3, respectively).
SYS	System as identified on FSAR Table 3.2.1-1.
SYM	Valve symbol which corresponds to valve type as follows: B - Butterfly Valves F - Flow Control Valves P - Pressure Control Valves R - Safety and Relief Valves V - Gate, Globe and Check Valves W - Three Way Valves
TAG NO.	This corresponds to the valve unique number plus safety train designation (e.g., tag no. "100SA" has its valve unique number"100" plus its safety train designation "SA").
NOM SIZE (IN)	Nominal pipe size (in inches) with which the valve is compatible.
VALVE TYPE	Refer to SYM above.
LOC ELEV	Location (building designation) and floor elevation where the equipment is located. The building designation is given by the following: RCB - Reactor Containment Building RAB - Reactor Auxiliary Building MST - Main Steam Tunnel ESWCTST - Emergency Service Water and Cooling Tower Structure SCRN STR Emergency Screen Wash Structure TK - Tank Building FHB - Fuel Handling Building FO - Fuel Oil Storage Tank Structure WPB - Waste Processing Building DGB - Diesel Generator Building ESWISS - Emergency Service Water Intake Screening Structure ESIS -Emergency Service Water (and Cooling Tower Makeup) Intake Structure
DESIGN PRES PSIG	This is the design pressure of the system in pounds per square inch gauge.
DESIGN TEMP(F)	This is the design temperature of the system in degrees Fahrenheit.
OPERATOR	This corresponds to the operator type as follows: D - Diaphragm Operator EH - Electro Hydraulic Operator H - Hand Operator M - Motor Operator P - Piston Operator

EXPLANATION AND LEGEND FOR TABLE 3.93-14 (Continued)

S - Solenoid Operator

PRES STD/ ANSI RATG	This is the pressure rating class of the valve as described in ANSI B16.5.
MANUFACTURER NAME	The name of the manufacturer which supplied the valve listed.
MANUFACTURER MODEL NO.	The model number that is supplied by the manufacturer.
FUNCTION	This is the safety function of the valve specified.

<u>Environmental Conditions</u>

(1) The operators on these motor operated valves are environmentally qualified to the program outlined in Westinghouse letter NS-CE-692 (C. Eicheldinger to V. B. Vassallo) dated 7/10/75.

(2) The qualification of the operators on these air operated valve is accomplished via failure mode and effects analysis which has demonstrated that these valves fail in a safe position. This program is outlined in Westinghouse letter NS-CE-755, dated 8/15/75.

(3) These valves inside RAB, WPB and FHB are qualified to meet temperature ranges stated in Section 9.4 for indoor application.

(4) These valves inside RCB are qualified to environmental conditions stated in Section 3.11.

(5) Although located in enclosed structures, these valves inside the intake structure, Diesel Generator Building, and Fuel Oil Storage Tank Building may be exposed to, and are qualified to, the outdoor temperature range in Section 9.4 for outdoor application. Heat tracing is provided when necessary for the process application.

(6) These valves are located outside RAB, and qualified to the outdoor temperature range stated in Section 9.4 for outdoor applications.

Note 1: Design rating may be 600 or 800 per ESR 95-01002.

TABLE 3.9.3-15

NON-NSSS SUPPLIED ACTIVE PUMPS

Pump	System	ANS Safety Class	Normal Mode	Active Mode	Manufacturer	Function
Containment Spray Pumps	CT	2	Off	On	Ingersoll-Rand	ECCS operation, safe-shutdown
Chilled Water Pumps	CH	3	On	On	Goulds Pump	Supply chilled water to HVAC systems
Auxiliary Feedwater Pumps	AF	3	Off	On	Ingersoll-Rand	Start-up, safe shut-down
Diesel Oil Transfer Pumps	FO	3	Off	On-Off	Goulds Pump	Safe shut-down
Chilled Water Condenser Recirculation Pump	CX	3	Off	On-Off	Goulds Pump	ESF operation
Emergency Screen Wash Pumps	SC	3	Off	On	Crane-Deming	Safe shut-down
Emergency Service Water Pumps	SW	3	Off	On	Ingersoll-Dresser	Safe shut-down
Emergency Service Water Booster Pumps	SW	3	Off	On	Goulds Pump	Safe shut-down

NOTE: This is historical information. See Section 3.9.3.2.1.

EXPLANATION AND LEGEND FOR TABLE 3.9.3-15

SYS — System as identified on FSAR Table 3.2.1-1.

PUMPS NAME — The specific pump name.

TAG NO. — This corresponds to the pump number plus its safety train designation (e.g., tag no. "1A-SA" has a pump number "1A" plus its safety train designation "SA").

LOC/ELEV — Location (building description) and floor elevation where the equipment is located. The building designation is given by the following:

RCB	-	Reactor Containment Building
RAB	-	Reactor Auxiliary Building
MST	-	Main Steam Tunnel
ESWCTST		-Emergency Service Water and Cooling Tower Structure
TK	-	Tank Building
FHB	-	Fuel Handling Building
FO	-	Fuel Oil Storage Tank Structure
WPB	-	Waste Processing Building
DGB	-	Diesel Generator Building

PUMP MANF. AND MODEL — The manufacturer and manufacturer's model number for the particular pump.

DRIVER MANF. AND MODEL — The manufacturer and manufacturer's model number for the particular driver.

RMP — Revolution per minute which given the specified gpm.

GPM — Gallons per minute by design.

HP — Rated horsepower of the pump.

TABLE 3.9.5-1

MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (in.)	No-Loss-of-Function Deflections (in.)
Upper Barrel		
Radial inward	4.1	8.2
Radial outward	1.0	1.0
Upper Package	0.10	0.15
Rod Cluster Guide Tubes	1.00	1.75

TABLE 3.9C.12-1

RESULTS OF STRESS ANALYSIS TYPE Y SOLENOID VALVES

				MIN. WALL THICKNESS		ACTUAL WALL THICKNESS	
MODEL NO.	SECTION MODULUS RATIO	AREA RATIO	ALLOWABLE STRESS RATIO	BODY IN.	NECK IN.	BODY IN.	NECK IN.
-015	1.37	2.25	1.17	0.250	0.277	0.506	0.614
-017	3.32	2.37	0.819	0.450	0.605	0.506	0.614

The section modulus and area ratios are greater than 1.10.

Actual wall thicknesses are greater than the minimum.

MODEL NO.	BONNET COMBINED STRESS, PSI @ I.D.	ALL STRESS PSI
All	17802	24,690

Combined stress is less than the allowable stress.

MODEL NO.	NATURAL FREQUENCY Hz	SEISMIC DEFLECTION σ, IN.	INTERNAL CLEARANCE σ, IN.
All	156	0.0023	0.005

Natural frequency is greater than 33Hz. Deflection is less than clearance.

TABLE 3.9C.12-2

RESULTS OF STRESS ANALYSIS TYPE T SOLENOID VALVES

				MIN. WALL THICKNESS		ACTUAL WALL THICKNESS	
MODEL NO.	SECTION MODULUS RATIO	AREA RATIO	ALLOWABLE STRESS RATIO	BODY IN.	NECK IN.	BODY IN.	NECK IN.
All	11.72	8.13	0.957	0.212	0.399	0.800	0.590

The section modulus and area ratios are greater than 1.10.

Actual wall thicknesses are greater than the minimum.

MODEL NO.	BONNET COMBINED STRESS, PSI @ I.D.	ALL STRESS PSI
All	3626	27,150

Combined stress is less than the allowable stress.

MODEL NO.	NATURAL FREQUENCY Hz	SEISMIC DEFLECTION σ, IN.	INTERNAL CLEARANCE σ, IN.
All	236	0.000997	0.010

Natural frequency is greater than 33Hz. Deflection is less than clearance.

TABLE 3.9C.14-1

STATIC SEISMIC ANALYSIS FREQUENCY AND STRESSES
1" DIAPHRAGM OPERATED GLOBE VALVES

(TAG NUMBERS 2AF V162SAB, 2AF V163SAB, 2AF V164SAB, 2AF V165SAB, 2AF V166SAB, 2AF V167SAB, 2CX V121SA, 2CX V244SB, 2CX V247SB, 2FW V89SAB, 2FW V90SAB,2FW V91SAB, 2FW V92SAB, 2FW V93SAB, 2FW V94SAB, 2MS V122SAB, 2MS V124SAB, 2MS V126SAB)

Description	Actual	Criteria
Natural Frequency	68.4 Hz	> 33 Hz*
Yoke Leg Stress Membrane & Bending (Faulted)	1800 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress Membrane & Bending (Faulted)	7780 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	2024 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	13824 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet Membrane & Bending (Max)	15805 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	8510 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	5.89 in^3	≥(1.1) 0.161 in^3
Area	11.98 in^2	≥ (1.1) 0.639 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-2

STATIC SEISMIC ANALYSIS FREQUENCY AND STRESSES
1.5" DIAPHRAGM OPERATED THREE WAY VALVES

(TAG NUMBERS 3CX W18SA, 3CX W32SA, 3CX W33SB, 3CX W25SB)

Description	Actual	Criteria
Natural Frequency	82.1 Hz	> 33 Hz*
Yoke Leg Stress Membrane & Bending (Faulted)	1399 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress Membrane & Bending (Faulted)	10634 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	2006 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	9006 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet Membrane & Bending (Max)	12128 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	23799 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	0.897 in^3	$\geq$ (1.1) 0.412 in^3
Area	2.209 in^2	$\geq$(1.1) 1.068 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-3

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES
- 1.5" ELECTROHYDRAULIC THREE WAY VALVES
(TAG NUMBER 3CX W17SA)

Description	Actual	Criteria
Natural Frequency	180 Hz	> 33 Hz*
Yoke Fasteners (Faulted)	2707 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet Membrane & Bending (Max)	2364 PSI	26250 PSI = 1.5 S
Valve Bonnet Bolts	4639 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	0.897 in^3	$\geq$ (1.1) 0.412 in^3
Area	2.209 in^2	$\geq$ (1.1) 1.068 in^2

* Critical for Static Analysis; actuator externally supported.

TABLE 3.9C.14-4

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES -
1.5" DIAPHRAGM OPERATED GLOBE VALVES

(TAG NUMBERS 3CX V122SA, 3CX V123SA, 3CX V245SB)

Description	Actual	Criteria
Natural Frequency	62.5 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	1112 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	2352 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	1259 PSI	< 14400 PSI = 2/3 Sy [.6 Sy]
Yoke Fasteners	5116 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	26179 PSI	<26250 PSI = 1.5 S
Valve Bonnet Bolts	42604 PSI	<50000 PSI = 2.0 S
Valve Body		
Section Modulus	0.897 in^3	≥ (1.1) 0.412 in^3
Area	2.209 in^2	≥ (1.1) 1.068 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-5

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES-
2.0" DIAPHRAGM OPERATED VALVES

(TAG NUMBERS 2MS V59SAB, 2MS V60SAB, 2MS V61SAB, 2BD P6SB, 2BD P7SB, 2BD P8SB)

Description	Actual	Criteria
Natural Frequency	35.2 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	3495 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	2418 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	4359 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	11186 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	14685 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	17538 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	4.07 in^3	$\geq$ (1.1) 0.731 in^3
Area	7.85 in^2	$\geq$ (1.1) 1.477 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-6

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES -
2.5” DIAPHRAGM OPERATED GLOBE VALVES

(TAG NUMBERS 3CX V63SB, 3CX V83SA)

Description	Actual	Criteria
Natural Frequency	79.5 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	2328 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	21181 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	3740 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	10024 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	21196 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	28325 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	3.66 in^3	≥ (1.1) 1.06 in^3
Area	4.73 in^2	≥ (1.1) 1.70 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-7

STATIC SEISMIC ANALYSIS FREQUENCY AND STRESSES
2.5" DIAPHRAGM OPERATED THREE-WAY VALVES

(TAG NUMBERS 3CX W1SA, 3CX W2SA, 3CX W3SB, 3CX W4SB, 3CX W5SB, 3CX W8SA, 3CX W12SB, 3CX W13SB, 3CX W15SA, 3CX W19SA, 3CX W20SA, 3CX W21SA, 3CX W22SB, 3CX W24SB, 3CX W26SB, 3CX W27SB, 3CX W29SB)

Description	Actual	Criteria
Natural Frequency	68.7 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	2073 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	18635 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	3132 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	17309 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	12181 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	24834 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	3.057 in^3	≥ (1.1) 1.064 in^3
Area	3.976 in^2	≥ (1.1) 1.704 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-8

STATIC SEISMIC ANALYSIS   FREQUENCY AND STRESSES
3.0" DIAPHRAGM OPERATED GLOBE VALVES

(TAG NUMBERS 2FW V123SAB, 2FW V124SAB, 2FW V125SAB)

Based on analysis, the maximum flange stress is 11727 psi and the maximum bolt stress is 33801 psi, which are less than the allowable stress limits.

TABLE 3.9C.14-9

STATIC SEISMIC ANALYSIS FREQUENCY AND STRESSES
3.O" DIAPHRAGM OPERATED GLOBE AND FLOW CONTROL VALVES

(This table supplements data on the following table for valves tagged 2FW V123SAB, 2FW V124SAB, 2FW V125SAB)

Description	Actual	Criteria
Natural Frequency	60.4 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	2540 PSI	
Yoke Plate Stress		
Membrane & Bending (Faulted)	3825 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	2163 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	16326 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	10057 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	28067 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	6.93 in^3	$\geq$ (1.1) 2.226 in^3
Area	8.69 in^2	$\geq$(1.1) 3.02 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-10

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES
- 3.0" ELECTROHYDRAULIC OPERATED GLOBE VALVES

(TAG NUMBERS 3SC V15SA, 3SC V26SA, 3SC V30SB, 3SC V31SB)

Description	Actual	Criteria
Natural Frequency	149 Hz	> 33 Hz*
Yoke Fasteners	8417 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	13611 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	22388 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	6.93 in^3	≥ (1.1) 2.226 in^3
Area	8.69 in^2	≥ (1.1) 3.02 in^2

* Criteria for Static Analysis; actuator externally supported.

TABLE 3.9C.14-11

STATIC SEISMIC ANALYSIS FREQUENCY AND STRESSES
3.0" DIAPHRAGM OPERATED GLOBE AND FLOW CONTROL VALVES

(TAG NUMBERS 2MS F1SAB, 2MS F2SAB, 2MS F3SAB)

Description	Actual	Criteria
Natural Frequency	60.4 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	2540 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	3825 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	2163 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	16326 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	10057 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	28067 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	6.93 in^3	$\geq$ (1.1) 2 226 in^3
Area	8.69 in^2	$\geq$ (1.1) 3.02 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-12

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES
- 3.0" ELECTROHYDRAULIC OPERATED FLOW CONTROL VALVES

(TAG NUMBERS 2AF F4SB, 2AF F5SB, 2AF F6SB)

Description	Actual	Criteria
Natural Frequency	115 Hz	> 33 Hz*
Yoke Fasteners	8703 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	15808 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	20575 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	3.66 in^3	$\geq$(1.1) 3.21 in^3
Area	4.73 in^2	$\geq$(1.1) 3.17 in^2

* Criteria for Static Analysis; actuator externally supported.

TABLE 3.9C.14-13

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES -
3.0" DIAPHRAGM OPERATED THREE-WAY VALVES

(TAG NUMBERS 3CX W7SA, 3CX W9SA, 3CX W10SB, 3CX W14SB, 3CX W16SA, 3CX W23SB)

Description	Actual	Criteria
Natural Frequency	68.7 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	2073 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	18635 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	3132 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	17309 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	12181 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	24834 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	3.057 in^3	$\geq$ (1.1) 1.724 in^3
Area	3.976 in^2	$\geq$ (1.1) 2.228 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-14

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES - 4.0" DIAPHRAGM OPERATED GLOBE VALVES

(TAG NUMBERS 2BD V11SN, 2BD V15SN, 2BD V19SN)

Description	Actual	Criteria
Natural Frequency	53.0 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	1671 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	13506 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	2804 PSI	< 14400 PSI = 2/3 [.6 Sy]
Yoke Fasteners	15059 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	18071 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	25360 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	13.538 in^3	≥ (1.1) 3.214 in^3
Area	13.057 in^2	≥ (1.1) 3.174 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-15

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES
- 4.0" DIAPHRAGM OPERATED GLOBE VALVES

(TAG NUMBER 2FP V44SA)

Description	Actual	Criteria
Natural Frequency	62.1 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	1220 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	2182 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	7092 PSI	< 10800 PSI = 2/3 [.6 Sy]
Yoke Fasteners	8531 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	10498 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	25594 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	6.22 in^3	$\geq$ (1.1) 3.21 in^3
Area	6.10 in^2	$\geq$ (1.1) 3.17 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-16

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES
40" ELECTROHYDRAULIC OPERATED PRESSURE CONTROL VALVES

(TAG NUMBERS 3AF P1SA, 3AF P2SB)

Description	Actual	Criteria
Natural Frequency	178 Hz	> 33 Hz*
Yoke Fasteners	8780 PSI	
Valve Bonnet		
Membrane & Bending (Max)	18573 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	30248 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	13.538 in^3	$\geq$ (1.1) 3.21 in^3
Area	13.057 in^2	$\geq$ (1.1) 3.17 in^2

* Criteria for Static Analysis; actuator externally supported.

TABLE 3.9C.14-17

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES
4.0" DIAPHRAGM OPERATED BUTTERFLY VALVES

(TAG NUMBERS 3CX B1SB, 3CX B4SA, 3CH B1SB, 3CH B3SA)

Description	Actual	Criteria
Natural Frequency	116 Hz	> 33 Hz*
Actuator Bracket Stress		
Membrane & Bending (Faulted)	1537 PSI	< 25330 PSI = 2/3 Sy
Actuator Bracket Fasteners	6375 PSI	< 70000 PSI = 2/3 Sy
Actuator Bracket Weld Stress	857 PSI	< 14400 PSI = 0.6 (2/3 Sy)
Valve Body		
Section Modulus	38.1 in^3	> 1.1 (3.21) in^3
Area	31.6 in^2	> 1.1 (3.17) in^2

* Criteria for Static Analysis

TABLE 3.9C.14-18

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES
6.0" DIAPHRAGM OPERATED GLOBE VALVES

(TAG NUMBER 2FP V45SA)

Description	Actual	Criteria
Natural Frequency	78.1 Hz	> 33 Hz*
Yoke Leg Stress		
Membrane & Bending (Faulted)	1968 PSI	< 24000 PSI = 2/3 Sy
Yoke Plate Stress		
Membrane & Bending (Faulted)	2440 PSI	< 25330 PSI = 2/3 Sy
Yoke Weld Stress (Faulted)	7871 PSI	< 10000 PSI = 2/3 [.6 Sy]
Yoke Fasteners	8299 PSI	< 70000 PSI = 2/3 Sy
Valve Bonnet		
Membrane & Bending (Max)	12159 PSI	< 26250 PSI = 1.5 S
Valve Bonnet Bolts	18812 PSI	< 50000 PSI = 2.0 S
Valve Body		
Section Modulus	15.50 in^3	$\geq$ (1.1) 8.50 in^3
Area	10.21 in^2	$\geq$ (1.1) 5.58 in^2

* Criteria for Static Analysis

TABLE 3.9C.14-19

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES - 12.0" DIAPHRAGM OPERATED BUTTERFLY VALVES

(TAG NUMBERS 2SW B88SAB, 2SW B89SA, 2SW B90SB)

Description	Actual	Criteria
Natural Frequency	43.7 Hz	> 33 Hz*
Actuator Bracket Stress		
Membrane & Bending (Faulted)	3861 PSI	< 25330 PSI = 2/3 Sy
Actuator Bracket Fasteners	9862 PSI	< 70000 PSI = 2/3 Sy
Actuator Bracket Weld Stress	5583 PSI	< 14400 PSI = 2/3 (0.6 Sy)
Valve Body		
Section Modulus	171.5 in^3	> 1.1 (43.8) in^3
Area	54.0 in_2	> 1.1 (14.6) in^2
Allowable Stress	17500 PSI	≥ 17500 PSI

* Criteria for Static Analysis

TABLE 3.9C.14-20

STATIC SEISMIC ANALYSIS - FREQUENCY AND STRESSES -
14.0” DIAPHRAGM OPERATED BUTTERFLY VALVES

(TAG NUMBERS 3SW B64SA, 3SW B65SB)

Description	Actual	Criteria
Natural Frequency	54.5 Hz	> 33 Hz*
Yoke Stress		
Membrane & Bending (Faulted)	1918 PSI	< 25330 PSI = 2/3 Sy
Actuator Bracket Fasteners	21294 PSI	< 70000 PSI = 2/3 Sy
Actuator Bracket Weld Stress (Faulted)	7804 PSI	< 14400 PSI = 2/3 (0.6 Sy)
Valve Body		
Section Modulus	485.8 in^3	> (1.1) 53.3 in^3
Area	138 in^2	> (1.1) 16.05 in^2
Allowable Stress	17500 PSI	≥ 17500 PSI

* Criteria for Static Analysis

TABLE 3.9C.15-1

STATIC SEISMIC ANALYSIS- FREQUENCY AND STRESS - 8'x10' BUTTERFLY VALVES

Maximum Disc Stress (Tensile)	0,900 psi
Maximum Disc Deflection	0.075 in.
Maximum Shear in Valve Shafts	4,990 psi
Maximum Bearing Stress (Compression)	
Valve Shafts in Body	7,555 psi
Maximum Tensile Stress in End Cover Bolts	231 psi
Maximum Valve Body End Cover Tensile Stress	5,700 psi
Natural Frequency - Extended Shafting	
1) Section Where Shaft Exits Valve Body	41.5 Hz
2) Typical Extended Section	38.1 Hz
Maximum Shear Stress in Extended Shafting	5,537 psi
Maximum Shear Stress, Floorstand Anchor Bolts	5,615 psi
Natural Frequency, Operator & Floorstand	383 Hz
Operator Floorstand	
Maximum Tensile Stress in Walls	5,350 psi
Maximum Deflection	0.0059 in.
Maximum Base Tensile Stress	6,515 psi
Maximum Bearing Pressure	1,000 psi
Maximum Shear on Thrust Collar Pin	6,666 psi
Bearing Stress of Pin on Collar	6,283 psi
Maximum Shear Stress on Extended Shaft Coupling Pins	13,415 psi
Bearing Stress on Coupling (From Pin)	10,190 psi
Bearing Stress on Extended Shaft Bearings	104 psi
Extended Shaft Bracket	
Shear Key Welds (Shear Stress)	497 psi
Bolt Stress	1,325 psi
Bracket Stress	1,283 psi
Bracket Natural Frequency	143.8 Hz
Backing Plate Maximum Tensile Stress	1,100 psi
Valve Thrust Collar	
Shear Stress on Pin	8,794 psi
Bearing Stress of Pin on Collar	8,288 psi
Bearing Stress Thrust Collar on Plate	1,125 psi
Maximum Shear Stress on Shafting at Collar	10,010 psi
Maximum Tensile in Thrust Plate	1,410 psi
Maximum Tensile in Studs	2,190 psi
Maximum Shear in Bearing Plate	8,430 psi

TABLE 3.10.1-1 SEISMIC INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY NOTE: (THIS IS HISTORICAL INFORMATION - SEE SECTION 3.10.1.1)

EQUIPMENT	MANUFACTURER	MODEL OR DRAWING	SAFETY RELATED FUNCTION	PURCHASED PRIOR TO 3/1/77 (YES/NO)(d)	CONFORMANCE TO IEEE-344 (71 or 75)	QUALIFICATION BY TEST OR ANALYSIS OR BOTH	SEISMIC(a) QUALIFICATION REFERENCE	COMMENTS
RTDs	Weed	N9004E-28	$\Delta T\text{-}T_{avg}$	No	75	Test	WCAP-11587	
	RdF	21205	RCS Temp.	No	75	Test	ESE-6	
	MINCO	S8809	RVLIS Temp. Comp.	No	75	Test	ESE-42A	
Pressure Transmitters	Barton	763	RCS Press. Pzr. Press. S/G Press. CCW Press.	No	75	Test	ESE-1A	
	Barton	753	RCS Press. Turbine First Stage Press.	No	75	Test	ESE-2	
	Veritrak	76PH2	RCS Press.	No	75	Test	ESE-1B	
	Tobar	32PA2/32PG2	VCT Press. Letdown Press. RCS Press.	No	75	Analysis	ESE-2C	
Differential Pressure Transmitters	Barton	764	Pzr. Level S/G Level	No	75	Test	ESE-3A	
	Barton	752	Boric Acid Tank Level RWST Level RV Level/RVLIS CCW Tank Level	No	75	Test	ESE-4	
	Tobar	32DP2	CCW Flow	No	75	Analysis	ESE-4C	
DP Switches	Barton	581A	RHR Miniflow Isolation	No	75	Test	ESE-40	
	Barton	581	RVLIS	No	75	Test	ESE-49A	
Sensors	Barton	351	Containment Press.	No	75	Test	ESE-21	

TABLE 3.10.1-1 SEISMIC INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY NOTE: (THIS IS HISTORICAL INFORMATION - SEE SECTION 3.10.1.1)

EQUIPMENT	MANUFACTURER	MODEL OR DRAWING	SAFETY RELATED FUNCTION	PURCHASED PRIOR TO 3/1/77 (YES/NO)(d)	CONFORMANCE TO IEEE-344 (71 or 75)	QUALIFICATION BY TEST OR ANALYSIS OR BOTH	SEISMIC(a) QUALIFICATION REFERENCE	COMMENTS
	Barton	353	RVLIS	No	75	Test	ESE-48A	
PAMS Indicators	W RID	VX252	PAMS	No	75	Test	ESE-14	
PAMS Recorders	W CID	Optimac 100	PAMS	No	75	Test	ESE-15	
Process Protection System	W ISD	7300	Analog Process Interface 2/Protection Complex	Yes	75	Test	ESE-13	
Nuclear Instrumentation System	W NICD	1054E26 Rev. D	Nuclear Interface w/ Protection Complex	Yes	75	Test	ESE-10	Scope is limited to Power Range
Reactor Trip Switchgear	W LVSD	DS416	Reactor Trip	No	75	Test	ESE-20	
Static Inverter	W PEDS	7.5 KVA 60 Hz/1φ	Vital Instrument Bus Power	No	75	Test	ESE-18	
Solid State Protection System & Safeguards Test Cabinets	W NICD	2 Train	Reactor Protection	Yes	75	Test	ESE-16	
Main Control Board & Termination Cabinets	Westinghouse	1139E34 1139E35 1139E36	Power Plant Control Board	No	75	Test	WCAP-10369 WCAP-10469	
Hydrogen Recombiners	W Sturtevant/ Halmar Electronics	A	Reduce H_2 in Containment Atmosphere	Yes	75	Test	SP-1	
Excore Nuclear Detectors	W IGTD	WL-24154	Power Range Neutron Level Detection	Yes	75	Test	ESE-8A	
RVLIS	W NSID	8086	Supply Information to Operator	No	75	Test	Various (e)	
1) PSMS	W NSID	PSMS/1	RVLIS	No	75	Test	ESE-53	
2) Plasma Display	W NSID	RVLIS 86	RVLIS	No	75	Test	ESE-61A	

TABLE 3.10.1-1 SEISMIC INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY
NOTE: (THIS IS HISTORICAL INFORMATION - SEE SECTION 3.10.1.1)

EQUIPMENT	MANUFACTURER	MODEL OR DRAWING	SAFETY RELATED FUNCTION	PURCHASED PRIOR TO 3/1/77 (YES/NO)(d)	CONFORMANCE TO IEEE-344 (71 or 75)	QUALIFICATION BY TEST OR ANALYSIS OR BOTH	SEISMIC(a) QUALIFICATION REFERENCE	COMMENTS
Thermocouple System	W IGTD	Top Inserted	Supply Information to Operator	NO	75	-	-	
1) T/C	CBK Industries	Type K	T/C System	No	75	Test	ESE-43A	
2) Connectors	ABB	ABB	T/C System	No	75	Test	145-NOME-EP-0038	
3) Reference Junction Box	W IGTD	WX-34794	T/C System	No	75	Test	ESE-44	
Centrifugal Charging Pump Motor	W LMD	8241D38	RCS Charging RCP Seal Injection HHSI	Yes	75	Analysis	AE-2	
RHR Pump Motor	W LMD	8246D34	Residual Removal LHSI	Yes	75	Analysis	AE-2	
Boric Acid Transfer Pump Motor	Chempump	862239	Check RFS	Yes	75	Analysis	AE-3	
Component Cooling Water Pump Motor	W LMD	8249D36	Provide Component Cooling Water to Various Equipment	Yes		Analysis	AE-2	
Limit Switches	NAMCO	EA 170 Series	ECCS Operation & Containment Isolation	Yes	71	Test	Franklin Institute Test FC-3879	
	NAMCO	D2400-X	ECCS Operation & Containment Isolation	Yes	71	Test	Franklin Institute Test FC-3879	
	NAMCO	EA-180 Series	ECCS Operation & Containment Isolation	No	75	Test	HE-3/HE-6[b]	
	NAMCO	EA-740-20100	Main Steam Isolation	No	75	Test	NAMCO Report Qtr-111 2/20/78	
Meter Operator	Limitorque	Various	ECCS Operation & Containment Isolation	Yes	71	Test	Various [c]	
Solenoid Valves	ASCO	FT831654 HT8300B54	ECCS Operation & Containment Isolation	Yes	71	Test	ASCO Report 103	
	ASCO	NP8321A6E	Main Steam Isolation	No	75	Test	AQS-21678/TR Rev. A	

Note: (a) AE-, ESE-, HE-, SP prefixes refer to EQDP's from WCAP-8587 Supplement 1.
(b) Applies only to EA 180 Series Switches purchased under WCAP-8587 program.
(c) Applicable report depends on operator specifics and is traceable through operator serial number and shop order number.
(d) Equipment purchased prior to 3/1/77 is qualified to the supplemental program (Reference 3.10.2-2).
(e) Reports applicable to RVLIS: EQDP's ESE-4, 15, 42, 48, 49, 53, and 61 as identified in WCAP-8587.

TABLE 3.10.1-2 NON-NSSS SEISMIC CATEGORY I ELECTRICAL AND INSTRUMENTATION EQUIPMENT QUALIFICATION
NOTE: (THIS IS HISTORICAL INFORMATION - SEE SECTION 3.10.1.2)

Equipment	Manufacturer	Function	Conformance to IEEE 344 (1971 or 1975)	Qualification by Test or Analysis or Both[(1)]	Summary of Results/ Comments [(1)]
6900 Volt Class IE Switchgear	Siemens-Allis	Supply of 6900 Volt AC power	1975		
480 Volt Class IE Switchgear and Station Services Transformers	Gould (ITE)	Supply of 480 Volt AC Power	1975		
480 Volt Class IE Motor Control Centers	Gould (ITE)	Supply of 480 Volt AC power to small loads	1975		
125 Volt Class IE Batteries and Racks	C & D	Supply of 125 Volt DC power	1975		
125 Volt DC Distribution Panels	Gould (ITE)	Supply of 125 Volt DC power	1975		
6900 Volt Class IE Vacuum Breakers	Siemens	Supply of 6900 Volt AC power	1975		
6900 Volt Transfer Switch	Eaton/Cutler-Hammer	Supply of 6900 Volt AC Power	1975		
Fire Rated Cable	Meggitt Safety System, Inc.	Power and Control Circuits	1975	Test	
125 Volt Class IE DC Battery Chargers	C & D	Maintenance and re-charge of batteries	1975		
Emergency Diesel Generators and Components	DeLaval	Standby power source	1975		
Containment Electrical Penetrations	Westinghouse	Conduction of power to components within containment	1975		
Cable Tray	Burndy-Husky	Support of Class IE Cables	1975		
Class IE Axial Flow Fan Motors	Joy Mnfg. Co./Reliance	Cooling Reactor Cavity & Reactor Supports	1975		
Class IE Electric Heating Coils	Brasch Manufacturing Co.	Heating Control Room	1975		
Dampers - Class IE Motor Operators	Ruskin/ITT	Control Air Flow	1975		
Class IE Containment Fan Cooler Motors	American Air Filter/Reliance	Cooling Containment	1975		

TABLE 3.10.1-2 NON-NSSS SEISMIC CATEGORY I ELECTRICAL AND INSTRUMENTATION EQUIPMENT QUALIFICATION NOTE: (THIS IS HISTORICAL INFORMATION - SEE SECTION 3.10.1.2)

Equipment	Manufacturer	Function	Conformance to IEEE 344 (1971 or 1975)	Qualification by Test or Analysis or Both(1)	Summary of Results/ Comments (1)
Water Chillers Comp. Pump Motor and Accessories	York/Reliance	Chilled Water for ESF Systems	1975		
Chilled Water Circulating Pump Motors		Circulates Water for ESF Systems	1975		
Air Handling Units	Bahnson/Reliance	Cooling for ESF Systems	1975		
Centrifugal Fans	Barry Blower	Ventilation for ESF Systems	1975		
Air Cleaning Units	CTI-Nuclear/Reliance	Airborne Radioactivity Removal	1975		
Butterfly Valves - Class IE Motor Operators	BIF/Limitorque	Containment Isolation and Control Air Flow	1975		
Emergency Service Water Pump Motors	GE	Drives Pump	1975		
Screen Wash Pump Motors	Reliance	Drives Pump	1975		
Motor Operators:					
Control Valves ITT/ND 435176	ITT General Controls (Electro-Hydraulic)	Actuates Valves	1975		
Station Valves Pacific 435022	Limitorque	Actuates Valves	1975		
M-32A Station Anchor Darling 435013	Limitorque	Actuates Valves	1975		
M36Y Station Valves Yarway 435056	Limitorque	Actuates Valves	1975		
Butterfly Valves Jamesbury 435082	Limitorque	Actuates Valves	1975		
Packless Valve Kerotest	Limitorque	Actuates Valves	1975		
Hydrogen Analyzers	Exo Sensor Inc.	Monitor the hydrogen concentration in containment	1975		
Auxiliary Feedwater Pump Motors	Westinghouse	Drives AFP	1975		
Fuel Oil Transfer Pump Motors	Westinghouse	Drives Pump	1975		
Containment Spray Pump Motors	Westinghouse	Drives CSP	1975		

TABLE 3.10.1-2 NON-NSSS SEISMIC CATEGORY I ELECTRICAL AND INSTRUMENTATION EQUIPMENT QUALIFICATION NOTE: (THIS IS HISTORICAL INFORMATION - SEE SECTION 3.10.1.2)

Equipment	Manufacturer	Function	Conformance to IEEE 344 (1971 or 1975)	Qualification by Test or Analysis or Both(1)	Summary of Results/ Comments (1)
Service Water Booster & Fuel Pool Cooling Pump Motors	Allis-Chalmers	Drives Pump	1975		
Fuel Pool Cooling Heat Exchanger Heater	Yuba	Remove heat from spent fuel	1975		
Solenoid Valve Spec M-73	Target Rock Corp.	Actuates Valve	1975		
Basket Strainers Pressure Switches 435163	Zurn (ITT Barton)	Annunciation	1975		
Temperature, Pressure Flow Transmitters	Rosemount	Control & Alarm	1975		
Solenoid Valves	Asco	Actuate pneumatic valve	1975		
Pressure Switch	Static-O-Ring	Alarm	1975		
Limit Switch	Namco	Valve position indication	1975		
Flow Nozzles	Vickery-Simms, Inc.	Control & Alarm	Non-electrical Component		
Orifice Plates	Vickery-Simms, Inc.	Control & Alarm	Non-electrical component		
Thermocouples Assemblies & Test Wells	Weed Instrument Co.	Control & Alarm	1975		
Auxiliary Control Panel	Reliance Electric Co.	Control & Alarm	1975		
Local Panels and Racks	Mercury Co. of Norwood	Control & Alarm	1975		
Local Pressure	Dresser Industries	Local Indication	Non-electrical component		
Local Dial Thermometers	Dresser Industries	Local Indication	Non-electrical component		
Auxiliary Relay Panels	Systems Controls Corp.	Control & Alarm	1975		
Isolation Panels	Consolidated Controls Corp	Control & Alarm	1975		
Transfer Panels	Systems Control Corp	Control & Alarm	1975		
Sequencer Panels	Systems Control Corp	Control & Alarm	1975		
Seismic Monitoring System	Kinemetrics, Inc.	Seismic Monitoring	1975		

TABLE 3.10.1-2 NON-NSSS SEISMIC CATEGORY I ELECTRICAL AND INSTRUMENTATION EQUIPMENT QUALIFICATION
NOTE: (THIS IS HISTORICAL INFORMATION - SEE SECTION 3.10.1.2)

Equipment	Manufacturer	Function	Conformance to IEEE 344 (1971 or 1975)	Qualification by Test or Analysis or Both(1)	Summary of Results/ Comments (1)
Instrument Tubing and Fittings	Parker and Swagelok	Instrument Connections	Non-electrical component		
Level Transmitters	Rosemount, Inc. and ITT Barton	Control & Alarm	1975		
Safety Related Radiation Monitors	General Atomic	Detect, Alarm & Monitor Radioactivity	1975		
Level Switches	Magnetrol International, Inc.	Control & Alarm	1975		

(1)Refer to applicable vendor seismic qualification report.

TABLE 3.11.0-1 NSSS SUPPLIED SAFETY-RELATED EQUIPMENT (NOTE: THIS IS HISTORICAL INFORMATION – SEE SECTION 3.11.0)

Equipment [(a)]	Manufacturer	Model or Drawing	Qualification Reference [(b)]	Qualification Level
RTD's	RdF	21205	ESE-6	1974
	MINCO	58809	ESE-42A	1974
Pressure Transmitters	Barton	763	ESE-1A	1974
	Barton	753	ESE-2	1974
	Veritrak	76PH2	ESE-1B	1974
	Tobar	32PA2/32PG2	ESE-2C	1974
Differential Pressure Transmitters	Barton	764	ESE-3A	1974
	Barton	752	ESE-4	1974
	Tobar	32DP2	ESE-4C	1974
D/P Switches	Barton	581A	ESE-40	1974
	Barton	581	ESE-49A	1974
Sensors	Barton	351	ESE-21	1974
	Barton	353	ESE-48A	1974
Indicators	W RID	VX252	ESE-14	1974
Recorders	W CID	Optimac 100	ESE-15	1974
Process Protection System	W ISD	7300	ESE-13	1974
Nuclear Instrumentation System	W NICD	1054E26 Rev. D	ESE-10	1974
Reactor Trip Switchgear	W LVSD	DS416	ESE-20	1974
Static Inverter	W PEDS	7.5 KVA	ESE-18	1974
SSPS & Test Cabinets	W NICD	2 Trains	ESE-16	1974
Main Control Board & Termination Cabinets	Westinghouse	1139E34	WCAP-10369	1974
		1139E35	WCAP-10469	
		1139E36		
Excore Neutron Detectors	W IGTD	WL-24154	ESE-8A	1974
RVLIS	W NSID	8086	Various [(c)]	-
1) PSMS	W NSID	PSMS/1	ESE-53	1974
2) Plasma Display	W NSID	RVLIS 86	ESE-61A	1974
Thermocouple System	W IGTD	Top inserted	-	-
1) T/C	CBK Industries	Type K	ESE-43A	1974
2) Connectors	ABB	ABB	145-NOME-EP-0038	1974
3) Ref. Junc. Box	W IGTD	WX-34794	ESE-44	1974
Centrifugal Charging Pump Motor	W LMD	8241D38	AE-2	1974
RHR Pump Motor	W LMD	8246D34	AE-2	1974
Boric Acid Transfer Pump Motor	Chempump	B62239	AE-3	1974
Component Cooling Water Motor	W LMD	8249D36	AE-2	1974
Valve Motor	Limitorque	Various	Various[(d)(f)]	1971
Operators	Limitorque	Various	Various[(d)]	1974
Solenoid Valves	ASCO	NP8321A6E	AQS-21678/ TR Rev. A	1974

TABLE 3.11.0-1 NSSS SUPPLIED SAFETY-RELATED EQUIPMENT (NOTE: THIS IS HISTORICAL INFORMATION – SEE SECTION 3.11.0)

Equipment [(a)]	Manufacturer	Model or Drawing	Qualification Reference [(b)]	Qualification Level
	ASCO	FT831654	NS-CE-775[(f)]	1971
	ASCO	HT8300B54	NS-CE-755[(f)]	1971
Limit Switches	NAMCO	EA-180 Series	HE-3/HE-6[(c)]	1974
	NAMCO	EA 740-20100	NAMCO Report Qtr-111 2/20/78 and Qtr-117 Rev. 0	1974
	NAMCO	EA-170 Series	Franklin Report FC-3879[(f)] and NAMCO Report Qtr-107	1971 1974
	NAMCO	D2400-2	Franklin Report FC-3879[(f)]	1971

NOTES:

a) Refer to Table 3.10.1-1 for equipment function.
b) AE-, ESE-, HE-, SP- prefixes refer to EQDP's from WCAP-8587.
c) Reports applicable to RVLIS: EQDP's -4, 15, 42A, 48A, 49A, 53, 61A.
d) Applicable report depends on operator specifics and is traceable through serial number and shop order number.
e) Applied only to switches purchased under the WCAP-8587 program.
f) Equipment to function in mild environment only.

TABLE 3.11.0-2 EBASCO PURCHASED SAFETY-RELATED EQUIPMENT (HARSH)
(NOTE: THIS IS HISTORICAL INFORMATION – SEE SECTION 3.11.0)

EQUIPMENT	MANUFACTURER	MODEL NUMBER	QUALIFICATION PER IEEE-323 (1971 OR 1974)
Power and Control Cable	Kerite	FR-II, FR-III, HTK, FR	1974
Instrumentation Cable	American Insulated Wire Corp.	EPR	1974
15 KV Power Cable	Anaconda	EPR	1974
Coaxial Cable	Rockbestos	RG-11/(RSS-6-108); RG-54/(RSS-6-105); RSS-6-104	1974
Control Cable	Rockbestos	Firewall III, Chemically XLPE	1974
Control Cable	Rockbestos	Firewall III, Irradiation XLPE	1974
Radiation Detectors	General Atomic	RD-8	1974
Radiation Detectors	General Atomic	RD-23	1974
Level-Transmitters	TDI-Gem.	XM 5400	1974
Transmitters	Rosemount	1153 Series B	1974
Transmitters	Gould	DR 3200	1974
Transmitters	Gould	PG 3200	1974
Transmitters	Barton	763; 764	1974
Skids	General Atomic	WRGM/RD-52/RD-72	1974
Radiation Alarms	General Atomic	RL-10	1974
Limit Switches	NAMCO	EA-180	1974
Electric Heating Coil	Brasch	XP-83-09576, CP-83-0073, XF-83-08780, XS-78-09579, CP-83-00105, XS-78-08648, CP-78-00386, XS-78-08049, CP-78-00389	1974
Terminal Blocks	General Electric	EB25, EB26, EB27	1974
Penetrations	Westinghouse	WX33400 Series	1974
Air Lock Penetrations	Conax	N11000 Series	1974
Position Indicator	Crosby	65320	1974
Level Switches	Magnetrol International	A153F	1974
Pressure Switches	ITT Barton	583A-0, 580A-0, 580-0	1974
RTDs and Thermocouples	Weed	612-1B-A-4-C-13-0-0, E4B250G	1974
Thermocouples	Conax	2310-9458-01, 7T95-10000-03	1974
Isokinetic Sampling	TEC	Model 2025	1974
Pneumatic Operator	Bettis	N521C, N721C, NT-820, NCB-315, NCB-520	1974
Pneumatic Operator	Bettis	CB-315; CB-415	1974
Local Control Stations	General Electric	CR 2940 Series	1974
480V MCCs	Telemecanique	Series 5600	1974
Solenoid Valves	Target Rock	79Q-005, -006, -014, -018, -021, -026, -1011105-2, 1015005-3, 1031010-2, 1032110-4, 1033005-2,	1974
Solenoid Valves	ASCO	NP8316, NP8320, NP8321	1974
Valve Operators	Limitorque	SB-00, SMB-00, SMB-000, SMC-04	1974
Pneu-Operator	Ralf Hilier	12 SA-A029	1974
Elec-Hyd Operator	ITT	NH91, NH92, NH94, NH95, NH96	1974
Containment Fan Cooler Motor	Reliance	TEAO, Frame 449T, Class H Type RN Insulation	1974

TABLE 3.11.0-2 EBASCO PURCHASED SAFETY-RELATED EQUIPMENT (HARSH)
(NOTE: THIS IS HISTORICAL INFORMATION – SEE SECTION 3.11.0)

EQUIPMENT	MANUFACTURER	MODEL NUMBER	QUALIFICATION PER IEEE-323 (1971 OR 1974)
Motors	Reliance	RH Insulation	1974
Motors	Siemens Allis	447TS	1974
Motors	Siemens Allis	445TS	1974
Motors	Westinghouse	5810H and 5008P39	1974
Triaxial Cable	Boston Insulated Wire and Cable Co.	CSPE/Tetzel, RG 11/U	1974
Instrumentation Cable	Anaconda	FR-EP	1974
Thermocouple Cable	Eaton-Samuel Moore	EPDM	1974
6.9 KV Switchgear	Siemens Allis	FB 500	1974
480 V Switchgear	Brown Boveri	Type LK	1974
Motors	Westinghouse	143T, 182T, 184T, 213T, 254T, 256T, 286TS, 404T	1974
ESW Pump Motors	General Electric	5K6356XC21A	1974
Chiller Motors	Reliance	4824-SM Insulation	1974
Diesel Gen. Aux. Motors	Siemens Allis	450T Type RGZ	1974(a)
Diesel Gen. Control Panel	RTE Delta Corp.	NA	1974(a)
Diesel Gen. Engine Control Panel	TDI	NA	1974(a)
Radiation Monitor	General Atomic	Various	1974
Aux. Control Panel	Reliance Electric	NA	1974
Aux. Equipment Panel	Consolidated	NA	1974
Aux Equipment Panel	Reliance	NA	1974
Chiller Control Panel	York	NA	1974
Aux. Relay Panel	System Control	NA	1974
Local Control Cabinet and Racks	Mercury Co.	NA	1974
Batteries	C&D Batteries	60LC-19	1971
Battery Changers	C&D Batteries	ARR130HK150F	----
Valve Operator	Paul Munroe Hydraulics	PD25210-200	1974
Diesel Gen. Comp. Starter	Telemecanique	Size 3 NEMA 12	1974
125 V DC Dist. Panel	BBC-Gould	FC-1	1974
Sync. Generator	Parsons	L-11043	1974

Note (a):Repair or replacement parts may be procured to the requirements of IEEE Standard 323-1983.

TABLE 3.11.0-3

CP&L PURCHASED SAFETY-RELATED EQUIPMENT (HARSH AND MILD)

(NOTE: THIS IS HISTORICAL INFORMATION – SEE SECTION 3.11.0)

EQUIPMENT	MANUFACTURER	MODEL NUMBER	IEEE-323 (1971 OR 1974)
Solenoid Valve	ASCO	FT Series	1974
Cable	General Electric	Vulkene Supreme	1974
Cable	Anaconda	NSIS	1974
Switchboard Wire	Brand Rex	Ultrol Switchboard Wire	1974
Cable	Okonite	Okozel Wire	1974
Coaxial Cable	Brand Rex	NA	1974
Limit Switch	NAMCO	EA-1970	1974
Conduit Seal	Conax	ECSA	1974
Splice Kit	Raychem	WCSF-N, NMCK, NHVT, NJRS, NPK	1974
Tape	Bishop	No. 17	1974
Tape	Okonite	35/T95	1974
Tape	Scotch	130C	1974
Terminal Block	General Electric	CR 151	1974
Terminal Block	Marathon	1500 NUC, 1600 NUC, 142-ST-NUC	1974
Draw-out type 480 V Breakers	Wyle/Siemens	RLN Series	1974
Thread Sealant	Patel Engineers	P-1	1974
Connector	Patel/EGS	880701 913601 913602	
RTD	Weed	N9004E-2B	1974
Transmitter	Rosemount	1153 1154	1974
Cable Connector	EGS	GB-1, GB-2, GB-3	1974
Transmitter	AMETEK (Gulton-Statham)	PG3200	1974

TABLE 3.11.1-1 SHEARON HARRIS NUCLEAR POWER PLANT SAFETY RELATED EQUIPMENT LOCATION CODES

IDEN	PC	AREA	EXCL	XMIN	XMAX	YMIN	YMAX	ZMIN	ZMAX	RCYL
WP11	1	WASTE PROC EL 211		1192.00	1382.90	1655.00	1945.00	211.00	234.90	0.00
WP21	2	WASTE PROC EL 236		1192.00	1382.90	1655.00	1945.00	235.00	259.90	0.00
WP21	2	WASTE PROC EL 236	EXCL	1288.00	1382.90	1766.00	1834.00	234.00	260.90	0.00
WP22	3	WP CTL RM & VAULT		1288.00	1382.90	1766.00	1834.00	235.00	259.90	0.00
AB52	4	RAB EL 305-CONT RM		1537.00	1764.00	1513.00	1699.90	304.00	323.00	0.00
AB52	4	RAB EL 305-CONT RM	EXCL	1737.00	1764.00	1572.00	1699.90	304.00	330.00	0.00
AB52	4	RAB EL 305-CONT RM	EXCL	1537.00	1565.00	1570.00	1699.90	303.00	324.00	0.00
AB52	4	RAB EL 305-CONT RM	EXCL	1537.00	1592.00	1640.00	1699.90	303.00	324.00	0.00
AB01	5	RAB EL 190		1383.00	1650.00	1513.00	1699.90	190.00	215.00	0.00
AB01	5	RAB EL 190	EXCL	1500.00	0.00	1700.00	0.00	189.00	450.00	65.00
AB21	6	RAB EL 236		1383.00	1740.00	1513.00	1699.90	235.00	259.90	0.00
AB21	6	RAB EL 236	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
AB31	7	RAB EL 261		1383.00	1710.00	1513.00	1699.90	260.00	284.90	0.00
AB31	7	RAB EL 261	EXCL	1476.00	1525.00	1570.00	1647.00	259.00	310.00	0.00
AB31	7	RAB EL 261	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
AB31	7	RAB EL 261	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
AB32	8	RAB EL 261-305		1476.00	1525.00	1570.00	1647.00	260.00	303.90	0.00
AB32	8	RAB EL 261-305	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
TA11	9	TANK AREA UNIT 1		1319.00	1382.90	1513.00	1654.90	230.00	340.00	0.00
SW11	10	SEC WASTE EL 216		1383.00	1453.00	1700.00	1900.00	216.00	234.90	0.00
SW11	10	SEC WASTE EL 216	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
SW21	11	SEC WASTE EL 236		1383.00	1453.00	1700.00	1900.00	235.00	259.90	0.00
SW21	11	SEC WASTE EL 236	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
TB31	12	TURB ELS 240-261		1278.00	1710.00	1345.00	1513.00	240.00	284.90	0.00
TB41	13	TURB ELS 286-314		1278.00	1710.00	1345.00	1513.00	285.00	340.00	0.00
FH11	14	FUEL HDLG EL 216		1453.00	1917.00	1700.00	1900.00	216.00	234.90	0.00
FH11	14	FUEL HDLG EL 216	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
AB43	15	RAB EL 286-SWGR RM		1383.00	1589.90	1513.00	1602.00	285.00	303.90	0.00
AB43	15	RAB EL 286-SWGR RM	EXCL	1565.00	1589.90	1513.00	1570.00	284.00	310.00	0.00
AB43	15	RAB EL 286-SWGR RM	EXCL	1476.00	1525.00	1570.00	1602.00	284.00	310.00	0.00
AB41	16	RAB EL 286		1383.00	1650.00	1513.00	1699.90	285.00	303.90	0.00

TABLE 3.11.1-1 SHEARON HARRIS NUCLEAR POWER PLANT SAFETY RELATED EQUIPMENT LOCATION CODES

IDEN	PC	AREA	EXCL	XMIN	XMAX	YMIN	YMAX	ZMIN	ZMAX	RCYL
AB41	16	RAB EL 286	EXCL	1476.00	1525.00	1570.00	1647.00	259.00	310.00	0.00
AB41	16	RAB EL 286	EXCL	1383.00	1565.00	1513.00	1602.00	284.00	310.00	0.00
AB41	16	RAB EL 286	EXCL	1565.00	1589.90	1570.00	1602.00	284.00	310.00	0.00
AB41	16	RAB EL 286	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
FH21	17	FUEL HDLG EL 236		1453.00	1917.00	1700.00	1900.00	235.00	259.90	0.00
FH21	17	FUEL HDLG EL 236	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
AB11	18	RAB EL 216		1383.00	1749.00	1513.00	1699.90	215.00	234.90	0.00
AB11	18	RAB EL 216	EXCL	1590.00	1710.00	1610.00	1699.90	214.00	236.90	0.00
AB11	18	RAB EL 216	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
WP31	19	WASTE PROC EL 261		1192.00	1382.90	1655.00	1945.00	260.00	289.90	0.00
WP31	19	WASTE PROC EL 261	EXCL	1192.00	1382.90	1766.00	1905.00	274.00	290.90	0.00
SW31	20	SEC WASTE EL 261		1383.00	1453.00	1700.00	1900.00	260.00	284.90	0.00
SW31	20	SEC WASTE EL 261	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
FH31	21	FUEL HDLG EL 261		1453.00	2016.00	1700.00	1900.00	260.00	284.90	0.00
FH31	21	FUEL HDLG EL 261	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
XY31	22	XFMR YARD		1380.00	1620.00	1145.00	1346.90	250.00	300.00	0.00
WT31	23	WATER TREAT BLDG		750.00	1042.00	2060.00	2310.00	250.00	350.00	0.00
WT31	23	WATER TREAT BLDG	EXCL	935.00	1026.00	2060.00	2106.90	249.00	351.00	0.00
DF31	24	DIESEL FO STR BLDG		2140.00	2240.00	1852.00	1948.00	240.00	300.00	0.00
BA31	25	AUX BLR AREA		724.00	1025.90	2032.00	2475.00	250.00	300.00	0.00
BA31	25	AUX BLR AREA	EXCL	823.90	1025.90	2400.00	2475.00	249.00	301.00	0.00
BA31	25	AUX BLR AREA	EXCL	724.00	1025.90	2106.90	2399.90	249.00	301.00	0.00
BA31	25	AUX BLR AREA	EXCL	724.00	949.90	2032.00	2106.90	249.00	301.00	0.00
IE31	26	INTAKE STR-EMER SW		44.00	173.00	1475.00	1685.00	230.00	300.00	0.00
SS31	27	EM SCREEN STRUCT		160.00	300.00	2130.00	2200.00	230.00	300.00	0.00
IS31	28	INTAKE STRUCT-SW		1560.00	1750.00	415.00	525.00	250.00	300.00	0.00
AB51	29	RAB EL 305		1458.00	1545.00	1513.00	1654.00	304.00	325.00	0.00
AB51	29	RAB EL 305	EXCL	1537.00	1545.00	1513.00	1570.00	303.00	321.00	0.00
WP41	30	WASTE PROC EL 276		1192.00	1382.90	1766.00	1905.00	275.00	289.90	0.00
WP51	31	WASTE PROC EL 291		1192.00	1382.90	1658.00	1945.00	290.00	340.00	0.00
FH41	32	FUEL HDLG EL 286		1411.00	2016.00	1700.00	1900.00	285.00	303.90	0.00
FH41	32	FUEL HDLG EL 286	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00

TABLE 3.11.1-1 SHEARON HARRIS NUCLEAR POWER PLANT SAFETY RELATED EQUIPMENT LOCATION CODES

IDEN	PC	AREA	EXCL	XMIN	XMAX	YMIN	YMAX	ZMIN	ZMAX	RCYL
CB11	33	RCB EL 221		1500.00	0.00	1700.00	0.00	190.00	235.00	65.00
CB21	34	RCB EL 236		1500.00	0.00	1700.00	0.00	236.00	260.00	65.00
CB31	35	RCB EL 261		1500.00	0.00	1700.00	0.00	261.00	285.00	65.00
CB41	36	RCB EL 286		1500.00	0.00	1700.00	0.00	286.00	450.00	65.00
IC31	37	INTAKE STRUCT-CW		1220.00	1335.00	620.00	780.00	250.00	300.00	0.00
CT31	38	COOL'G TOWER		1210.00	1500.00	420.00	150.00	250.00	650.00	0.00
DG31	39	DIESEL GEN BLDG		1573.00	1727.00	1057.00	1180.00	240.00	350.00	0.00
2S31	40	230KV SWYD		330.00	790.00	525.00	1180.00	250.00	280.00	0.00
FH51	41	FUEL HDLG EL 305		1411.00	2016.00	1700.00	1900.00	304.00	332.90	0.00
FH51	41	FUEL HDLG EL 305	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
FH61	42	FUEL HDLG EL 324		1411.00	2016.00	1700.00	1900.00	323.00	340.00	0.00
FH61	42	FUEL HDLG EL 324	EXCL	1500.00	0.00	1700.00	0.00	190.00	450.00	65.00
CF01	43	INT STR-CPE FR RIV		7425.00	7575.00	7725.00	7875.00	137.00	230.00	0.00
MD01	44	MAIN DAM SPILLWAY		5425.00	5575.00	5725.00	5875.00	195.00	270.00	0.00
SB31	45	SERVICE BLDG		950.00	1055.00	1700.00	1955.00	250.00	350.00	0.00
GS31	46	GAS STORAGE BLDG		965.00	1040.00	2400.00	2575.00	250.00	305.00	0.00
SS31	47	500 kV SWITCHYARD		3006.00	3914.00	1911.00	2700.00	250.00	280.00	0.00
TA12	48	TANK AREA		1917.00	1981.00	1513.00	1656.00	236.00	340.00	0.00
YD31	49	Y D R AREA #1		390.00	1650.00	60.00	1800.00	240.00	275.00	0.00
YD32	50	Y D R AREA #2		1650.00	2600.00	60.00	1800.00	240.00	275.00	0.00
YD33	51	Y D 4 AREA #3		1650.00	2600.00	1800.00	3400.00	240.00	275.00	0.00
YD34	52	Y D 4 AREA #4		390.00	1650.00	1800.00	3400.00	240.00	275.00	0.00

Table 3.11C-1

EQ Thermal Lag Analyses in Containment

Description of Limiting Mass and Energies Used in Analysis	Computer Code Used in Analysis	Equipment Analyzed	Maximum Analyzed Equipment Temperature (°F)	Maximum Equipment Qualification Temperature (°F)
102% Power, 1.4 ft^2 with MFIV failure MSLB (using pre SGR/PUR M&Es)[2]	CONTEMPT-LT Mod 28	Containment Fan Coolers Low Voltage Penetration Module	264.6 247	330 340
102% Power, 1.4 ft^2 with MFIV failure MSLB (using pre SGR/PUR M&Es)[2]	CONTEMPT-LT Mod 28	Reuter Stokes Ion Chamber	264	365
102% Power, 1.4 ft^2 with MFIV failure MSLB (using pre SGR/PUR M&Es)	CONTEMPT-LT Mod 28	134 MIL Okonite Single Conductor BIW Rockbestos Thermocouple Rockbestos Coaxial Okonite-Multiple Conductor Okozel in Conduit	344.6[1] 327.4 342.1[3] 336.4 305 294.5[4]	346 340 342.1 346 341 346
102% Power, 1.4 ft^2 with MFIV failure MSLB (using pre SGR/PUR M&Es)	GOTHIC Version 6.1b	90 Mil, Okonite Single Conductor PA/PG 3200 Pressure Transmitter	338[5] 261.6[6]	346 266

Notes for Table 3.11C-1:

1. Maximum temperature is at the surface of the jacket. The insulation temperature is less than that of the jacket. For this reason, the margin predicted for the 90 Mil conductor by GOTHIC is larger than that predicted for the 134 Mil conductor by CONTEMPT.
2. Evaluated using pre-SGR/PUR mass and energies (M&Es). Since the maximum temperature predicted for the cables using post SGR/PUR M&Es did not change significantly with respect to analyses using pre-SGR/PUR M&Es and the margin between the maximum analyzed and qualified temperatures is large, no further analysis using post SGR/PUR M&Es was performed.
3. Maximum temperature is at surface of jacket. Surface temperature at the shield or insulation surfaces would be significantly lower based on pre-SGR/PUR analyses.
4. Analysis credits thermal lag of conduit.
5. Maximum temperature is at the surface of the insulation.
6. Analysis credits insulation effect of mounting bracket and thermal lag effect of air inside transmitter housing. Temperature is maximum temperature of the air in the housing.

Table 3.11E-1

COEFFICIENTS FOR CALCULATION OF AVERAGE HEAT TRANSFER COEFFICIENTS

Coefficients for Calculation of Average Heat Transfer Coefficient
of a Circular Cylinder in a Gas Flowing
Normal to the Cylinder Axis from Reference 3.11E-2

$Re_{Df} = \frac{V_{\infty} D_o}{v}$	C	n
0.4 - 4	0.891	0.330
4 - 40	0.821	0.385
40 - 4,000	0.825	0.466
4,000 - 40,000	0.174	0.618
40,000 - 400,000	0.0239	0.805

Table 3.11E-2

EQ Thermal Lag Analyses in the Main Steam Line Tunnel

Description of Limiting Mass and Energies Used in Analysis	Computer Code Used in Analysis	Equipment Analyzed	Maximum Analyzed Equipment Temperature (°F)	Maximum Equipment Qualification Temperature (°F)
102 Percent power, 0.5 ft break MSLB (pre SGR/PUR)	GOTHIC Version 3.4d	ASCO Valve Body w/0.25 inch insulation ASCO Coil Housing NAMCO Limit Switch	258.3 389.8 329.4	346 411 340
70 percent power, 0.5 ft^2 break MSLB (post SGR/PUR)	GOTHIC Version 6.1 b	Okonite single conductor 90 MIl inside conduit	302(1)	340

NOTES:

(1) Analysis credits thermal lag of conduit. Temperature represents maximum temperature of inner surface of conduit.

FIGURE	TITLE
3.6A-17-PLOT-A	REACTOR AUXILIARY BUILDING, PLOT OF PIPE BREAK LOCATIONS, SAFETY INJECTION PIPING
3.6A-18	REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE
3.6A-18-CALC	REACTOR AUXILIARY BUILDING, SUMMARY OF PIPE BREAK LOCATIONS, SAFETY INJECTION PIPING
3.6A-18-PLOT-A	REACTOR AUXILIARY BUILDING, PLOT OF PIPE BREAK LOCATIONS, SAFETY INJECTION PIPING
3.6A-19	REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE
3.6A-19-CALC	DELETED BY AMENDMENT 27
3.6A-19-PLOT-A	DELETED BY AMENDMENT 27
3.6A-20	REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE
3.6A-20-CALC	CONTAINMENT BUILDING, SUMMARY OF PIPE BREAK LOCATIONS, REACTOR COOLANT PIPING (LOOPS 1, 2 AND 3)
3.6A-20-PLOT-A	DELETED BY AMENDMENT 51
3.6A-20-PLOT-B	DELETED BY AMENDMENT 51
3.6A-20-PLOT-C	DELETED BY AMENDMENT 51
3.6A-20.1-CALC	CONTAINMENT BUILDING, SUMMARY OF PIPE BREAK LOCATIONS, REACTOR COOLANT PIPING (LOOPS 1, 2 AND 3)
3.6A-20.2-CALC	CONTAINMENT BUILDING – SUMMARY OF PIPE BREAK LOCATIONS – SAFETY INJECTION – COLD LEG
3.6A-20.3-CALC	CONTAINMENT BUILDING – SUMMARY OF PIPE BREAK LOCATIONS – SAFETY INJECTION – HOT LEG
3.6A-21	REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE
3.6A-21-CALC	CONTAINMENT BUILDING – SUMMARY OF PIPE BREAK LOCATIONS – RHR SYSTEM
3.6A-22	REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE
3.6A-23	REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE
3.6A-23-CALC	REACTOR CONTAINMENT BUILDING – SUMMARY OF PIPE BREAK LOCATIONS – PRESSURIZER SURGE
3.6A-24	REFER TO FSAR TABLE 1.6-3 FOR DESIGN DOCUMENT INCORPORATED BY REFERENCE
3.6A-24-CALC	CONTAINMENT BUILDING, SUMMARY OF PIPE BREAK LOCATIONS, BLOWDOWN PIPING (LOOPS 1, 2, AND 3)
3.6A-24-PLOT-A	CONTAINMENT BUILDING, PLOT OF PIPE BREAK LOCATIONS, BLOWDOWN PIPING (LOOP 1)

FIGURE 3.3.1-1

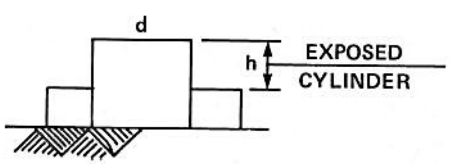
CONTAINMENT WALL WIND PRESSURE COEFFICIENTS AND DESIGN WIND PRESSURE

V_{30} = 179 MPH

$q = 0.00256\, V_{30}^{2} = 82$ PSF

q_F = 133 PSF

PRESSURE COEFFICIENTS Cpe	0.9	0.8	0.7	0.6	0.4	0.2	0	-0.2	-0.4	-0.5	-0.55	-0.6	-0.8	-0.9	-1.0	-1.1	-1.2	-1.25
WIND PRESSURE q_F Cpe	120	106	93	80	53	27	0	-27	-53	-67	-73	-80	-106	-120	-133	-146	-160	-166



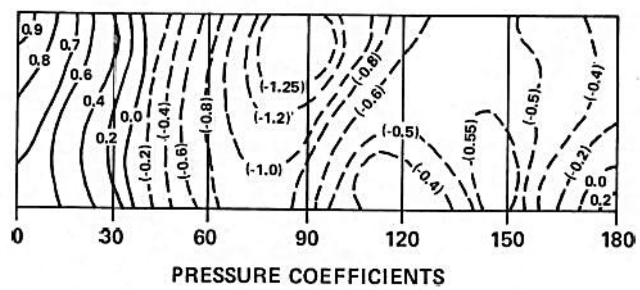


FIGURE 3.3.1-2

CONTAINMENT DOME WIND PRESSURE COEFFICIENTS AND DESIGN WIND PRESSURE

PRESSURE COEFFICIENTS C_{pe}	0.9	0.8	0.7	0.6	0.5	0.4	0.3	0.2	0.1	0	-0.1	-0.2	-0.3	-0.4	-0.5	-0.6	-0.7	-0.8	-0.9	-1.0	-1.1	-1.2	-1.3	-1.4	-1.5	-1.6
WIND PRESSURE $q_F\ C_{pe}$	135	120	105	90	75	60	45	30	15	0	-15	-30	-45	-60	-75	-90	-105	-120	-135	-150	-165	-180	-195	-210	-215	-240

V_{30} = 179 MPH

$q = 0.00256\ V_{30}^2$ = 82 PSF

q_F = 150 PSF

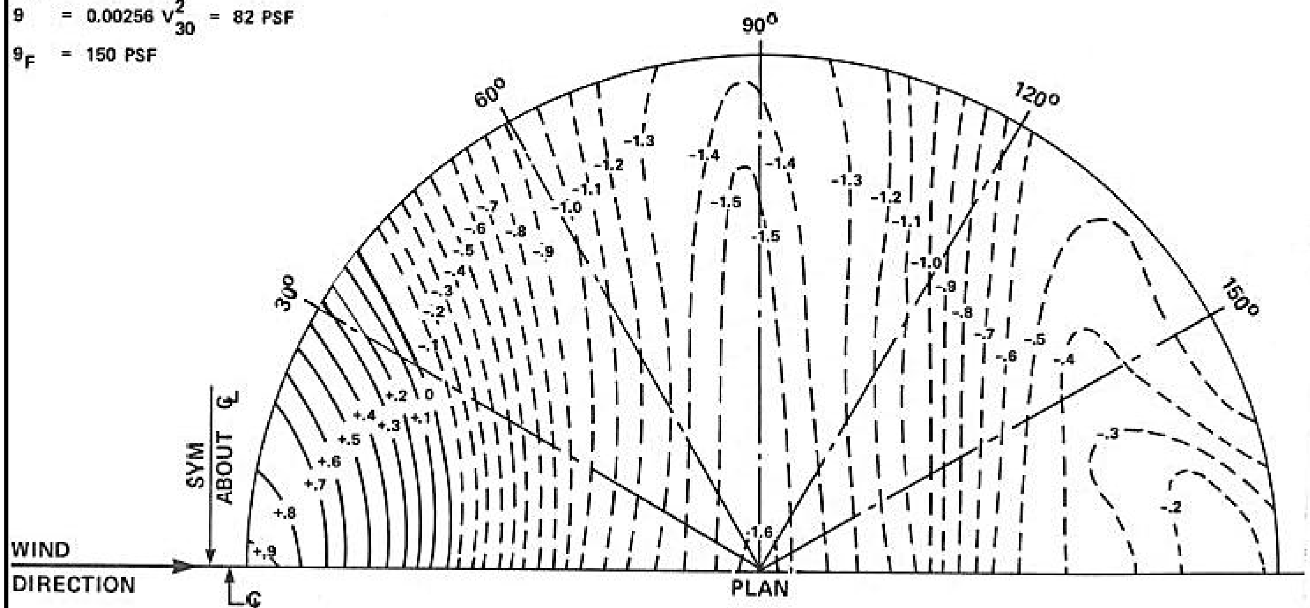


FIGURE 3.3.2-1

CONTAINMENT WALL TORNADO PRESSURE COEFFICIENTS AND DESIGN TORNADO PRESSURE

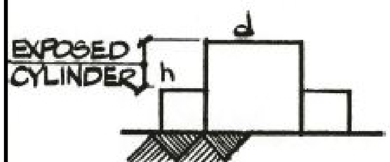
$V_T = 360$ mph

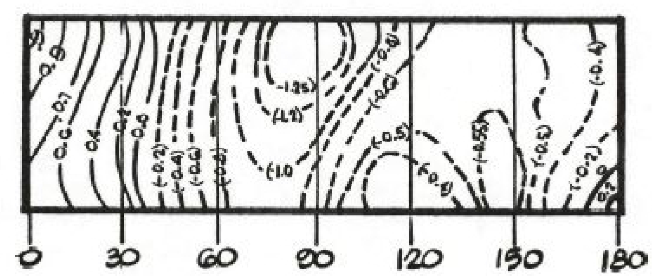
$q = 0.00256 V_T^2 = 332$ psf

$p = -3$ psi $= -432$ psf

$P_j = q\,C_{pe} + 0.5p$

PRESSURE COEFFICIENTS Cpe	0	0.2	0.4	0.6	0.7	0.8	0.9	-0.2	-0.4	-0.5	-0.55	-0.6	-0.8	-0.9	-1.0	-1.1	-1.2	-1.25
TORNADO WIND PRESSURE q Cpe	0	66	133	199	232	266	299	-66	-133	-166	-183	-199	-266	-299	-332	-365	-398	-415
DESIGN TORNADO PRESSURE q Cpe+0.5p	-216	-150	-83	-17	16	50	83	-282	-349	-382	-399	-415	-482	-515	-548	-581	-614	-631





PRESSURE COEFFICIENTS

FIGURE 3.3.2-2

CONTAINMENT DOME TORNADO PRESSURE COEFFICIENTS AND DESIGN TORNADO PRESSURE

PRESSURE COEFFICIENTS Cpc	0.9	0.8	0.7	0.6	0.5	0.4	0.3	0.2	0.1	0	-0.1	-0.2	-0.3	-0.4	-0.5	-0.6	-0.7	-0.8	-0.9	-1.0	-1.1	-1.2	-1.3	-1.4	-1.5	-1.6
TORNADO WIND PRESSURE q Cpc	299	266	232	199	166	133	100	66	33	0	-33	-66	-100	-133	-166	-199	-232	-266	-299	-332	-365	-398	-432	-465	-498	-531
DESIGN TORNADO PRESSURE q Cpc + 0.5p	83	50	16	-17	-50	-83	-116	-150	-183	-216	-249	-282	-316	-349	-382	-415	-448	-482	-515	-548	-581	-614	-648	-681	-714	-747

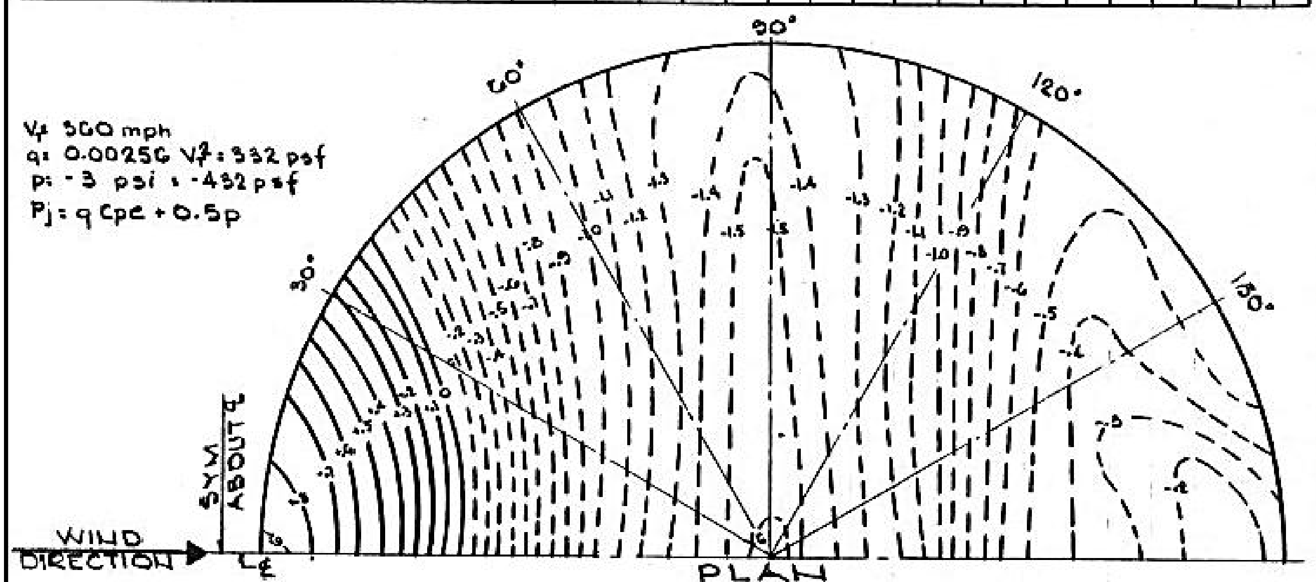


FIGURE 3.3.2-3

TORNADO EFFECTIVE VELOCITY PRESSURE PROFILE

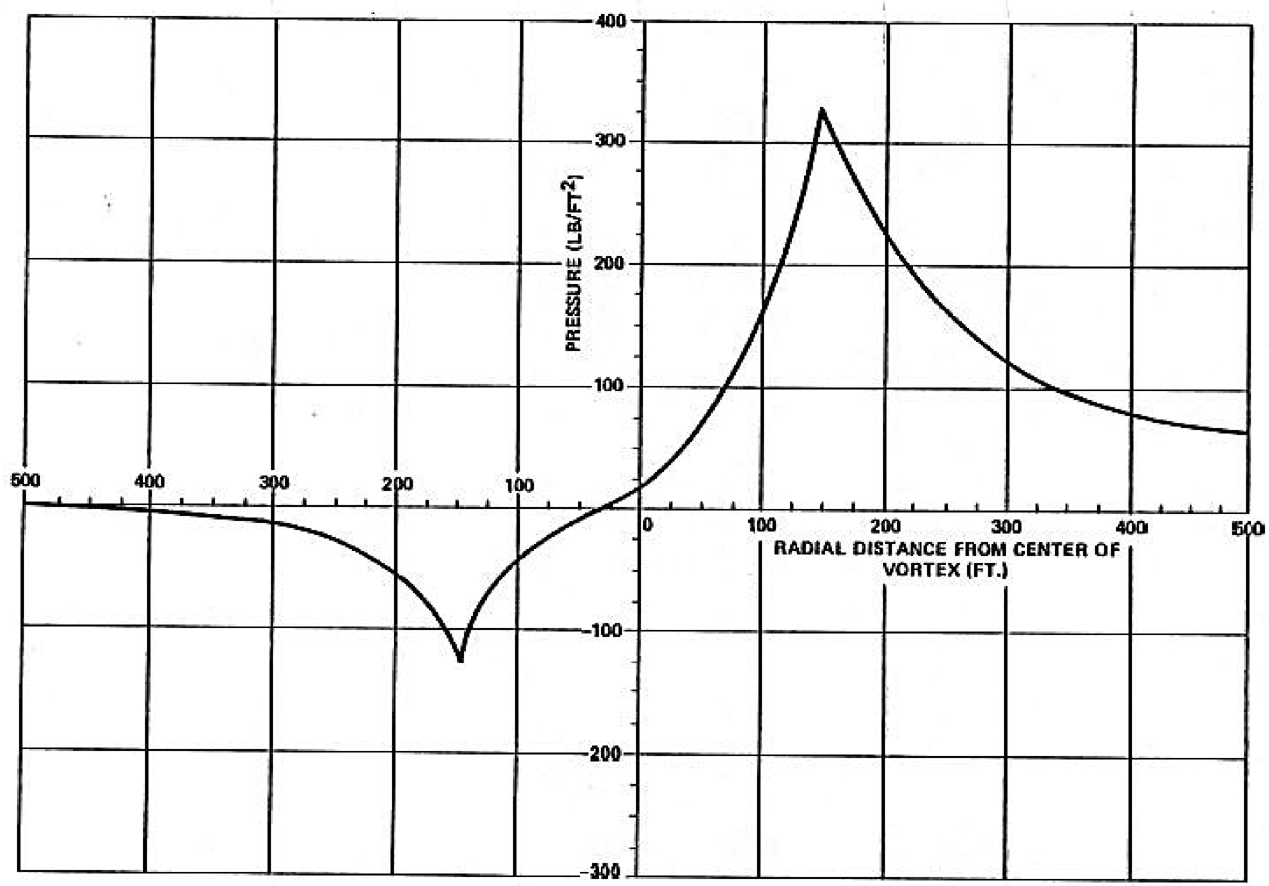


FIGURE 3.3.2-4

TORNADO ATMOSPHERIC PRESSURE FIELD PROFILE

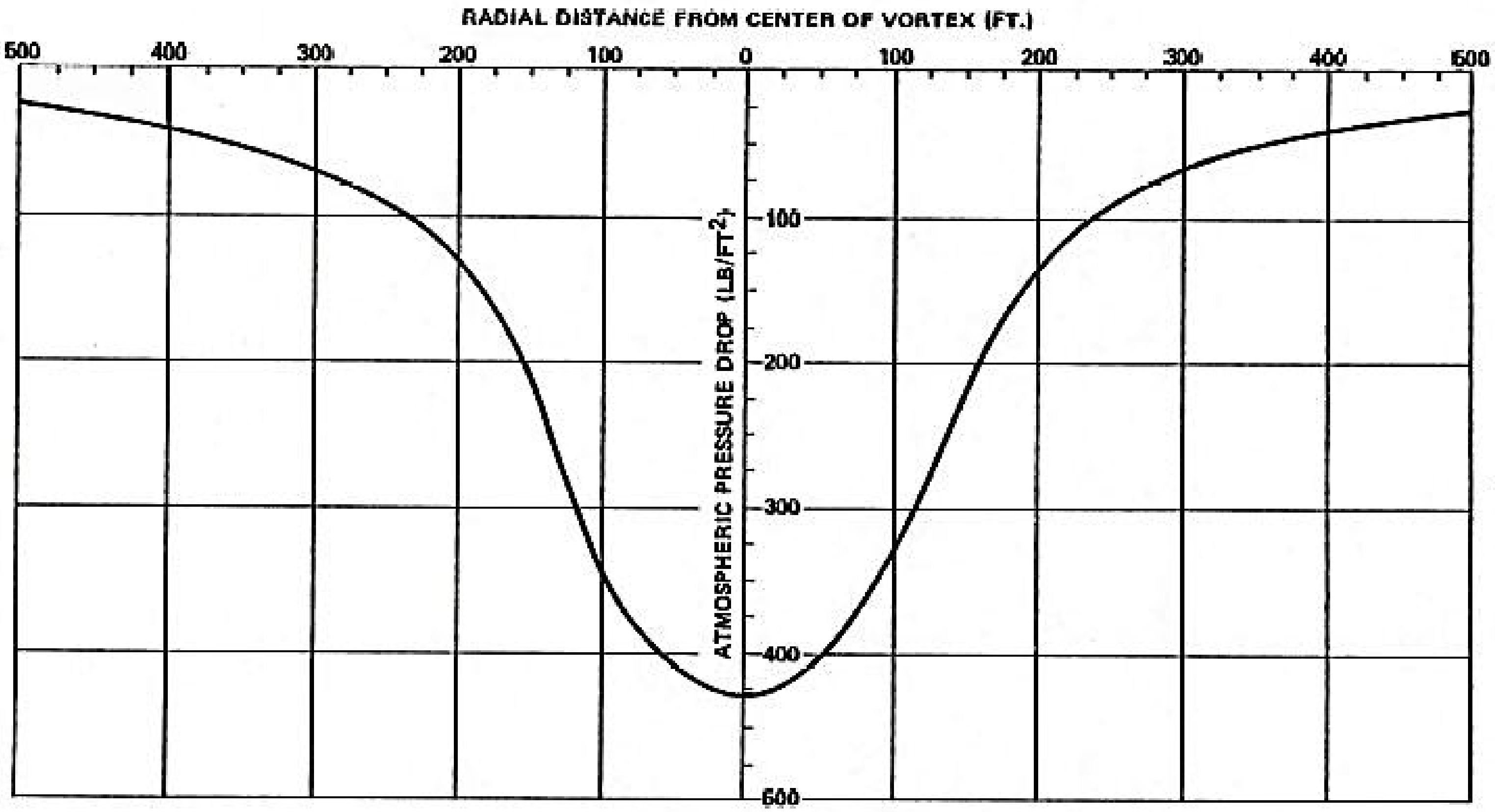


FIGURE 3.3.2-5

TORNADO PRESSURE DISTRIBUTION ON A STRUCTURE

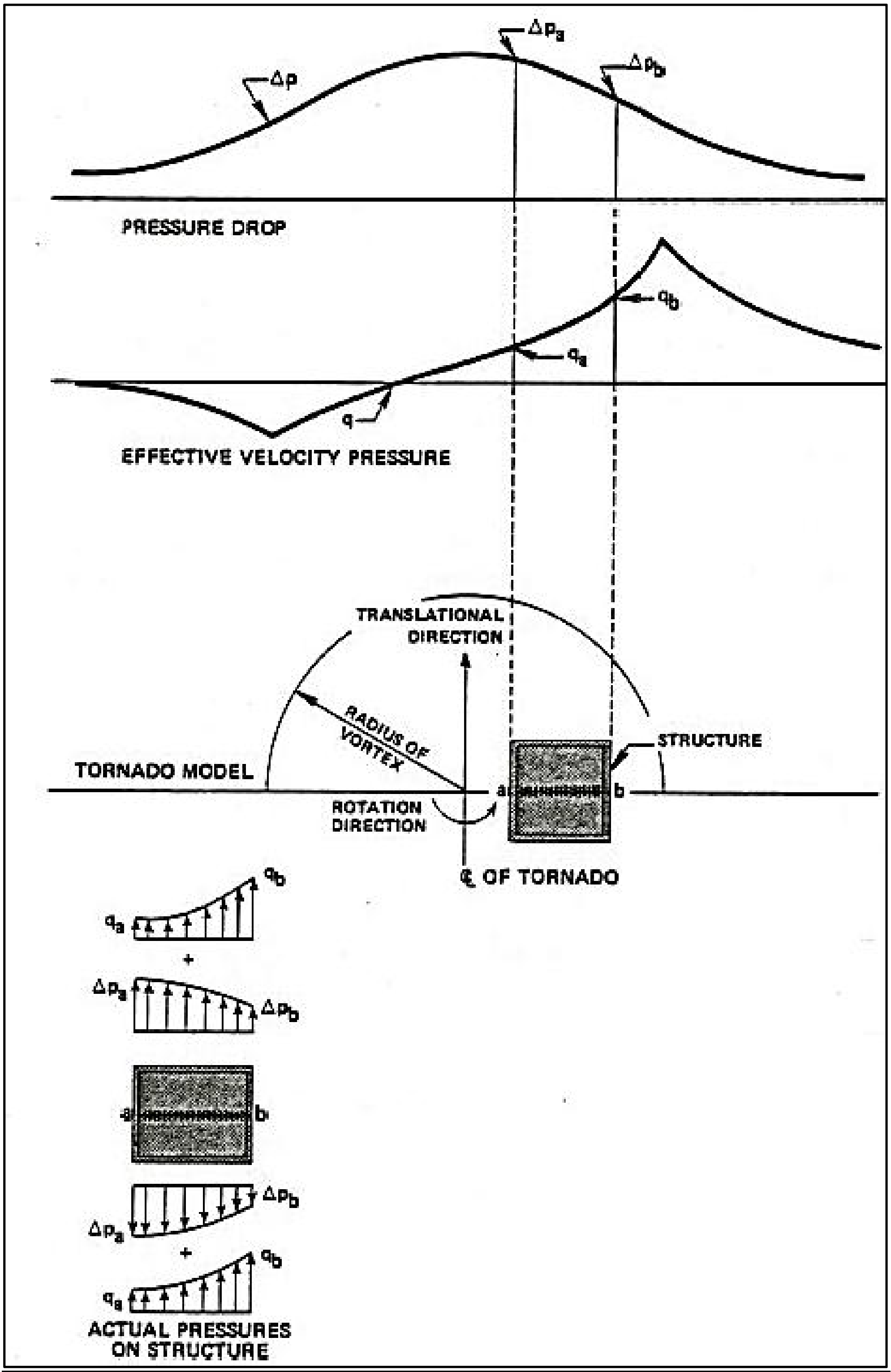


FIGURE 3.4.1-2

PLANT BUILDINGS WATERSTOPS IN SEISMIC GAP

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-3

FUEL HANDLING BUILDING EXTERIOR WALL ELEVATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-4

WASTE PROCESSING BUILDING EXTERIOR WALL ELEVATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-6

TANK BUILDING EXTERIOR WALL ELEVATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-7

TURBINE BUILDING EXTERIOR WALL PART ELEVATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-8

FUEL HANDLING UNLOADING AREA EXTERIOR WALL ELEVATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-9

DIESEL GENERATOR BUILDING EXTERIOR WALL ELEVATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-10

DIESEL FUEL OIL STORAGE TANK BUILDING EXTERIOR WALL ELEVATIONS SHEET 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-10

DIESEL FUEL OIL STORAGE TANK BUILDING EXTERIOR WALL ELEVATION SHEET 2

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-11

E.S.W. INTAKE SCREEN STRUCTURE WALL ELEVATIONS SHEET 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-11

E.S.W. INTAKE SCREEN STRUCTURE WALL ELEVATIONS SHEET 2

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-12

EMERGENCY SERVICE WATER AND COOLING TOWER MAKE-UP WATER INTAKE STRUCTURE WALL ELEVATIONS SHEET 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-12

EMERGENCY SERVICE WATER AND COOLING TOWER MAKE-UP WATER INTAKE STRUCTURE WALL ELEVATIONS SHEET 2

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-13

ELECTRICAL MANHOLE (TYPICAL)

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-14

AUXILIARY AND EMERGENCY POWER SYSTEM CABLES YARD DUCT RUNS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-15

CONCRETE CONTAINMENT STRUCTURE GENERAL ARRANGEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.4.1-16

CONTAINMENT STRUCTURE POROUS CONCRETE DRAINS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.5.1-01

SAFETY-RELATED STRUCTURES SYSTEMS AND COMPONENT PROTECTED AGAINST TORNADO MISSILES

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.5.1-03

LOW TRAJECTORY TURBINE MISSILE STRIKE ZONES

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6.1-1

LOSS OF REACTOR COOLANT ACCIDENT BOUNDARY LIMITS

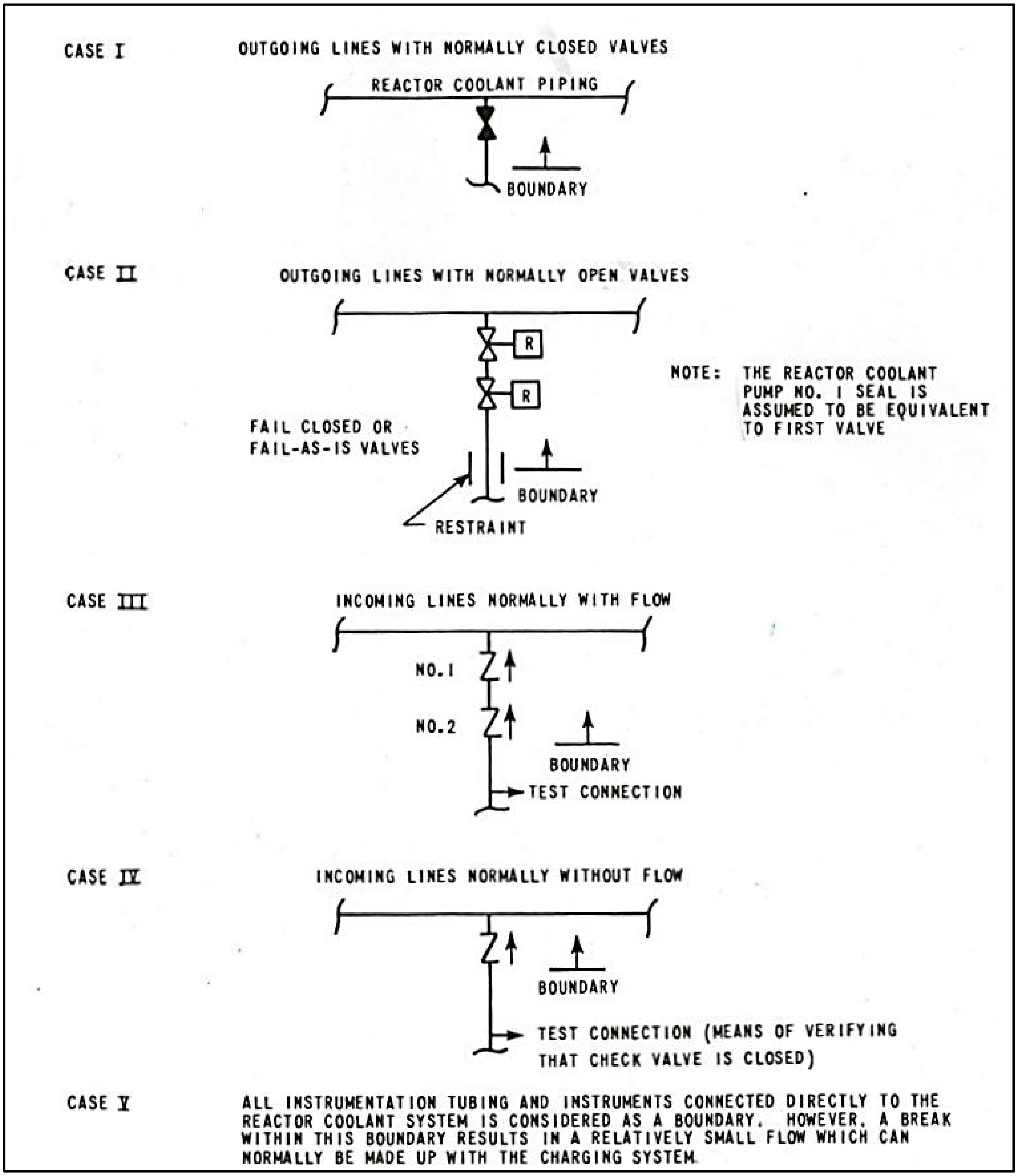


FIGURE 3.6.2-1

ILLUSTRATION OF MAIN STEAM AND FEEDWATER PIPING OUTSIDE CONTAINMENT

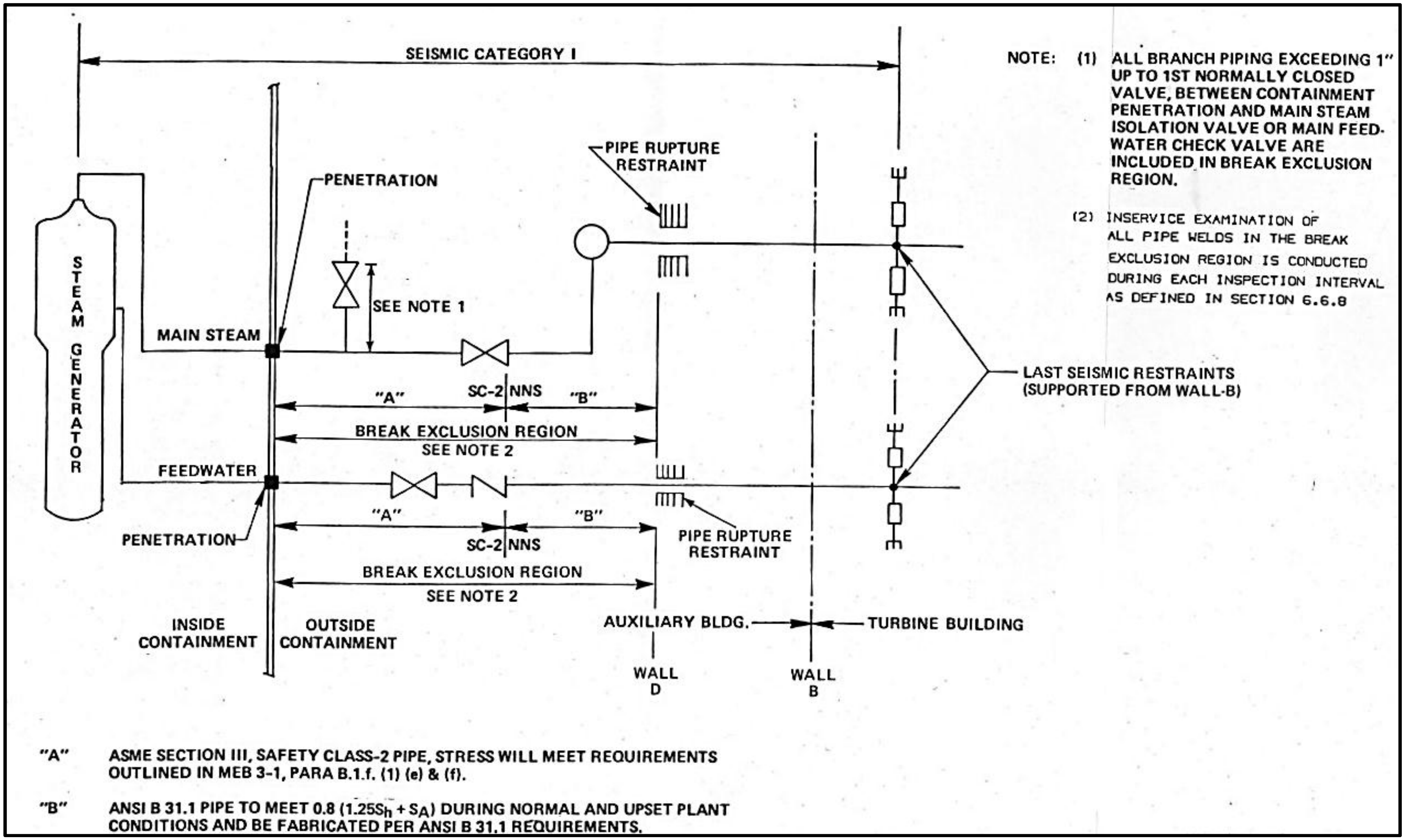
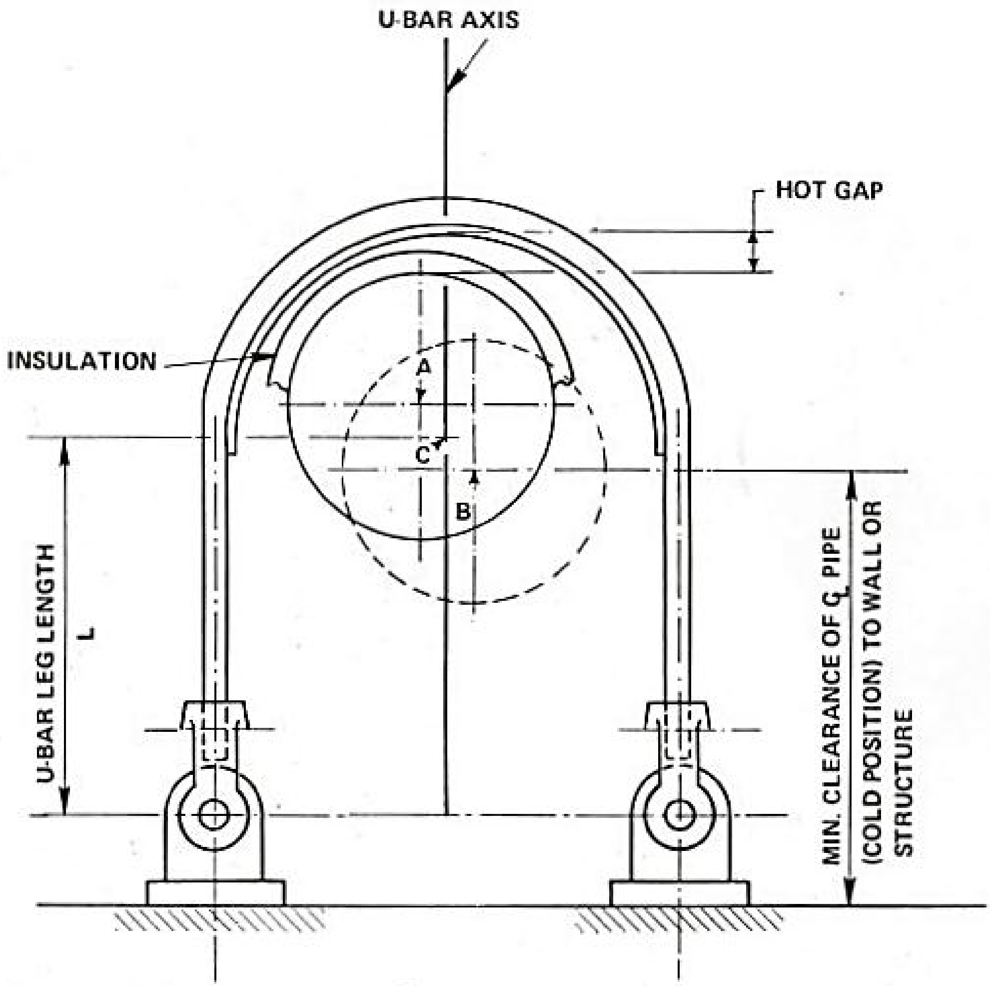


FIGURE 3.6.2-2

U-BAR RESTRAINT



A = CENTERLINE OF PIPE IN HOT POSITION

B = CENTERLINE OF PIPE IN COLD POSITION

C = CENTER OF U-BAR CURVED PORTION; GENERALLY. C IS THE MID-POINT OF STRAIGHT LINE AB

FIGURE 3.6.2-3

PLATE-TYPE RESTRAINT

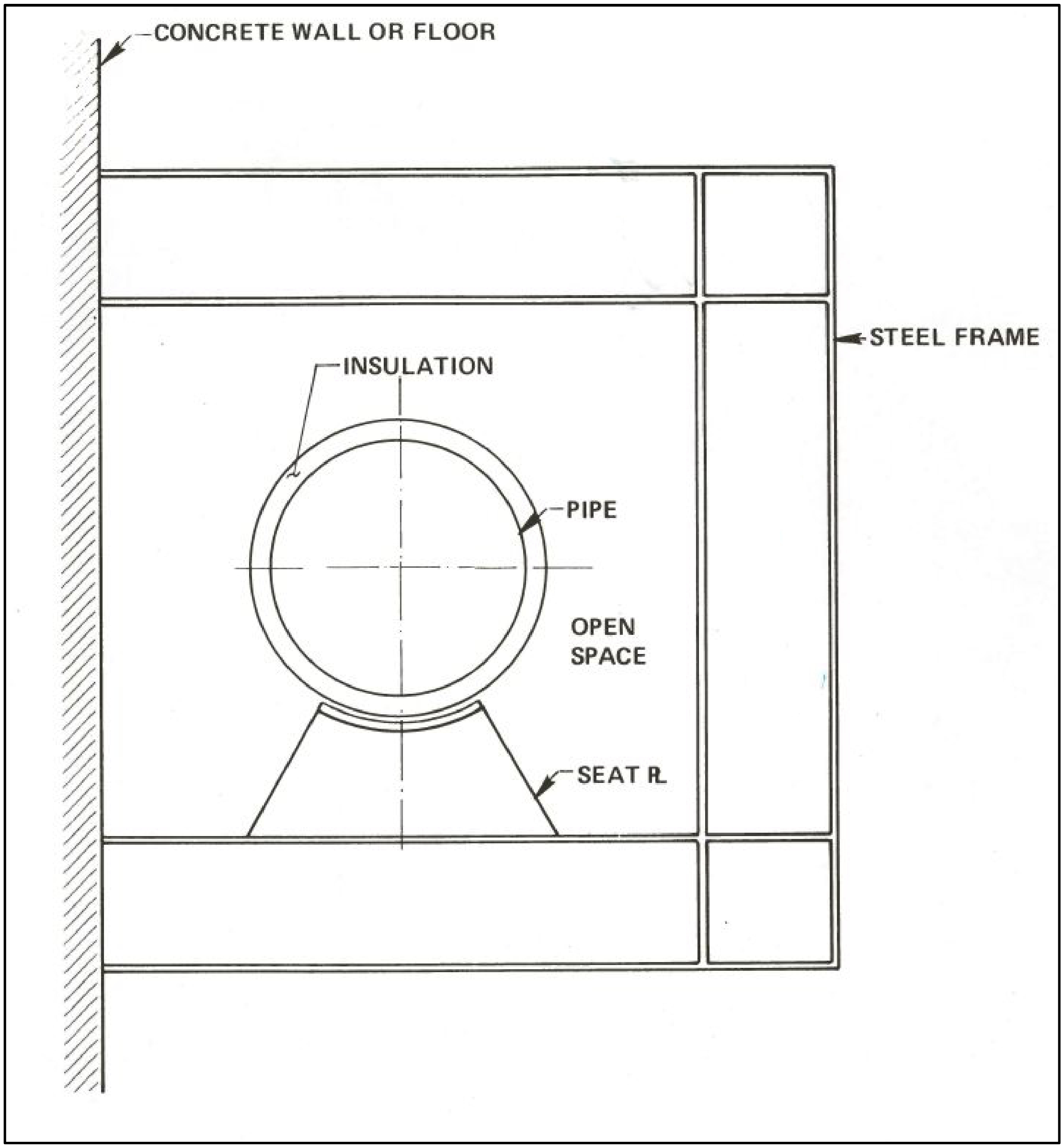


FIGURE 3.6.2-4

CRUSHABLE MATERIAL – TYPE RESTRAINT

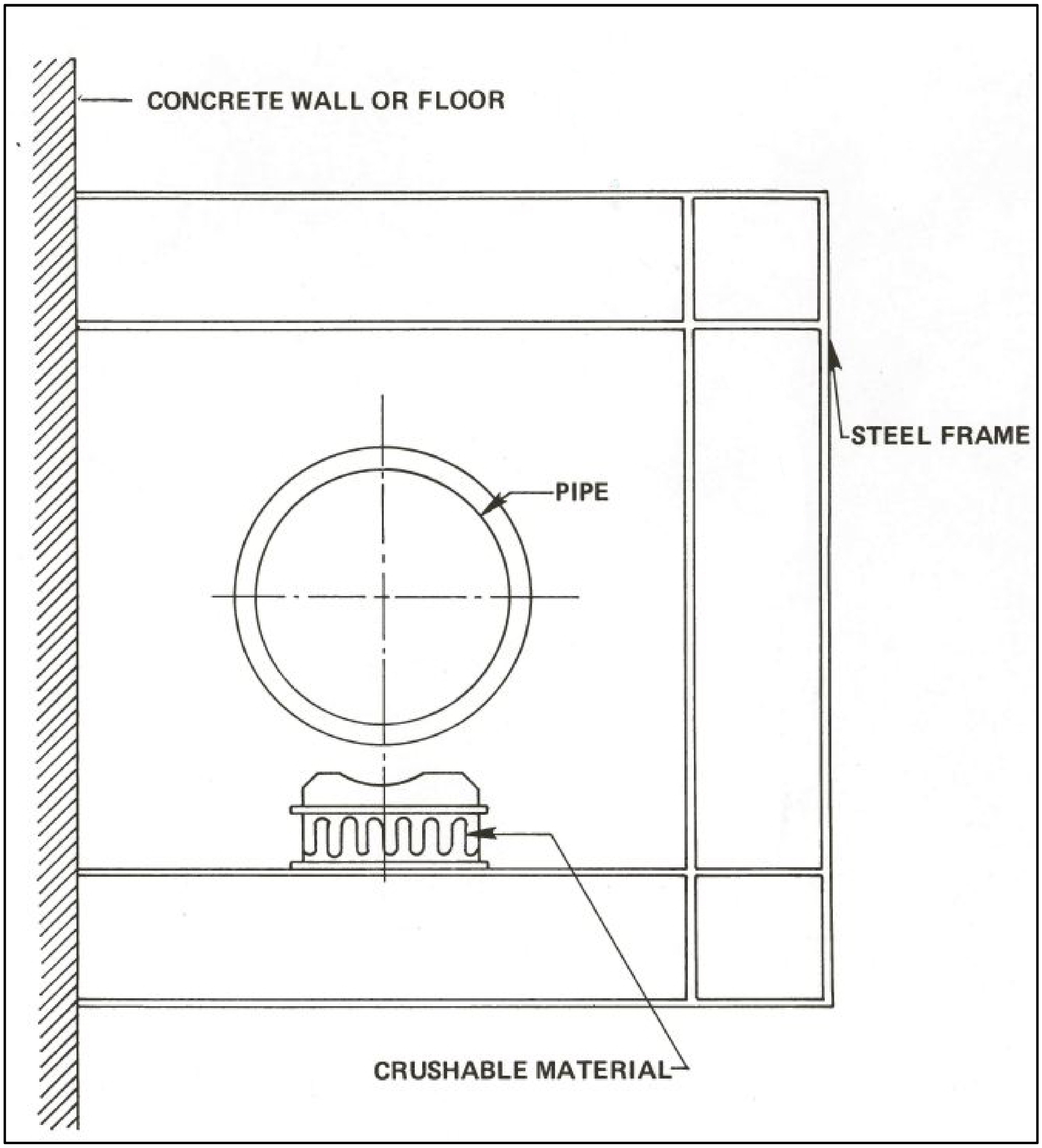


FIGURE 3.6.2-5

JET DIVERGENCE FOR WET STEAM AND SATURATED WATER

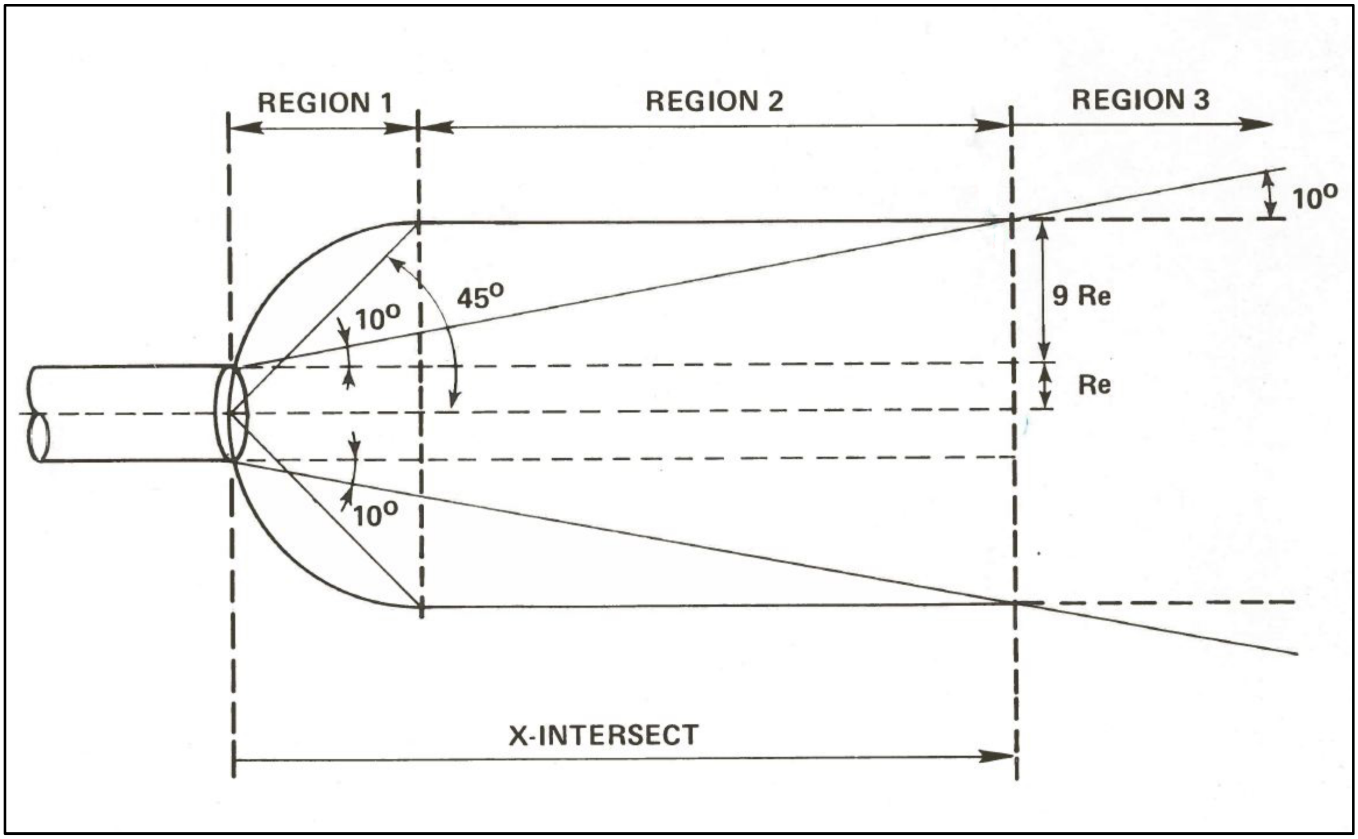


FIGURE 3.6.2-6

JET DIVERGENCE FOR DRY STEAM & NONFLASHING WATER

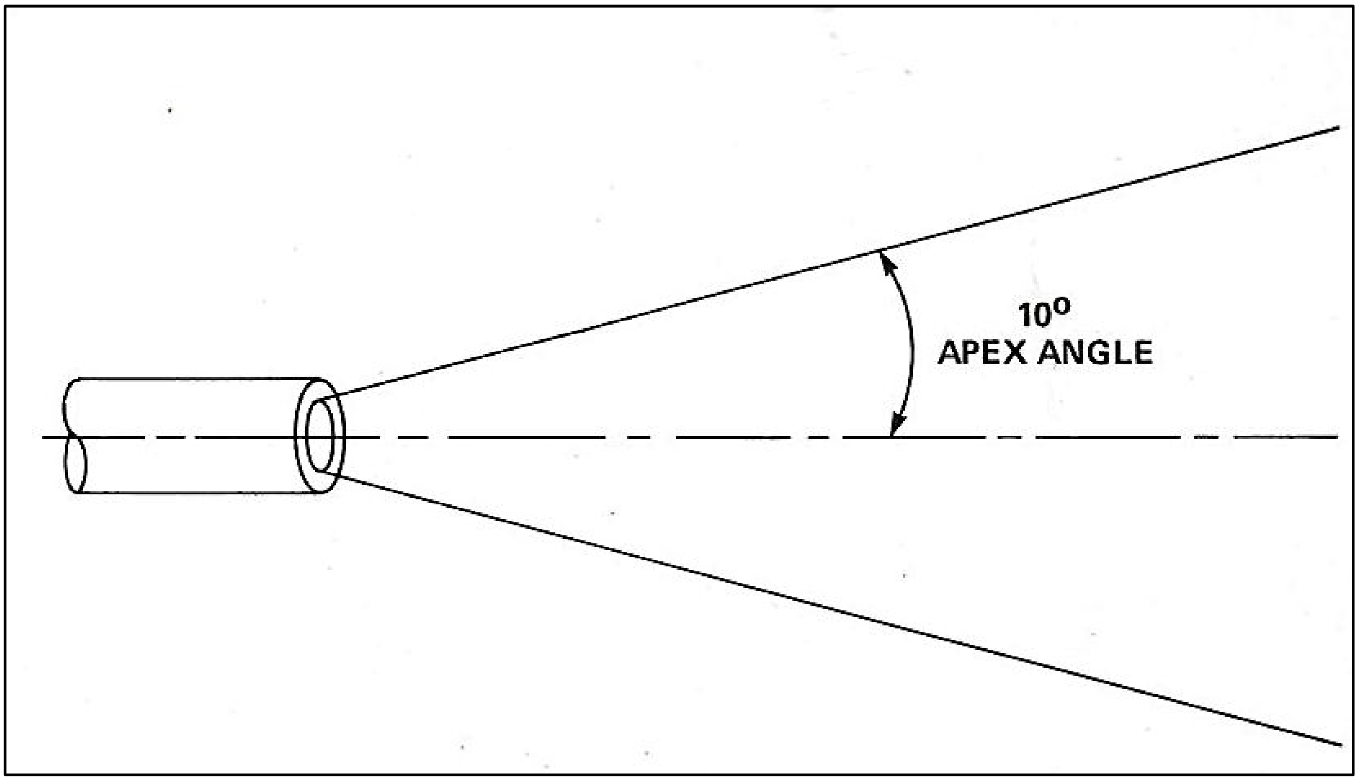


FIGURE 3.6.2-7

RADIAL (DISK-SHAPED) JET EMANATING FROM CIRCUMFERENTIAL BREAK

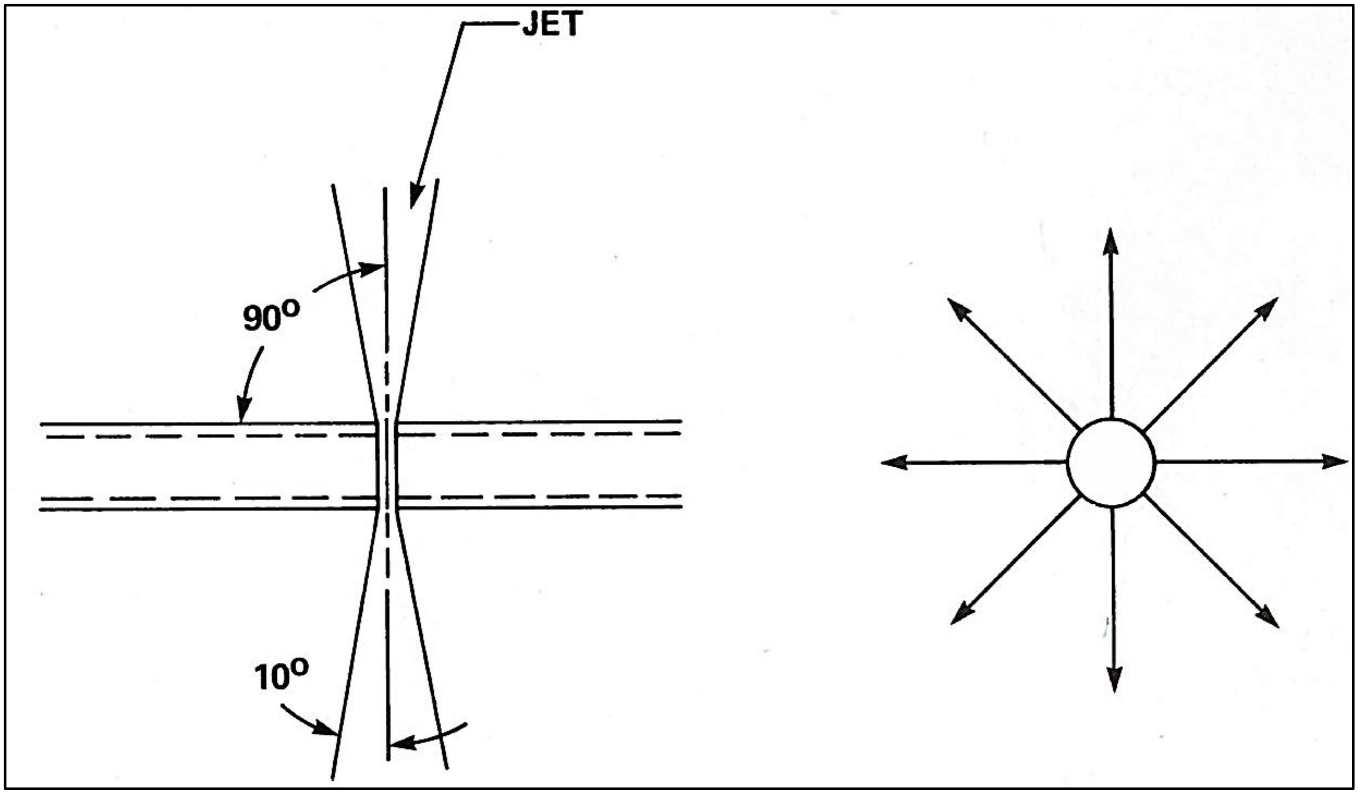


FIGURE 3.6.2-8

CIRCUMFERENTIAL BREAK-LIMITED SEPARATION

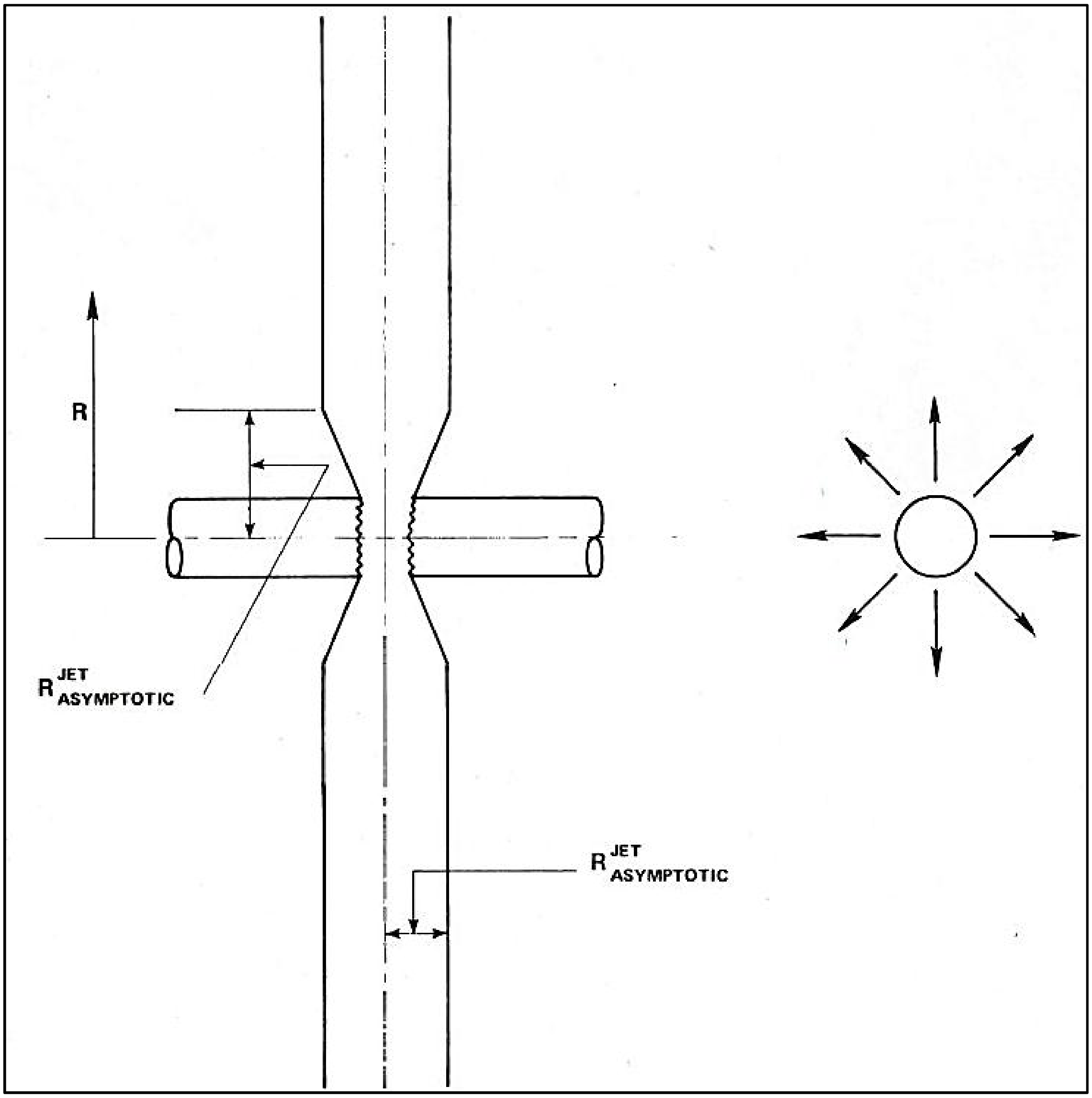


FIGURE 3.6A-1-CALC

REACTOR CONTAINMENT BUILDING
SUMMARY OF PIPE BREAK LOCATIONS
MAIN STEAM & FEEDWATER PIPING

SUMMARY OF CALCULATIONS

MS & FD WATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI- TERIA(a)	BREAK TYPE(b)
3Ø1	R-HFW-67-1	ST. GEN. 1A-SN NOZZLE	1.Ø14	TE	G
322	R-HFW-67-5	PENET M-4 INSIDE CONTAINMENT	Ø.79	TE	G
1	R-HMS-1-1	STM. GEN. NOZZLE	Ø.76	TE	G
23	R-HMS-1-5	PENET M-3 INSIDE CONTAINMENT	Ø.69	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-MS/FW-1155-39 FIGURE 3.6A-1, FIGURE 3.6A-5, FIGURE 3.6A-7

FIGURE 3.6A-1-CALC (Continued)

SUMMARY OF CALCULATIONS

MAIN STEAM PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	APPLICABLE CRITERIA	BREAK TYPE*
1	R-HMS-2-1	STM GEN 1B-SN NOZZLE	Ø.54	TERMINAL END	ALL BREAKS ARE GUILLOTINE TYPE (SEE PAGE 8)
13	R-HMS-2-5	CONT. PENETRATION M-2	Ø.84	TERMINAL END	

FEEDWATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	APPLICABLE CRITERIA	BREAK TYPE*
3Ø1	R-HFW-68-1	STM GEN 1B-SN NOZZLE	1.Ø85	TERMINAL END	ALL BREAKS ARE GUILLOTINE TYPE (SEE PAGE 8)
313	R-HFW-68-6	SR ELBOW	1.25	HRI	
3133	R-HFW-68-7	BUTT WELD	1.212	HRI	
318	R-HFW-68-5	CONT. PENETRATION M-5	1.2	TERMINAL END	

* G = GUILLOTINE (CIRCUMFERENTIAL)
L = SLOT (LONGITUDINAL)

REF. NO:
CALC. NO: PRA-W-MS/FW-1157-3, FIGURE 3.6A-1, FIGURE 3.6A-5, FIGURE 3.6A-7

FIGURE 3.6A-1-CALC (Continued)

SUMMARY OF CALCULATIONS

MS & FW PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (a)	BREAK TYPE (b)
1	R-HMS-3-1	STM GEN 1C-SN NOZZLE	Ø.45	TE	G
23	R-HMS-3-5	PENET M-1 INSIDE CONT.	Ø.64	TE	G
3Ø1	R-HFW-69-1	STM. GEN. NOZZLE	1.1Ø8	TE	G
325	R-HFW-69-5	PENET M6 INSIDE CONT.	Ø.72	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

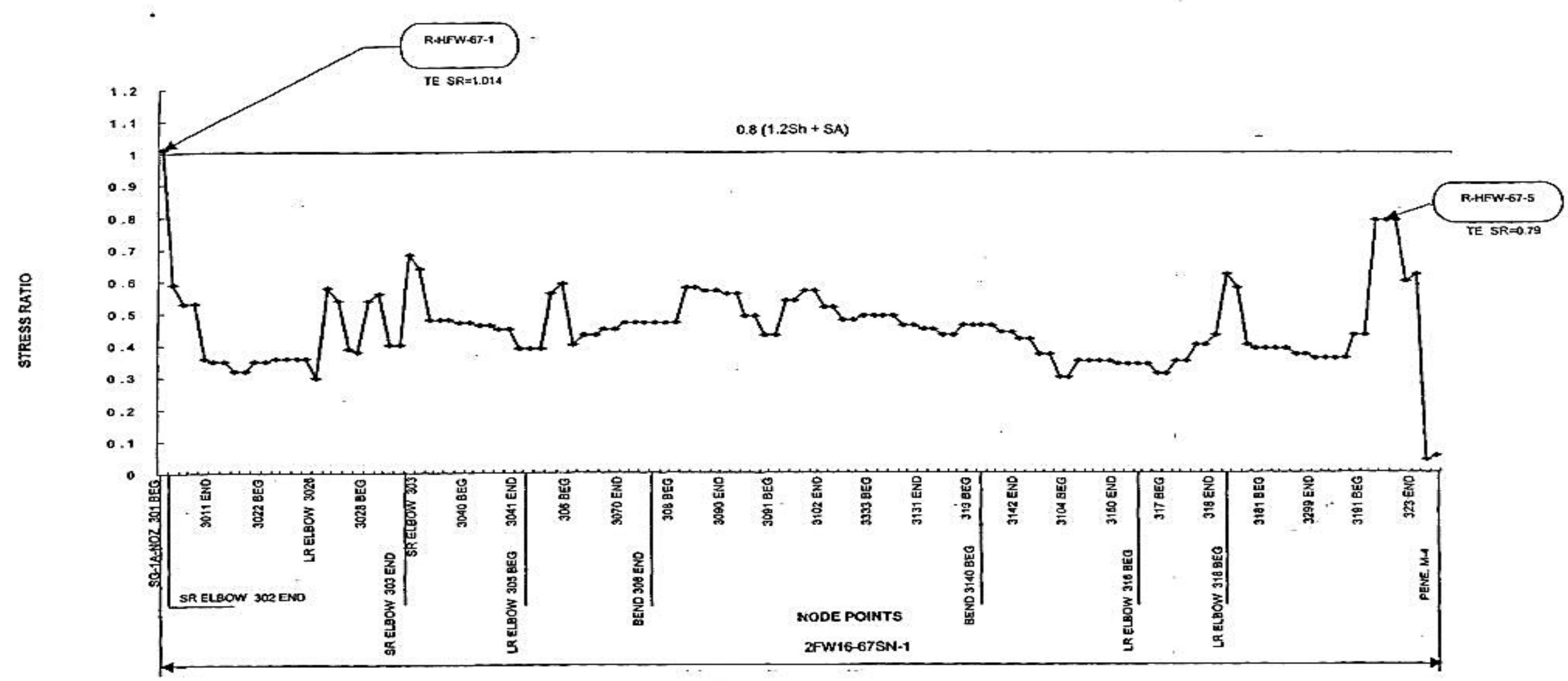
MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-MS/FW-1156-4Ø FIGURE 3.6A-1, FIGURE 3.6A-5, FIGURE 3.6A-7

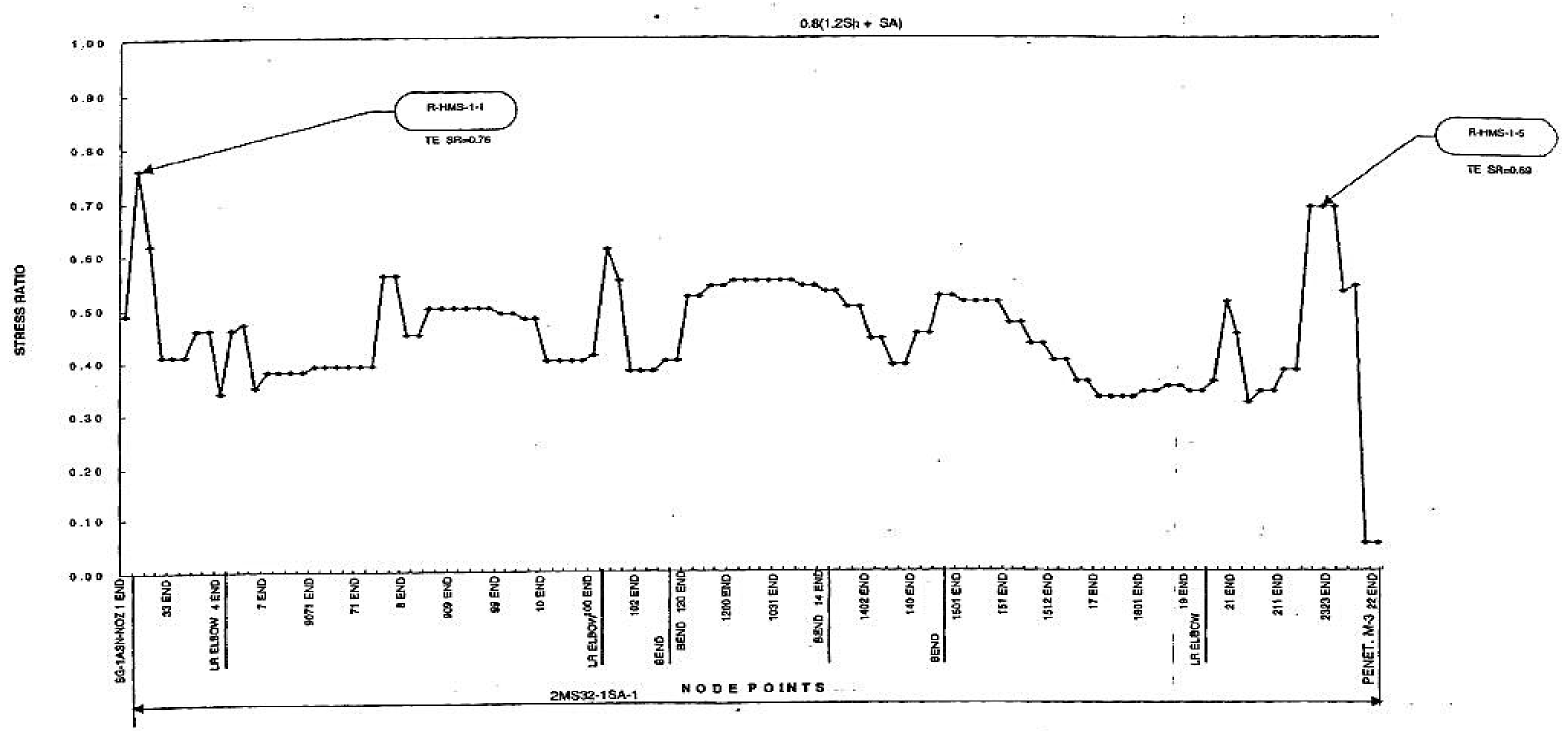
FIGURE 3.6A-1-PLOT-A

REACTOR & REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS MAIN STEAM & FEEDWATER PIPING LOOP #1



REF. NO.:
CALC. NO.: PRA-W-MS/FW-1155-39 FIGURE 3.6A-1, FIGURE 3.6A-5, FIGURE 3.6A-7, AND FIGURE 3.6A-1-CALC

FIGURE 3.6A-1-PLOT-A (Continued)



REF. NO.:
CALC. NO.: PRA-W-MS/FW-1155-39, FIGURE 3.6A-1, FIGURE 3.6A-5, FIGURE 3.6A-7, AND FIGURE 3.6A-1-CALC

FIGURE 3.6A-1-PLOT-B

REACTOR & REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS MAIN STEAM & FEEDWATER PIPING LOOP #2

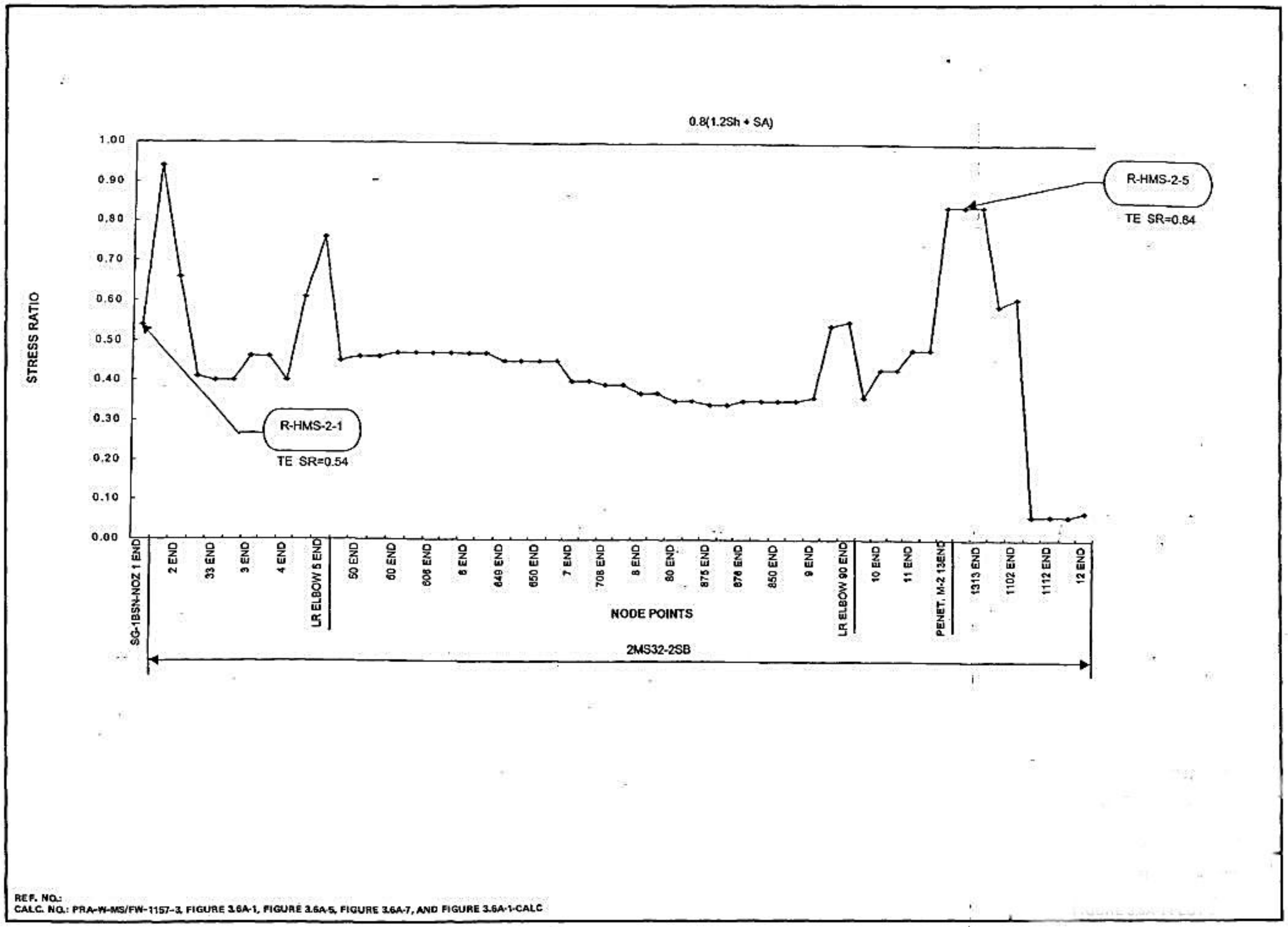


FIGURE 3.6A-1-PLOT-B (Continued)

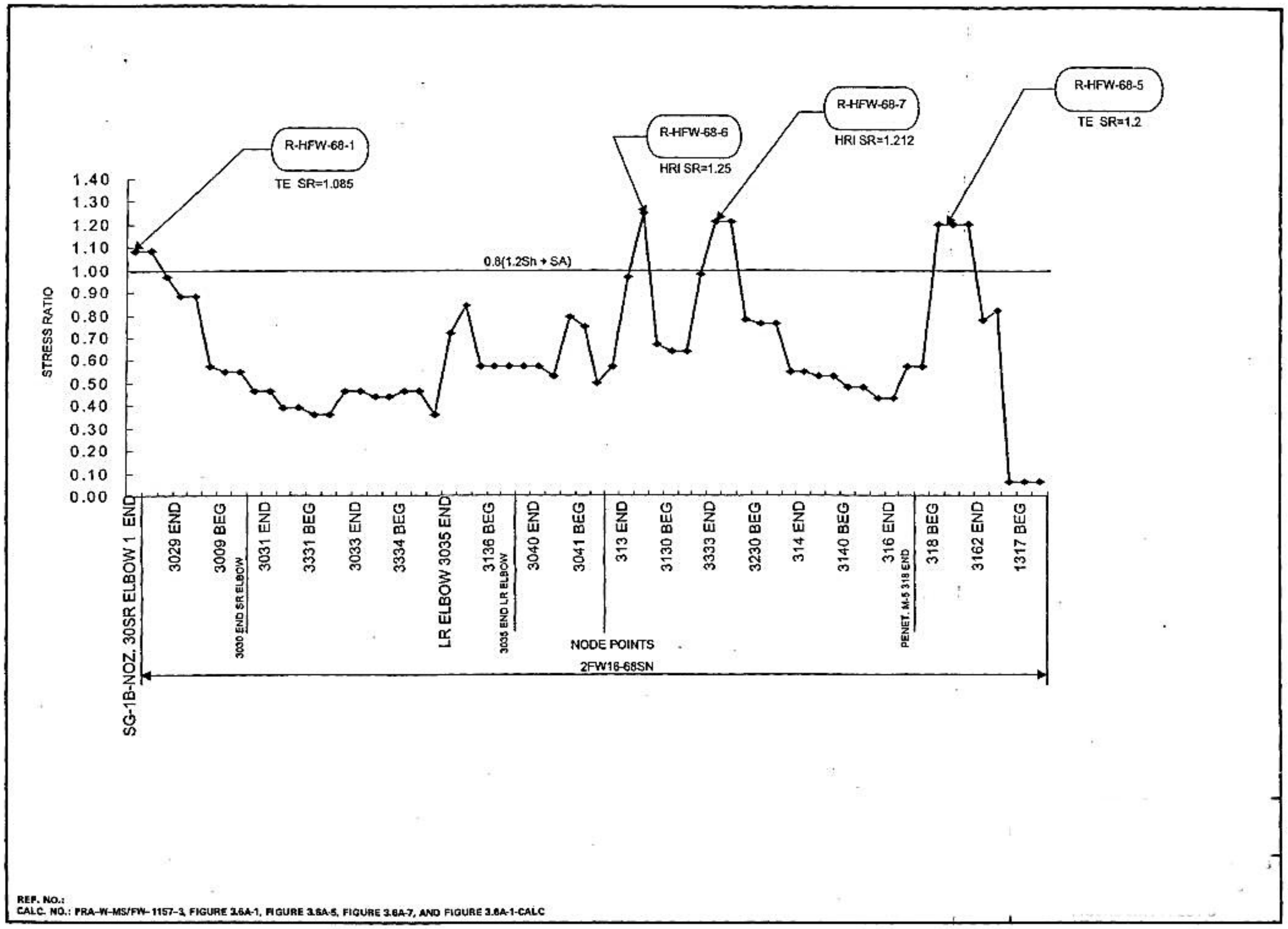
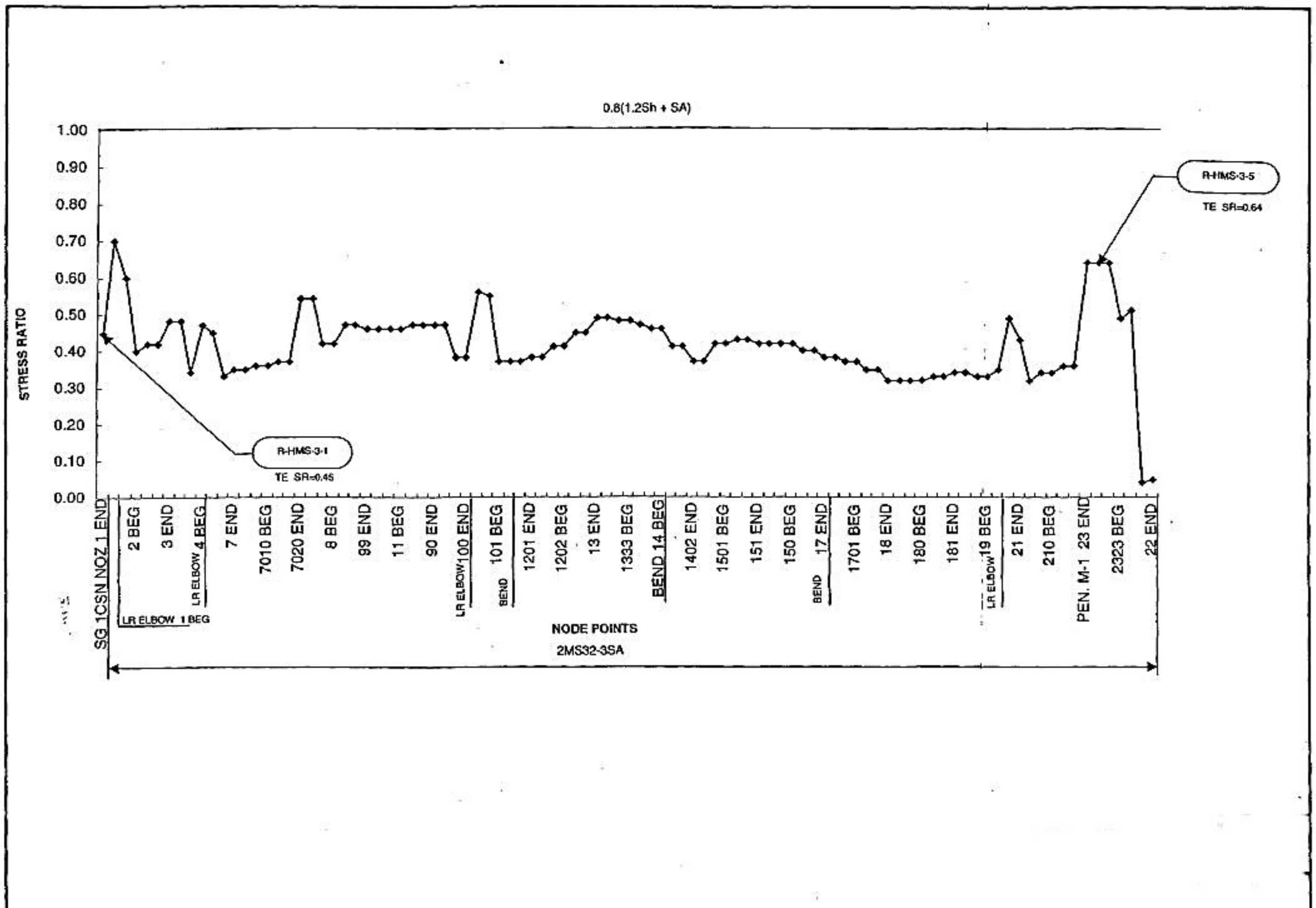


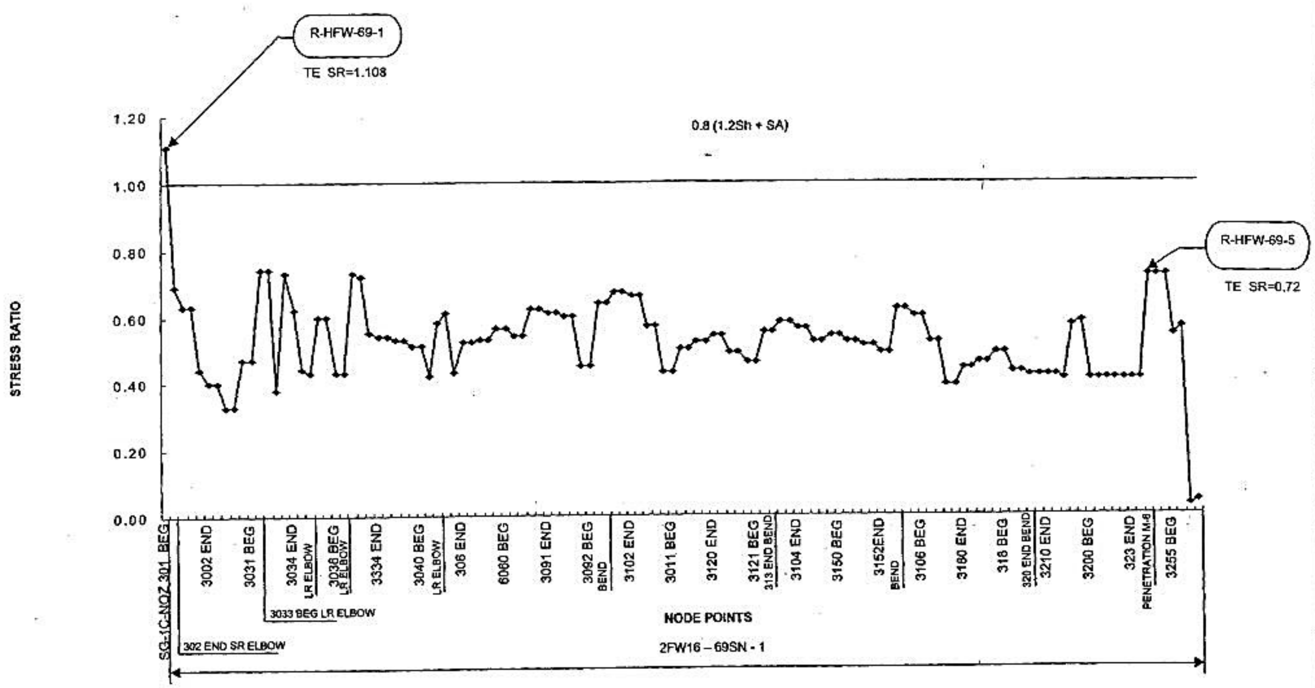
FIGURE 3.6A-1-PLOT-C

REACTOR & REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS MAIN STEAM & FEEDWATER PIPING LOOP #3



REF. NO.:
CALC. NO.: PRA-W-MS/FW-1156-40 FIGURE 3.6A-1, FIGURE 3.6A-6, FIGURE 3.6A-7, AND FIGURE 3.6A-1-CALC

FIGURE 3.6A-1-PLOT-C (Continued)



REF. NO.:
CALC. NO.: PRA-W-MS/FW-1156-40 FIGURE 3.6A-1, FIGURE 3.6A-5, FIGURE 3.6A-7, AND FIGURE 3.6A-1-CALC

FIGURE 3.6A-5-CALC

REACTOR & REACTOR AUXILIARY
SUMMARY OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING

SUMMARY OF CALCULATIONS

AUXILIARY FEEDWATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
1	A-HAF-1-1	NOZZ. IA-SA ST.GEN. AUX FD.	Ø.292	TE	G
426	A-HAF-4-1	ANCHOR POINT	Ø.463	TE	G
238	A-HAF-1-4	ANCHOR POINT	Ø.46Ø	TE	G
80	A-HAF-6-1	ANCHOR POINT	Ø.234	TE	G
498	A-HAF-3-1	NOZZ. IB-SB ST.GEN. AUX FD.	Ø.234	TE	G
459	A-HAF-13-2	FLOW RESTRICTOR	Ø.372	TE	G
10	A-HAF-12-2	FLOW RESTRICTOR	Ø.369	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-AF-71-1-7, FIGURE 3.6A-5, FIGURE 3.6A-8

FIGURE 3.6A-5-CALC (Continued)

SUMMARY OF CALCULATIONS

AUXILIARY FEEDWATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
238	A-HAF-1-4	ANCHOR POINT	Ø.27	TE	G
2438	A-HAF-94-1	VALVE	Ø.32	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-AF-71-5-11, FIGURE 3.6A-5, FIGURE 3.6A-8

FIGURE 3.6A-5-CALC (Continued)

SUMMARY OF CALCULATIONS

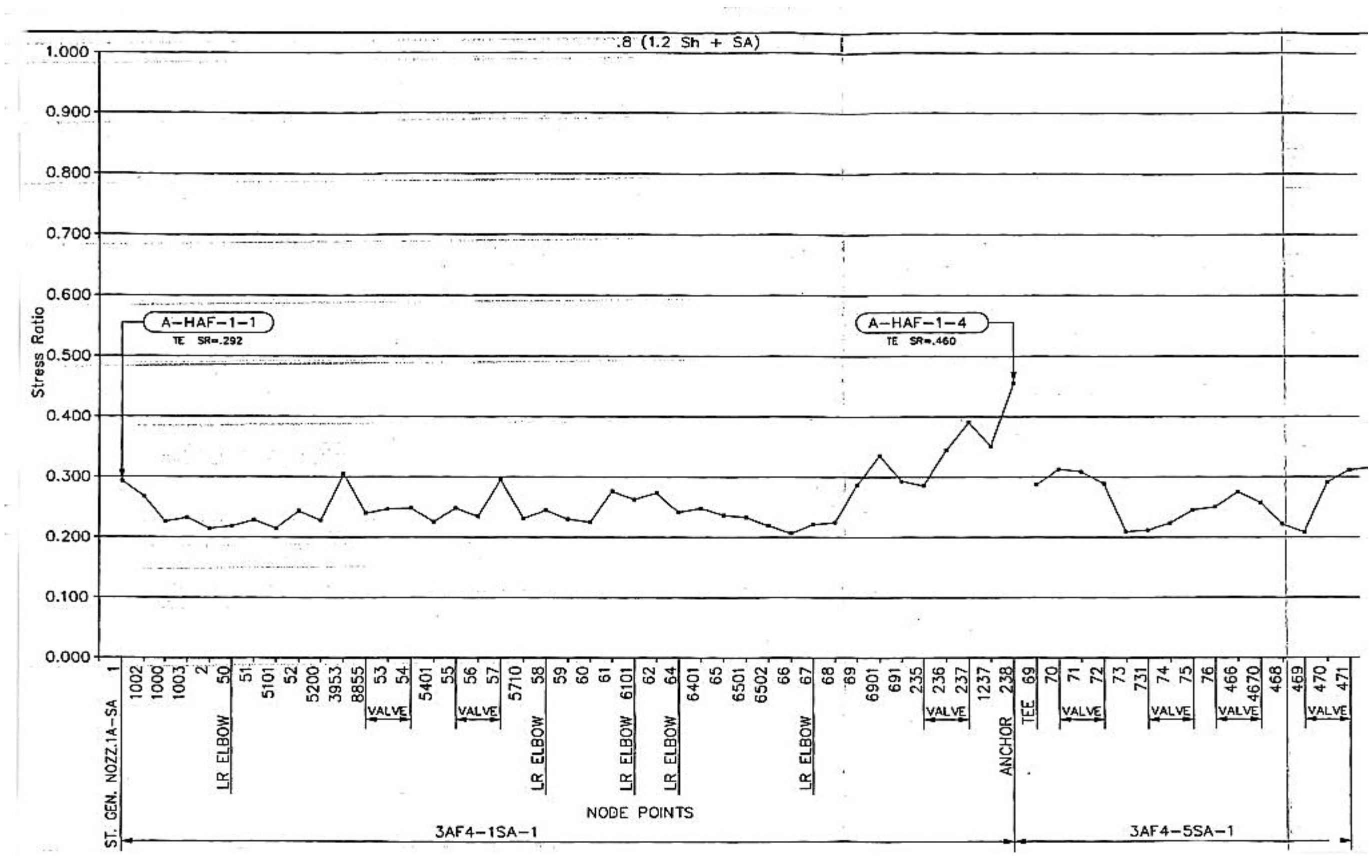
THE AUXILIARY FEEDWATER STEAM DRIVEN TURBINE PUMP DOES NOT OPERATE UNDER NORMAL PLANT CONDITIONS. THEREFORE, ALL ASSOCIATED PIPING FROM THE AF PUMP NOZZ 1X-SAB UP TO AND INCLUDING THE CHECK VALVES 3AF-V-31SB-1, 3AF-V-34SB-1 & 3AF-V-37-SB-1 ARE CONSIDERED MODERATE ENERGY PIPING AND ONLY POSTULATED CRACKS WILL OCCURE IN FLOODING ANALYSIS.

SINCE NO BREAKS ARE POSTULATED, NO PIPE WHIP RESTRAINTS ARE DESIGNATED WITHIN THE SUBJECT PIPING.

REF. NO.:
CALC. NO: PRA-W-AF-71-2-8, FIGURE 3.6A-5, FIGURE 3.6A-8

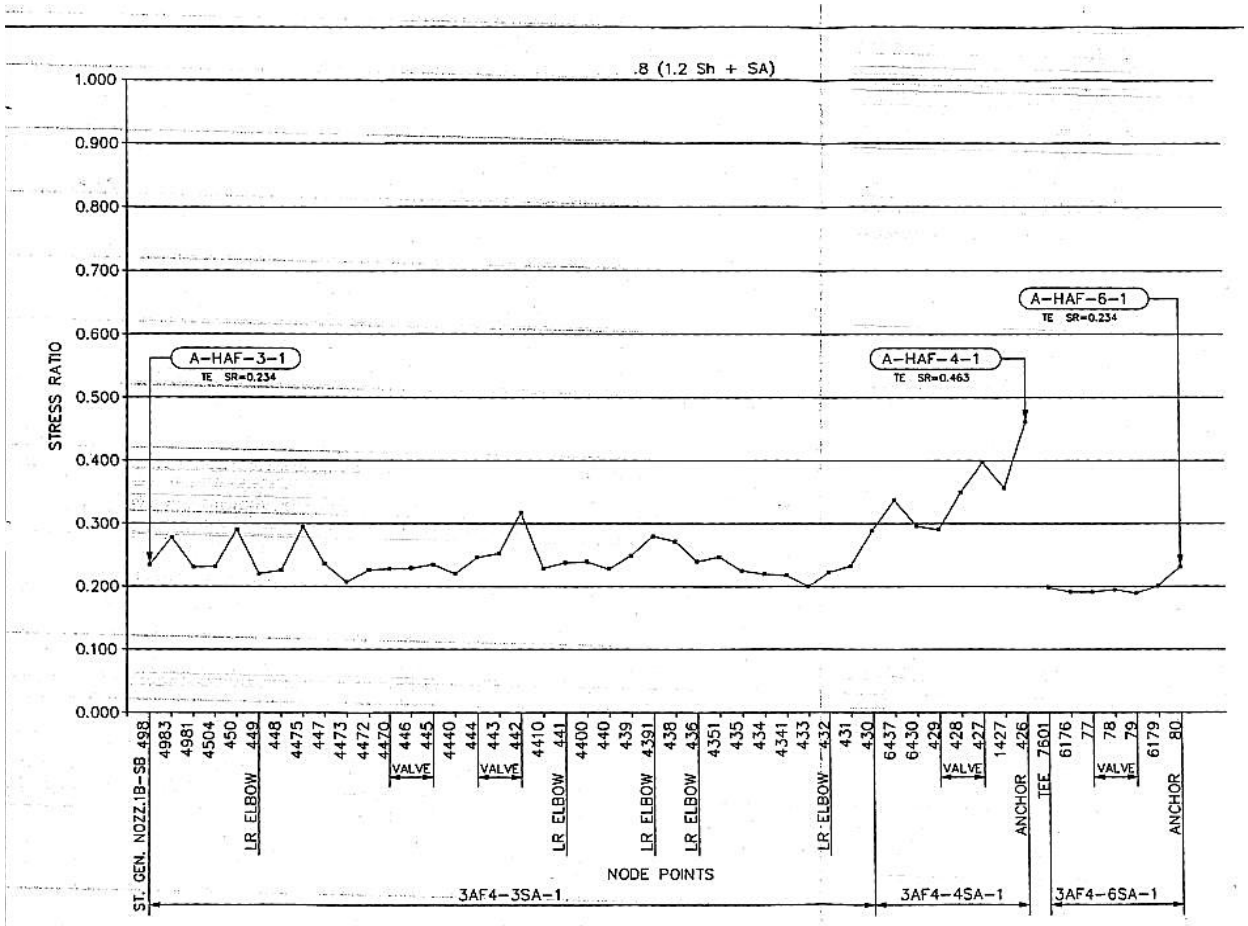
FIGURE 3.6A-5-PLOT-A

REACTOR & REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS AUXILIARY FEEDWATER PIPING



REF.NO.
CALC. NO.: PRA-W-AF-71-1-7, FIGURE 3.6A-5, FIGURE 3.6A-7-CALC

FIGURE 3.6A-5-PLOT-A (Continued)



REF.NO.
CALC. NO.: PRA-W-AF-71-1-7, FIGURE 3.6A-5, FIGURE 3.6A-7-CALC

FIGURE 3.6A-5-PLOT-A (Continued)

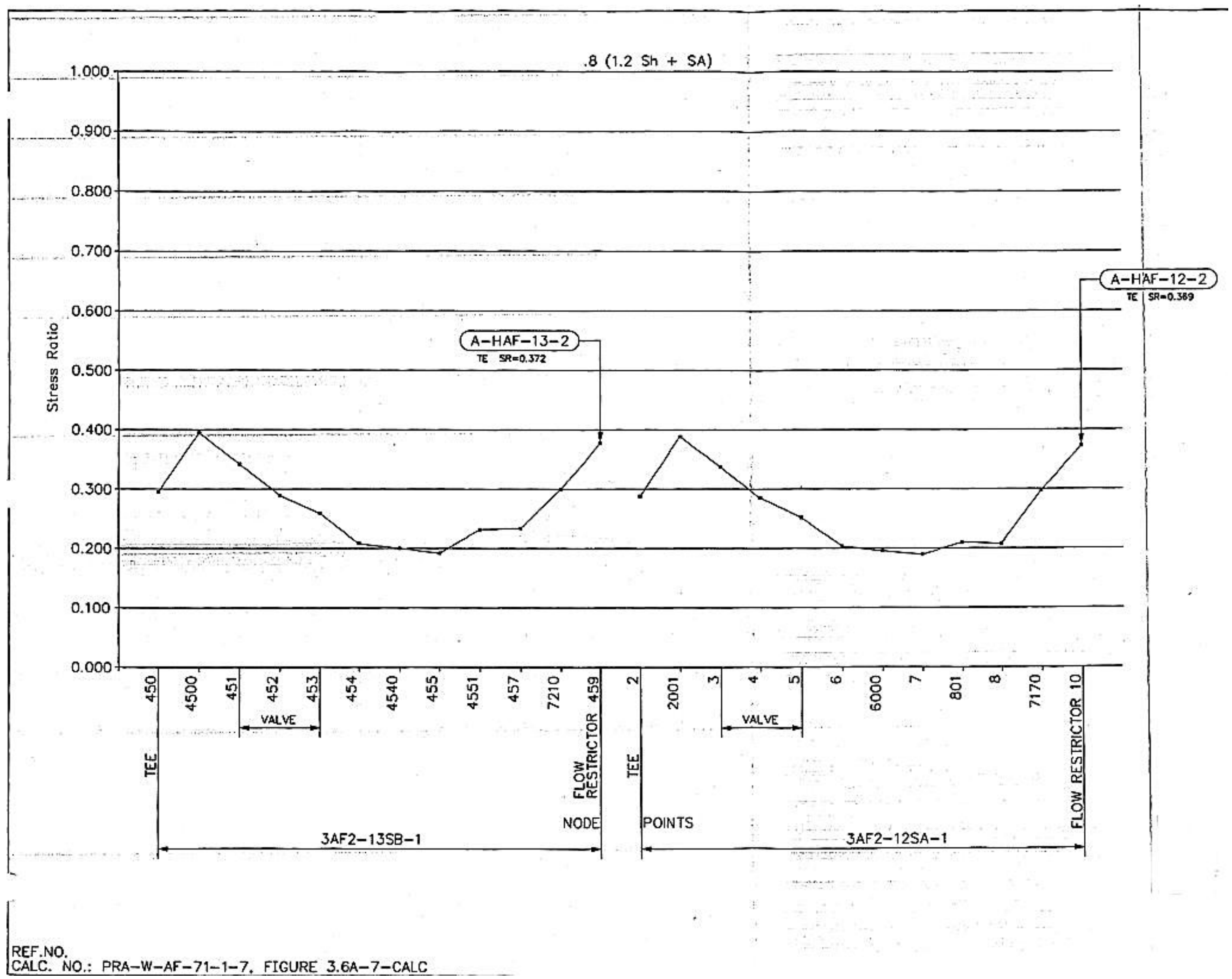


FIGURE 3.6A-5-PLOT-B

REACTOR & REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING

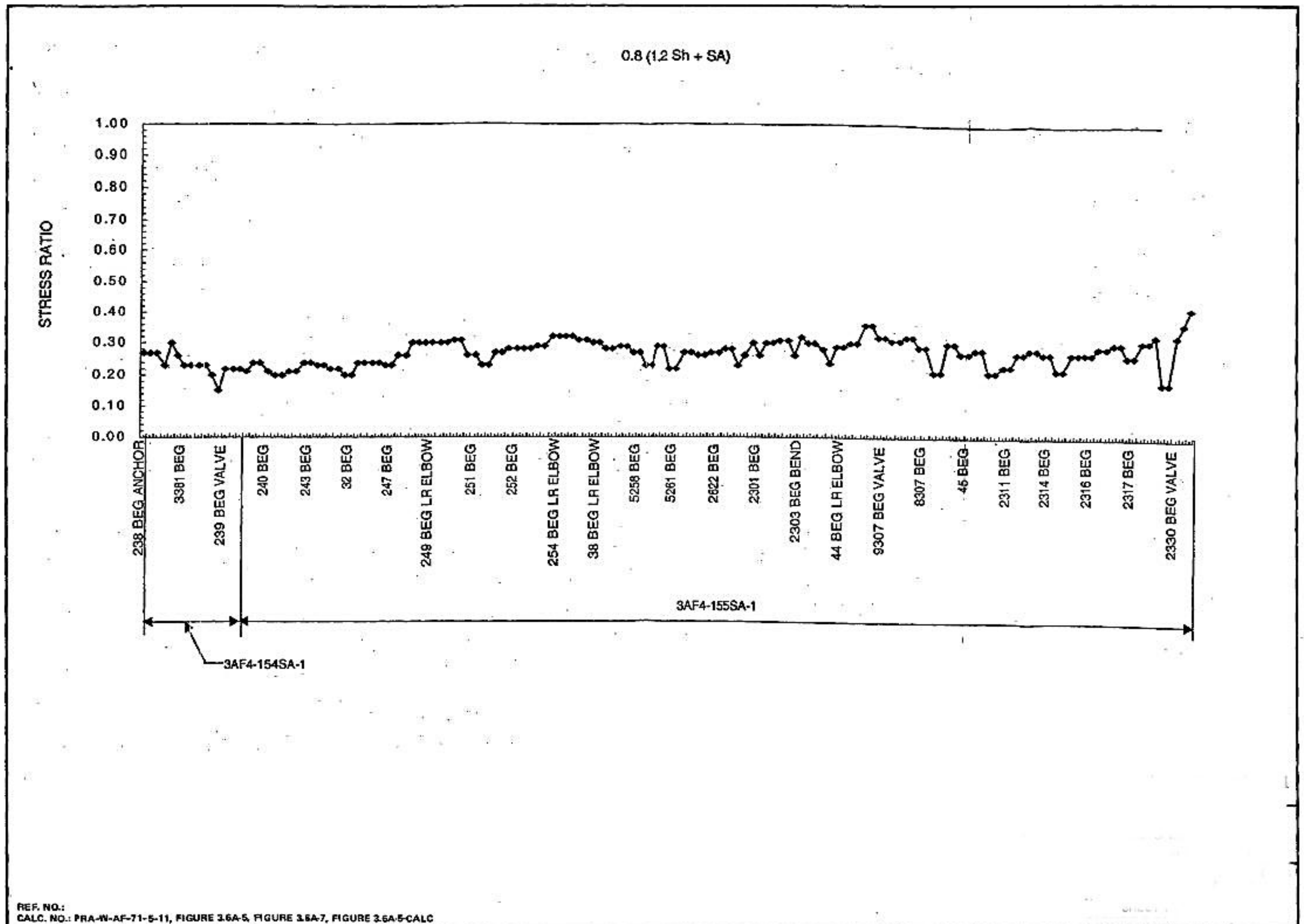


FIGURE 3.6A-5-PLOT-B (Continued)

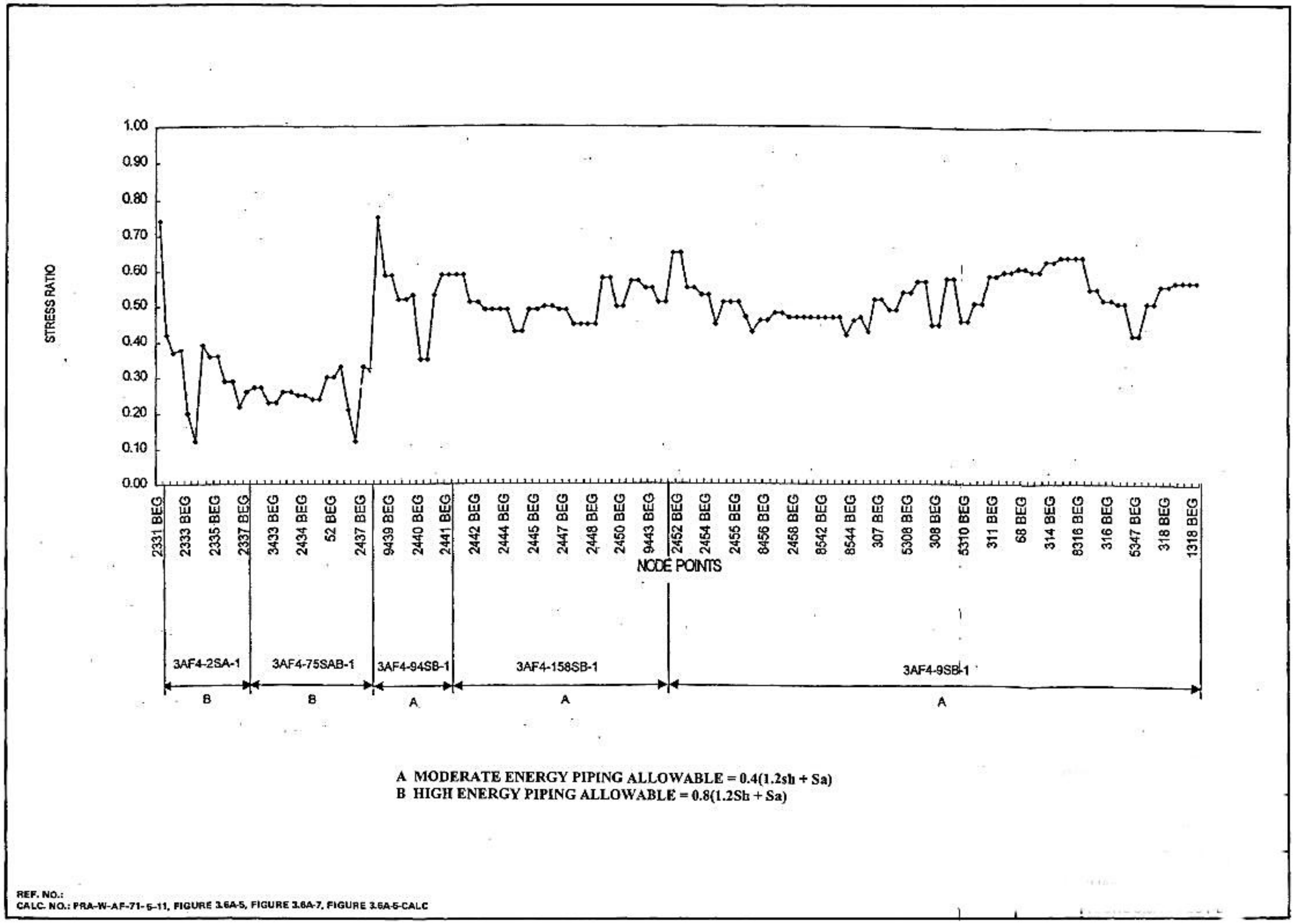


FIGURE 3.6A-5-PLOT-B (Continued)

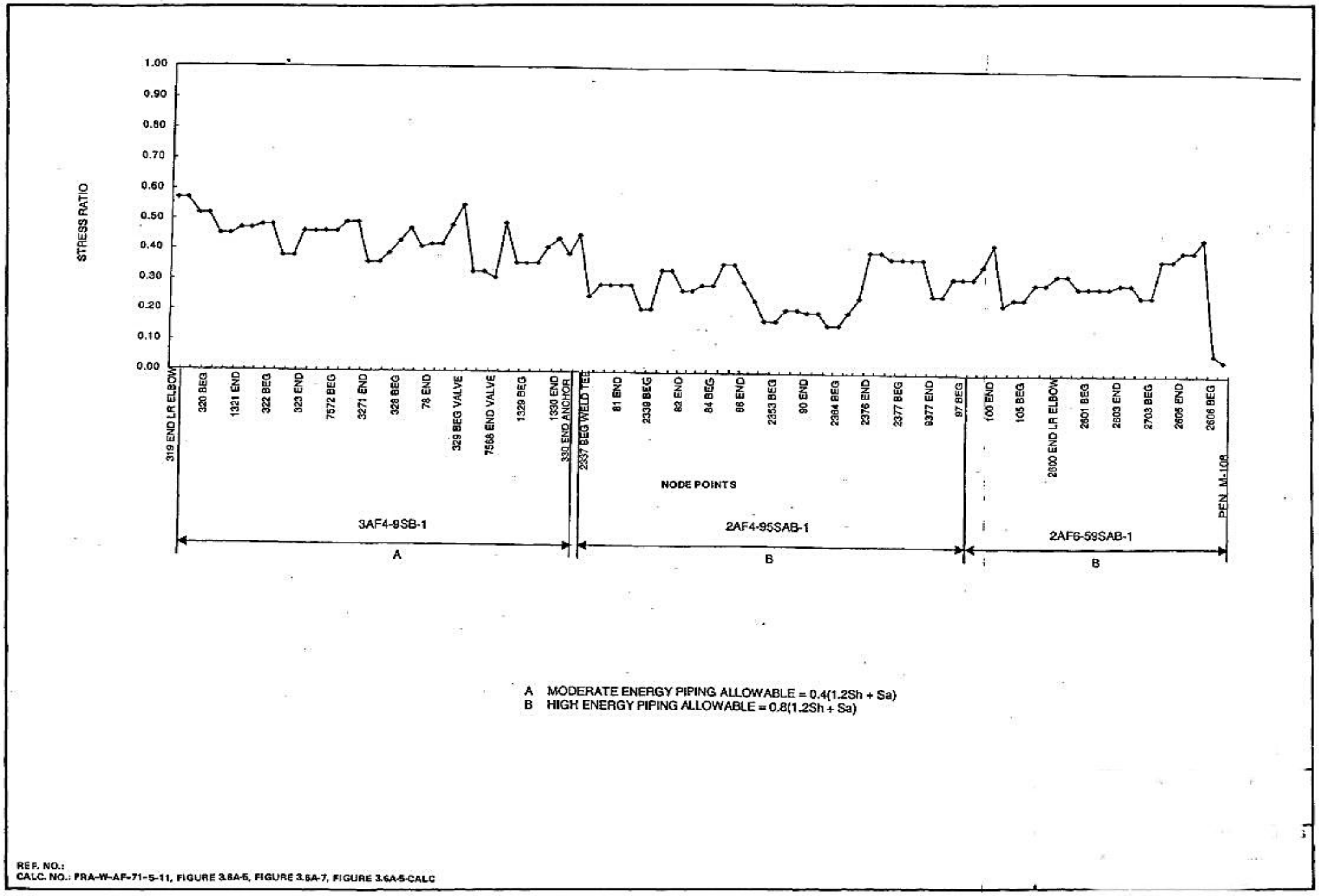


FIGURE 3.6A-6-CALC

TURBINE BUILDING, SUMMARY OF PIPE BREAK LOCATIONS, FEEDWATER PIPING

SUMMARY OF CALCULATIONS

FEEDWATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
2721	A-HFW-12-1	END OF LR ELBOW	N/A	NOTE 1	G
272Ø	A-HFW-12-2	END OF LR ELBOW	N/A	NOTE 1	G
2723	A-HFW-12-3	END OF LR ELBOW	N/A	NOTE 1	G
274	A-HFW-12-4	END OF LR ELBOW	N/A	NOTE 1	G
44Ø2	A-HFW-16-1	END OF LR ELBOW	N/A	NOTE 1	G
441	A-HFW-16-2	END OF LR ELBOW	N/A	NOTE 1	G
4444	A-HFW-16-3	END OF LR ELBOW	N/A	NOTE 1	G
445	A-HFW-16-4	END OF LR ELBOW	N/A	NOTE 1	G
369	A-HFW-14-1	END OF LR ELBOW	N/A	NOTE 1	G
3691	A-HFW-14-2	END OF LR ELBOW	N/A	NOTE 1	G
3693	A-HFW-14-3	END OF LR ELBOW	N/A	NOTE 1	G
37Ø	A-HFW-14-4	END OF LR ELBOW	N/A	NOTE 1	G

NOTE 1: BREAKS ARE POSTULATED FOR EVALUATION OF BEX REGION.

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-FW-72-1A-45, FIGURE 3.6A-6, FIGURE 3.6A-33

AMENDMENT NO. 51

SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	TURBINE BUILDING SUMMARY OF PIPE BREAK LOCATIONS FEEDWATER PIPING	FIGURE 3.6A-6-CALC

FIGURE 3.6A-6-PLOT-A

TURBINE BUILDING PLOT OF PIPE BREAK LOCATIONS
FEEDWATER PIPING

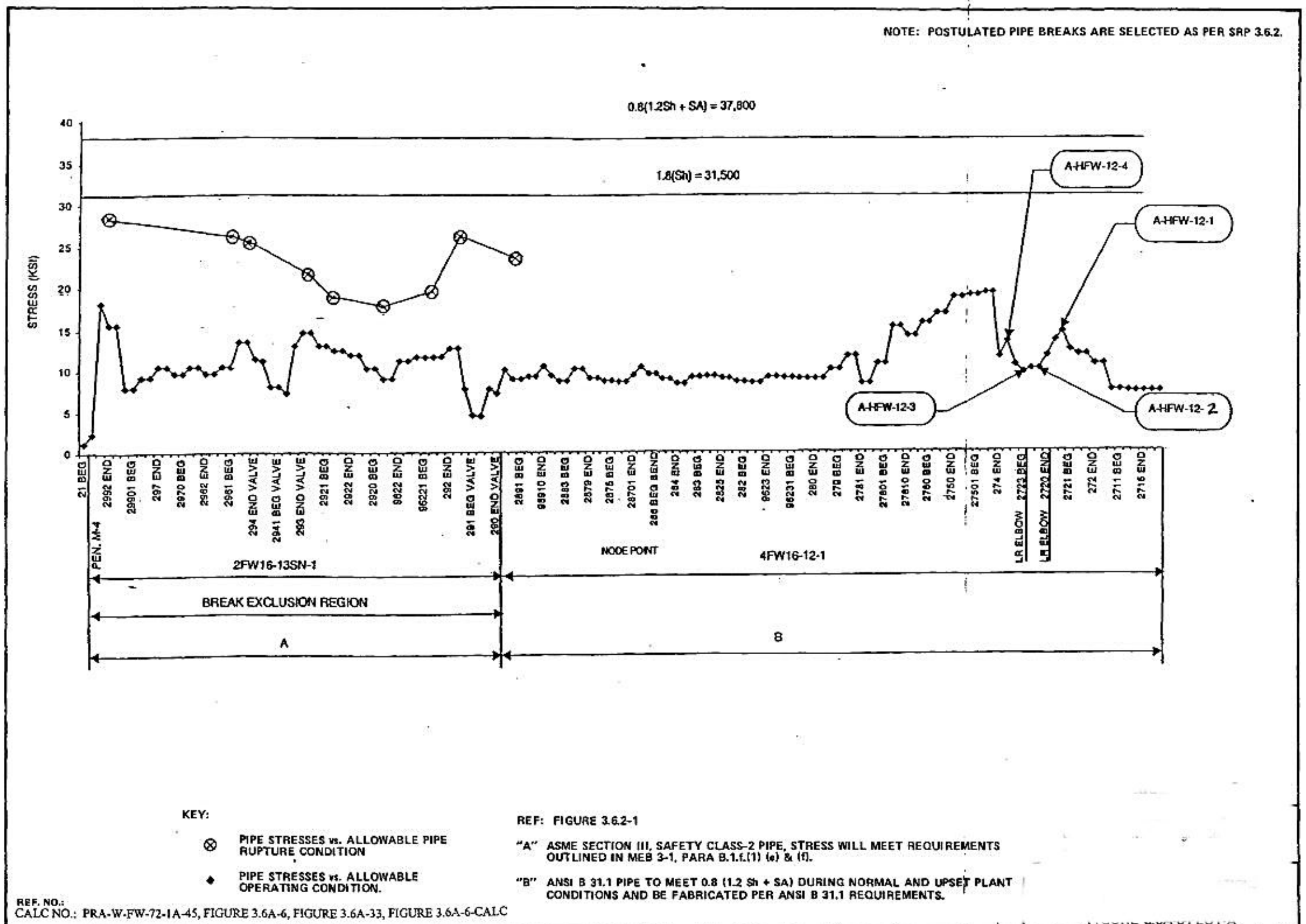


FIGURE 3.6A-6-PLOT-A (Continued)

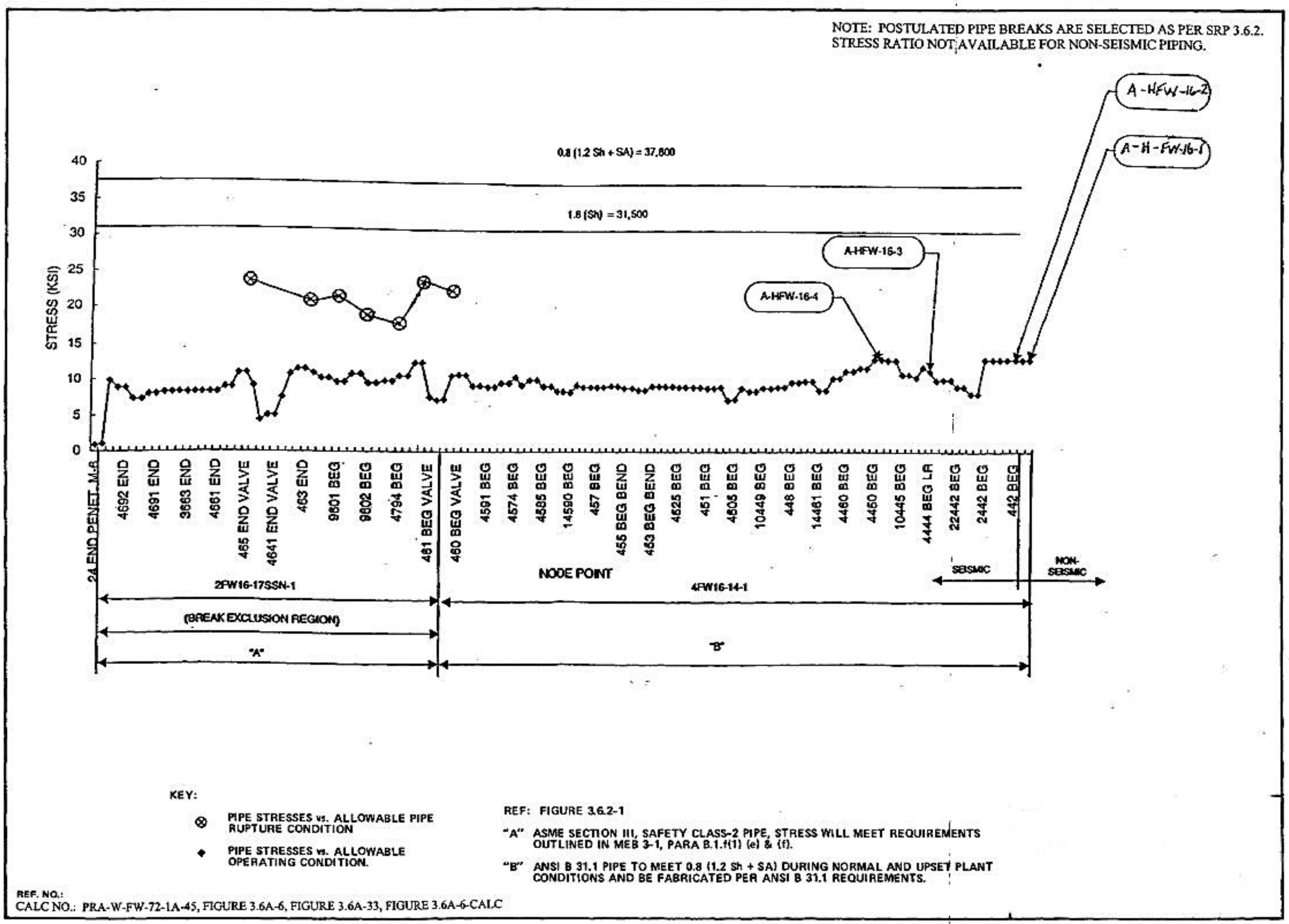


FIGURE 3.6A-6-PLOT-A (Continued)

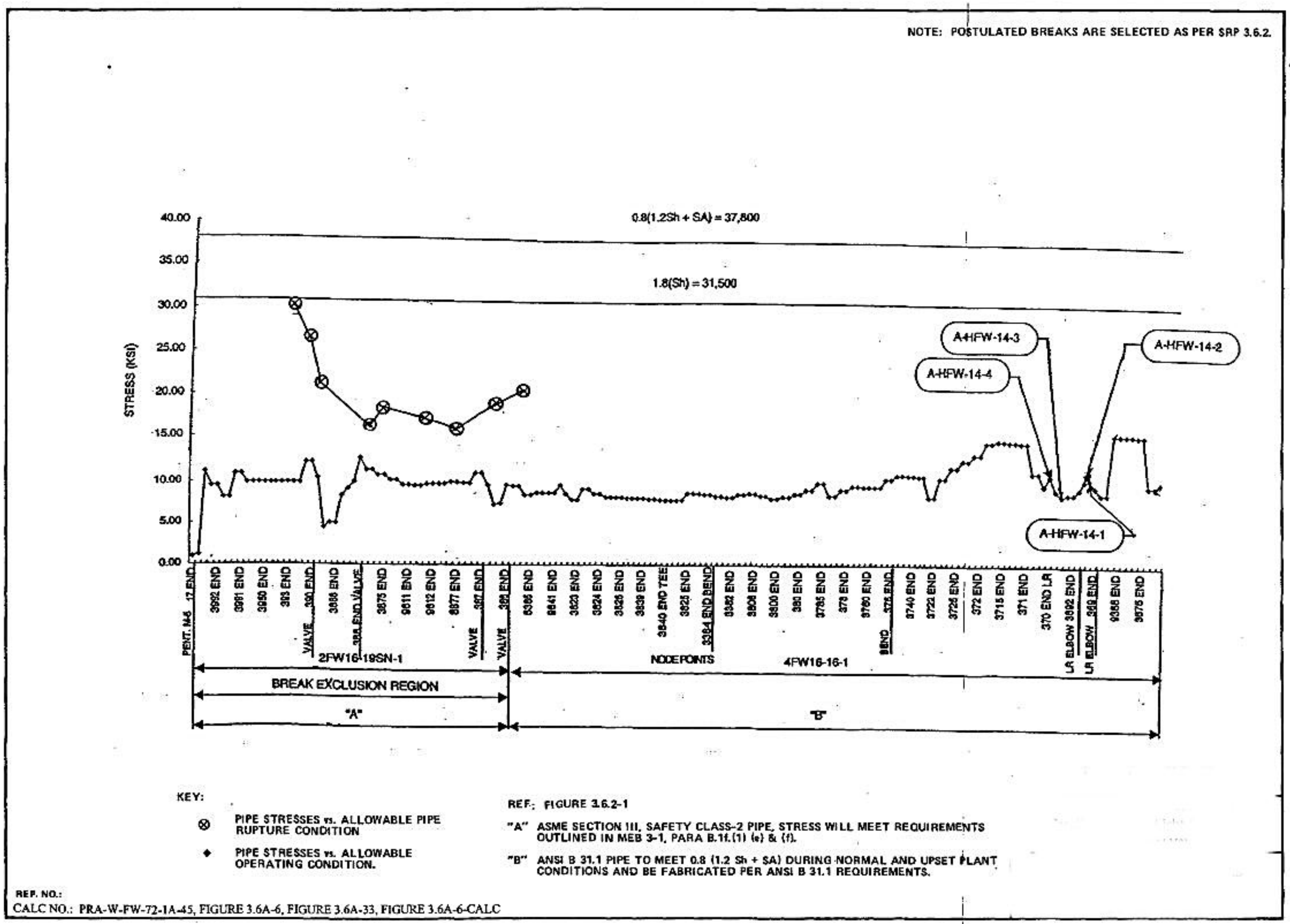


FIGURE 3.6A-7-PLOT-C

REACTOR & REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING

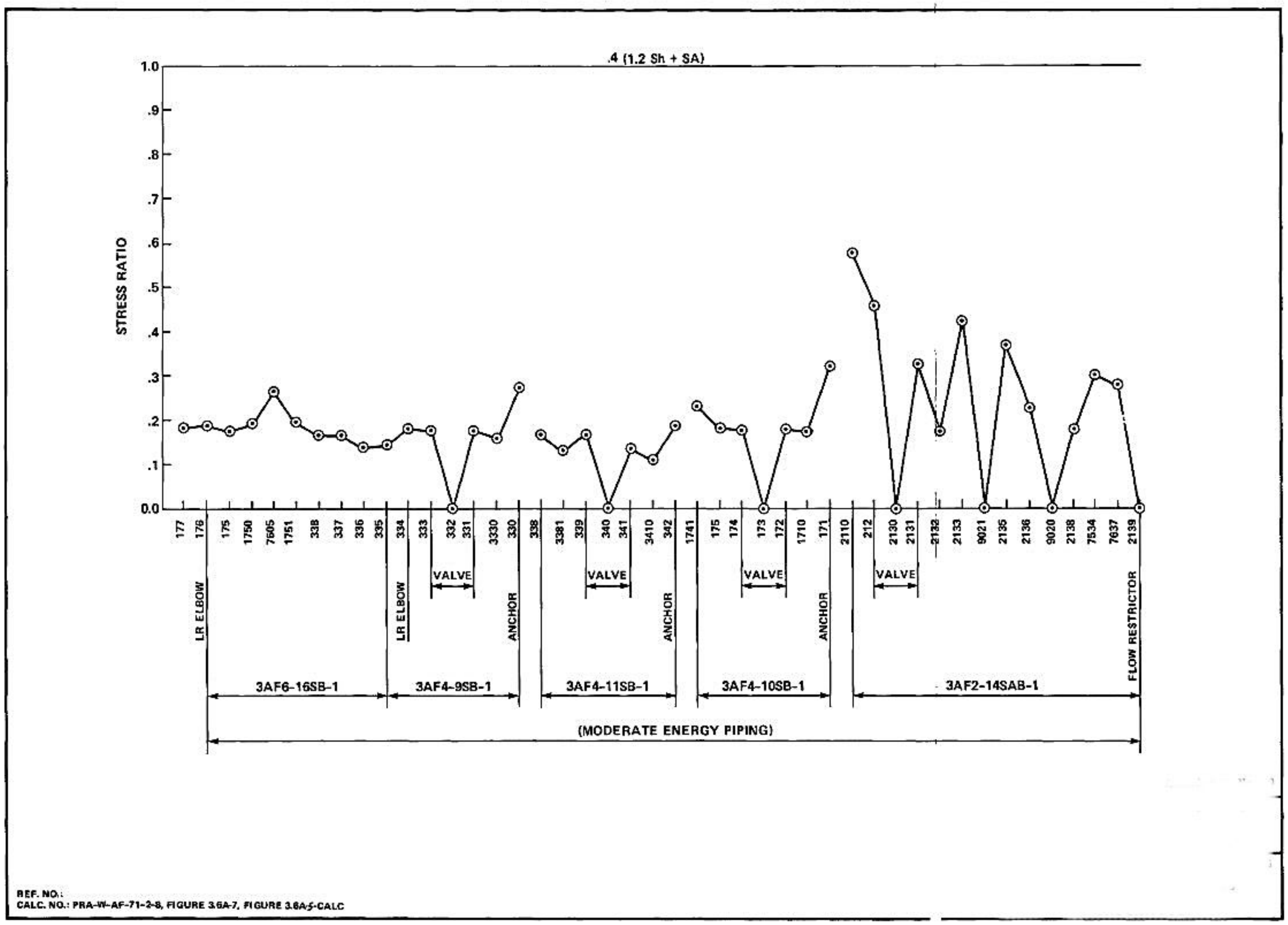
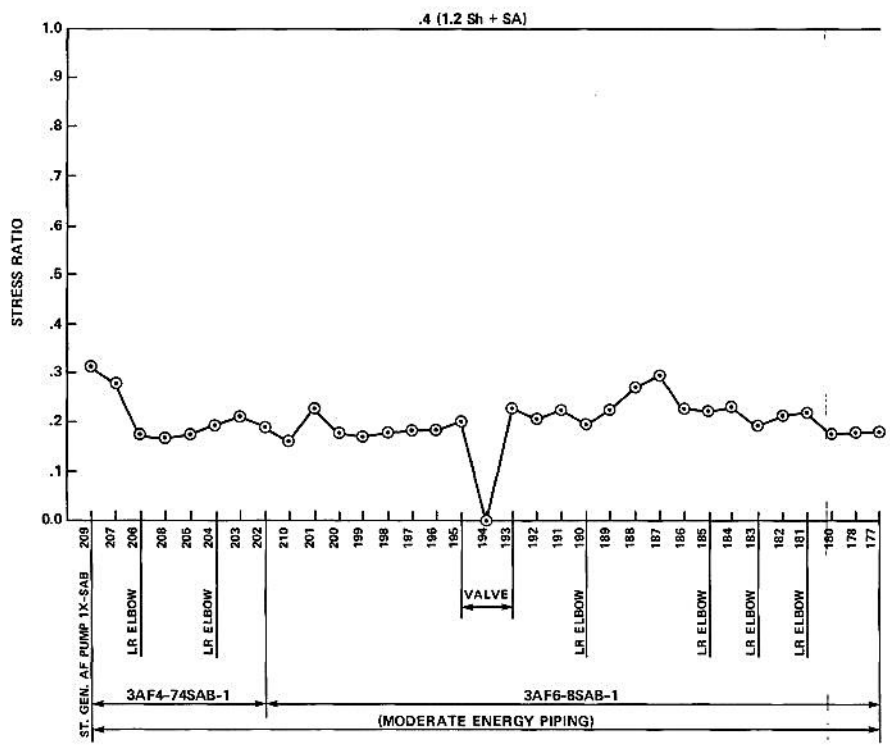


FIGURE 3.6A-7-PLOT-C (Continued)



REF. NO.:
CALC. NO.: PRA-W-AF-71-2-8, FIGURE 3.6A-7, FIGURE 3.6A-5-CALC

FIGURE 3.6A-8.1-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING LOOP #1

SUMMARY OF CALCULATIONS

AF PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
101	R-HAF-93-1	ST. GEN. NOZZ. 1A-SN	0.53	TE	G
41	R-HAF-59-6	PENET M-108	0.49	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-AF-200-1-24, FIGURE 3.6A-8

FIGURE 3.6A-8.1-CALC LOOP #2(Continued)

SUMMARY OF CALCULATIONS

AUXILIARY FEEDWATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
1001	R-HAF-92-1	ST. GEN. NOZZ. 1B-SN	0.772	TE	C
32	R-HAF-7-6	PENET. M-109	0.61	TE	C

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-AF-200-2-23, FIGURE 3.6A-8

FIGURE 3.6A-8.1-CALC LOOP #3(Continued)

SUMMARY OF CALCULATIONS

AUXILIARY FEEDWATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
101	R-HAF-91-1	ST. GEN. NOZZ. 1C-SN	0.52	TE	G
6050	R-HAF-60-6	PENET. M-110	0.44	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

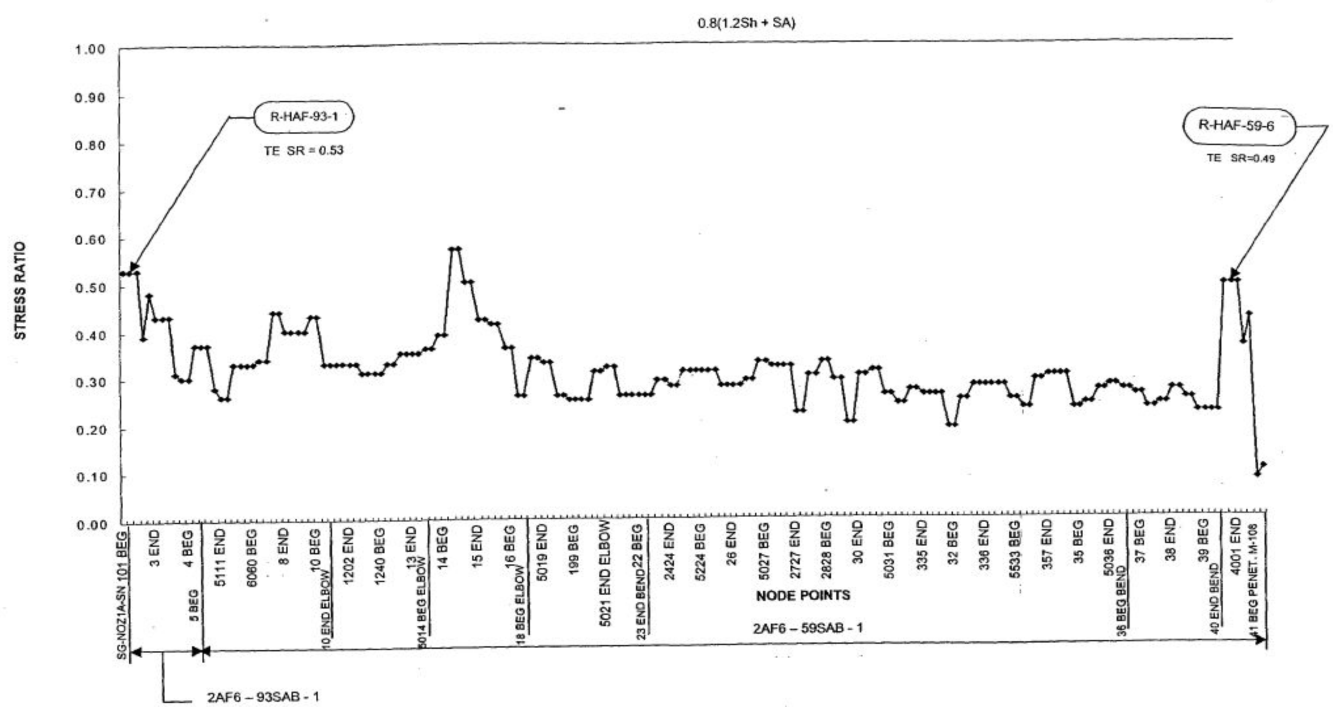
MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-AF-200-3-22, FIGURE 3.6A-8

FIGURE 3.6A-8.1-PLOT-A

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING LOOP #1



REF. NO.:
CALC. NO.: PRA-W-AF-200-1-24, FIGURE 3.6A-8, FIGURE 3.6A-8.1-CALC

FIGURE 3.6A-8.1-PLOT-B

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING LOOP #2

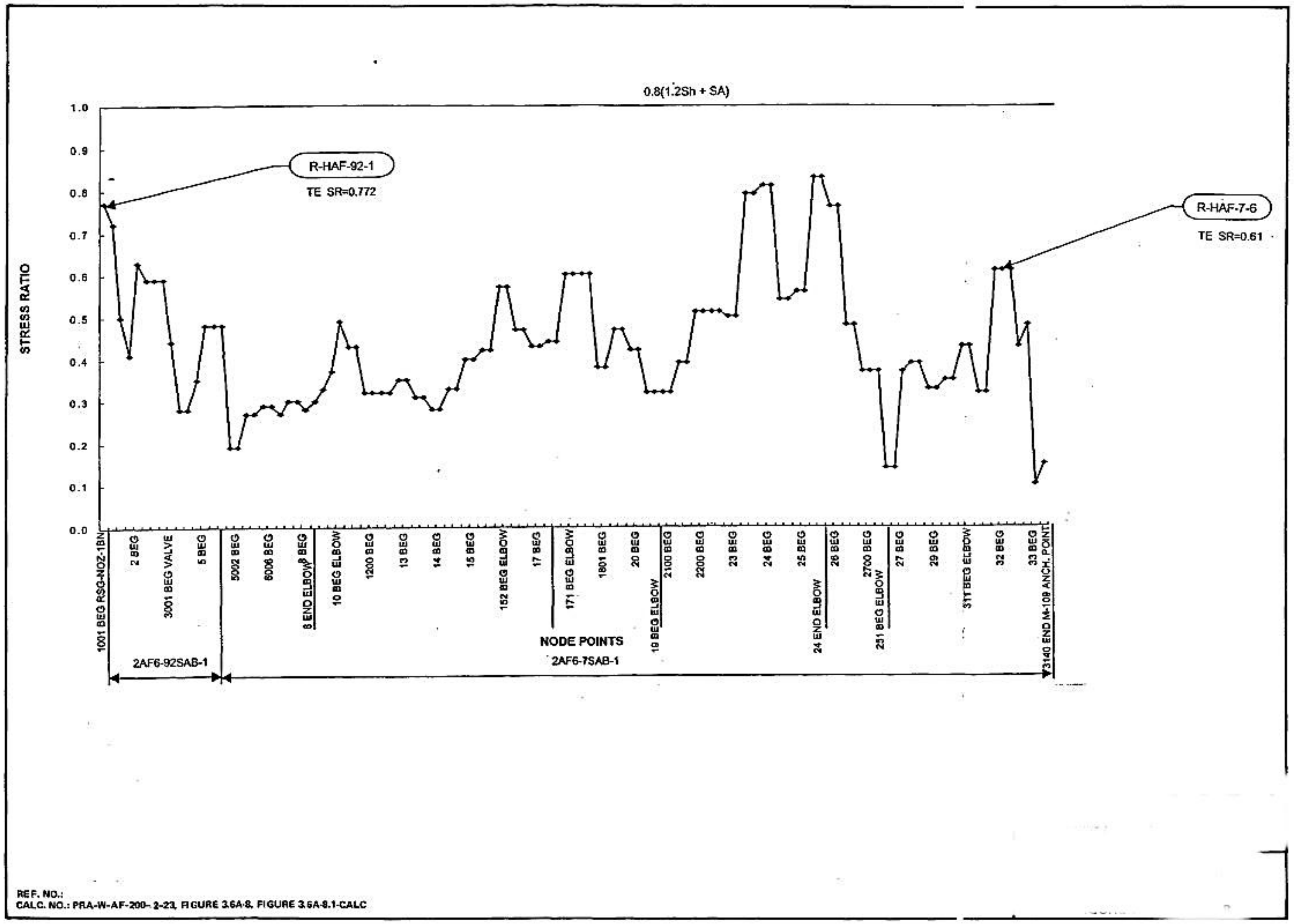


FIGURE 3.6A-8.1-PLOT-C

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING LOOP #3

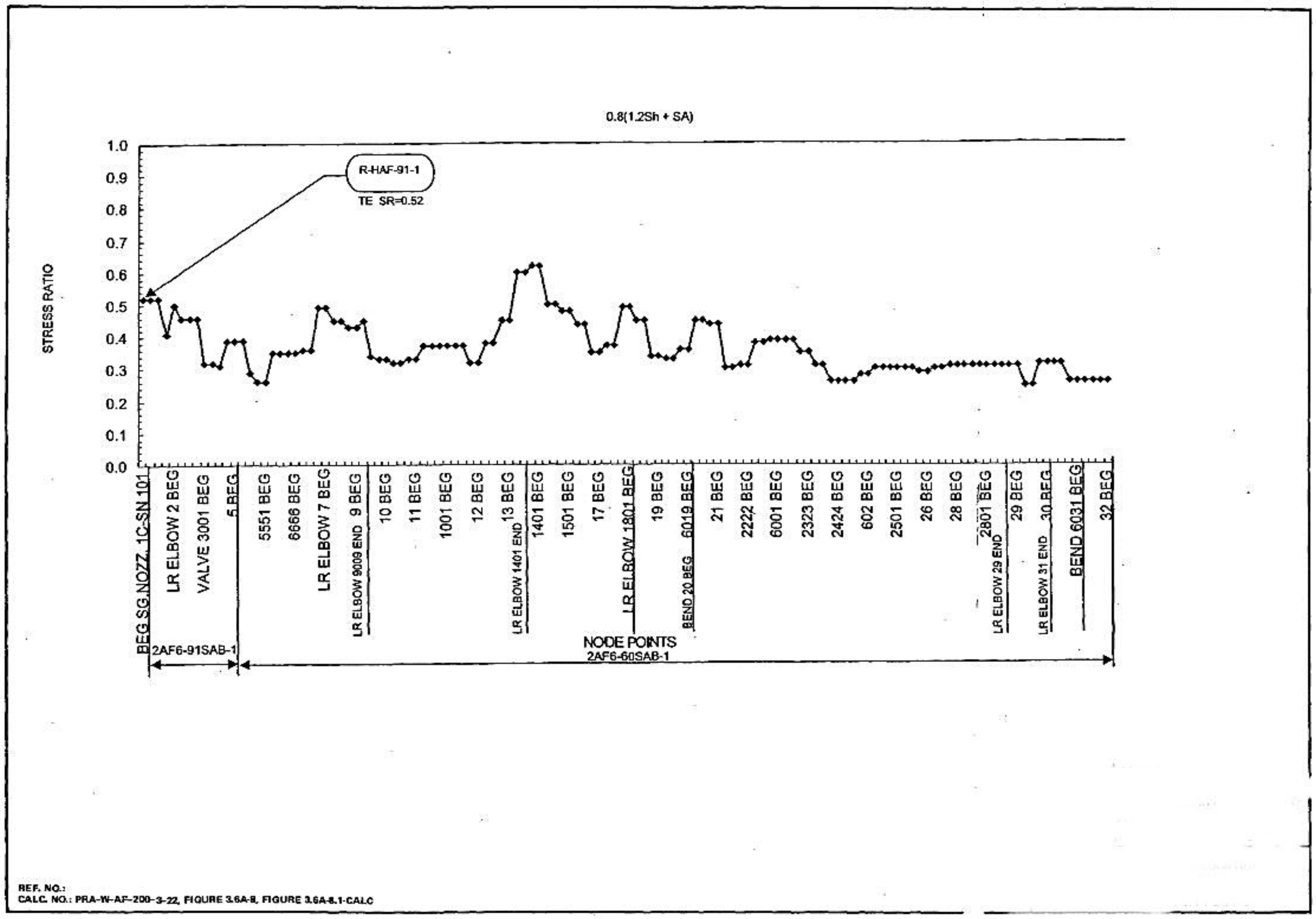
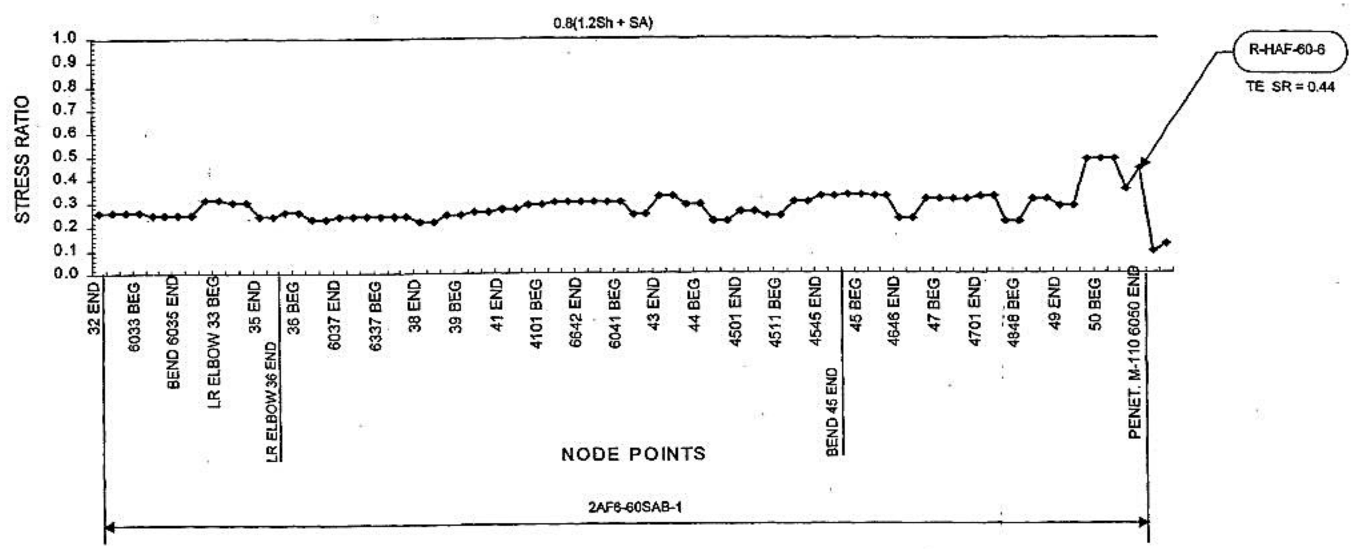


FIGURE 3.6A-8.1-PLOT-C (Continued)



REF. NO.:
CALC. NO.: PRA-W-AF-200-3-22, FIGURE 3.6A-8, FIGURE 3.6A-8.1-CALC

FIGURE 3.6A-8.2-CALC

RAB & TUNNEL AREA SUMMARY OF PIPE BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING

SUMMARY OF CALCULATIONS

AUXILIARY FEEDWATER PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
426	A-HAF-4-1	ANCHOR POINT	Ø.31	TE	G
3693	A-HAF-98-1	VALVE	Ø.32	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-AF-71-6-9, FIGURE 3.6A-8 & FIGURE 3.6A-Ø5

FIGURE 3.6A-8.2-CALC (Continued)

SUMMARY OF CALCULATIONS

AUXILIARY FEEDWATER PIPE BREAK LOCATION

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
3844	A-HAF-77-1	INLET TO CHECK VALVE	Ø.35	TE	G
8Ø	A-HAF-6-1	ANCHOR POINT	Ø.29Ø	TE	G
4Ø76	A-HAF-6Ø-6	PENET. M-11Ø	Ø.Ø7	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

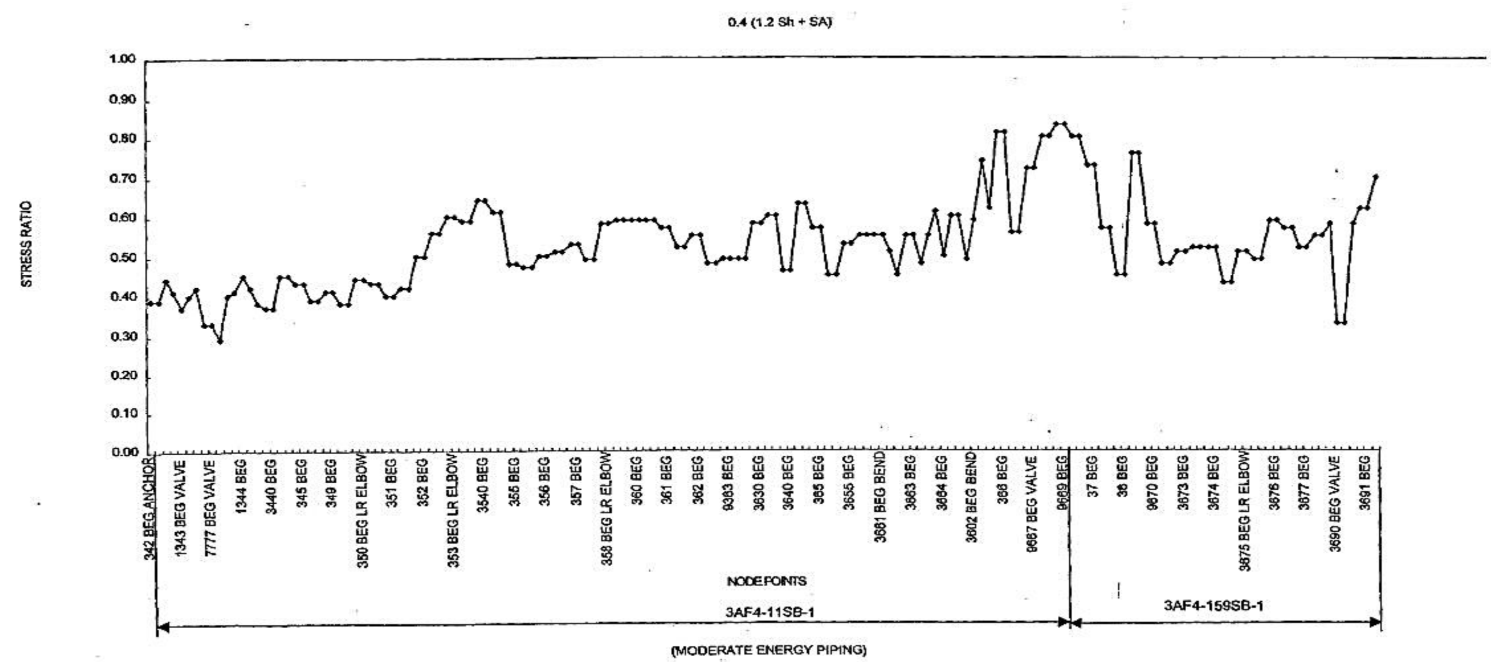
MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-AF-71-7-1Ø, FIGURE 3.6A-8 & FIGURE 3.6A-Ø5

FIGURE 3.6A-8.2-PLOT-A

TUNNEL AREA PLOT OF BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING



REF. NO.:
CALC. NO.: PRA-W-AF-71- 6-9, FIGURE 3.6A-8, FIGURE 3.6A-8.2-CALC

Figure 3.6A-8.2-PLOT-A (Continued)

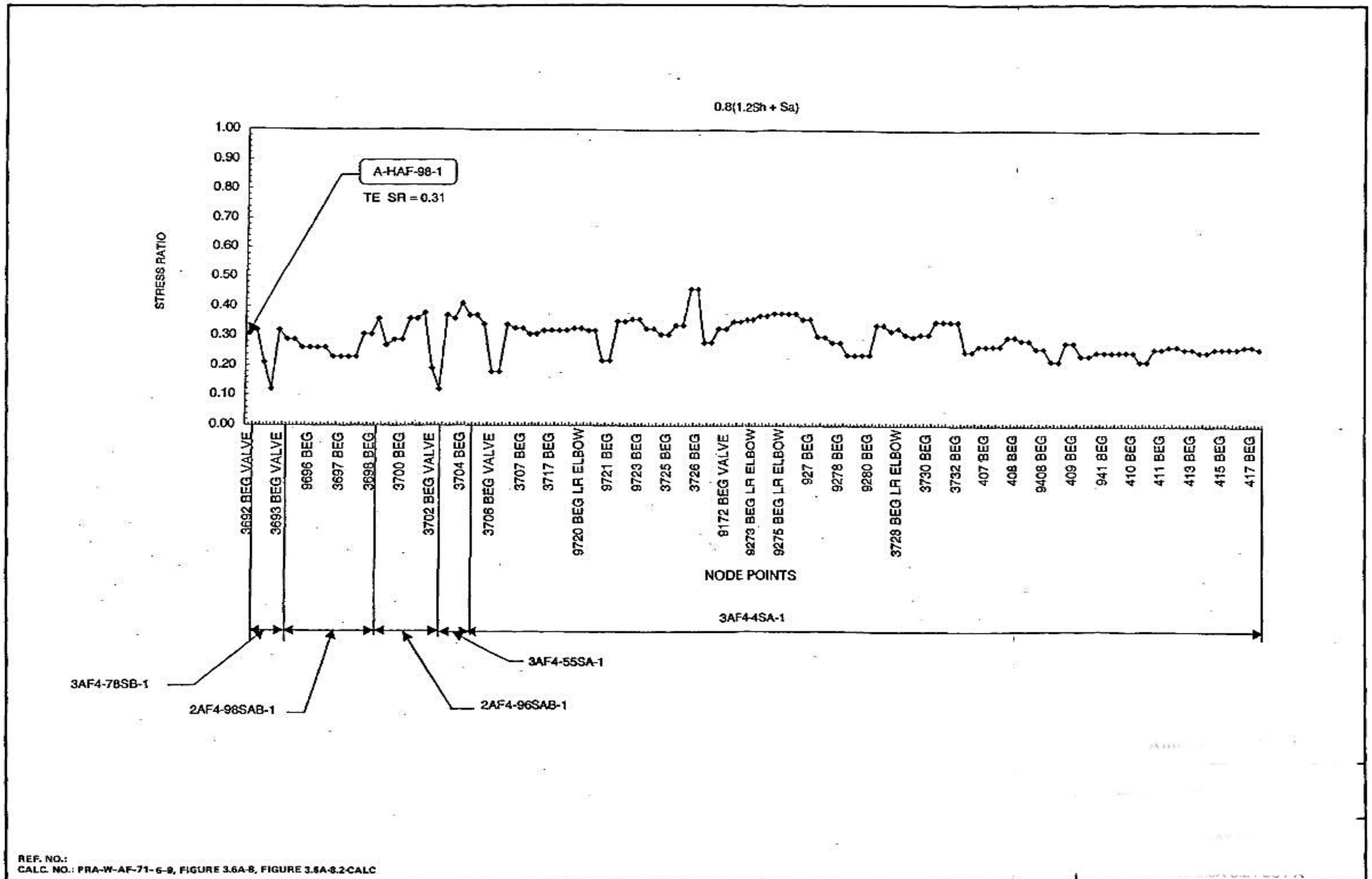


Figure 3.6A-8.2-PLOT-A (Continued)

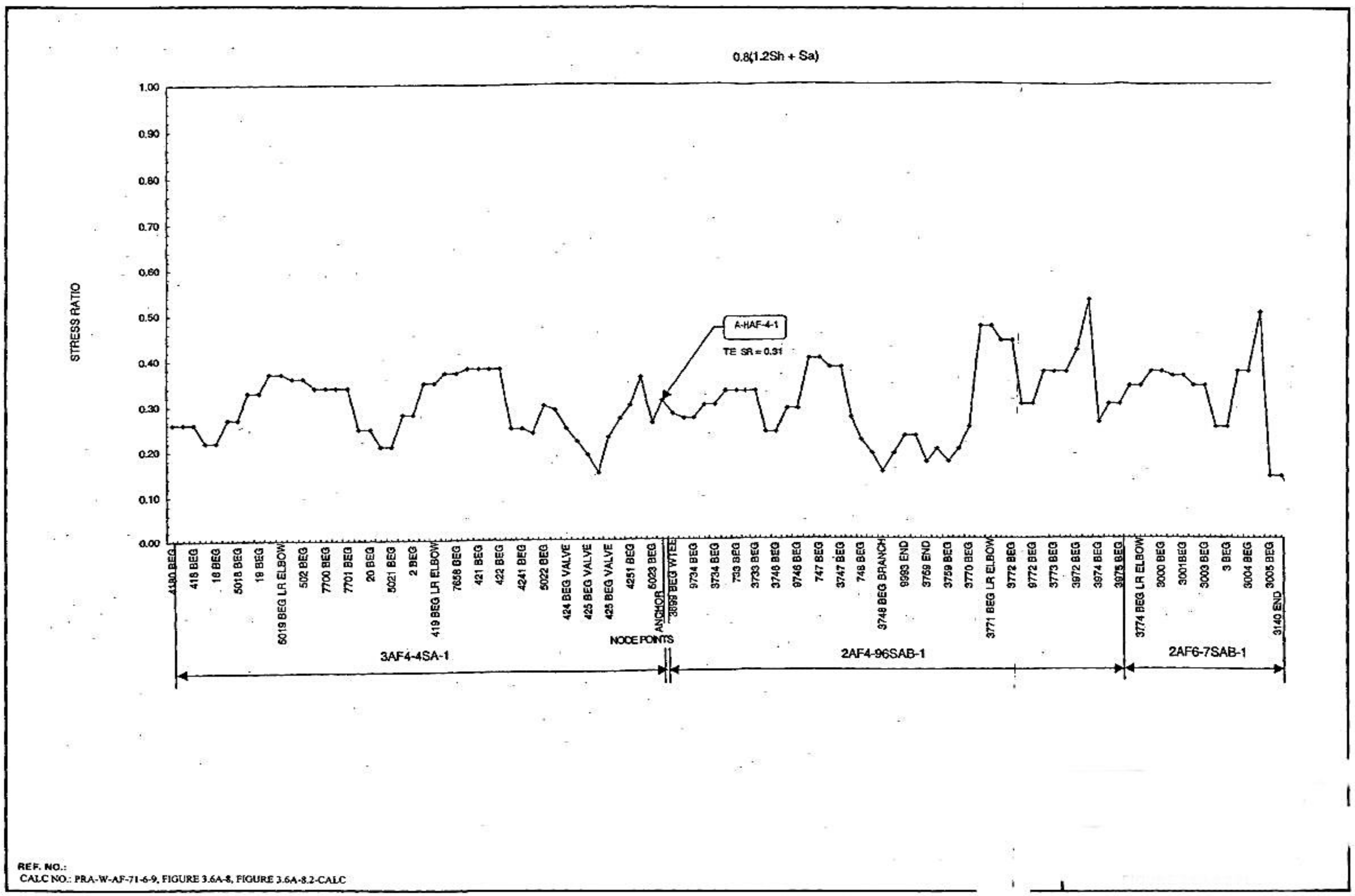
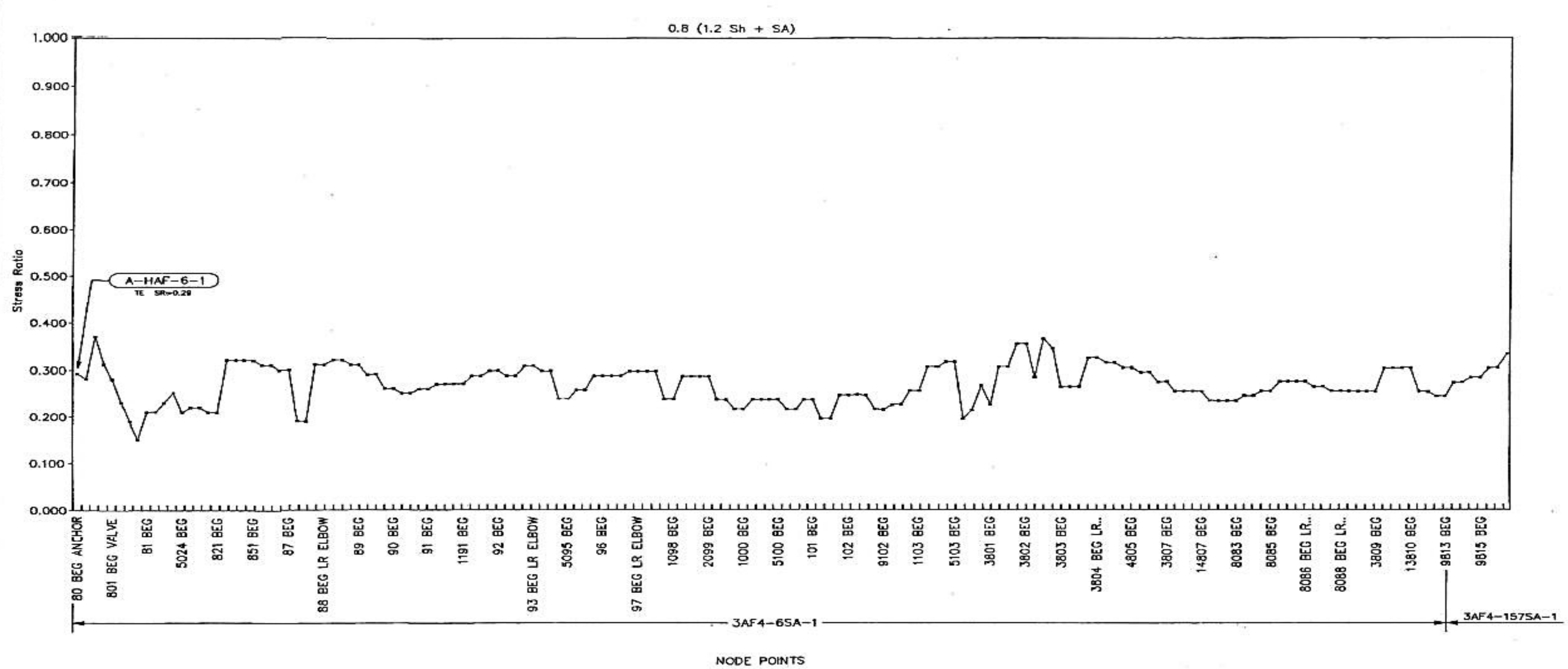


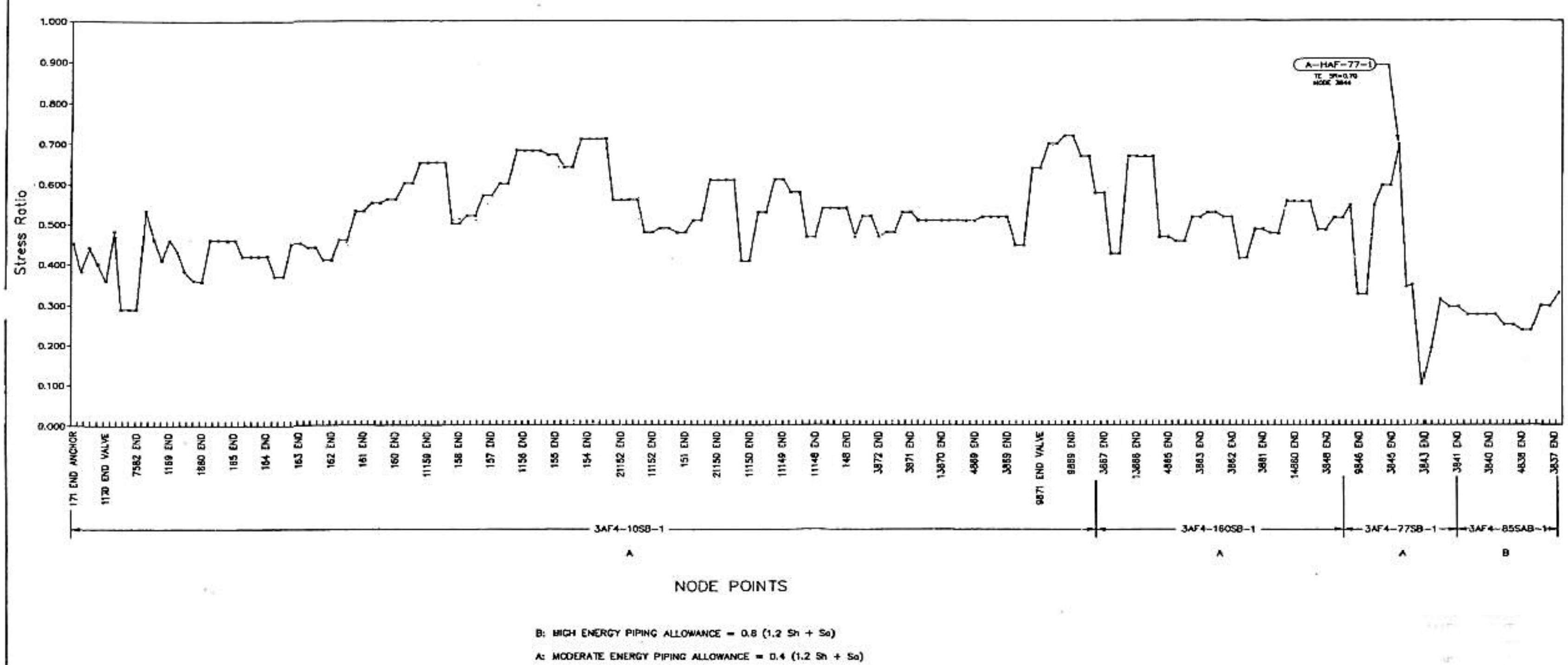
FIGURE 3.6A-8.2-PLOT-B

TUNNEL AREA PLOT OF BREAK LOCATIONS
AUXILIARY FEEDWATER PIPING



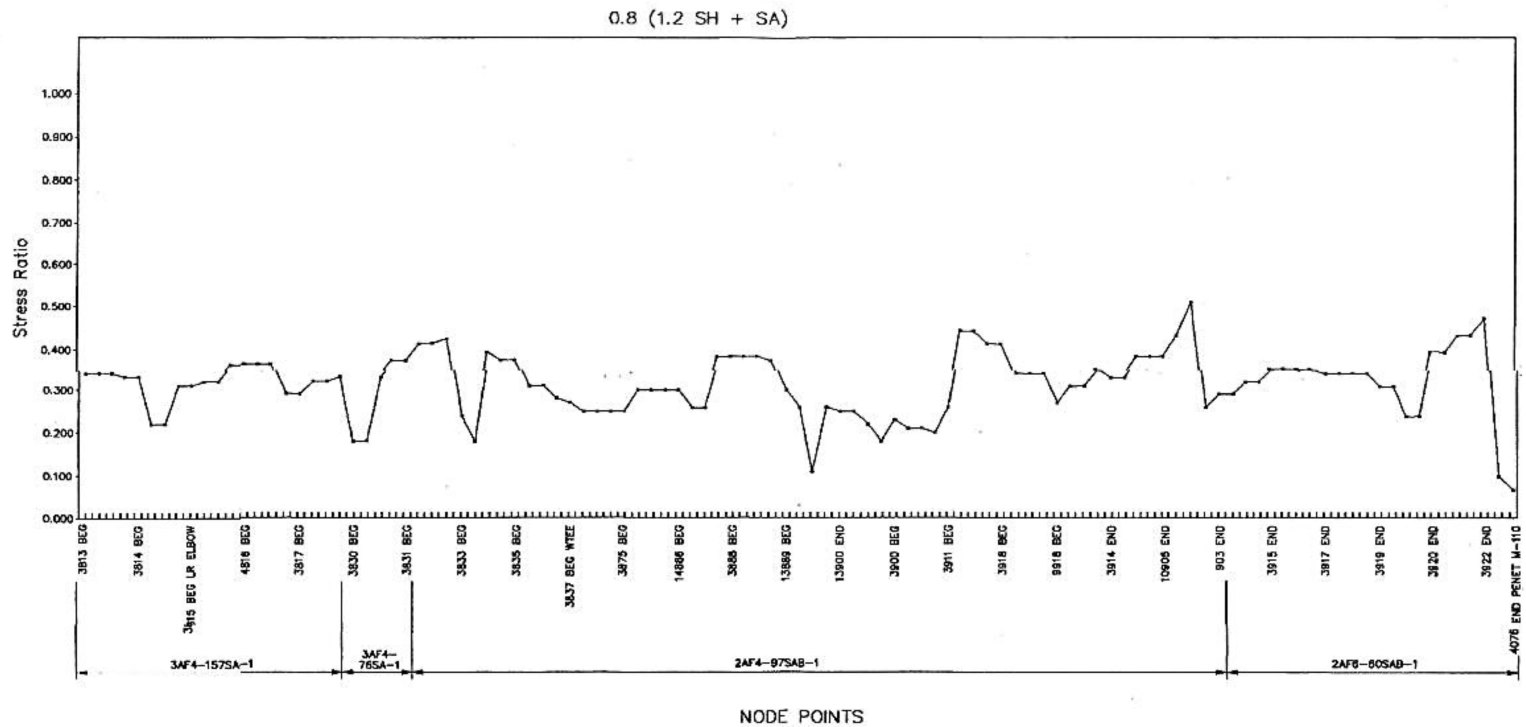
REF. NO.
CALC. NO.: PRA-W-AF-71-7-10, FIGURE 3.6A-8, FIGURE 3.6A-8.2-CALC

FIGURE 3.6A-8.2-PLOT-B (Continued)



REF. NO.
CALC. NO.: PRA-W-AF-71-7-10, FIGURE 3.6A-8, FIGURE 3.6A-8.2-CALC

FIGURE 3.6A-8.2-PLOT-B (Continued)



REF. NO.
CALC. NO.: PRA-W-AF-71-7-10, FIGURE 3.6A-8, FIGURE 3.6A-8.2-CALC

FIGURE 3.6A-9-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
CVCS & RC PIPING

SUMMARY OF CALCULATIONS

CVCS NORMAL CHARGING PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
BOP W					
32	R-HCS-118-1	DOWN STREAM END OF VALVE	0.302	HRI*	G
33	R-HCS-118-2	UP STREAM END OF VALVE	0.324	HRI*	G
3200	R-HCS-118-3	3X1 BRANCH CONN	0.736	HRU	G
35	R-HRC-45-4	DOWN STREAM END OF VALVE	0.330	HRI*	G
39	R-HRC-45-1	BRANCH CONN	0.262	TE	G
1	R-HCS-83-4	ANCHOR POINT (CLASS-2)			

REF. W CLASS 1 ANALYSIS LTR CQLB418 DTD 11-19-84

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

HRU = HIGH RELATIVE USAGE FACTOR

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:
CALC NO PRA-W-CS-3016-1C-35, FIGURE 3.6A-9, FIGURE 3.6A-10

FIGURE 3.6A-9-CAL (Continued)

SUMMARY OF CALCULATIONS

CS PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO	CRI-TERIA[a]	BREAK TYPE[b]
16	R-HCS-140-1	3 X 2 RED ELL	0.551	TE	G
24	R-HCS-90-1	3 X 3 X 2 RED TEE	0.455	TE	G
40	R-HCS-91-1	3 X 3 X 2 RED TEE	0.984	TE	G
47	R-HCS-139-1	3 X 3 X 2 RED TEE	0.403	TE	G
53	R-HCS-88-1	REG. HEAT EXCH. NOZZLE	0.412	TE	G
9	R-HCS-92-1	3 X 2 RED ELL	0.627	TE	G
31	R-HCS-138-1	3 X 2 RED ELL	0.385	TE	G
150	R-HCS-89-1	3 X 3 X 2 RED TEE	1.142	TE	G
1	R-HCS-96-1	PENETRATION M-7	0.044	TE	G
1491	R-HCS-89-2	UPSTREAM END OF VALVE	1.175	HRI	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-3003-1C-41, FIGURE 3.6A-9, FIGURE 3.5A-10, FIGURE 3.6A-11

FIGURE 3.6A-9-CAL (Continued)

SUMMARY OF CALCULATIONS

PRESSURIZER SAFETY & RELIEF PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
1	R-HCS-83-1	BRANCH CONN.	0.295	TE	G
36	R-HCS-83-4	ANCHOR	0.714	TE	G
5002	R-HCS-84-1	WELDED TEE	1.277	HRI	G
5003	R-HRC-84-2	END OF VALVE	0.989	TE	G
139	R-HCS-85-4	WELDED TEE	0.381	TE	G
110	R-HCS-85-1	UPSTREAM END OF VALVE	0.384	TE	G
1080	R-HCS-117-3	3X1 BRANCH	0.736 *	HRU	G
108	R-HCS-117-1	DOWNSTREAM END OF VALVE	0.593 *	HRU	G
107	R-HCS-117-2	UPSTREAM END OF VALVE	0.593 *	HRU	G
105	R-HRC-28-4	DOWNSTREAM END OF VALVE	0.444 *	HRU	G
102	R-HRC-28-1	BRANCH CONN.	-	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

HRU: HIGH RELATIVE USAGE FACTOR

* USAGE FACTOR

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:
CALC NO PRA-W-CS=3014-1C-37 FIGURE 3.6A-9, FIGURE 3.6A-10

FIGURE 3.6A-9-PLOT-A

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
CVCS & RC PIPING

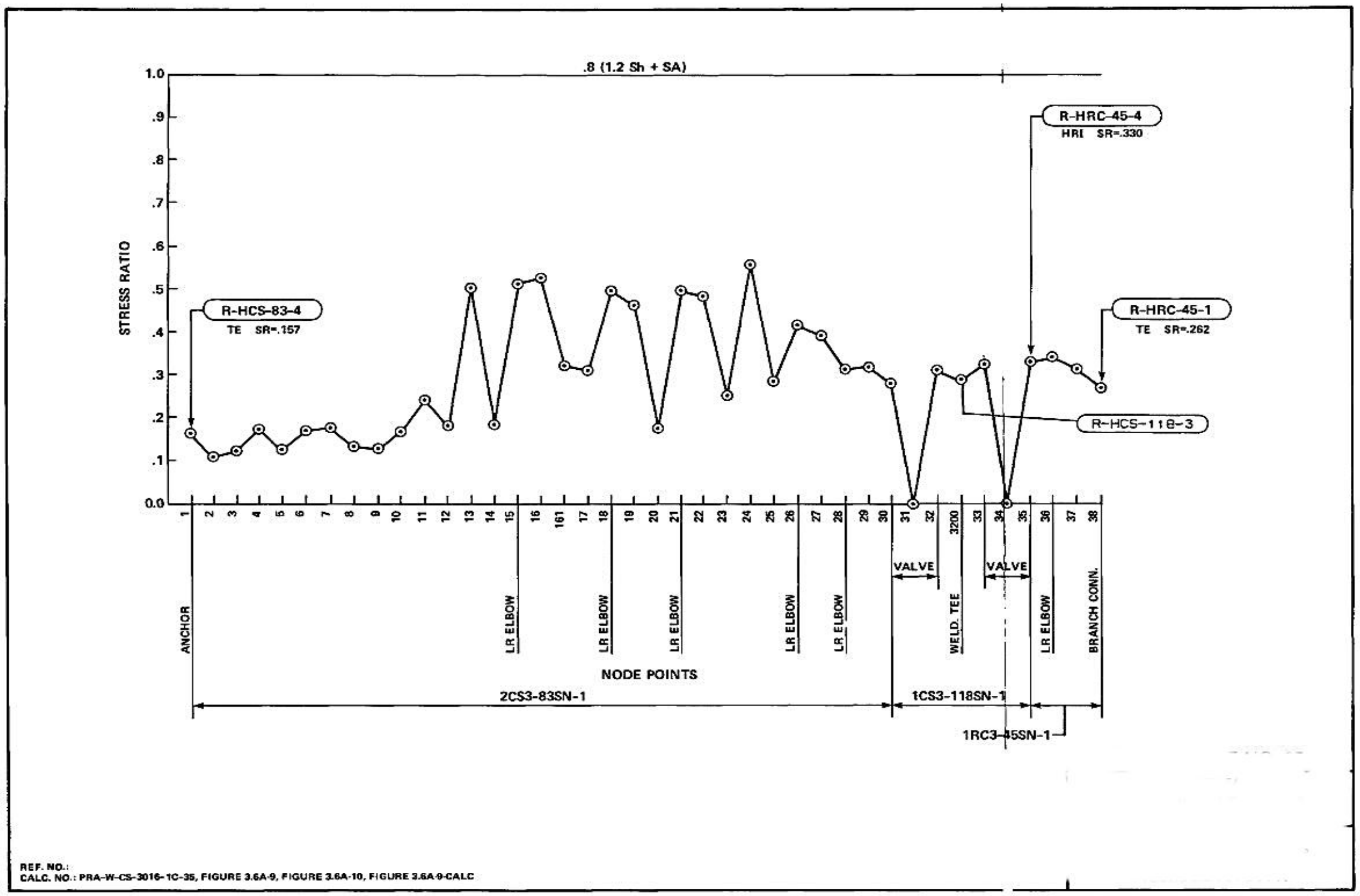
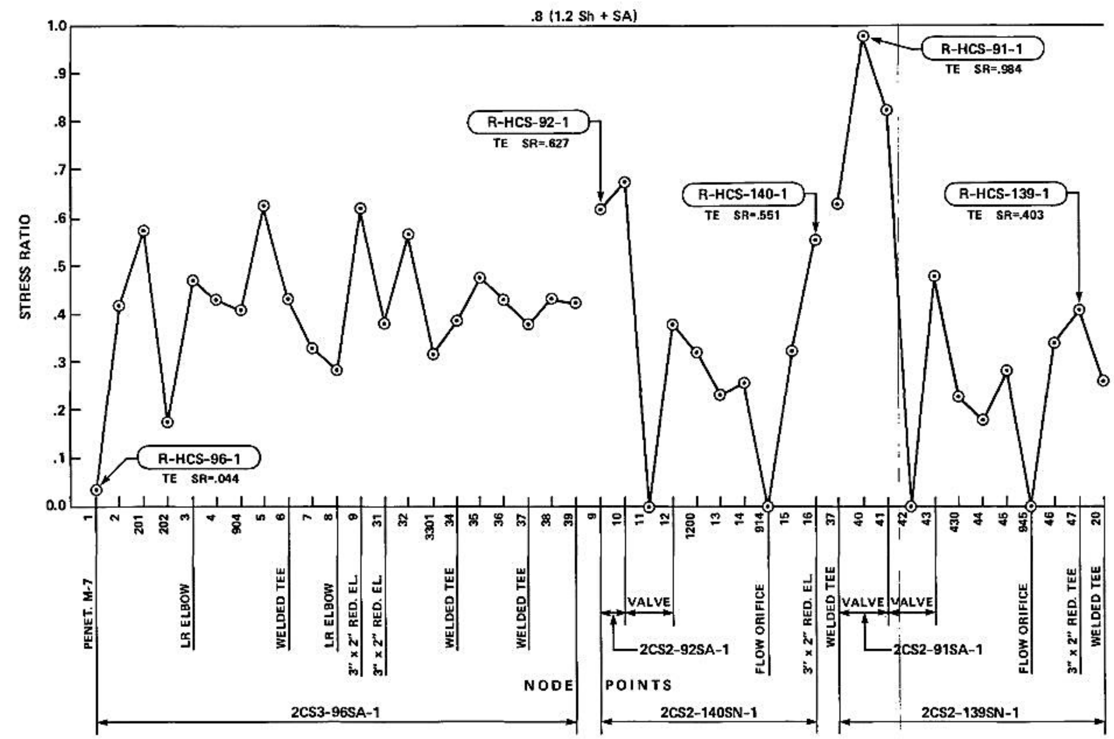


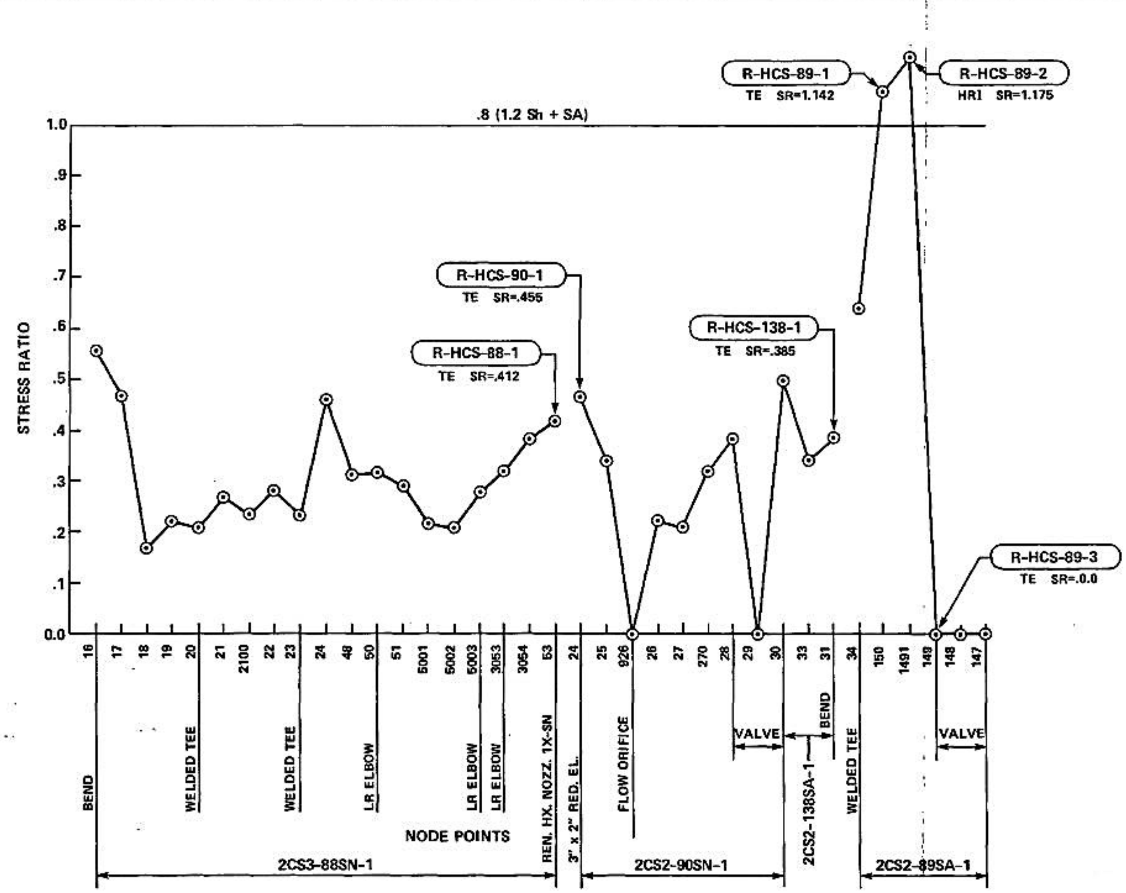
FIGURE 3.6A-9-PLOT-B

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING



REF. NO.:
CALC. NO.: PRA-W-CS-3003-1C-41, FIGURE 3.6A-9, FIGURE 3.6A-10, FIGURE 3.6A-11, FIGURE 3.6A-9-CALC

FIGURE 3.6A-9-PLOT-B (Continued)



REF. NO.:
CALC NO.: PRA-W-CS-3003-1C-41, FIGURE 3.6A-9, FIGURE 3.6A-10, FIGURE 3.6A-11, FIGURE 3.6A-9-CALC

FIGURE 3.6A-9-PLOT-C

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
CVCS & RC PIPING

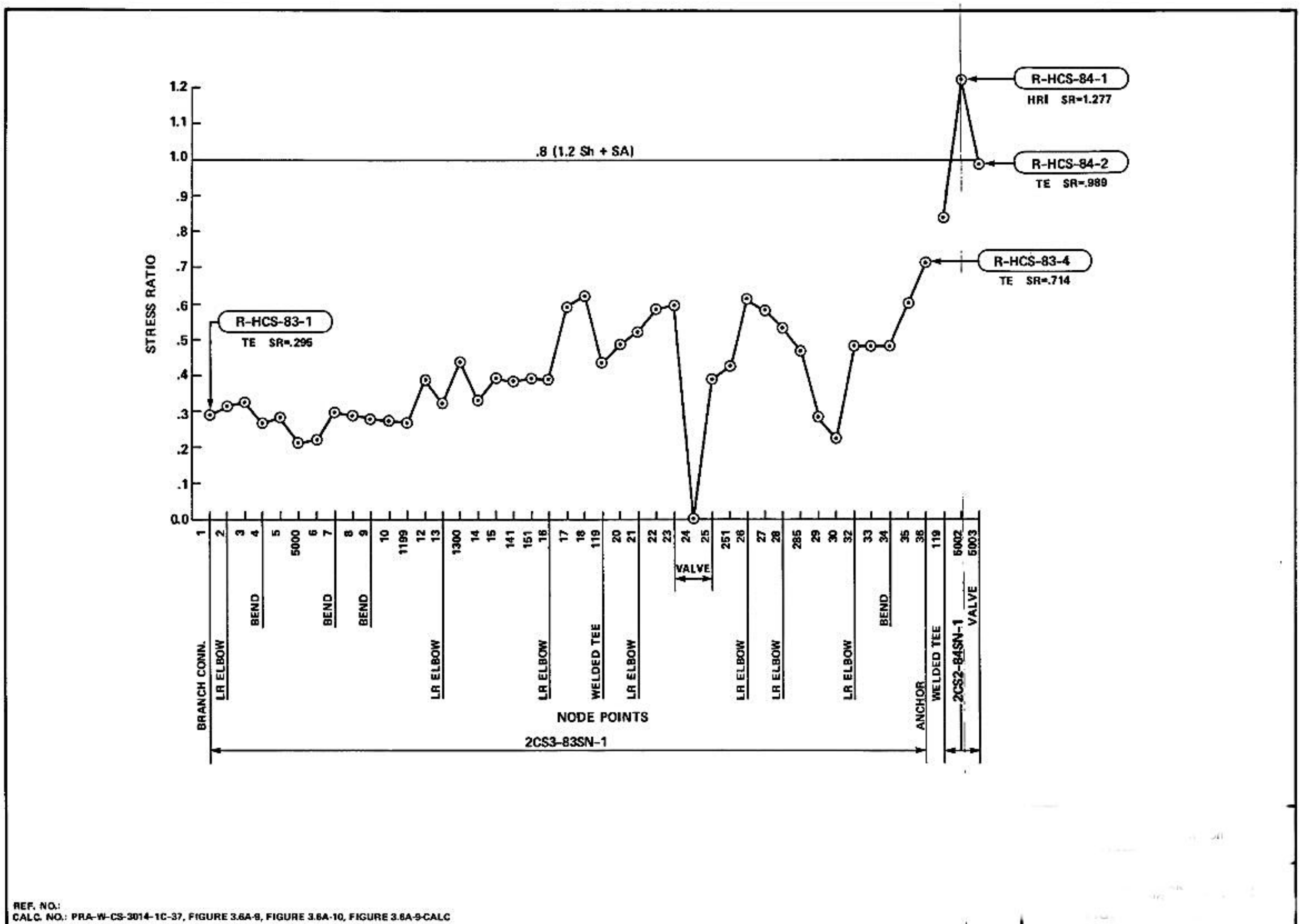
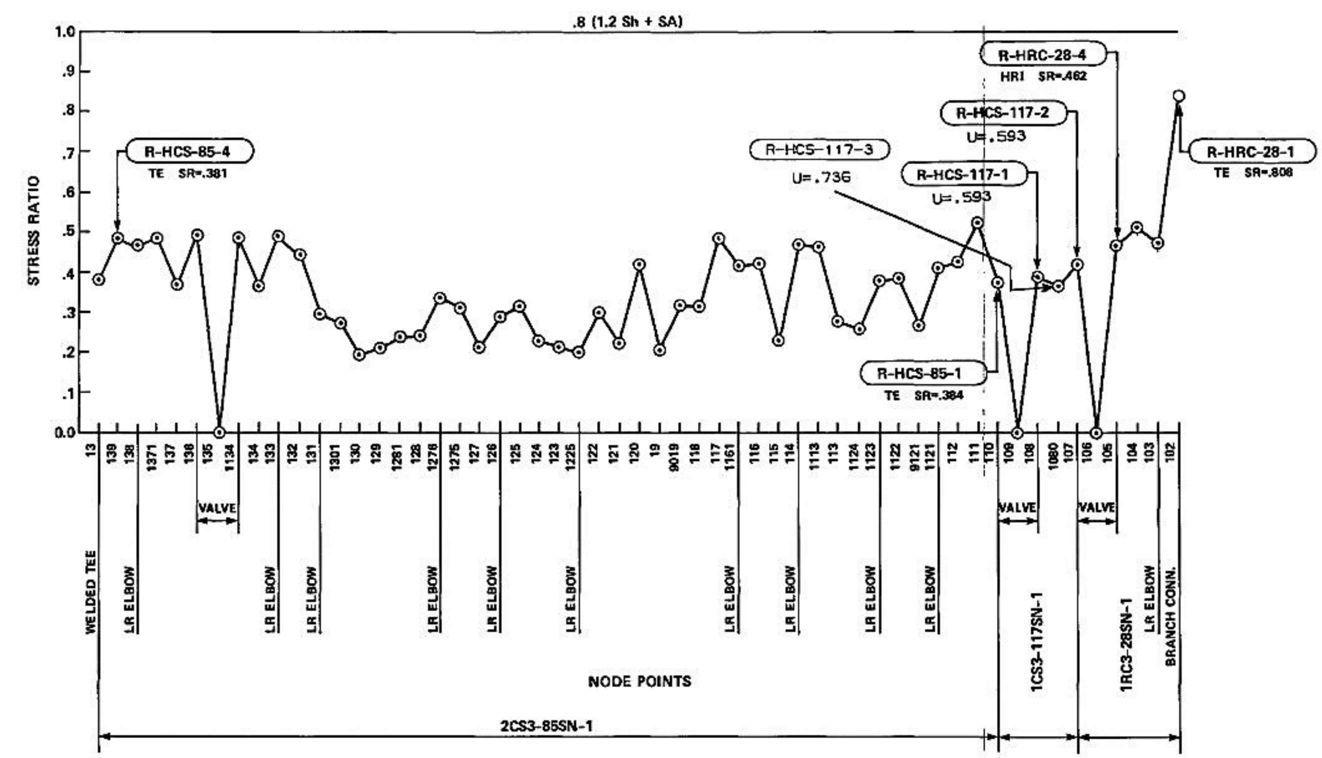


FIGURE 3.6A-9-PLOT-C (Continued)



REF. NO.:
CALC. NO.: PRA-W-CS-3014-1C-37, FIGURE 3.6A-9, FIGURE 3.6A-10, FIGURE 3.6A-9-CALC

FIGURE 3.6A-10-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
CVCS & RC PIPING

SUMMARY OF CALCULATIONS

CVCS & RC PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
2	R-HRC-23-1	BRANCH CONNECTION	0.800	TE*	G
19	R-HRC-23-6	UP STREAM OF VALVE END	0.422	HRI*	G
7	R-HRC-71-1	3"X2" REDUCER	0.941 †	HRU	G
11	R-HRC-71-4	UP STREAM OF VALVE END	0.205	TE	G
51	R-HRC-87-1	CONN. TO REFEN. HT. EXCH.	0.367	TE	G

*CLASS 1 PIPING BREAKS ARE POSTULATED AS PER WESTINGHOUSE CRITERIA REF. 8.

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

HRU - HIGH RELATIVE USAGE FACTOR

† - USAGE FACTOR

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-3011-1C-34, FIGURE 3.6A-10

FIGURE 3.6A-10-CALC (Continued)

SUMMARY OF CALCULATIONS

CVCS PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
84	R-HCS-95-1	PENET. M-8	0.027	TE	G
18	R-HCS-114-1	HX NOZZLE	0.352	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC NO.: PRA-W-CS-3006-IC-33, FIGURE 3.6A-19, FIGURE 3.6A-10

FIGURE 3.6A-10-CALC (Continued)

SUMMARY OF CALCULATIONS

SEAL INJECTION LINES PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI- TERIA (A)	BREAK TYPE (B)
BOP	W*					
48	2020	R-HCS-16-1	RC PUMP NOZZLE	-	TE	G
1	-	R-HCS-19-1	PEN. # M-9	-	TE	G
43	1030	R-HCS-17-1	RC PUMP NOZZLE	-	TE	G
1001	-	R-HCS-20-1	PEN # M-10	-	TE	G
1	1030	R-HCS-18-1	RC PUMP NOZZLE	-	TE	G
70	-	R-HCS-21-1	PEN # M-11	-	TE	G

*W LTR CQL-8418 DTD 11-19-84

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:

CALC NO.: PRA-W-CS-0057

FIGURE 3.6A.10 & 3.6A.11

FIGURE 3.6A-10-PLOT-A

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
CVCS & RC PIPING

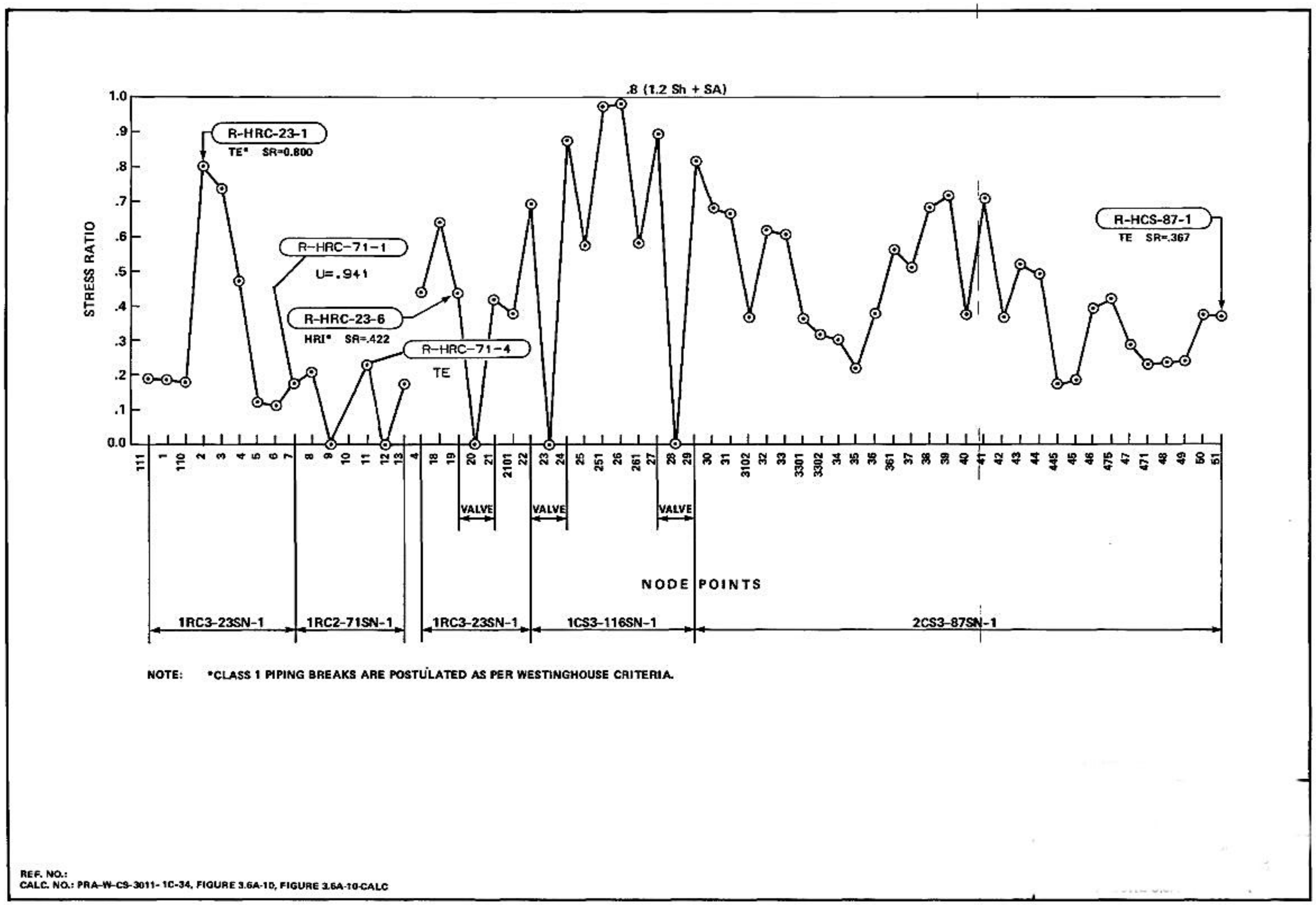


FIGURE 3.6A-10-PLOT-B

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING

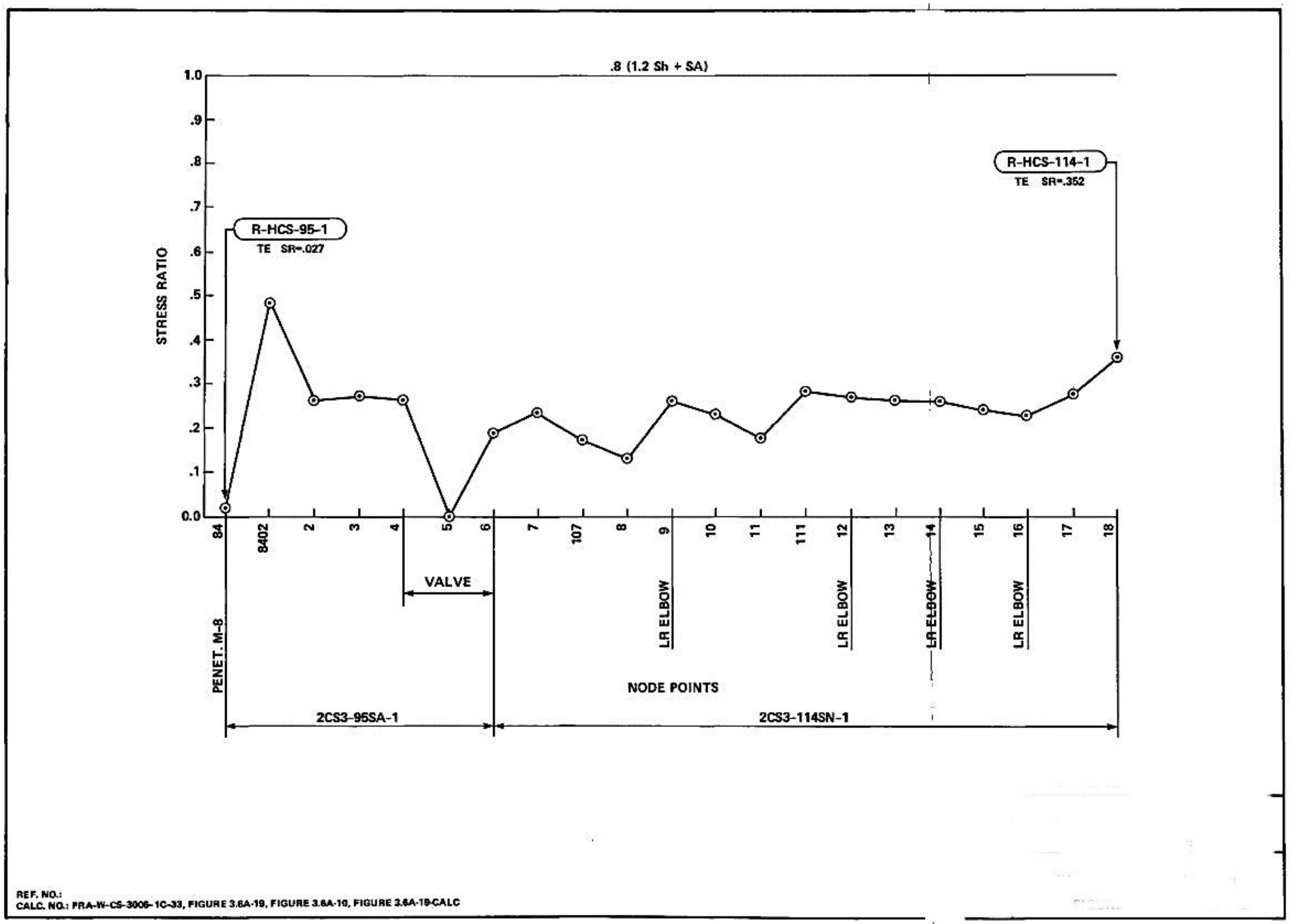


FIGURE 3.6A-12-CALC

REACTOR AUXILIARY BUILDING SUMMARY OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING

SUMMARY OF CALCULATIONS

CS PIPING PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRITERIA(a)	BREAK TYPE(b)
127	A-HCS-300-3	ANCHOR POINT	0.246	TE	G
186	A-HCS-294-1	#1 PUMP NOZZ. IA-SA	0.000	TE	G
2972	A-HCS-296-5	MIN. FLOW ORIFICE	0.000	TE	G
202	A-HCS-292-1	#3 PUMP NOZZ. IC-SAB	0.000	TE	G
2372	A-HCS-297-5	MIN. FLOW ORIFICE	0.000	TE	G
3396	A-HCS-305-3	ANCHOR POINT @ E-WALL	0.194	TE	G
348	A-HSI-84-4	ANCHOR POINT	0.232	TE	G
170	A-HCS-337-1	#2 PUMP NOZZ. IB-SB	0.000	TE	G
3242	A-HCS-298-5	MIN. FLOW ORIFICE	0.000	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-141-1&2-27, FIGURE 3.6A-12

FIGURE 3.6A-12-CALC (Continued)

SUMMARY OF CALCULATIONS

CS & SI PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA[a]	BREAK TYPE[b]
84	A-HCS-95-1	PENET. M-8	0.039	TE	G
127	A-HCS-300-3	ANCHOR POINT	0.437	TE	G
202	A-HSI-50-1	ANCHOR POINT	0.246	TE	G
198	A-HSI-51-2	DOWN STREAM END OF VALVE	0.384	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-141-3-28, FIGURE 3.6A-12, FIGURE 3.6A-18

FIGURE 3.6A-12-CALC (Continued)

SUMMARY OF CALCULATIONS

CS PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA[a]	BREAK TYPE[b]
1	A-HCS-19-1	PENETRATION M-9	.008	TE	G*
460	A-HCS-67-3	ANCHOR POINT	.122	TE	G*

*NOTES

(1) THERE IS NO ESSENTIAL EQUIPMENT LOCATED IN THE VICINITY OF THE SUBJECT PIPING.

(2) THE STEADY-STATE JET FORCE IS NEGLIGIBLE.

(3) DUE TO NOTES 1 & 2 REASONS NO BREAKS & ASSOCIATED JETS ARE NOT POSTULATED OR SHOWN ON RUPTURE DRAWINGS

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-3098-1A-30, FIGURE 3.6A-12, FIGURE 3.6A-25

FIGURE 3.6A-12-CALC (Continued)

SUMMARY OF CALCULATIONS

CVCS RC PUMP SEAL INJ. PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
2085	A-HCS-311-1	ANCHOR POINT	.104	TE	G
1001	A-HCS-20-1	PENETRATION M-10	.010	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIA

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-3099-1A-31, FIGURE 3.6A-12, FIGURE 3.6A-26

FIGURE 3.6A-12-CALC (Continued)

SUMMARY OF CALCULATIONS

CVCS-SEAL INJECTION PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA[a]	BREAK TYPE[b]
2085	A-HCS-311-1	RIGID ANCHOR ON 2CS2-311-SN-1	0.224	TE	G
5155	A-HCS-311-4	RIGID ANCHOR ON 2CS2-311-SN-1	0.249	TE	G
5194	A-HCS-312-1	RIGID ANCHOR ON 2CS2-312-SN-1	0.294	TE	G
70	A-HCS-21-1	PENET. M-11	0.007	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-3108-1A-32, FIGURE 3.6A-12, FIGURE 3.6A-26, FIGURE 3.6A-27

FIGURE 3.6A-12-CALC (Continued)

SUMMARY OF CALCULATIONS

CVCS LETDOWN PIPE BREAK LOCATION

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI- TERIA[a]	BREAK TYPE[b]
870	A-HCS-93-3	STRESS ANCHOR POINT	0.149	TE*	G

NOTE: * PIPE WHIP RESTRAINTS ARE NOT PROVIDED FOR THE POSTULATED PIPE BREAK BECAUSE THE PIPE IS NOT NEAR ANY ESSENTIAL COMPONENTS AND IS WITHIN ITS OWN PROTECTED SHIELD.

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-143-1-1A-36, FIGURE 3.6A-12, FIGURE 3.6A-26

FIGURE 3.6A-12-CALC (Continued)

SUMMARY OF CALCULATIONS

CS PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (a)	BREAK TYPE (b)
460	A-HCS-305-9	ANCHOR POINT	0.323	TE	G
455	A-HCS-306-1	ANCHOR POINT	0.416	TE	G
2257	A-HCS-305-6	ANCHOR POINT	0.170	TE	G
3396	A-HCS-305-3	ANCHOR POINT	0.130	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

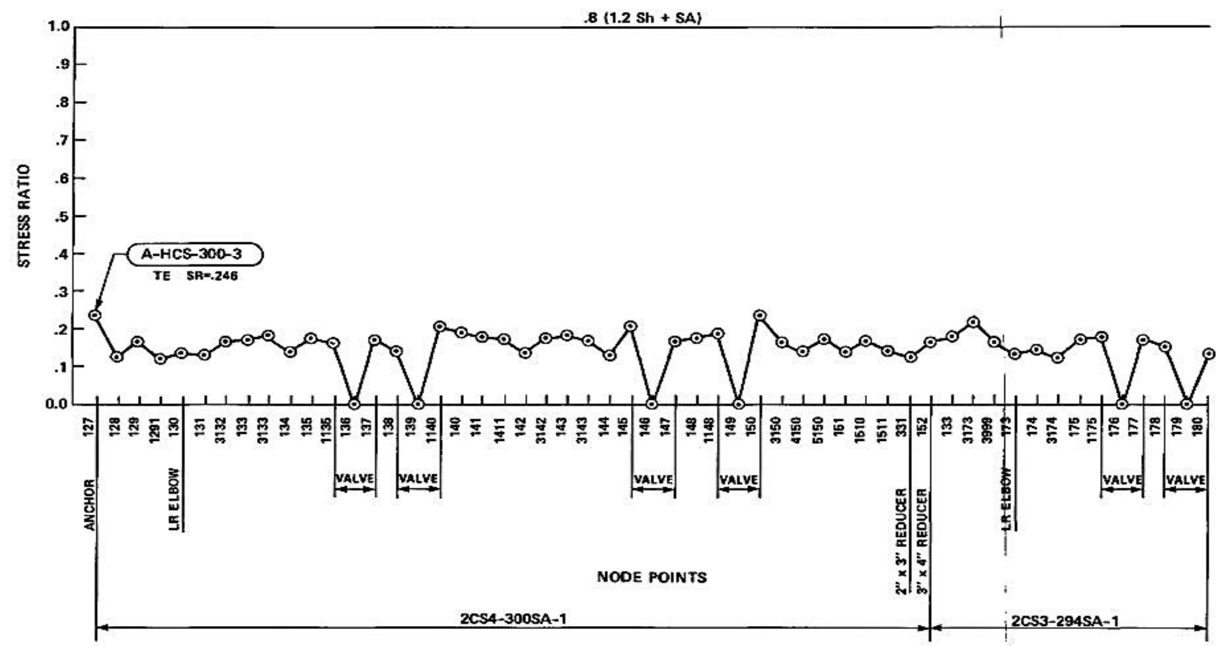
MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-3096-1&2-1A-29, FIGURE 3.6A-12, FIGURE 3.6A-13

FIGURE 3.6A-12-PLOT-A

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING



REF. NO.:
CALC. NO.: PRA-W-CS-141- 1&2-27, FIGURE 3.6A-12, FIGURE 3.6A-12-CALC

FIGURE 3.6A-12-PLOT-A (Continued)

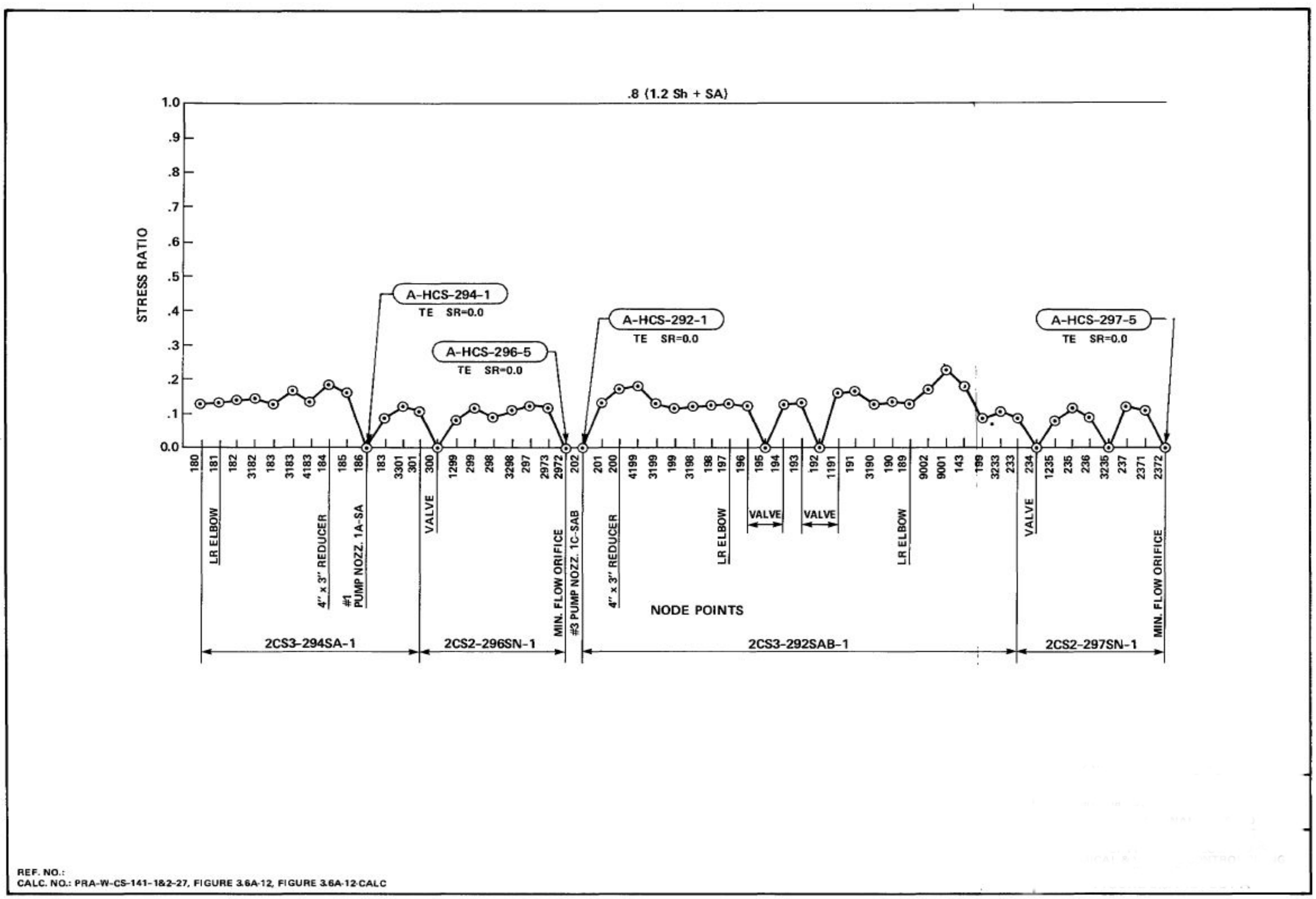


FIGURE 3.6A-12-PLOT-A (Continued)

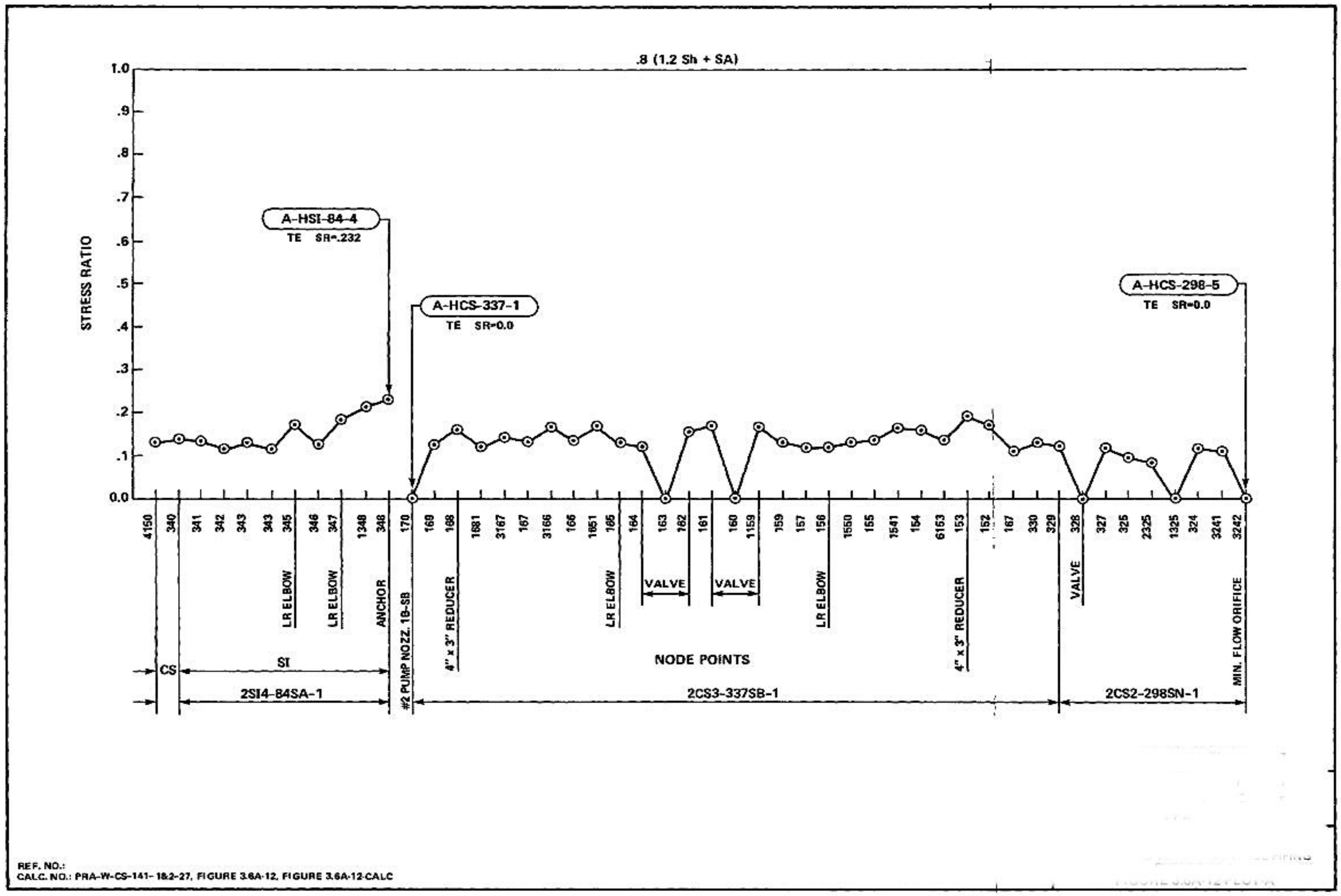
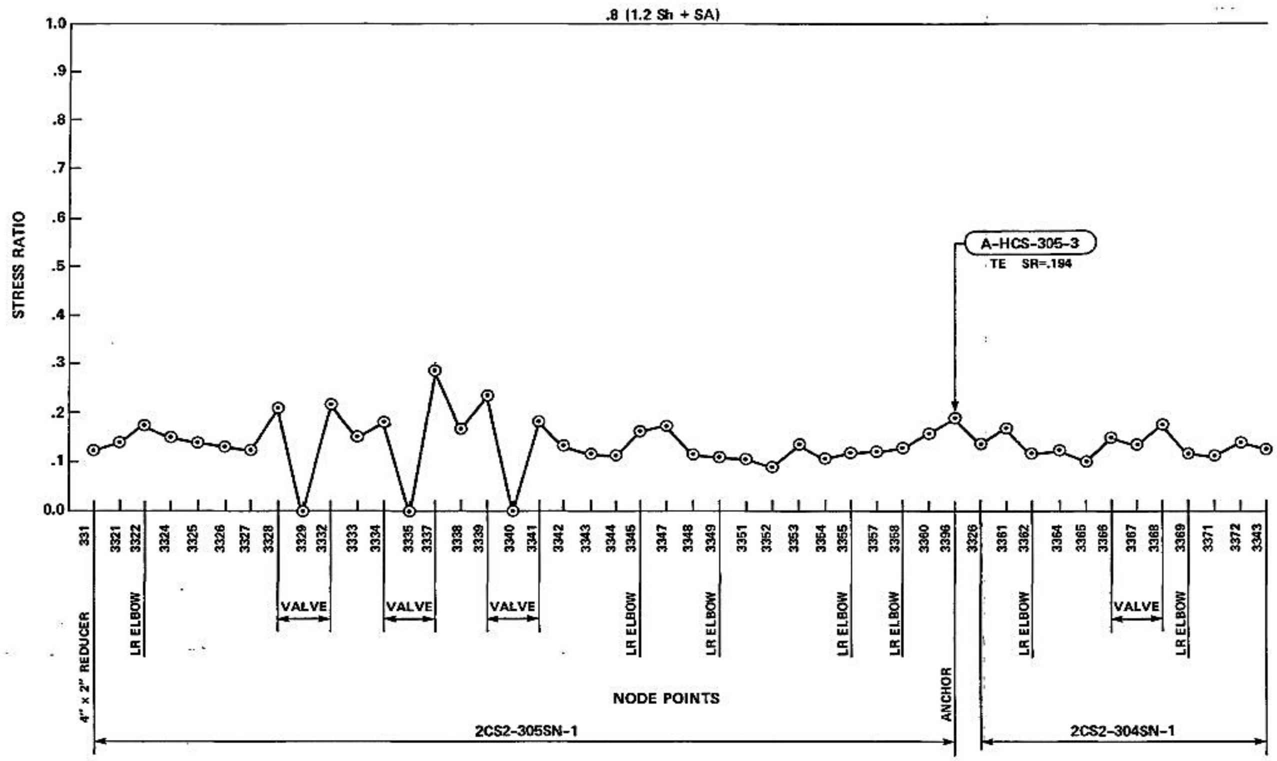


FIGURE 3.6A-12-PLOT-A (Continued)



AMENDMENT NO. 48

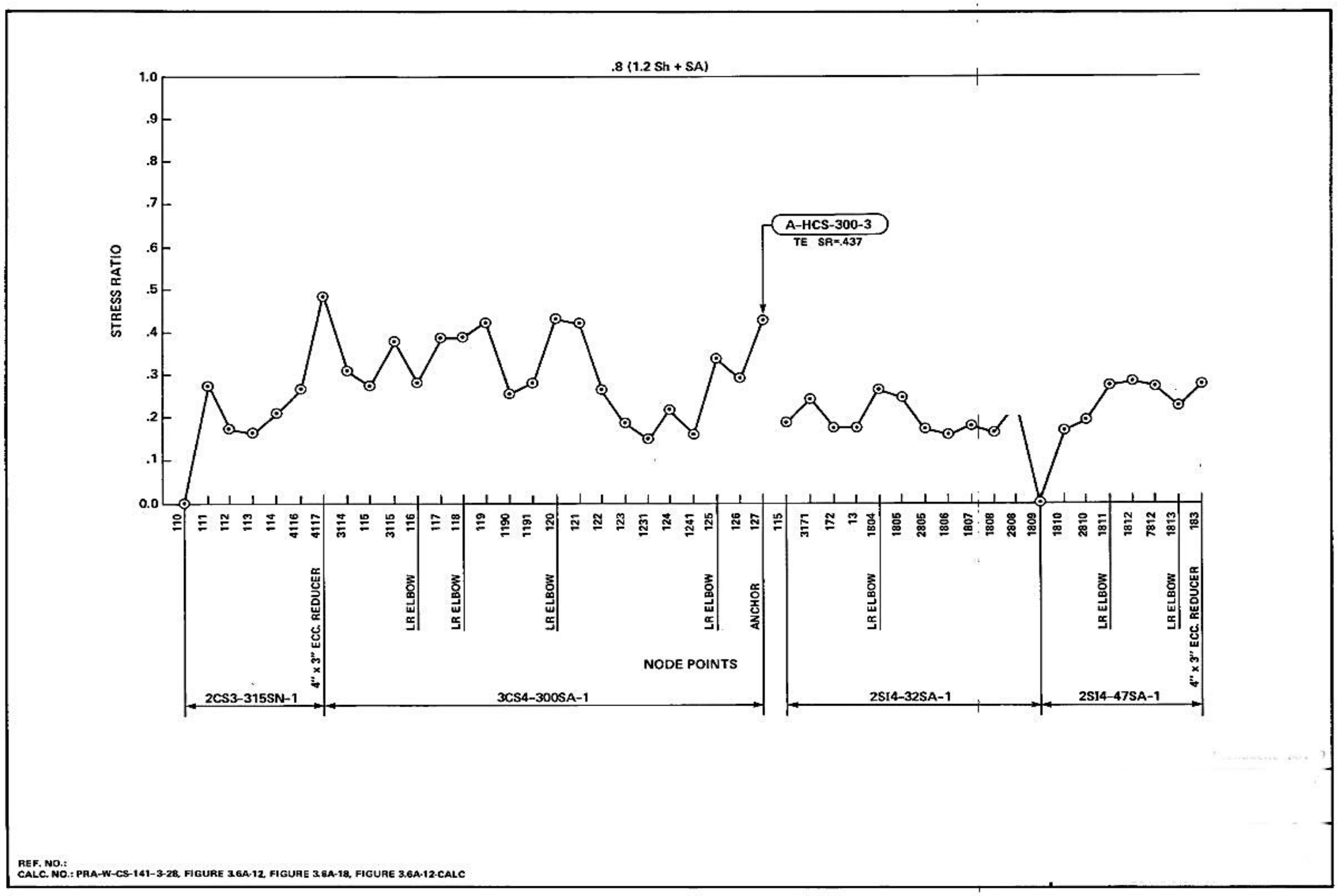
SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

REACTOR AUXILIARY BUILDING
PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING
SHEET 4 OF 4
FIGURE 3.6A-12-PLOT-A

REF. NO.:
CALC NO.: PRA-W-CS-141-1&2-27, FIGURE 3.6A-12, FIGURE 3.6A-12-CALC

FIGURE 3.6A-12-PLOT-B

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING



REF. NO.:
CALC. NO.: PR

FIGURE 3.6A-12-PLOT-B (Continued)

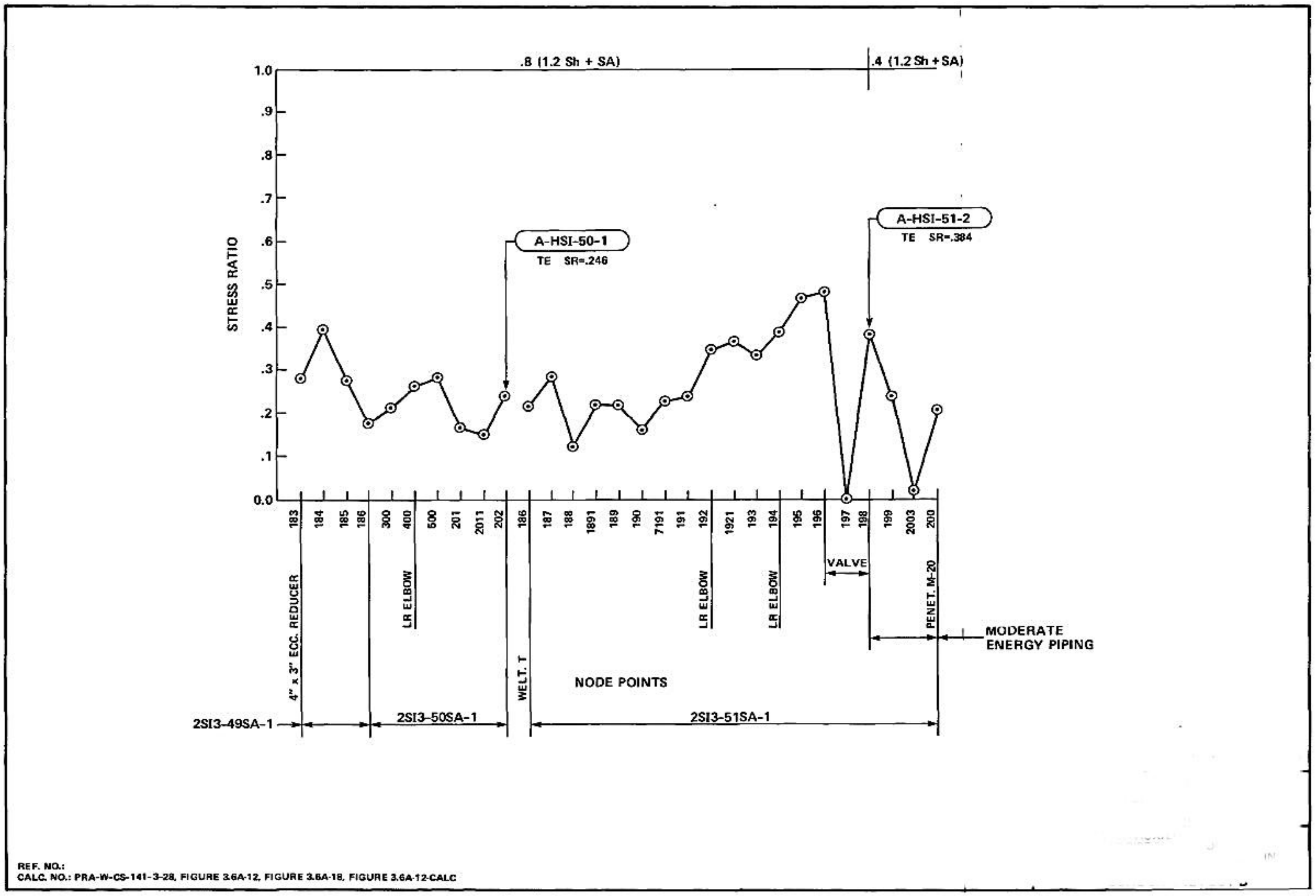


FIGURE 3.6A-12-PLOT-C

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING

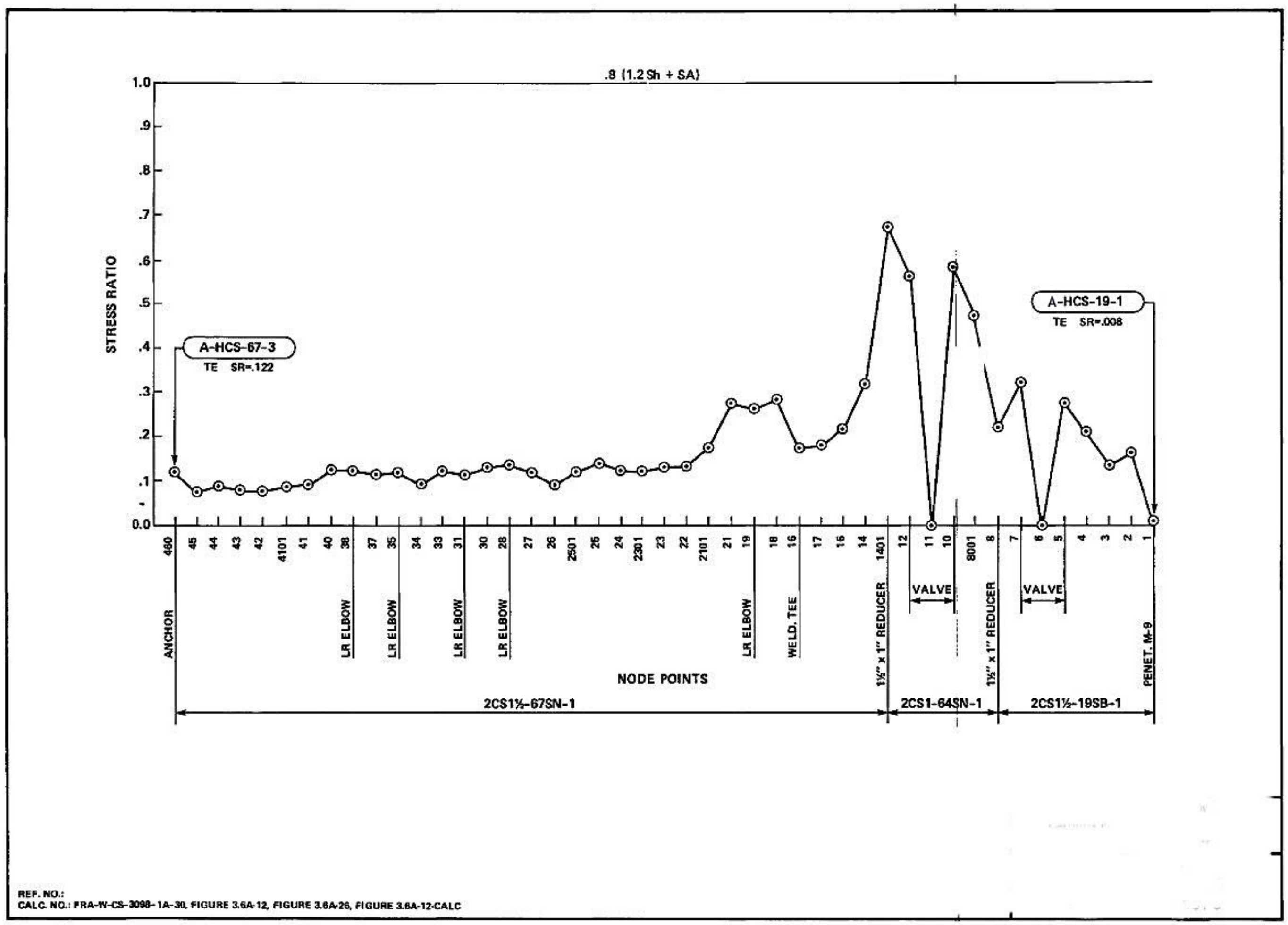


FIGURE 3.6A-12-PLOT-D

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING

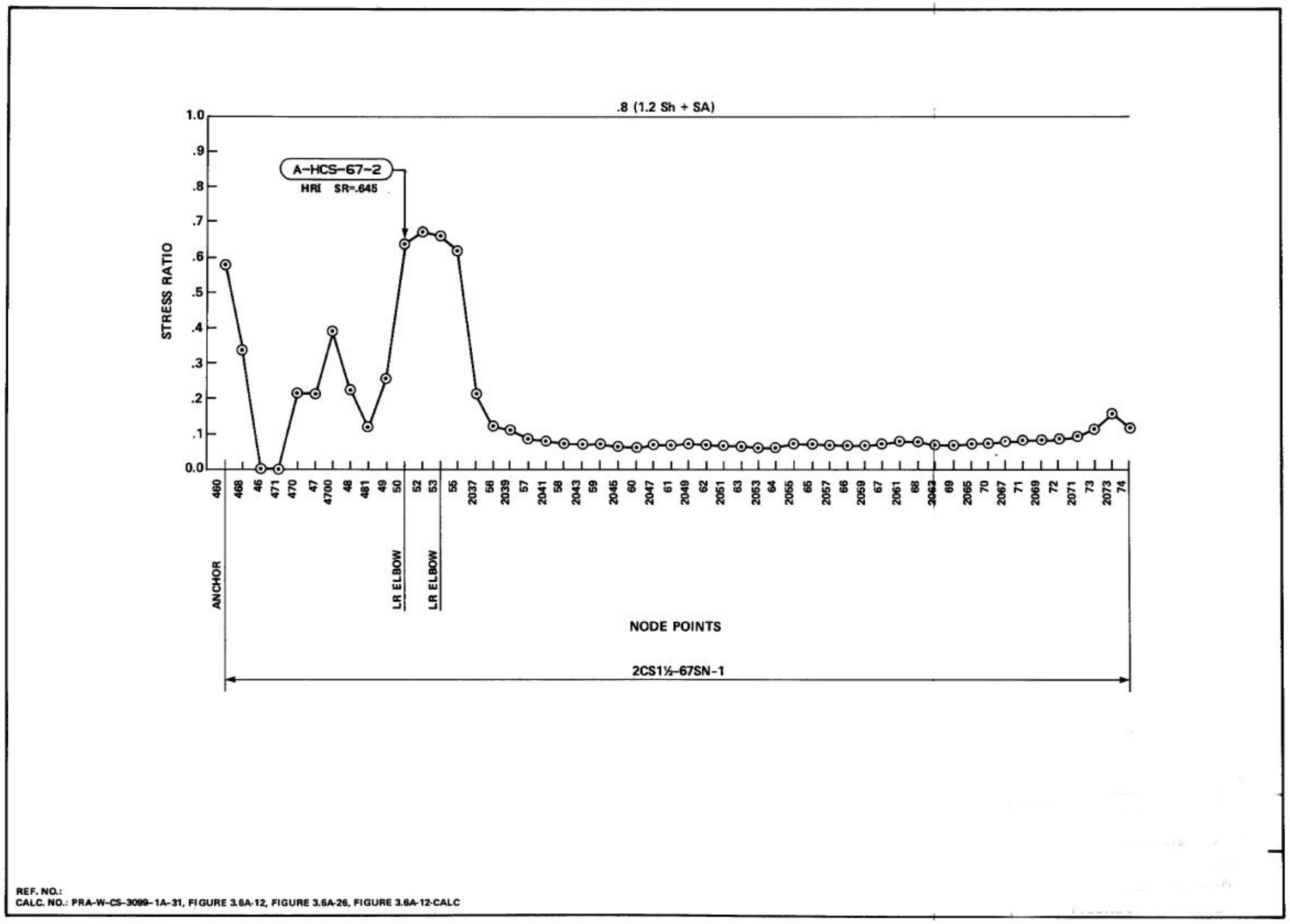


FIGURE 3.6A-12-PLOT-D (Continued)

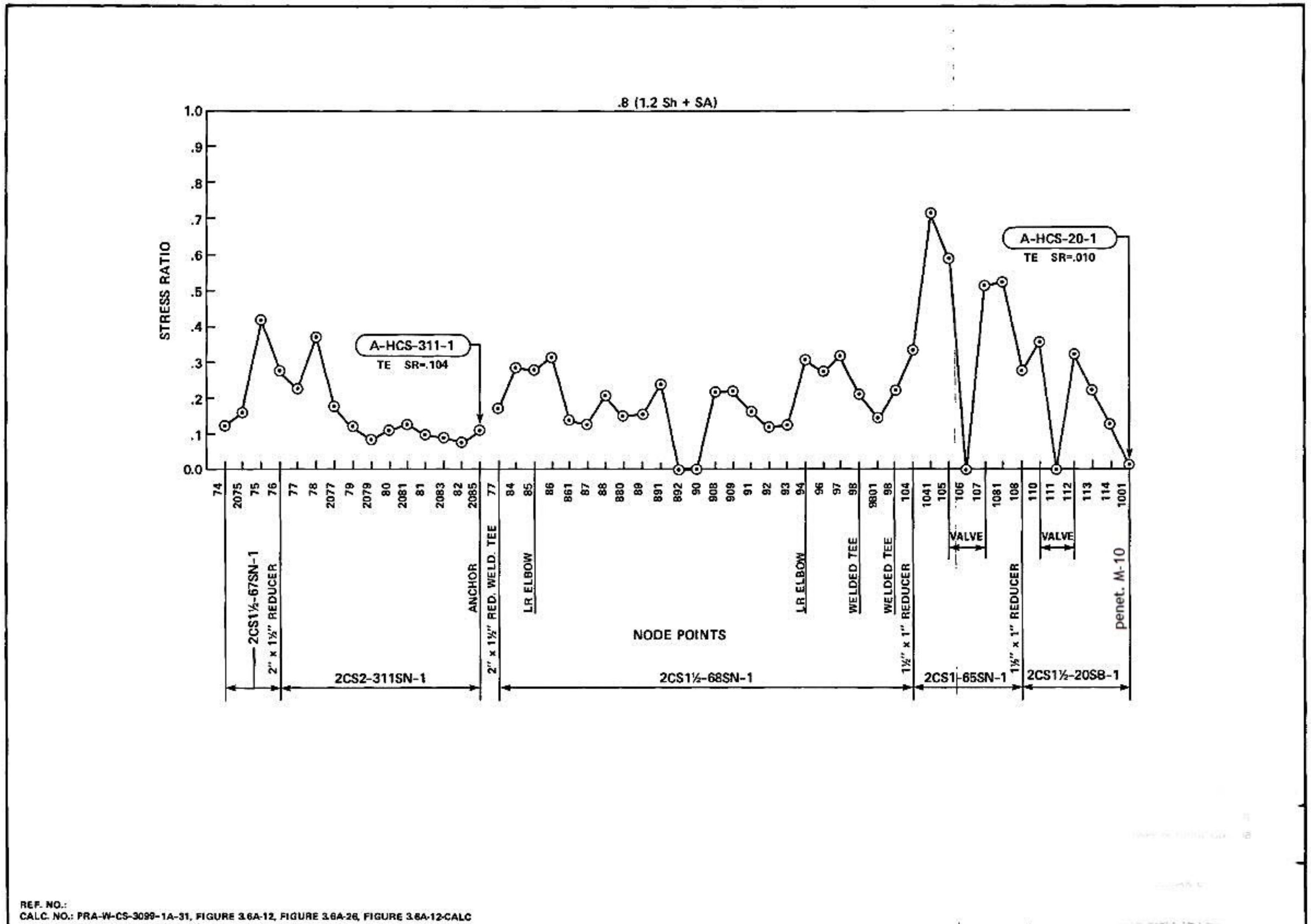


FIGURE 3.6A-12-PLOT-E

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING

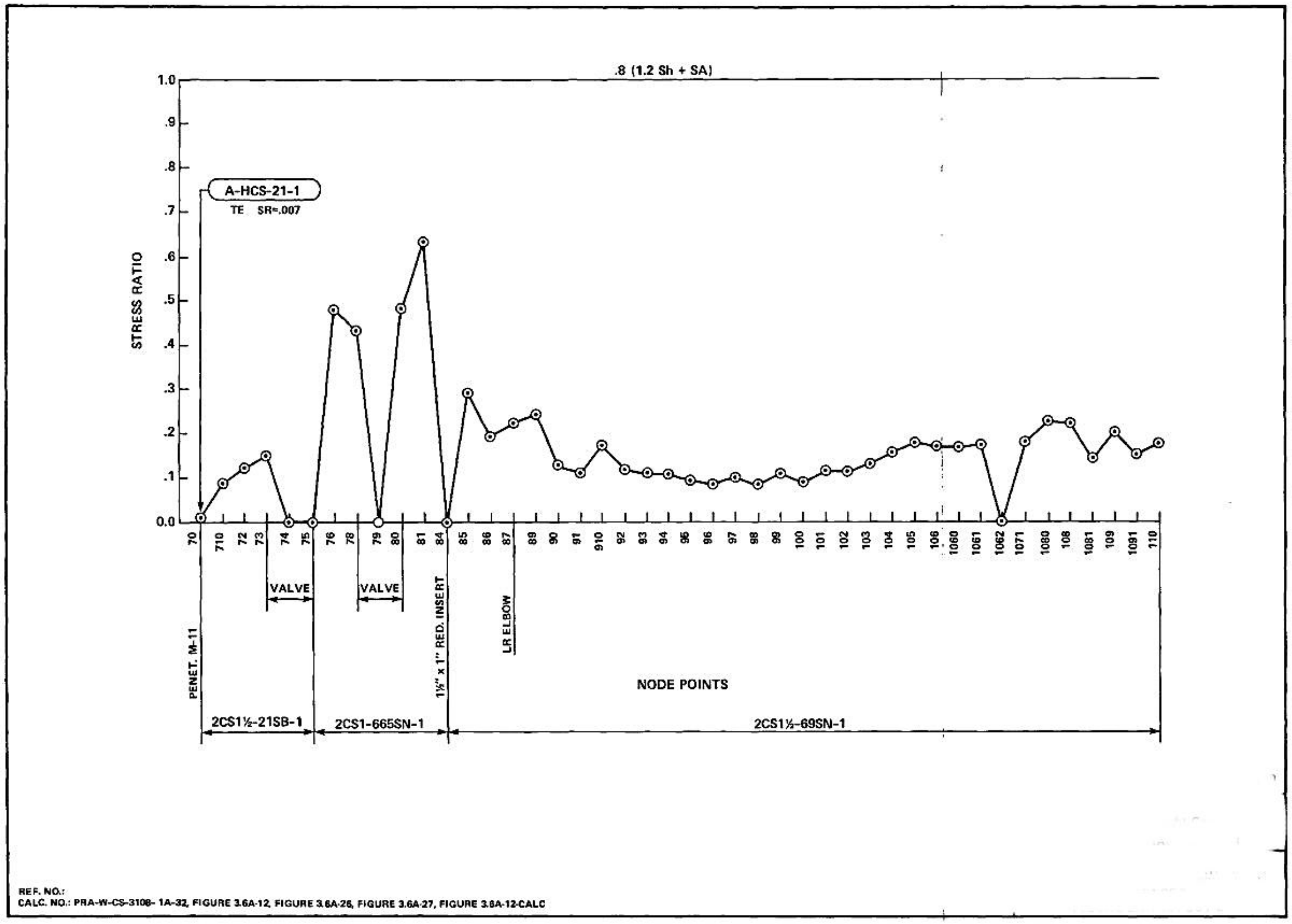


FIGURE 3.6A-12-PLOT-E (Continued)

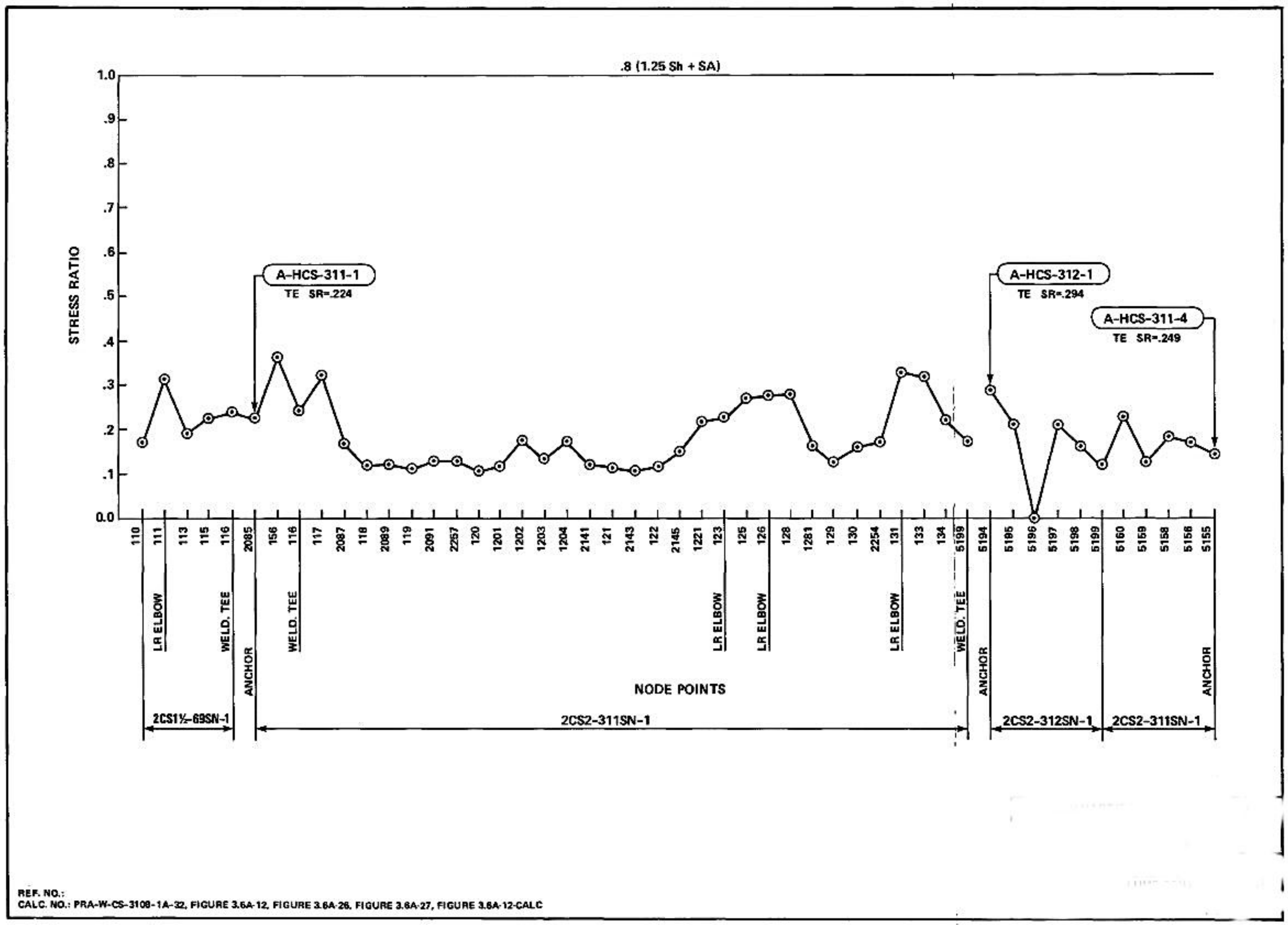


FIGURE 3.6A-12-PLOT-F

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING

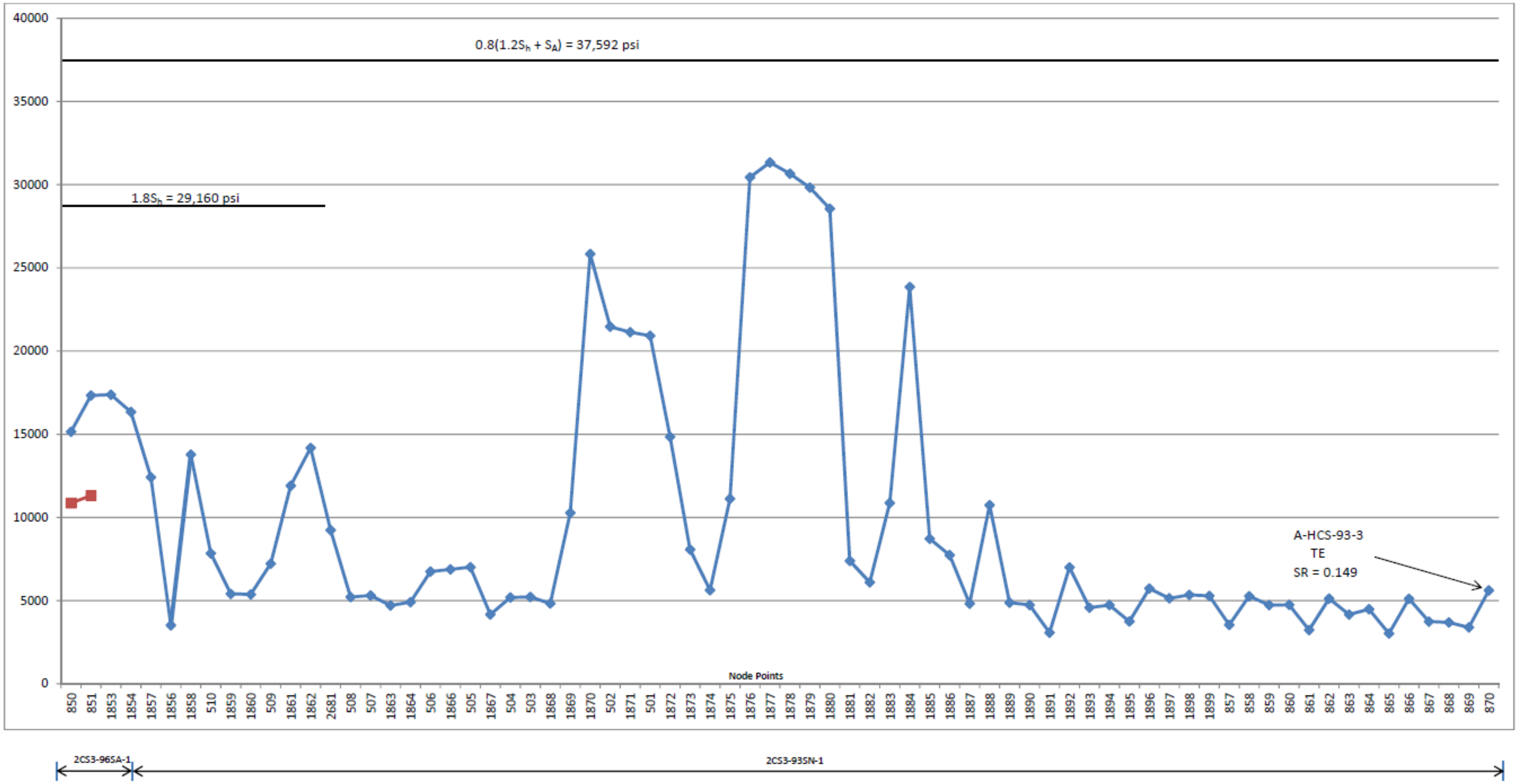


Fig: CVCS Letdown Pipe Stress vs. Allowables

FIGURE 3.6A-12-PLOT-G

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL

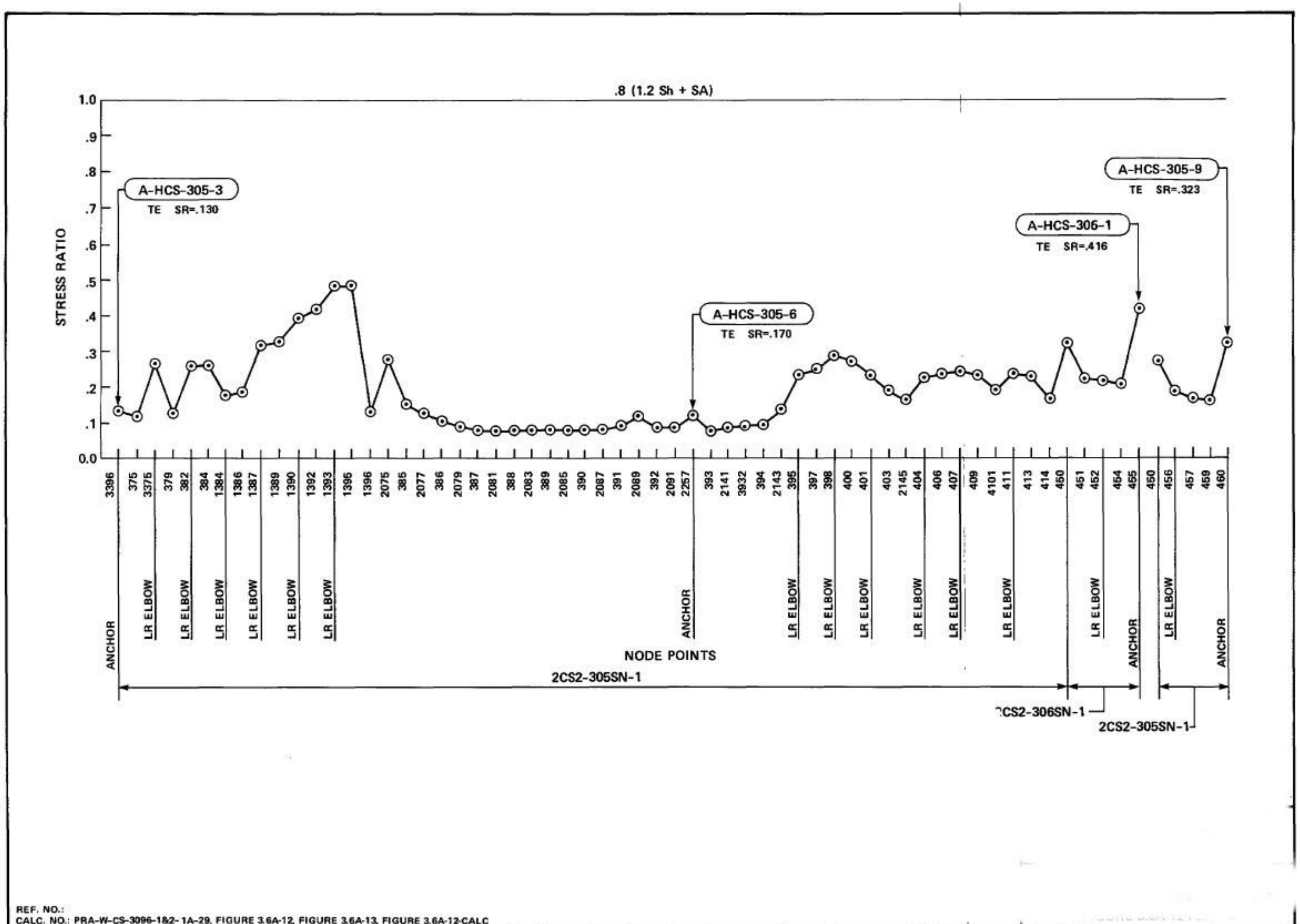


FIGURE 3.6A-14 CALC

REACTOR CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
RC LOOP DRAIN LINES

SUMMARY OF CALCULATIONS

RC LOOP 2&3 DRAINS PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI- TERIA (A)	BREAK TYPE (B)
BOP	W					
167	1130	R-HRC-41-1	RCL NOZZLE	-	TE	G
171	1190	R-HRC-41-2	END OF VALVE	-	TE	G
2	1030	R-HRC-63-1	RCL NOZZLE	-	TE	G
6	1150	R-HRC-63-6	END OF VALVE	-	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:
CALC NO.: PRA-W-RC-0056

FIGURE 3.6A-14 CALC Continued)

REACTOR CONTAINMENT BUILDING SUMMARY OF BREAK LOCATIONS
PRESSURIZER SPRAY & AUXILIARY SPRAY PIPING

SUMMARY OF CALCULATIONS 741

PRESSURIZER SPRAY PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
BOP	W					
1	1402	R-HRC-231-1	PZR. NOZZLE	Ø.1175	TE	G
3	5010	R-HRC-231-2	BUTT WELD @ ELBOW	0.939	HRI	G.S
4	5020	R-HRC-231-3	6 X 4 REDUCER	0.8+	HRI	G.S
5	5030	R-HRC-122-1	6 X 4 REDUCER	0.8+	HRI	G.S
16	3640	R-HRC-122-2	6 X 3/4 BRANCH	0.6	HRI	G.S
18	4500	R-HRC-122-3	6 X 6 X 4 SP. TEE	0.6	HRI	G.S
146	3020	R-HRC-25-1	COLD LEG NOZZLE	-	TE.	G
226.1	1010	R-HRC-44-1	COLD LEG NOZZLE	-	TE	G
1300	5210	R-HRC-122-4	6 X 2 REDUCER	0.950+	HRI	G.S
5132	5220	R-HRC-123-1	6 X 2 REDUCER	0.950+	HRI	G
5137	5230	R-HRC-123-2	VALVE SOCKET WELD	0.817	HRI	G
5139	5270	R-HCS-558-1	VALVE SOCKET WELD	0.817	HRI	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

+ = HIGH STRESS PIONT-HSP (>2.4Sm)

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:

CALC NO PRA-W-RD-3029/3016-10-54

FIGURE 3.6A-14 CALC Continued)

REACTOR CONTAINMENT BUILDING SUMMARY OF BREAK LOCATIONS
PRESSURIZER SPRAY & AUXILIARY SPRAY PIPING

SUMMARY OF CALCULATIONS

PRESSURIZER SPRAY PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
BOP	W					
6141	5320	A-HCS-658-2	2 X 1 REDUCER	0.53	HSP	G
5143	6040	A-HCS-658-3	VALVE SOCKET WELD	0.817	HRI	G
2	—		ELBOW B.W.	0.939+		

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

\+ = HIGH STRESS POINT-HSP (>2.4Sm)

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.
CALC NO. PRA-W-RC-3029/3014-1C-54

FIGURE 3.6A-15

CONTAINMENT BUILDING – BREAK & RESTRAINT LOCATIONS – REACTOR COOLANT PIPING – PARTIAL PLANS & SECTIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-15 CALC

REACTOR CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
PRESSURIZER SAFETY RELIEF PIPING

SUMMARY OF CALCULATIONS

PRESSURIZER SAFETY & RELIEF PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI- TERIA (A)	BREAK TYPE (B)
BOP	W					
1	5020	R-HRC-128-1	NOZZLE #4C	0.0681	TE	G
10	4230	R-HRC-128-2	6" X 3/4" BRANCH	0.941	HRI	G/S
1201	4320	R-HRC-128-3	BUTT WELD	–	TE	G
3301	2320	R-HRC-124-3	BUTT WELD	–	TE	G
36	2230	R-HRC-124-2	6" X 3/4" BRANCH	0.941	HRI	G/S
45	2020	R-HRC-124-1	NOZZLE #4A	0.0681	TE	G
5101	3320	R-HRC-126-3	BUTT WELD	–	TE	G
53	3250	R-HRC-126-2	6" X 3/4" BRANCH	0.941	HRI	G/S
62	3020	R-HRC-126-1	NOZZLE #4B	0.0681	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENEGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CACL NO PRA-W-RC-3041-1-C-50

FIGURE 3.6A-15 CALC Continued)

SUMMARY OF CALCULATIONS

PRESSURIZER SAFETY & RELIEF PIPE BREAK LOCATIONS

NODE POINT BOP	NODE POINT W	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(A)	BREAK TYPE(B)
111	5020	R-HRC-135-1	NOZZLE # 1X-5N	-	TE	G
7007	5250	R-HRC-135-2	6" X 3/4" BRANCH	0.950	HRI	G/S
1111	8000	R-HRC-135-3	6"X 3" TEE	0.950	HRI	G/S
1115	7000	R-HRC-135-4	6"X 3" TEE	0.950	HRI	G/S
1117	5530	R-HRC-135-5	LG. END OF 6 X 3 RED	0.950	HRI	G/S
9117	5540	R-HRC-141-1	SM. END OF 6 X 3 RED	0.950	HRI	G
1119	5580	R-HRC-141-2	LR. ELBOW	0.412	HRI	G
9119	5620	R-HRC-141-3	LR. ELBOW	0.412	HRI	G
9120	5540	R-HRC-141-4	LR. ELBOW	0.412	HRI	G
1120	5680	R-HRC-141-5	LR. ELBOW	0.412	HRI	G
9121	5710	R-HRC-141-6	VALVE END	0.412	HRI	G
9123	5740	R-HRC-141-7	VALVE END	0.412	HRI	G
1125	5770	R-HRC-141-8	VALVE END	-	TE	G
1110	8030	R-HRC-139-1	6" X 3" TEE	0.412	HRI	G
1109	8050	R-HRC-139-2	LR ELBOW	0.412	HRI	G
1192	8090	R-HRC-139-3	LR ELBOW	0.412	HRI	G
1193	8120	R-HRC-139-4	VAVLE END	0.412	HRI	G
9195	8150	R-HRC-139-5	VALVE END	0.412	HRI	G
1197	8190	R-HRC-139-6	VALVE END	-	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:
CACL NO PRA-W-RC-3041-1C-50

FIGURE 3.6A-15 CALC Continued)

SUMMARY OF CALCULATIONS

PRESSURIZER SAFETY & RELIEF PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI- TERIA(A)	BREAK TYPE(B)
BOP	W					
9115	7030	R-HRC-140-1	6" X 3" TEE	0.412	HRI	G
9116	7050	R-HRC-140-2	LR ELBOW	0.412	HRI	G
1171	7090	R-HRC-140-3	LR ELBOW	0.412	HRI	G
9172	7120	R-HRC-140-4	VALVE END	0.412	HRI	G
9174	7150	R-HRC-140-5	VALVE END	0.412	HRI	G
1176	7190	R-HRC-140-6	VALVE END	0.0	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF NO.
CALC NO PRA-W-RC-3041-IC-50

FIGURE 3.6A-17 CALC

REACTOR AUXILIARY BUILDING SUMMARY OF PIPE BREAK LOCATIONS
SAFETY INJECTION PIPING

SUMMARY OF CALCULATIONS

SAFETY INJECTION PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (A)	BREAK TYPE (B)
5	A-HSI-3-3	UPSTREAM END OF VALVE 2SI-V501SB	0.330	TE	G
348	A-HSI-84-4	PENET THRU RAB WALL E ANCHOR	0.528	TE	G
164	A-HSI-44-1	ANCHOR, SI-H-414	0.342	TE	G
75	A-HSI-44-2	BIT INLET NOZZLE	0.201	TE	G
42	A-HSI-11-1	BIT OUTLET NOZZLE	0.209	TE	G
19	A-HSI-11-2	UPSTREAM END OF 2SI-V506SA-1	0.301	TE	G
9	A-HSI-12-1	UPSTREAM END OF 2SI-V505SB-1	0.301	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

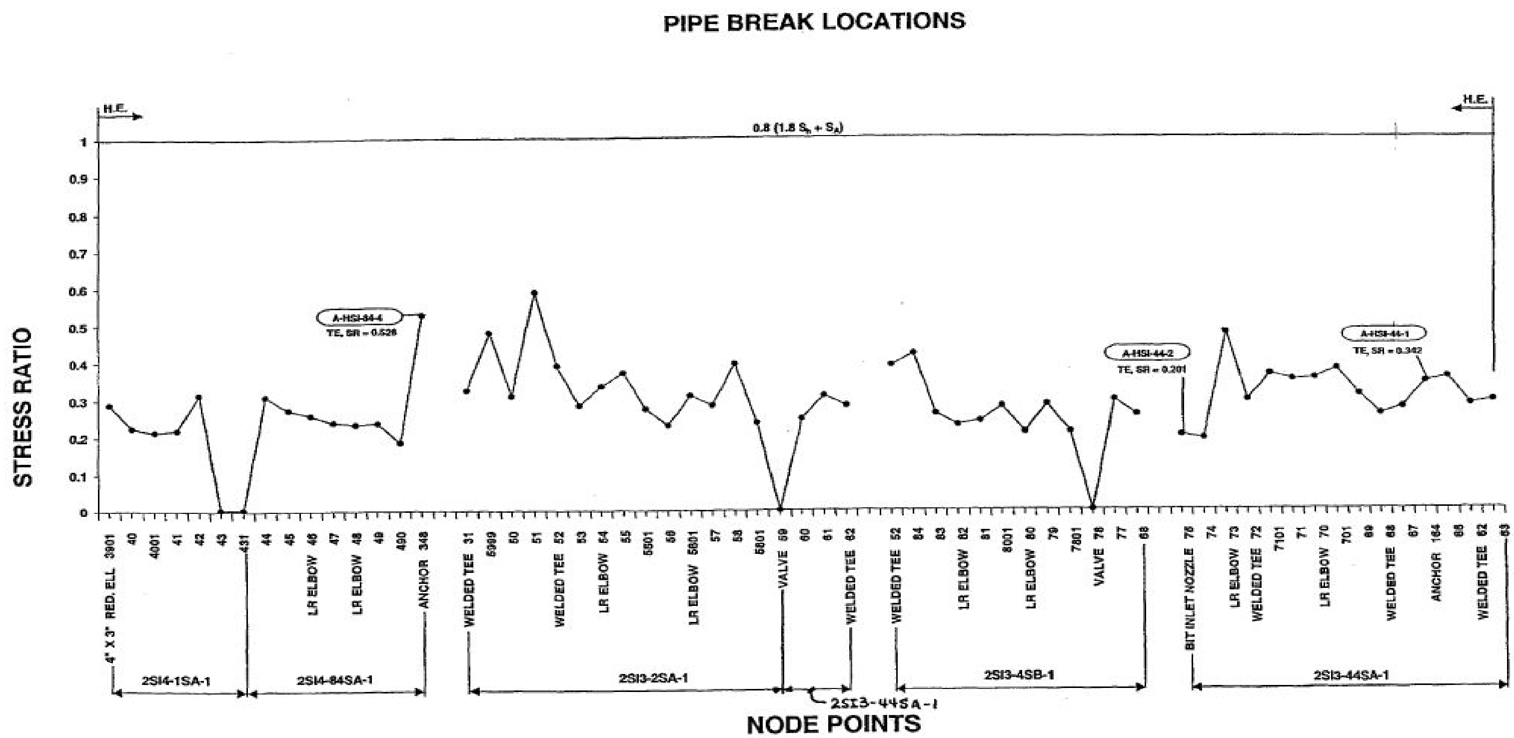
MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-SI-3027-1A-25, FIGURE 3.6A-27, FIGURE 3.6A-19
PRA-W-SI-3026-1A-100

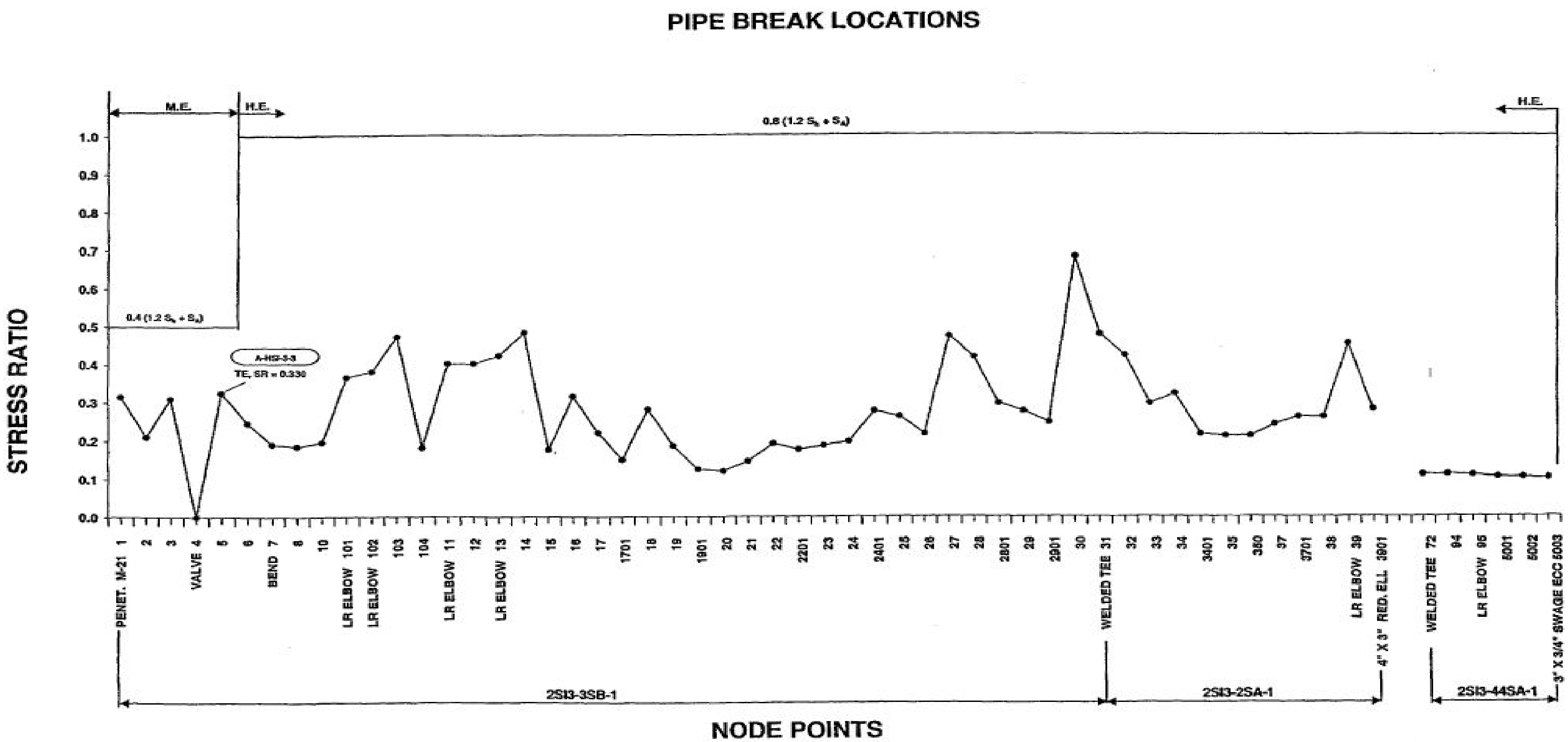
FIGURE 3.6A-17-PLOT-A

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
SAFETY INJECTION PIPING



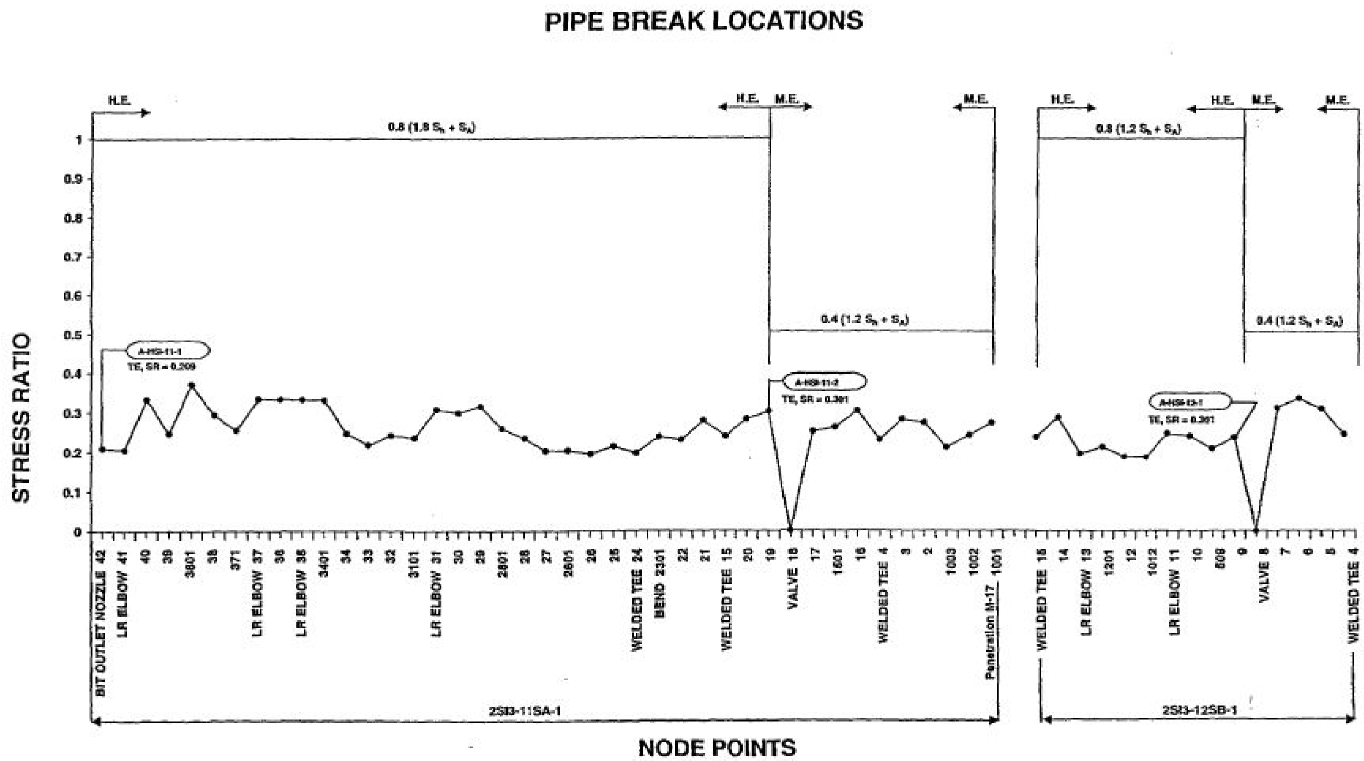
REF. NO.:
CALC. NO. PRA-W-SI-3027-1A-25, FIGURE 3.6A-17, FIGURE 3.6A-19, FIGURE 3.6A-17-CALC
CALC. NO. PRA-W-SI-3026-1A-100

FIGURE 3.6A-17-PLOT-A (Continued)



REF. NO.:
CALC. NO. PRA-W-SI-3Ø27-1A-25

FIGURE 3.6A-17-PLOT-A (Continued)



REF. NO.:
CALC. NO. PRA-W-SI-3026-1A-100

FIGURE 3.6A-18-CALC

REACTOR AUXILIARY BUILDING SUMMARY OF PIPE BREAK LOCATIONS
SAFETY INJECTION PIPING

SUMMARY OF CALCULATIONS

SAFETY INJECTION PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA[a]	BREAK TYPE[b]
202	A-HSI-50-1	ANCHOR POINT	0.131	TE	G
5	A-HSI-50-4	UP STREAM END OF VALVE	0.280	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

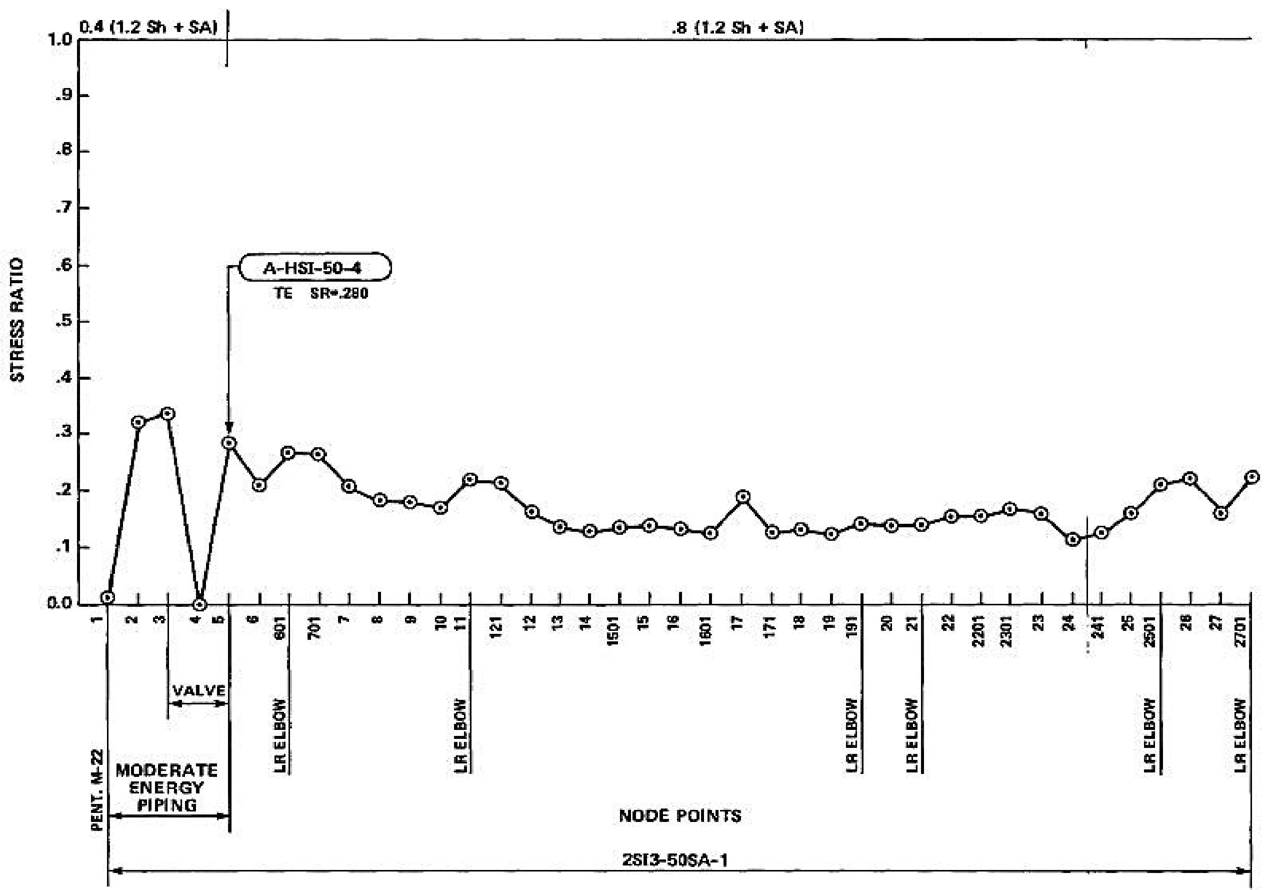
MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-SI-3025-1A-26, FIGURE 3.6A-18

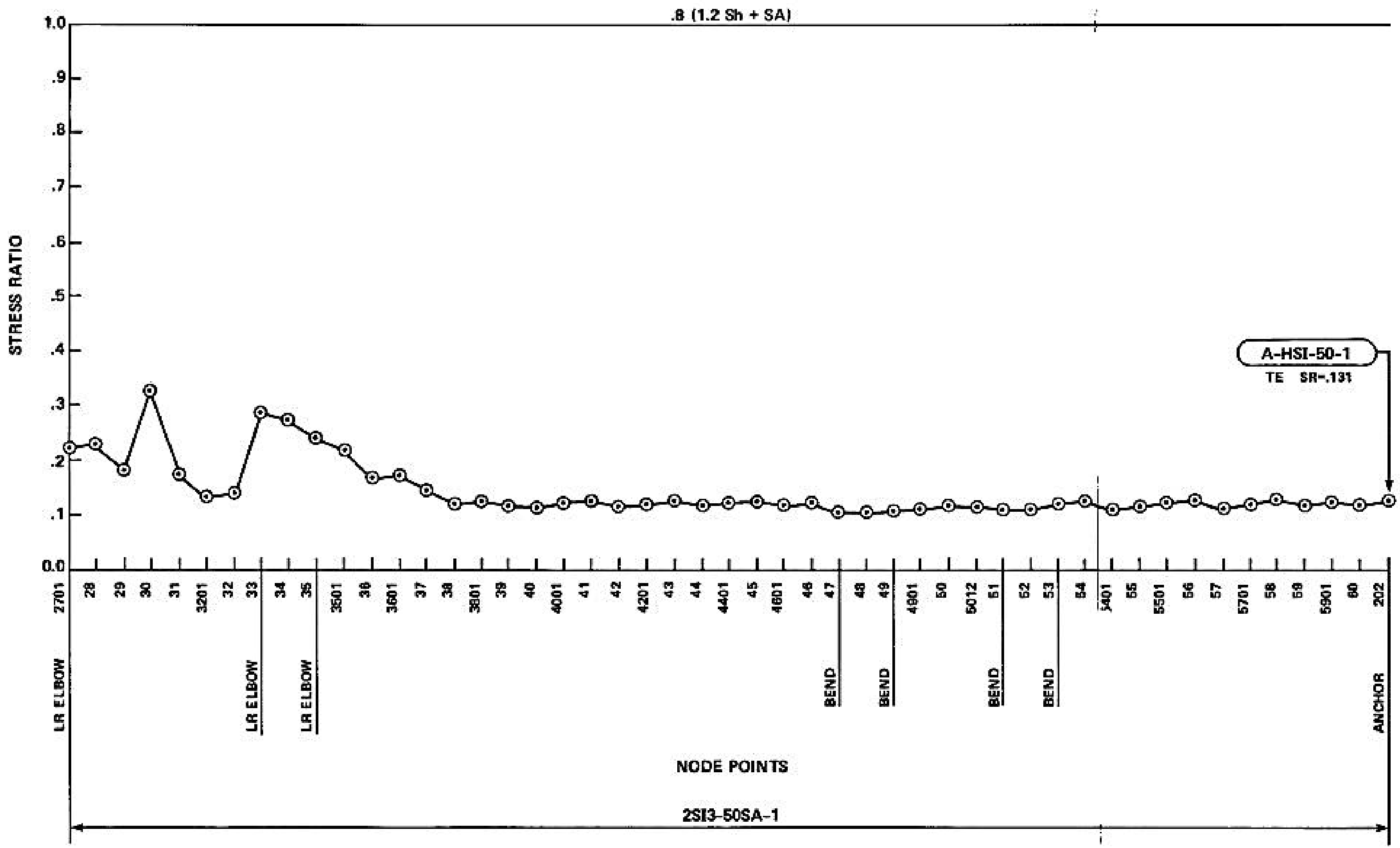
FIGURE 3.6A-18-PLOT-A

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
SAFETY INJECTION PIPING



REF. NO.:
CALC. NO.: PRA-W-SI-3025-1A-26, FIGURE 3.6A-18, FIGURE 3.6A-18-CALC

FIGURE 3.6A-18-PLOT-A (Continued)



REF. NO.:
CALC. NO.: PRA-W-SI-3025-1A-26, FIGURE 3.6A-18, FIGURE 3.6A-18-CALC

FIGURE 3.6A-20.1-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
REACTOR COOLANT PIPING – LOOP #1

SUMMARY OF CALCULATIONS

SI & RC PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TYPE (A)	BREAK TYPE (B)
1	R-HSI-158-1	ACCUM. NOZZ. 1A-SA	–	TE	G
31	R-HRC-26-6	BRANCH CONN.	–	*TE	G

*BREAKS ARE POSTULATED ON CLASS 1 PIPING AS PER W CRITERIA

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-SI-3032-1C-44, FIGURE 3.6A-20, FIGURE 3.6A-22

FIGURE 3.6A-20.1-CALC (Continued)

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
REACTOR COOLANT PIPING – LOOP #2

SUMMARY OF CALCULATIONS

RC & SI PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
31	R-HRC-46-4	LR ELBOW	Ø.761	HRI*	G
33	R-HRC-46-6	RCL NOZZLE	-	TE	G
1	R-HSI-159-1	ACCUM. NOZZ. 1B-SB	-	TE	G

*CLASS 1 PIPING BREAKS ARE POSTULATED AS PER WESTINGHOUSE CRITERIA

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-SI-3Ø33-1C-42, FIGURE 3.6A-2Ø, FIGURE 3.6A-22

FIGURE 3.6A-20.1-CALC (Continued)

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
REACTOR COOLANT PIPING – LOOP #3

SUMMARY OF CALCULATIONS

SI & RC PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA (a)	BREAK TYPE (b)
2	R-HRC-65-6	BRANCH CONN.	–	*TE	G
34	R-HSI-160-1	ACCUM. NOZZ. 1C-SA	–	TE	G

NOTE: *BREAKS ARE SELECTED PER W CRITERIA (LETTER CQL 8418)

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-SI-3034-1C-43, FIGURE 3.6A-20, FIGURE 3.6A-22

FIGURE 3.6A-20.2-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
SAFETY INJECTION – COLD LEG

SUMMARY OF CALCULATIONS

HI & LO HD COLD LEG PIPE BREAK LOCATIONS

NODE POINT BOP	NODE POINT W	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRITERIA (A)	BREAK TYPE (B)
502	1030	R-HRC-27-4	RCL NOZZLE	–	TE	G
503	1050	R-HRC-27-3	LR ELBOW	0.20	HRU	G
504	1060	R-HRC-27-2	LR ELBOW	0.20	HRU	G
505	1070	R-HRC-27-1	END OF VALVE	0.24	TE	G
65	1030	R-HRC-47-4	RCL NOZZLE	–	TE	G
64	1050	R-HRC-47-3	LR ELBOW	0.20	HRU	G
63	1060	R-HRC-47-2	LR ELBOW	0.20	HRU	G
62	1080	R-HRC-47-1	END OF VALVE	0.24	TE	G
9051	1015	R-HRC-66-6	RCL NOZZLE	–	TE	G
9050	1020	R-HRC-66-5	LR ELBOW	0.20	HRU	G
9049	1025	R-HRC-66-4	LR ELBOW	0.20	HRU	G
9048	1040	R-HRC-66-3	LR ELBOW	0.20	HRU	G
9047	1045	R-HRC-66-2	LR ELBOW	0.20	HRU	G
9046	1055	R-HRC-66-1	END OF VALVE	0.24	TE	G
	1021		LR ELBOW	0.090	HRI	G
	1022		LR ELBOW	0.090	HRI	G
	1041		LR ELBOW	0.090	HRI	G
	1042		LR ELBOW	0.090	HRI	G

(A) HIGH-ENERGY SYSTEMS:
- TE = TERMINAL END
- HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT
- HRU = HIGH RELATIVE USAGE FACTOR

MODERATE-ENERGY SYSTEMS:
- HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:
- G = GUILLOTINE (CIRCUMFERENTIAL)
- S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:
- C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC NO.: PRA-W-RC-0046

FIGURE 3.6A-20

FIGURE 3.6A-20.3-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
SAFETY INJECTION – HOT LEG

SUMMARY OF CALCULATIONS

HOT LEG INJECTION PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(A)	BREAK TYPE(B)
BOP	W					
550	1020	R-HRC-10-6	RCL NOZZLE	-	TE	G
544	1060	R-HRC-10-1	END OF VALVE	0.270	TE	G
2	1020	R-HRC-29-4	RCL NOZZLE	-	TE	G
5	1060	R-HRC-29-1	END OF VALVE	0.270	TE	G
75	1010	R-HRC-49-4	RCL NOZZLE	-	TE	G
72	1025	R-HRC-49-1	END OF VALVE	0.270	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:
CALC NO.: PRA-W-SI-0055

FIGURE 3.6A.20

FIGURE 3.6A-21-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS – RHR SYSTEM

SUMMARY OF CALCULATIONS

RC-RHR PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI- TERIA(A)	BREAK TYPE(B)
26	A-HRC-51-1	BRANCH CONN	-	TE	G
21	A-HRC-51-6	UP STREAM END OF VALVE	-	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:
CALC NO PRA-W-CLASS-1-PIPING-46

FIGURE 3.6A.21 & 3.6A.22

FIGURE 3.6A-21-CALC (Continued)

SUMMARY OF CALCULATIONS

RC-RHR PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(A)	BREAK TYPE(B)
26	R-HRC-12-1	BRANCH CONN	-	TE	G
21	R-HRC-12-6	UP STREAM END OF VALVE	-	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF.NO.:
CALC NO PRA-W-CLASS-1-PIPING-46 FIGURE 3.6A-21 & 3.6A-22

FIGURE 3.6A-23-CALC

REACTOR CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
PRESSURIZER SURGE

SUMMARY OF CALCULATIONS

PZR SURGE PIPE BREAK LOCATIONS

NODE POINT		BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(A)	BREAK TYPE(B)
BOP	W					
1	145Ø	R-4RC-35-1	PZR NOZZLE	Ø.7266	TE	G
6	719Ø	R-HRC-35-4	14 x 1 BRANCH	Ø.35	HRU	G
12	7Ø1Ø	R-HRC-35-6	BRANCH FROM RC LEGS	Ø.85	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

HRU = HIGH RELATIVE USAGE FACTOR

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-RC-ØØ46, FIGURE 3.6A.23

AMENDMENT NO. 56

FIGURE 3.6A-24-CALC

CONTAINMENT BUILDING SUMMARY OF PIPE BREAK LOCATIONS
BLOWDOWN PIPING LOOP #1

SUMMARY OF CALCULATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	APPLICABLE CRITERIA	BREAK TYPE*
5ØØ	R-HBD-3-1	CONT. PEN. M-51	Ø.2Ø	TERMINAL END	G
536	R-HBD-1-1	STM. GEN. 1A-SN BD NOZZLE NO. 3Ø	Ø.76	TERMINAL END	G
561	R-HBD-65-1	STM. GEN. 1A-SN BD NOZZLE NO. 32	Ø.56	TERMINAL END	G

REF. NO:
CALC. NO: PRA-W-BD-3Ø7Ø-1C-17, FIGURE 3.6A-24

FIGURE 3.6A-24-CALC LOOP #2(Continued)

SUMMARY OF CALCULATIONS

BLOWDOWN SYSTEM PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
329	R-HBD-5-1	ST. GEN. 1B-SN, NOZZ. #30	0.85	TE	G
354	R-HBD-66-1	ST. GEN. 1B-SN, NOZZ. #32	0.44	TE	G
300	R-HBD-7-1	PENET. M-52	0.21	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-BD-3069-1C-18, FIGURE 3.6A-24

FIGURE 3.6A-24-CALC LOOP #3(Continued)

SUMMARY OF CALCULATIONS

BLOWDOWN SYSTEM PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
179	R-HBD-67-1	ST. GEN. NOZZ. #32	Ø.67	TE	G
4136	R-HBD-9-1	ST. GEN. NOZZ. #3Ø	Ø.75	TE	G
1ØØ	R-HBD-11-1	PENET. M-53	Ø.21	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO:
CALC. NO: PRA-W-BD-3Ø68-1C-19, FIGURE 3.6A-24

FIGURE 3.6A-24-PLOT-A

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
BLOWDOWN PIPING – LOOP #1

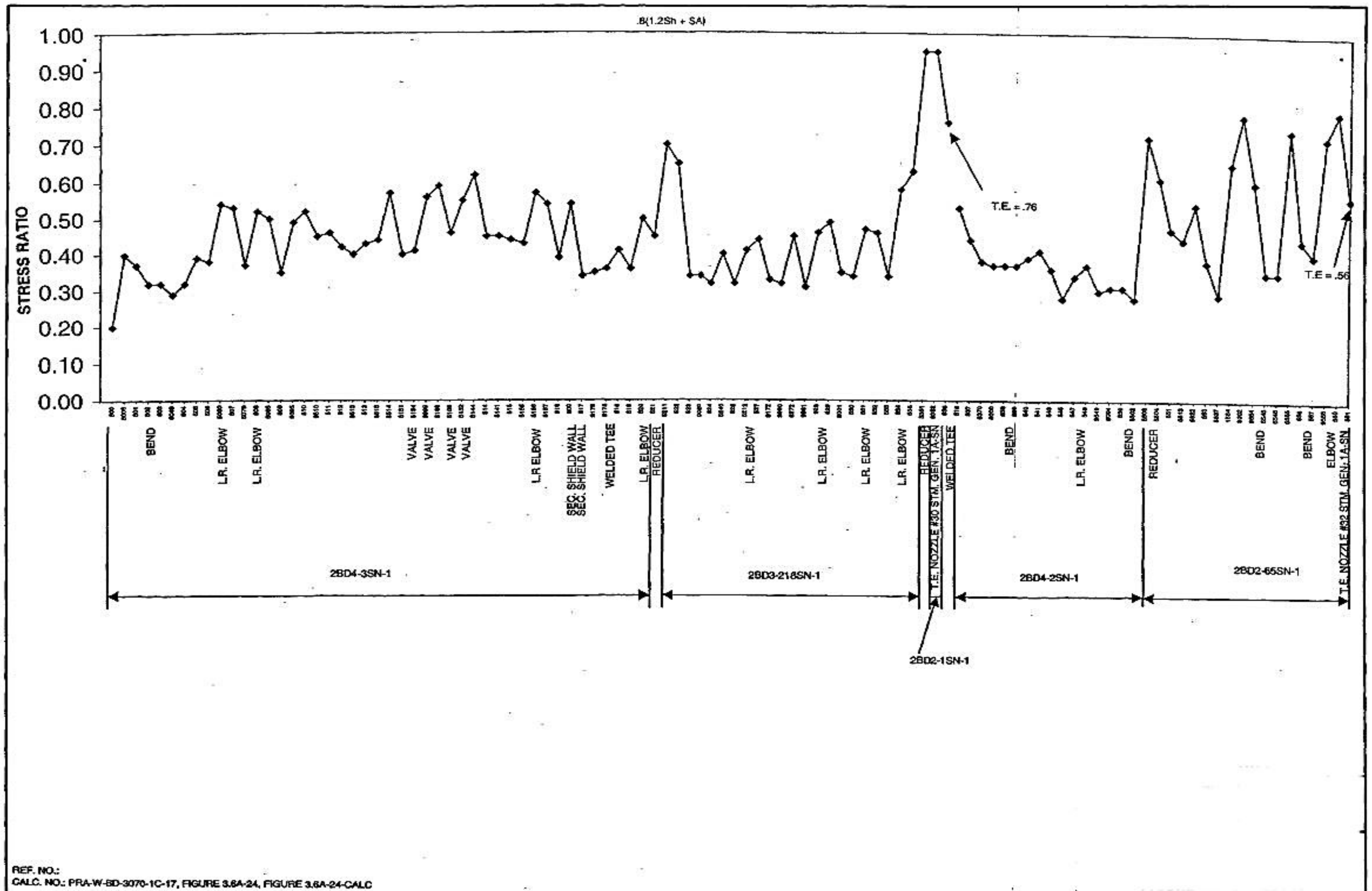
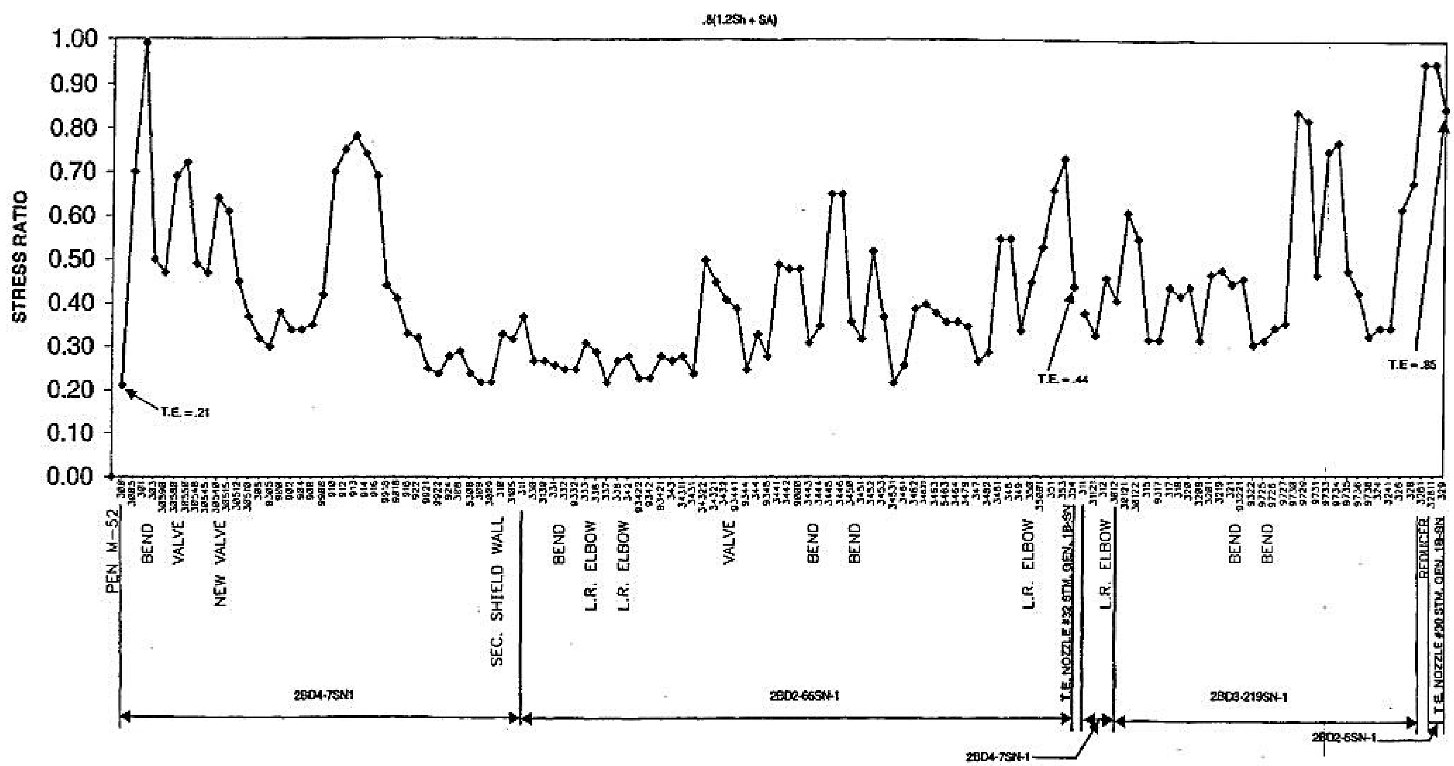


FIGURE 3.6A-24-PLOT-B

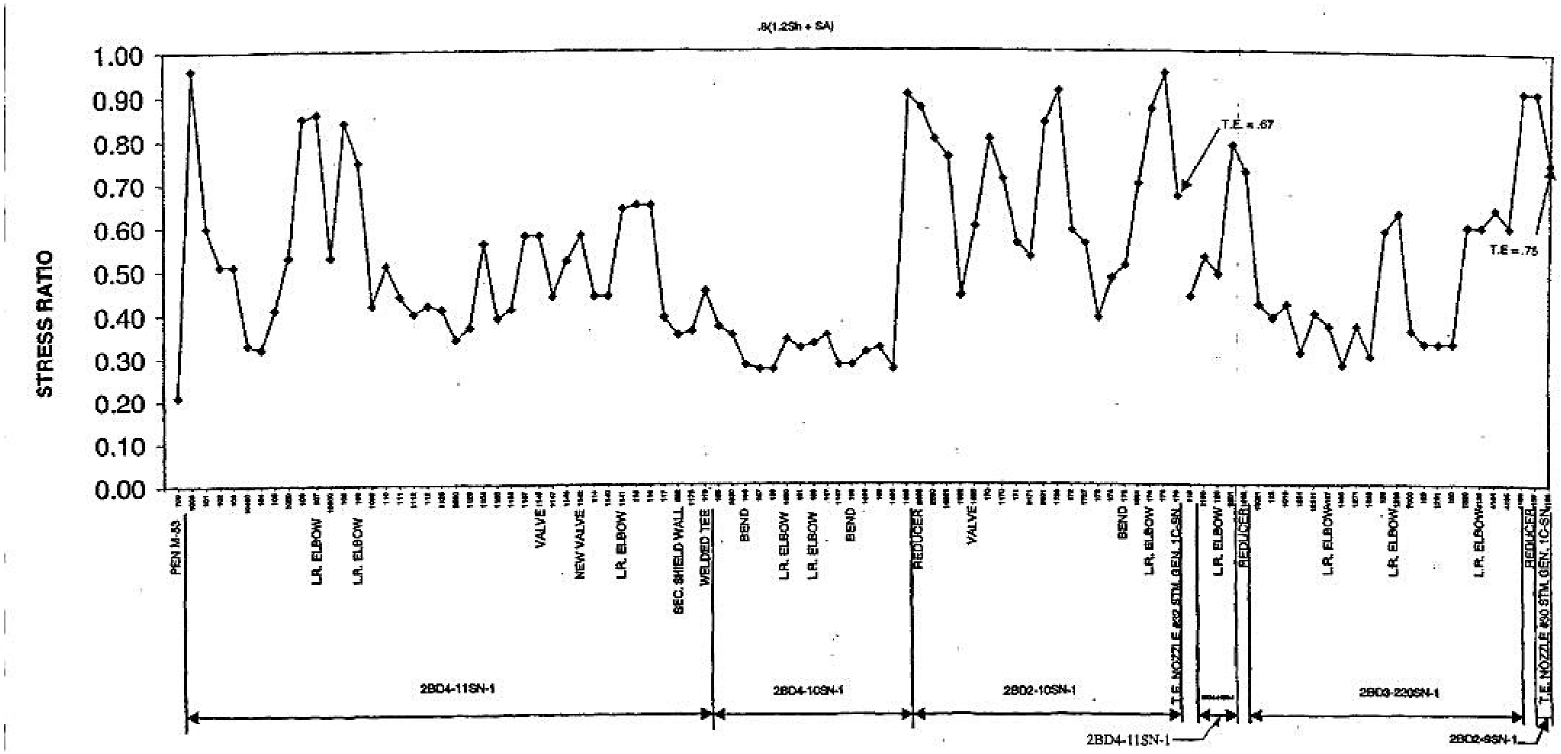
CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
BLOWDOWN PIPING – LOOP #2



REF NO.:
CALC. NO.: PRA-W-BD-3Ø69-1C-18, FIGURE 3.6A-24

FIGURE 3.6A-24-PLOT-C

CONTAINMENT BUILDING PLOT OF PIPE BREAK LOCATIONS
BLOWDOWN PIPING – LOOP #3



REF. NO.:
CALC. NO.: PRA-W-BD-3068-1C-19, FIGURE 3.6A-24, FIGURE 3.6A-24-CALC

FIGURE 3.6A-25-CALC

REACTOR AUXILIARY BUILDING SUMMARY OF PIPE BREAK LOCATIONS
STEAM GENERATOR BLOWDOWN PIPING

SUMMARY OF CALCULATIONS

BLOWDOWN SYSTEM PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
1614	A-HBD-13-4	ANCHOR POINT	0.235	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-BD-290-1-4, FIGURE 3.6A-25

FIGURE 3.6A-25-CALC (Continued)

SUMMARY OF CALCULATIONS

BLOWDOWN SYSTEM PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRITERIA[a]	BREAK TYPE[b]
1176	A-HBD-15-4	ANCHOR PT. 1176	0.191	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-BD-290-2-5, FIGURE 3.6A-25

FIGURE 3.6A-25-CALC (Continued)

SUMMARY OF CALCULATIONS

BLOWDOWN SYSTEM PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
1375	A-HBD-14-3	ANCHOR PT. 1375	0.206	TE	G

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END

HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)

S = SLOT (LONGITUDINAL)

MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-BD-290-3-6, FIGURE 3.6A-25

FIGURE 3.6A-25-PLOT-A

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
STEAM GENERATOR BLOWDOWN PIPING

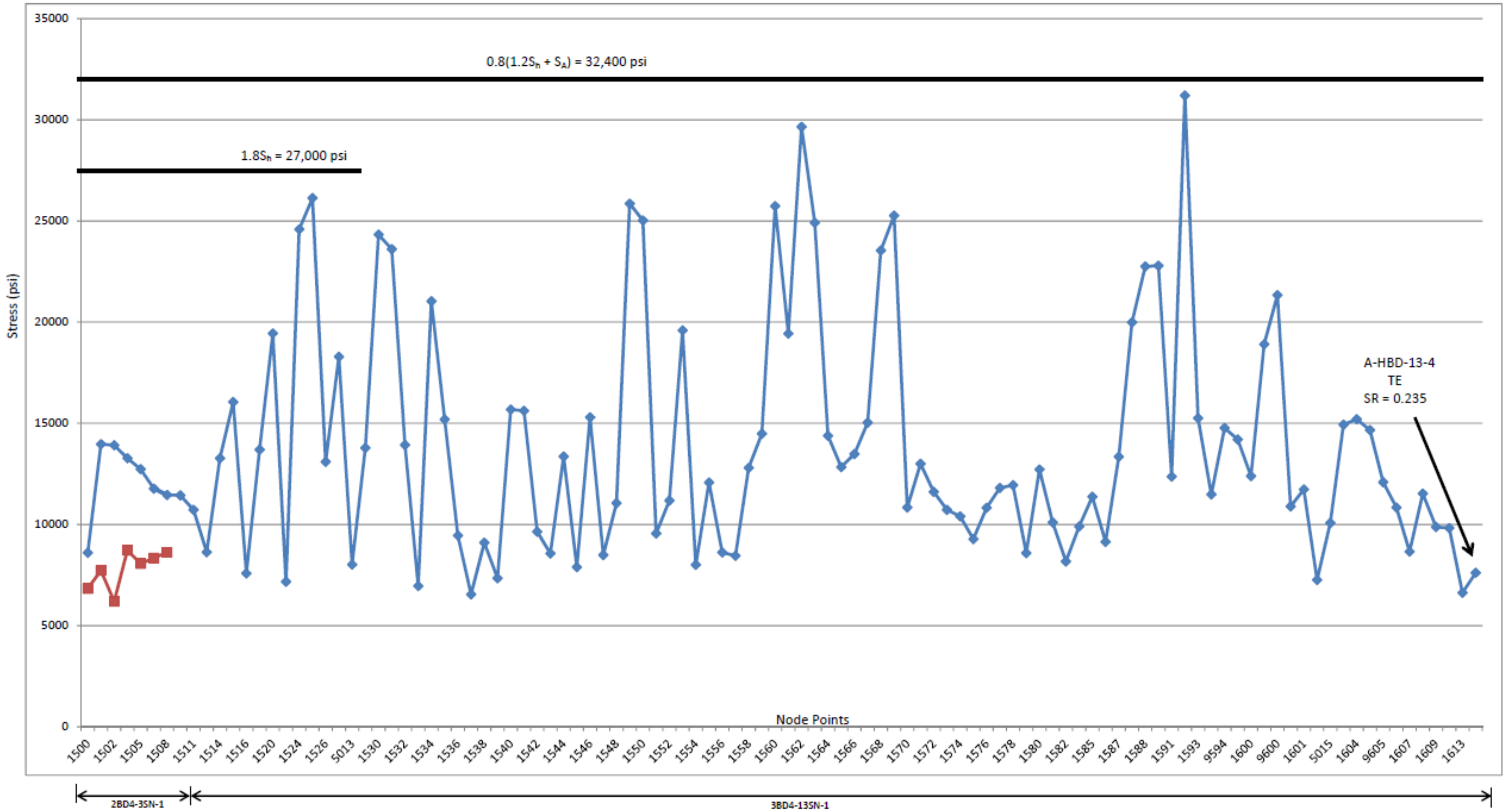


Fig: Blowdown System Pipe Stress vs. Allowables

FIGURE 3.6A-25-PLOT-B

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
STEAM GENERATOR BLOWDOWN PIPING

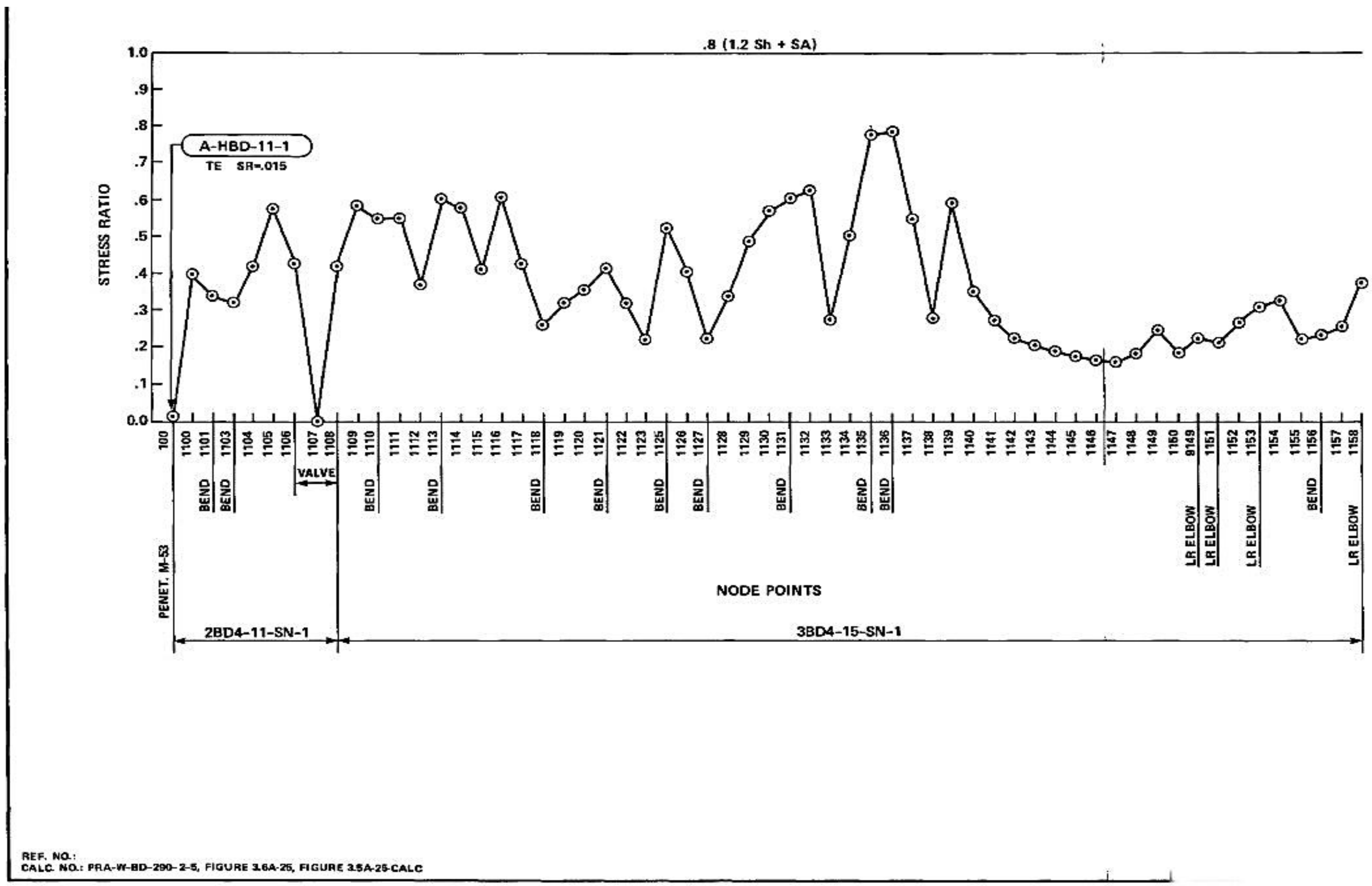


FIGURE 3.6A-25-PLOT-B

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
STEAM GENERATOR BLOWDOWN PIPING

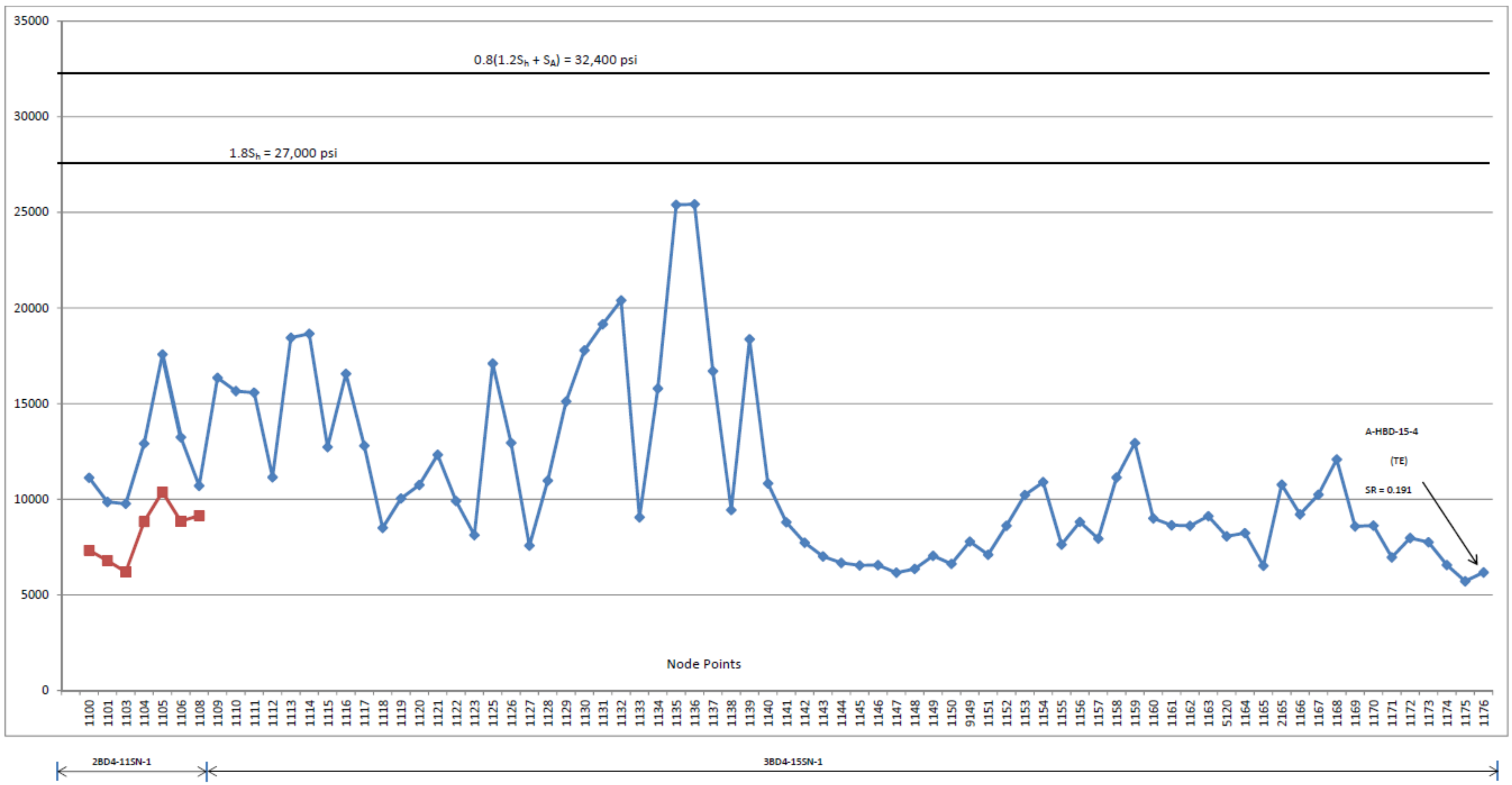


Fig: Blowdown System Pipe Stress vs. Allowables

FIGURE 3.6A-25-PLOT-C

REACTOR AUXILIARY BUILDING PLOT OF PIPE BREAK LOCATIONS
STEAM GENERATOR BLOWDOWN PIPING

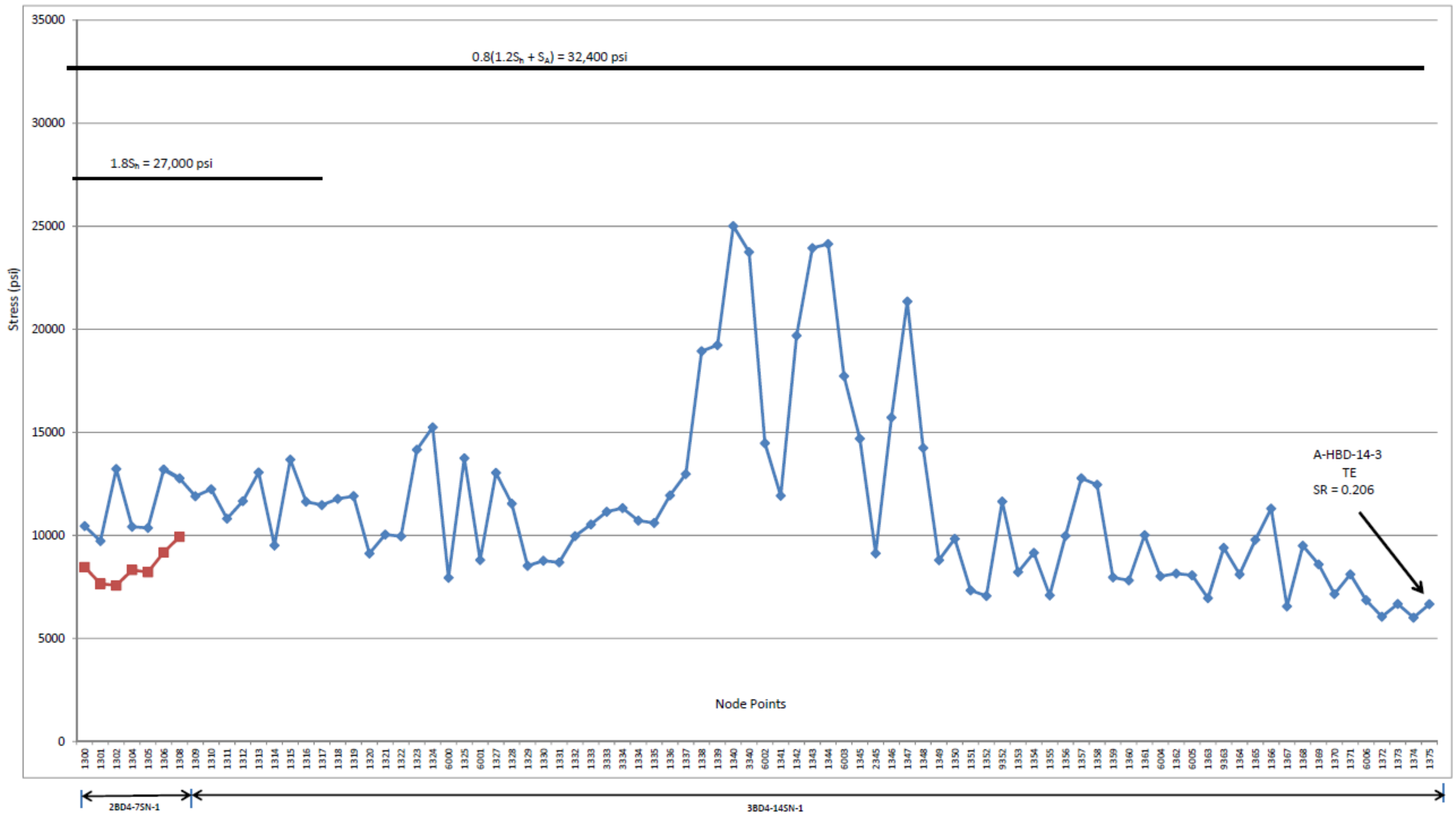


Fig: Blowdown System Pipe Stress vs. Allowables

FIGURE 3.6A-26-CALC

COMPOSITE PIPING – SHIELDED PIPE TUNNEL REACTOR AUXILIARY BUILDING
SUMMARY OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING

SUMMARY OF CALCULATIONS

CVCS PIPE BREAK LOCATIONS

NODE POINT	BREAK NUMBER	PHYSICAL DESCRIPTION OF BREAK POINT	STRESS RATIO/ USAGE FACTOR	CRI-TERIA(a)	BREAK TYPE(b)
870	A-HCS-93-3	ANCHOR POINT	.181	TE	G
856	A-HCS-93-5	ANCHOR POINT	.892	TE	G
952	A-HCS-385-1	ANCHOR POINT	0.0	TE *	G
956	A-HCS-209-2	ONE END OF VALVE	.558	TE *	G

*THESE POSTULATED BREAK POINTS ARE NOT SHOWN BECAUSE OF THE FOLLOWING REASONS:

1) THERE ARE NO ESSENTIAL EQUIPMENT LOCATED IN THE VICINITY OF THE PIPE BREAKS.
2) THE STEADY-STATE JET FORCE IS NEGLIGIBLE.
3) WITH THE ABOVE 2 REASONS, NO PIPE WHIP RESTRAINTS HAVE BEEN SPECIFIED.

(A) HIGH-ENERGY SYSTEMS:

TE = TERMINAL END
HRI = HIGH RELATIVE INTERMEDIATE STRESS POINT

MODERATE-ENERGY SYSTEMS:

HM = HIGH MODERATE ENERGY STRESS POINT

(B) HIGH-ENERGY SYSTEMS:

G = GUILLOTINE (CIRCUMFERENTIAL)
S = SLOT (LONGITUDINAL)

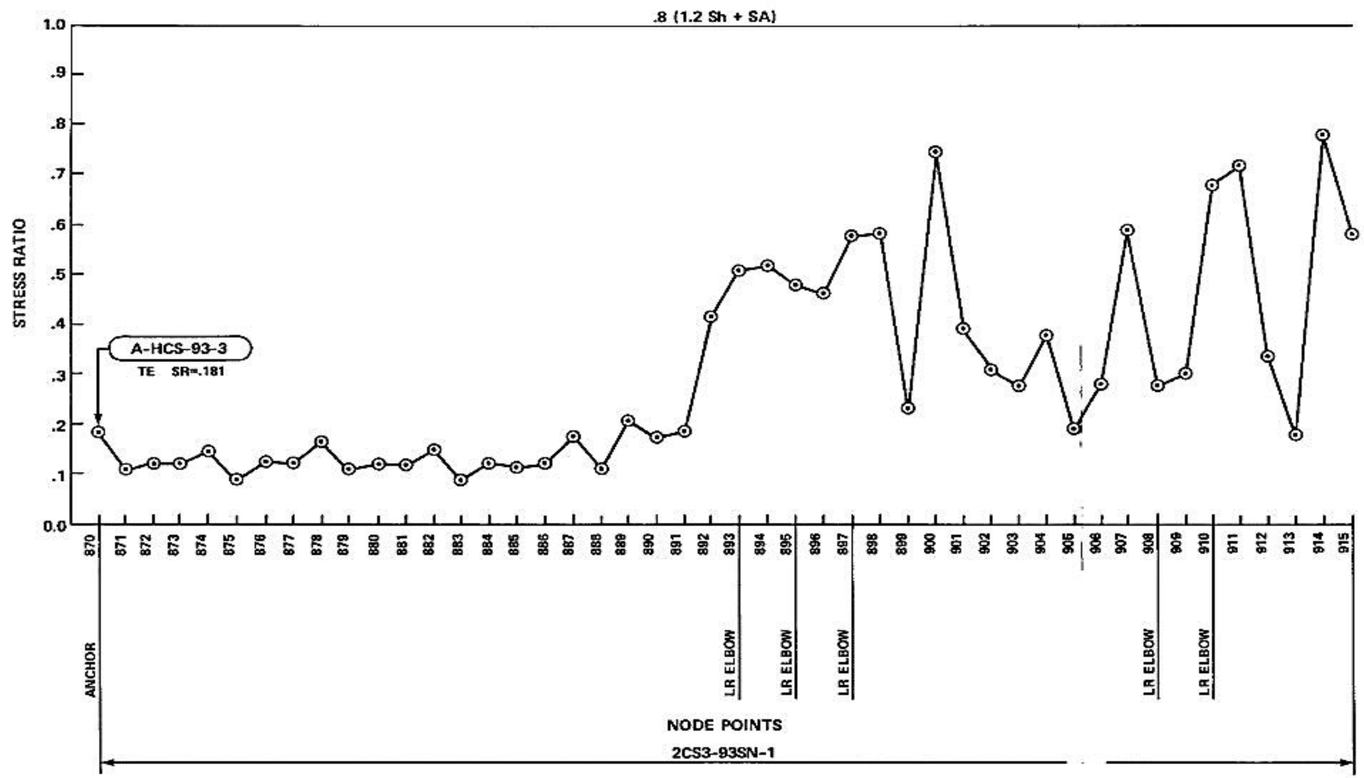
MODERATE-ENERGY SYSTEMS:

C = THROUGH-WALL LEAKAGE CRACK

REF. NO.:
CALC. NO: PRA-W-CS-143-2-1A-38, FIGURE 3.6A-26, FIGURE 3.6A-27

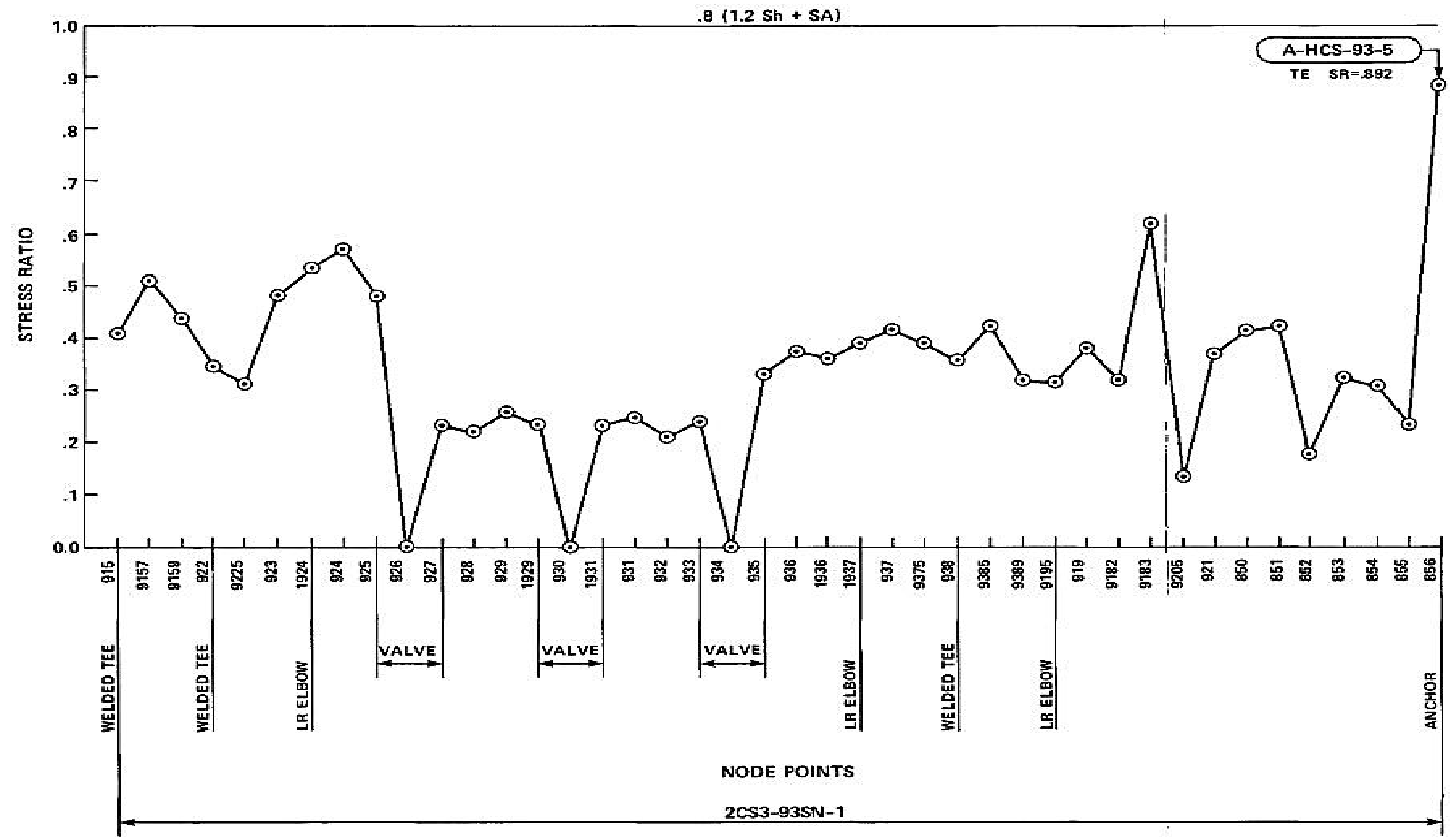
FIGURE 3.6A-26-PLOT-A

COMPOSITE PIPING – SHIELDED PIPE TUNNEL REACTOR AUXILIARY BUILDING – PLOT OF PIPE BREAK LOCATIONS
CHEMICAL & VOLUME CONTROL PIPING



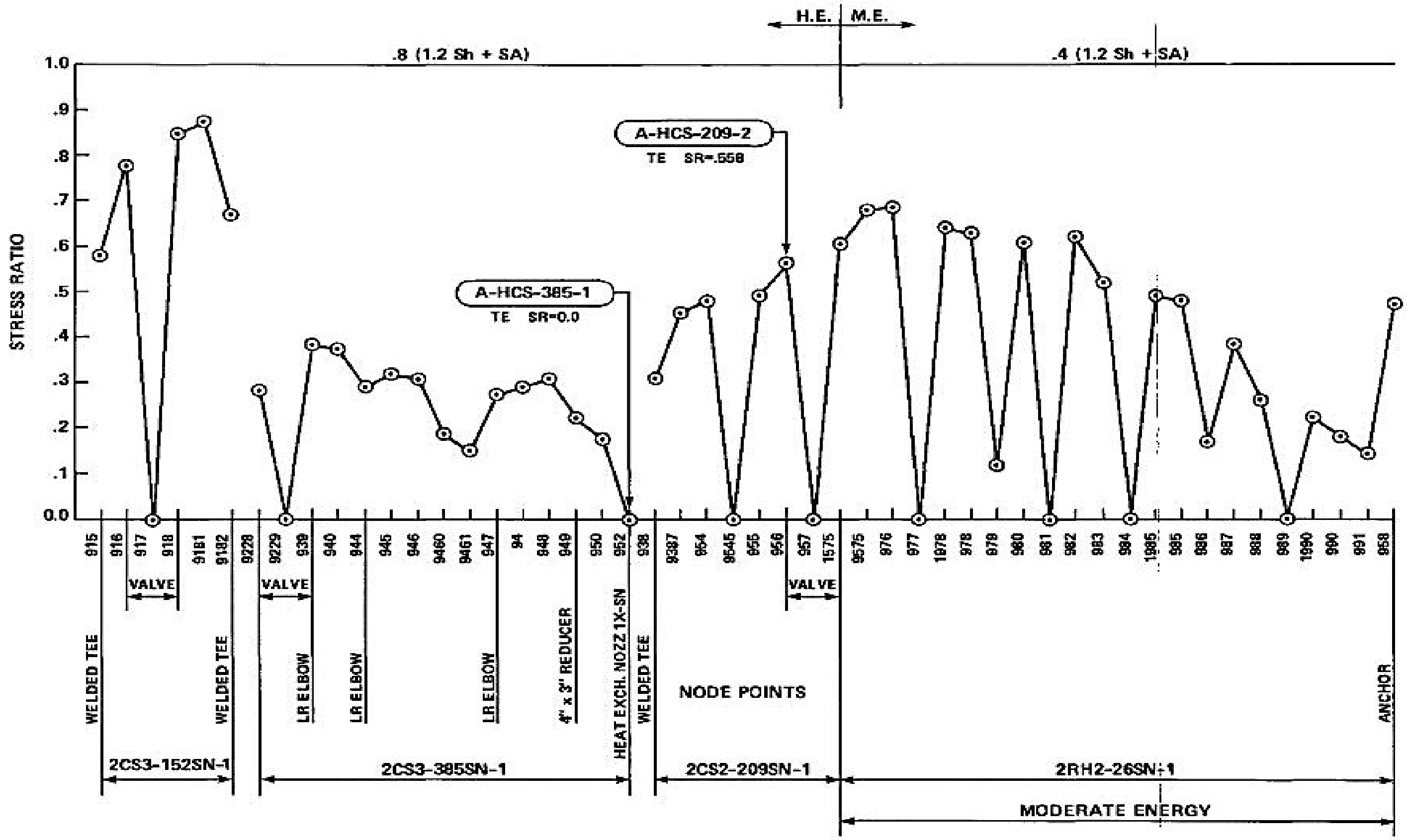
REF. NO.:
CALC. NO.: PRA-W-CS-143-2-1A-38, FIGURE 3.6A-26, FIGURE 3.6A-27, FIGURE 3.6A-26-CALC

FIGURE 3.6A-26-PLOT-A (Continued)



REF. NO.:
CALC. NO.: PRA-W-CS-143-2-1A-38, FIGURE 3.6A-26, FIGURE 3.6A-27, FIGURE 3.6A-28 CALC

FIGURE 3.6A-26-PLOT-A (Continued)



REF. NO.:
CALC. NO.: PRA-W-CS-143-2-1A-38, FIGURE 3.6A-26, FIGURE 3.6A-27, FIGURE 3.6A-26-CALC

FIGURE 3.6A-29

MATHEMATICAL MODEL MAIN STEAM AND FEEDWATER PIPING
INSIDE CONTAINMENT – LOOP 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-30

MATHEMATICAL MODEL MAIN STEAM AND FEEDWATER PIPING
INSIDE CONTAINMENT – LOOP 2

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-31

MATHEMATICAL MODEL MAIN STEAM AND FEEDWATER PIPING
INSIDE CONTAINMENT – LOOP 3

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-32

MATHEMATICAL MODEL MAIN STEAM PIPING OUTSIDE CONTAINMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-32 – PLOT-A

PLOT FOR MAIN STEAM PIPING OUTSIDE CONTAINMENT – LOOPS #1, #2 & #3

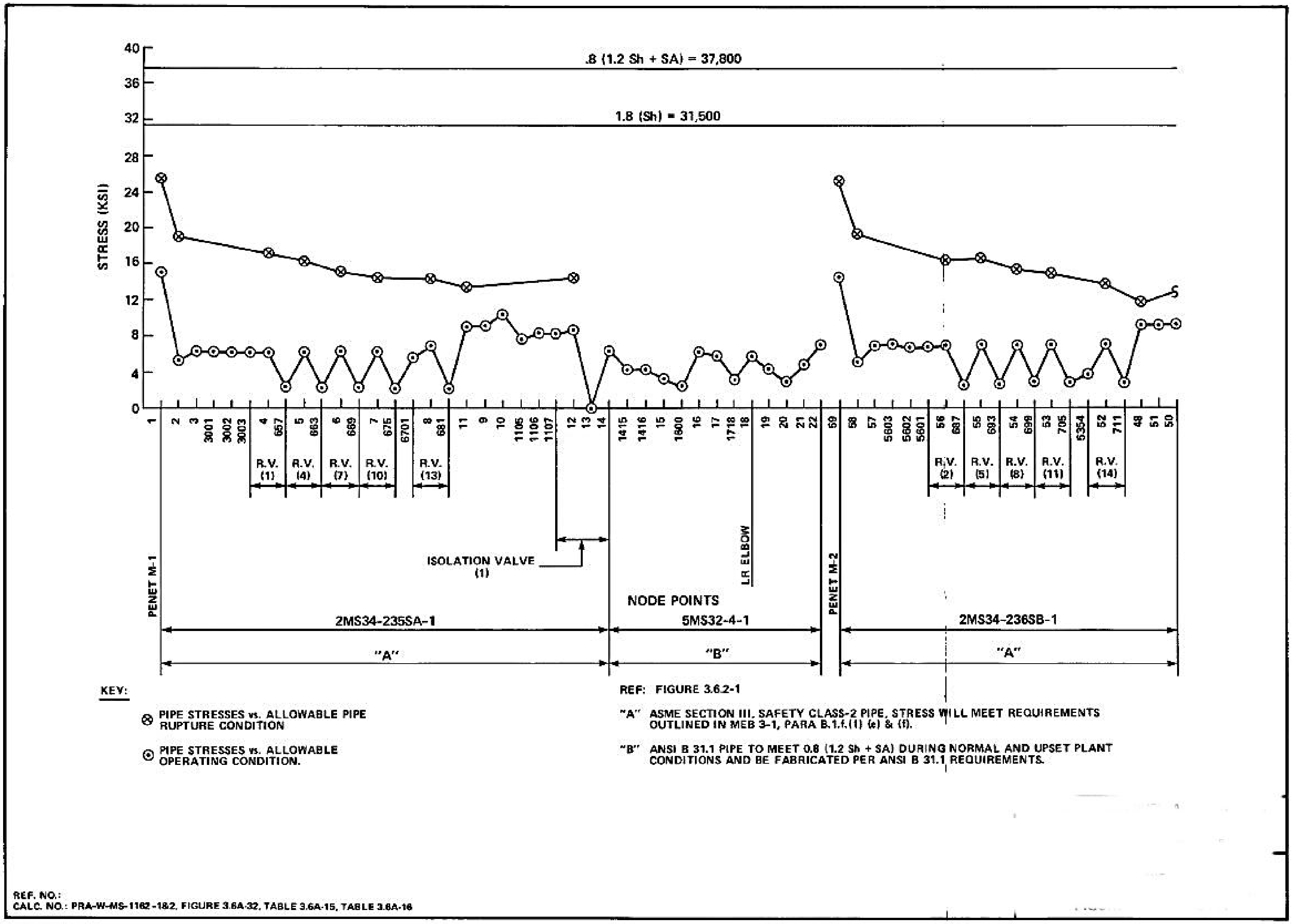


FIGURE 3.6A-32 – PLOT-A (Continued)

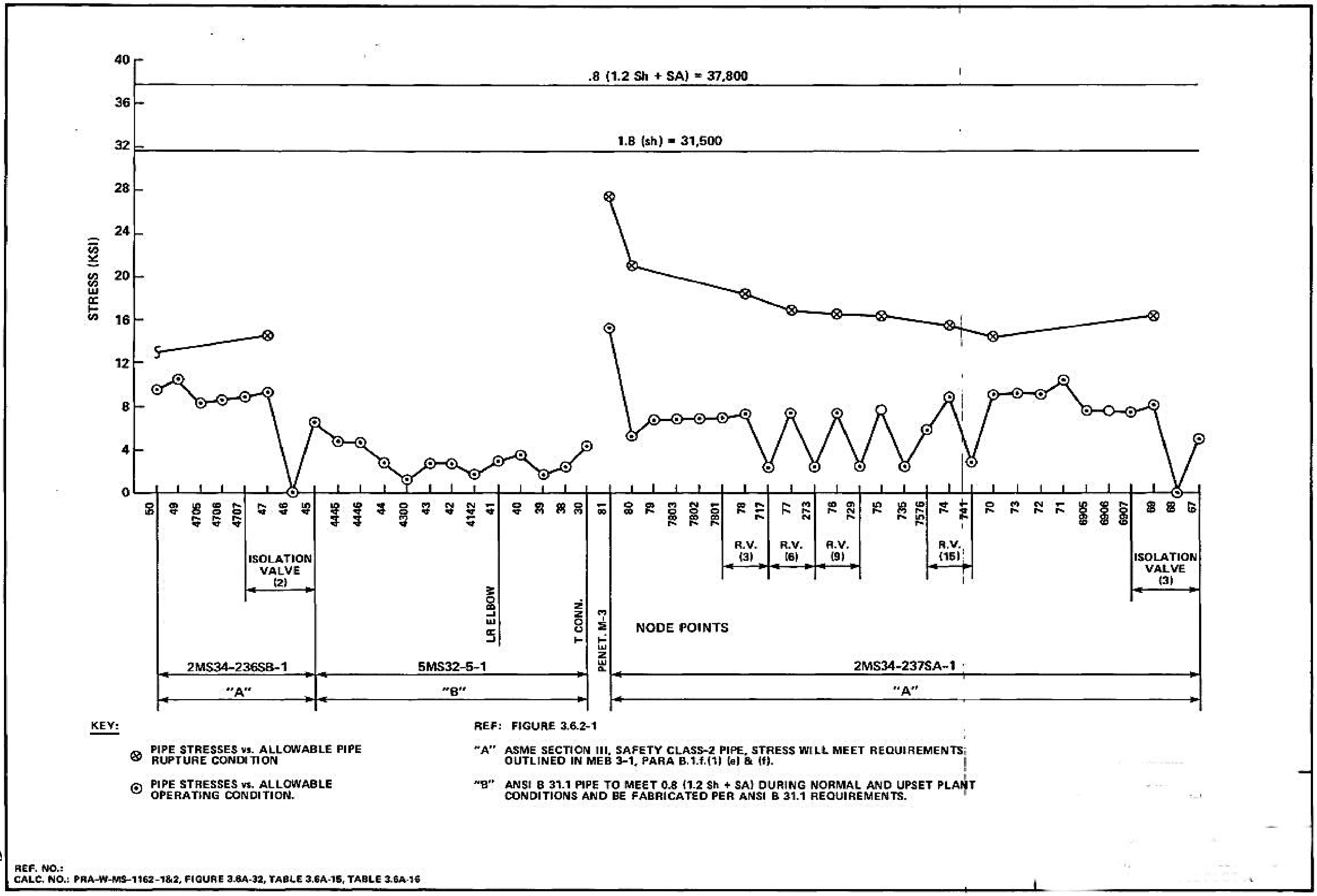


FIGURE 3.6A-32 – PLOT-A (Continued)

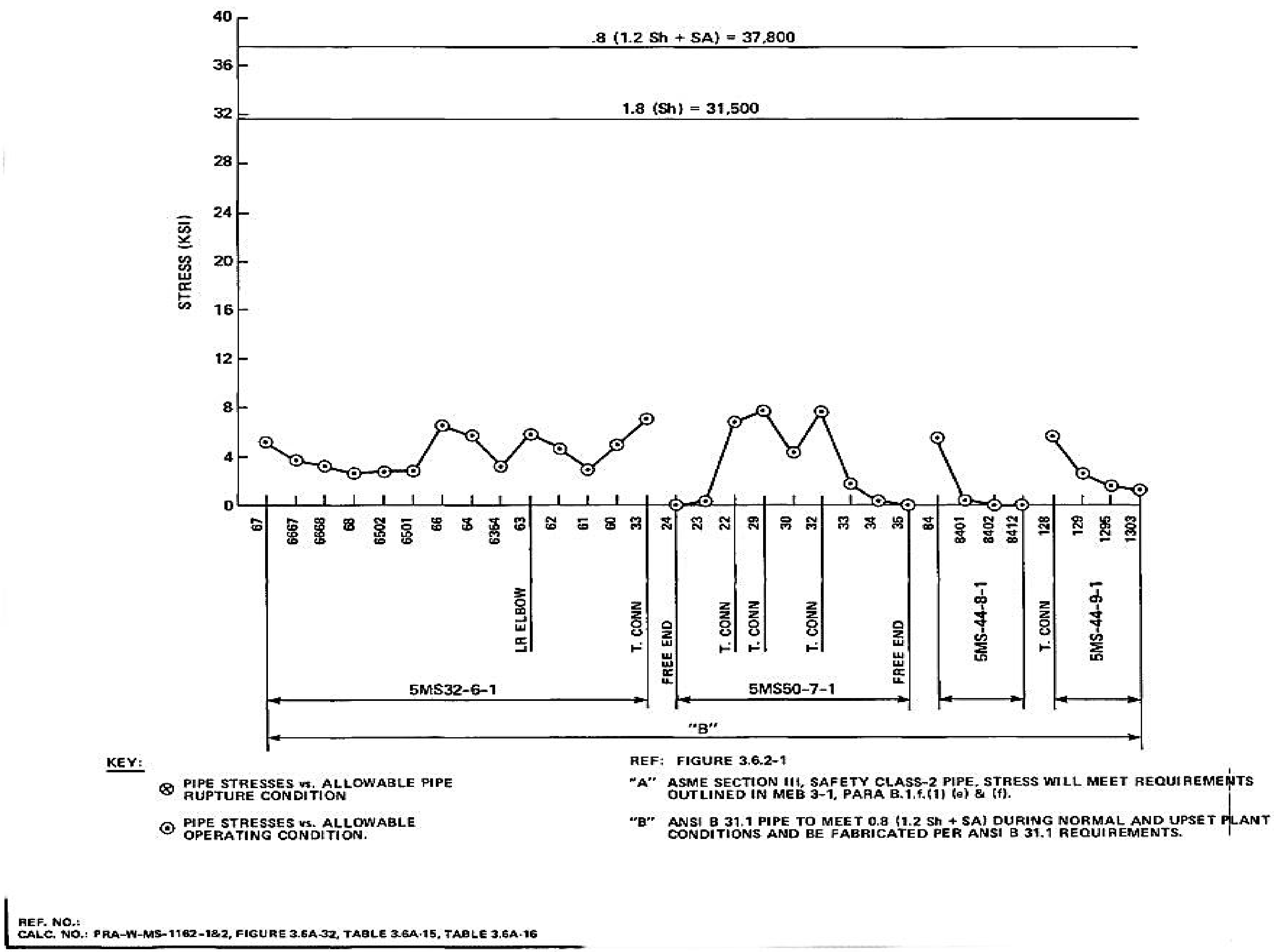


FIGURE 3.6A-32.1

MATHEMATICAL MODEL MAIN STEAM PIPING OUTSIDE CONTAINMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-32.2

MATHEMATICAL MODEL MAIN STEAM PIPING OUTSIDE CONTAINMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-32.2-CALC

PLOT FOR MATHEMATICAL MODEL MAIN STEAM PIPING OUTSIDE CONTAINMENT

SUMMARY OF CALCULATIONS

THE MAIN STEAM TO THE STEAM GENERATOR AUXILIARY FEED PUMP TURBINE DOES NOT OPERATE UNDER NORMAL PLANT CONDITIONS, THEREFORE, ALL ASSOCIATED PIPING FROM VALVES 2MS-V-8SB-1 AND 2MS-V-9SA-1 TO THE AUXILIARY FEED PUMP NOZZLE STEAM INLET ARE CONSIDERED MODERATE ENERGY PIPING AND ONLY POSTULATED CRACKS WILL OCCURR IN FLOODING ANALYSIS.

SINCE PIPING UPSTREAM OF VALVES 2MS-V-8SB-1 AND 2MS-V-9SA-1 IS CLASSIFIED AS "BREAK EXCLUSION REGION" THUS NO BREAKS ARE POSTULATED AND NO PIPE WHIP RESTRAINTS ARE DESIGNED WITHIN THE SUBJECT PIPING.

REF. NO.:
CALC. NO.: PRA-W-MS-6713-21, FIGURE 3.6A-32.2, FIGURE 3.6A-32

FIGURE 3.6A-32.2 – PLOT-A

PLOT FOR MATHEMATICAL MODEL MAIN STEAM PIPING OUTSIDE CONTAINMENT

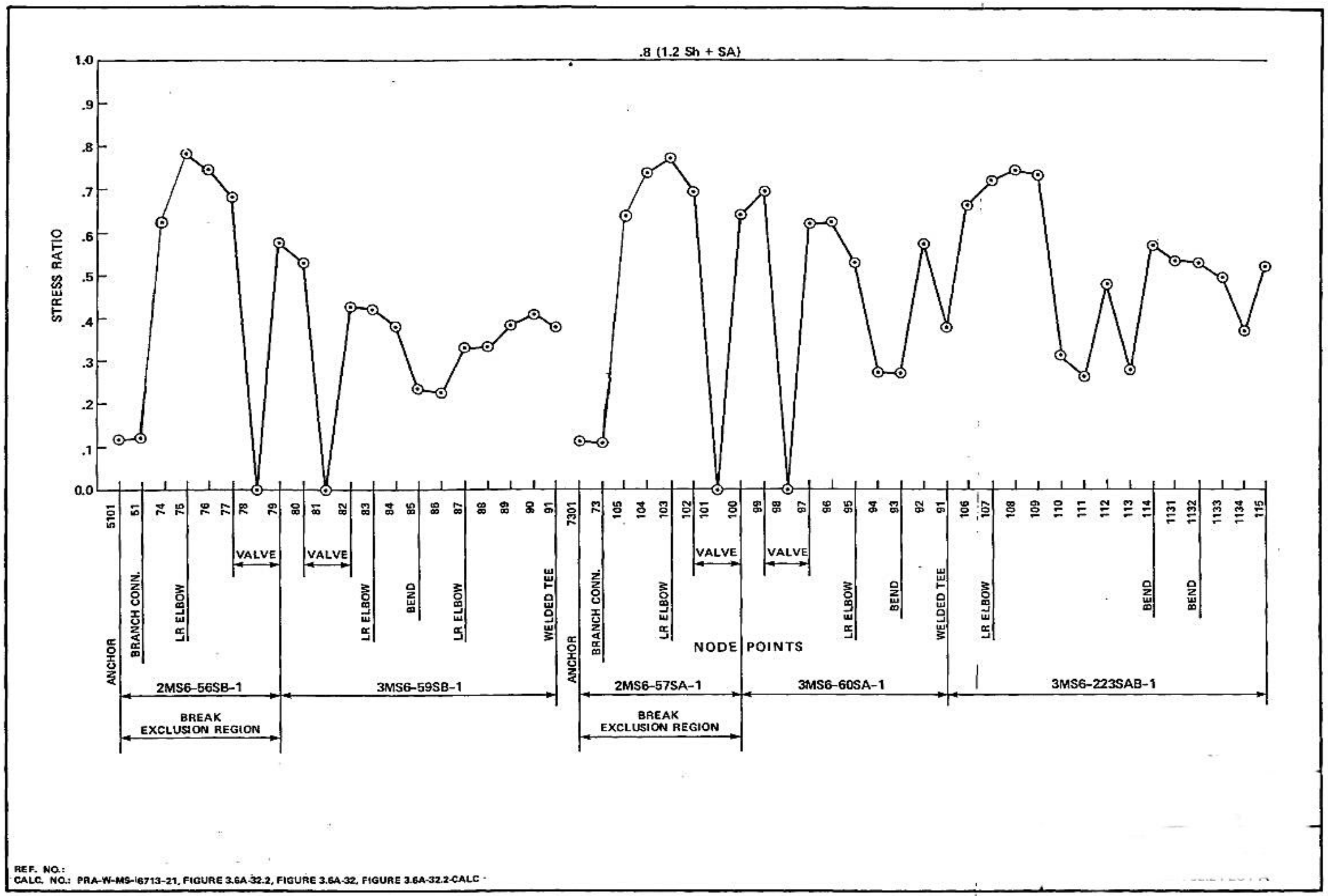


FIGURE 3.6A-33.1

MATHEMATICAL MODEL FEED WATER PIPING OUTSIDE CONTAINMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.6A-33

MATHEMATICAL MODEL FEED WATER PIPING OUTSIDE CONTAINMENT

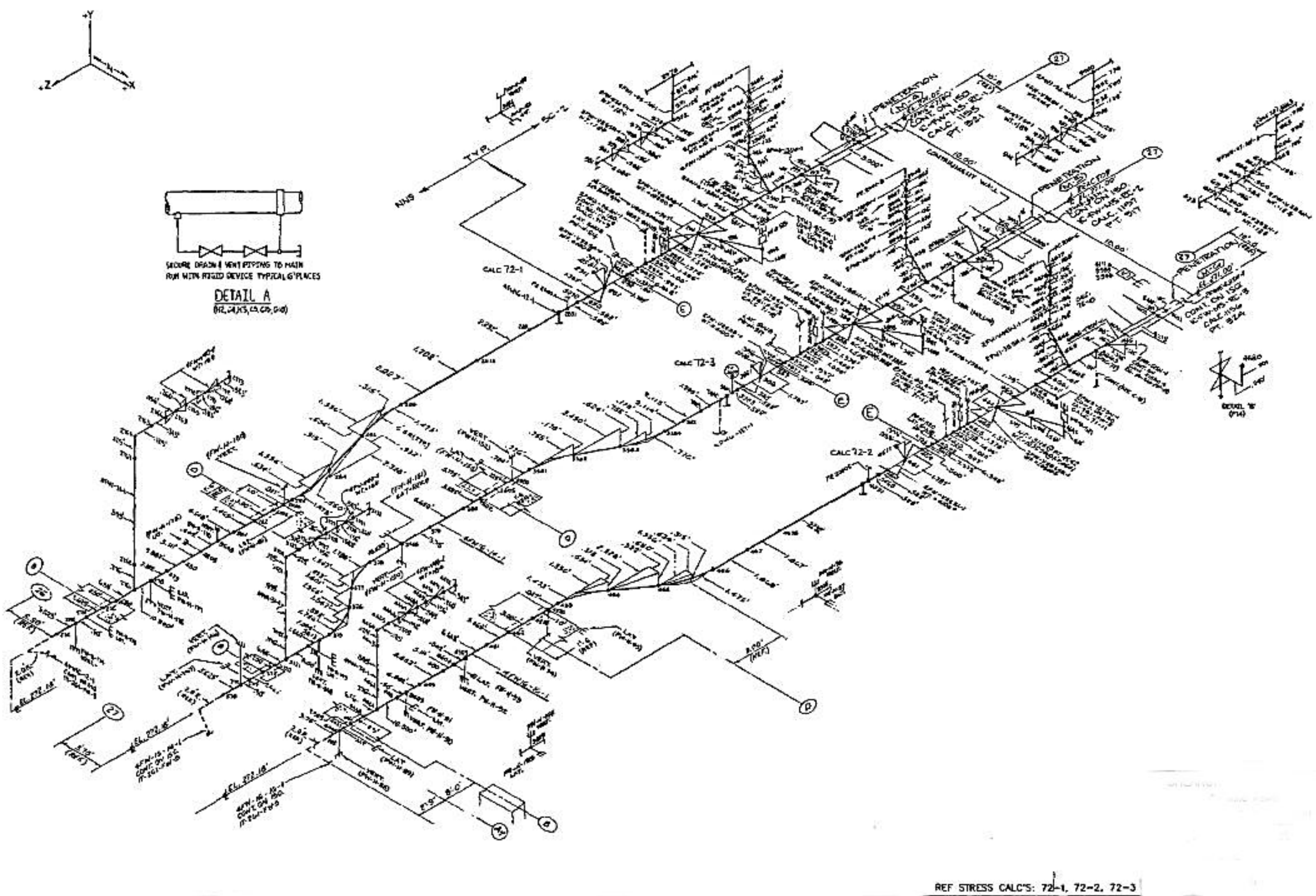


FIGURE 3.6A-34

STEAM TUNNEL PRESSURE AND TEMPERATURE NODALIZATION MODEL FOR MAIN FEEDWATER LINE BREAK

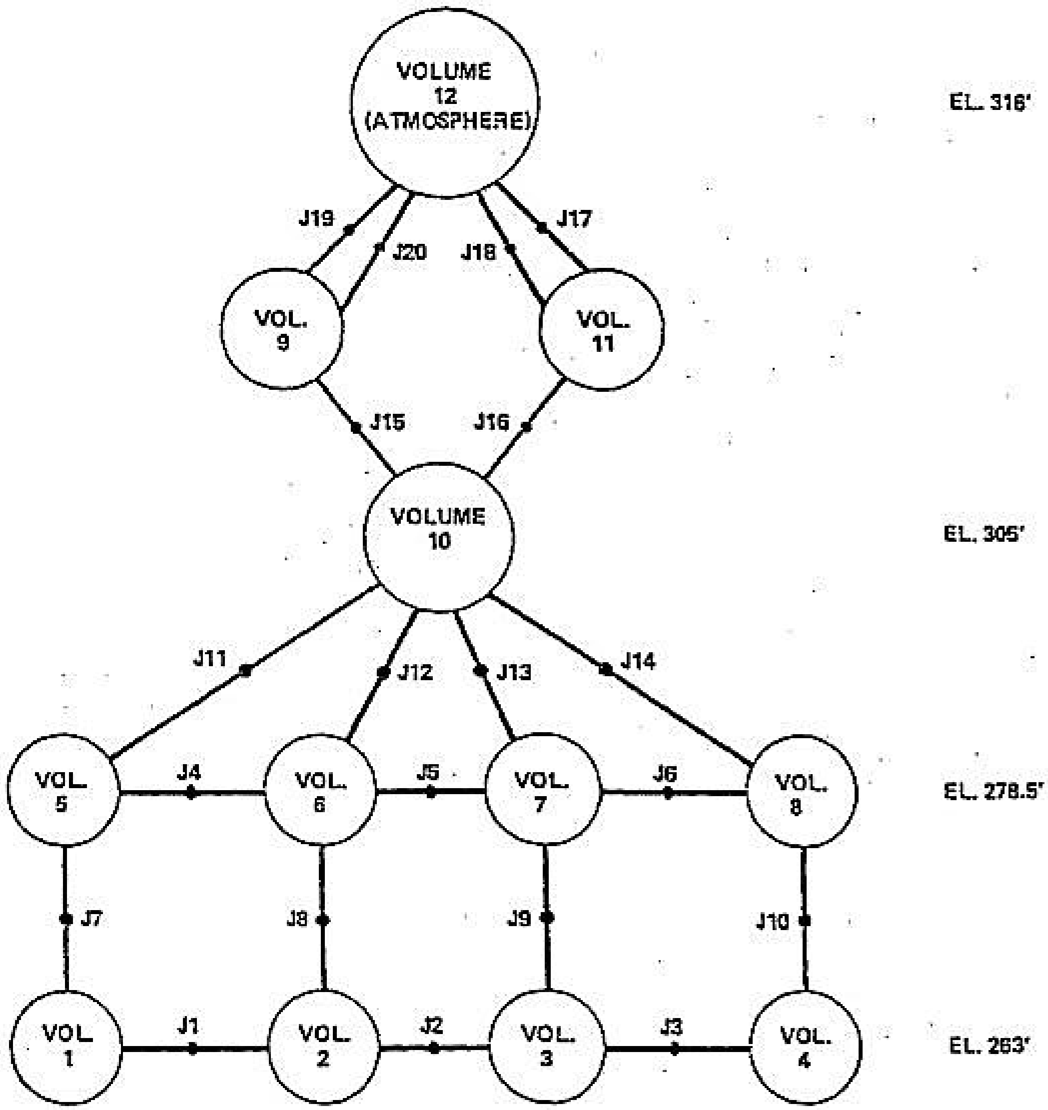


FIGURE 3.6A-34a

STEAM TUNNEL PRESSURE NODALIZATION MODEL FOR MAIN STEAM LINE BREAK

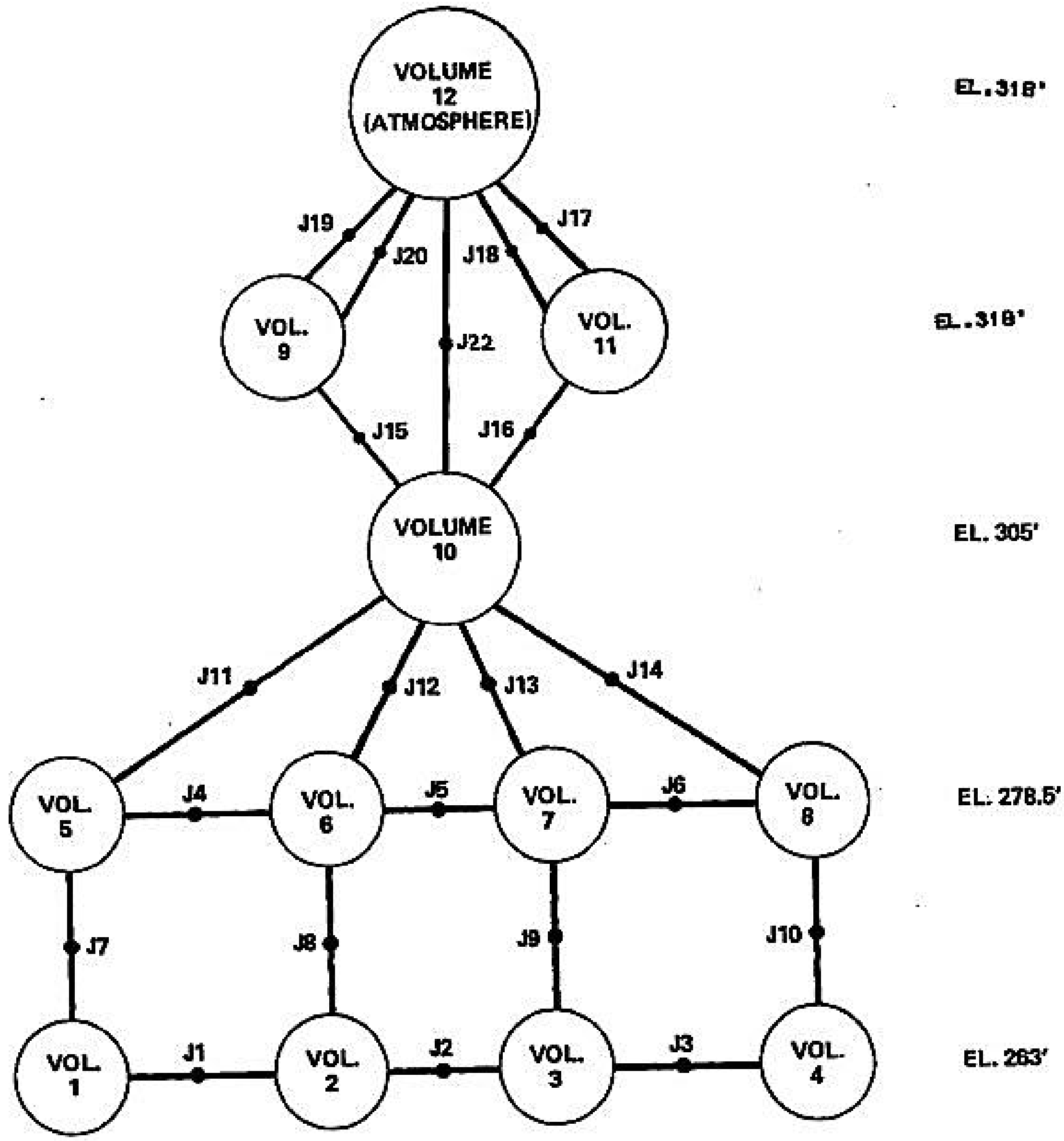


FIGURE 3.6A-34b

STEAM TUNNEL TEMPERATURE NODALIZATION MODEL FOR MAIN STEAM LINE BREAK

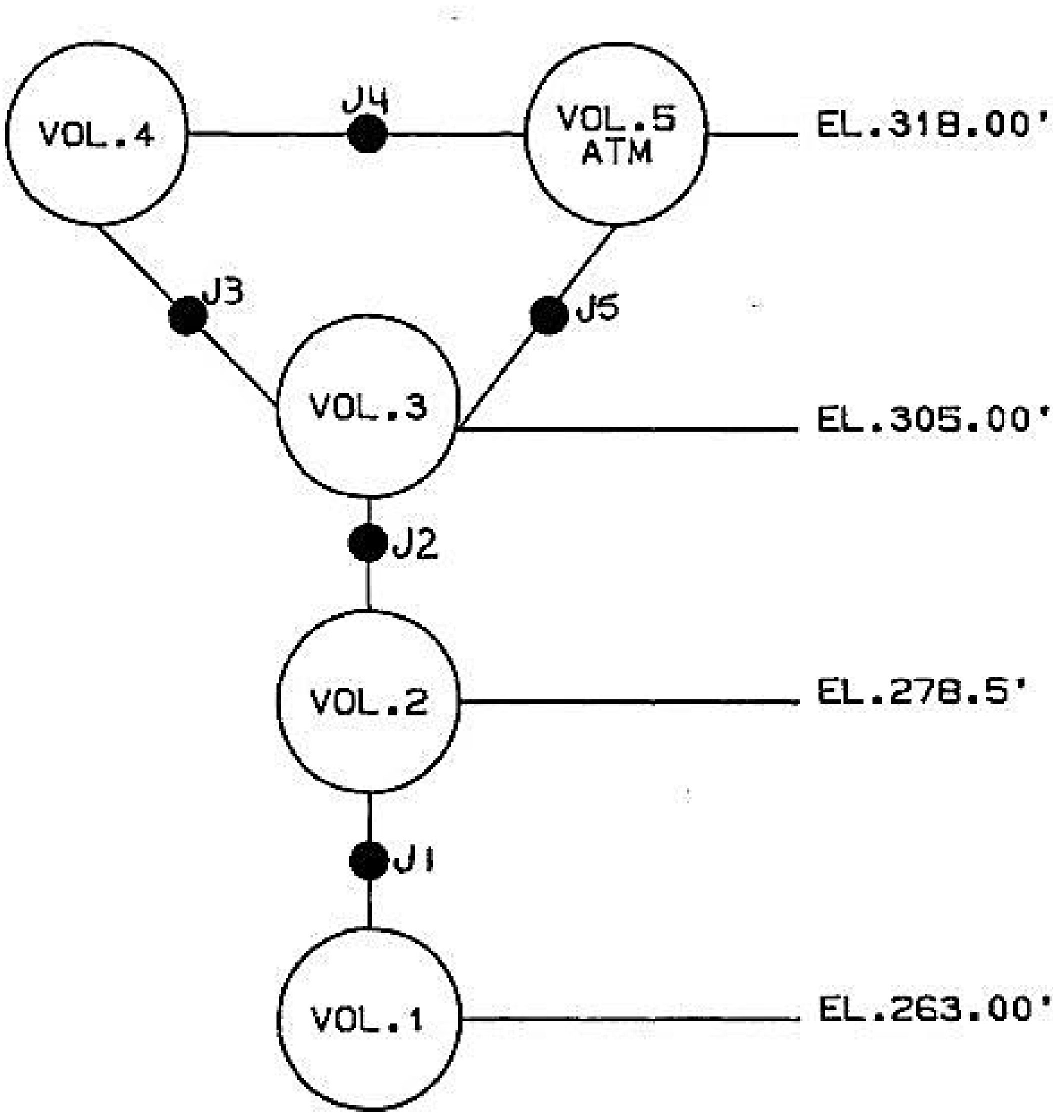


FIGURE 3.6A-35

MSLB – PRESSURE VS TIME MAIN STEAM TUNNEL (1.4 FT^2 MSLB AT 102% POWER)

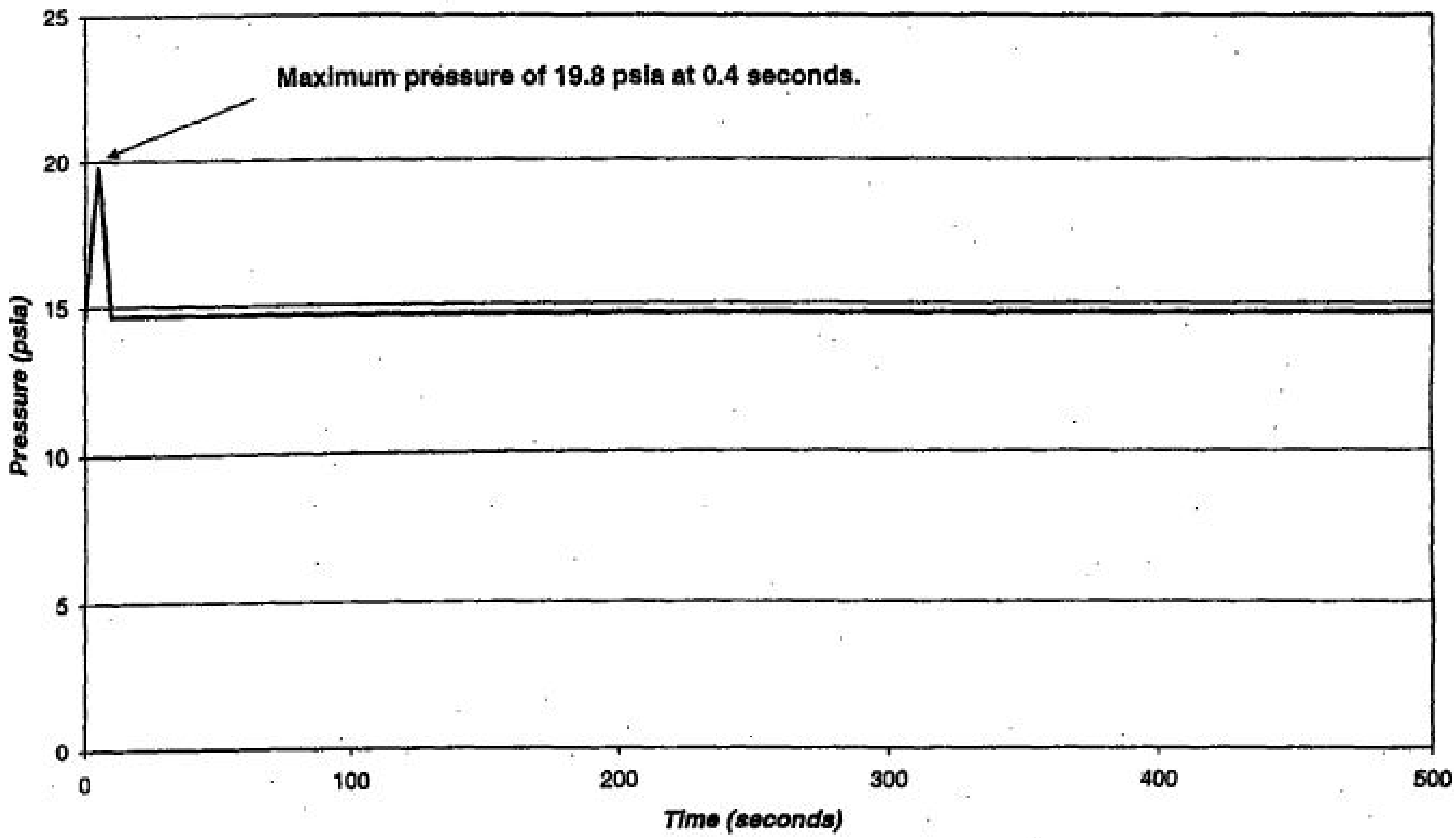


FIGURE 3.6A-40

MSLB – PRESSURE VS TIME MAIN STEAM TUNNEL (1.4 FT^2 MSLB AT 102% POWER)

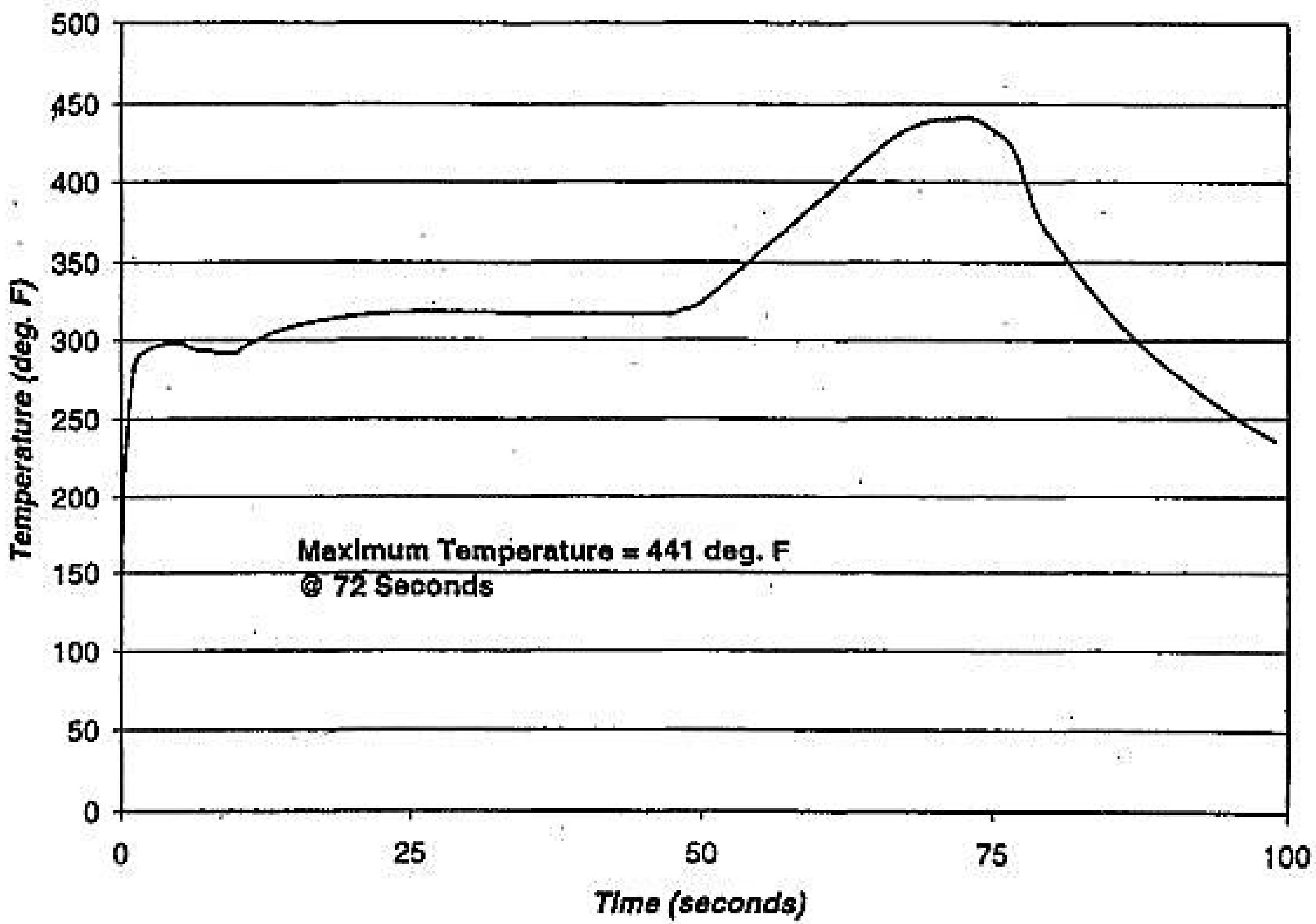


FIGURE 3.6A-41

FWLB – PRESSURE VS TIME

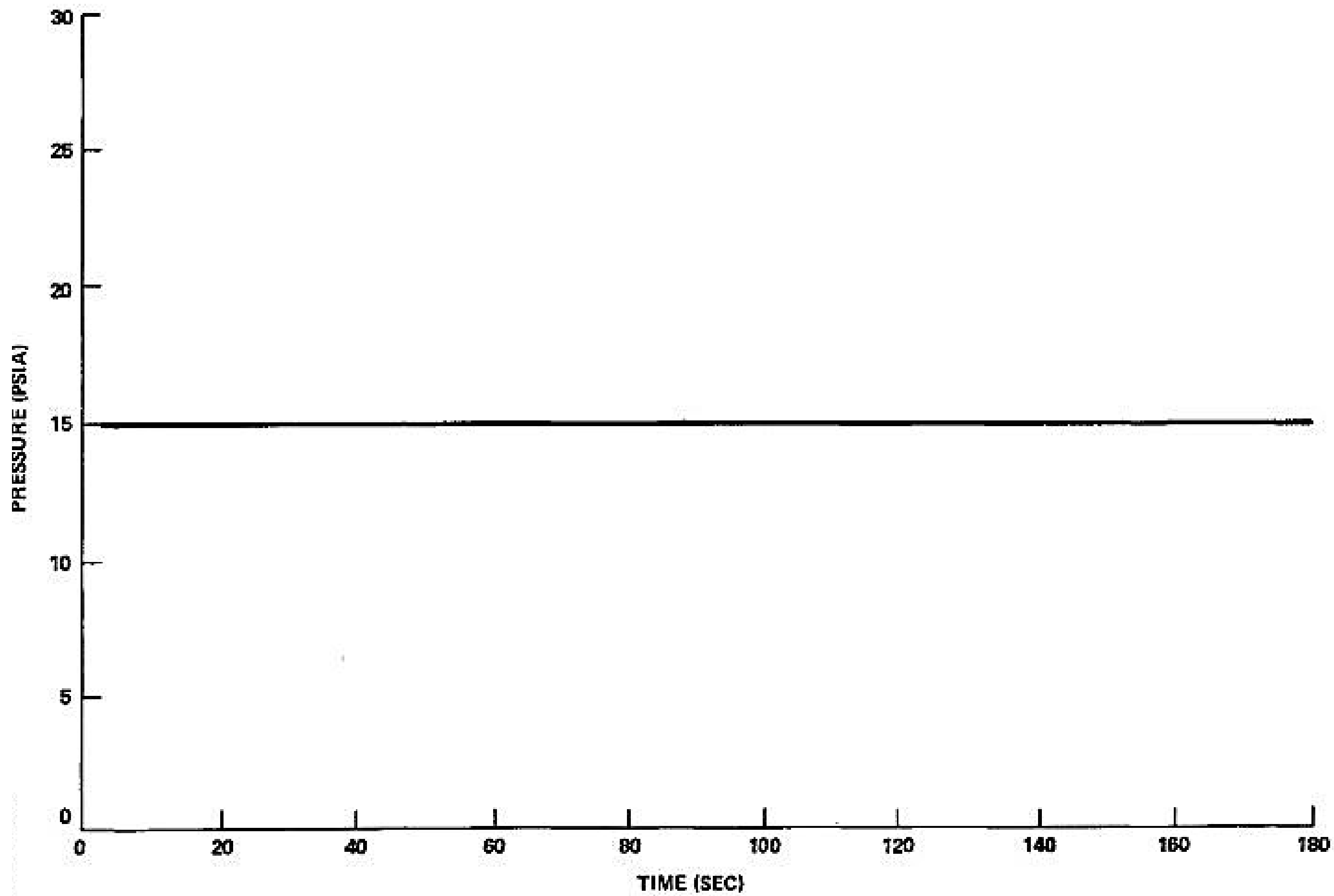


FIGURE 3.6A-42

FWLB – TEMPERATURE VS TIME

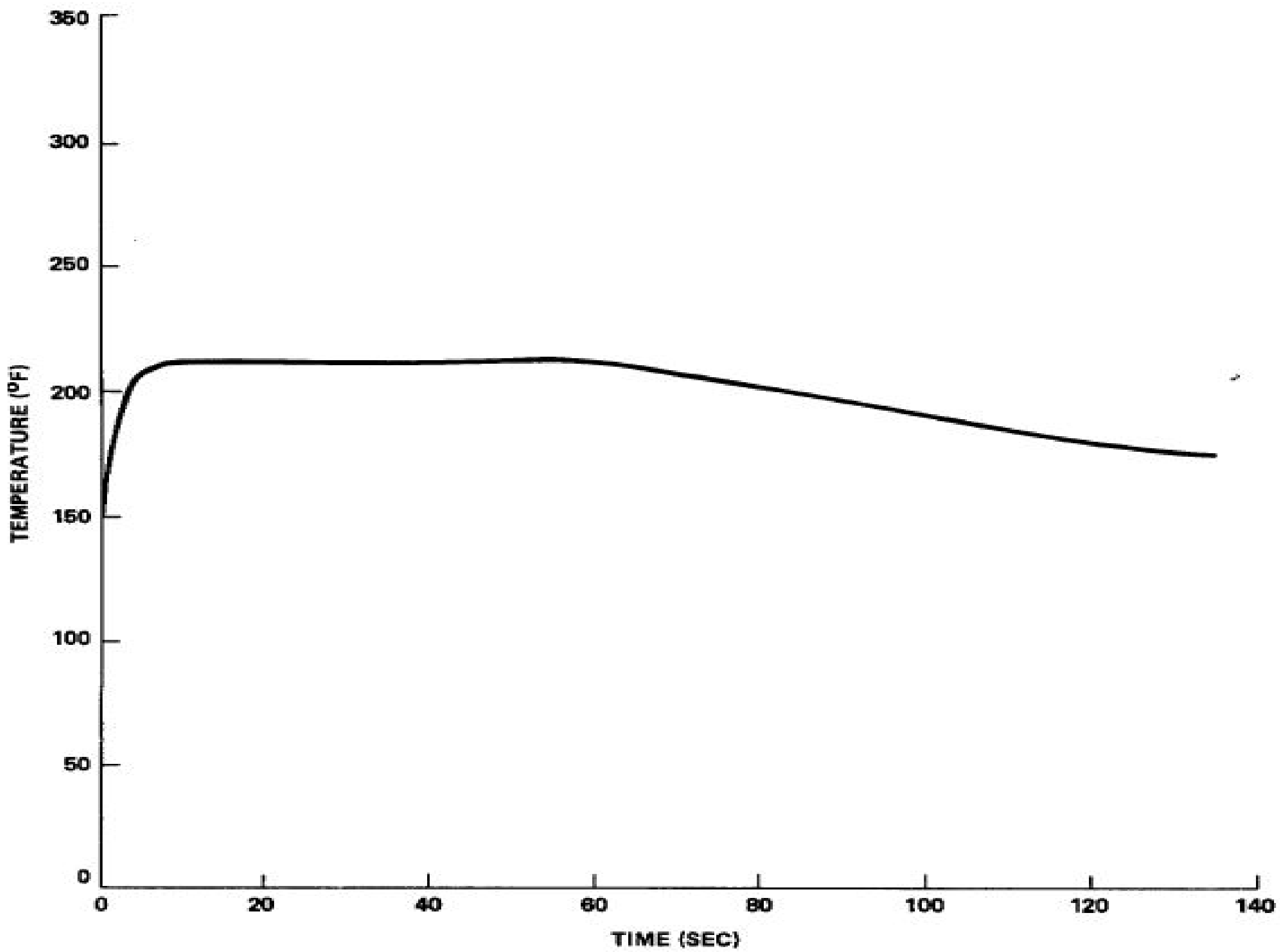
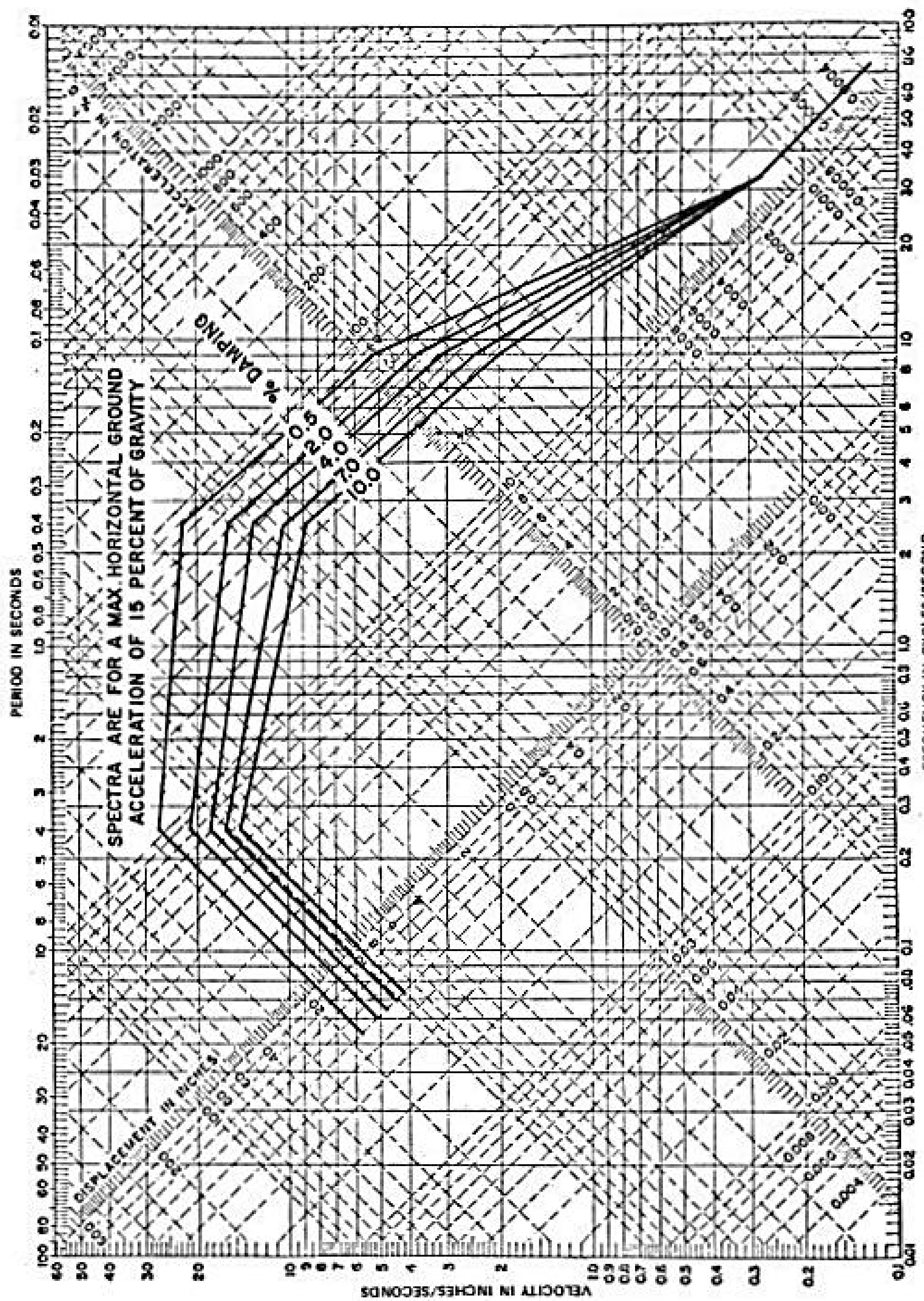


FIGURE 3.7.1-1

HORIZONTAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE

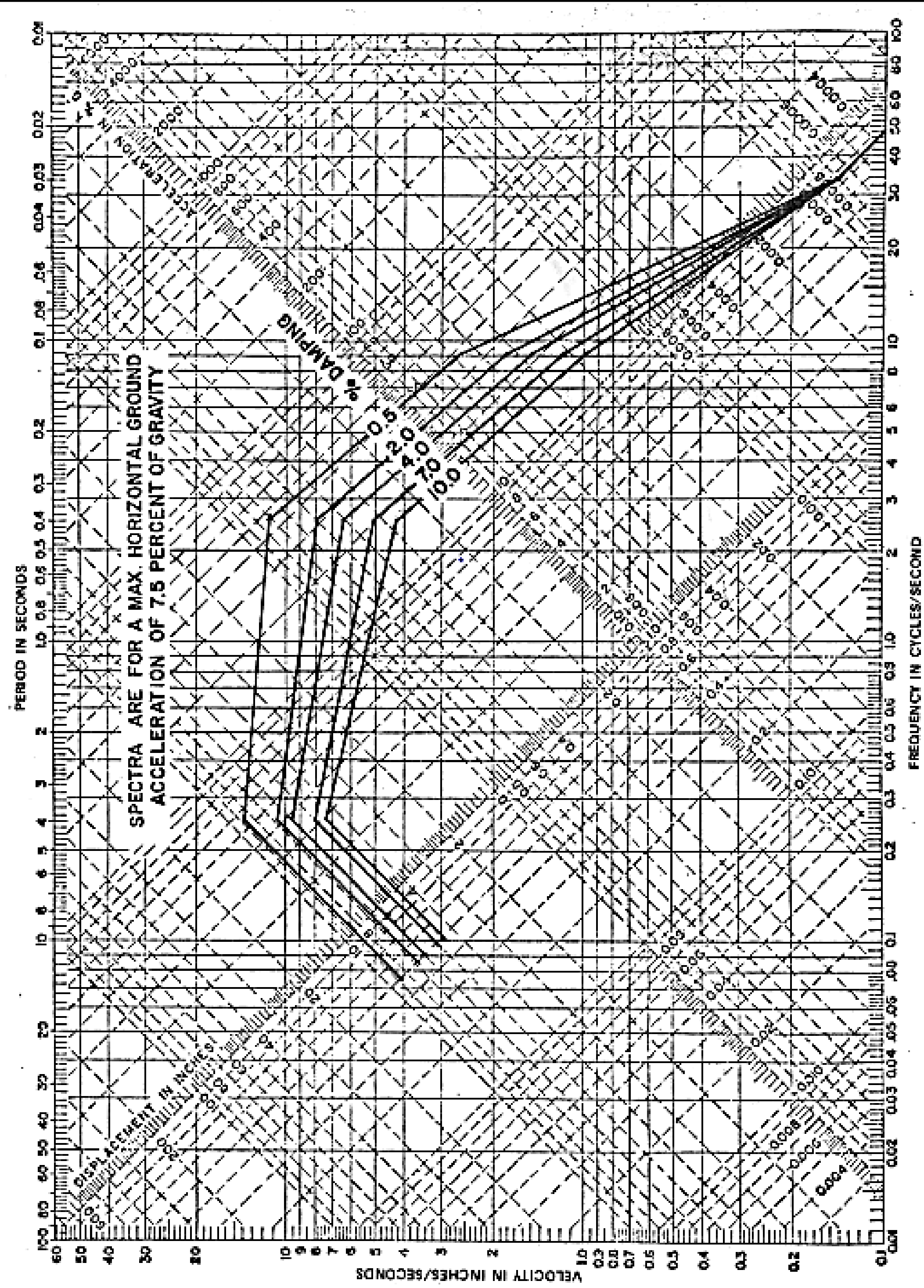


HORIZONTAL RESPONSE SPECTRA

DESIGN BASIS EARTHQUAKE

FIGURE 3.7.1-2

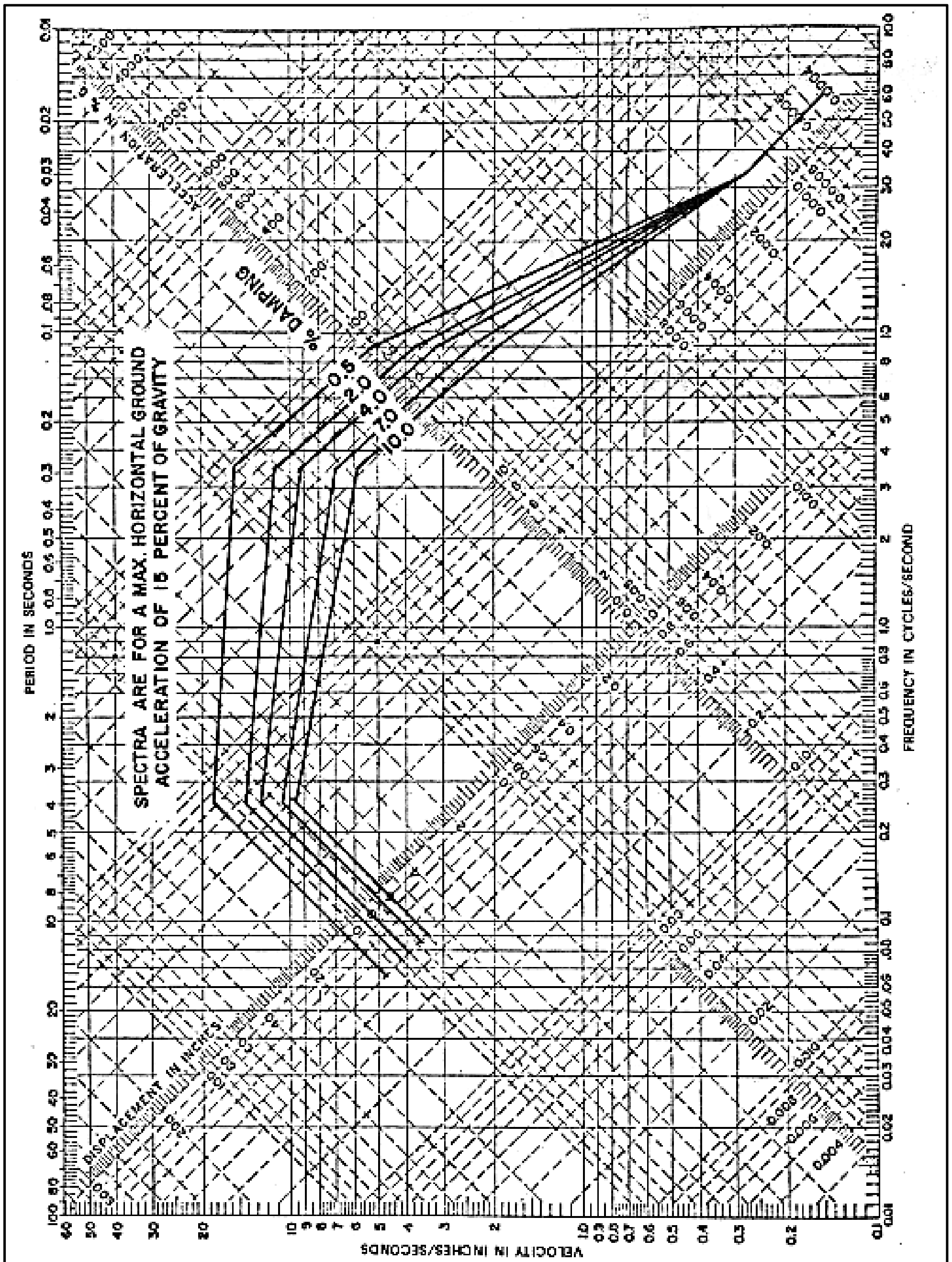
HORIZONTAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE



HORIZONTAL RESPONSE SPECTRA
OPERATING BASIS EARTHQUAKE

FIGURE 3.7.1-3

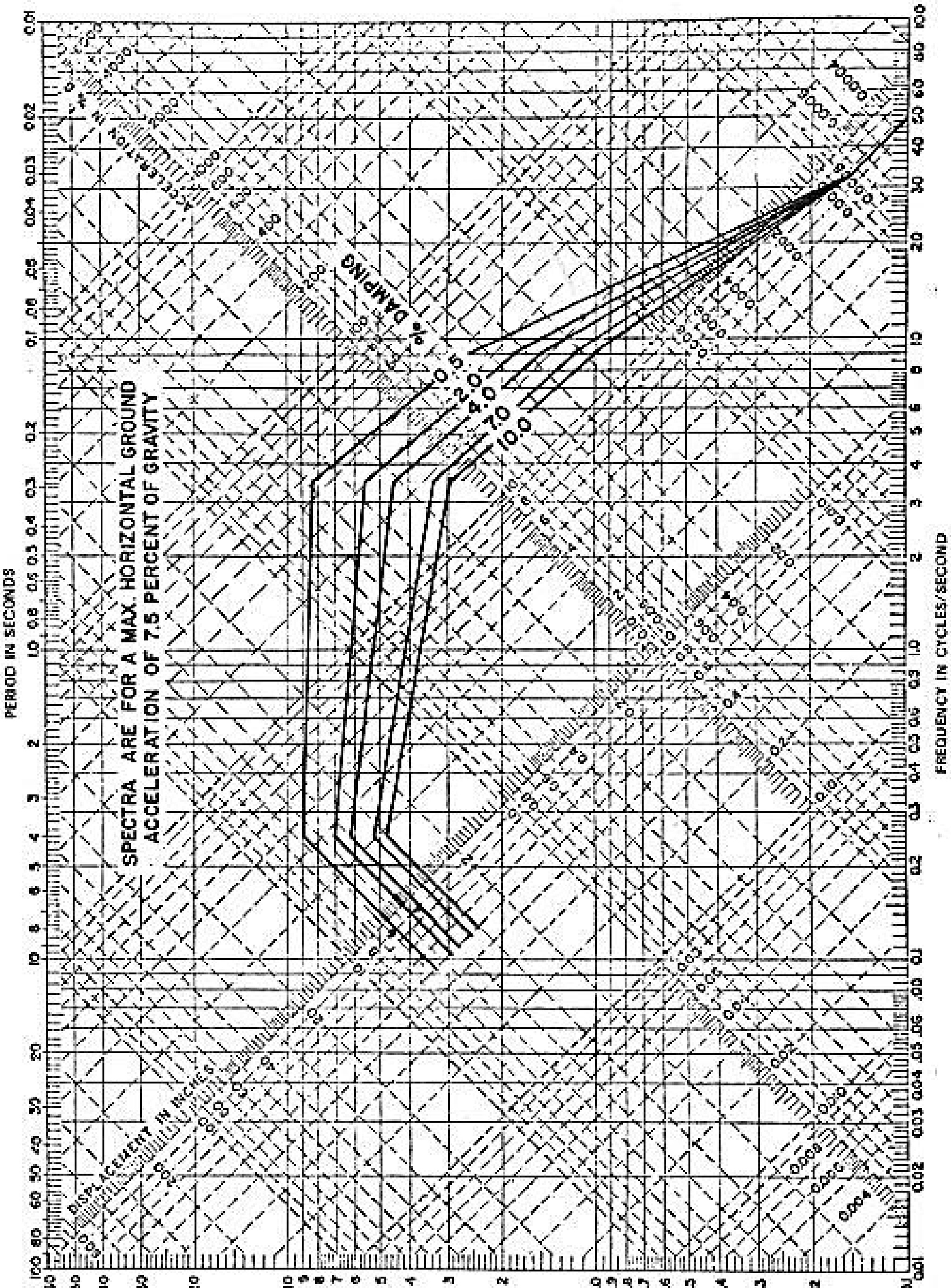
VERTICAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE



VERTICLE RESPONSE SPECTRA
DESIGN BASIS EARTHQUAKE

FIGURE 3.7.1-4

VERTICAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE



VERTICLE RESPONSE SPECTRA
OPERATING BASIS EARTHQUAKE

FIGURE 3.7.1-5

HORIZONTAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE DAMS AND DIKES

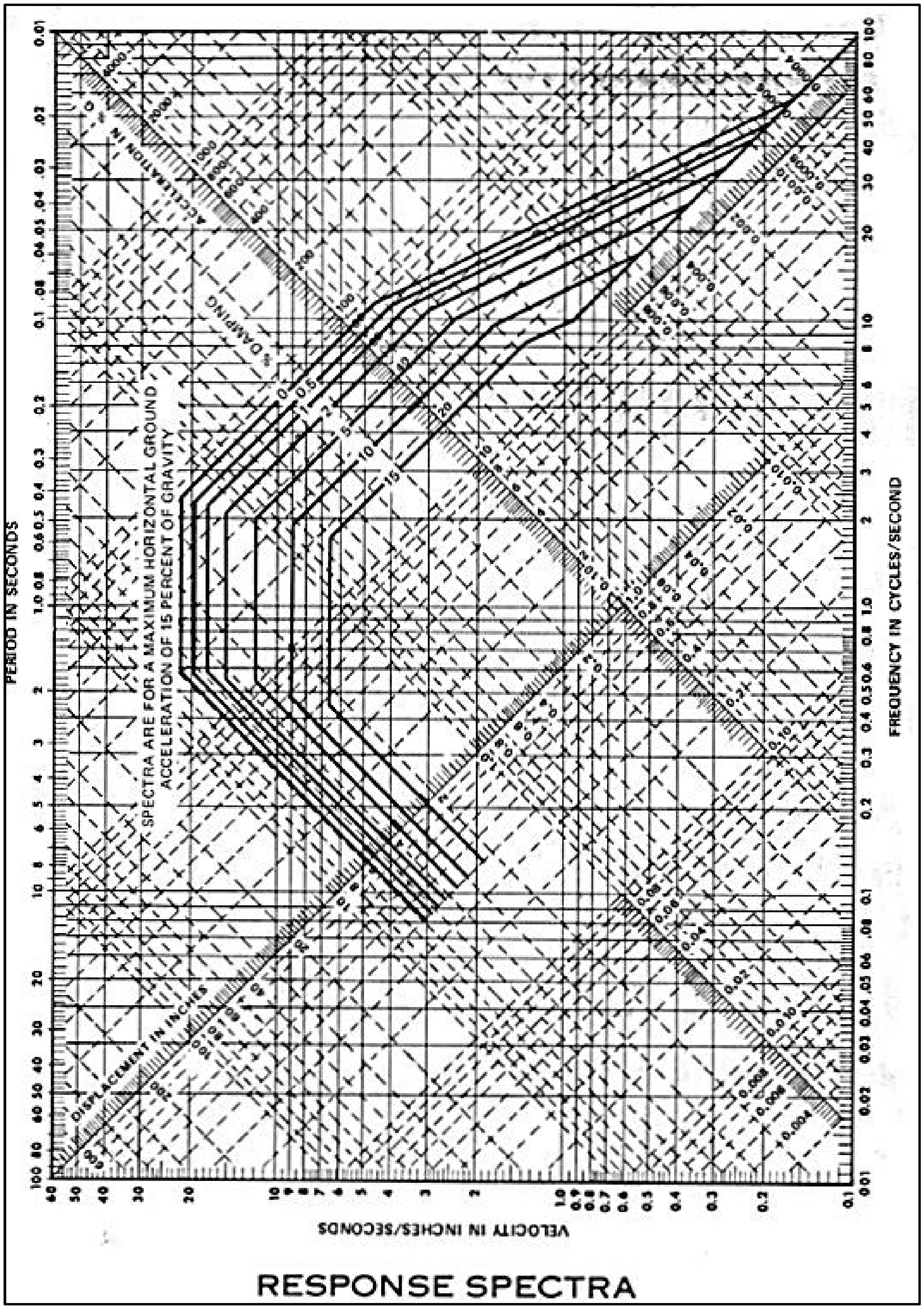
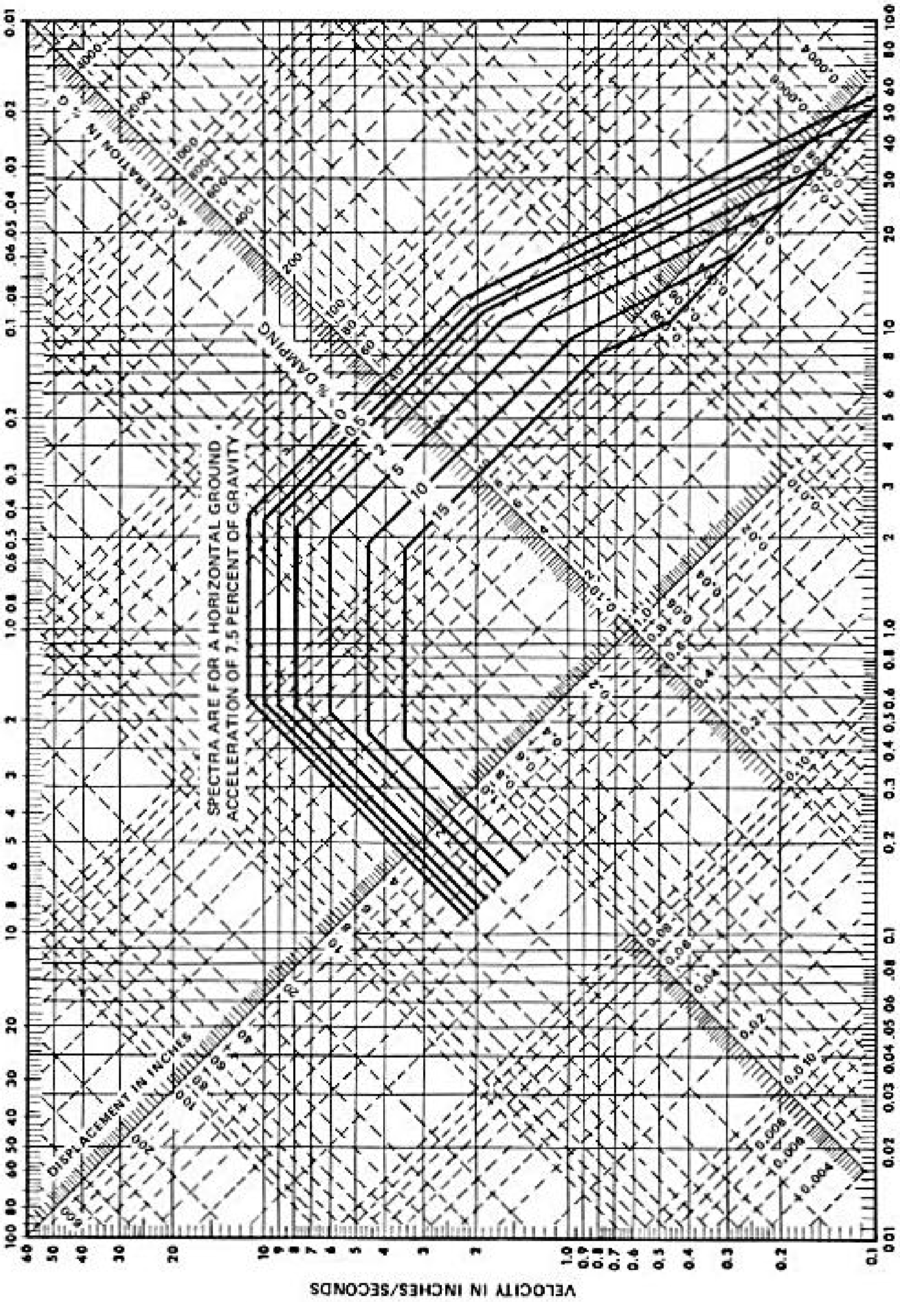


FIGURE 3.7.1-6

HORIZONTAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE DAMS AND DIKES



RESPONSE SPECTRA

FIGURE 3.7.1-7

VERTICAL DESIGN RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE DAMS AND DIKES

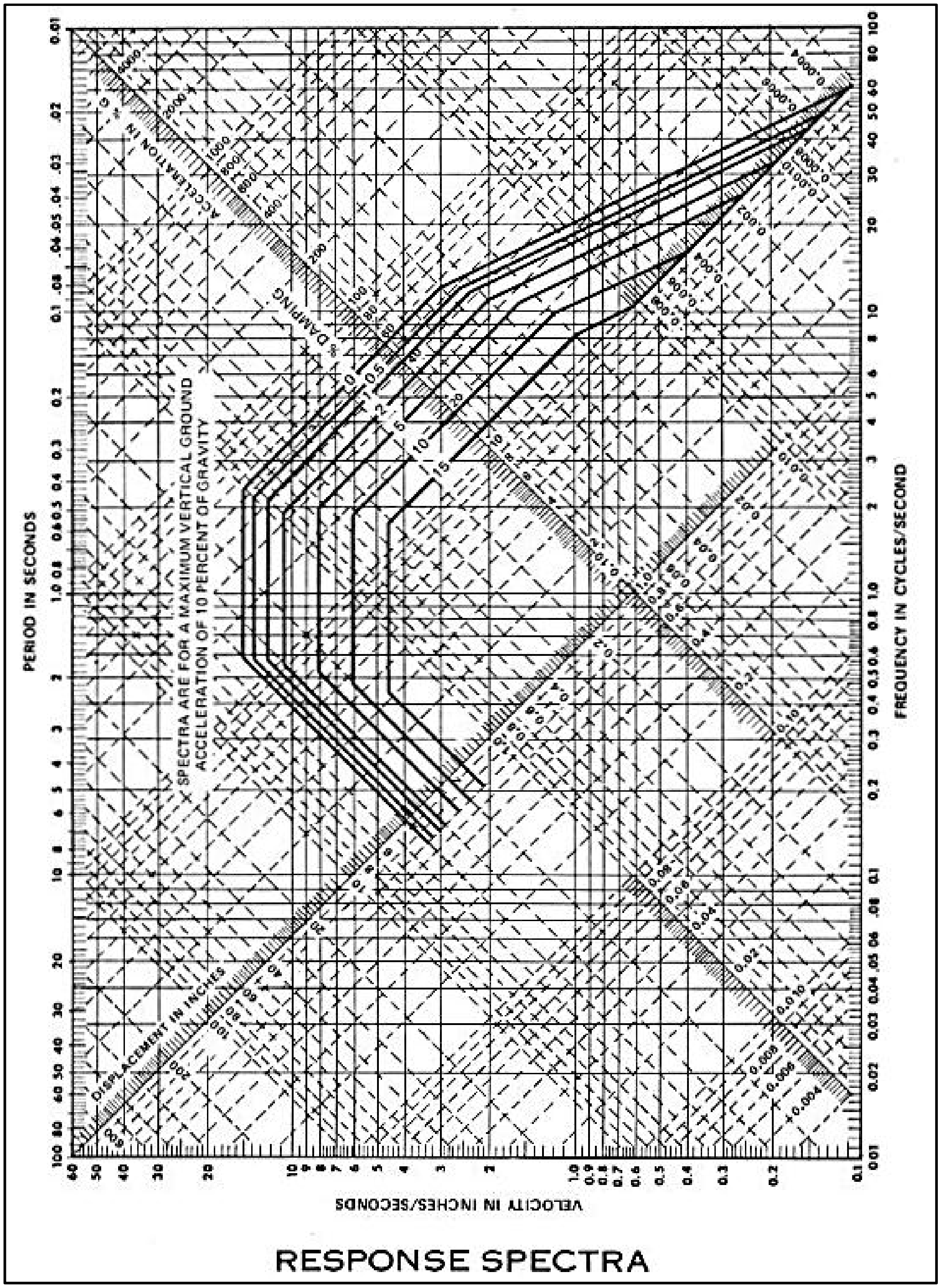


FIGURE 3.7.1-8

VERTICAL DESIGN RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE DAMS AND DIKES

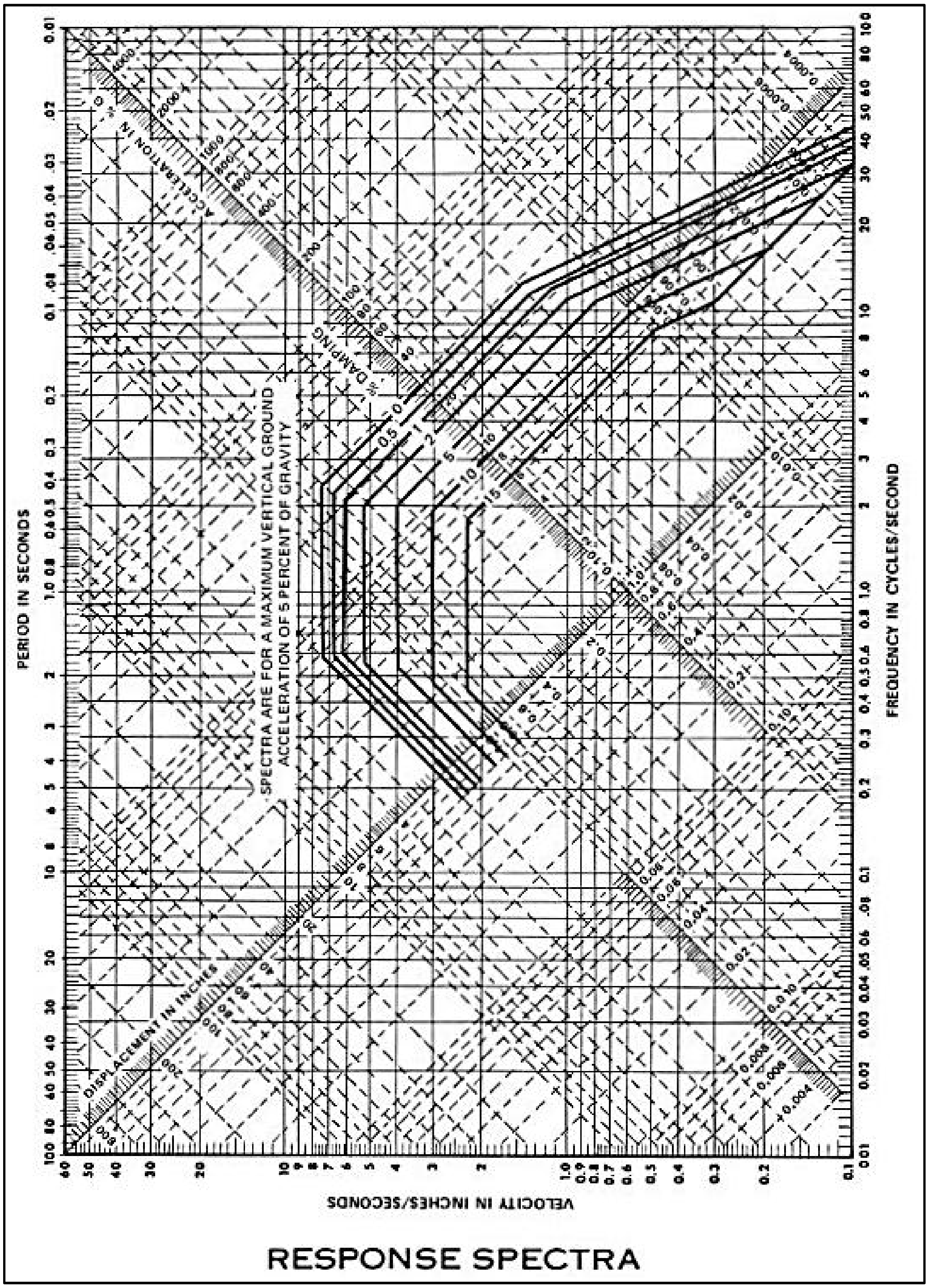


FIGURE 3.7.1-9

HORIZONTAL SIMULATED EARTHQUAKE ACCELEROGRAM – MAX. GROUND ACCELERATION 0.15 G OF TIME 10.00 SECONDS

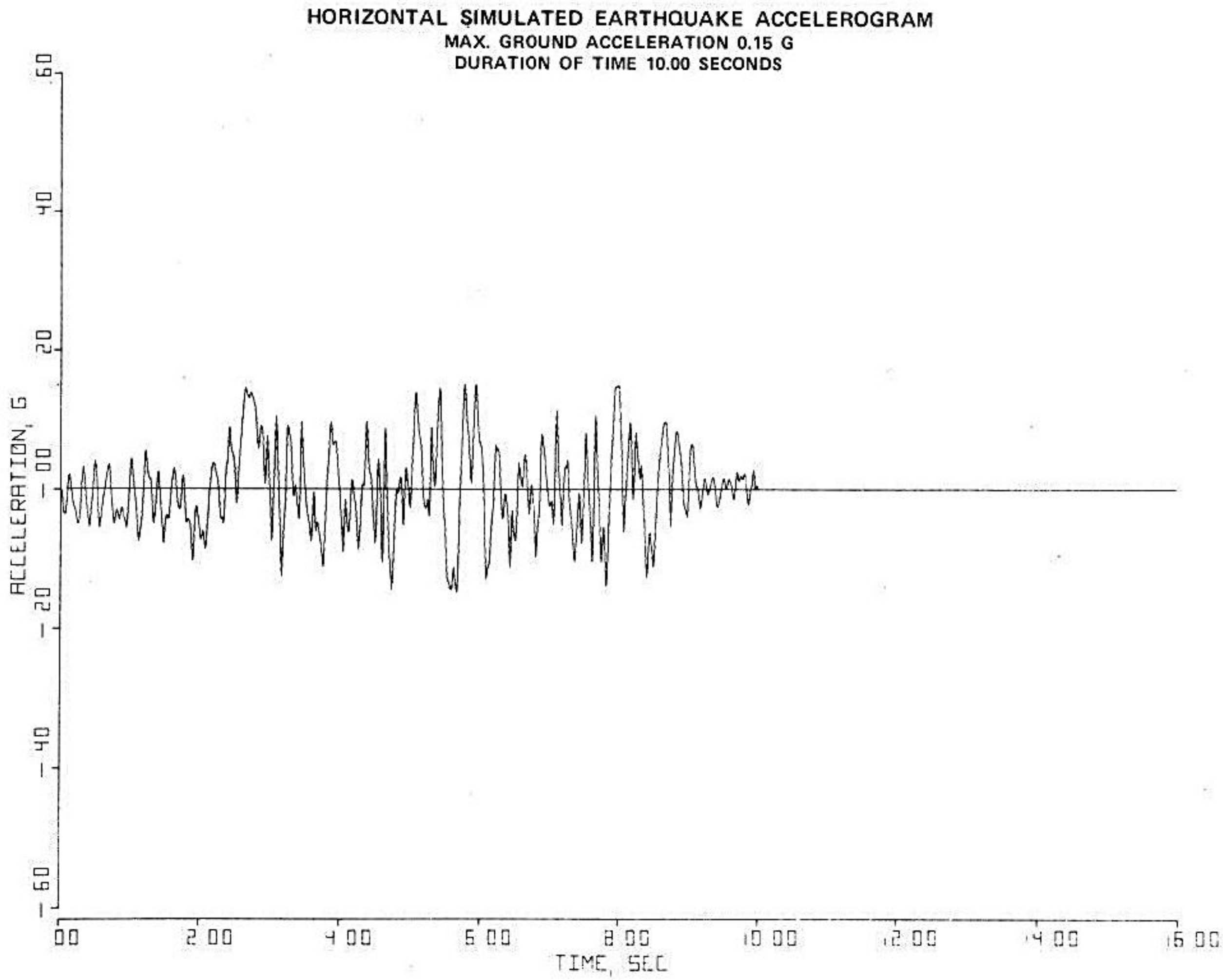


FIGURE 3.7.1-10

HORIZONTAL SIMULATED EARTHQUAKE ACCELEROGRAM – MAX. GROUND ACCELERATION 0.075 G OF TIME 10.00 SECONDS

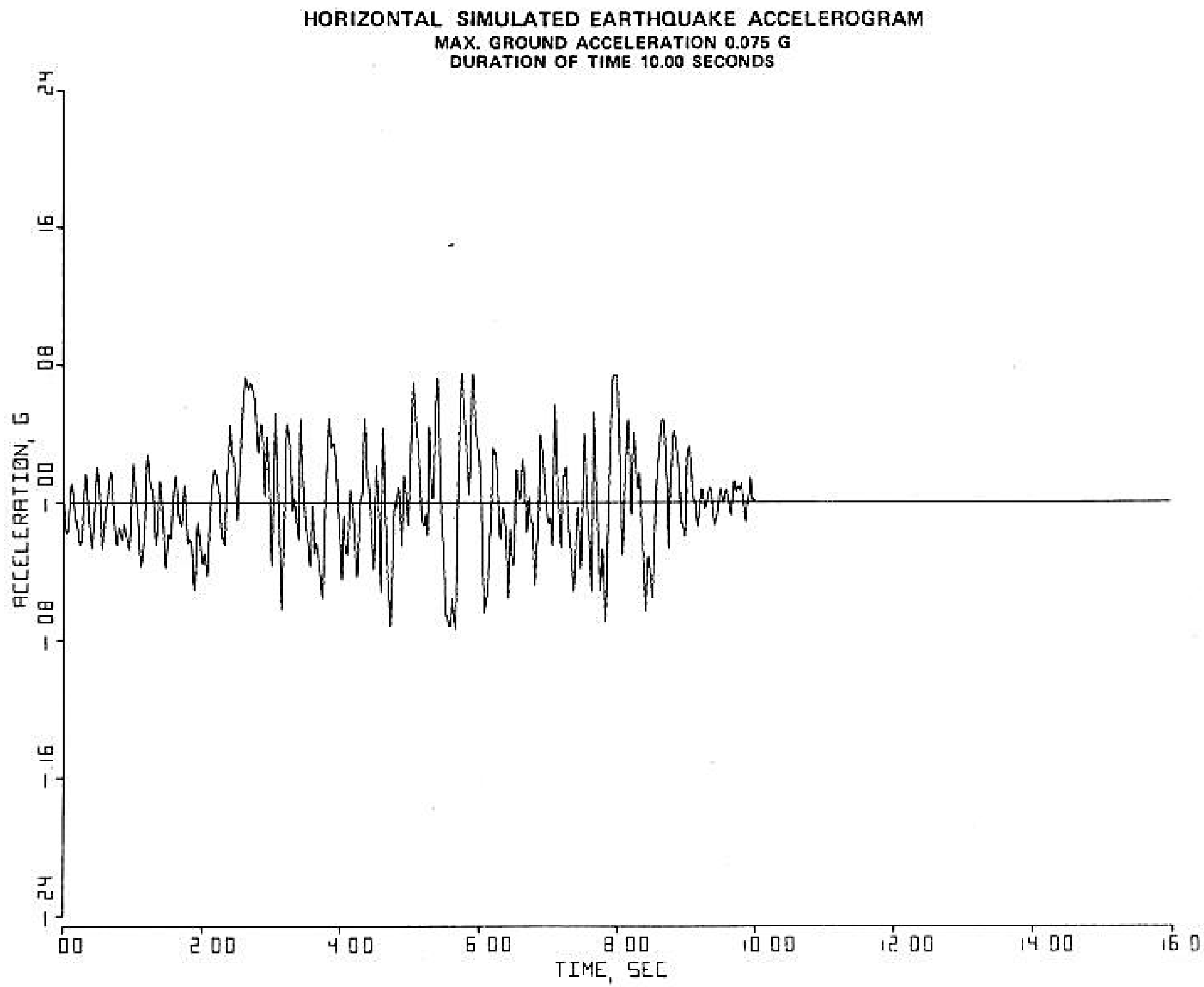


FIGURE 3.7.1-11

VERTICAL SIMULATED EARTHQUAKE ACCELEROGRAM MAX GROUND ACCELERATION 0.15 G DURATION OF TIME 10.00 SECONDS

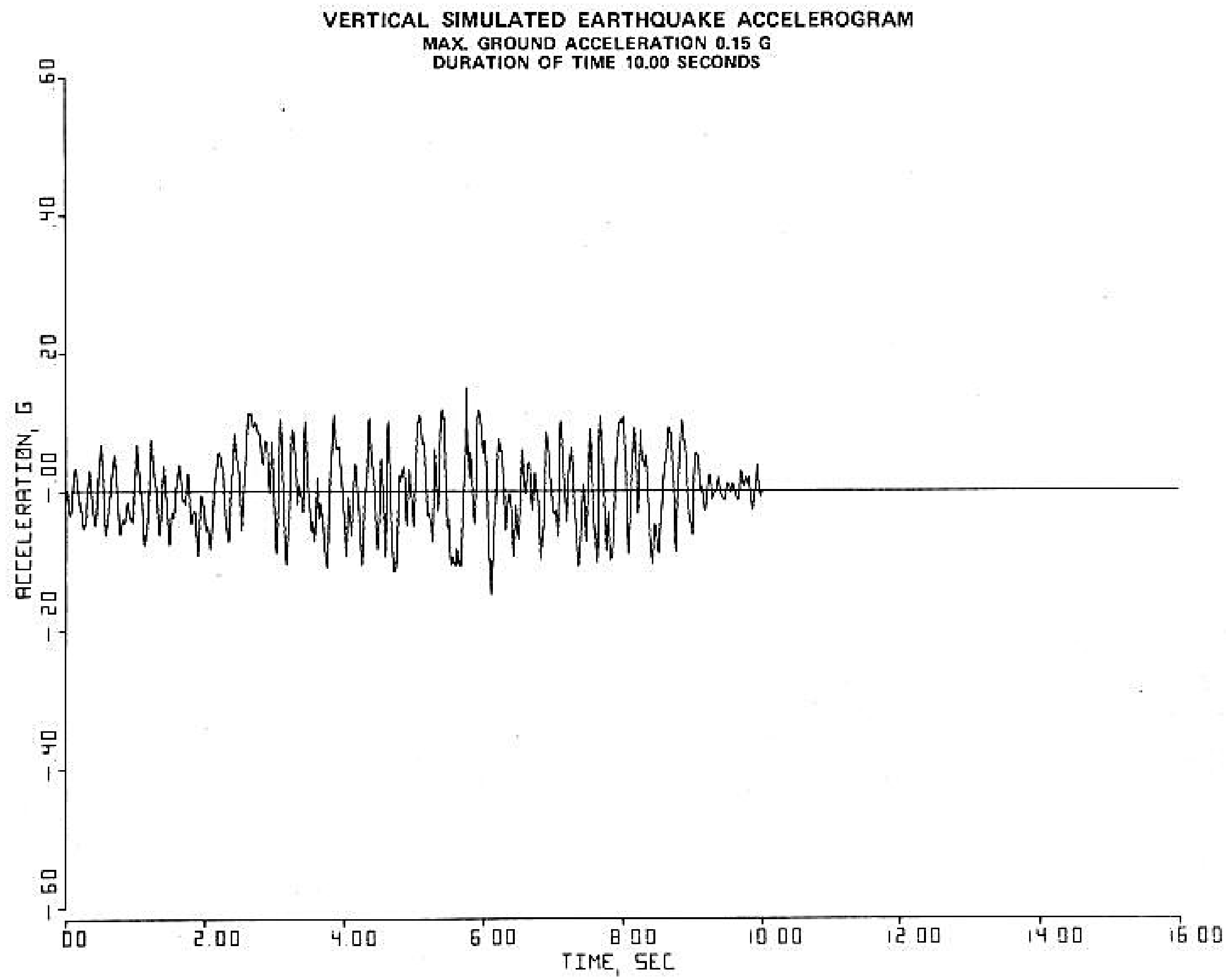


FIGURE 3.7.1-12

VERTICAL SIMULATED EARTHQUAKE ACCELEROGRAM MAX GROUND ACCELERATION 0.075 G DURATION OF TIME 10.00 SECONDS

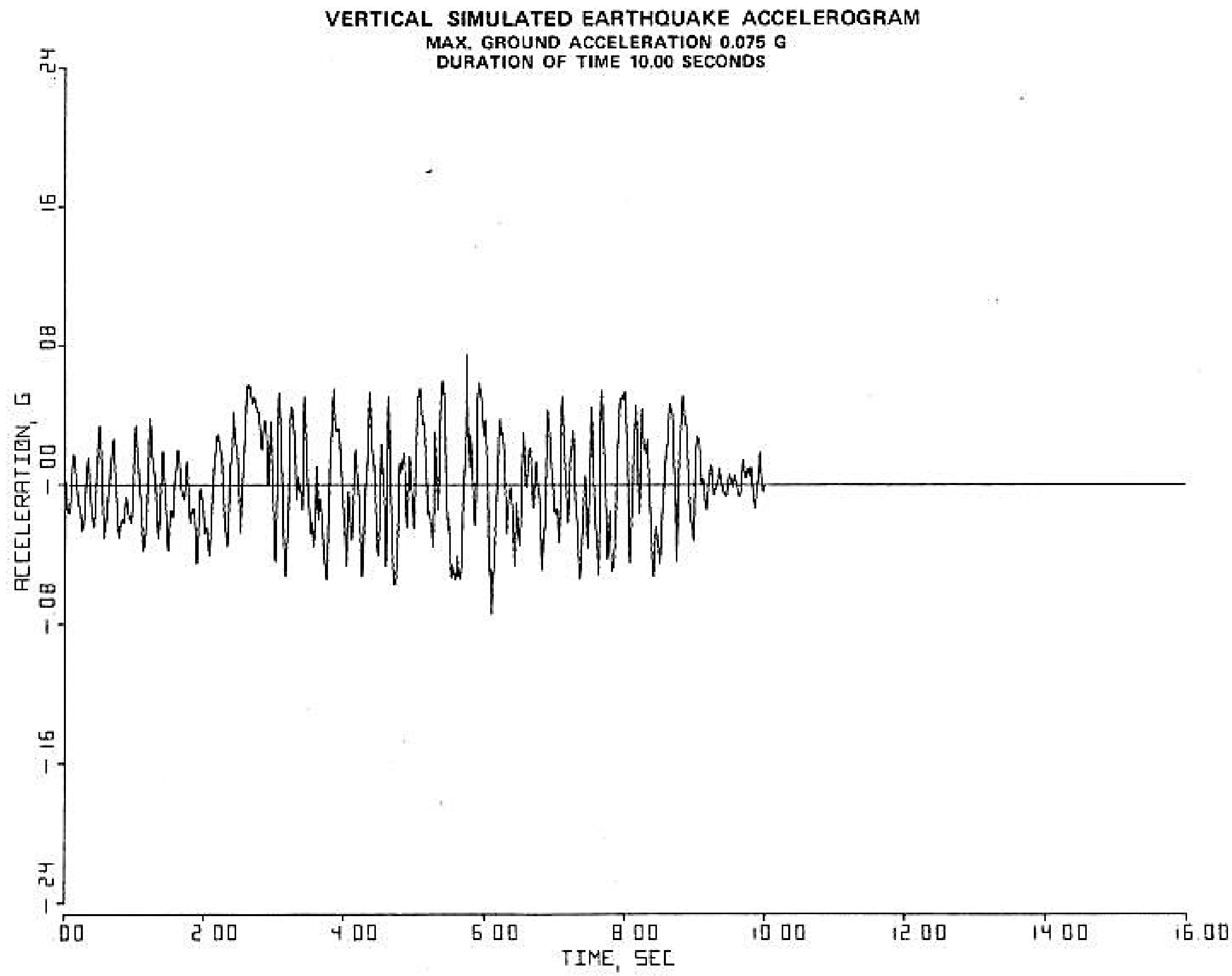


FIGURE 3.7.1-13

SSE HORIZONTAL DESIGN RESPONSE SPECTRA FOR 2 PERCENT DAMPING

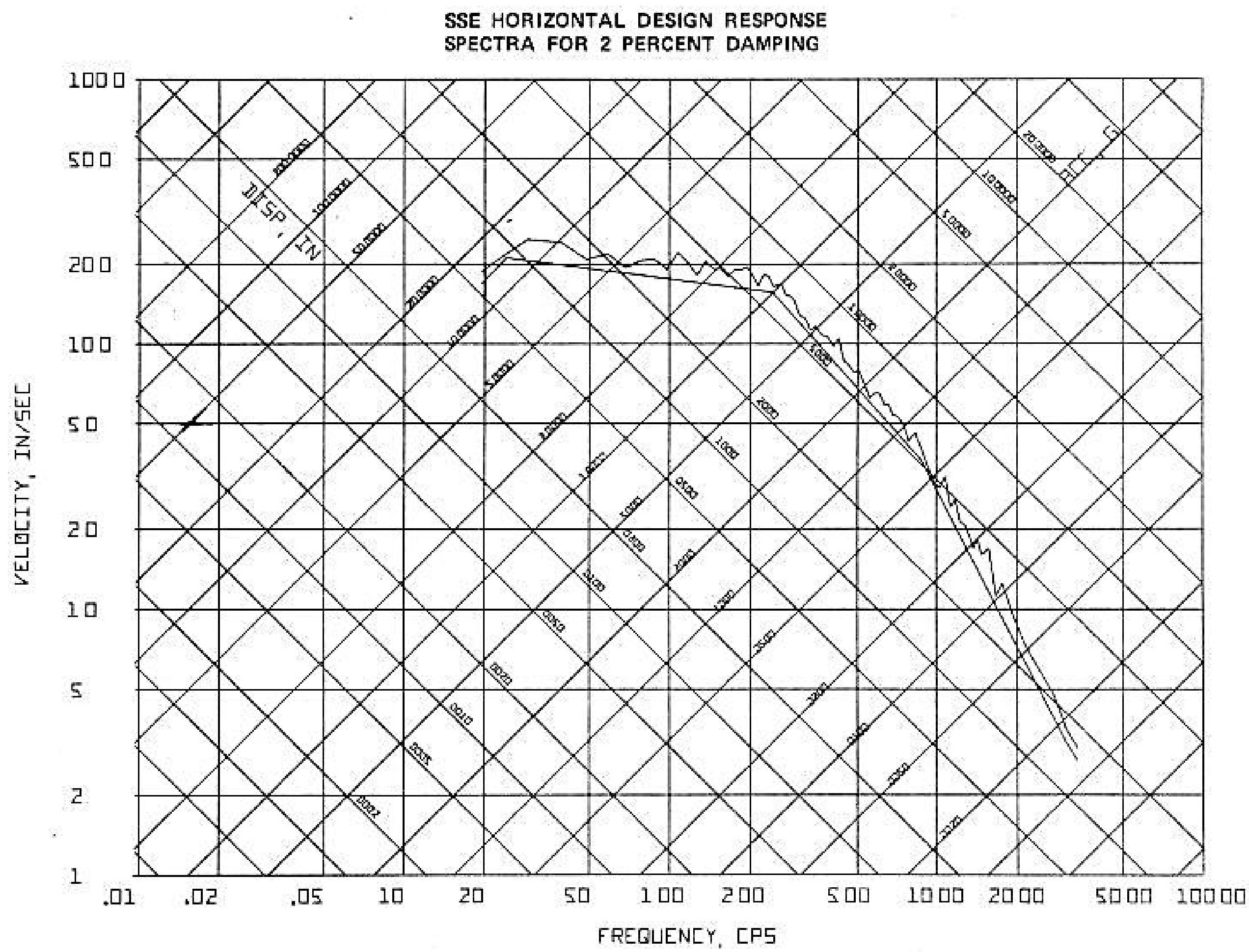


FIGURE 3.7.1-14

SSE HORIZONTAL DESIGN RESPONSE SPECTRA FOR 4 PERCENT DAMPING

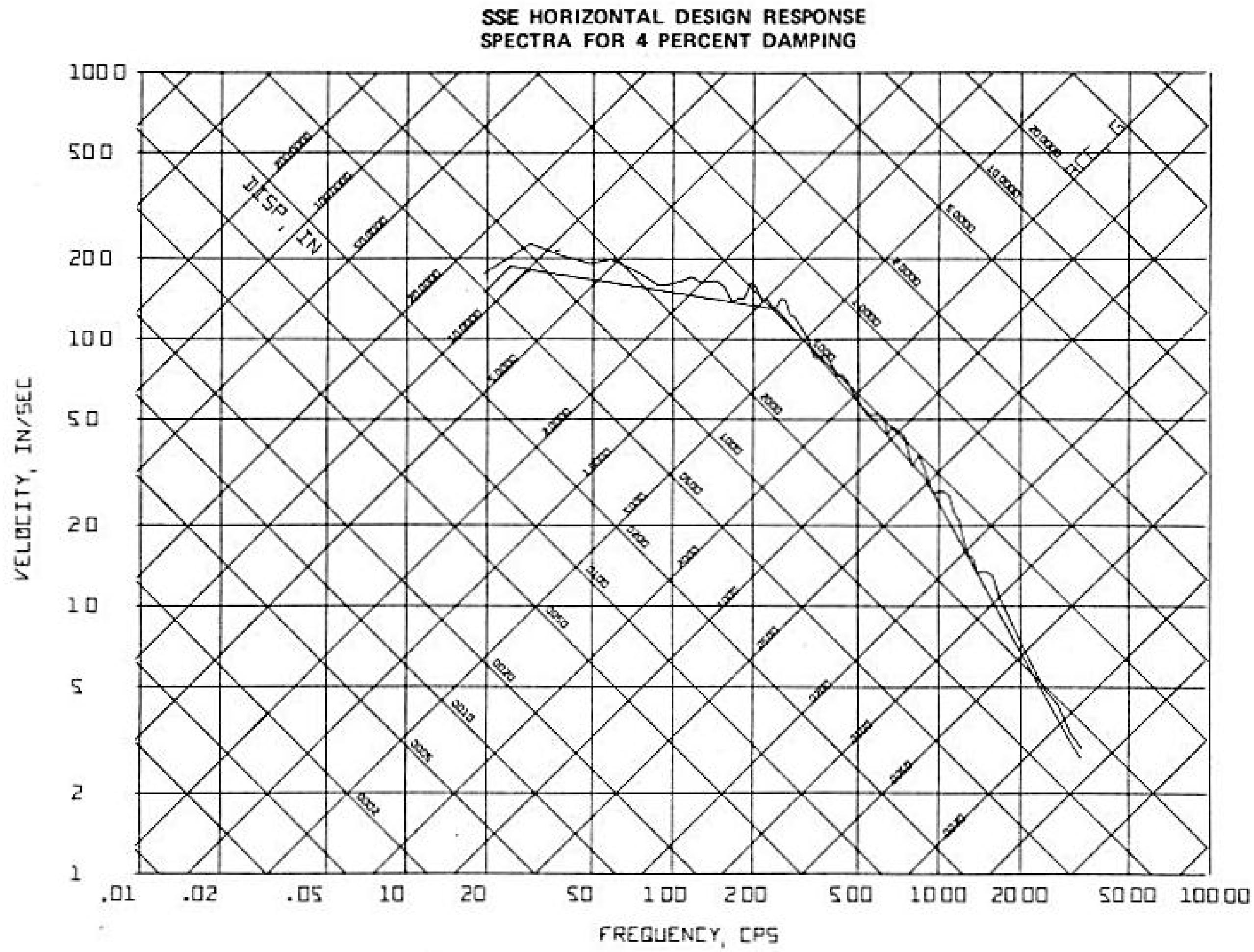


FIGURE 3.7.1-15

SSE HORIZONTAL DESIGN RESPONSE SPECTRA FOR 7 PERCENT DAMPING

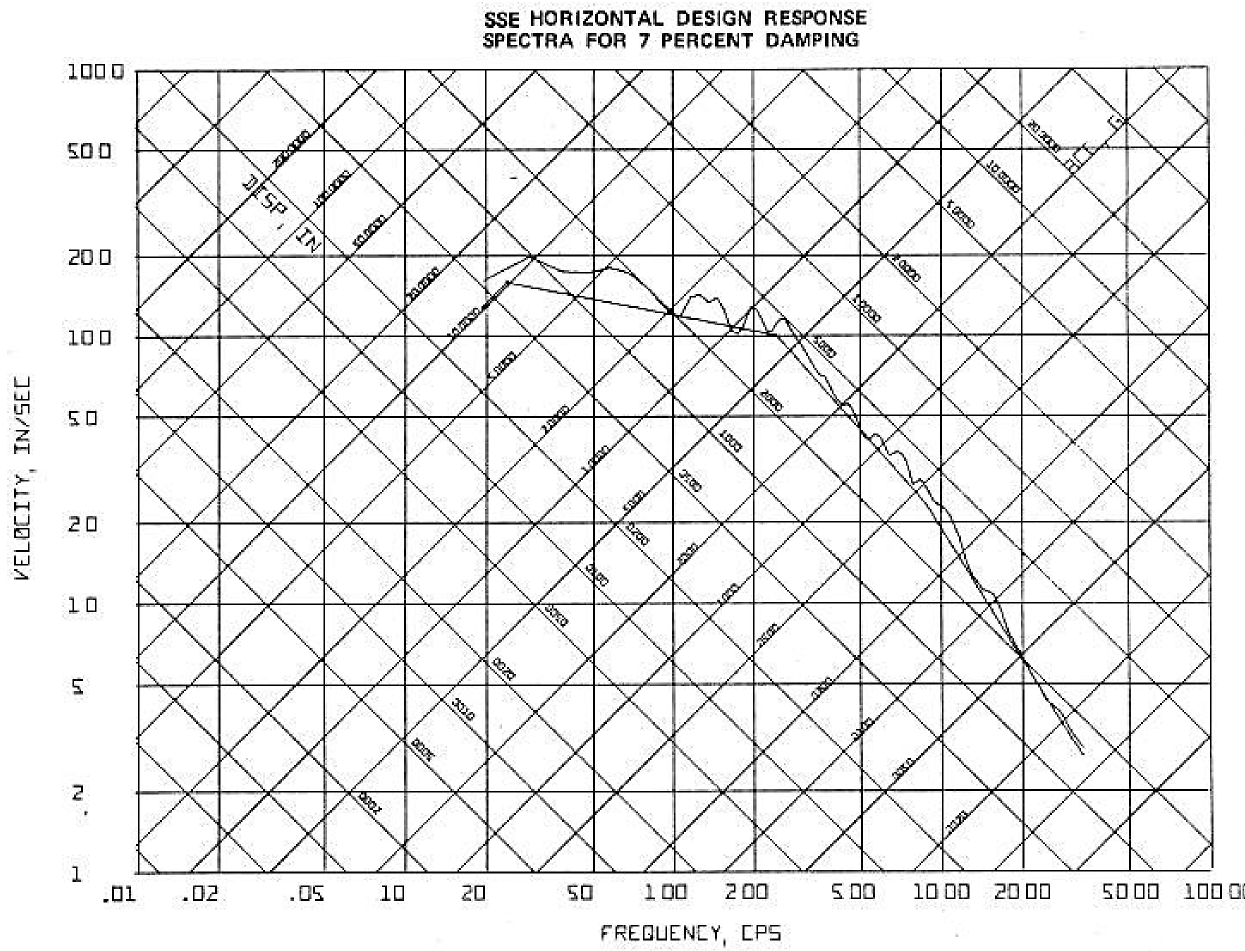


FIGURE 3.7.1-16

OBE HORIZONTAL DESIGN RESPONSE SPECTRA FOR 2 PERCENT DAMPING

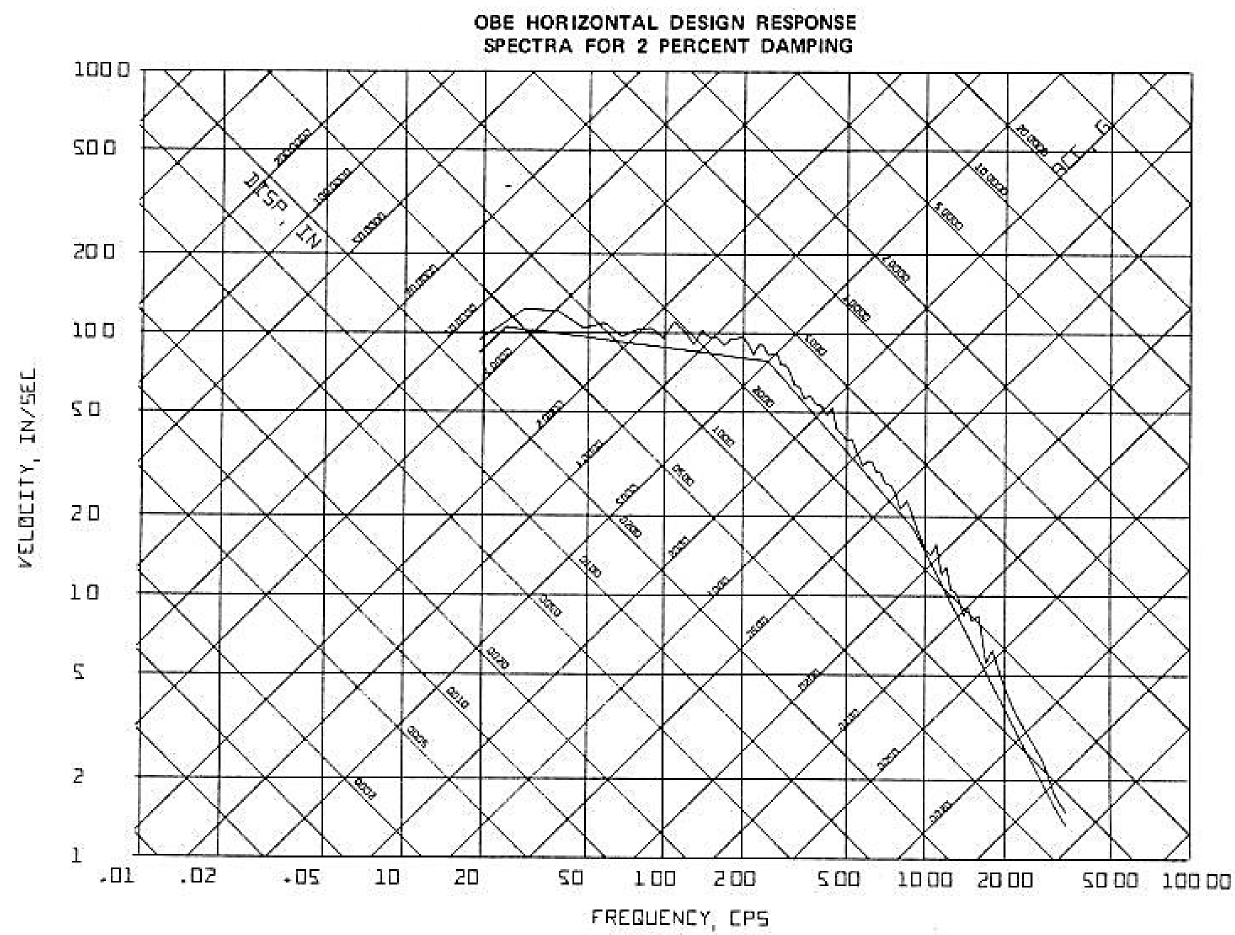


FIGURE 3.7.1-17

OBE HORIZONTAL DESIGN RESPONSE SPECTRA FOR 4 PERCENT DAMPING

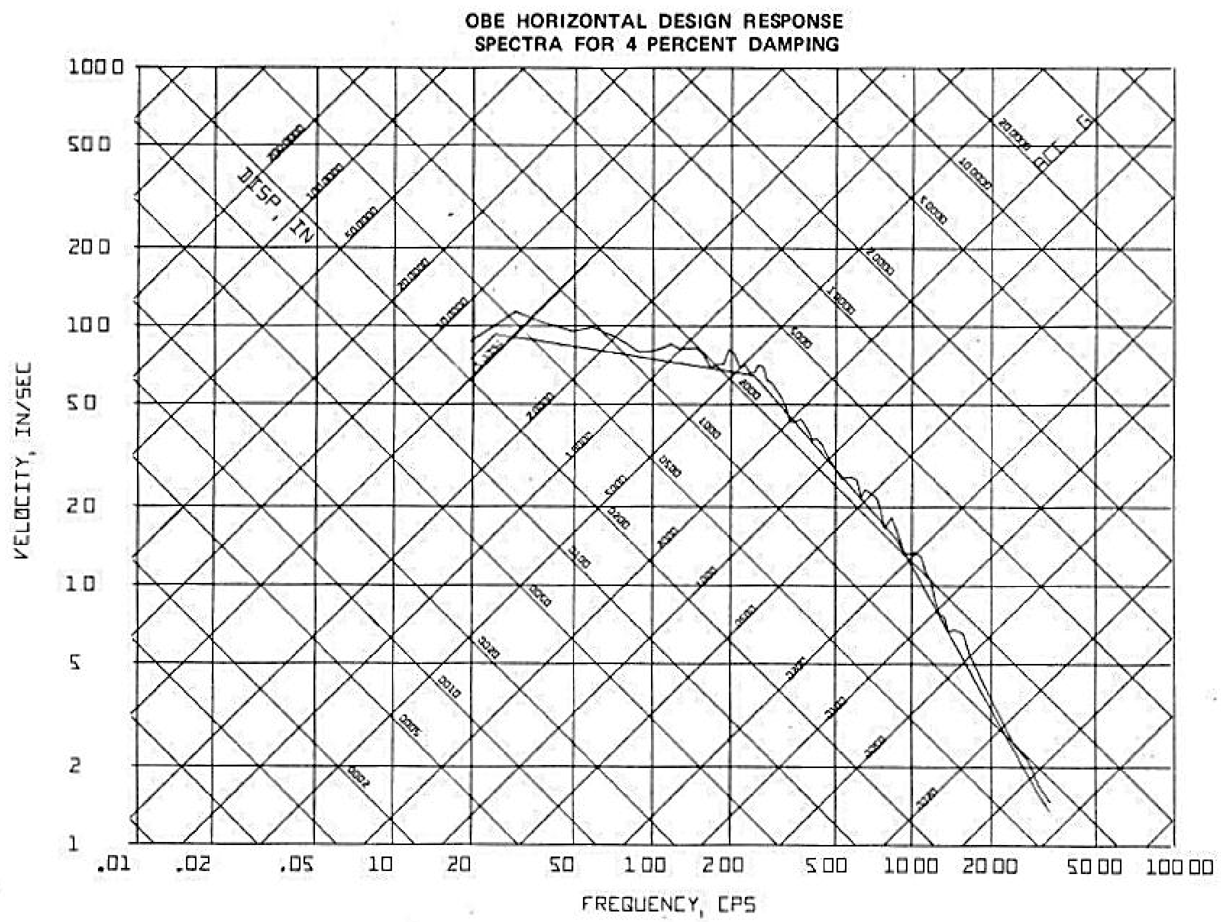


FIGURE 3.7.1-18

OBE HORIZONTAL DESIGN RESPONSE SPECTRA FOR 7 PERCENT DAMPING

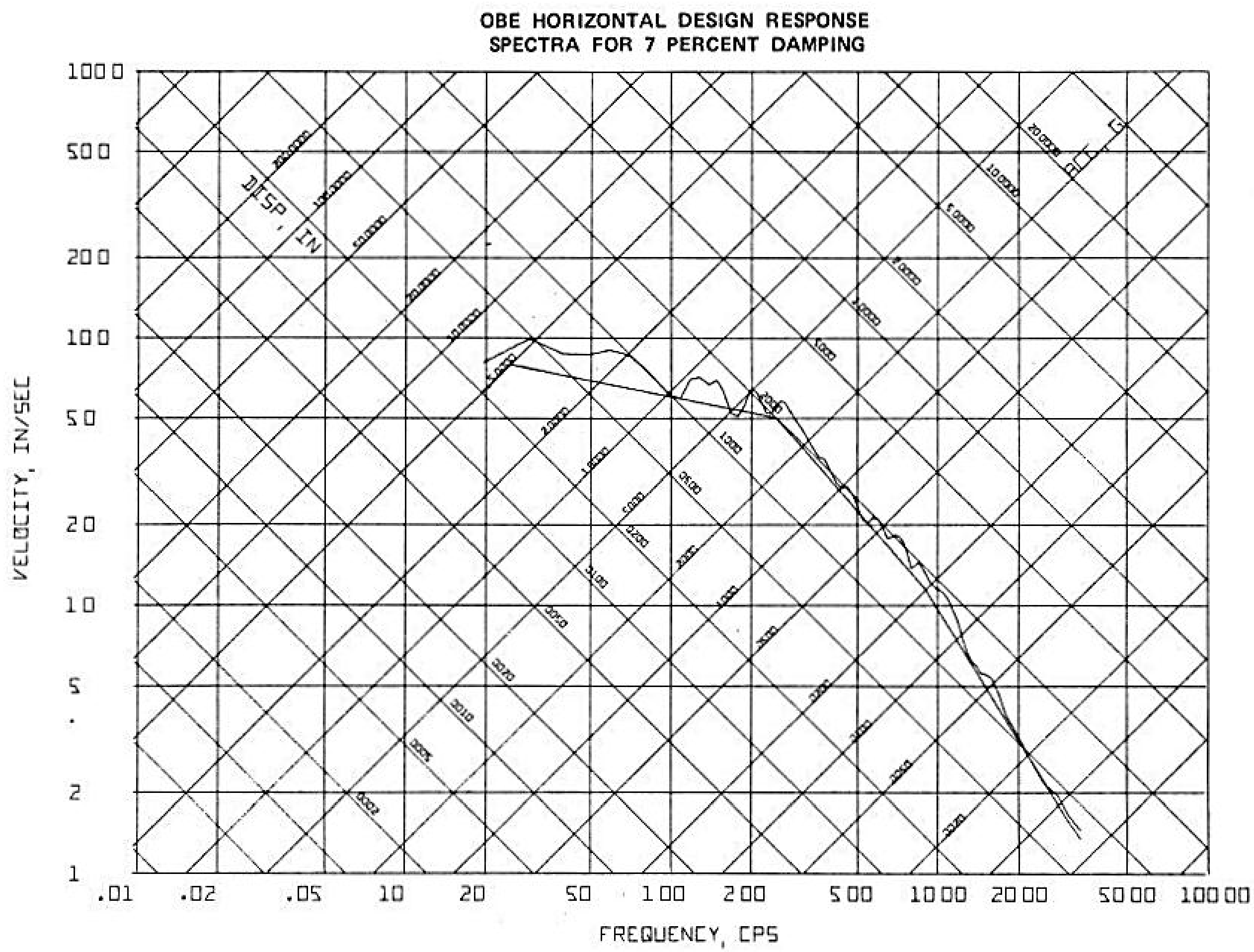


FIGURE 3.7.1-19

SSE VERTICAL DESIGN RESPONSE SPECTRA FOR 2 PERCENT DAMPING

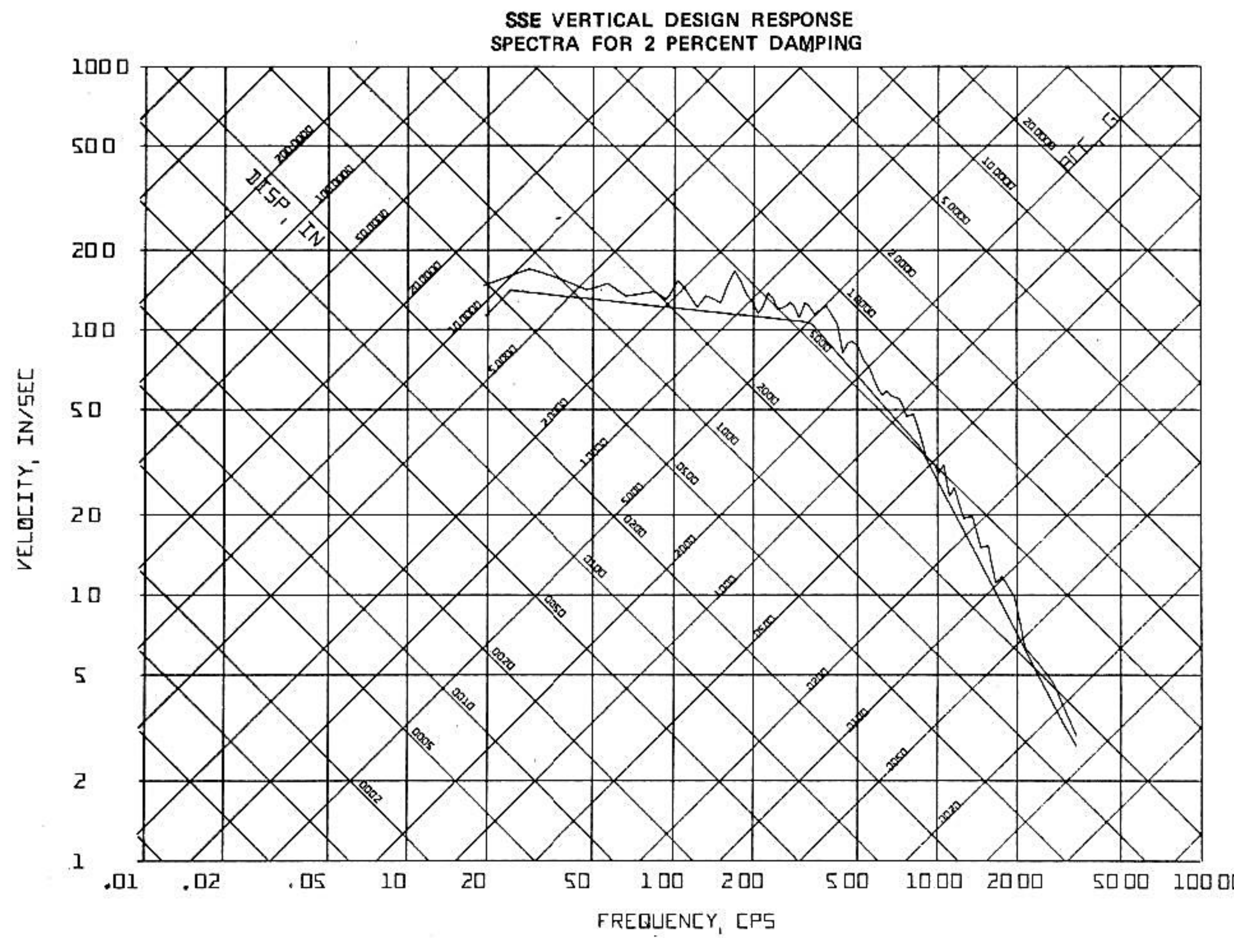


FIGURE 3.7.1-20

SSE VERTICAL DESIGN RESPONSE SPECTRA FOR 4 PERCENT DAMPING

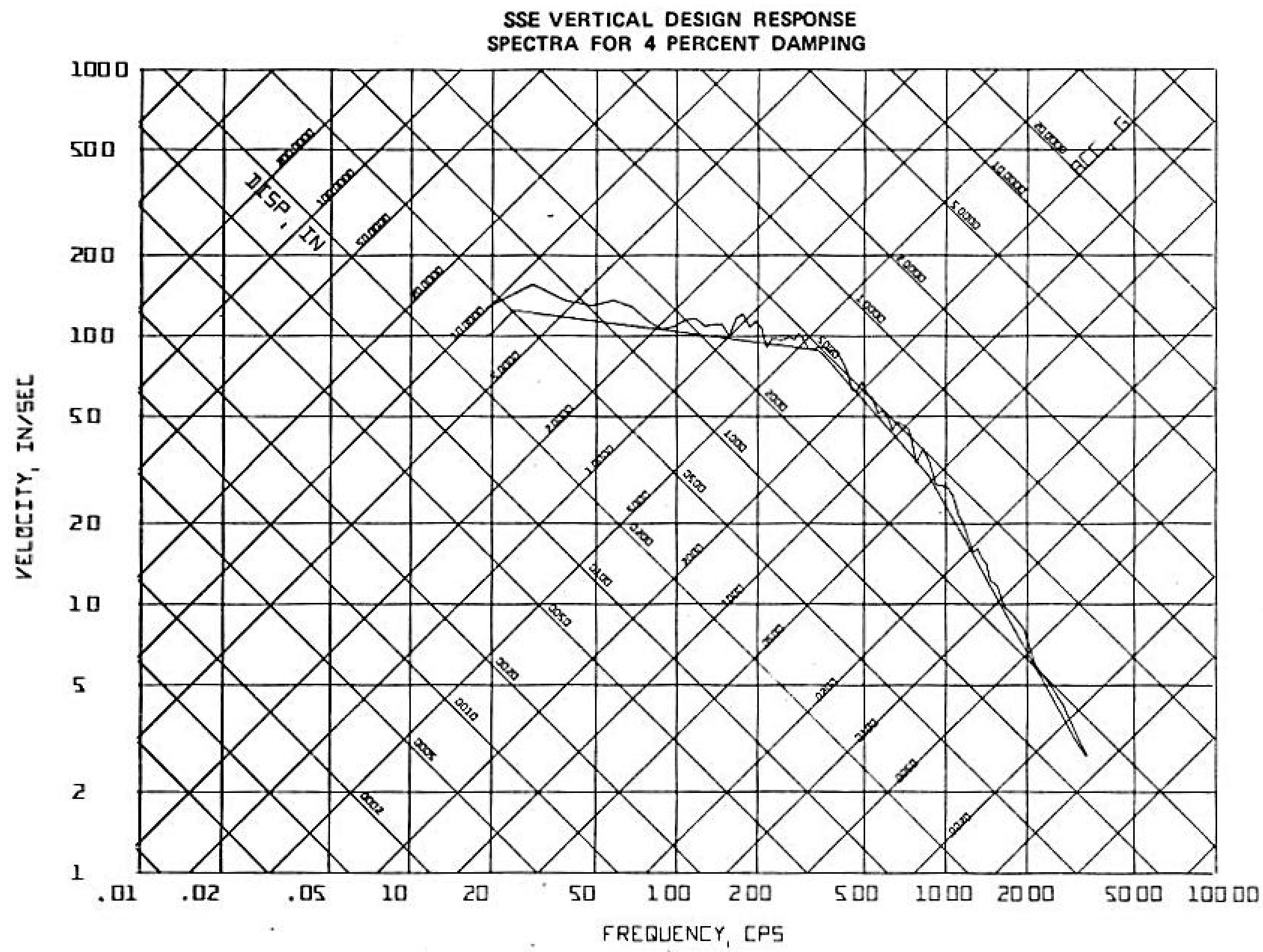


FIGURE 3.7.1-21

SSE VERTICAL DESIGN RESPONSE SPECTRA FOR 7 PERCENT DAMPING

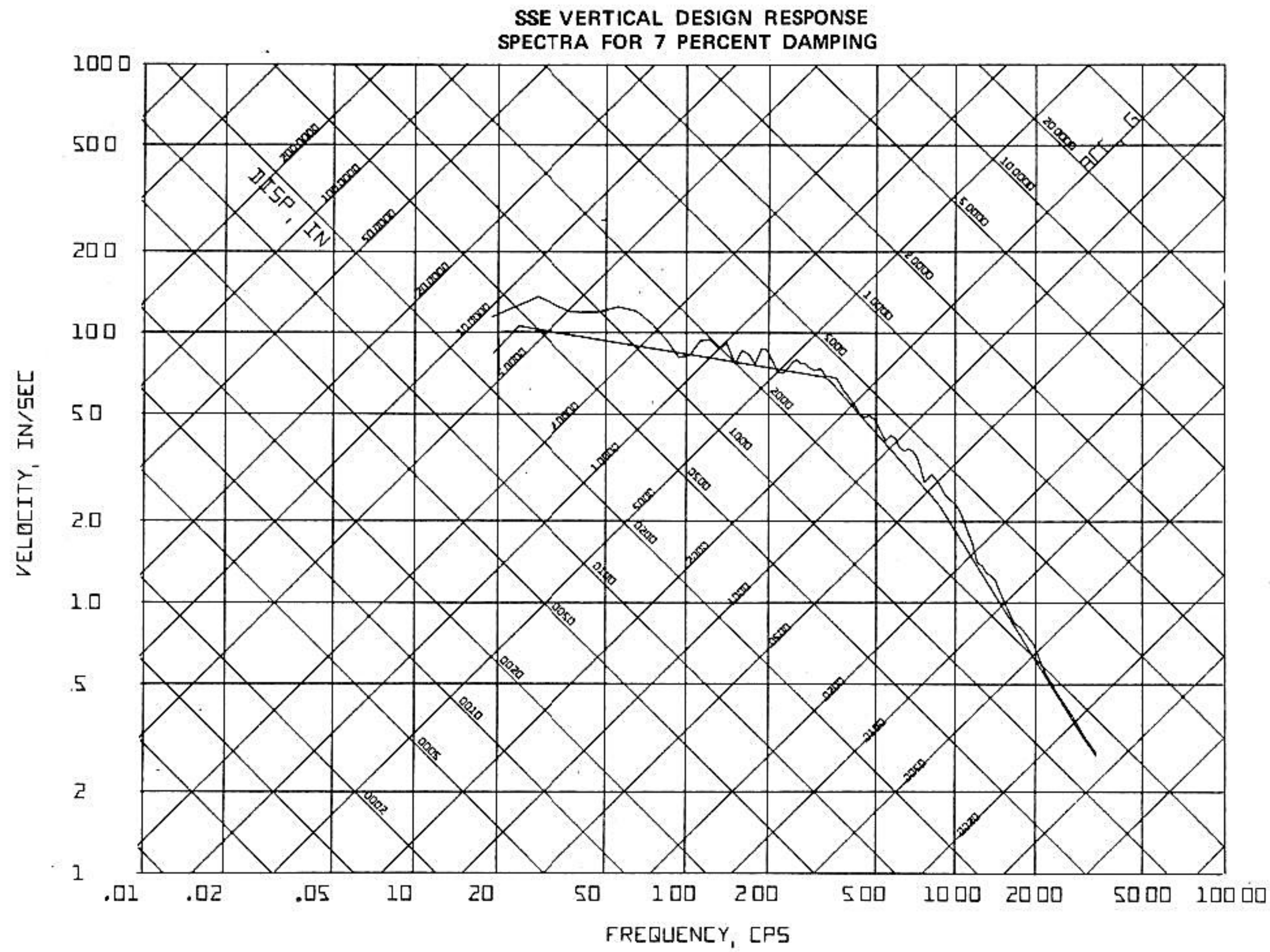


FIGURE 3.7.1-22

OBE VERTICAL DESIGN RESPONSE SPECTRA FOR 2 PERCENT DAMPING

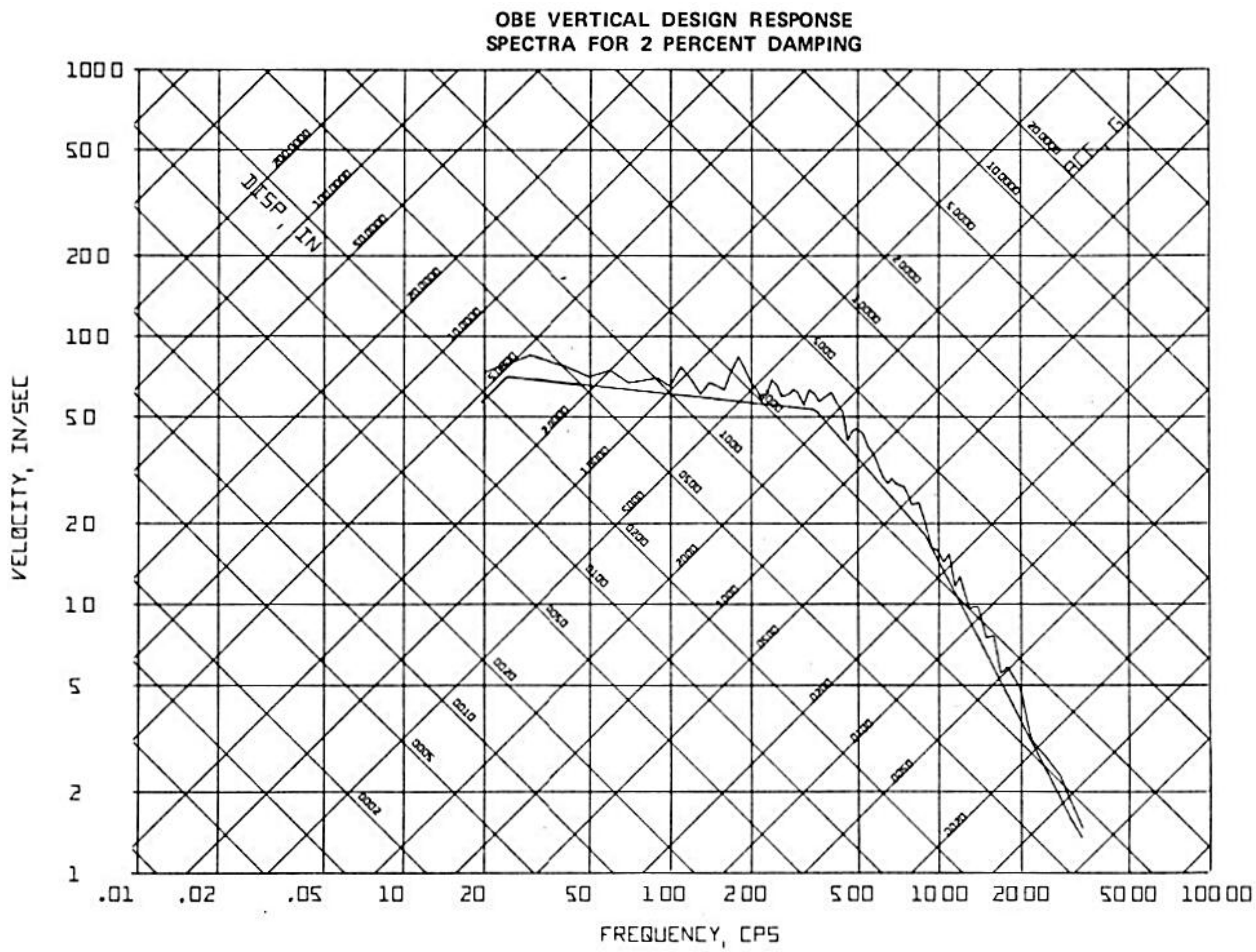


FIGURE 3.7.1-23

OBE VERTICAL DESIGN RESPONSE SPECTRA FOR 4 PERCENT DAMPING

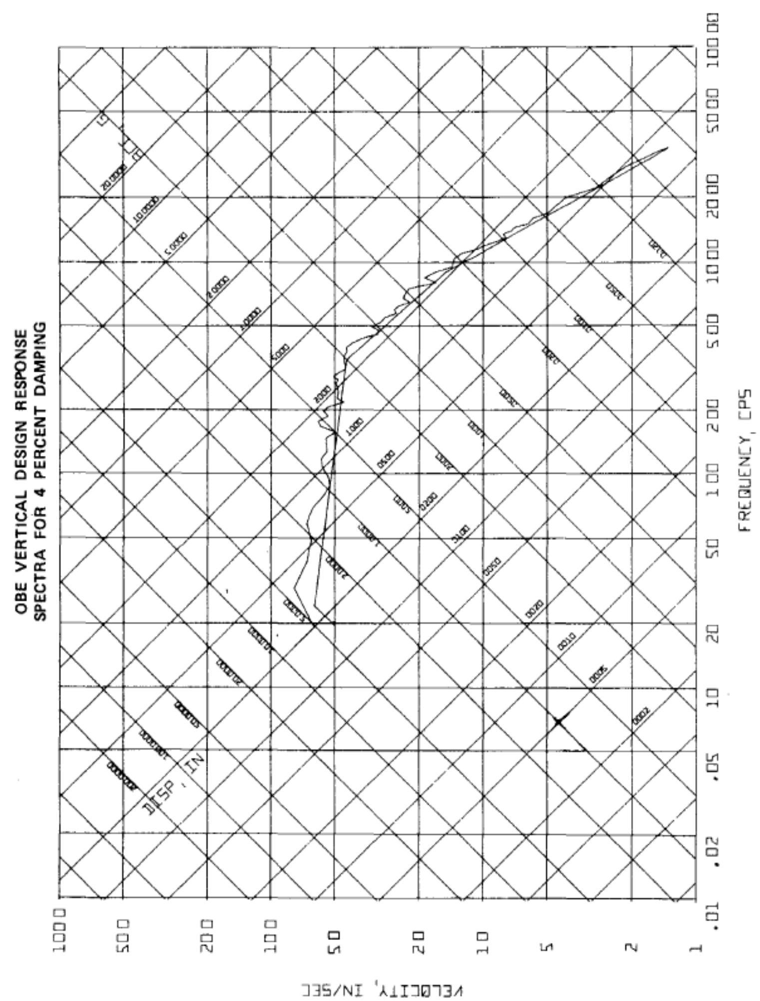


FIGURE 3.7.1-24

OBE VERTICAL DESIGN RESPONSE SPECTRA FOR 7 PERCENT DAMPING

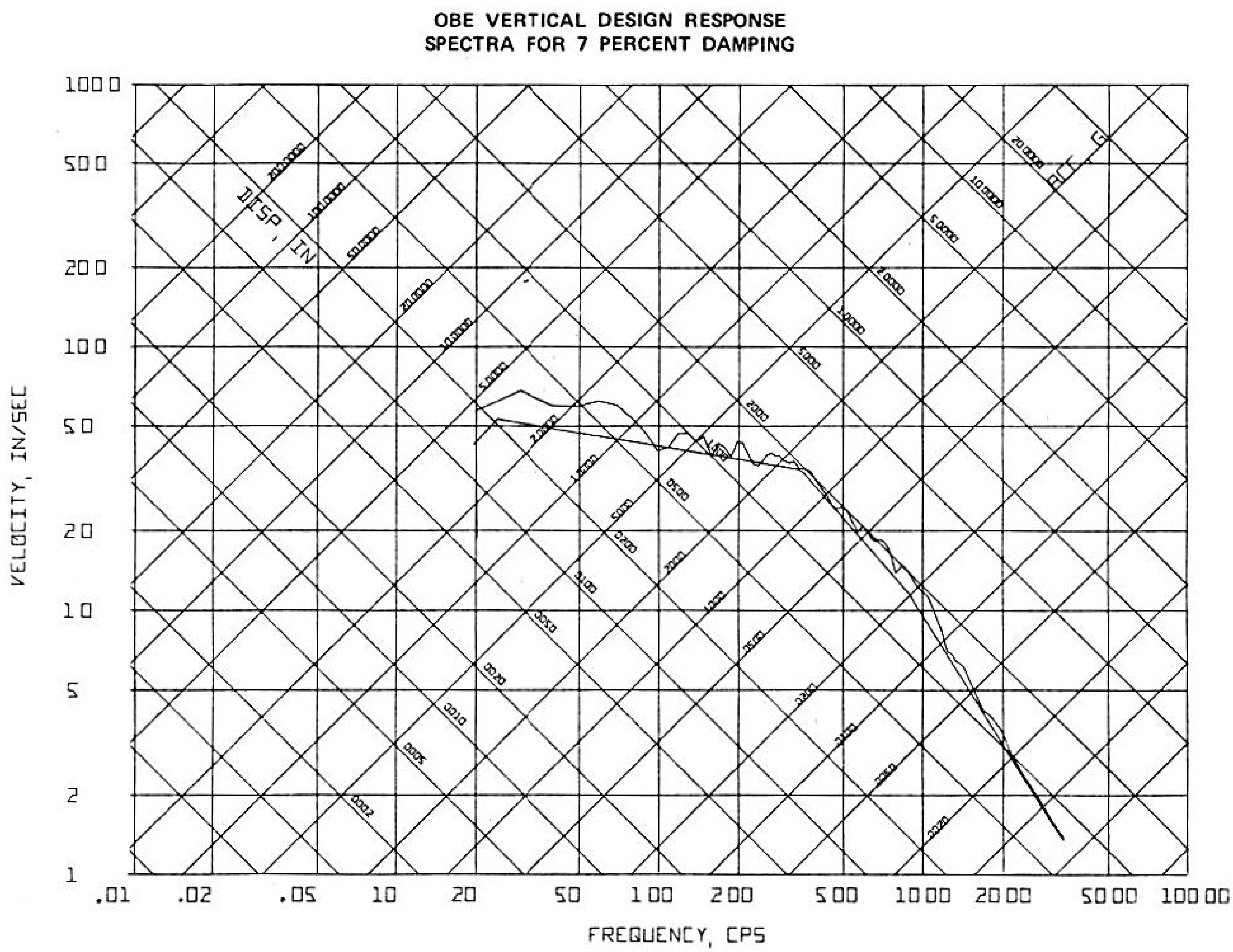


FIGURE 3.7.1-25

E-W HORIZONTAL EARTHQUAKE ACCELEROGRAM
MAX. GROUND ACCELERATION 0.15 G
DURATION OF TIME 10.00 SECONDS
STATISTICALLY INDEPENDENT

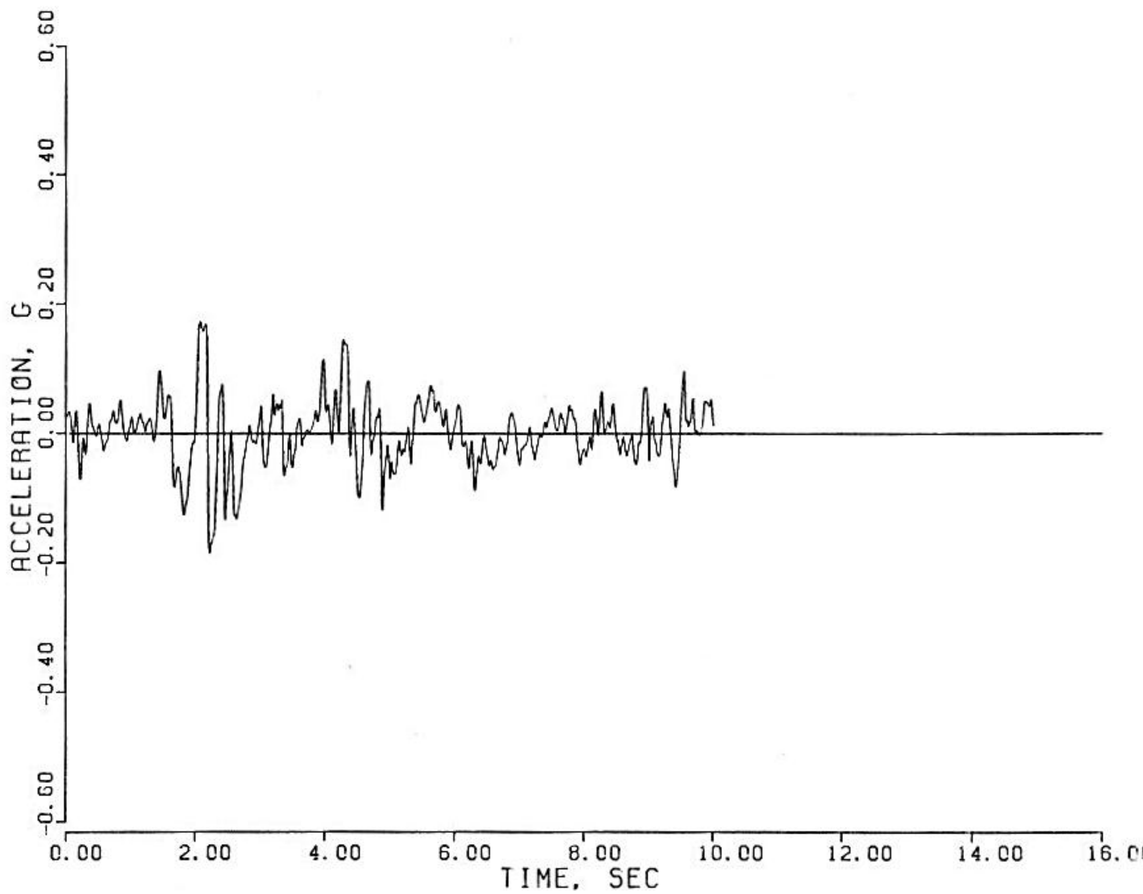


FIGURE 3.7.1-26

E-W HORIZONTAL EARTHQUAKE ACCELEROGRAM
MAX. GROUND ACCELERATION 0.075 G
DURATION OF TIME 10.00 SECONDS
STATISTICALLY INDEPENDENT

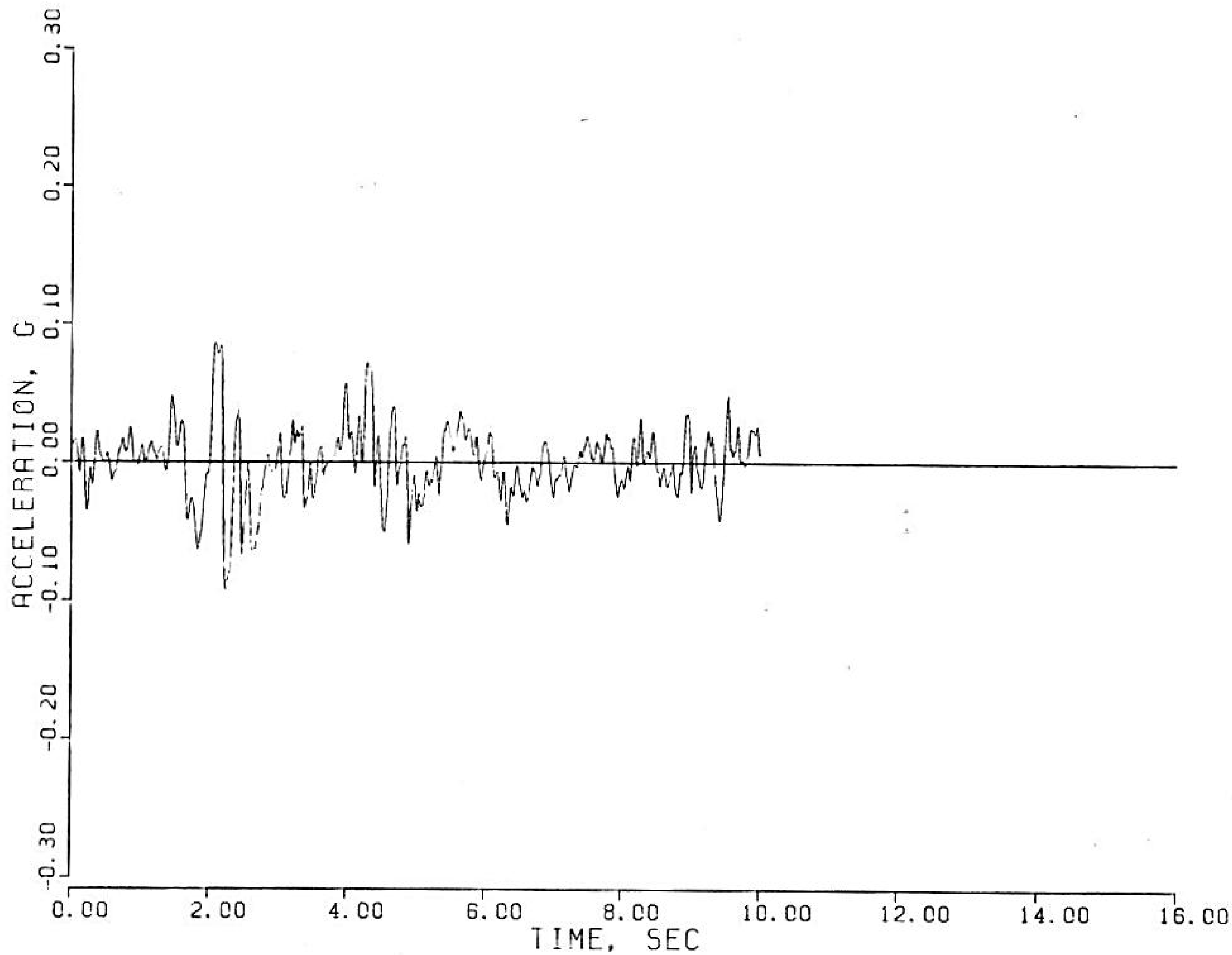


FIGURE 3.7.1-27

VERTICAL EARTHQUAKE ACCELEROGRAM
MAX. GROUND ACCELERATION 0.15 G
DURATION OF TIME 10.00 SECONDS
STATISTICALLY INDEPENDENT

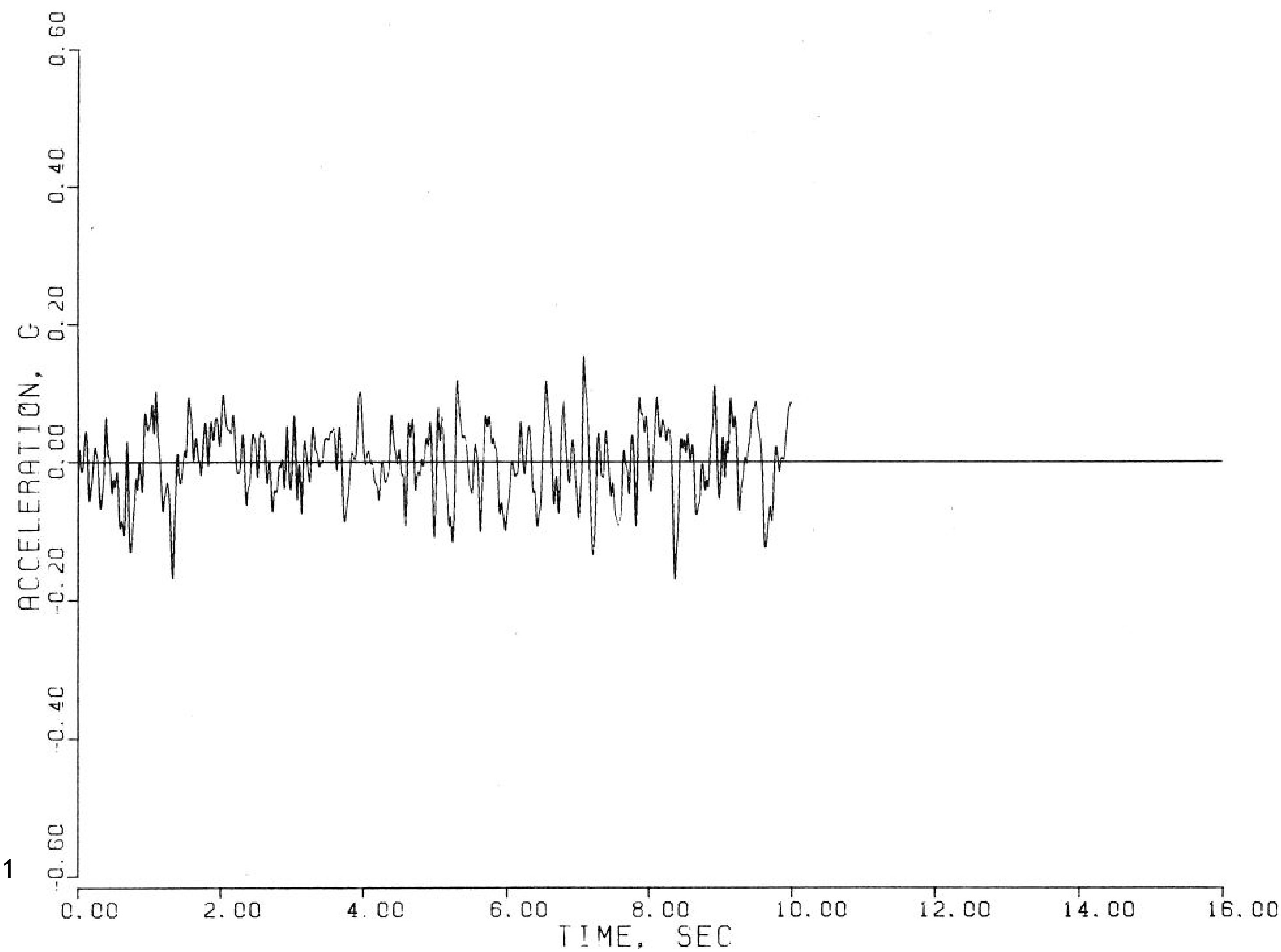


FIGURE 3.7.1-28

VERTICAL EARTHQUAKE ACCELEROGRAM
MAX. GROUND ACCELERATION 0.075 G
DURATION OF TIME 10.00 SECONDS
STATISTICALLY INDEPENDENT

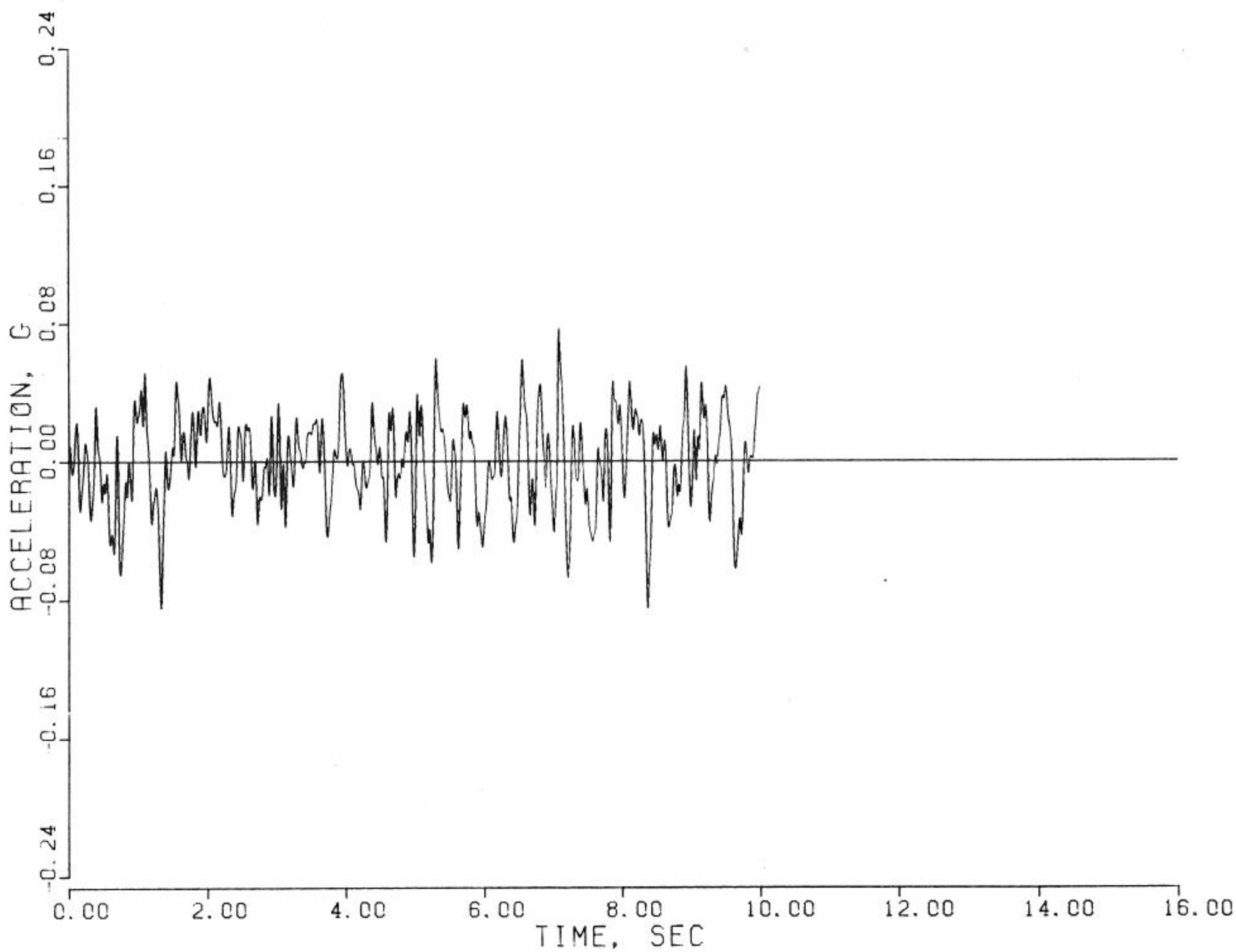


FIGURE 3.7.1-29

SSE E-W HORIZONTAL DESIGN RESPONSE
SPECTRA FOR 2 PERCENT DAMPING
STATISTICALLY INDEPENDENT

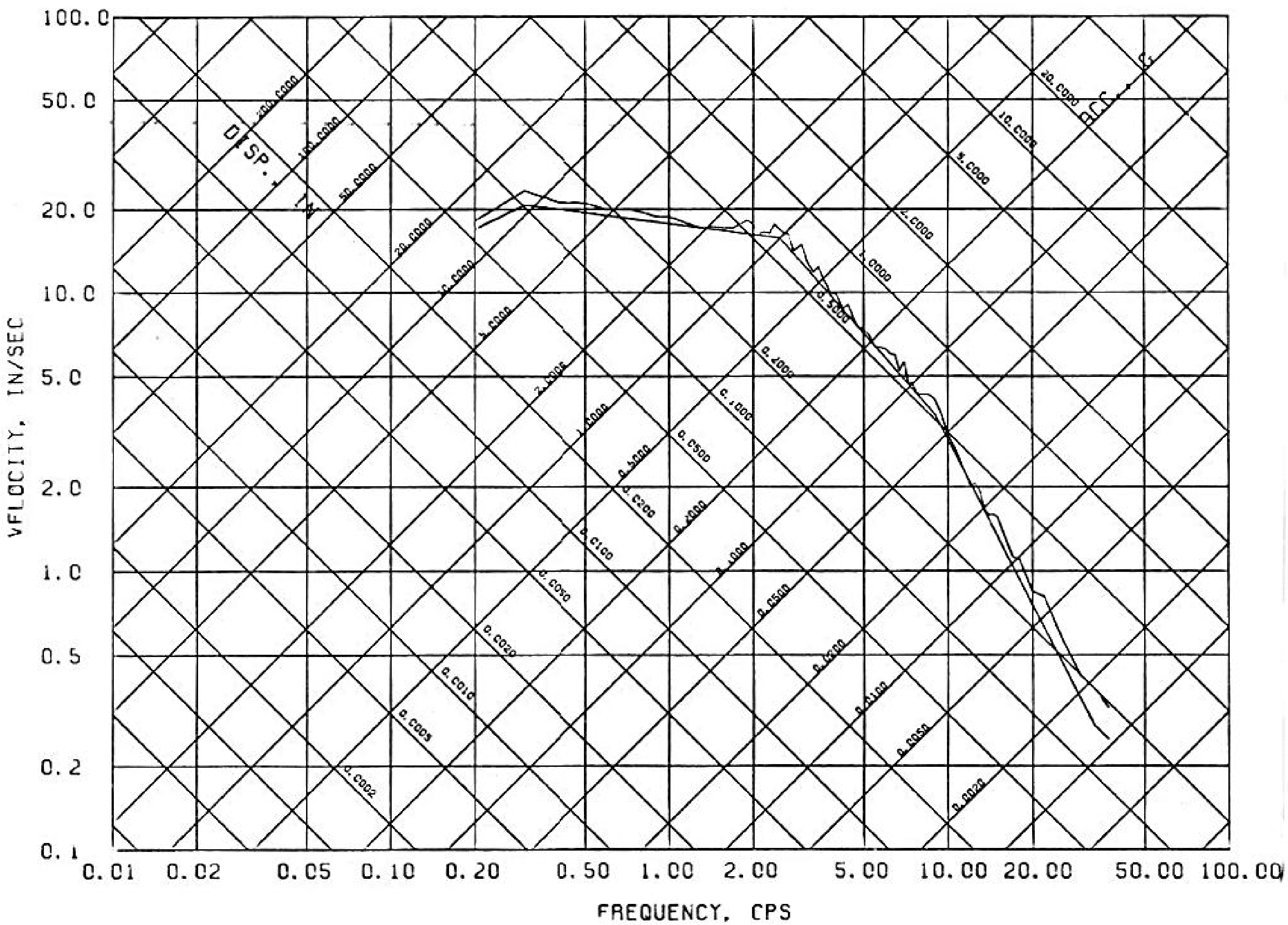


FIGURE 3.7.1-30

SSE E-W HORIZONTAL DESIGN RESPONSE SPECTRA FOR 4 PERCENT DAMPING STATISTICALLY INDEPENDENT

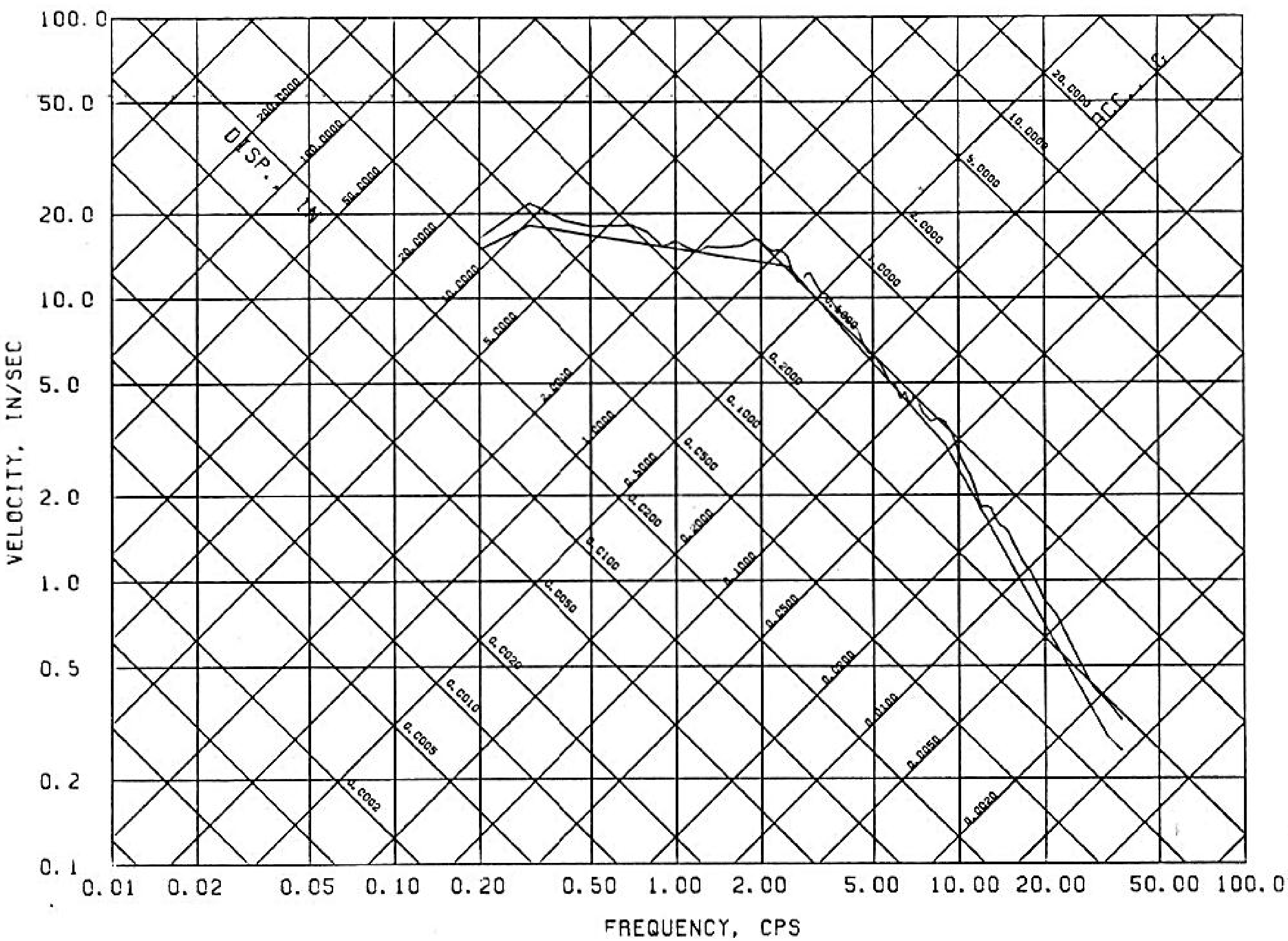


FIGURE 3.7.1-31

SSE E-W HORIZONTAL DESIGN RESPONSE
SPECTRA FOR 7 PERCENT DAMPING
STATISTICALLY INDEPENDENT

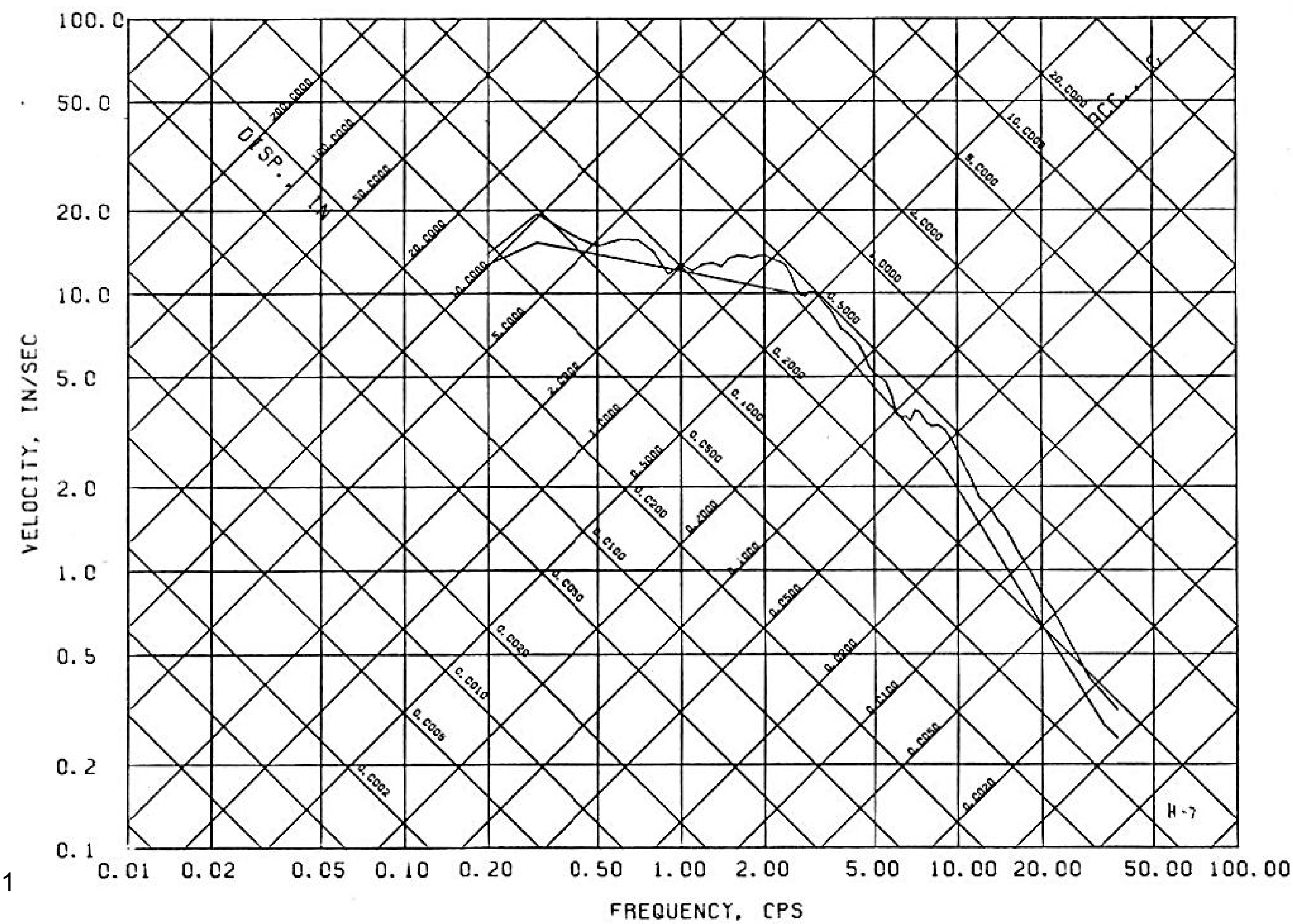


FIGURE 3.7.1-32

SSE VERTICAL DESIGN RESPONSE SPECTRA
FOR 2 PERCENT DAMPING
STATISTICALLY INDEPENDENT

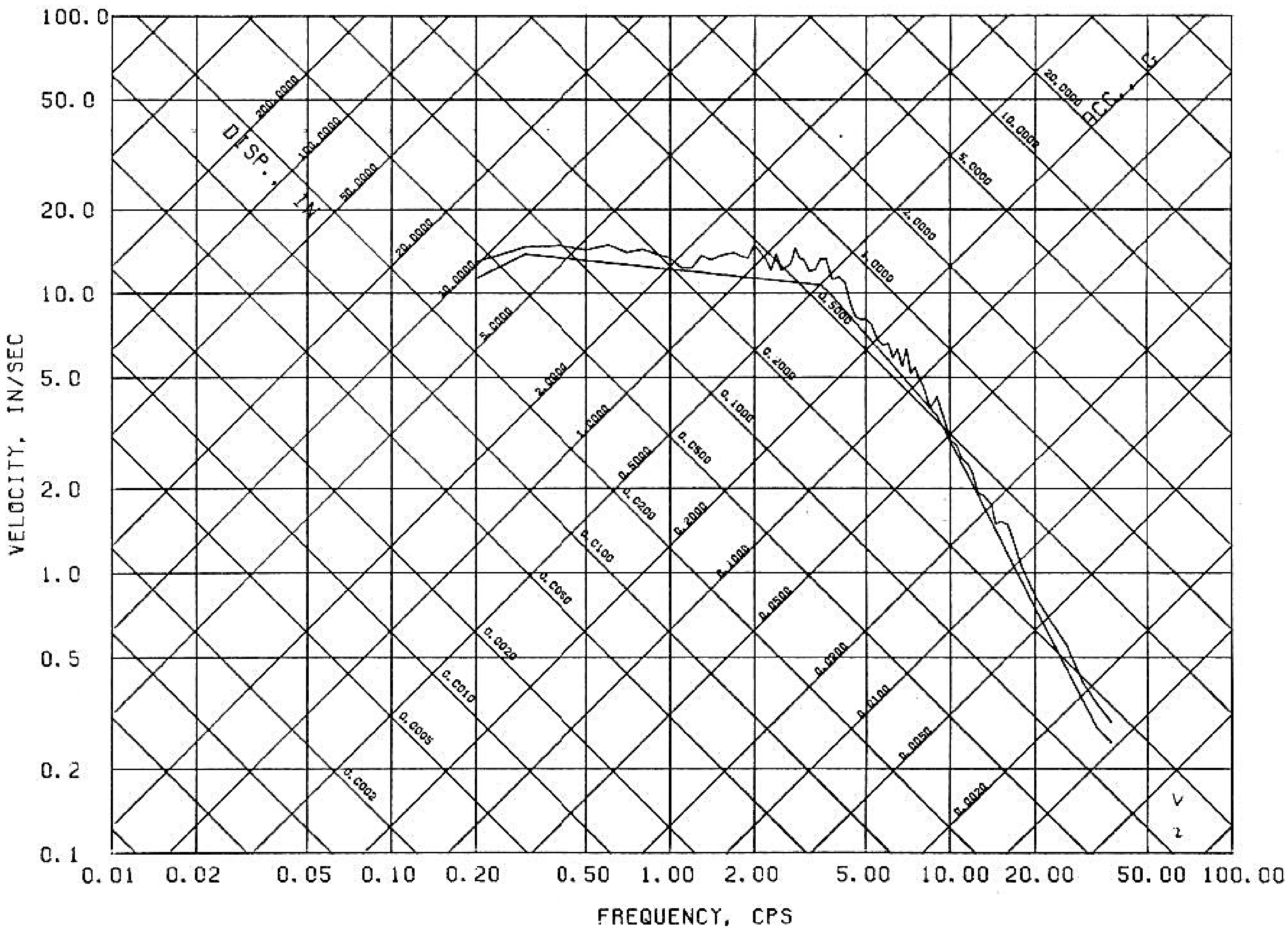


FIGURE 3.7.1-33

SSE VERTICAL DESIGN RESPONSE SPECTRA
FOR 4 PERCENT DAMPING
STATISTICALLY INDEPENDENT

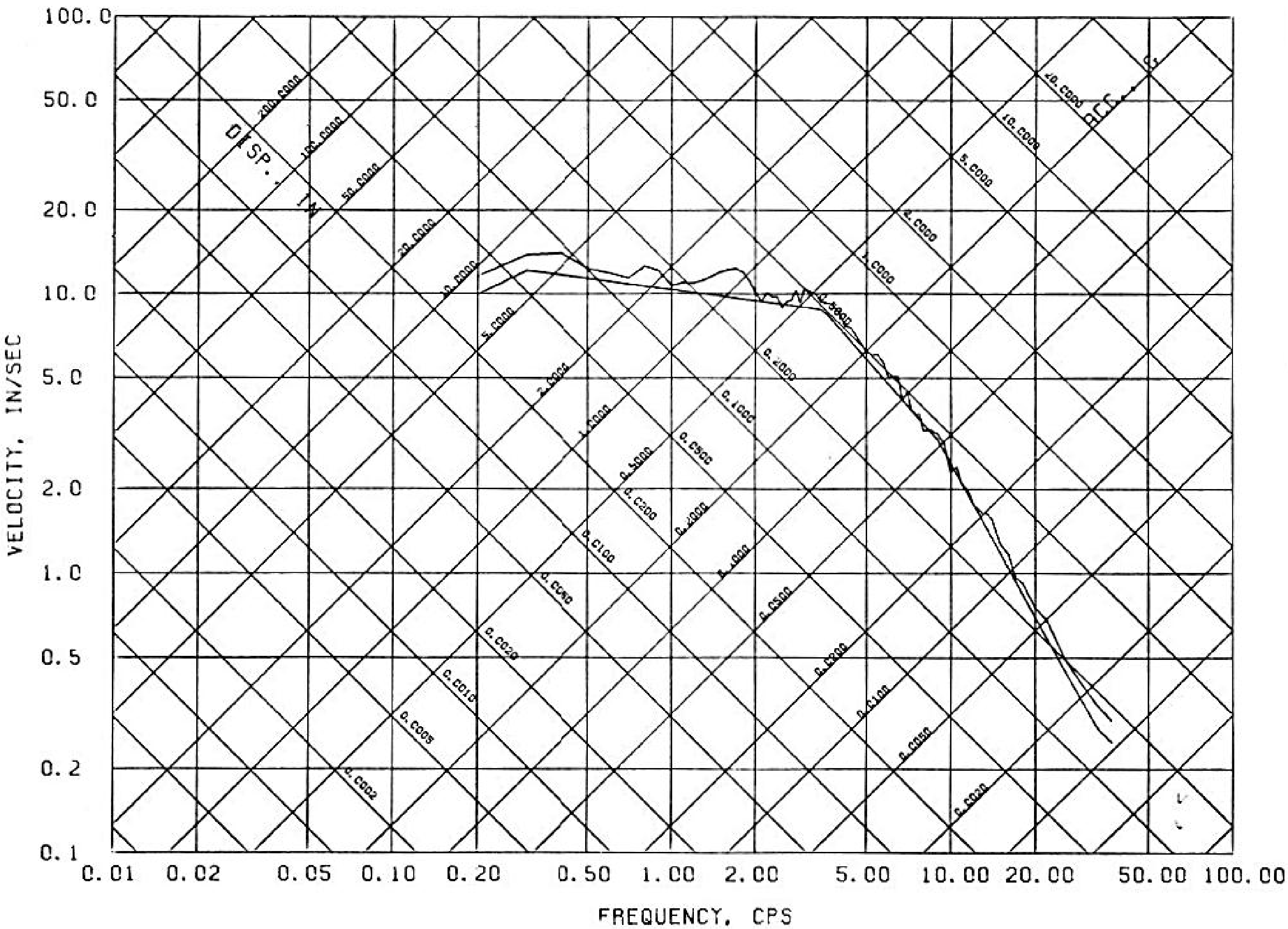


FIGURE 3.7.1-34

SSE VERTICAL DESIGN RESPONSE SPECTRA
FOR 7 PERCENT DAMPING
STATISTICALLY INDEPENDENT

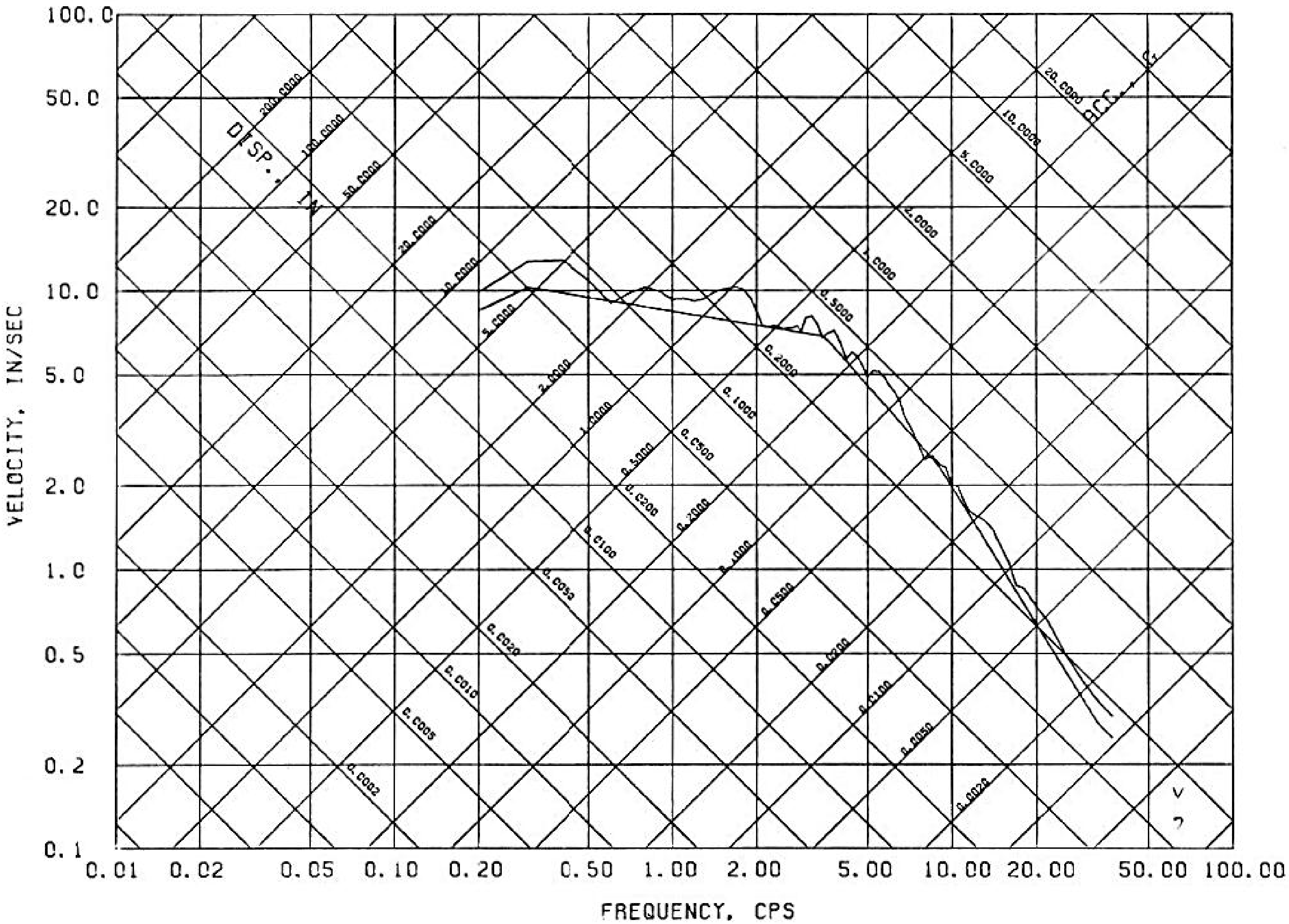


FIGURE 3.7.1-35

HORIZONTAL SIMULATED EARTHQUAKE ACCELEROGRAM
1 PERCENT DAMPING, .15 MAX G SSE
DAMS AND DIKES

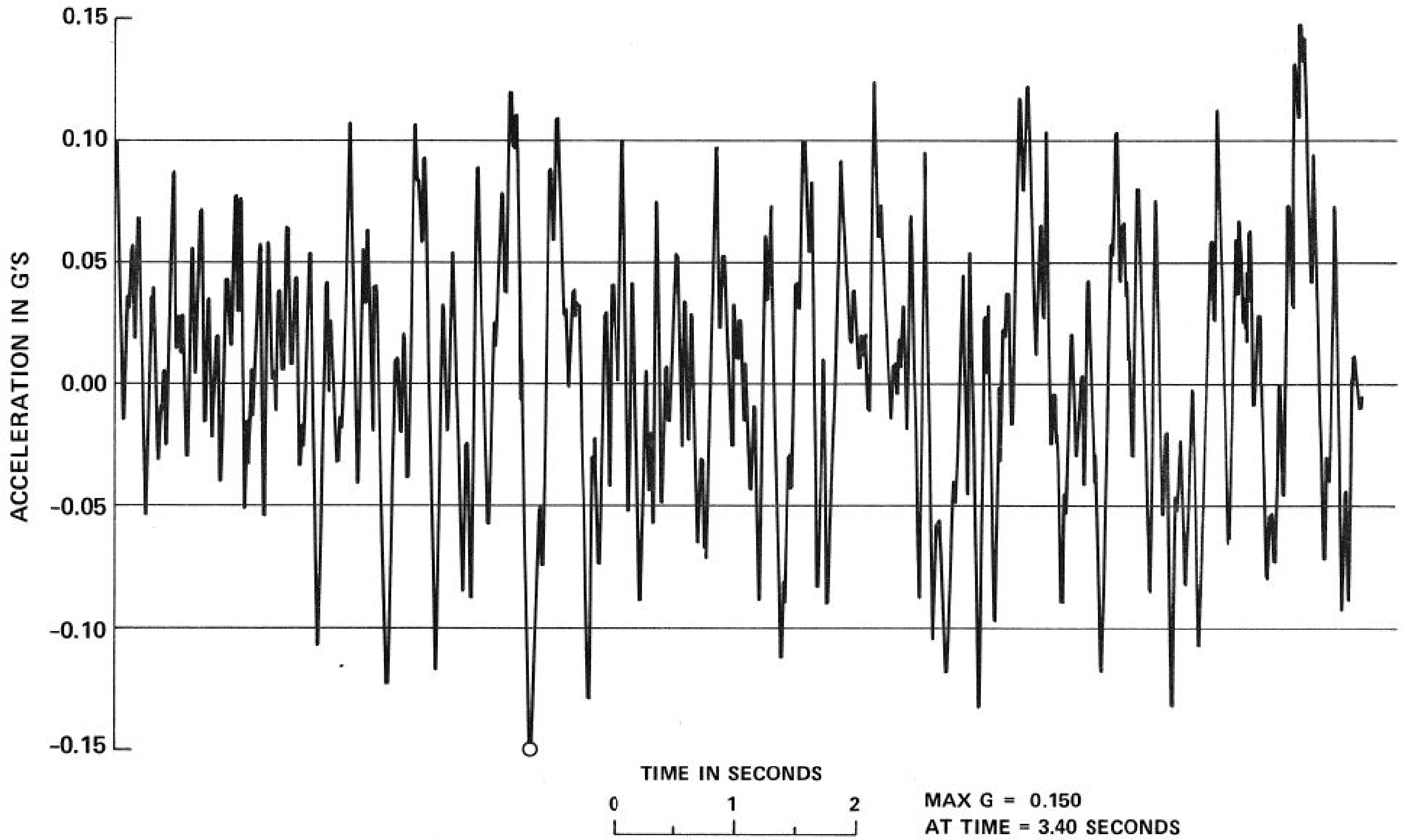


FIGURE 3.7.1-36

HORIZONTAL SIMULATED EARTHQUAKE ACCELEROGRAM
2 PERCENT DAMPING, .15 MAX G SSE
DAMS AND DIKES

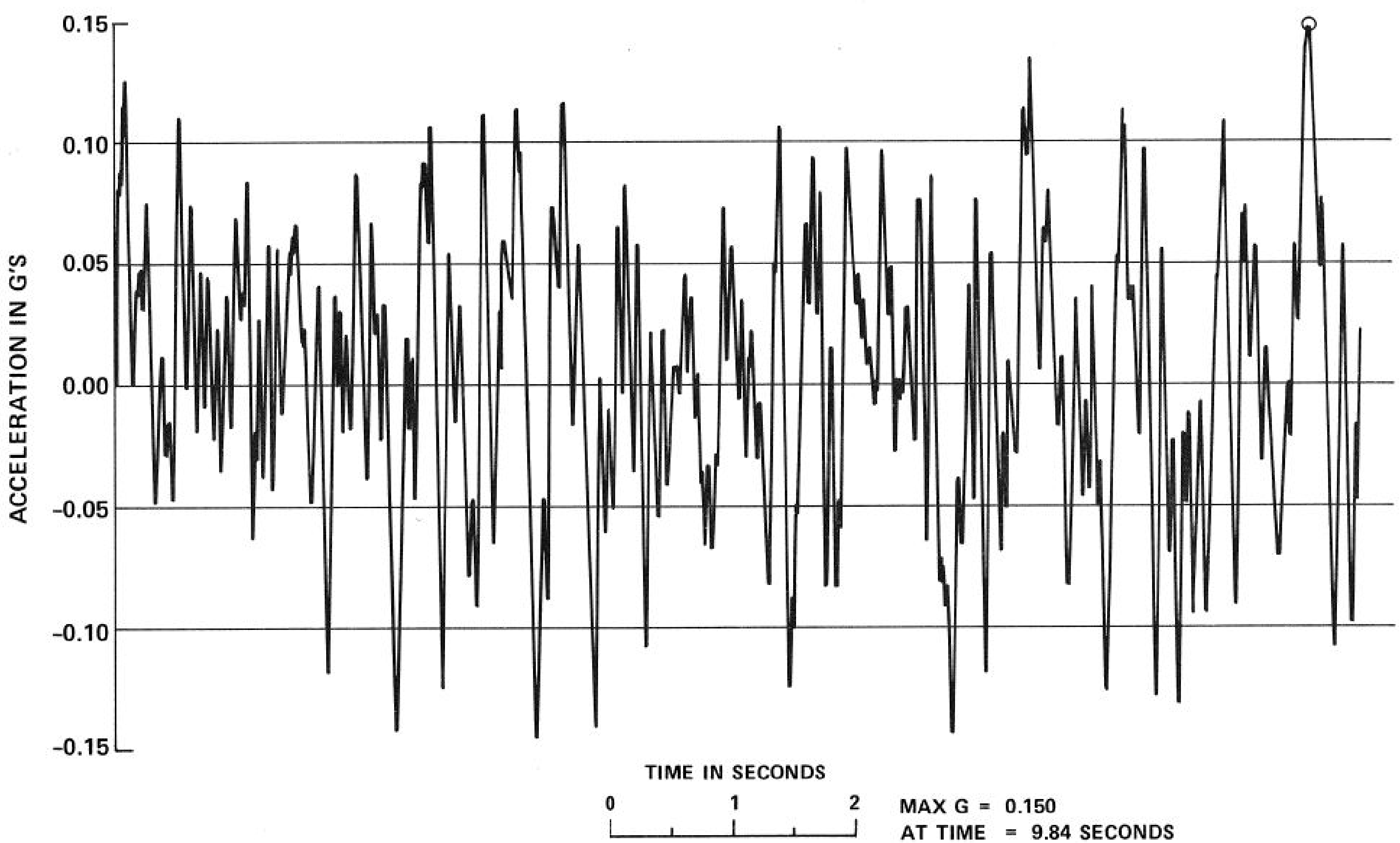


FIGURE 3.7.1-37

HORIZONTAL SIMULATED EARTHQUAKE ACCELEROGRAM
5 PERCENT DAMPING, .15 MAX G SSE
DAMS AND DIKES

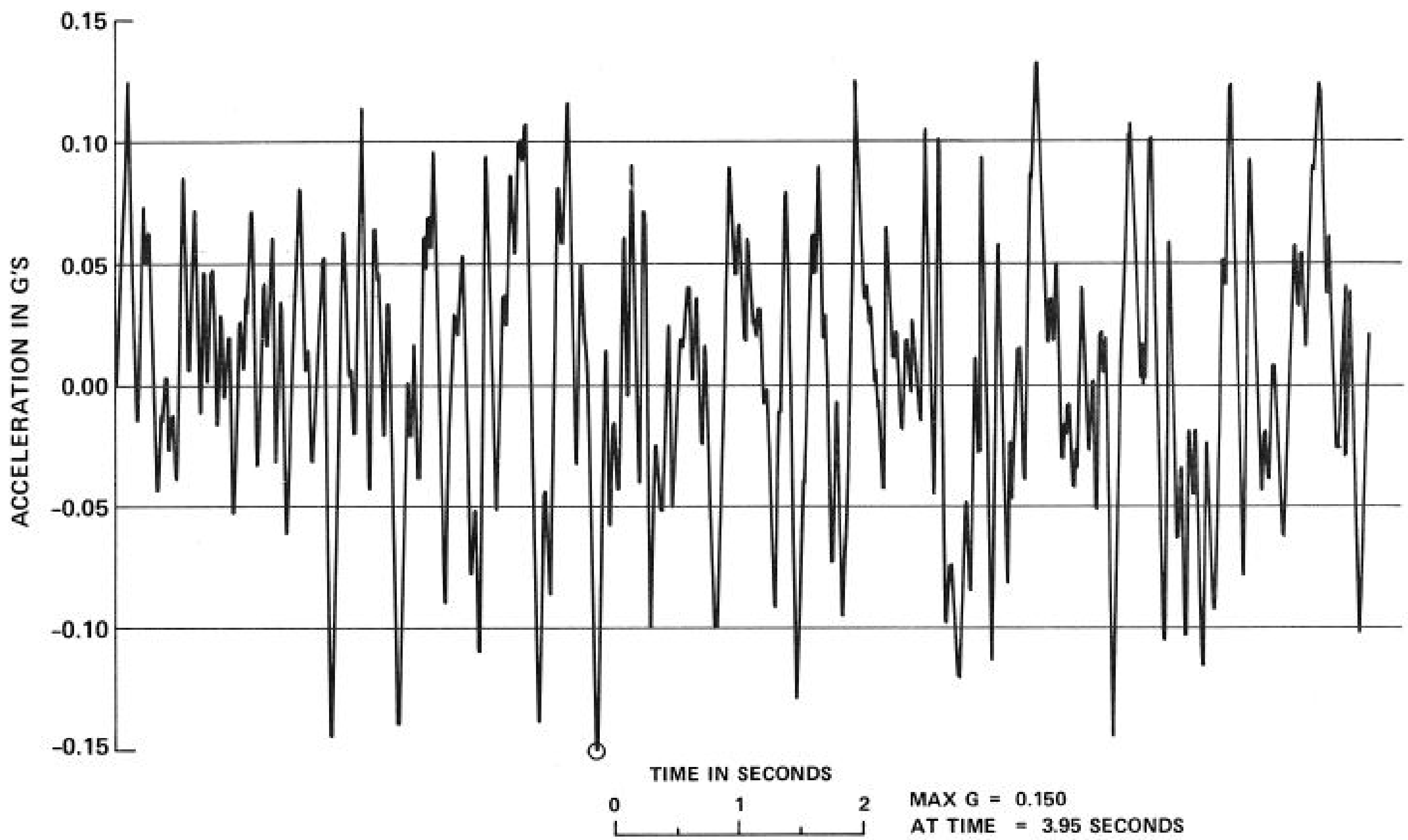


FIGURE 3.7.1-38

SSE HORIZONTAL DESIGN RESPONSE SPECTRA
1 PERCENT DAMPING, .15 MAX. G
DAMS AND DIKES

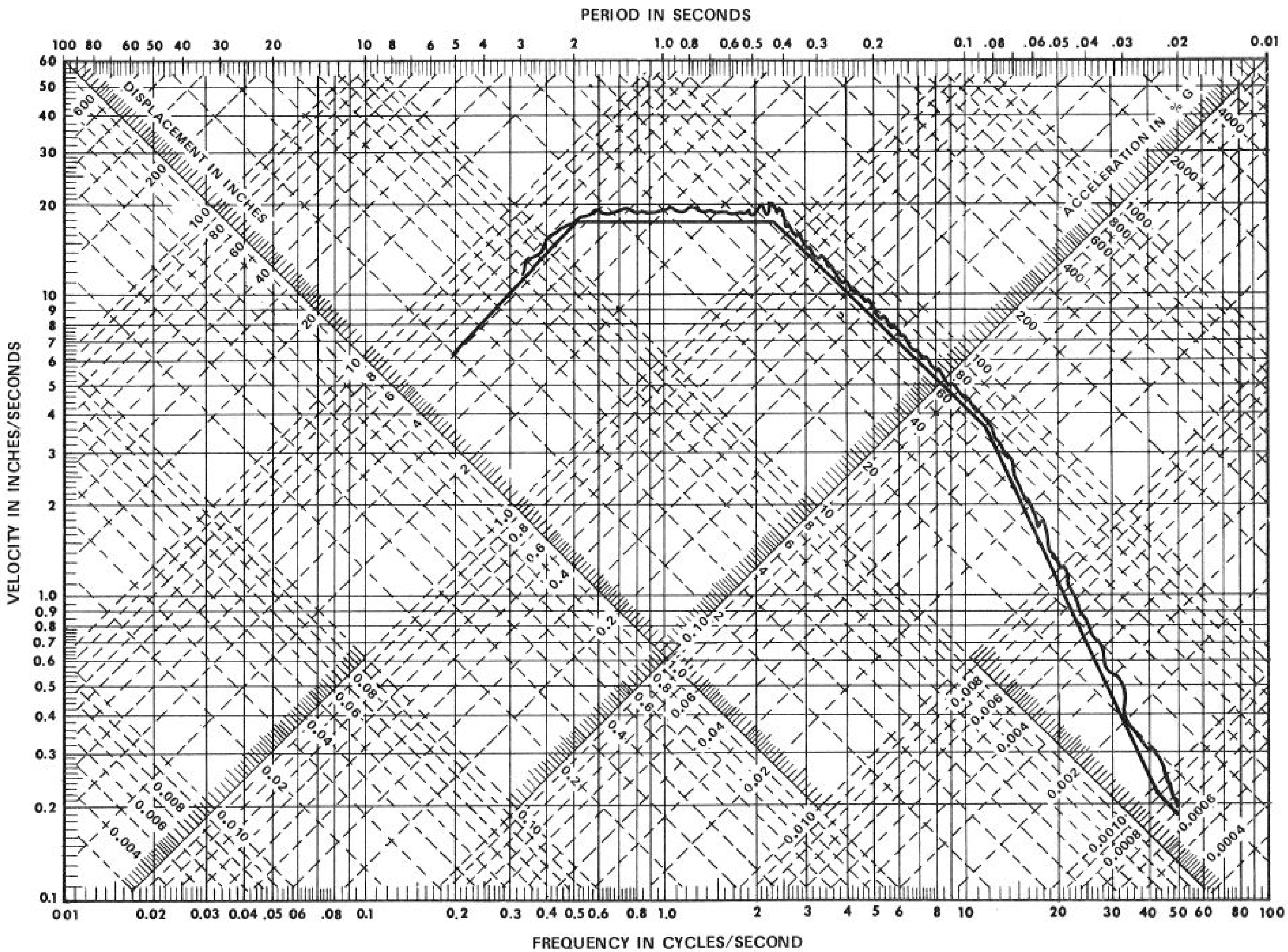


FIGURE 3.7.1-39

SSE HORIZONTAL DESIGN RESPONSE SPECTRA
2 PERCENT DAMPING, .15 MAX. G
DAMS AND DIKES

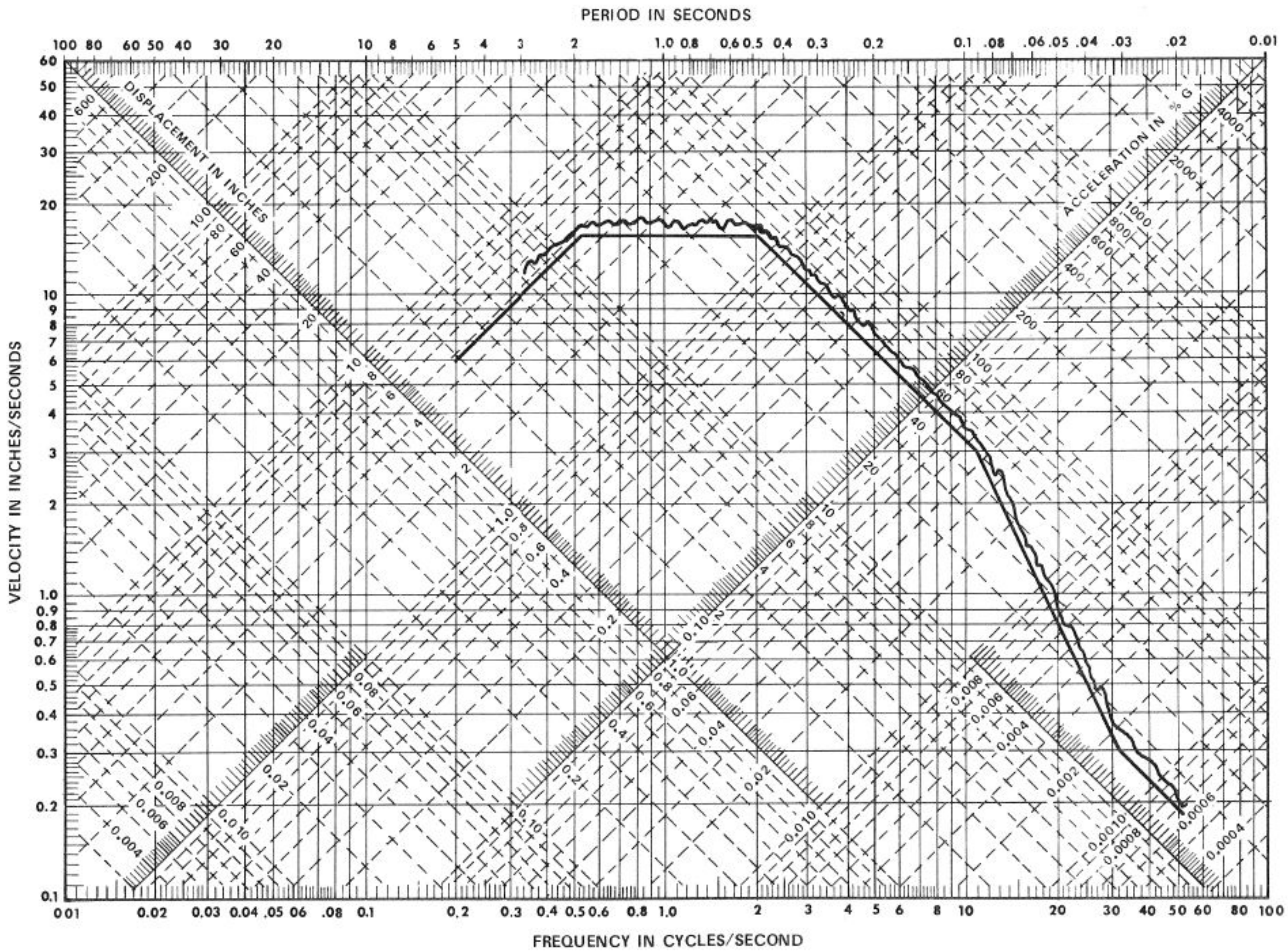


FIGURE 3.7.1-40

SSE HORIZONTAL DESIGN RESPONSE SPECTRA
5 PERCENT DAMPING, .15 MAX. G
DAMS AND DIKES

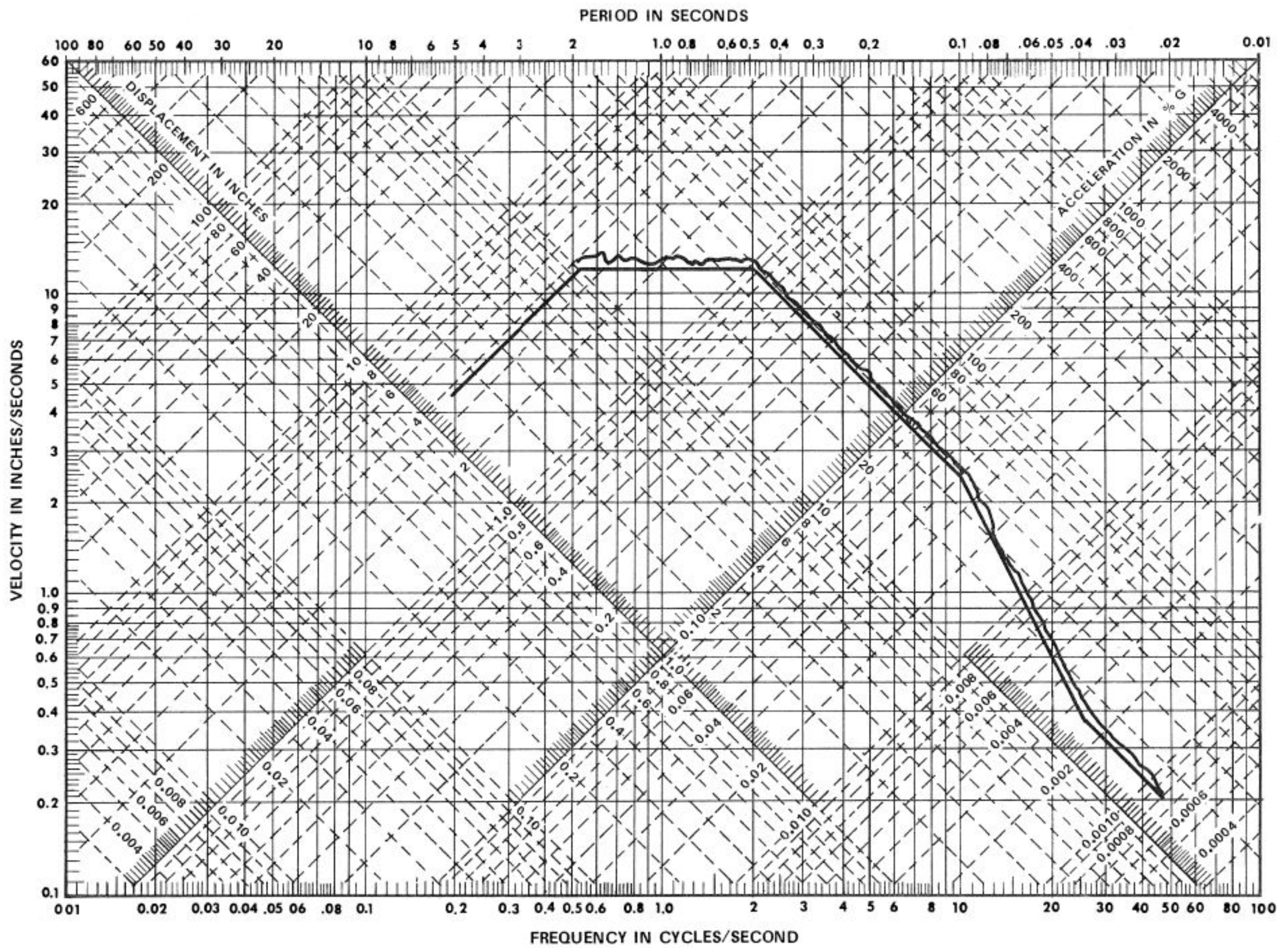


FIGURE 3.7.2-1

CONTAINMENT STRUCTURE MATHEMATICAL MODEL

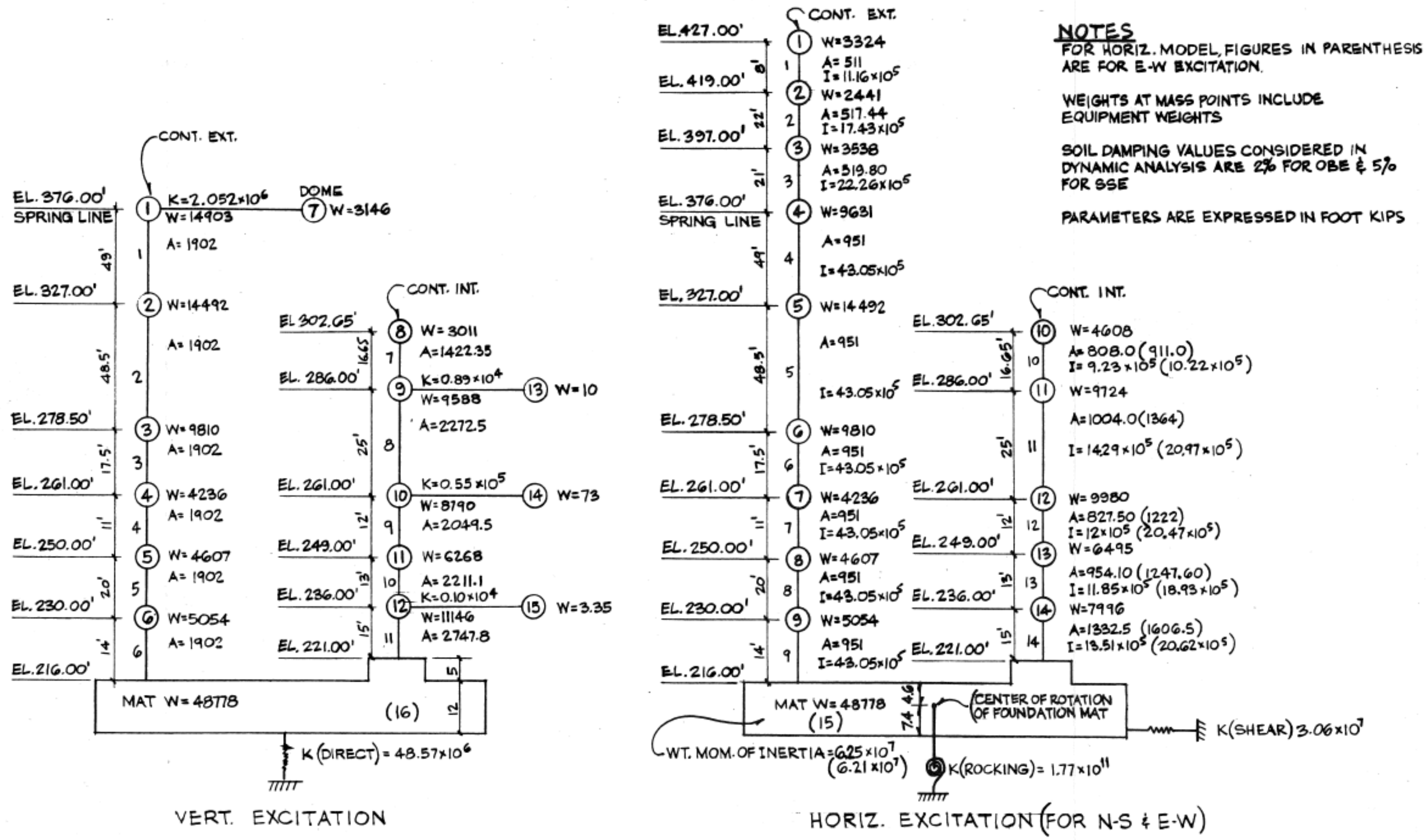


FIGURE 3.7.2-2

3D COUPLED STRUCTURE – EQUIPMENT DYNAMIC MODEL

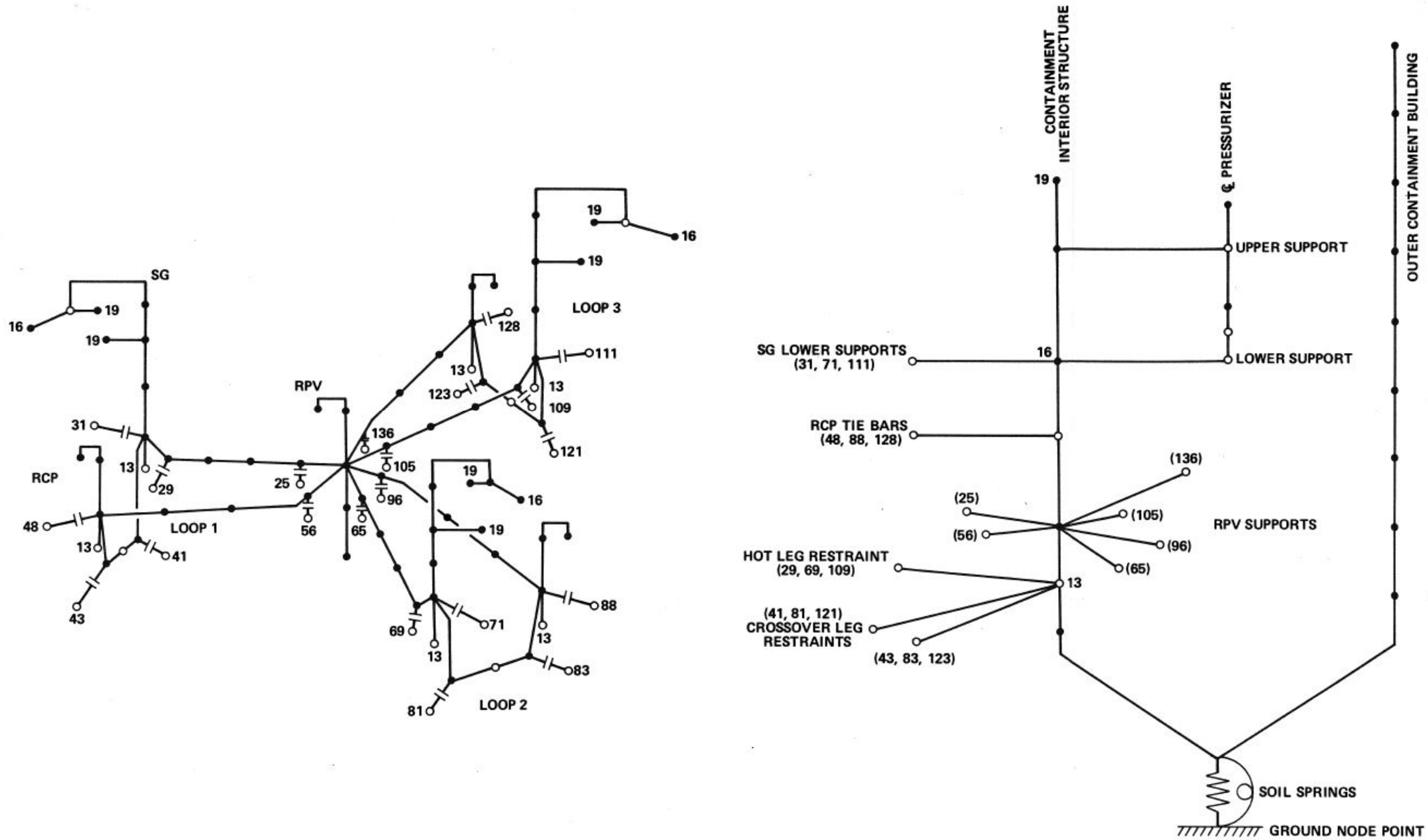


FIGURE 3.7.2-3

REACTOR AUXILIARY BUILDING MATHEMATICAL MODEL

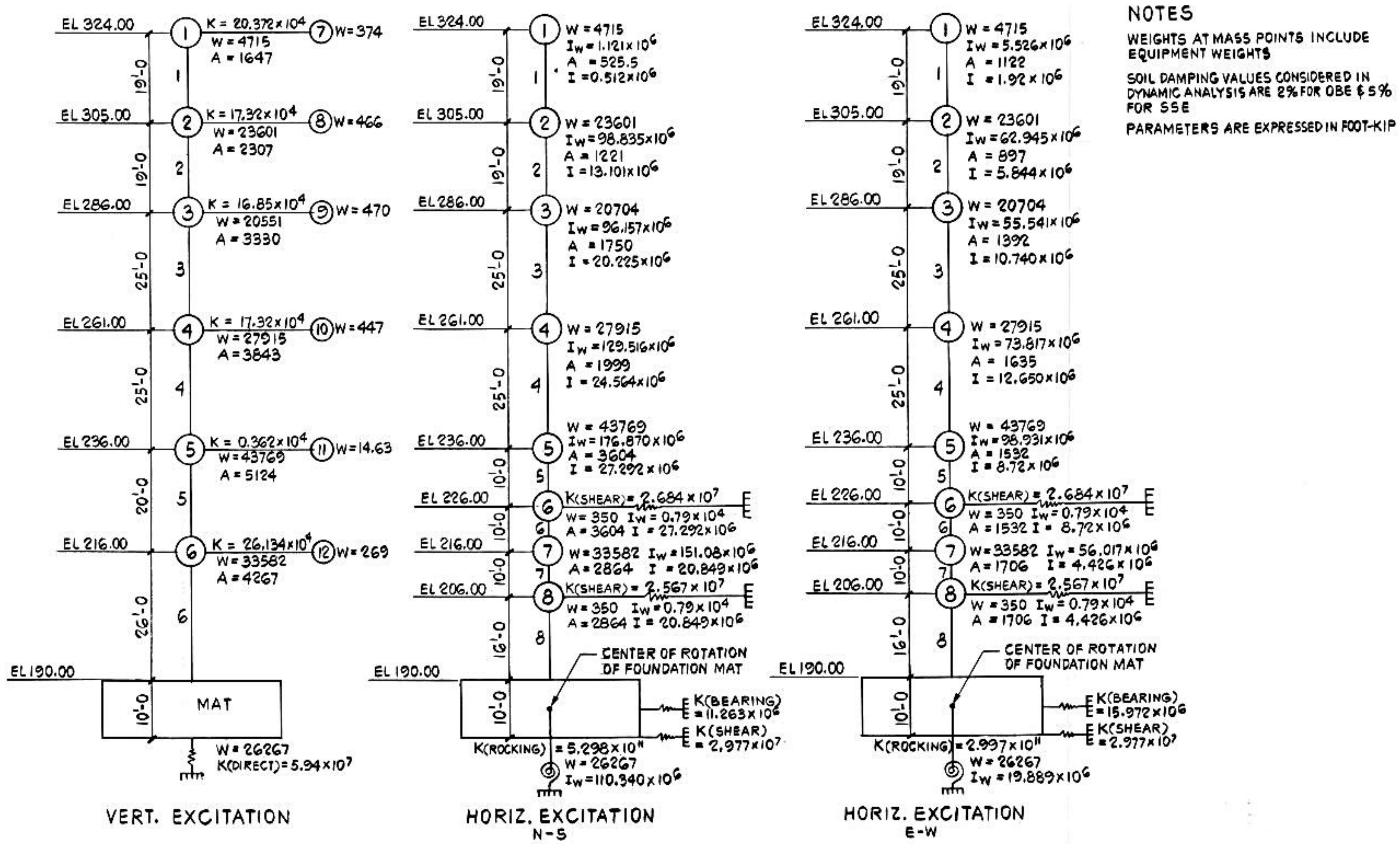


FIGURE 3.7.2-4

REACTOR AUXILIARY BUILDING MATHEMATICAL MODEL

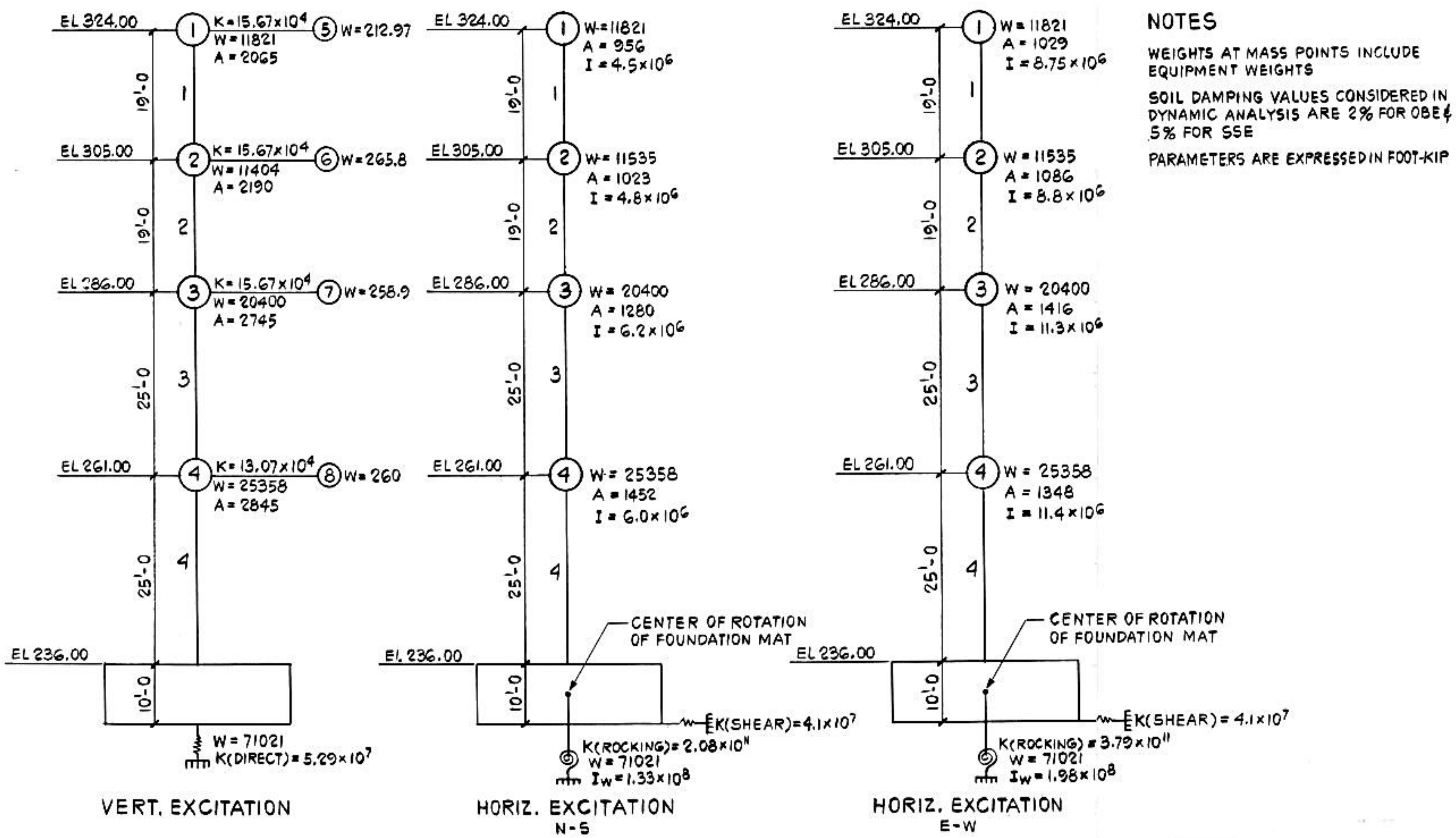


FIGURE 3.7.2-5

FUEL HANDLING BUILDING MATHEMATICAL MODELS

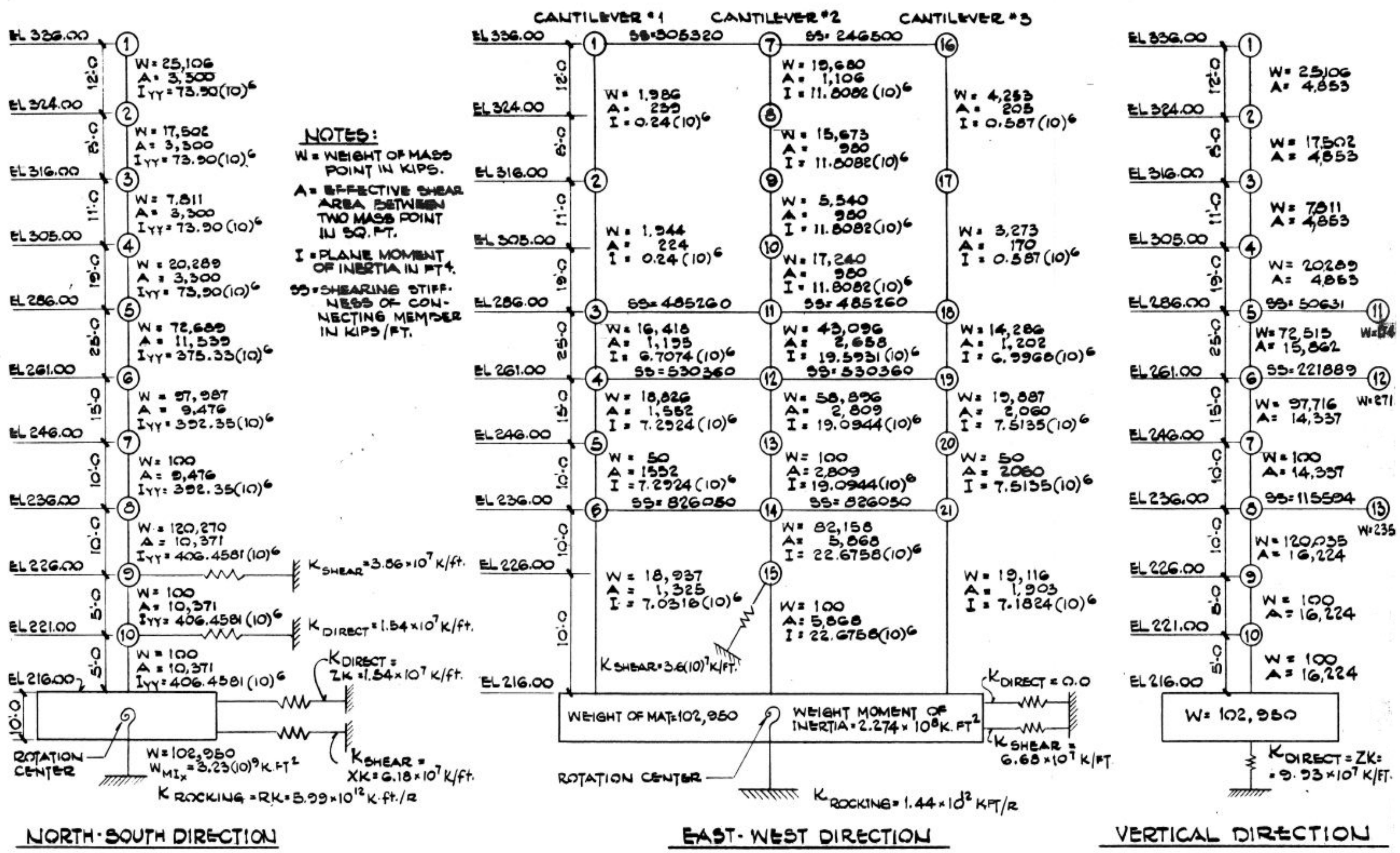


FIGURE 3.7.2-6

WASTE PROCESSING BUILDING MATHEMATICAL MODELS

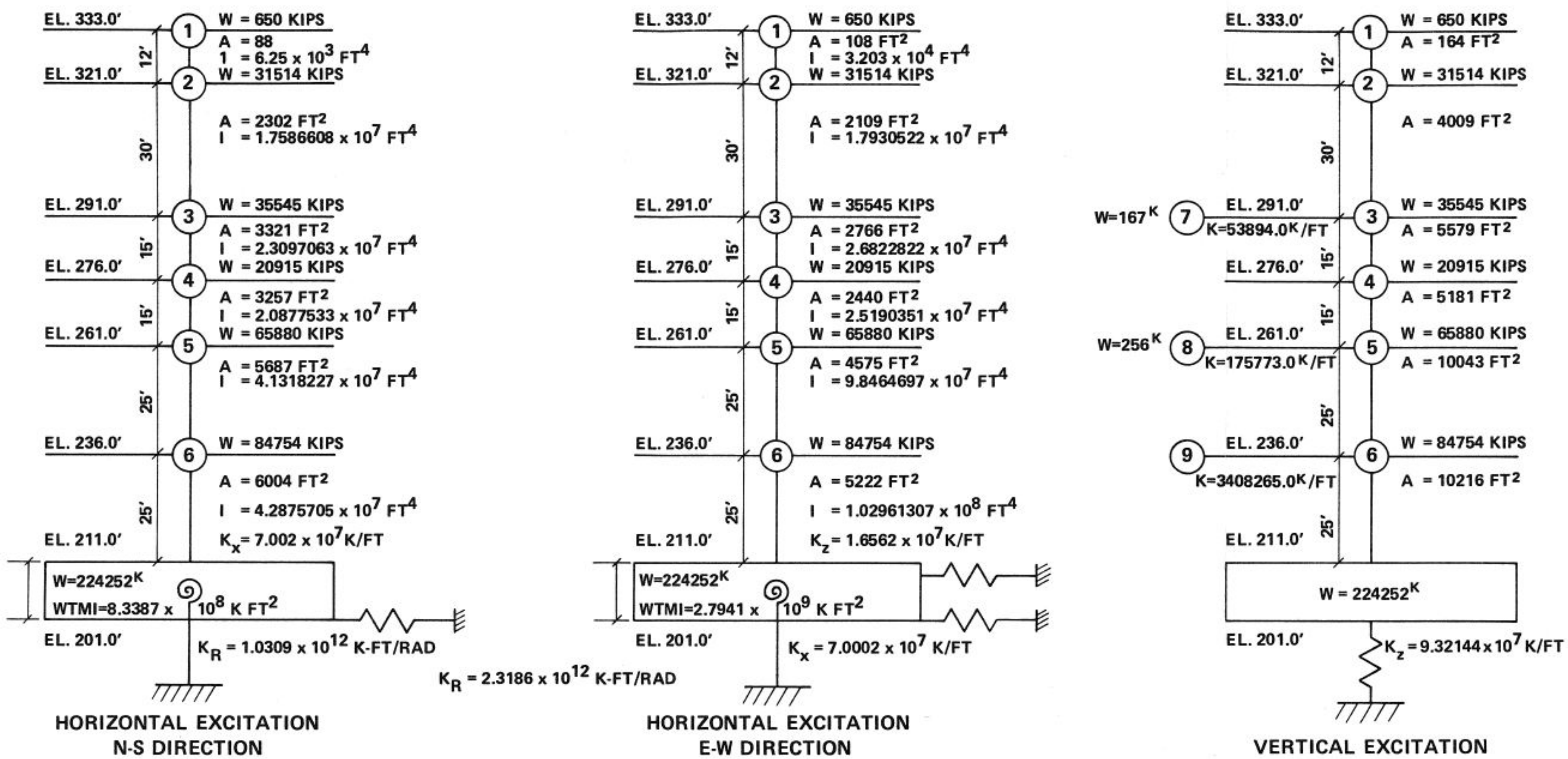


FIGURE 3.7.2-7

CONTAINMENT STRUCTURE – FLOOR RESPONSE SPECTRA
OPERATING FLOOR ELEV. 286'-0 BE
NORTH-SOUTH DIRECTION
1 PERCENT CRITICAL DAMPING

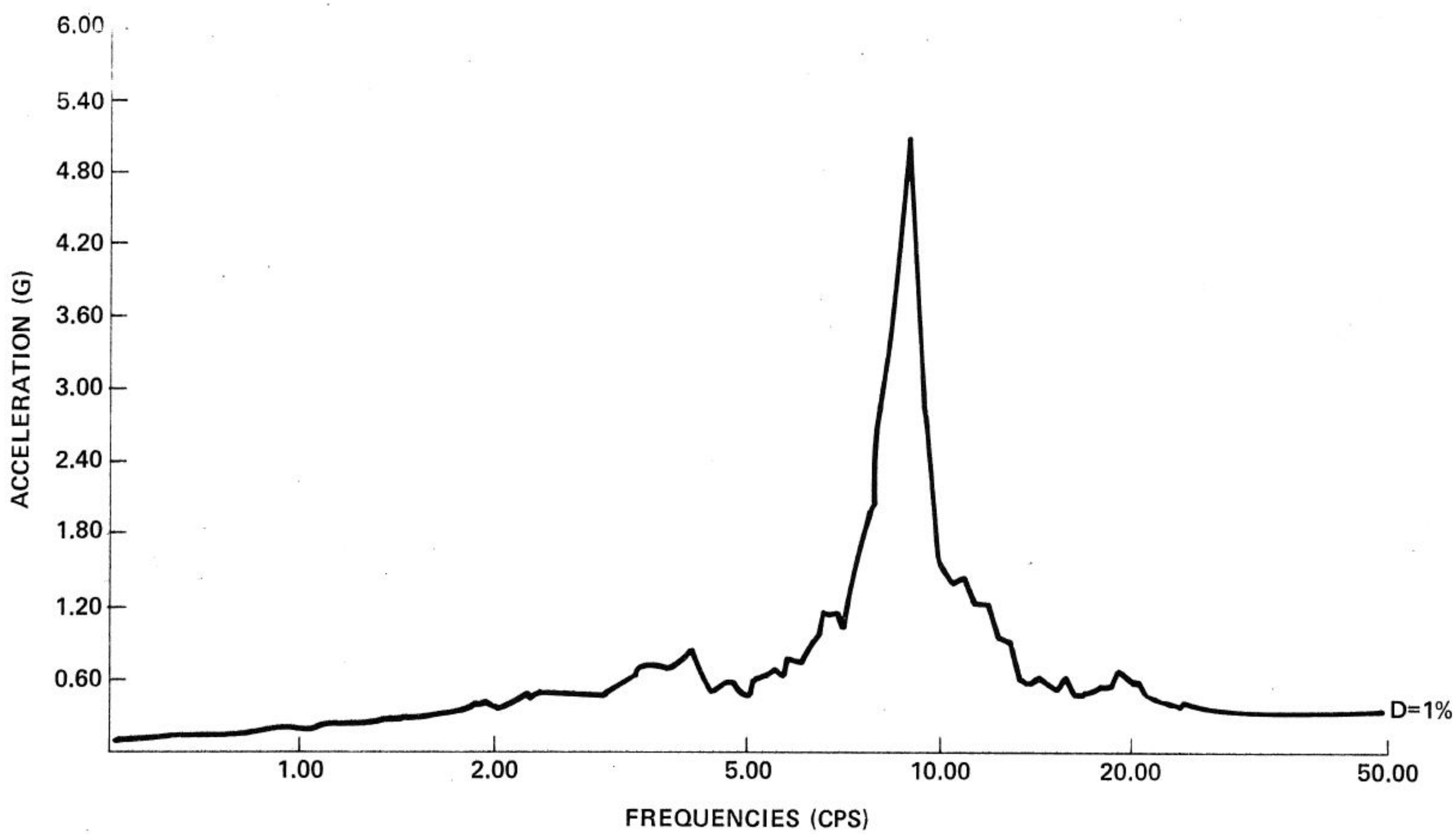


FIGURE 3.7.2-8

VERTICAL SOIL COLUMN MODELS

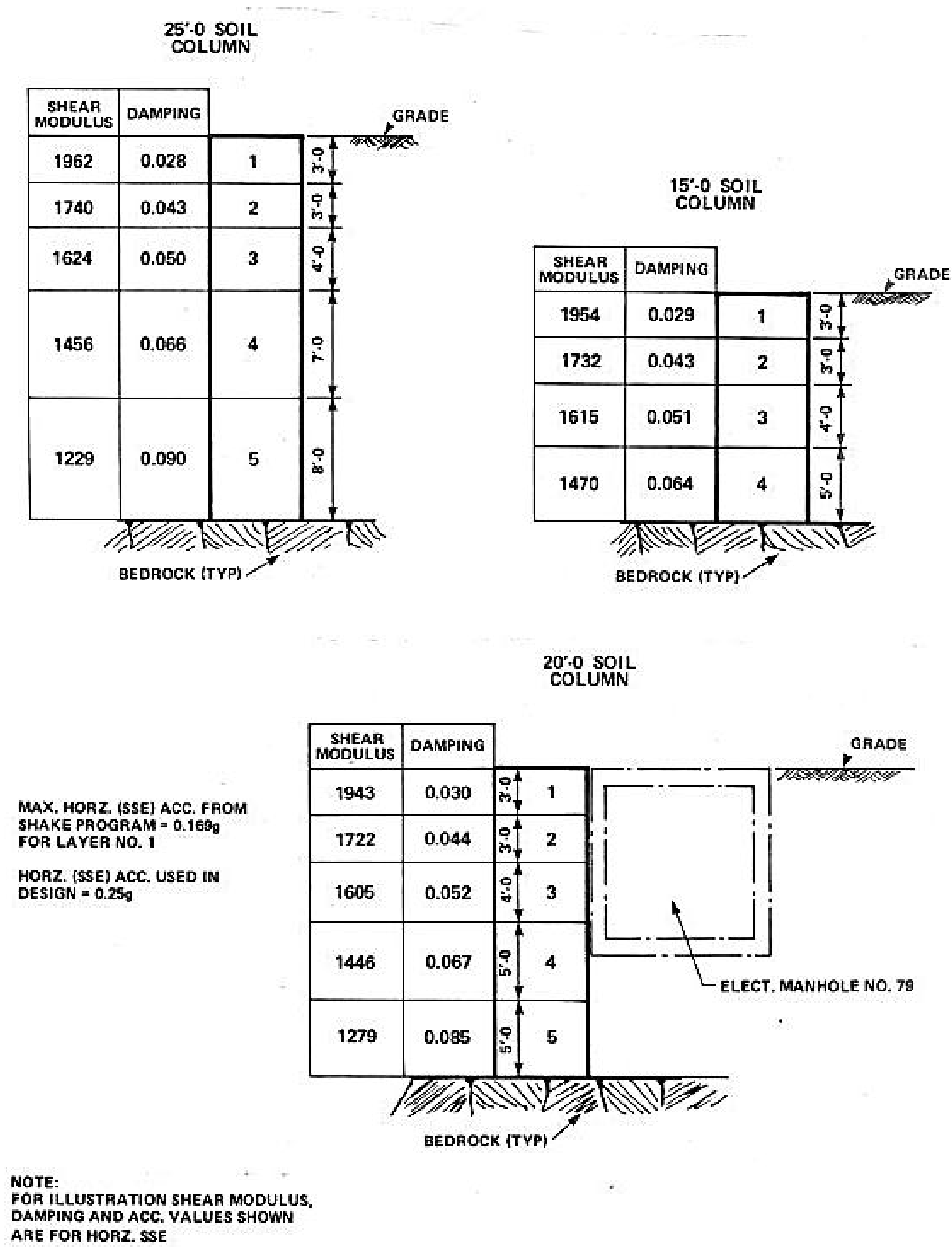
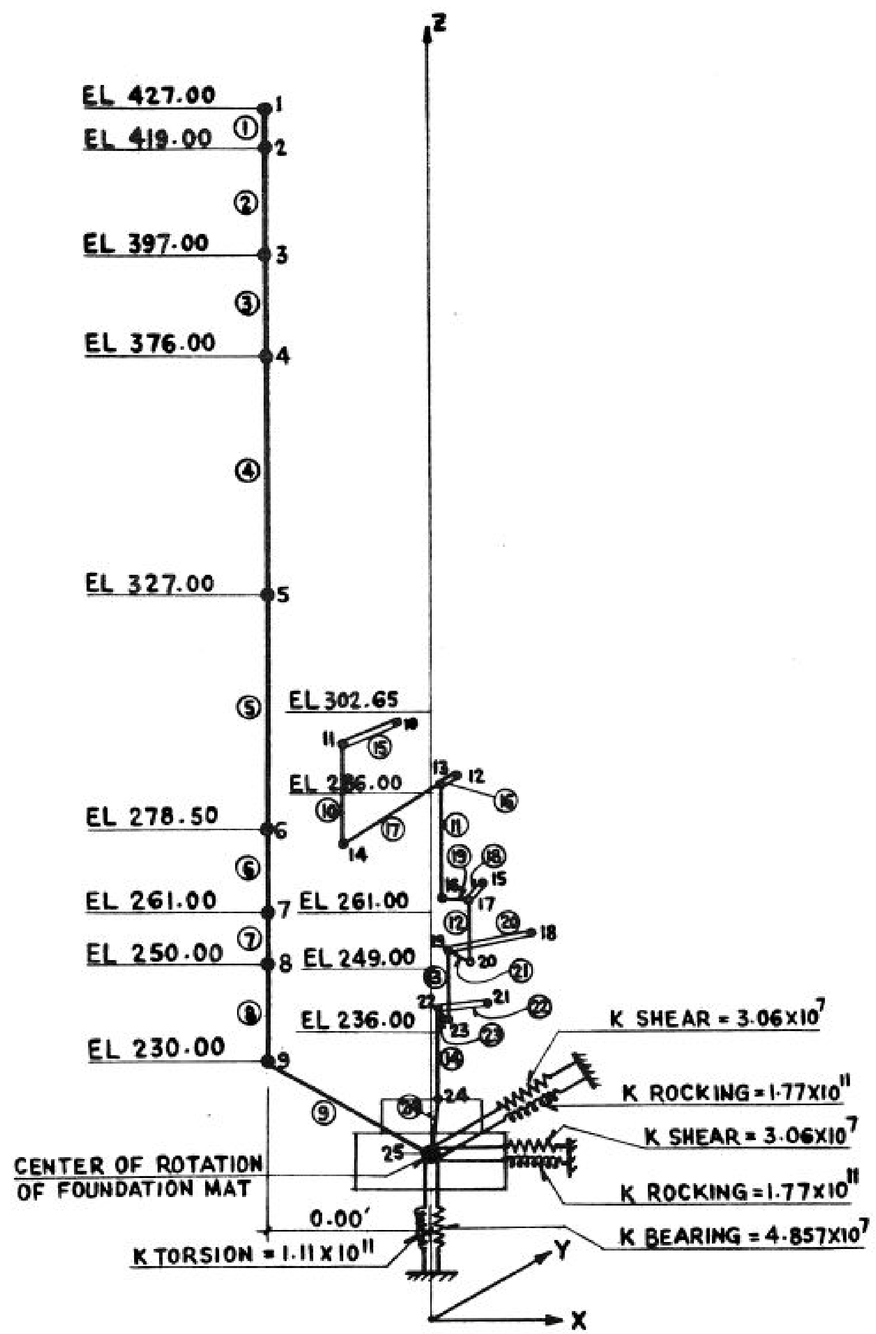


FIGURE 3.7.2-9

CONTAINMENT STRUCTURE 3D TORSIONAL MODEL



3D MODEL

JOINT PROPERTIES

NODE	COORDINATE (FT) X	COORDINATE (FT) Y	COORDINATE (FT) Z	WT. OF LUMPED MASS (KIPS)	WT. MOMENT OF INERTIA Jz (KIP-FT²)
1	0.00	0.00	427.0	3324	4.90×10^5
2	0.00	0.00	419.0	2441	8.00×10^5
3	0.00	0.00	397.0	3538	1.16×10^6
4	0.00	0.00	376.0	9631	3.682×10^7
5	0.00	0.00	327.0	14492	6.562×10^7
6	0.00	0.00	278.5	9810	4.442×10^7
7	0.00	0.00	261.0	4236	1.918×10^7
8	0.00	0.00	250.0	4607	2.086×10^7
9	0.00	0.00	230.0	5054	2.288×10^7
10	-2.59	-4.28	302.65	4608	5.28×10^6
11	-5.72	-12.75	302.65		
12	-0.11	1.08	286.0	9724	1.30×10^7
13	-0.11	0.35	286.0		
14	-5.72	-12.75	286.0		
15	-0.03	4.65	261.0	9980	1.312×10^7
16	-0.11	0.35	261.0		
17	0.06	4.17	261.0		
18	-0.66	9.18	249.0	6495	6.79×10^6
19	-0.67	4.35	249.0		
20	0.06	4.17	249.0		
21	-0.28	5.92	236.0	7996	9.62×10^6
22	-1.15	4.87	236.0		
23	-0.67	4.35	236.0		
24	-1.15	4.87	221.0		
25	0.00	0.00	211.4	48778	1.22×10^8

MEMBER PROPERTIES

MEMBER	Ax (FT²)	Ay (FT²)	Az (FT²)	Ix (FT4)	Iy (FT4)	Ip (FT4)
1	511.0	511.0	1022.0	11.16×10^5	11.16×10^5	22.32×10^5
2	517.44	517.44	1034.88	17.43×10^5	17.43×10^5	34.86×10^5
3	951.0	951.0	1902.0	22.26×10^5	22.26×10^5	44.52×10^5
4	951.0	951.0	1902.0	43.05×10^5	43.05×10^5	86.10×10^5
5	951.0	951.0	1902.0	43.05×10^5	43.05×10^5	86.10×10^5
6	951.0	951.0	1902.0	43.05×10^5	43.05×10^5	86.10×10^5
7	951.0	951.0	1902.0	43.05×10^5	43.05×10^5	86.10×10^5
8	951.0	951.0	1902.0	43.05×10^5	43.05×10^5	86.10×10^5
9	951.0	951.0	1902.0	43.05×10^5	43.05×10^5	86.10×10^5
10	808.0	911.0	1422.35	9.86×10^5	9.12×10^5	18.98×10^5
11	1004.0	1364.0	2272.5	1.98×10^6	1.43×10^6	3.41×10^6
12	827.5	1222.0	2049.5	1.86×10^6	1.21×10^6	3.07×10^6
13	954.1	1247.6	2211.1	1.89×10^6	1.22×10^6	3.11×10^6
14	1332.5	1606.5	2747.8	2.03×10^6	1.27×10^6	3.30×10^6

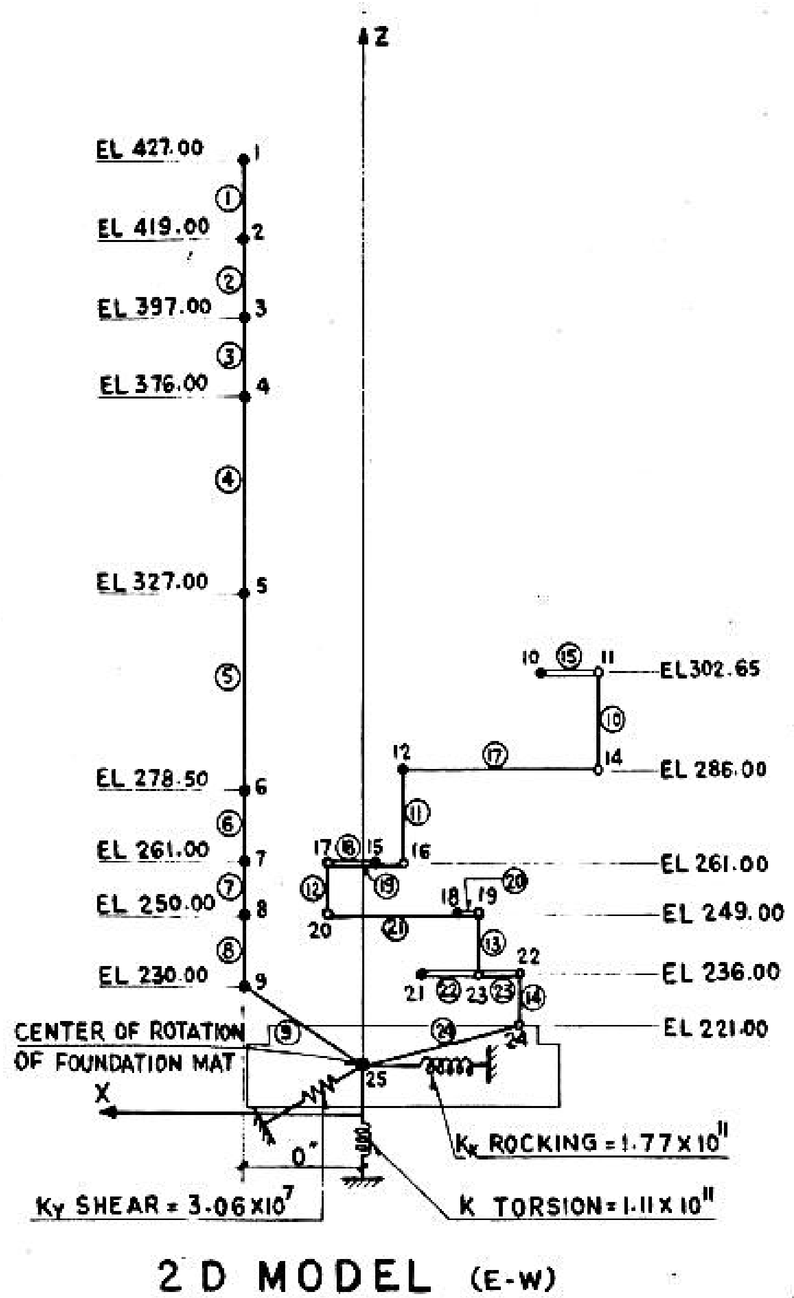
NOTES:

X - AXIS = N-S DIRECTION.
Y - AXIS = E-W DIRECTION
Z - AXIS = VERT. DIRECTION

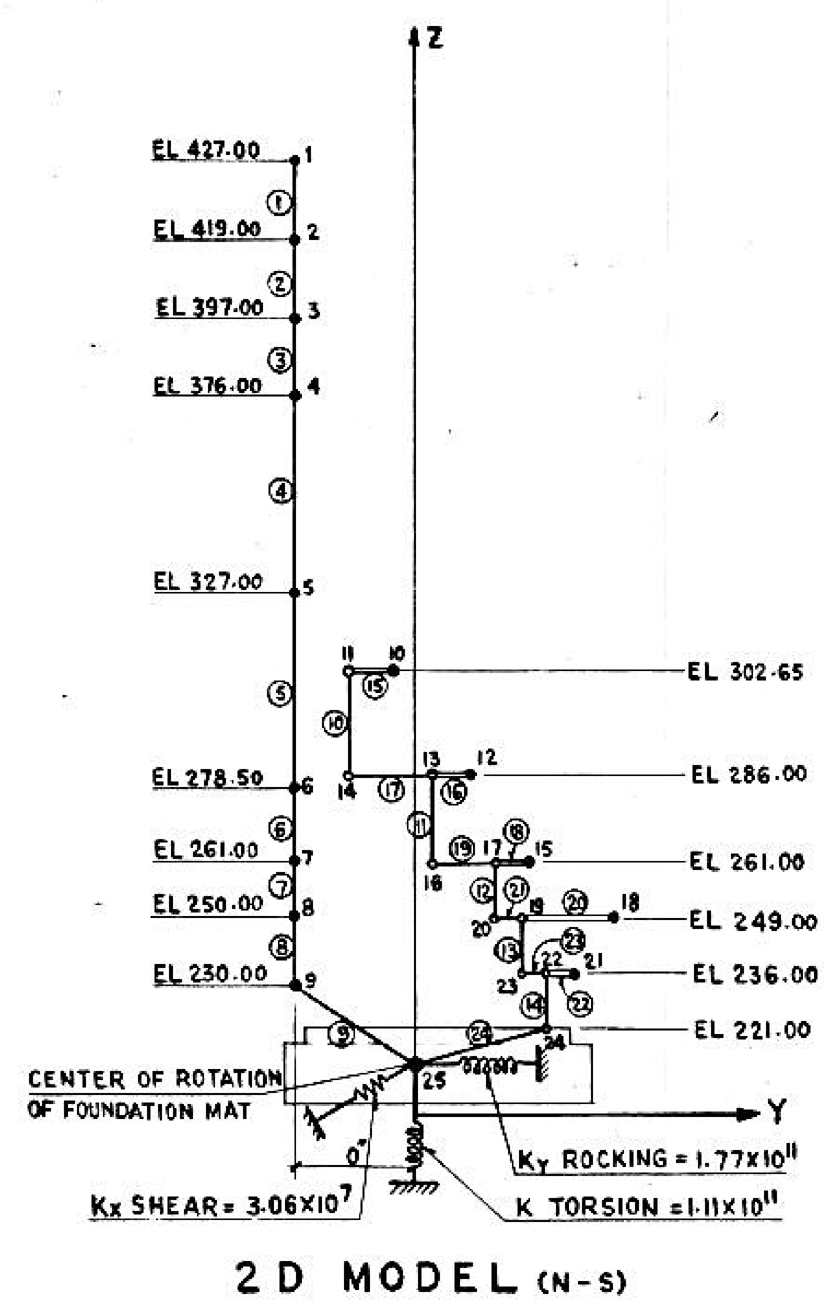
WEIGHT OF MASSES & WEIGHT MOMENTS OF INERTIA INCLUDE EQUIPMENT WEIGHT.

FIGURE 3.7.2-10

CONTAINMENT STRUCTURE 2D TORSIONAL MODELS



NODE	COORD. (FT) X	COORD. (FT) Z
1	0.00	427.0
2	0.00	419.0
3	0.00	397.0
4	0.00	376.0
5	0.00	327.0
6	0.00	278.5
7	0.00	261.0
8	0.00	250.0
9	0.00	230.0
10	-2.59	302.65
11	-5.72	302.65
12	-0.11	286.0
13		
14	-5.72	286.0
15	-0.03	261.0
16	-0.11	261.0
17	0.06	261.0
18	-0.66	249.0
19	-0.67	249.0
20	0.06	249.0
21	-0.28	236.0
22	-1.15	236.0
23	-0.67	236.0
24	-1.15	221.0
25	0.00	211.4

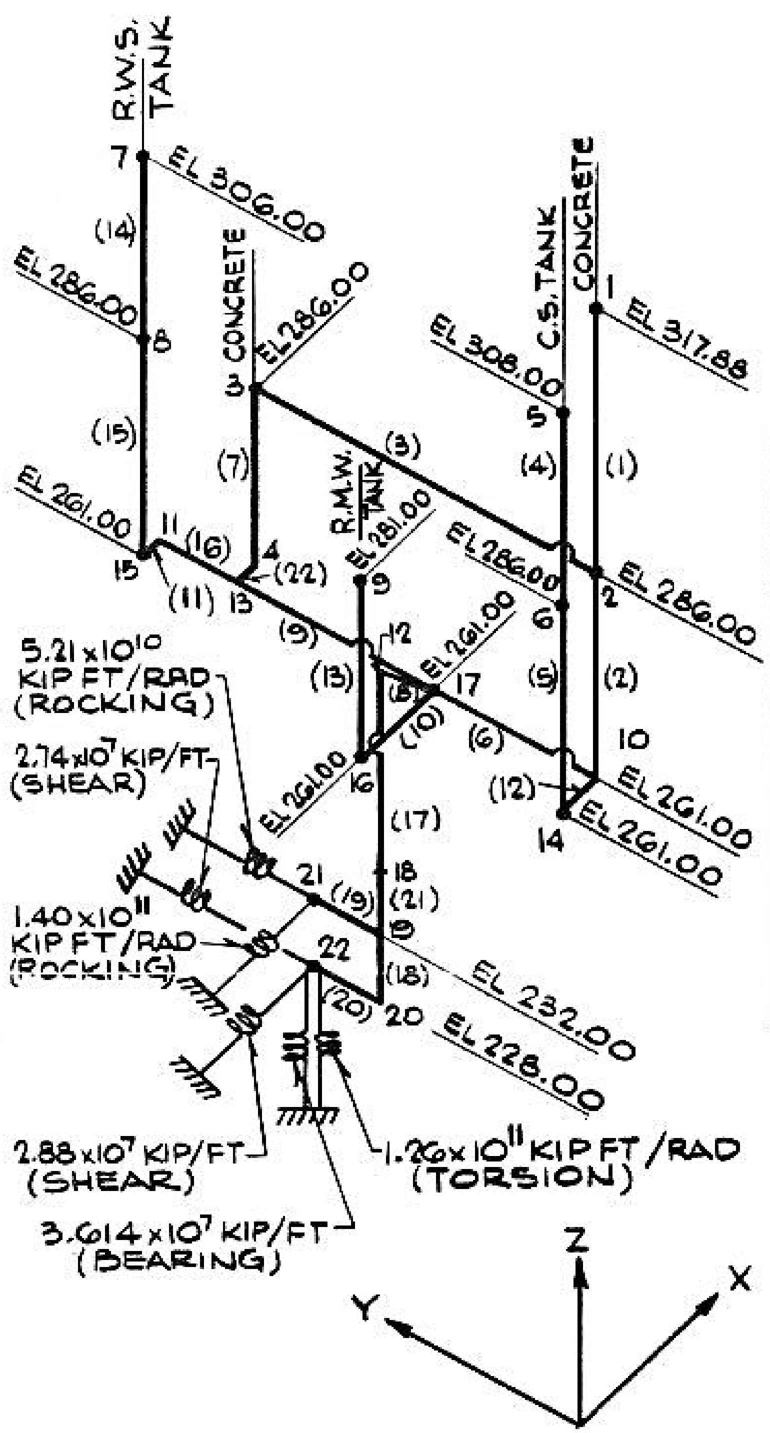


NODE	COORD. (FT) Y	COORD. (FT) Z
1	0.00	427.0
2	0.00	419.0
3	0.00	397.0
4	0.00	376.0
5	0.00	327.0
6	0.00	278.5
7	0.00	261.0
8	0.00	250.0
9	0.00	230.0
10	-4.28	302.65
11	-12.75	302.65
12	1.08	286.0
13	0.35	286.0
14	-12.75	286.0
15	4.65	261.0
16	0.35	261.0
17	4.17	261.0
18	9.18	249.0
19	4.35	249.0
20	4.17	249.0
21	5.92	236.0
22	4.87	236.0
23	4.35	236.0
24	4.87	221.0
25	0.00	211.4

NOTES:-
X-AXIS = N-S DIRECTION
Y-AXIS = E-W DIRECTION
Z-AXIS = VERTICAL DIRECTION
WEIGHT OF MASSES & WEIGHT MOMENTS OF INERTIA INCLUDE EQUIPMENT WEIGHT.

FIGURE 3.7.2-11

TANK BUILDING 3-D DYNAMIC/TORSIONAL MODEL



JOINT PROPERTIES

NODE	COORDINATE (FT) X	Y	Z	WT OF LUMPED MASS (KIPS)	WT. MOMENT OF INERTIA (KIP-FT2) Jx	Jy	Jz
1	41.33	26.05	94.88	2780	1.12×10^6	1.509×10^6	2.322×10^6
2	41.33	26.05	63.00	2881	1495×10^6	1.867×10^6	2.930×10^6
3	45.32	96.63	63.00	2093	1.810×10^6	1.324×10^6	2.936×10^6
4	45.32	96.63	38.00	–	–	–	–
5	36.00	27.00	85.00	893	1.25×10^5	1.25×10^5	1.78×10^5
6	36.00	27.00	63.00	1878	2.74×10^5	2.74×10^5	3.76×10^5
7	40.00	113.75	83.00	1030	1.65×10^5	1.65×10^5	2.61×10^5
8	40.00	113.75	63.00	2275	3.18×10^5	3.18×10^5	5.76×10^5
9	30.00	69.50	58.00	380	299×10^4	299×10^4	3.46×10^4
10	41.33	26.05	38.00	–	–	–	–
11	41.33	113.75	38.00	–	–	–	–
12	41.20	79.91	38.00	–	–	–	–
13	41.33	96.63	38.00	–	–	–	–
14	36.00	27.00	38.00	1018	1.55×10^5	1.55×10^5	2.036×10^5
15	40.00	113.75	38.00	1290	2.30×10^5	2.30×10^5	3.27×10^5
16	30.00	69.50	38.00	380	2.99×10^4	2.99×10^4	3.46×10^4
17	41.33	70.46	38.00	10164	2.014×10^7	5.585×10^6	2.493×10^7
18	41.20	79.91	13.00	–	–	–	–
19	41.20	79.91	9.00	–	–	–	–
20	41.20	79.91	5.00	–	–	–	–
21	39.63	80.44	9.00	12031	2.285×10^7	6.933×10^6	2.777×10^7
22	39.63	80.44	5.00	6016	1.143×10^7	3.467×10^6	1.388×10^7

MEMBER PROPERTIES

MEMBER	Ax (FT2)	Ay (FT2)	Az (FT2)	Iy (FT4)	Iz (FT4)	Ip (FT4)
1	383.00	383.00	317.90	3.78×10^5	2.875×10^5	6.785×10^5
2	383.00	383.00	317.90	3.78×10^5	2.875×10^5	6.785×10^5
4	14.22	14.22	14.22	6.233×10^3	6.233×10^3	1.252×10^4
5	23.69	23.69	23.69	1.035×10^4	1.035×10^4	2.085×10^4
6	191.00	191.00	191.00	1.715×10^5	4.621×10^5	6.336×10^5
7	382.00	382.00	538.62	5.875×10^5	6.585×10^5	1.547×10^6
9	191.00	191.00	191.00	1.715×10^5	4.621×10^5	6.336×10^5
13	8.80	8.80	8.80	1.765×10^3	1.765×10^3	3.525×10^3
14	16.00	16.00	16.00	8.863×10^3	8.863×10^3	1.778×10^4
15	26.65	26.65	26.65	1.476×10^4	1.476×10^4	2.968×10^4
16	191.00	191.00	191.00	1.715×10^5	4.621×10^5	6.336×10^5
17	738.00	738.00	1092.24	1.126×10^6	3.790×10^6	5.07×10^6
3	INFIN	ITELY	RIGID	MEM	BER	
8	"		"	"		
10	"		"	"		
11	"		"	"		
12	"		"	"		
18	"		"	"		
19	"		"	"		
20	"		"	"		
21	"		"	"		
22	"		"	"		

FIGURE 3.7.2-12

MULTIDEGREE OF FREEDOM SYSTEM

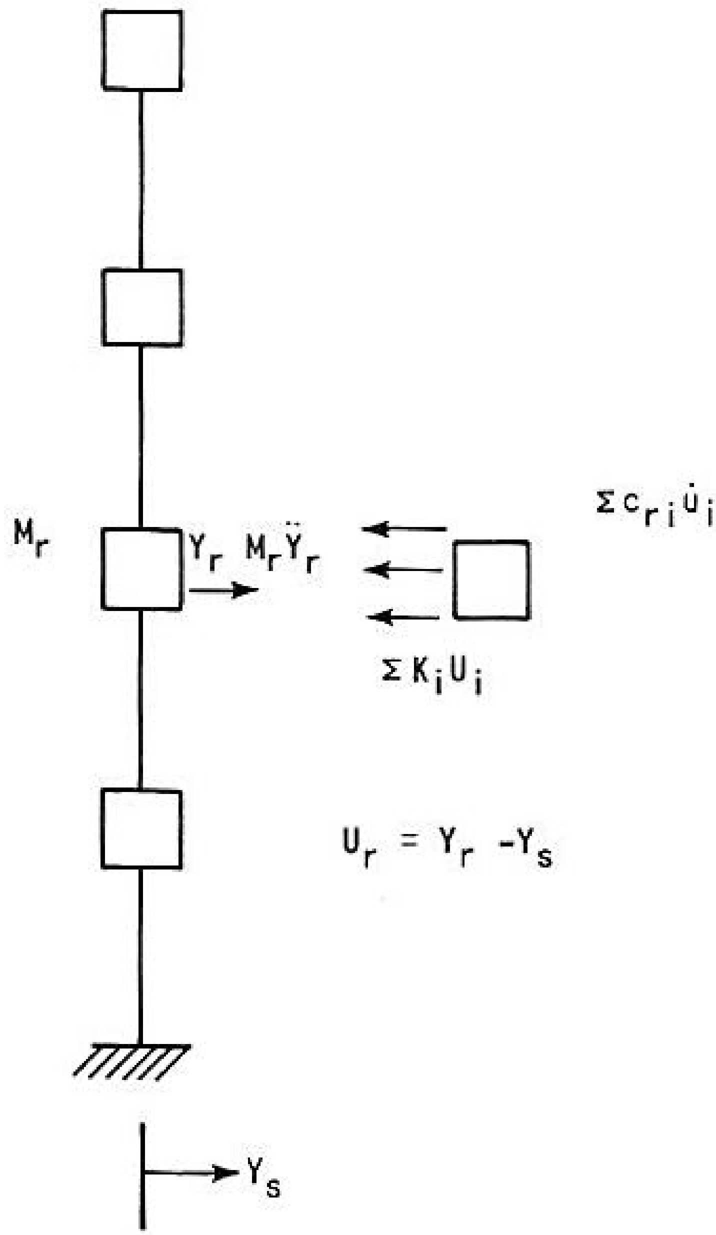


FIGURE 3.7.2-13

SCREEN STRUCTURE MATHEMATICAL MODEL

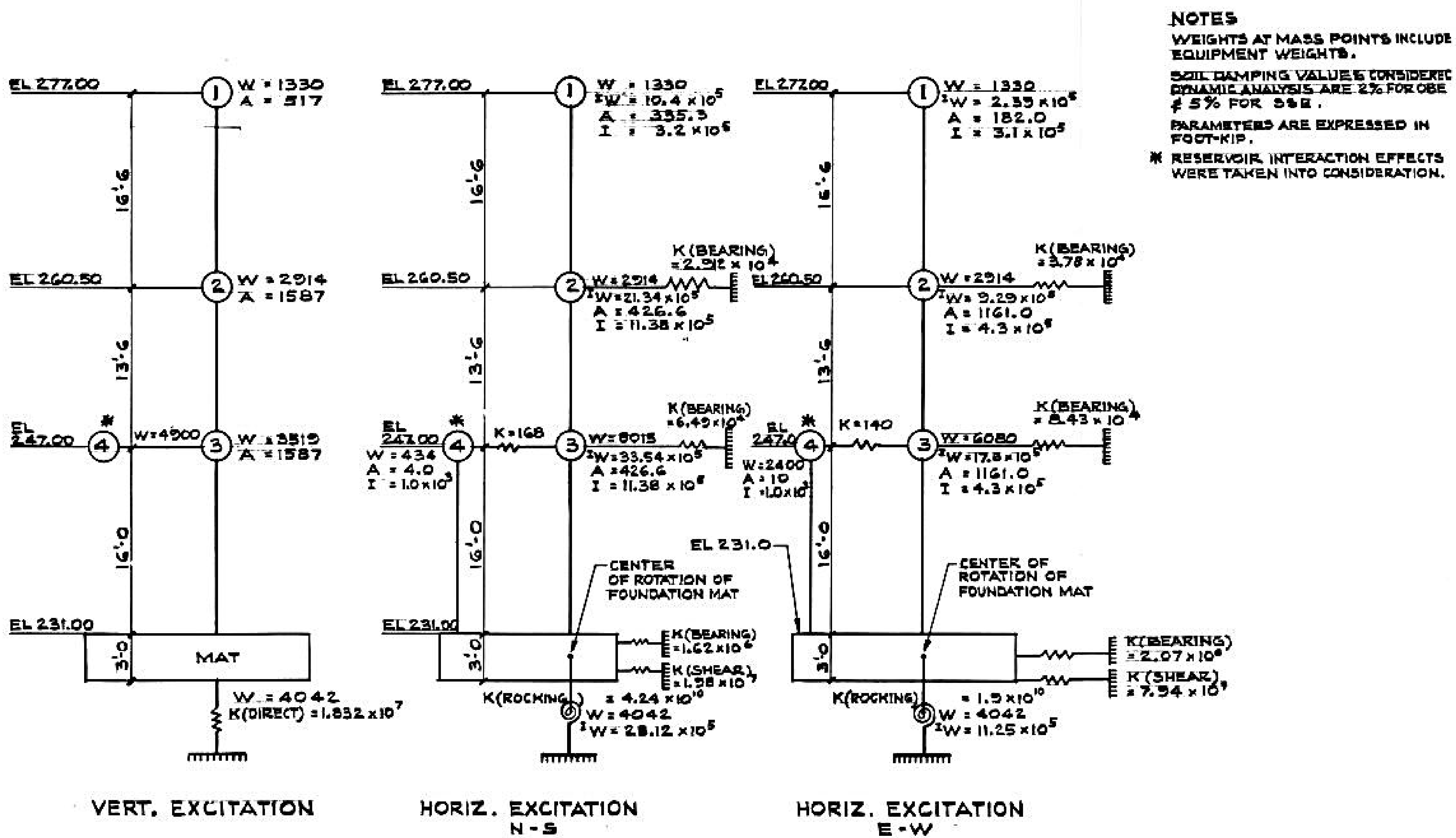
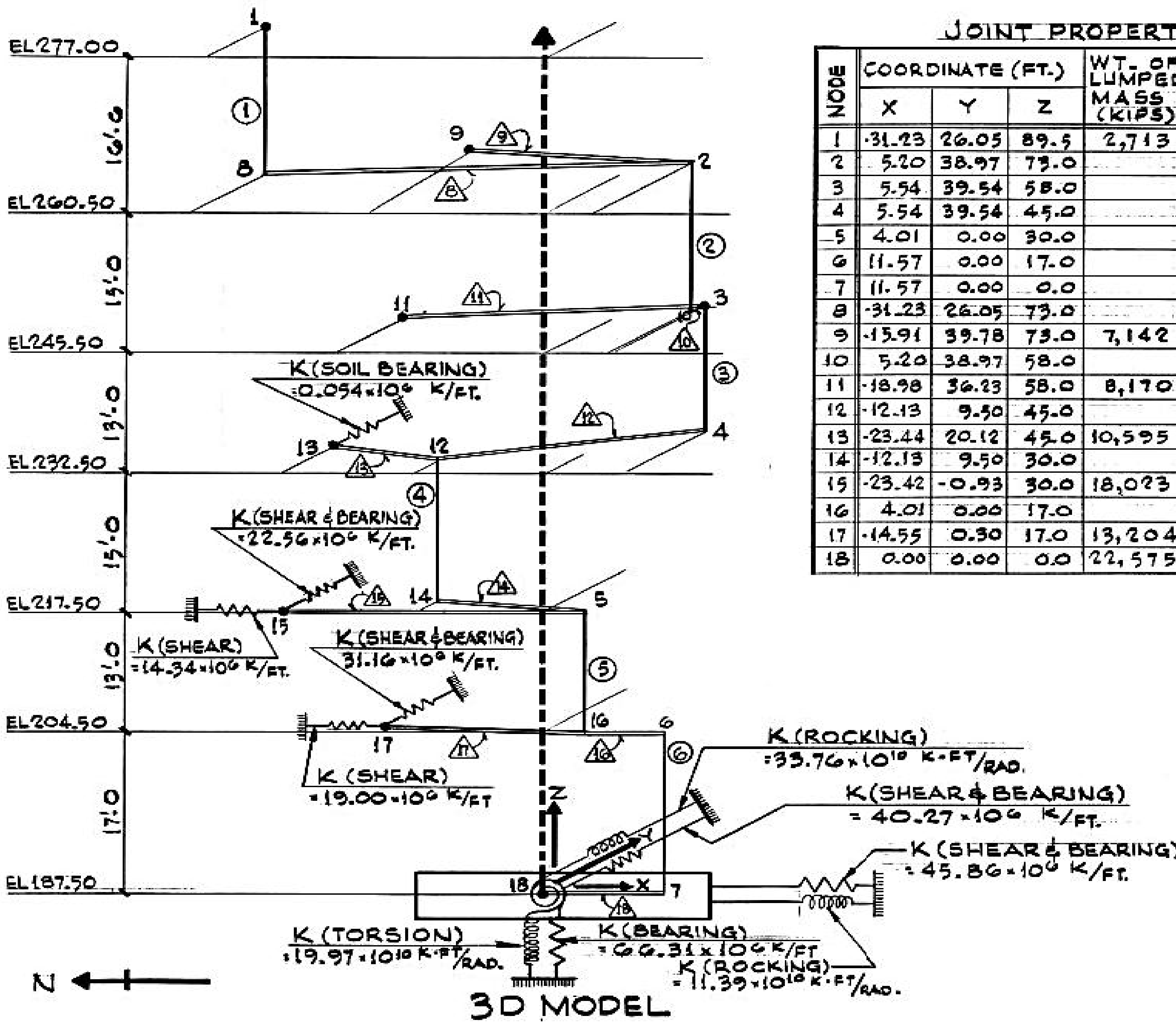


FIGURE 3.7.2-14

E.S.W.S. INTAKE STRUCTURE 3D TORSIONAL MODEL



JOINT PROPERTIES

NODE	COORDINATE (FT.) X	Y	Z	WT. OF LUMPED MASS (KIPS)	WT. MOMENT OF INERTIA Jz (KIP·FT²)
1	-31.23	26.05	89.5	2,713	2.58×10⁶
2	5.20	38.97	73.0		
3	5.54	39.54	58.0		
4	5.54	39.54	45.0		
5	4.01	0.00	30.0		
6	11.57	0.00	17.0		
7	11.57	0.00	0.0		
8	-31.23	26.05	73.0		
9	-15.91	39.78	73.0	7,142	12.01×10⁶
10	5.20	38.97	58.0		
11	-18.98	36.23	58.0	8,170	14.93×10⁶
12	-12.13	9.50	45.0		
13	-23.44	20.12	45.0	10,595	27.64×10⁶
14	-12.13	9.50	30.0		
15	-23.42	-0.93	30.0	18,023	55.83×10⁶
16	4.01	0.00	17.0		
17	-14.55	0.30	17.0	13,204	58.38×10⁶
18	0.00	0.00	0.0	22,575	45.29×10⁶

MEMBER PROPERTIES

MEMBER	Az (FT²)	Ix (FT⁴)	Iy (FT⁴)	Ip (FT⁴)
1	744	0.28×10⁶	0.70×10⁶	1.15×10⁶
2	2,925	2.97×10⁶	3.46×10⁶	7.85×10⁶
3	2,868	2.92×10⁶	3.60×10⁶	7.59×10⁶
4	7,716	20.76×10⁶	3.91×10⁶	24.98×10⁶
5	6,010	18.07×10⁶	3.61×10⁶	23.00×10⁶
6	4,555	13.92×10⁶	2.52×10⁶	18.02×10⁶

NOTES:

X - AXIS = N-S DIRECTION
Y - AXIS = E-W DIRECTION
Z - AXIS = VERT. DIRECTION

WEIGHT OF MASSES & WEIGHT MOMENTS OF INERTIA INCLUDE EQUIPMENT WEIGHT & SAND FILL.

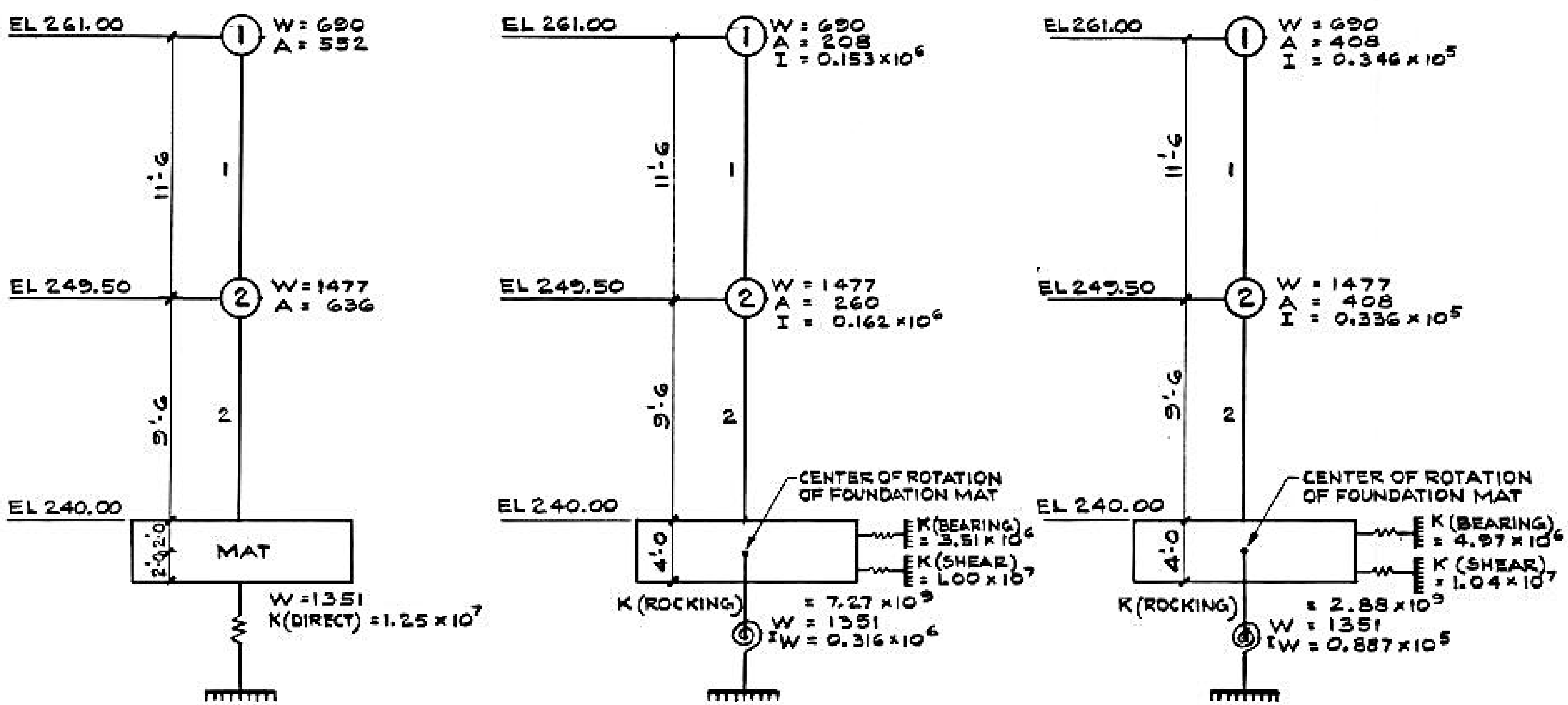
△ = HORIZONTAL RIGID BAR

○ = VERTICAL ELEMENT

FIGURE 3.7.2-15

E.S.W.S. DISCHARGE STRUCTURE MATHEMATICAL MODEL

NOTES
WEIGHTS AT MASS POINTS INCLUDE EQUIPMENT WEIGHTS.
SOIL DAMPING VALUES CONSIDERED IN DYNAMIC ANALYSIS ARE 2% FOR OBE & 5% FOR SSE
PARAMETERS ARE EXPRESSED IN FOOT-KIP



VERT. EXCITATION

HORIZ. EXCITATION N - S

HORIZ. EXCITATION E - W

FIGURE 3.7.2-16

DIESEL FUEL OIL STORAGE TANK BUILDING – MATHEMATICAL MODEL

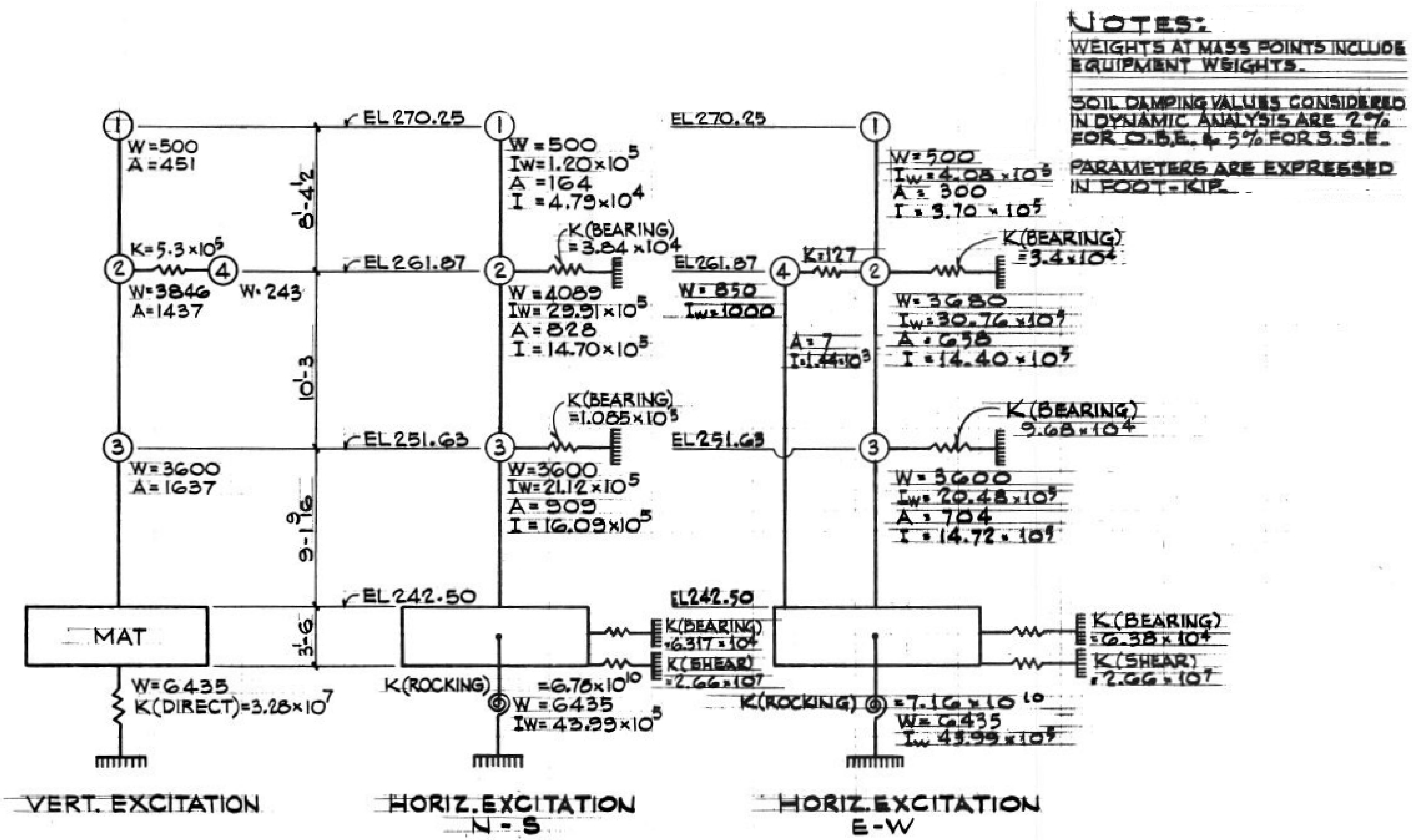
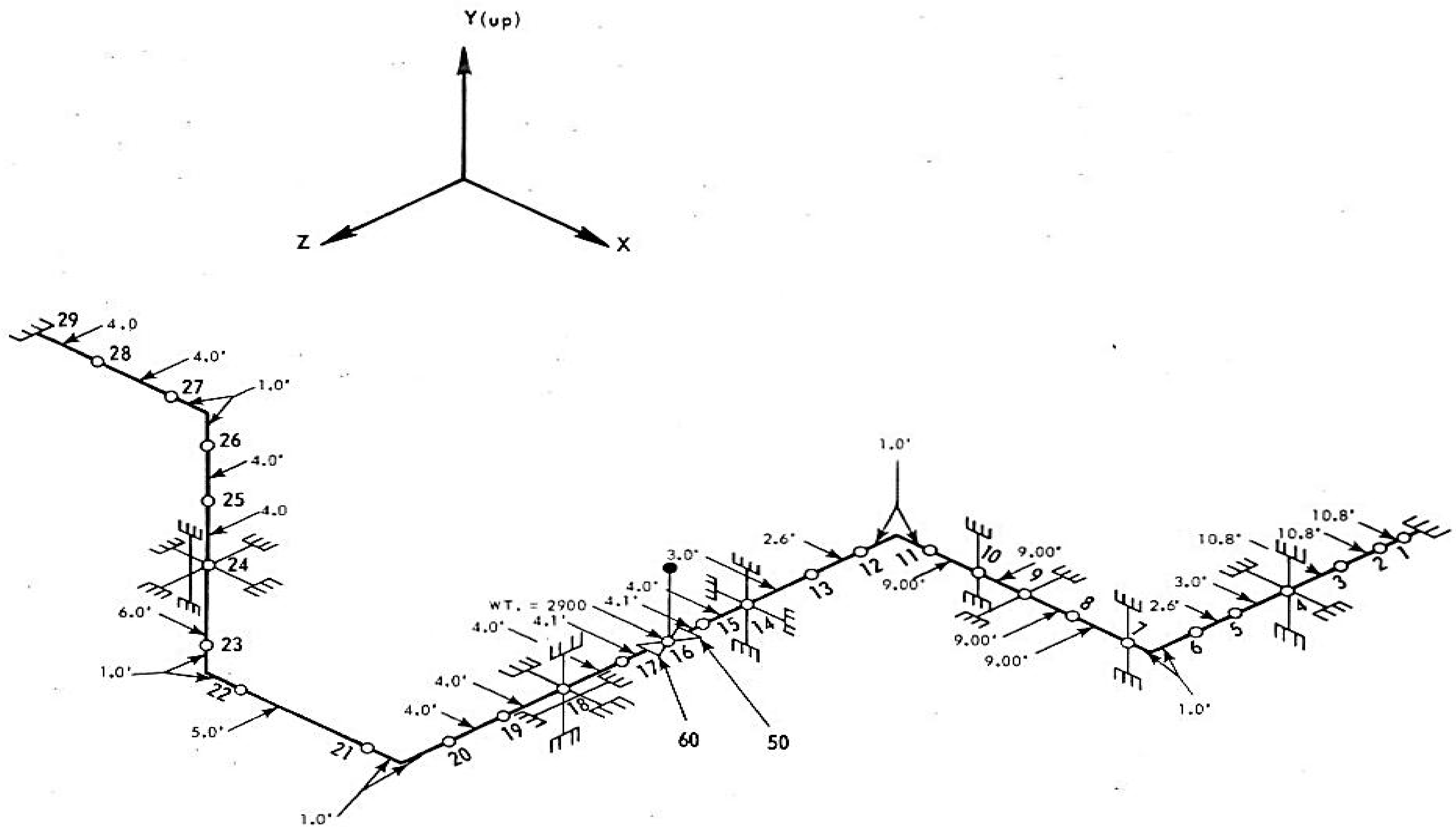


FIGURE 3.7.3-2

SEISMIC PROTECTION ANALYSIS SAMPLE PROBLEM NO. 1

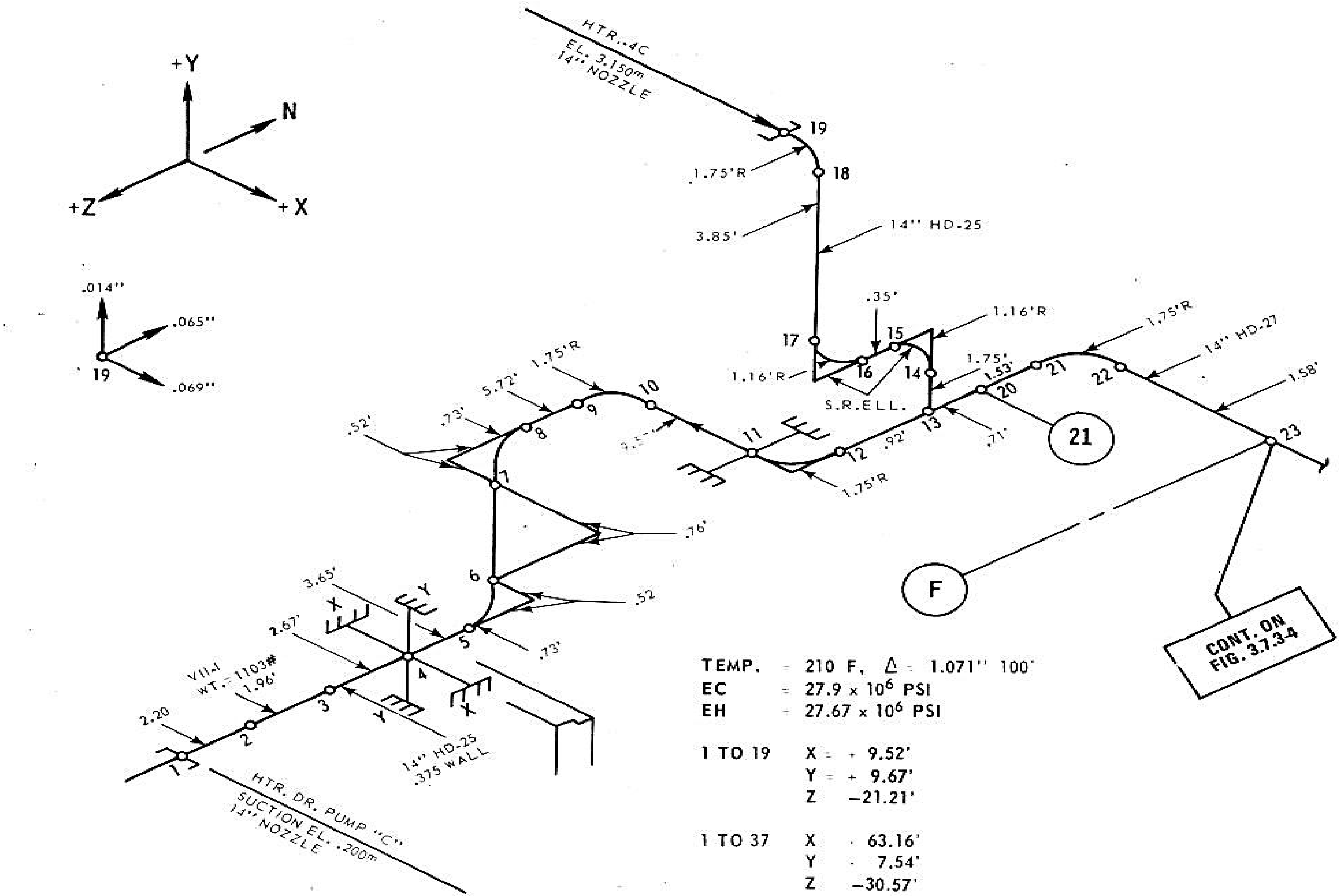


1 TO 29 X ⟶ 54.0'
Y ⟶ 16.0'
Z ⟶ 70.8'

8'' SCH. 40 C.S. WATER FILLING PIPING

FIGURE 3.7.3-3

SEISMIC PROTECTION ANALYSIS SAMPLE PROBLEM NO. 2 SHEET 1



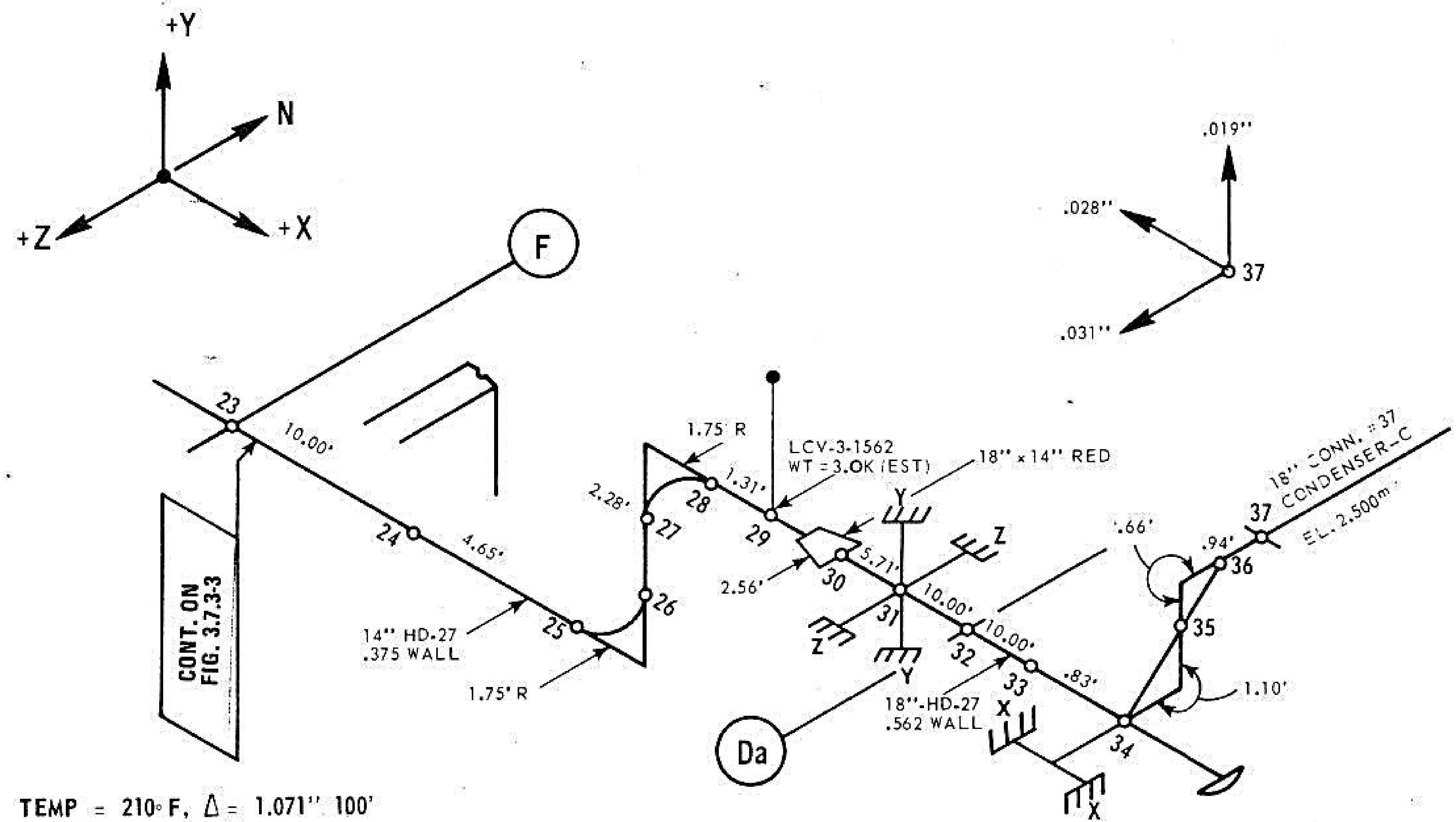
TEMP. = 210 F, Δ = 1.071'' 100'
EC = 27.9 x 10^6 PSI
EH = 27.67 x 10^6 PSI

1 TO 19 X = + 9.52'
Y = + 9.67'
Z −21.21'

1 TO 37 X · 63.16'
Y · 7.54'
Z −30.57'

FIGURE 3.7.3-4

SEISMIC PROTECTION ANALYSIS SAMPLE PROBLEM NO. 2 SHEET 2



TEMP = 210° F, Δ = 1.071'' 100'

EC = 27.9 x 10^6 PSI

EH = 27.67 x 10^6 PSI

FIGURE 3.7.3-5

SEISMIC PROTECTION ANALYSIS
SAMPLE PROBLEM NO. 3

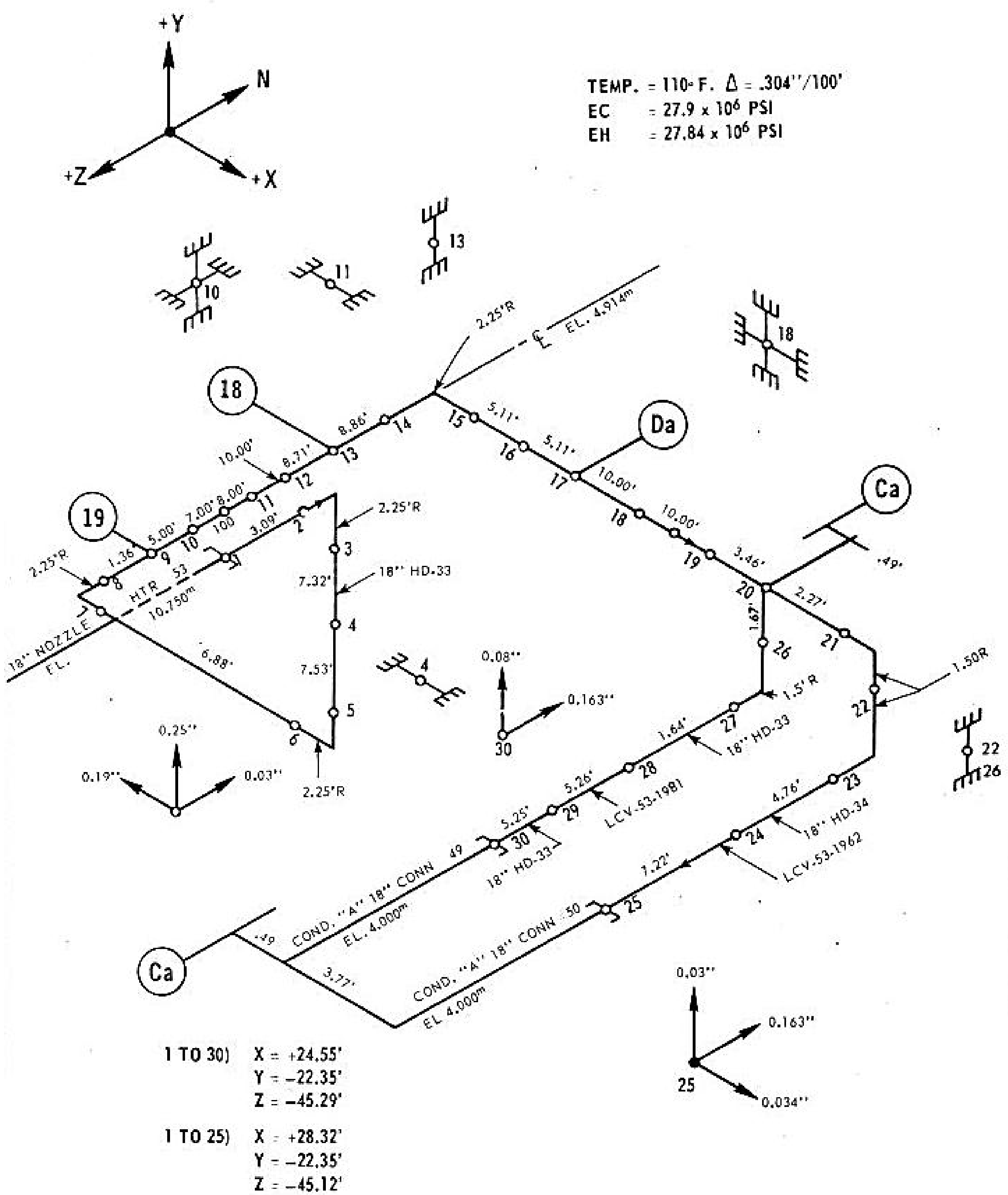


FIGURE 3.7.3-6

SEISMIC PROTECTION OF PIPING

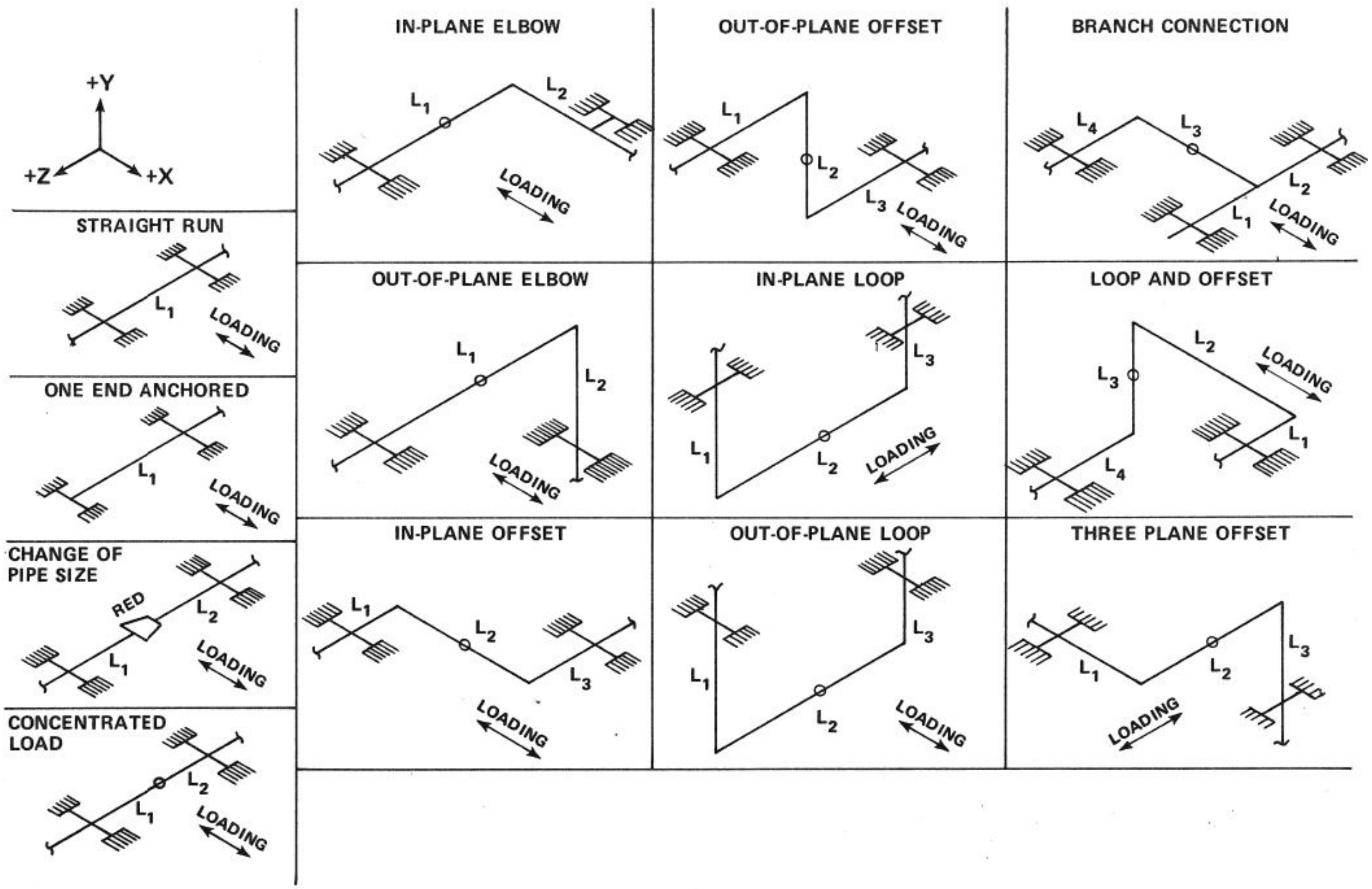
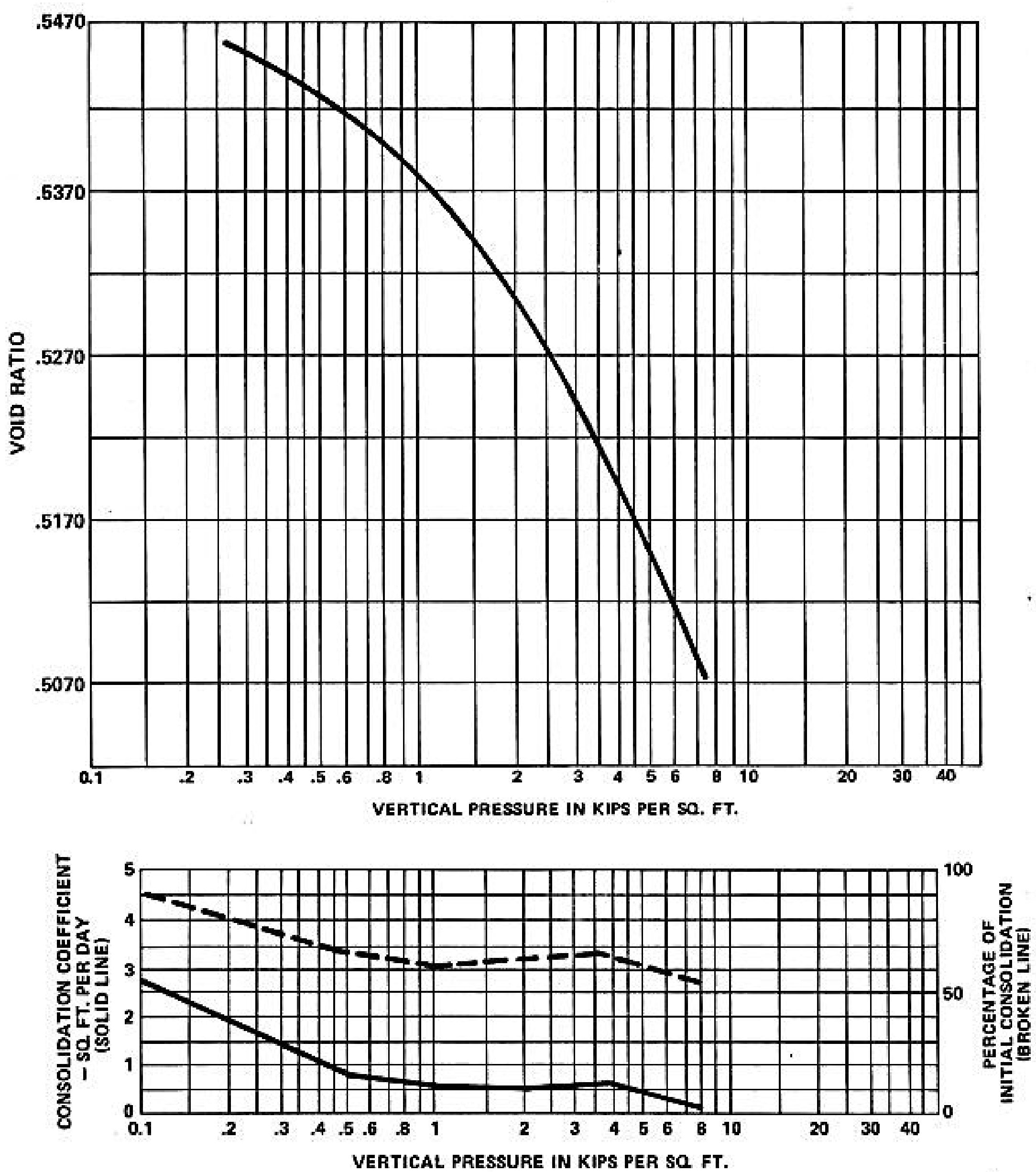


FIGURE 3.7.3-8

TIME SETTLEMENT CURVE



COMPRESSION INDEX .108 @ 8000 psf
UNIT WEIGHT (W) 127.6 pcf; (d) 108.6 pcf
WATER CONTENT 17.5%
SATURATION 85.3%

97% COMPACTION
STANDARD PROCTOR

CONSOLIDATION TEST
BORING NO. TPZ SAMPLE NO. ____
ELEV. OR DEPTH ______ JOB NO. RA-503
LAW ENGINEERING TESTING COMPANY

SOURCE: WOODWARD-MOORHOUSE & ASSOCIATES, INC.

FIGURE 3.7.3-9

PROFILE OF SERVICE WATER INTAKE

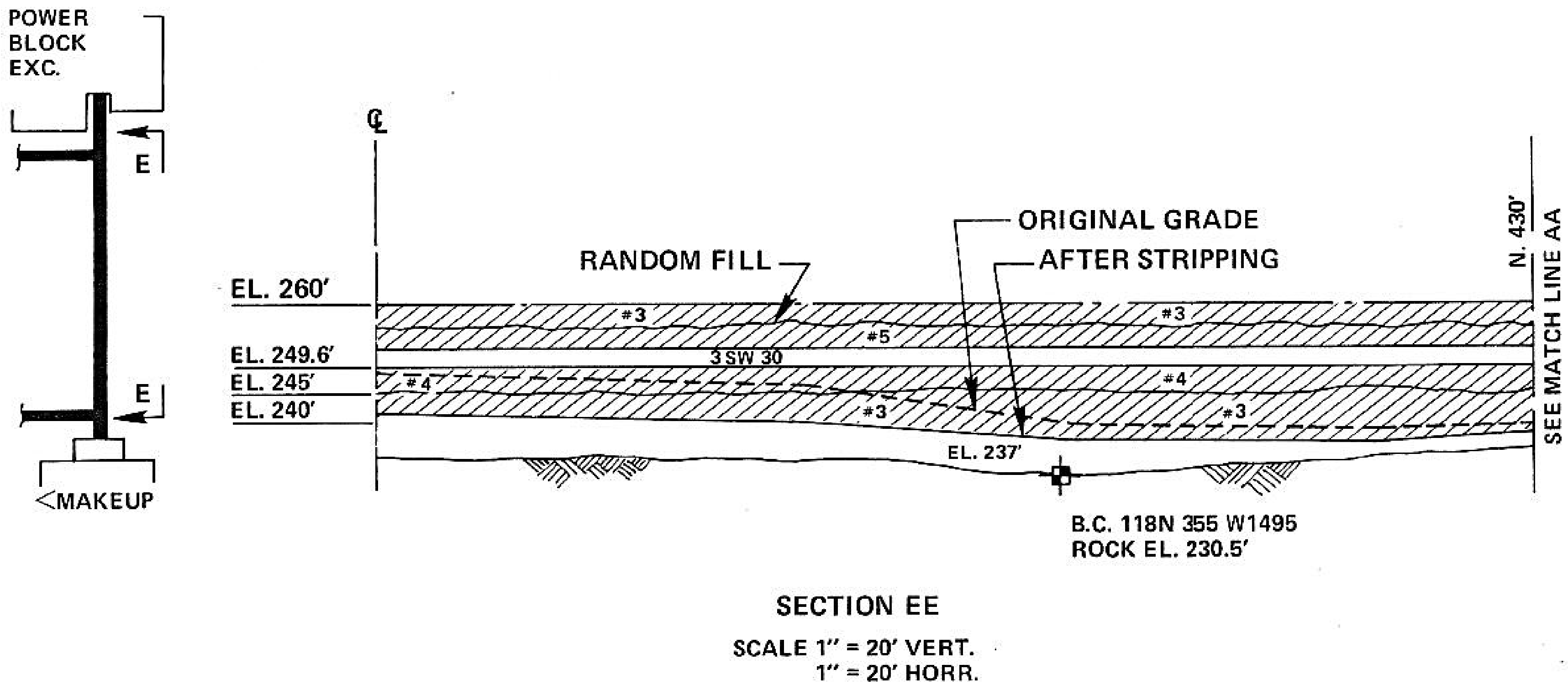


FIGURE 3.7.3-10

PROFILE OF SERVICE WATER INTAKE

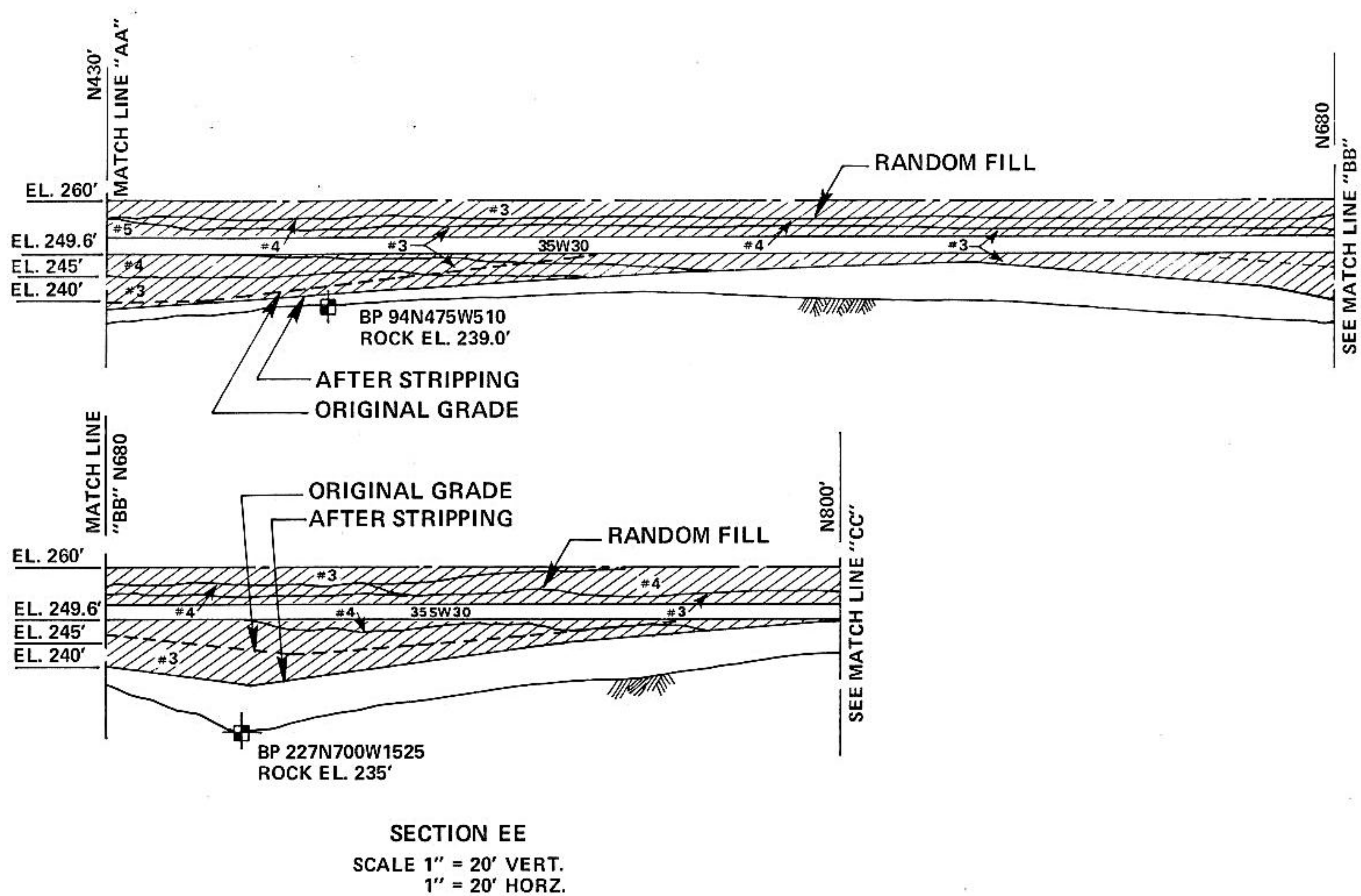


FIGURE 3.7.3-11

PROFILE OF SERVICE WATER INTAKE

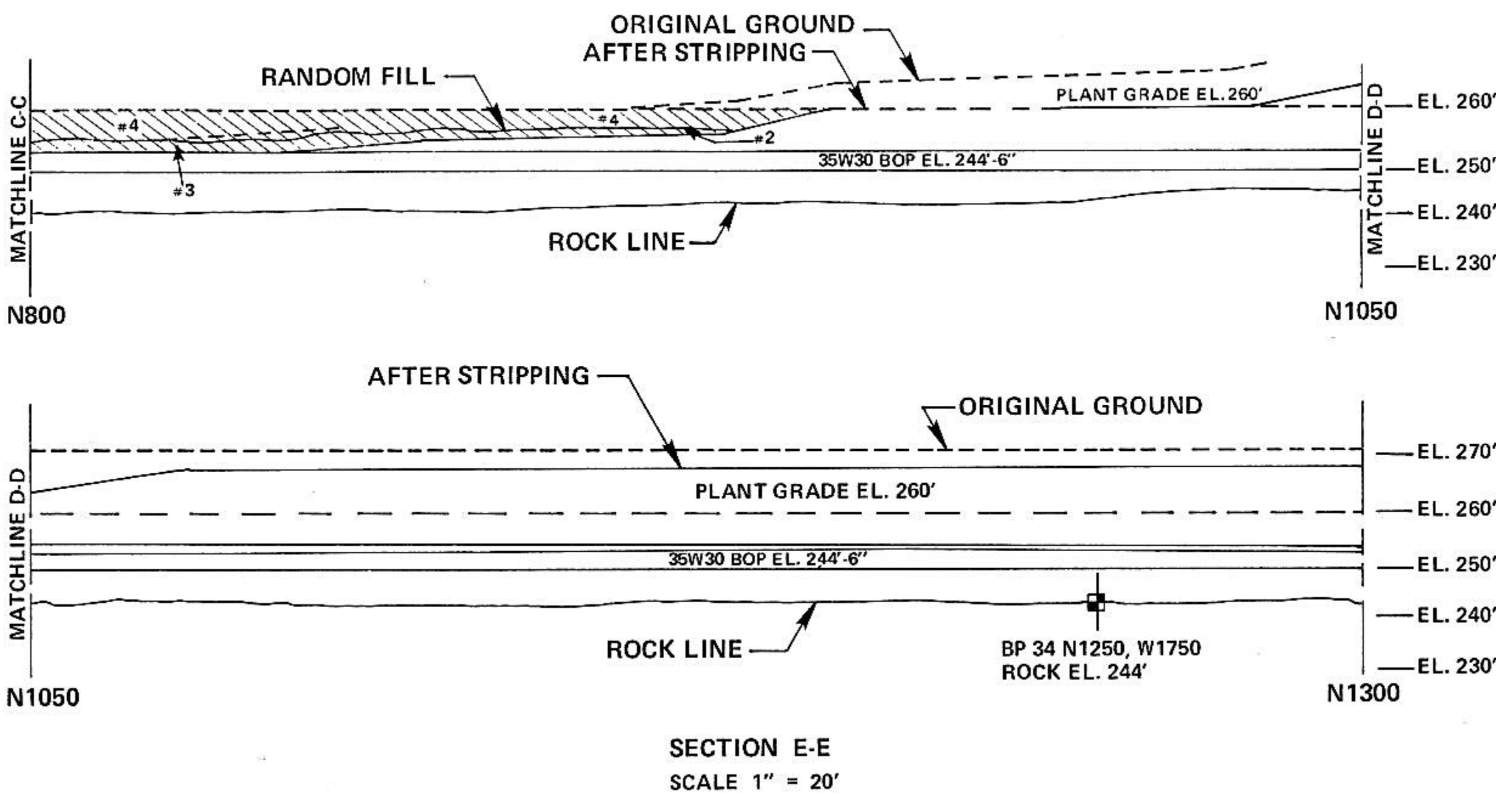


FIGURE 3.7.3-12

MATHEMATICAL MODEL OF PIPING SYSTEM

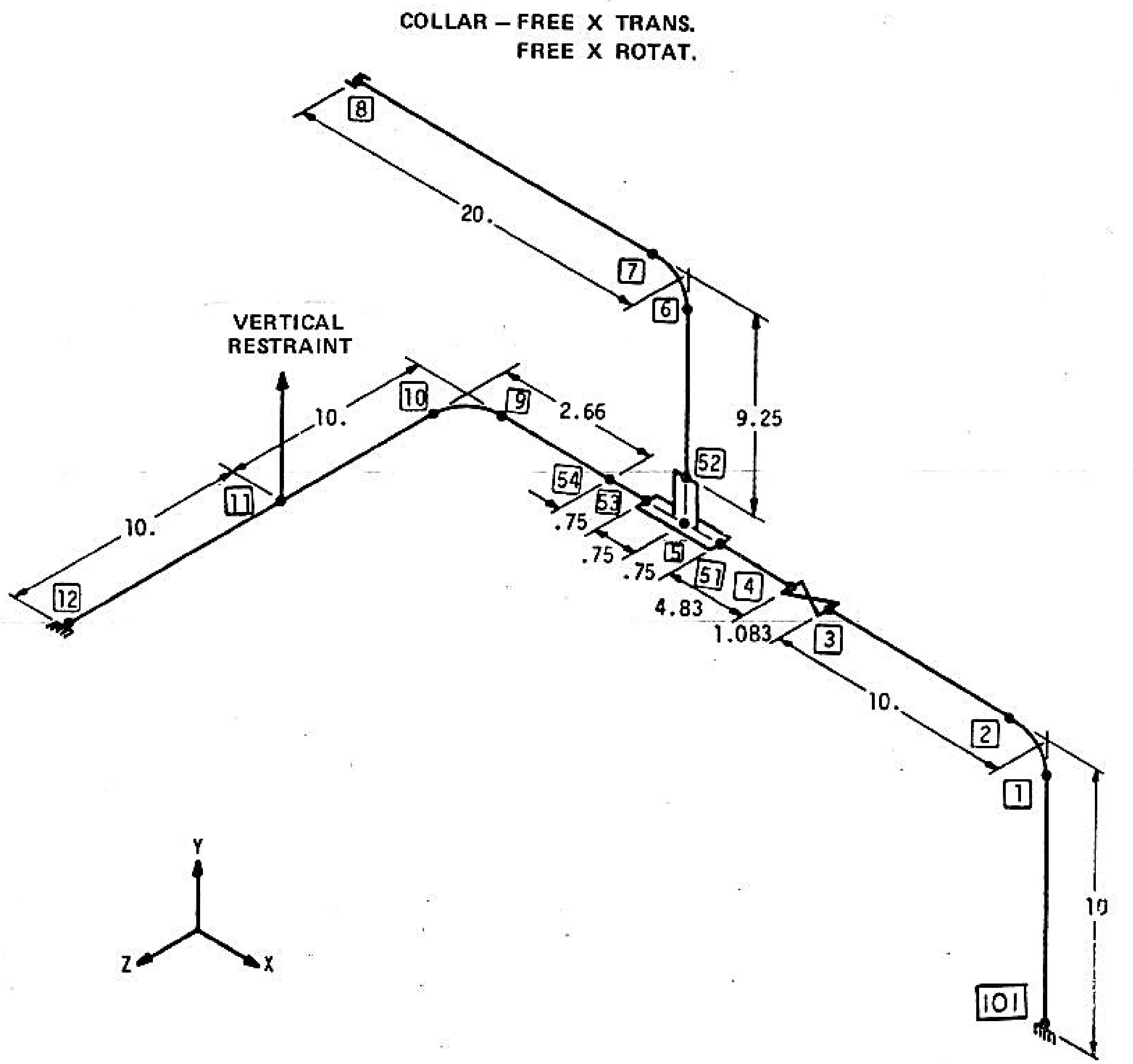


FIGURE 3.8.1-1

CONCRETE CONTAINMENT STRUCTURE – GENERAL ARRANGEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-2

CONCRETE CONTAINMENT STRUCTURE – MAT, MASONRY & REINFORCING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-3

CONCRETE CONTAINMENT STRUCTURE CYLINDER WALL REINFORCEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-4

CONCRETE CONTAINMENT STRUCTURE SEISMIC REINFORCEMENT

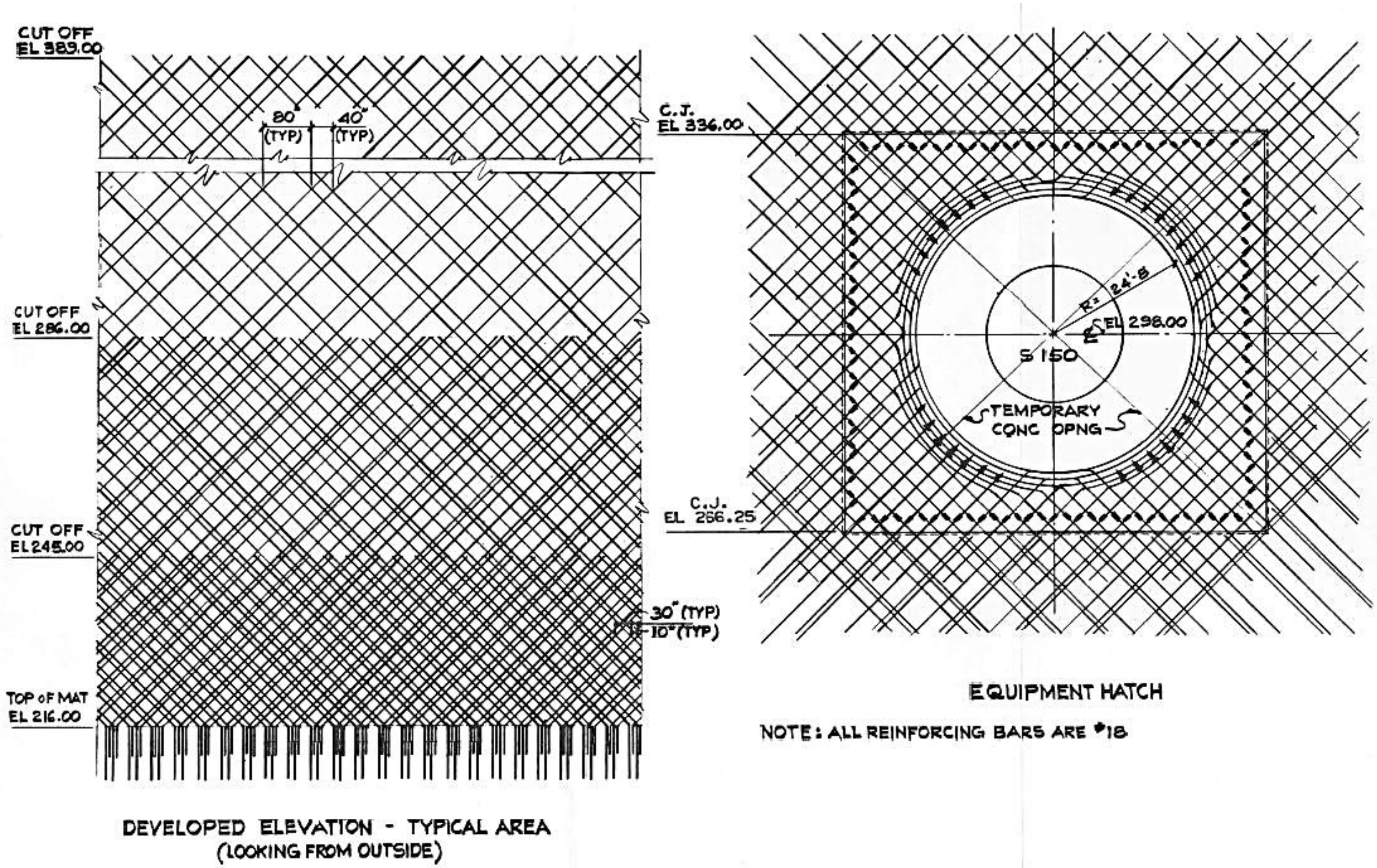


FIGURE 3.8.1-5

CONCRETE CONTAINMENT STRUCTURE – SEISMIC REINFORCEMENT

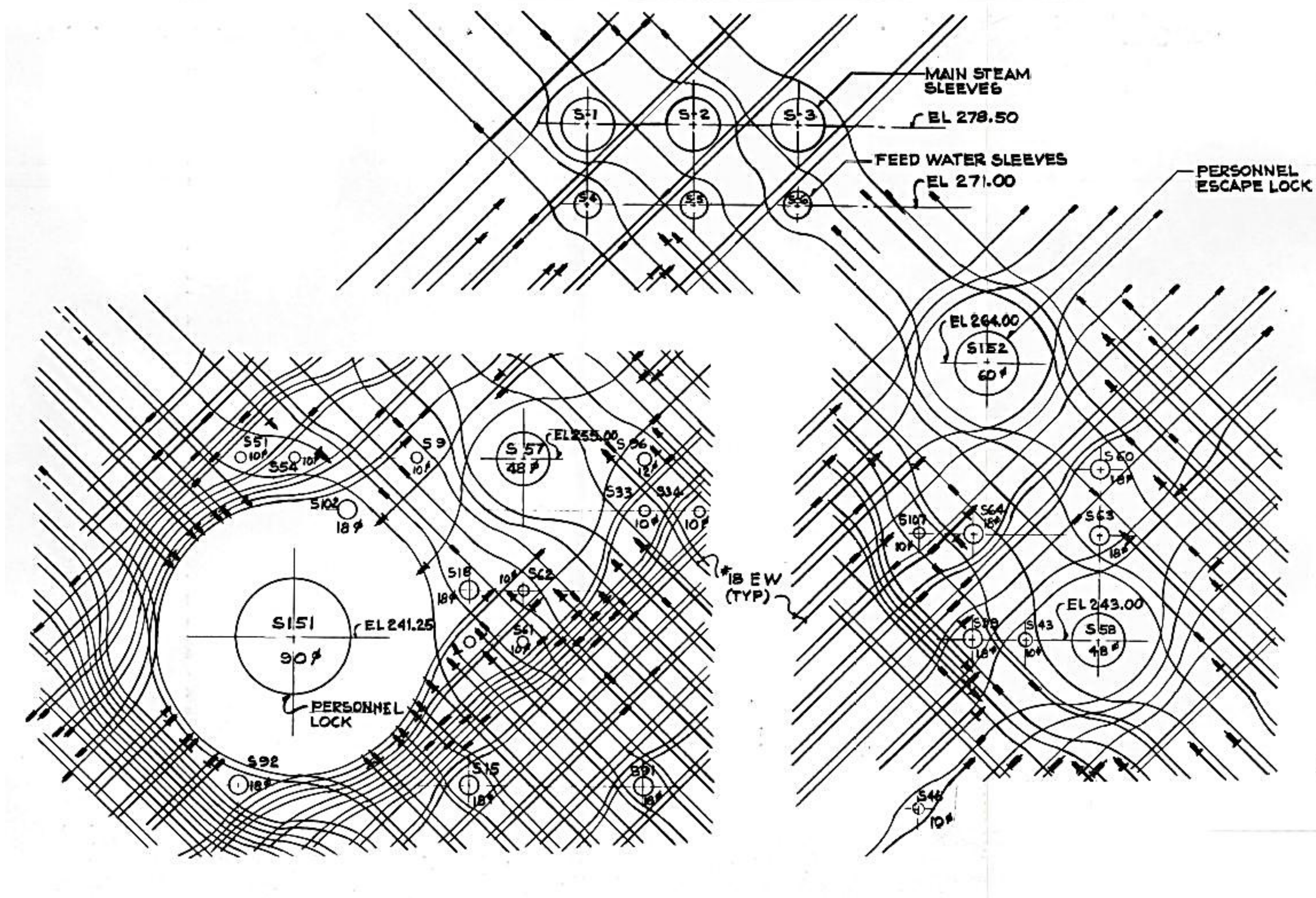


FIGURE 3.8.1-6

CONCRETE CONTAINMENT STRUCTURE – EQUIPMENT HATCH REINFORCING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-7

CONCRETE CONTAINMENT STRUCTURE PERSONNEL AIR LOCK AND PEN. S57 REINFORCING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-8

CONCRETE CONTAINMENT STRUCTURE PERSONNEL ESCAPE LOCK AND PEN. S58 REINFORCING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-9

CONCRETE CONTAINMENT STRUCTURE – MS & FW PENETRATION REINFORCING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-10

CONCRETE CONTAINMENT STRUCTURE MS & FW PENETRATION ATTACHMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-11

CONCRETE CONTAINMENT STRUCTURES – SMALL PENETRATION REINFORCING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-12

CONCRETE CONTAINMENT STRUCTURE – LINER DETAIL

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-13

CONCRETE CONTAINMENT STRUCTURE – LINER DETAILS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-14

CONCRETE CONTAINMENT STRUCTURE EQUIPMENT HATCH PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-15

CONCRETE CONTAINMENT STRUCTURE PERSONNEL LOCK PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-16

CONCRETE CONTAINMENT STRUCTURE ESCAPE LOCK PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-17

CONCRETE CONTAINMENT BUILDING MECHANICAL TYPE I PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-18

CONCRETE CONTAINMENT BUILDING MECHANICAL TYPE II PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-19

CONCRETE CONTAINMENT BUILDING ELECTRICAL TYPE III PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-20

CONCRETE CONTAINMENT BUILDING FUEL TRANSFER TUBE PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-21

CONCRETE CONTAINMENT STRUCTURE VALVE CHAMBER

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-22

CONCRETE CONTAINMENT STRUCTURE – DOME REINFORCEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-23

CONCRETE CONTAINMENT STRUCTURE DOME REINFORCEMENT SHEET 2

Security-Related Information - Figure Withheld Under 10 CFR 2.390

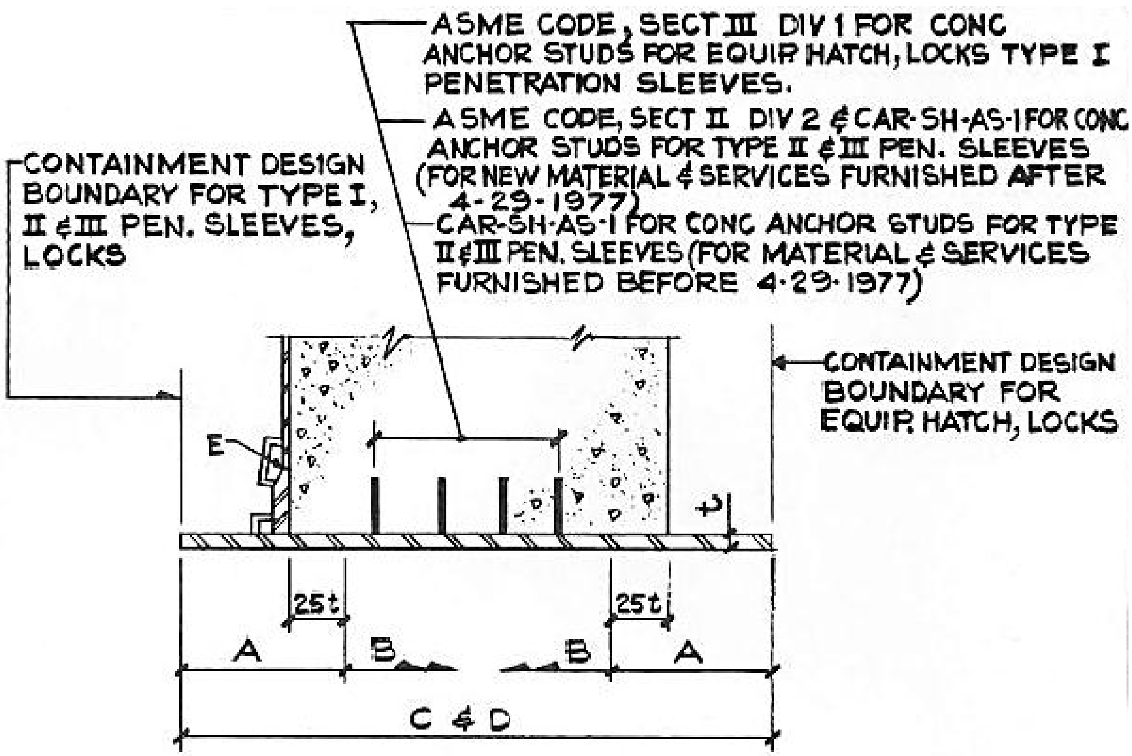
FIGURE 3.8.1-24

CONCRETE CONTAINMENT STRUCTURE BOUNDARIES

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-25

CONCRETE CONTAINMENT STRUCTURE PENETRATION BOUNDARY



A - DIV. BOUNDARY FOR ASME CODE, SECT III, DIV 1 WITH SUBSECTION NE (FOR NEW MATERIAL & SERVICES FURNISHED AFTER 4-29-1977) FOR TYPE II & III PENETRATION SLEEVES.

B - DIV. BOUNDARY FOR ASME CODE SECT III, DIV 2 (FOR NEW MATERIAL & SERVICES FURNISHED AFTER 4-29-1977) FOR TYPE II & III PENE. SLEEVES

C - DIV. BOUNDARY FOR ASME CODE SECT III DIV 1 WITH SUBSECTION NA FOR EQUIP. HATCH, LOCKS TYPE I PEN SLEEVES.

D - DIV. BOUNDARY FOR ASME CODE, SECT III, DIV 1 WITH SUBSECTION NA FOR TYPE II & III PEN SLEEVES (FOR MATERIAL & SERVICES FURNISHED BEFORE 4-29-1977)

E - ATTACHMENT WELD BETWEEN ASME CODE SECTION III DIVISION 1 ITEMS AND LINER ARE IN ACCORDANCE WITH THE ASME CODE SECTION III DIVISION 2

DIV. BOUNDARIES FOR THE TYPE II & III PENETRATION SLEEVES

LOCKS AND EQUIPMENT HATCH

FIGURE 3.8.1-26

CONCRETE CONTAINMENT STRUCTURE MAT STRUCTURAL RESPONSES

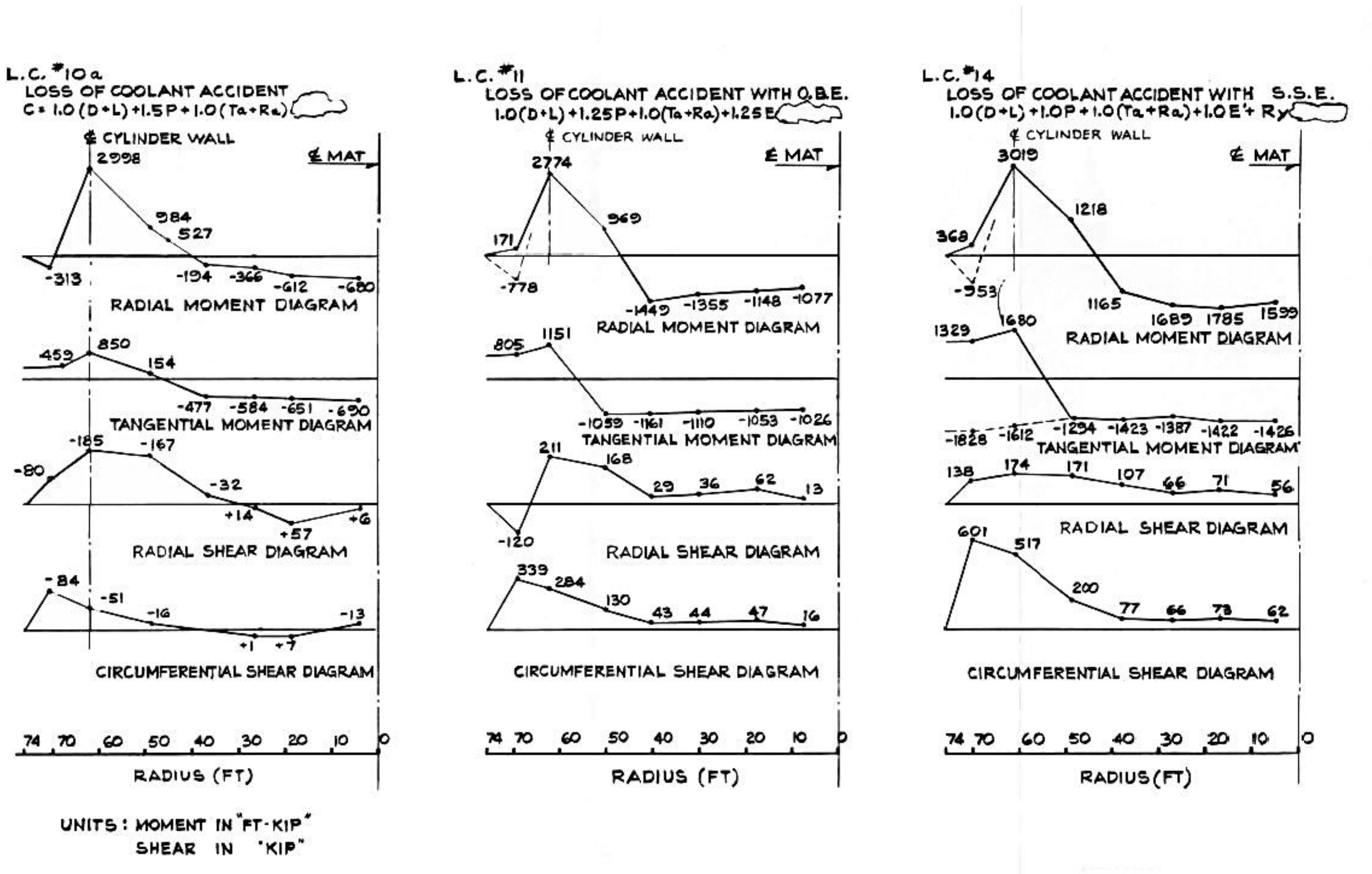
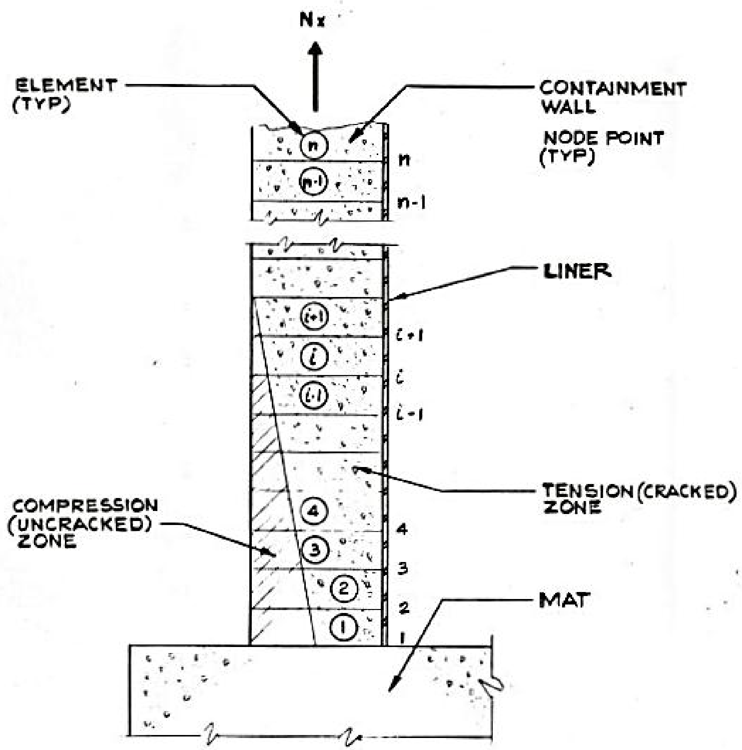
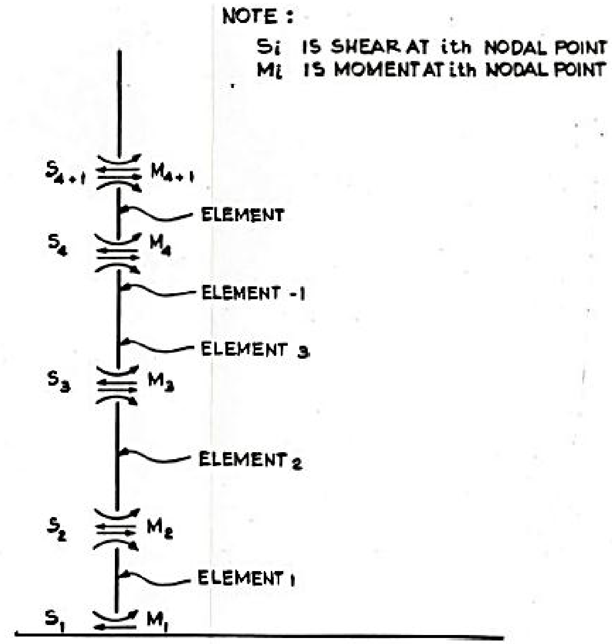


FIGURE 3.8.1-27

CONCRETE CONTAINMENT STRUCTURE CYLINDRICAL WALL
FINITE ELEMENT MODEL AND MODELING OF CRACKS



MODELING OF CRACKS



FINITE ELEMENT MODEL

FIGURE 3.8.1-28

CONCRETE CONTAINMENT STRUCTURE AXISYMMETRIC LOADS
FINITE ELEMENT MODELS
COMPARATIVE STUDY RESULTS

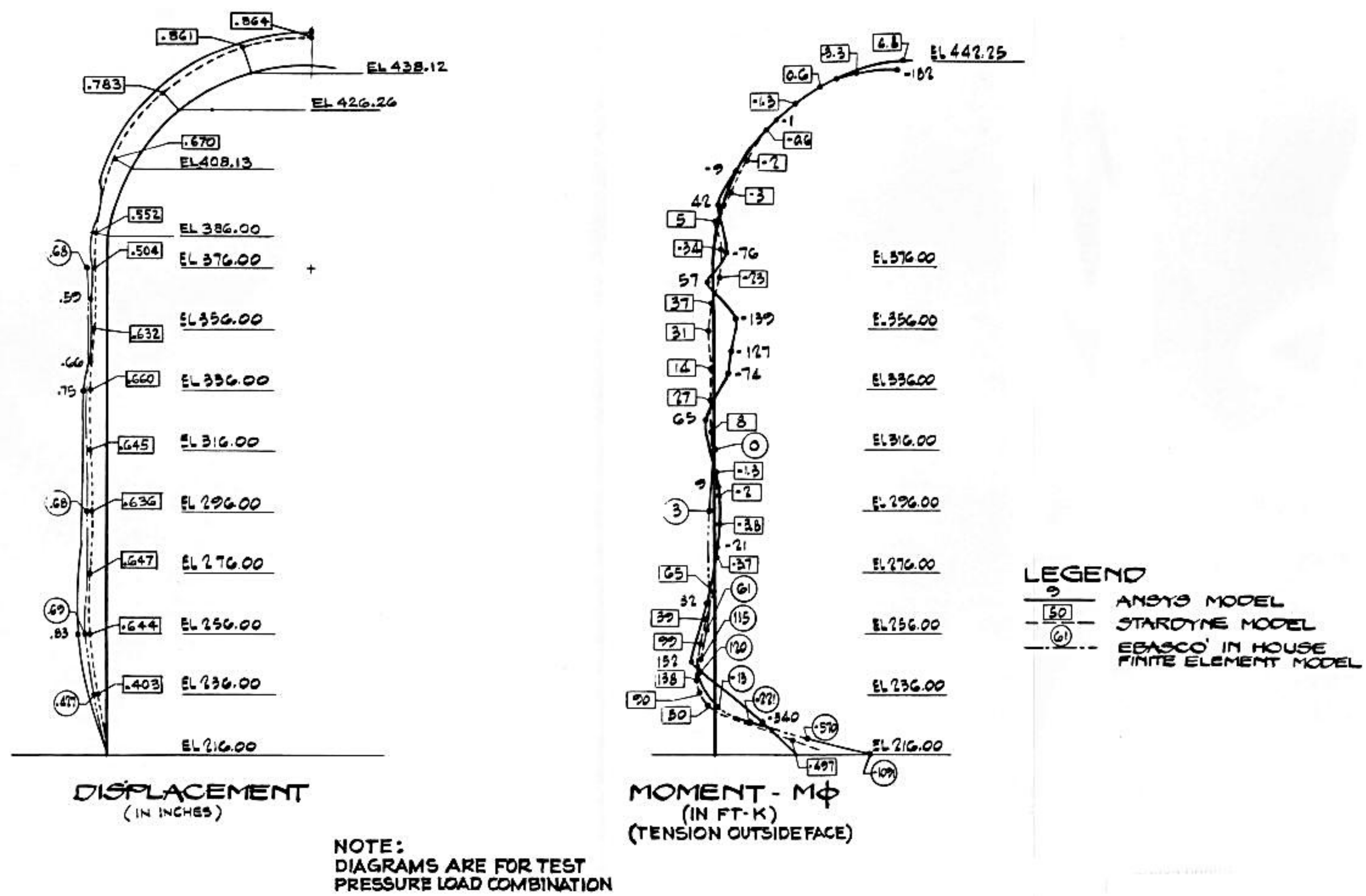


FIGURE 3.8.1-29

CONCRETE CONTAINMENT STRUCTURE – CYLINDRICAL WALL AND DOME STRUCTURE REPONSES

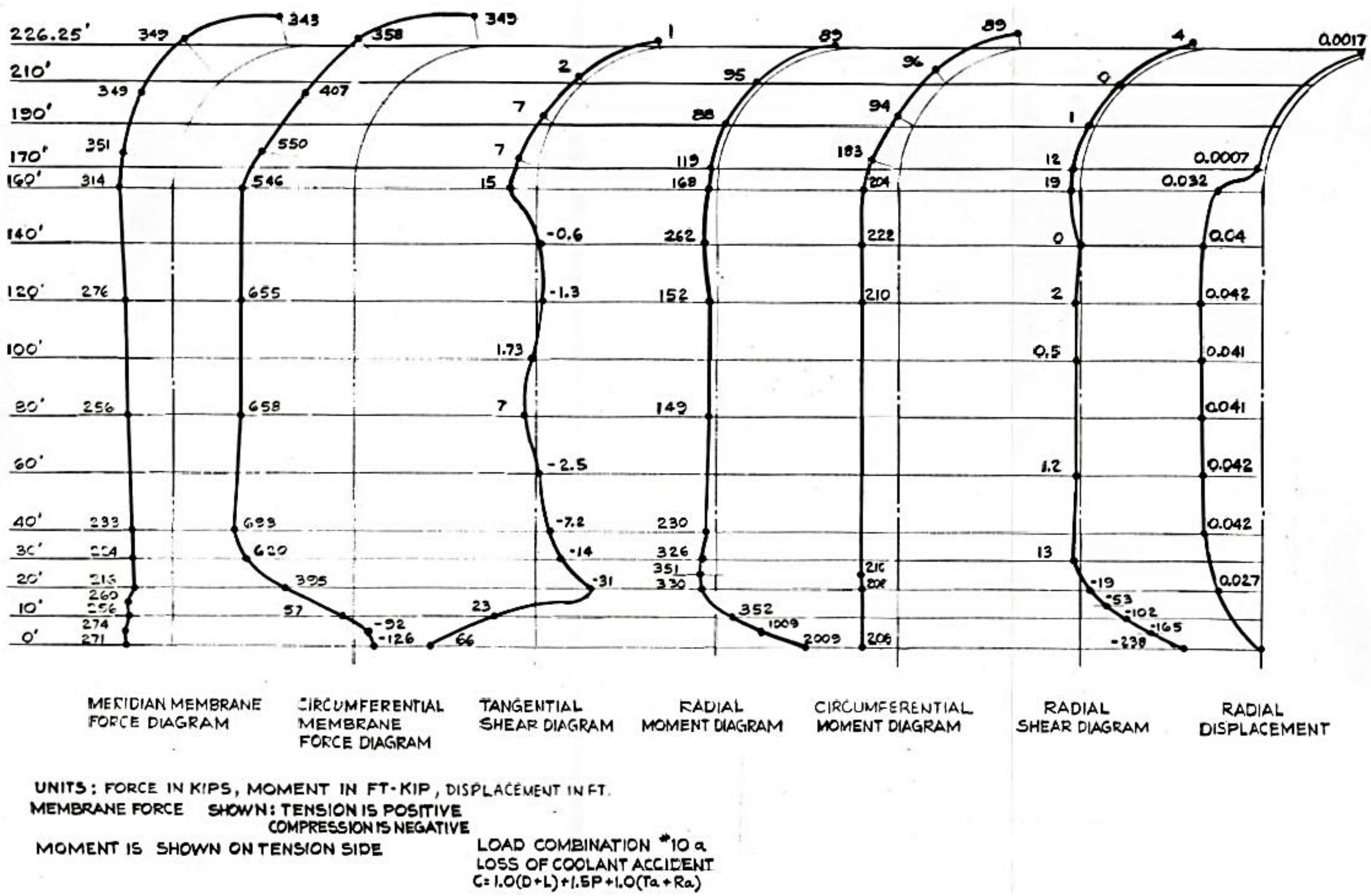


FIGURE 3.8.1-30

CONCRETE CONTAINMENT STRUCTURE – CYLINDRICAL WALL AND DOME STRUCTURE REPONSES

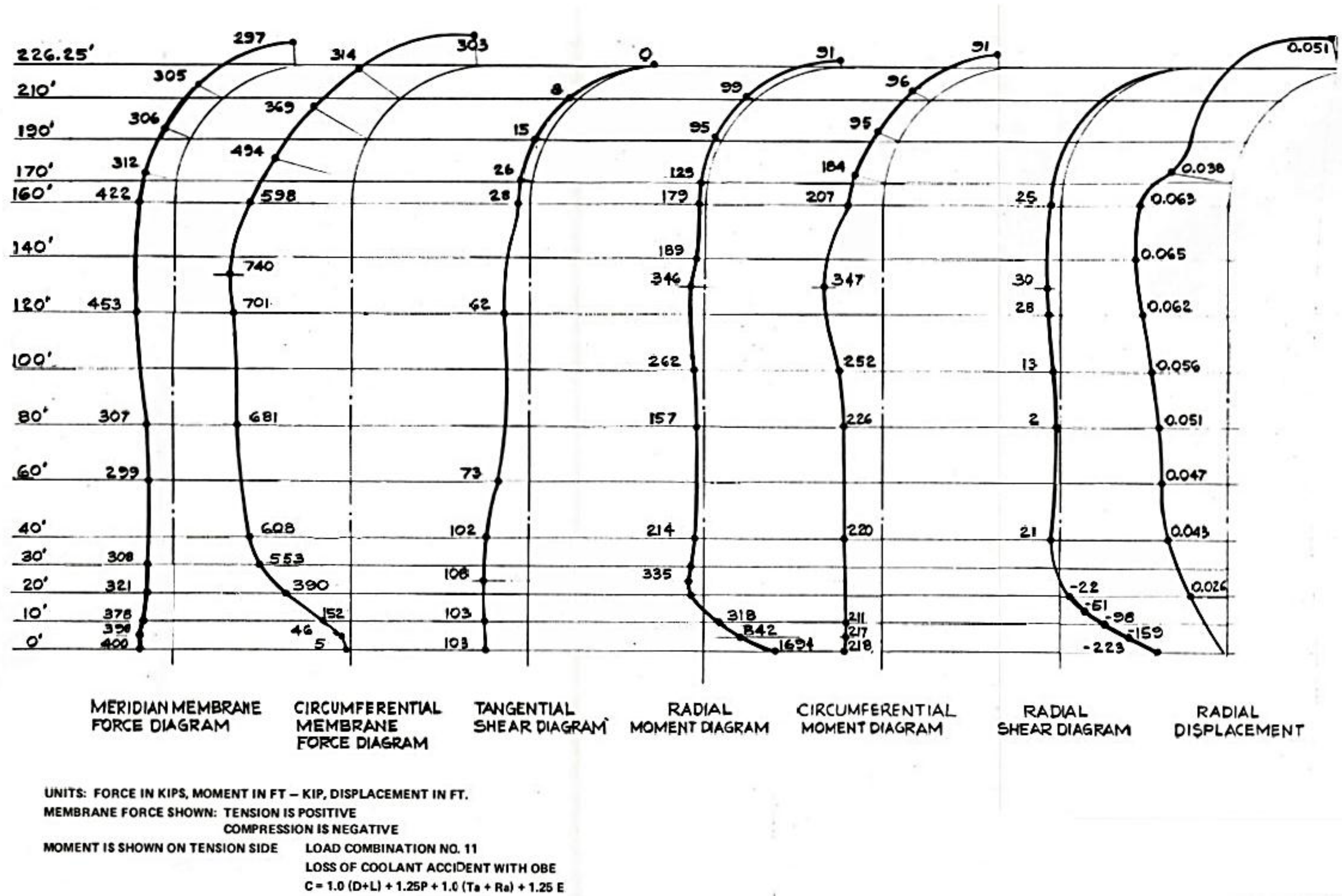


FIGURE 3.8.1-31

CONCRETE CONTAINMENT STRUCTURE – CYLINDRICAL WALL AND DOME STRUCTURE REPONSES

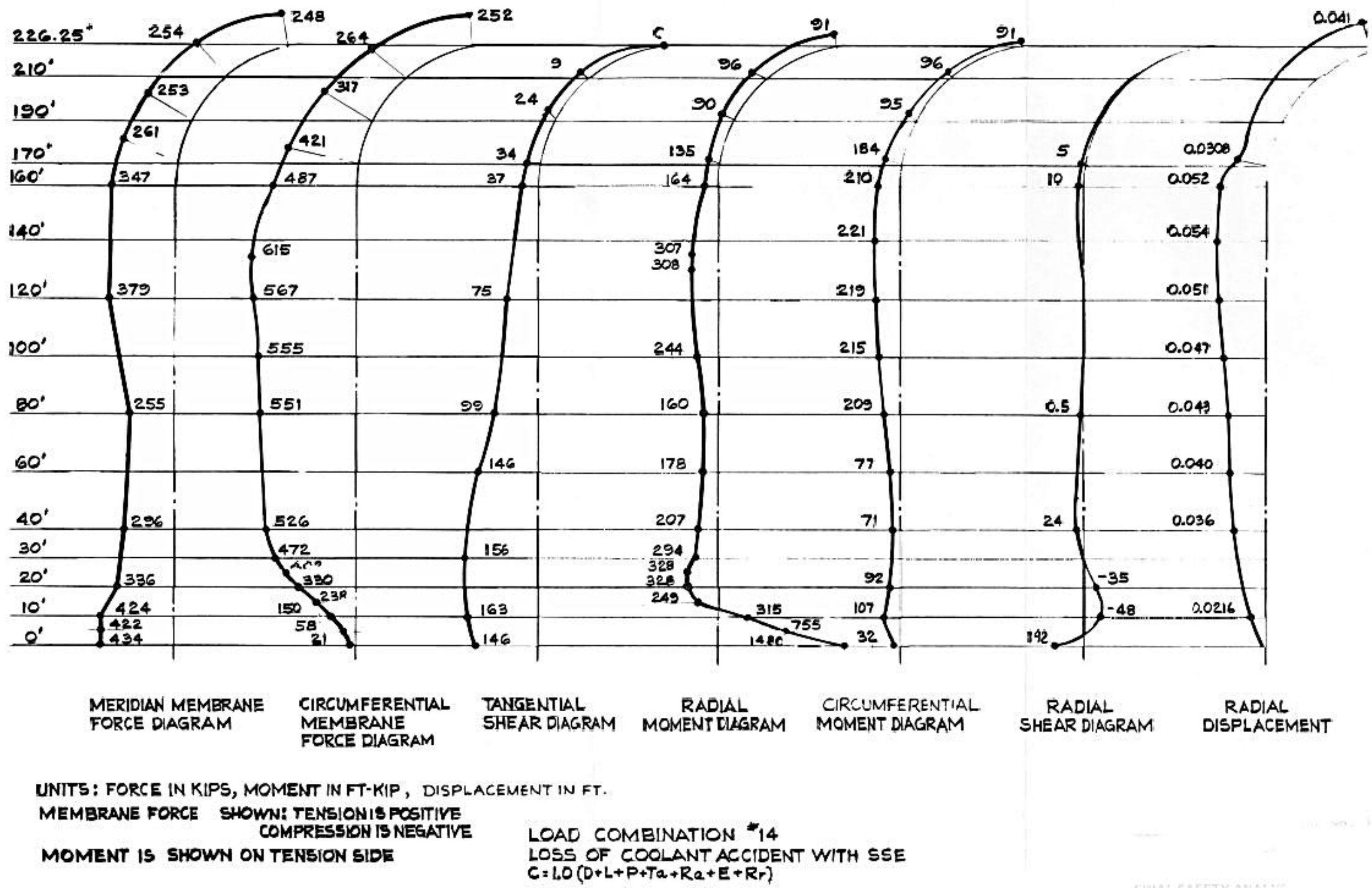


FIGURE 3.8.1-32

CONCRETE CONTAINMENT STRUCTURE
TEST OF 5/8” DIAMETER X 4” LONG HEADED
STUDS IN TENSION-CONCRETE IN TENSION

CURVE	SPRING CONSTANT K/IN.	ULTIMATE LOAD K	ULTIMATE DEFORMATION IN
1	400	16.50	.046
2	311	16.25	.070
3	750	15.50	.037
4	275	13.75	.125

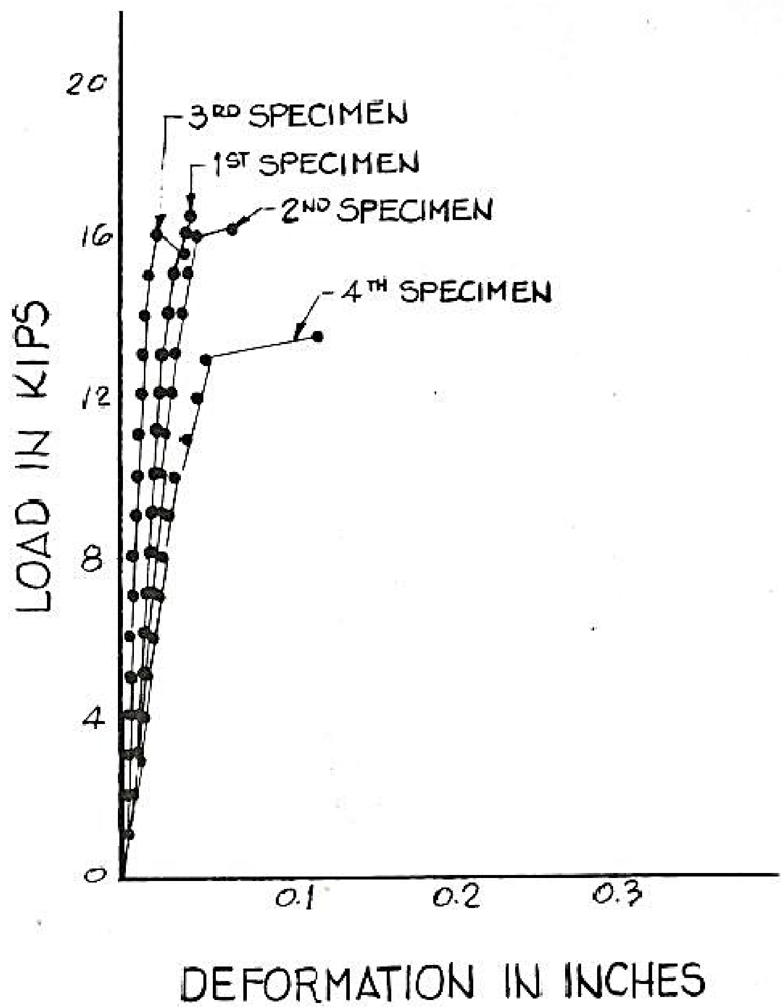


FIGURE 3.8.1-33

CONCRETE CONTAINMENT STRUCTURE TEST OF
5/8" DIAMETER X 4" LONG HEADED STUDS IN SHEAR-CONCRETE IN TENSION

CURE	SPRING CONSTANT K/IN.	ULTIMATE LOAD K	ULTIMATE DEFORMATION IN.
1	333	20.0	0.367
2	267	19.9	0.413
3	172	18.2	0.469
4	200	20.5	0.546

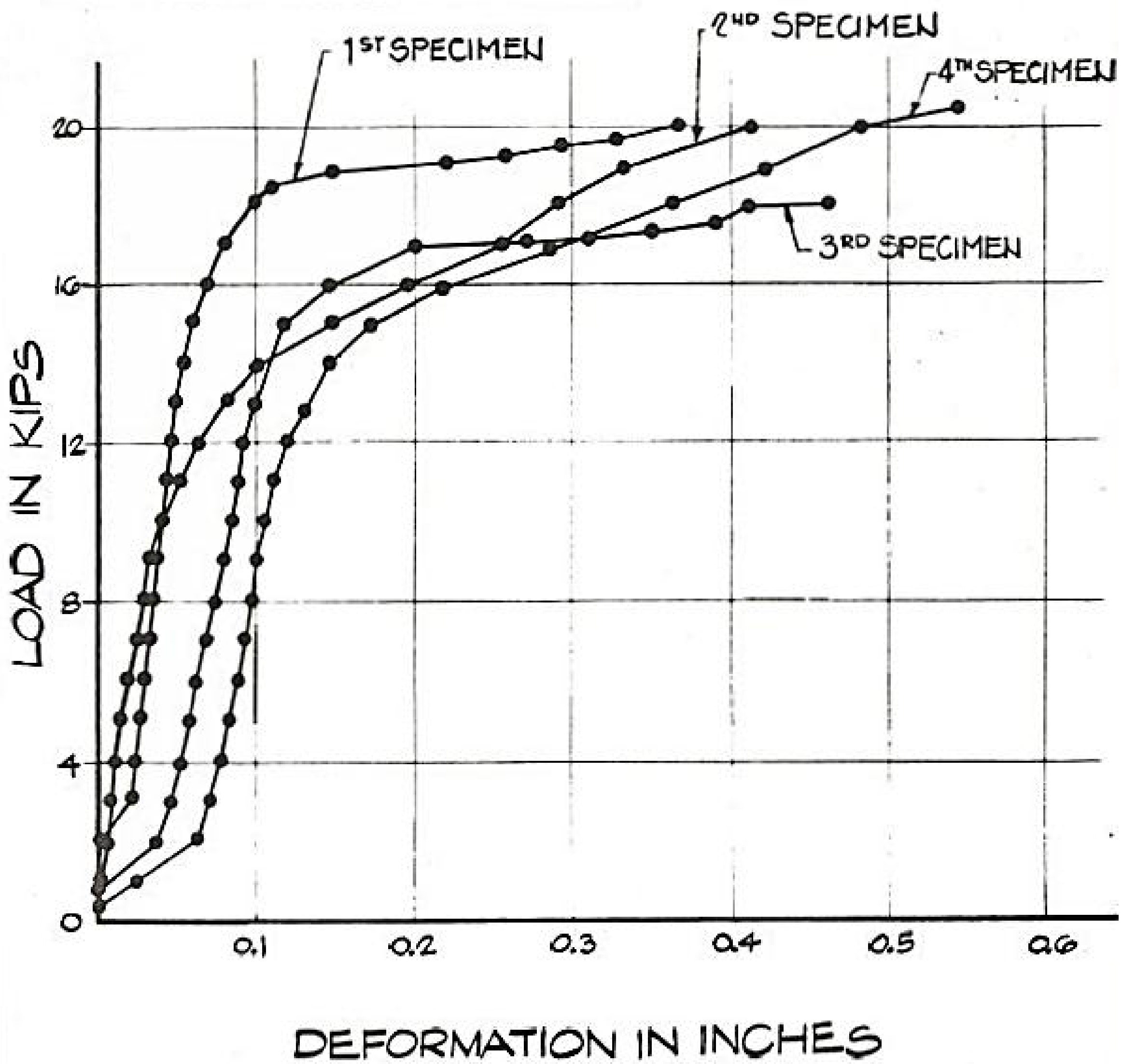


FIGURE 3.8.1-34

CONCRETE CONTAINMENT STRUCTURE TEST OF
5/8" DIAMETER X 4" LONG HEADED STUDS IN TENSION – CONCRETE UNLOADED

CURVE	SPRING CONSTANT K/IN.	ULTIMATE LOAD K	ULTIMATE DEFORMATION IN.
1	700	18.2	.06
2	560	18.0	.056
3	785	16.0	.035
4	480	18.6	.096

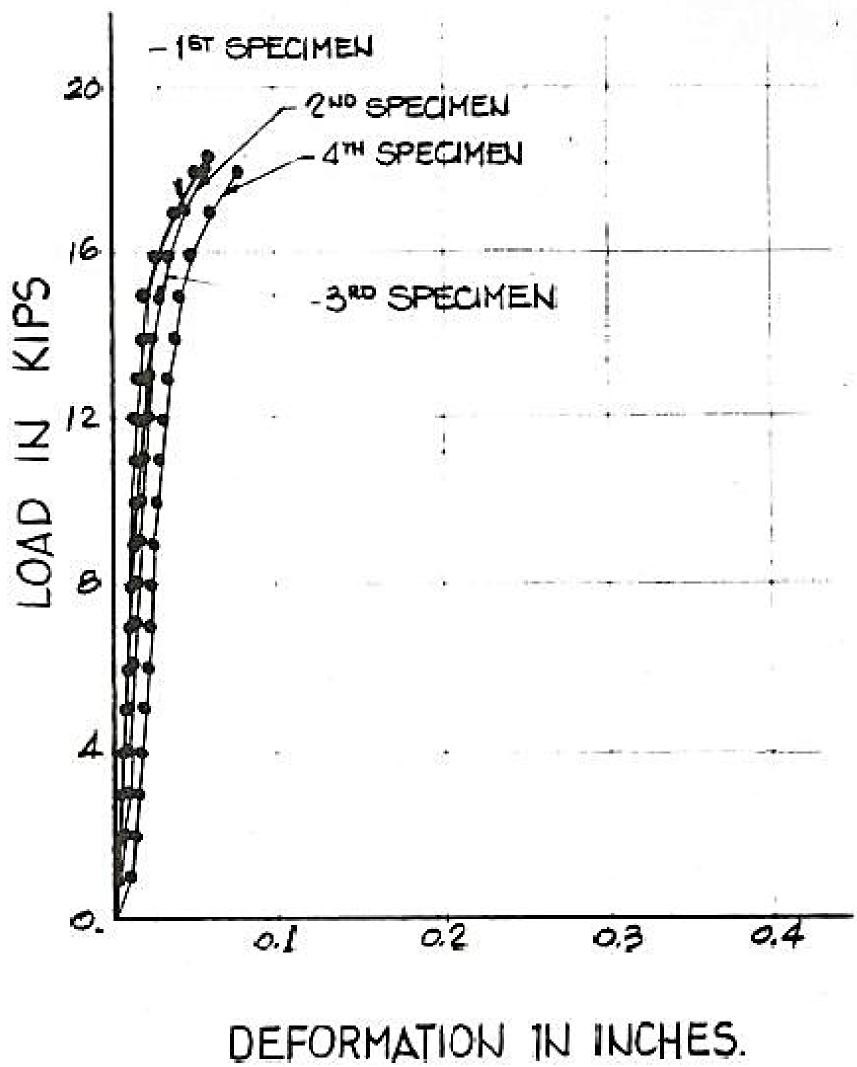


FIGURE 3.8.1-35

CONCRETE CONTAINMENT STRUCTURE TEST OF
5/8” DIAMETER X 4” LONG HEADED STUDS IN SHEAR – CONCRETE UNLOADED

CURVE	SPRING CONSTANT K/IN.	ULTIMATE LOAD K	ULTIMATE DEFORMATION IN
1	182	20	.211
2	490	20.7	.410
3	400	20.0	.227
4	467	20.7	.226

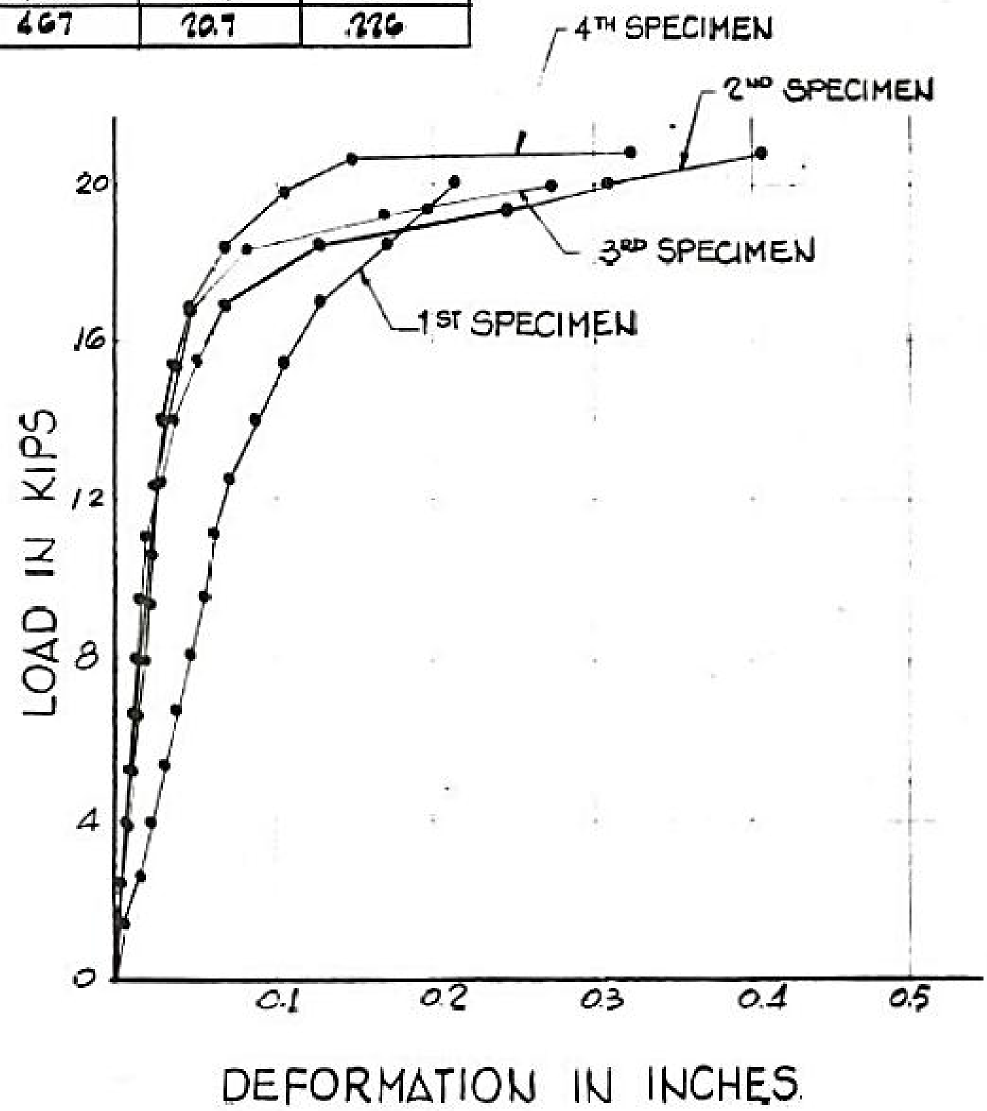


FIGURE 3.8.1-36

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT MODEL FOR LINER ANCHORAGE ANALYSIS

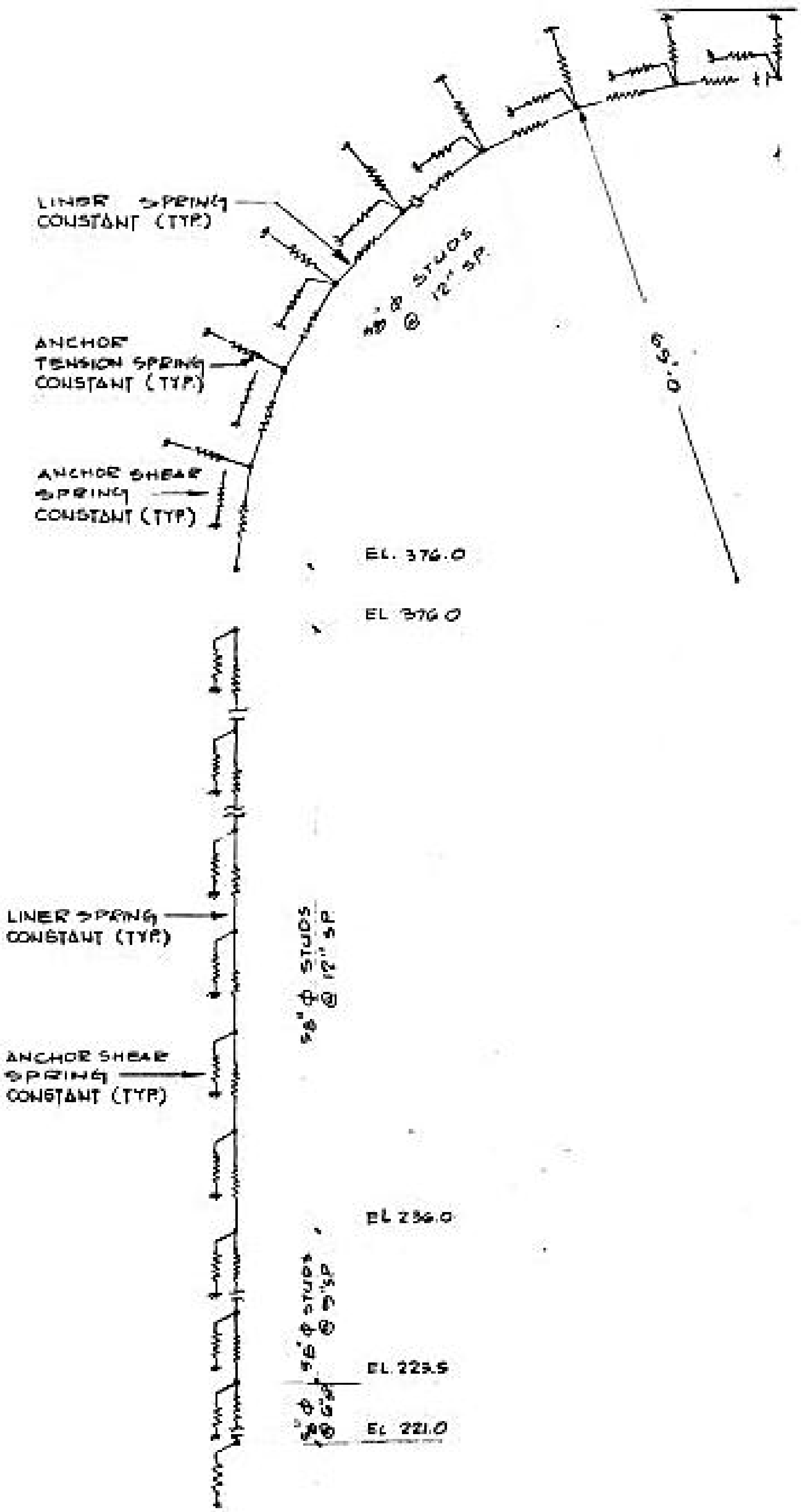


FIGURE 3.8.1-37

CONCRETE CONTAINMENT BUILDING FINITE ELEMENT MODEL
OF WALL MAT. LINER CONNECTION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-38

CONCRETE CONTAINMENT STRUCTURE – WALL MAT LINER CONNECTION STRUCTURAL RESPONSES

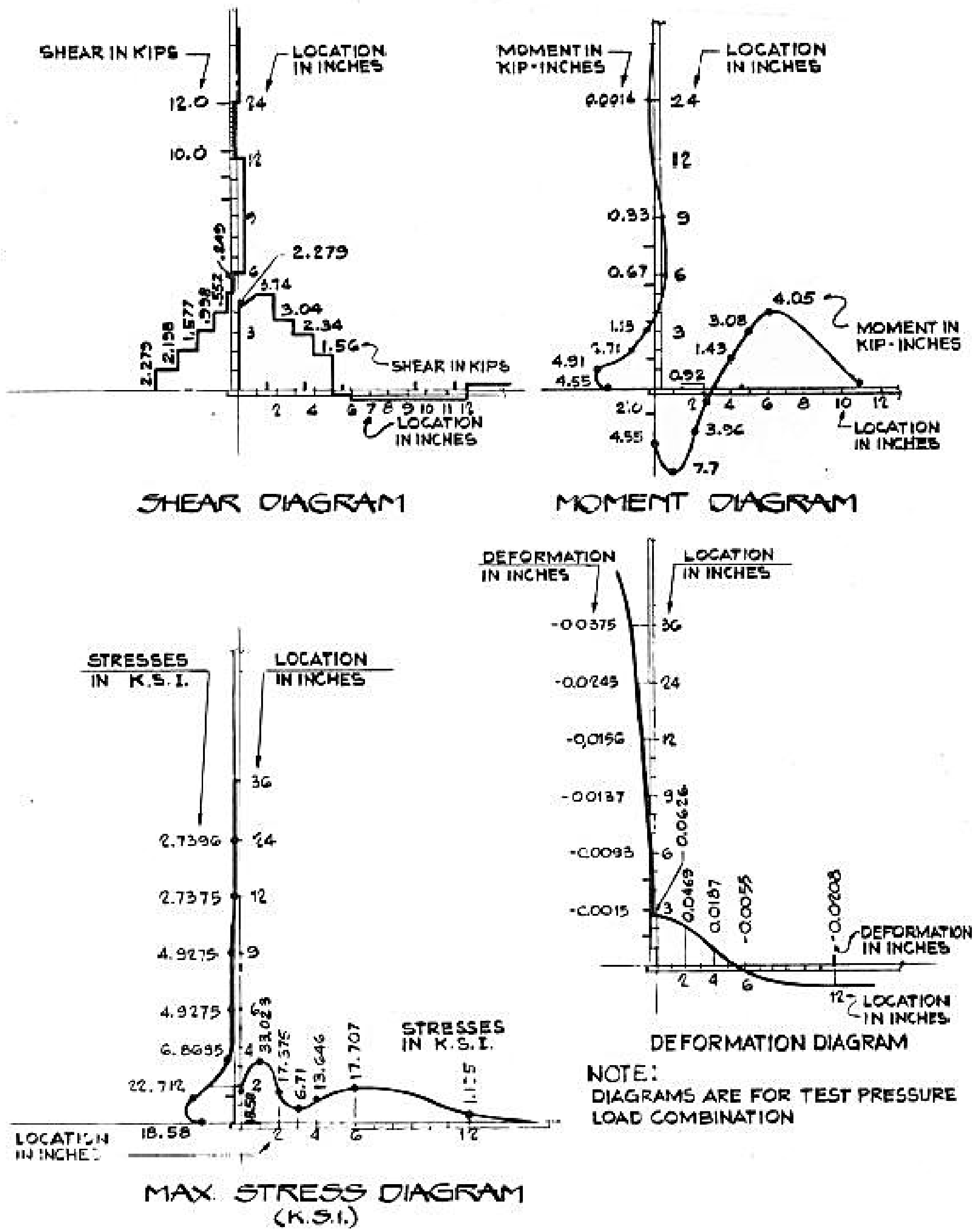


FIGURE 3.8.1-39

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT MODEL OF LINER PLATE AT CRANE GIRDER BRACKET

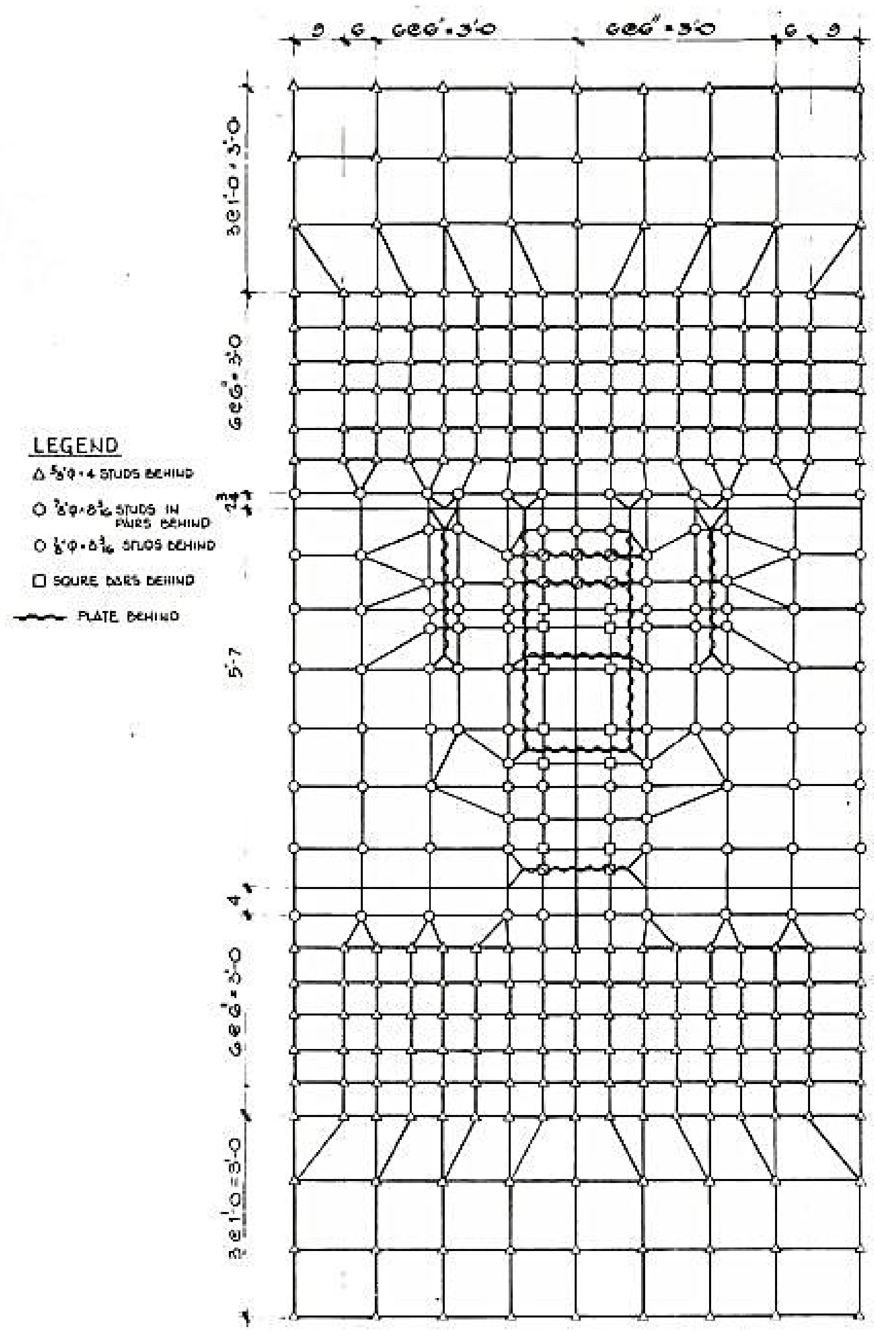
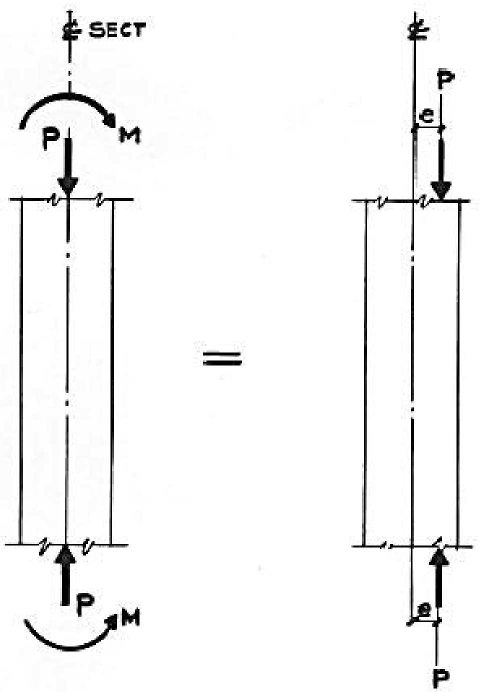
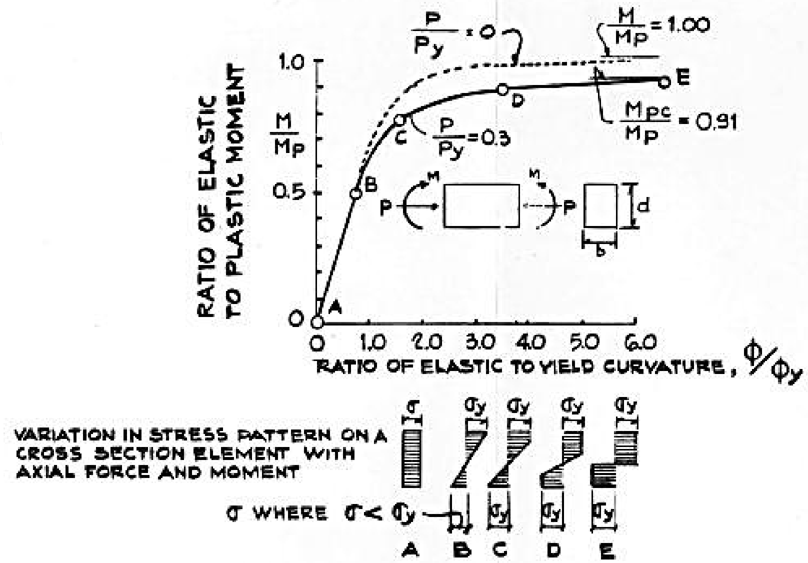


FIGURE 3.8.1-40

CONCRETE CONTAINMENT STRUCTURE PLASTIC MOMENT CAPACITY WITH AXIAL FORCE PRESENT



ECCENTRICITY e (INCHES) IN FIG. 3.8.1-41,42,43



MOMENT-CURVATURE RELATIONSHIP FOR RECTANGULAR SECTION WITH AXIAL FORCE AND MOMENT

ADAPTED FROM REF. 3.8.1-50

M_{PC} IS THE REDUCED PLASTIC MOMENT CAPACITY WHEN AXIAL FORCE IS PRESENT M_P IS THE FULL PLASTIC MOMENT CAPACITY

FIGURE 3.8.1-41

CONCRETE CONTAINMENT STRUCTURE 3/8" LINER FORCE – STRAIN DIAGRAM

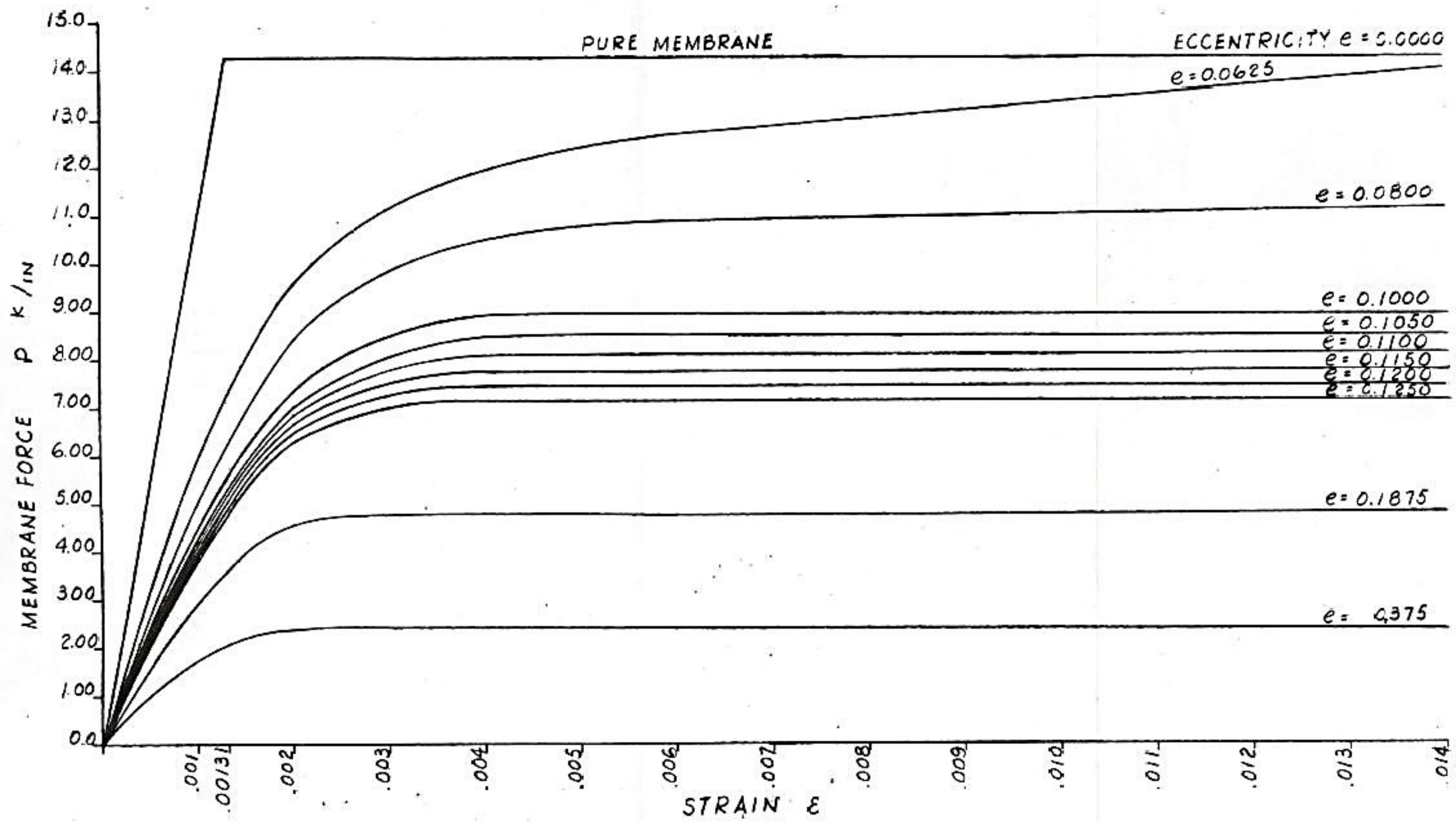


FIGURE 3.8.1-42

CONCRETE CONTAINMENT STRUCTURE 3/8" LINER MOMENT– STRAIN DIAGRAM

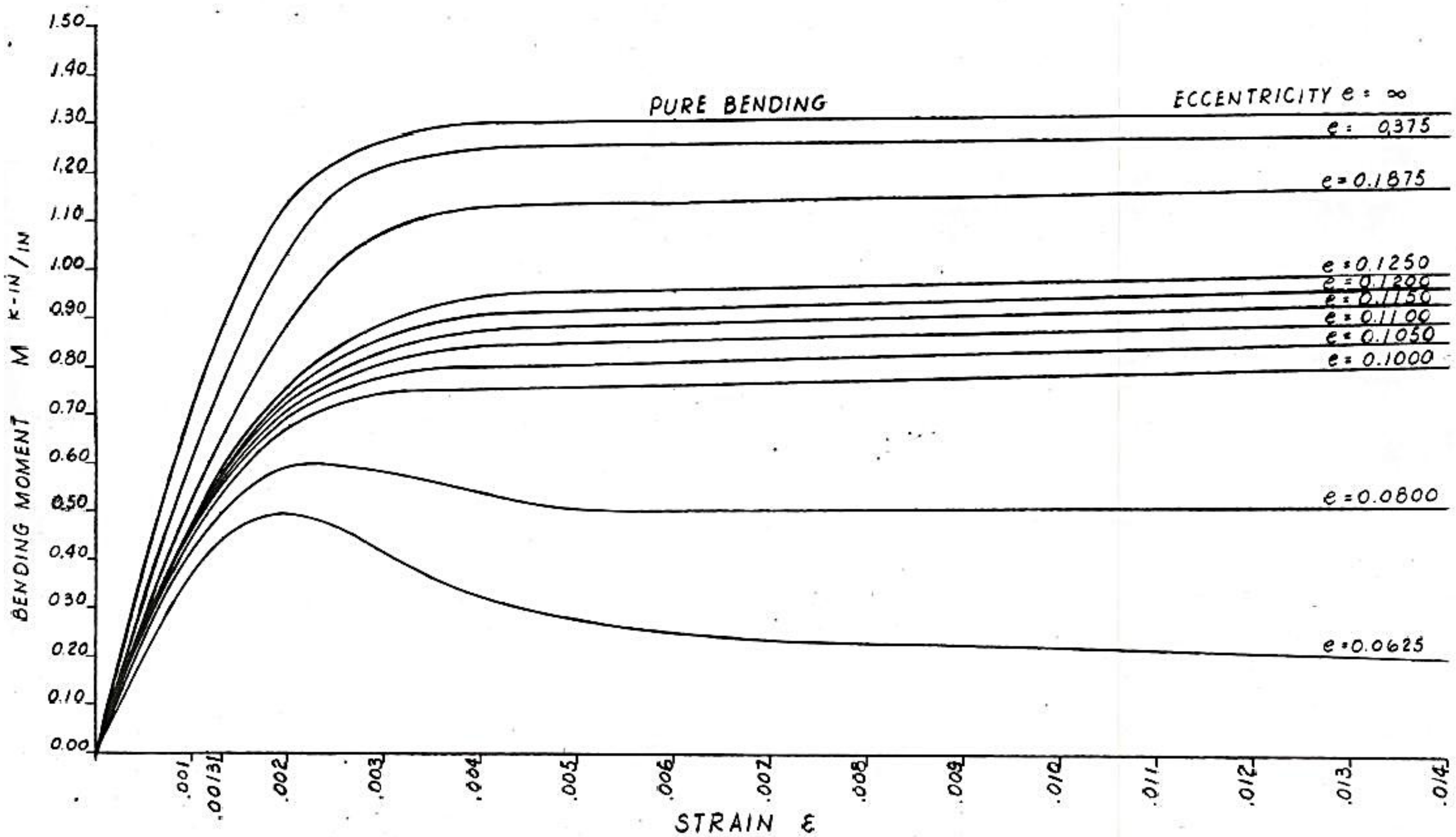


FIGURE 3.8.1-43

CONCRETE CONTAINMENT STRUCTURE 3/8" LINER FORCE – MOMENT CAPACITY DIAGRAM

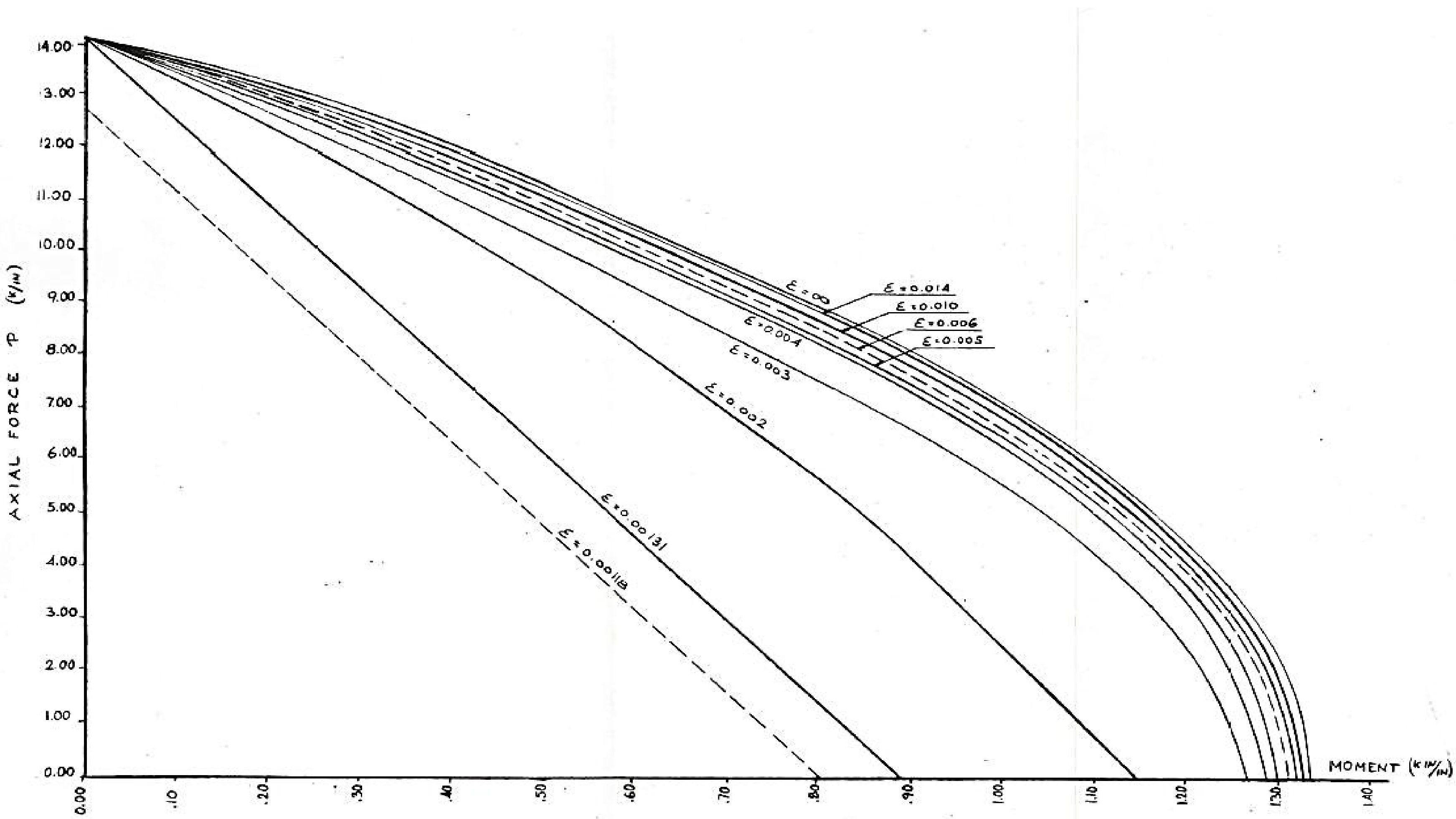


FIGURE 3.8.1-44

LINER REINFORCEMENT FOR CONCRETE PLACEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-45

CONCRETE CONTAINMENT STRUCTURE
STRUCTURAL INTEGRITY TEST – RADIAL DISPLACEMENT MEASUREMENT LOCATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-46

CONCRETE CONTAINMENT STRUCTURE
STRUCTURAL INTEGRITY TEST – VERTICAL DISPLACEMENT MEASUREMENT LOCATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-47

CONCRETE CONTAINMENT STRUCTURE
STRUCTURAL INTEGRITY TEST – WALL STRAIN MEASUREMENT LOCATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-48

CONCRETE CONTAINMENT INTEGRITY TEST
PENETRATION STRAIN MEASUREMENT LOCATIONS SHEET 1

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-49

CONCRETE CONTAINMENT – INTEGRITY TEST -
PENETRATION STRAIN MEASUREMENT LOCATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.1-50

CONCRETE CONTAINMENT STRUCTURE STRUCTURAL INTEGRITY TEST - CRACK MAPPING & TEMPERATURE MEASUREMENT LOCATIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.2-1

CONCRETE CONTAINMENT STRUCTURE SUMP PENETRATION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.2-2

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT MODEL OF EQUIPMENT HATCH

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.2-3

CONTAINMENT PENETRATIONS TEST OF 1” DIAMETER X 16”
LONG BENT ANCHORAGE IN TENSION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.2-4

CONTAINMENT PENETRATIONS TEST OF 1” DIAMETER X 16” LONG
BENT ANCHORAGE IN SHEAR IN PLANE OF CURVATURE

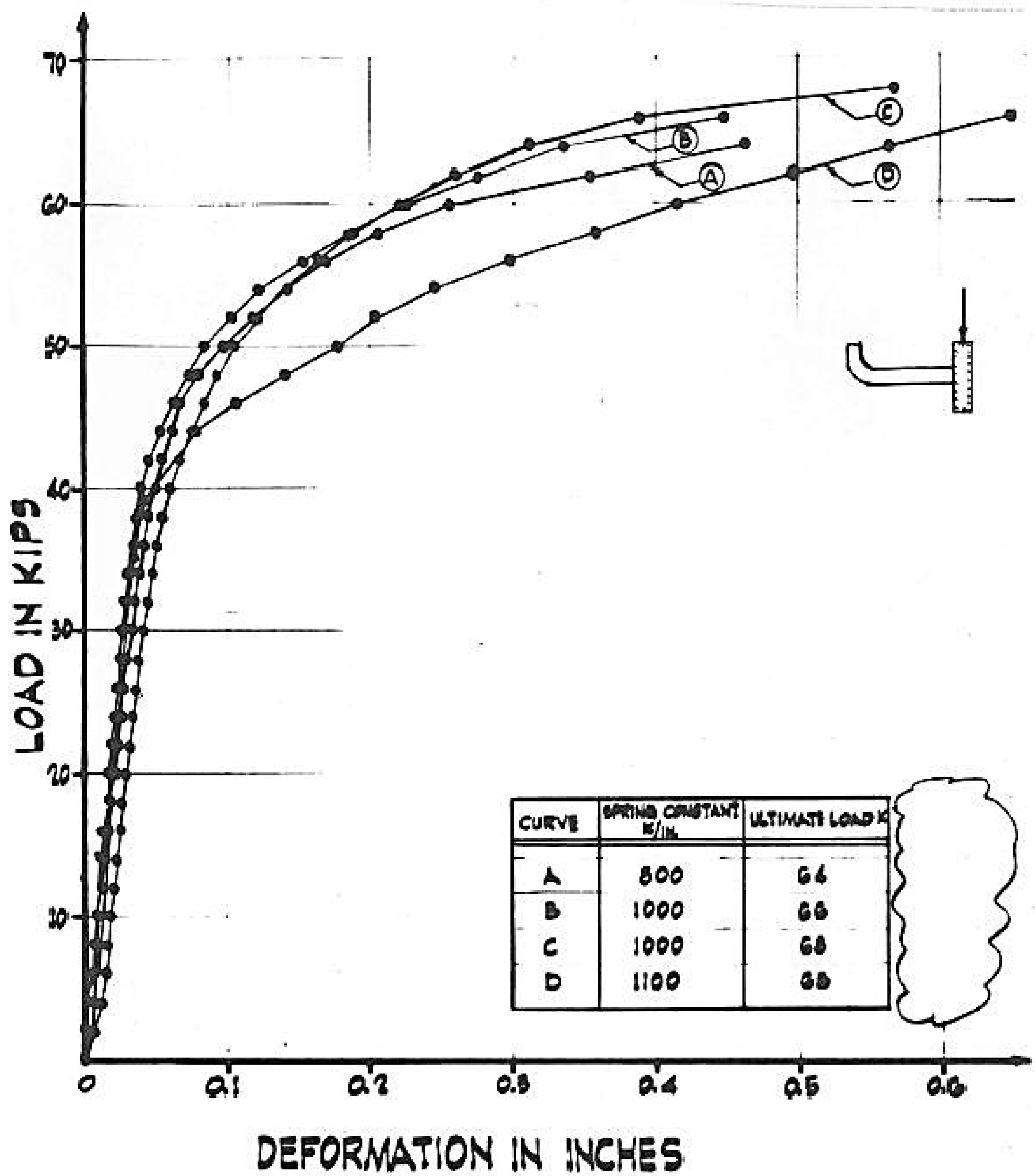


FIGURE 3.8.2-5

CONTAINMENT PENETRATIONS TEST OF 1” DIAMETER X 16” LONG BENT ANCHORAGE IN SHEAR PERPENDICULAR TO THE PLAN OF CURVATURE

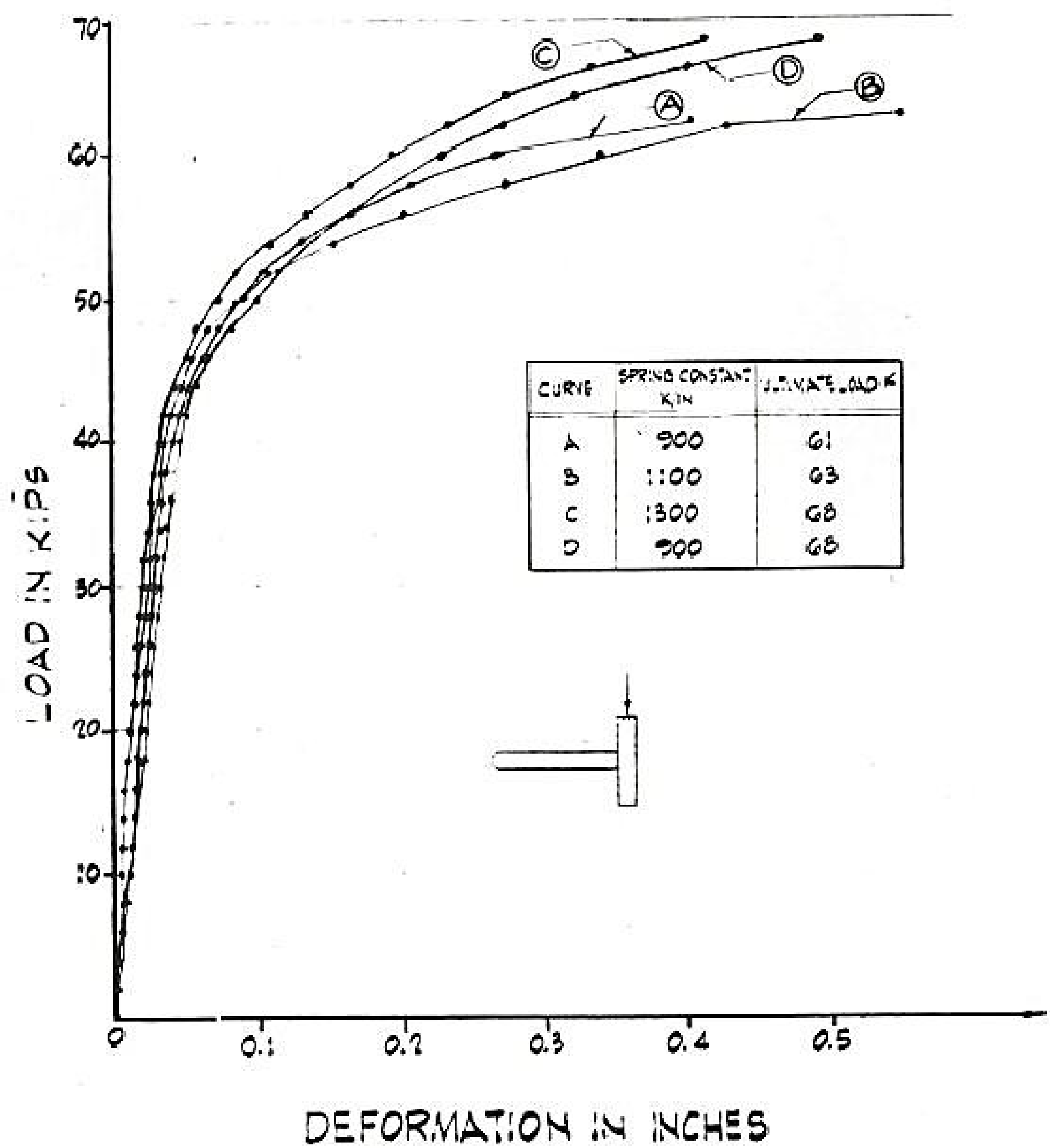


FIGURE 3.8.2-6

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT OF MODEL OF PERSONNEL LOCK

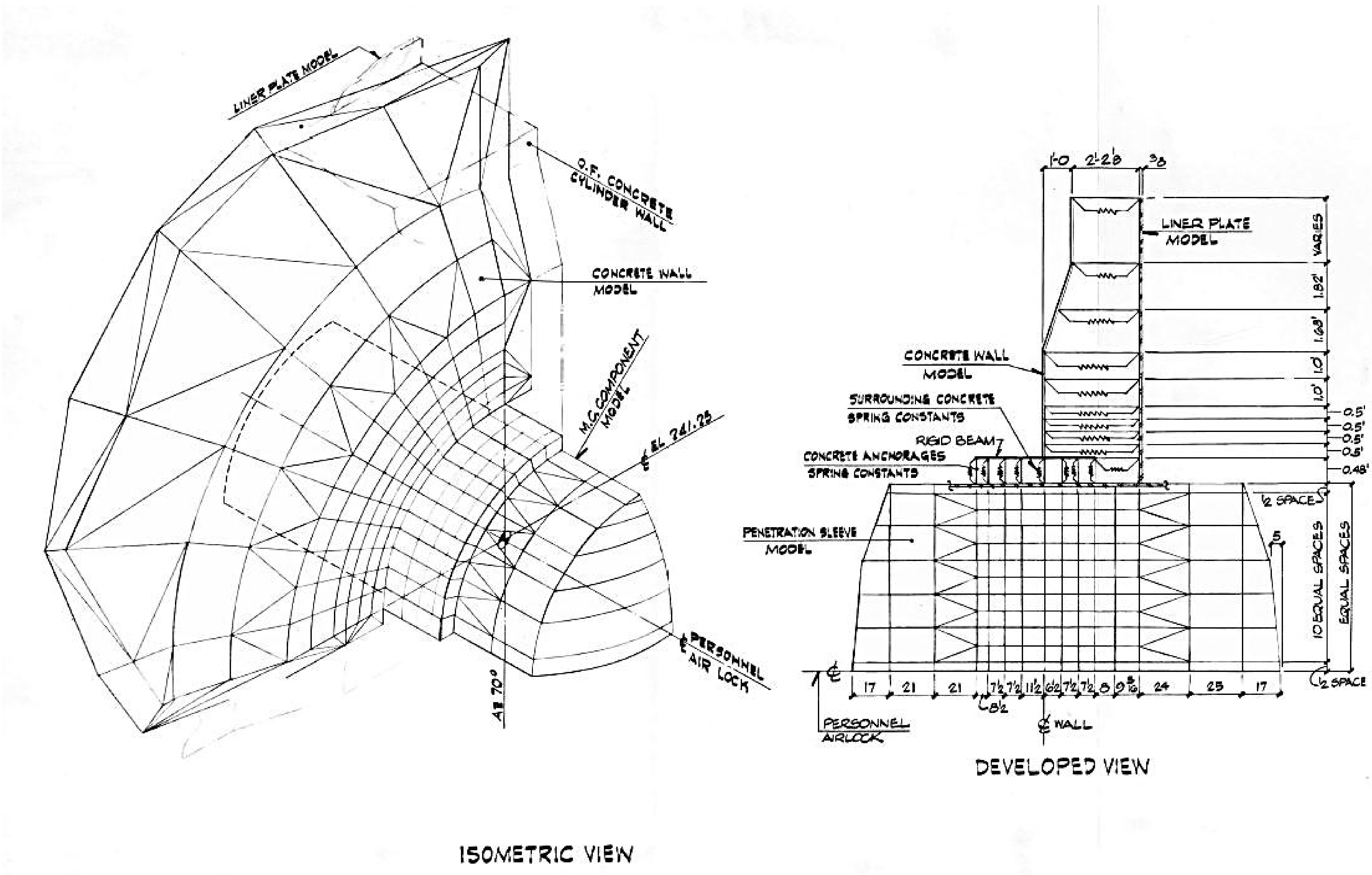
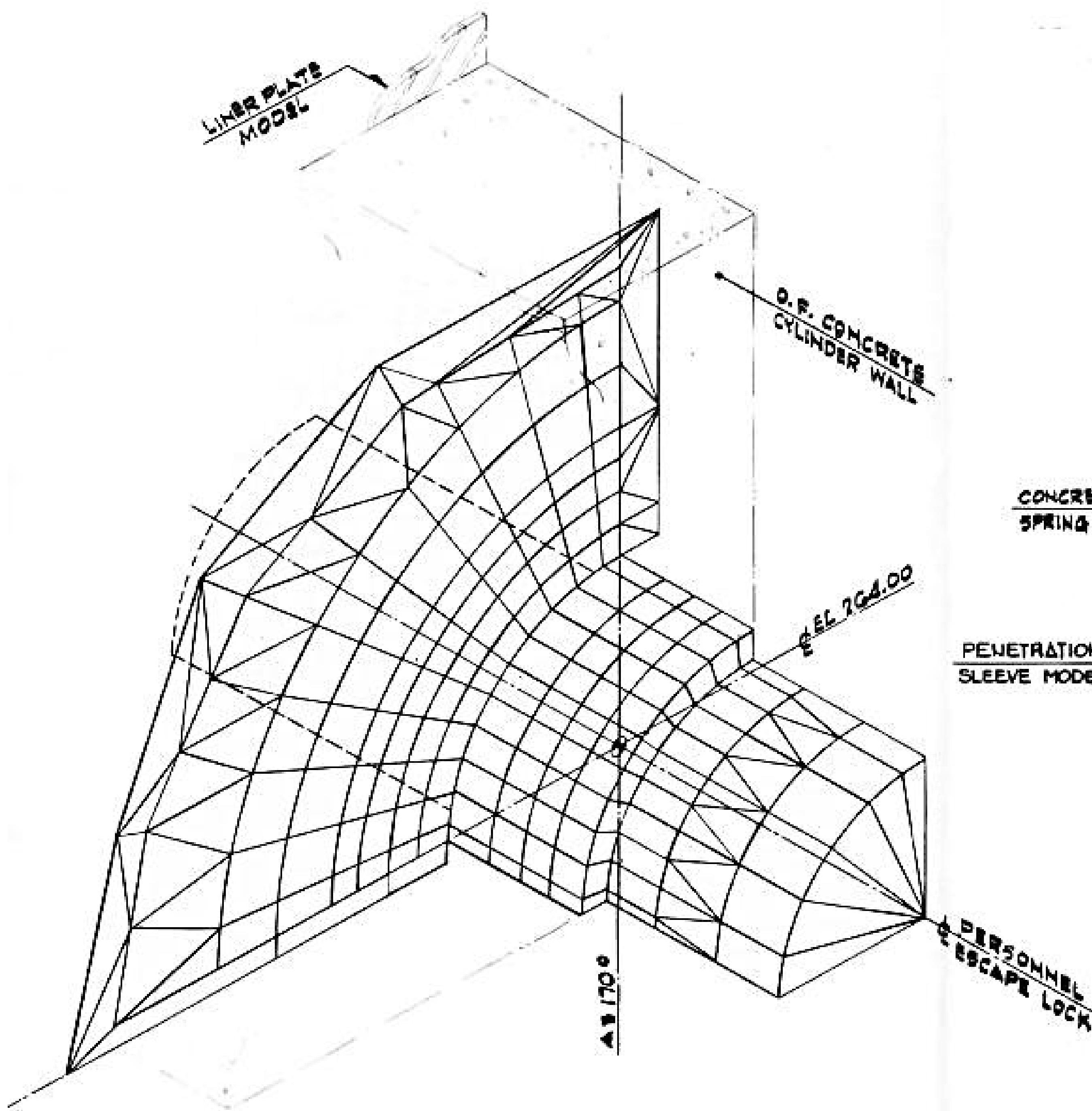
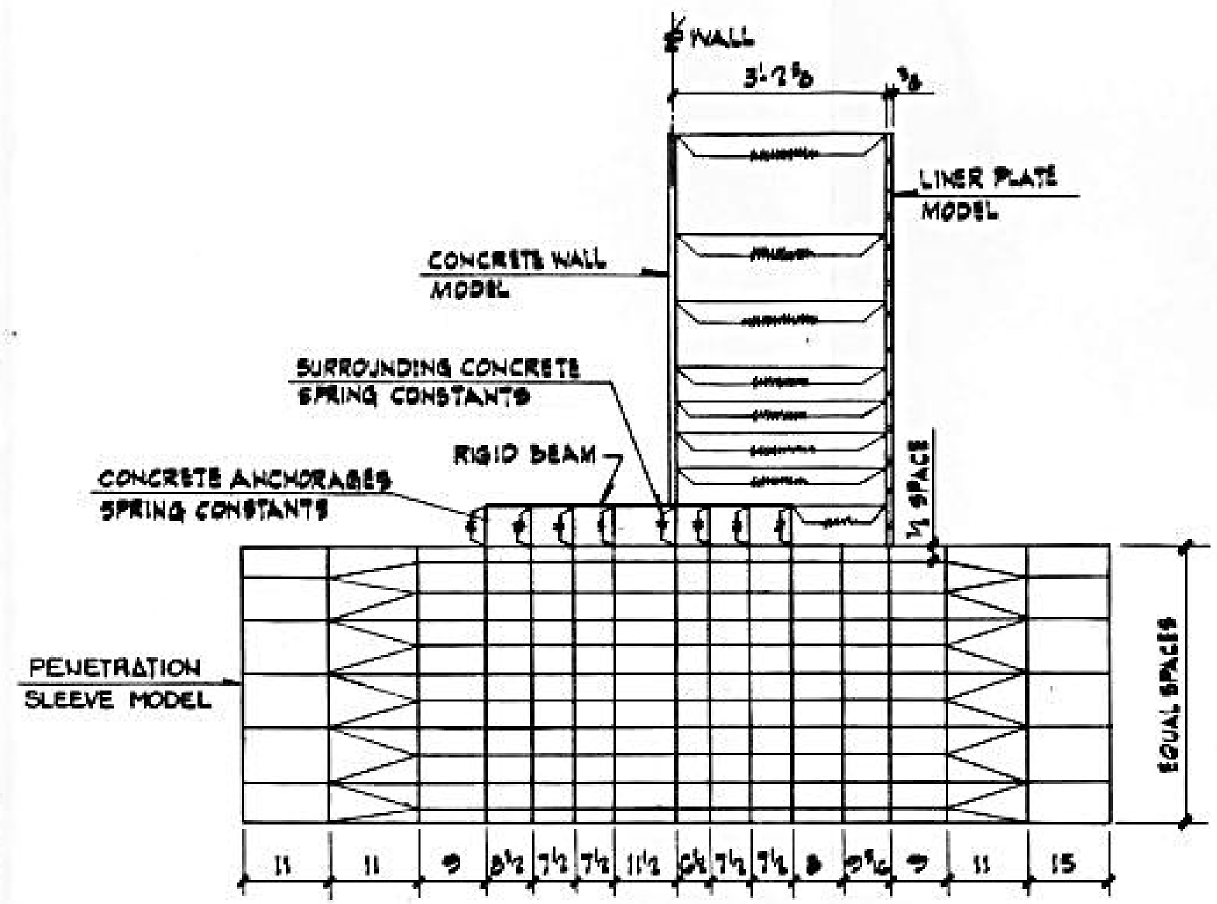


FIGURE 3.8.2-7

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT OF MODEL ESCAPE LOCK



ISOMETRIC VIEW



DEVELOPED VIEW

FIGURE 3.8.2-9

CONTAINMENT PENETRATIONS TEST OF MS & FW PENETRATION ATTACHMENT IN TENSION

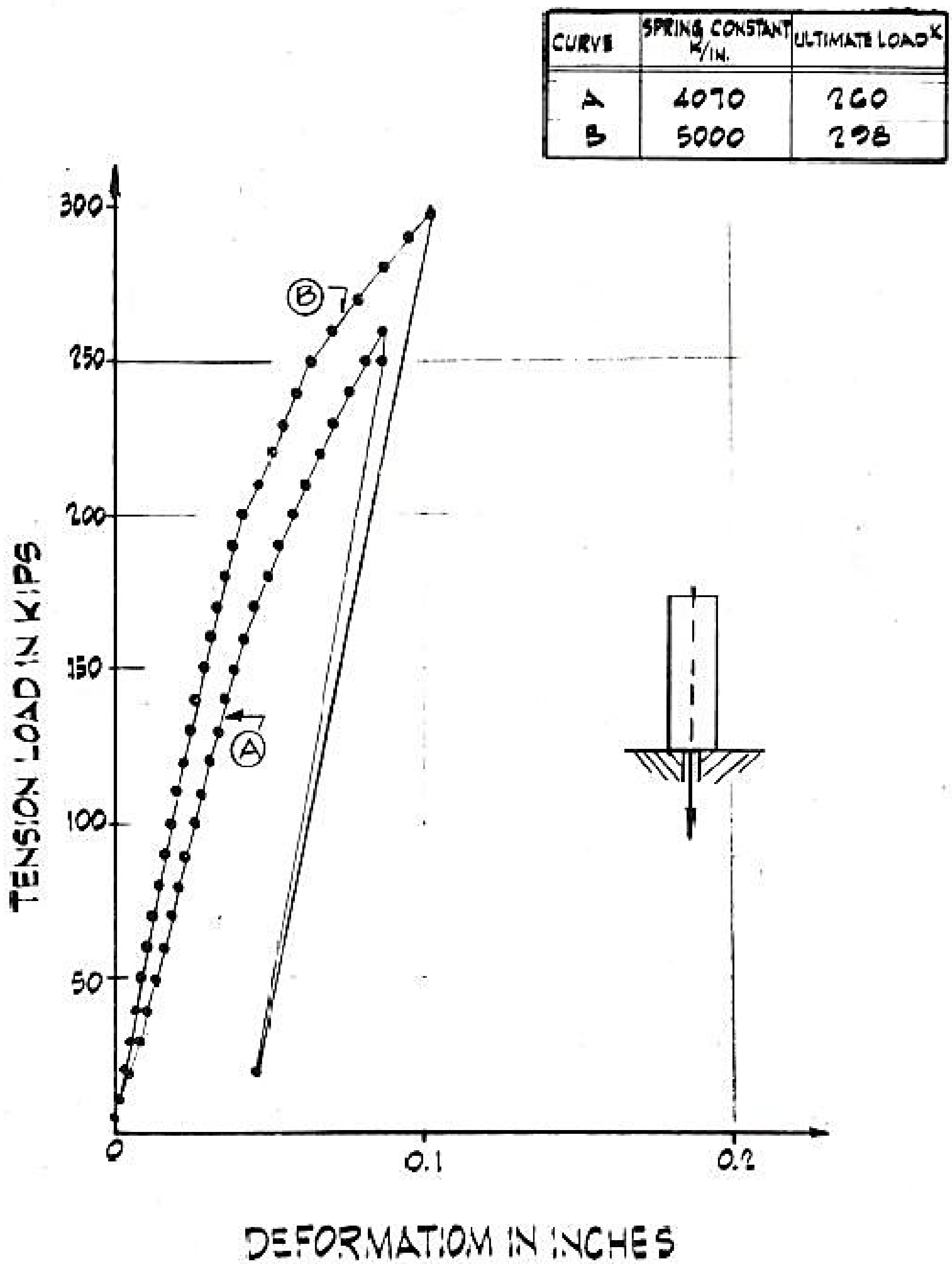


FIGURE 3.8.2-10

CONTAINMENT PENETRATIONS TEST OF MS & FW PENETRATION ATTACHMENT IN SHEAR PERPENDICULAR TO THE PLATE

CURVE	SPRING CONSTANT K/IN	ULTIMATE LOAD K
E	4100	295
F	4100	288

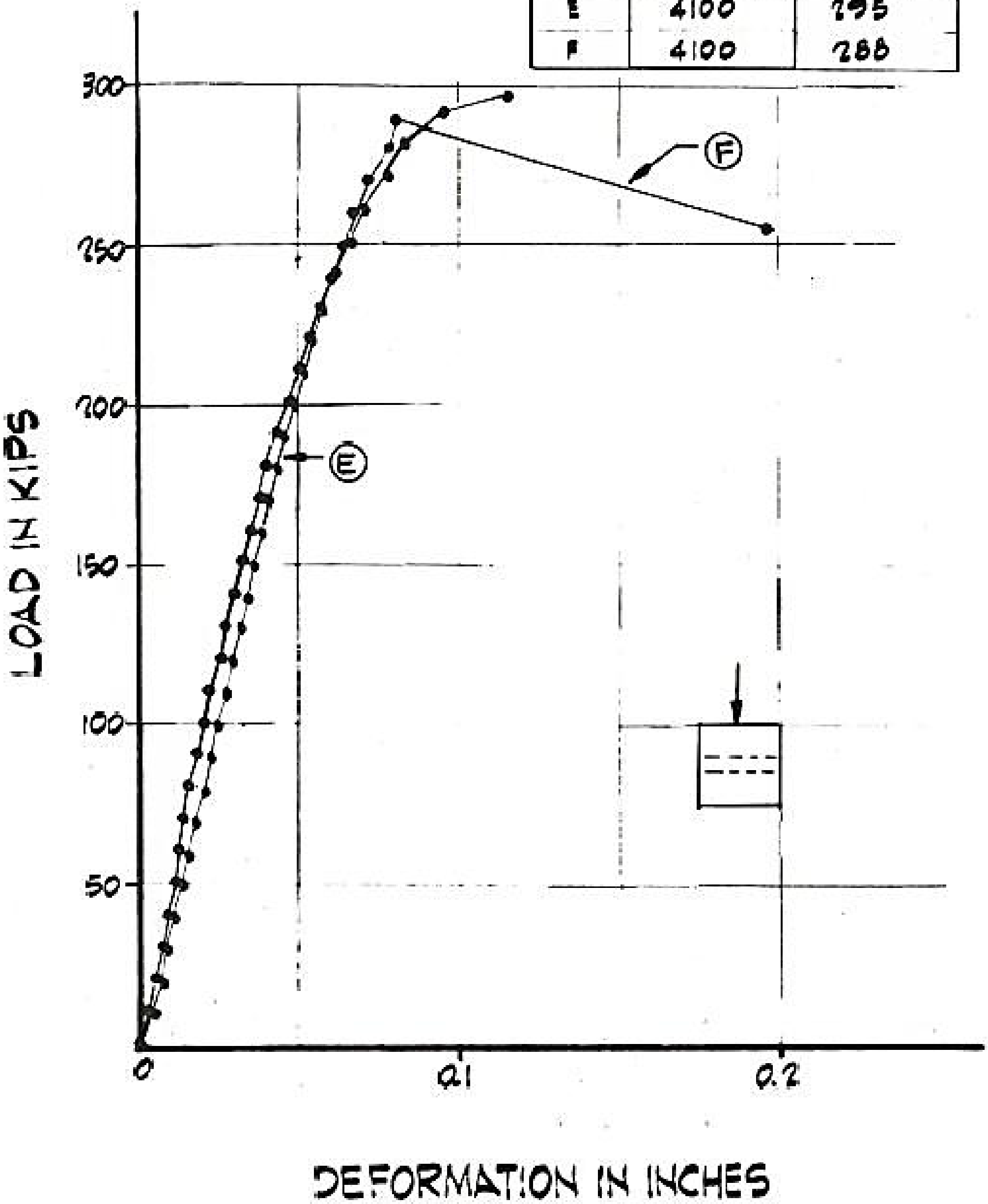
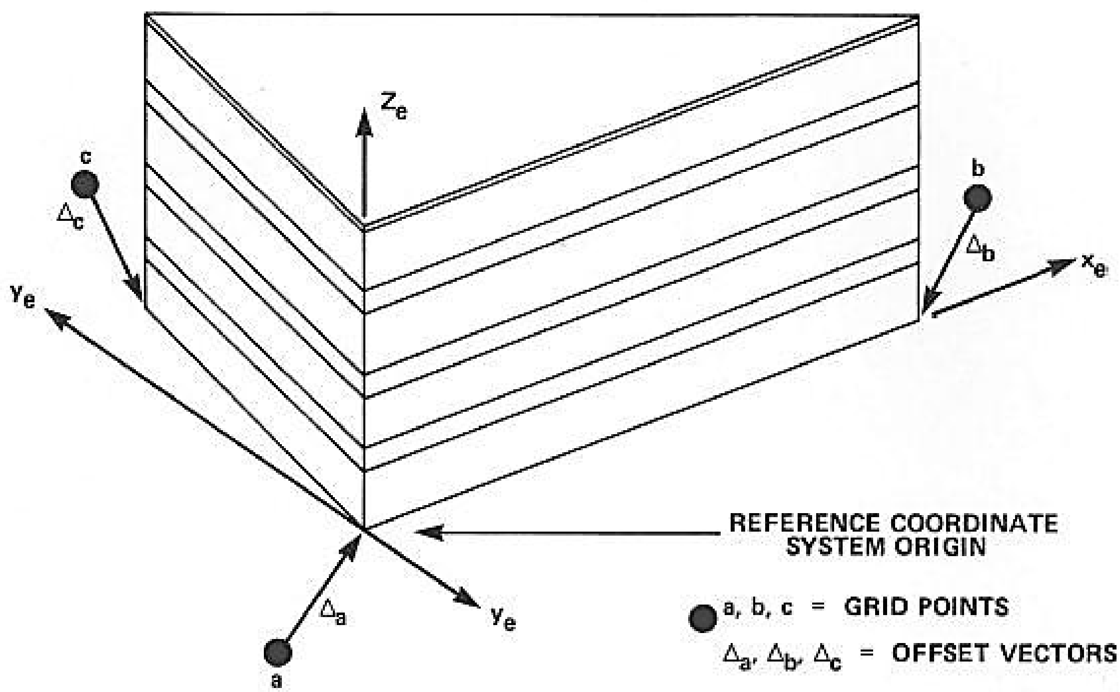


FIGURE 3.8.2-11

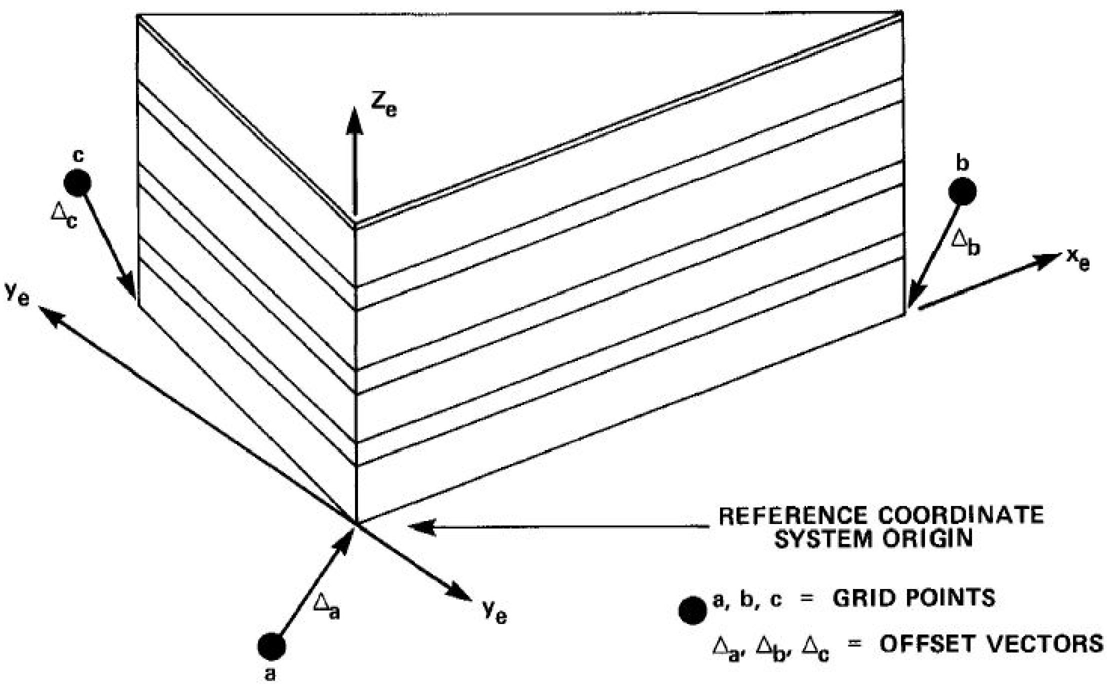
CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT MODEL OF CONCRETE CRACKING ELEMENT



TRIANGULAR LAYERED ELEMENT

FIGURE 3.8.2-14

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT MODEL OF CONCRETE CRACKING ELEMENT



TRIANGULAR LAYERED ELEMENT

FIGURE 3.8.2-15

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT MODEL OF MAIN STEAM & FEEDWATER AREA SUBSTRUCTURE

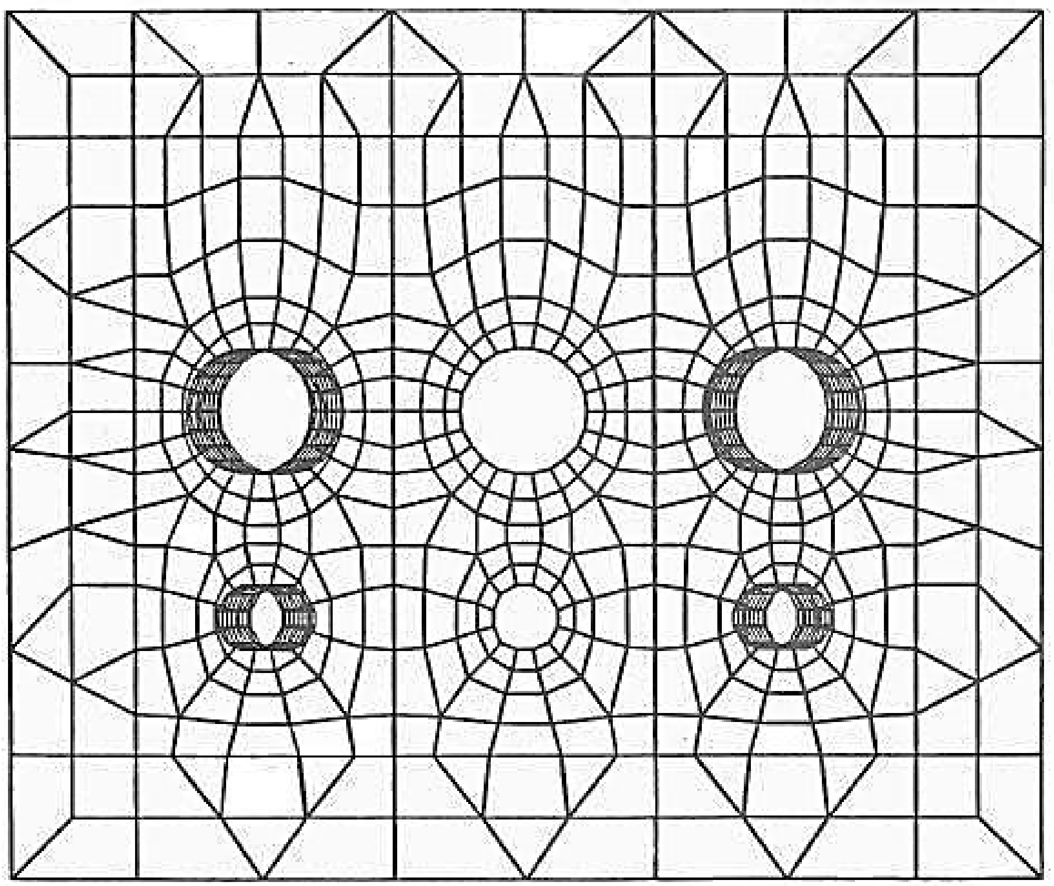


FIGURE 3.8.2-16

CONCRETE CONTAINMENT STRUCTURE FINITE ELEMENT MODEL
OF SUBSTRUCTURE OF MAIN STEAM & FEEDWATER ASSEMBLIES

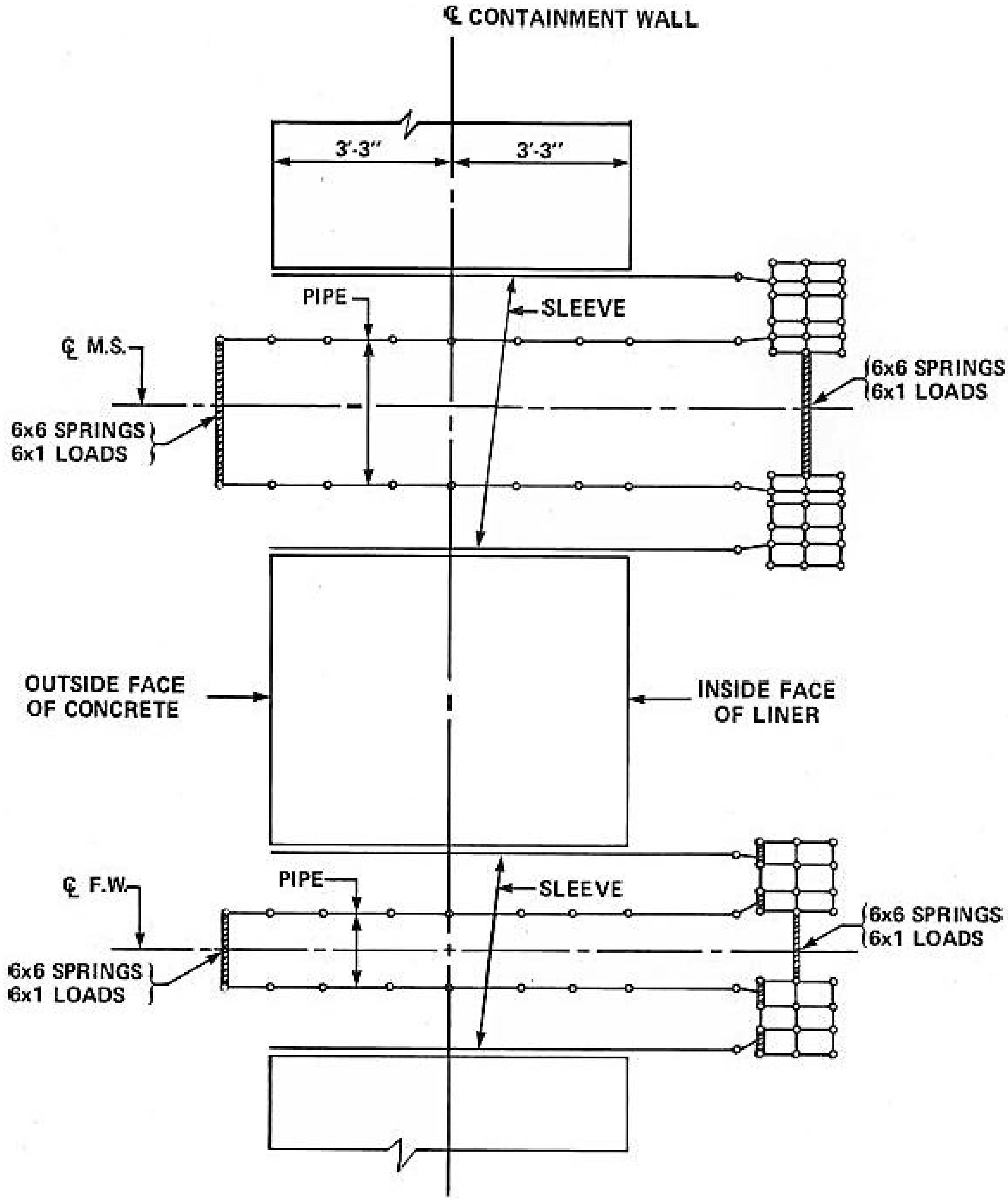


FIGURE 3.8.2-17

CONCRETE CONTAINMENT STRUCTURE SPRING CONNECTION BETWEEN MAIN STEAM PIPE SLEEVE & CONTAINMENT WALL

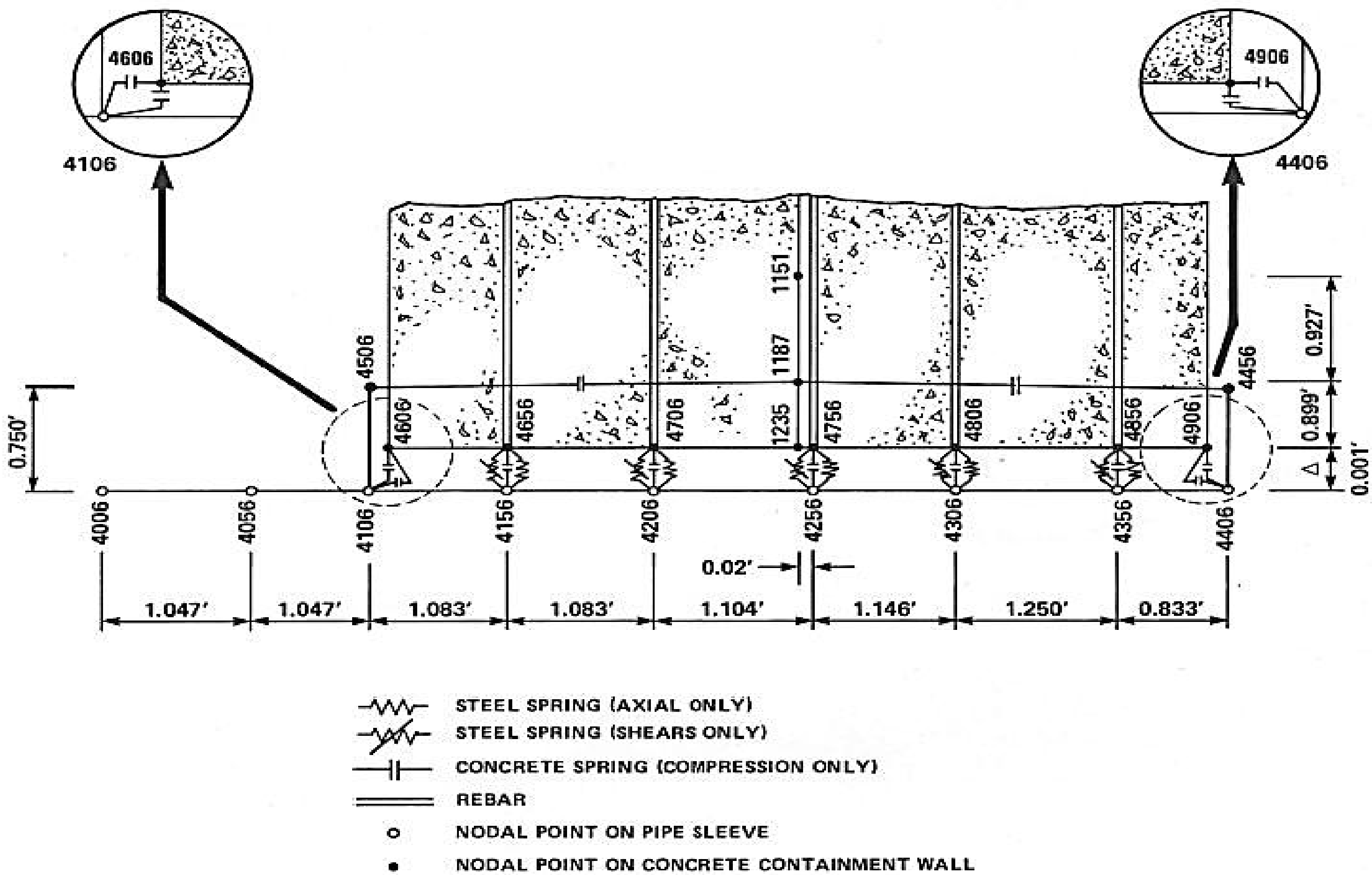


FIGURE 3.8.2-18

CONCRETE CONTAINMENT STRUCTURE ELEVATION OF SPRING CONNECTION BETWEEN MAIN STEAM PIPE SLEEVE & CONTAINMENT WALL

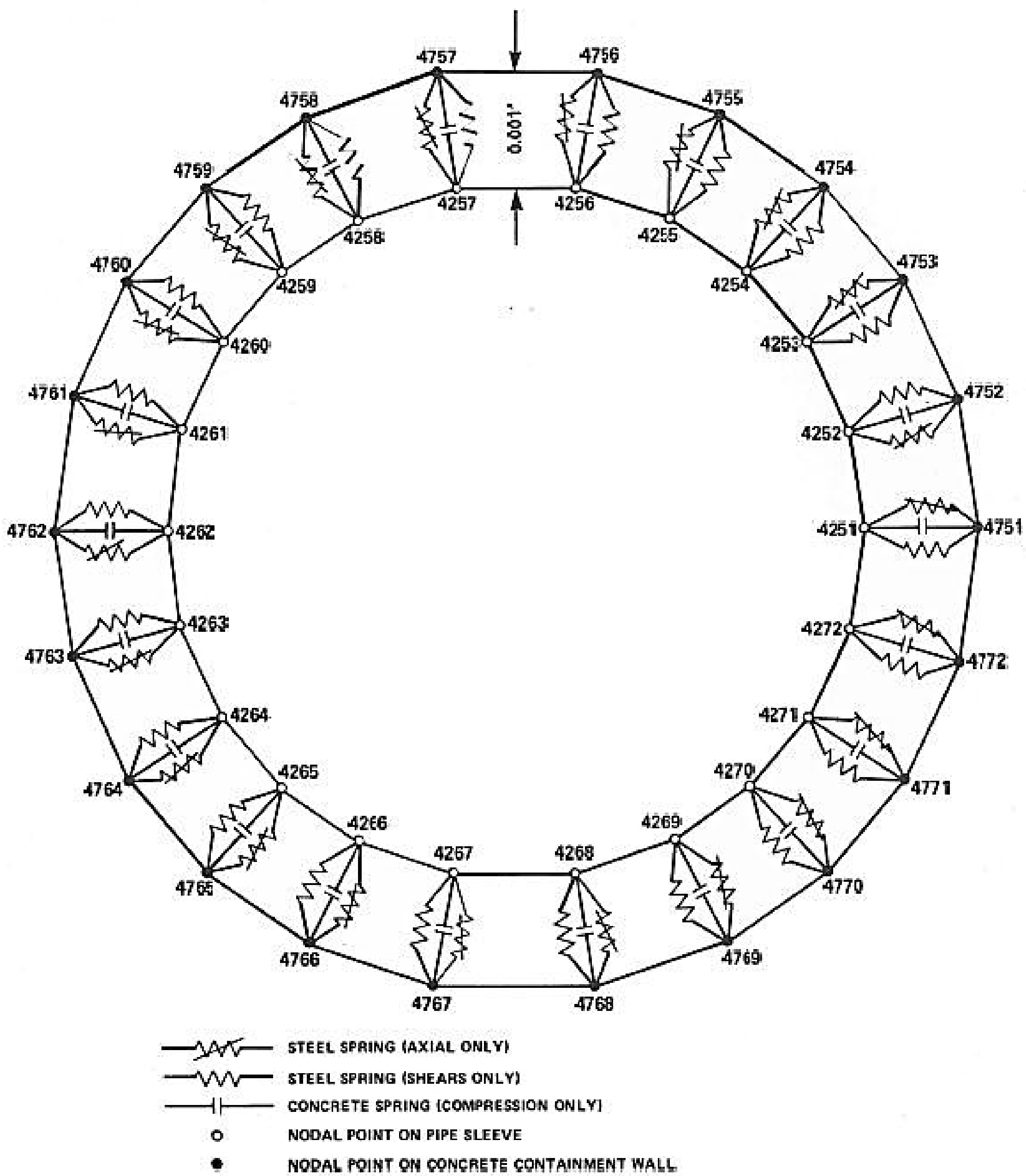
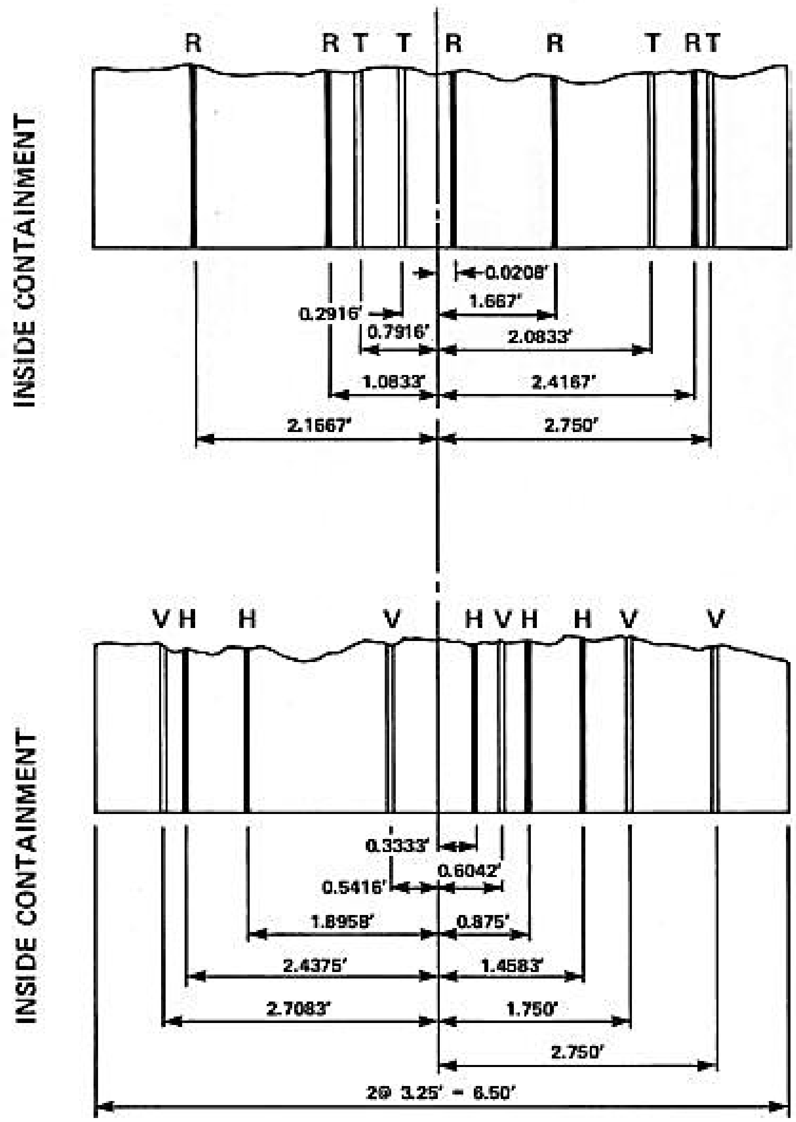


FIGURE 3.8.2-19

CONCRETE CONTAINMENT STRUCTURE TYPICAL CONCRETE SECTION WITH REBARS FOR NON-LINEAR CRACKING ELEMENT



NOTATION:

R = RADIAL REBAR

T = TANGETIAL REBAR

H = HORIZONTAL REBAR

V = VERTICAL REBAR

FIGURE 3.8.3-1

CONCRETE CONTAINMENT INTERNAL STRUCTURES
GENERAL ARRANGEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-2

CONCRETE CONTAINMENT INTERNAL STRUCTURE GENERAL ARRANGEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-3

CONCRETE CONTAINMENT INTERNAL STRUCTURE – PRIMARY SHIELD WALL REINFORCEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-4

CONCRETE CONTAINMENT INTERNAL STRUCTURE SECONDARY SHIELD WALL REINFORCEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-5

CONCRETE CONTAINMENT INTERNAL STRUCTURE REFUELING CAVITY WALL REINFORCEMENT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-6

CONCRETE CONTAINMENT INTERIOR STRUCTURE – MAT MASONRY & REINFORCING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-8

CONCRETE CONTAINMENT INTERNAL STRUCTURES
STEAM GENERATOR AND R.C. PUMP PEDESTALS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-9

REACTOR VESSEL SUPPORT SYSTEM

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-10

STEAM GENERATOR SUPPORT SYSTEM

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-13

PRIMARY SHIELD WALL DESIGN PRESSURE DISTRIBUTION

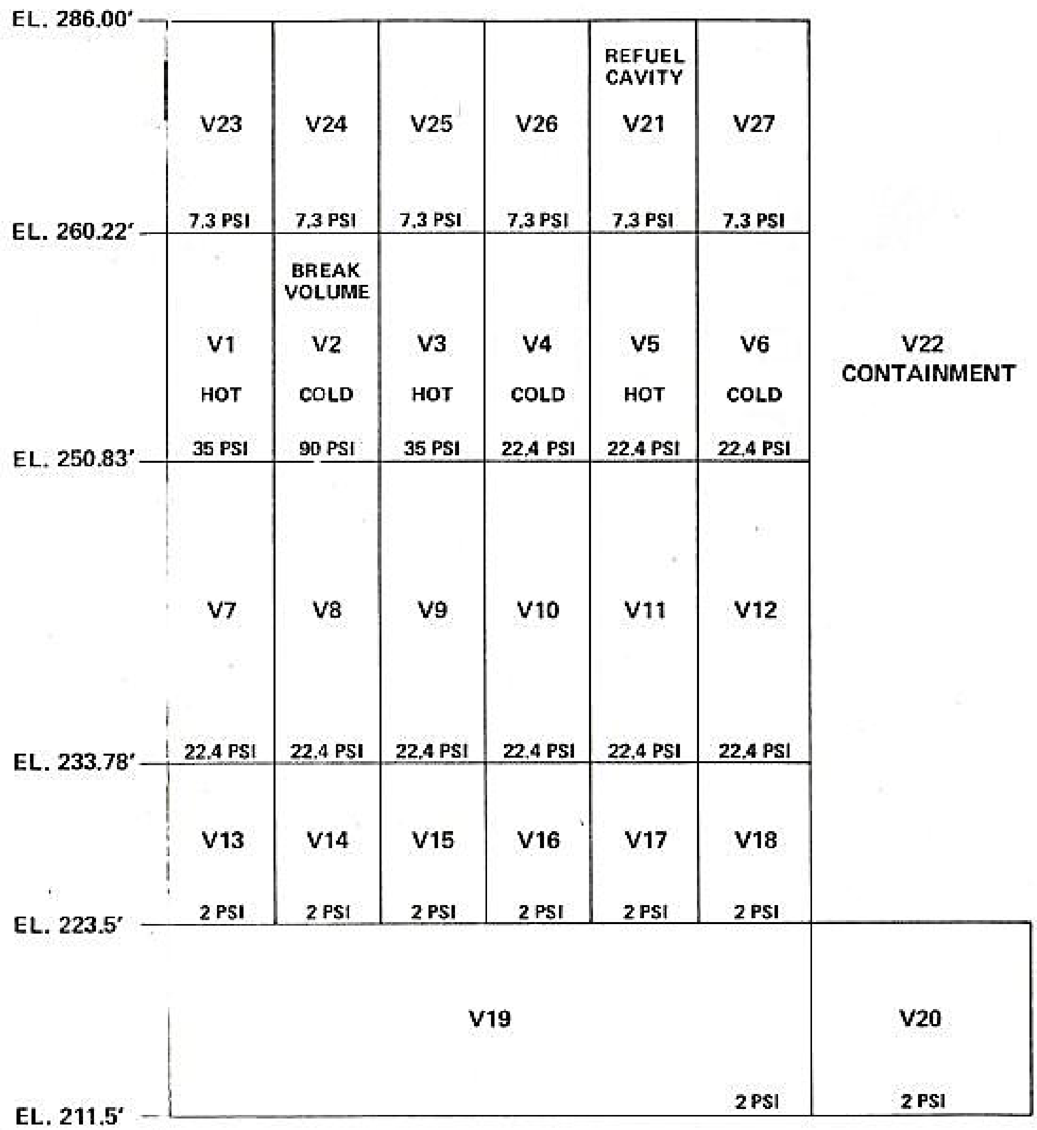


FIGURE 3.8.3-14

STEAM GENERATOR PRESSURIZER SUBCOMPARTMENT DESIGN PRESSURE DISTRIBUTION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-15

CRDM ROOM WALL, DESIGN PRESSURE DISTRIBUTION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.3-16

CONCRETE CONTAINMENT TYPICAL SECTION
STRUCTURAL PLATFORM FRAMING

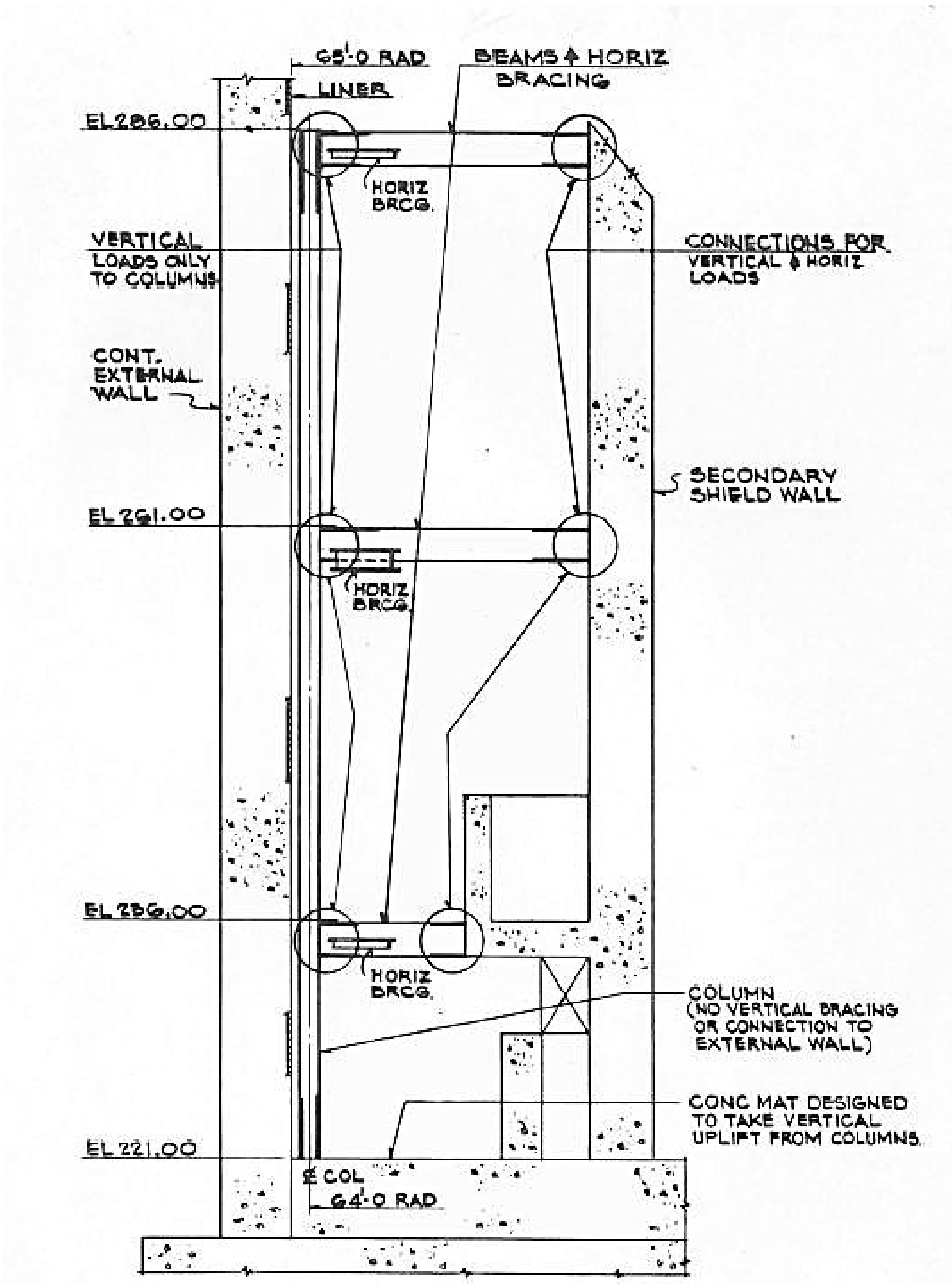


FIGURE 3.8.4-1

GENERAL LAYOUT OF SEISMIC I BUILDINGS AT PLANT ISLAND

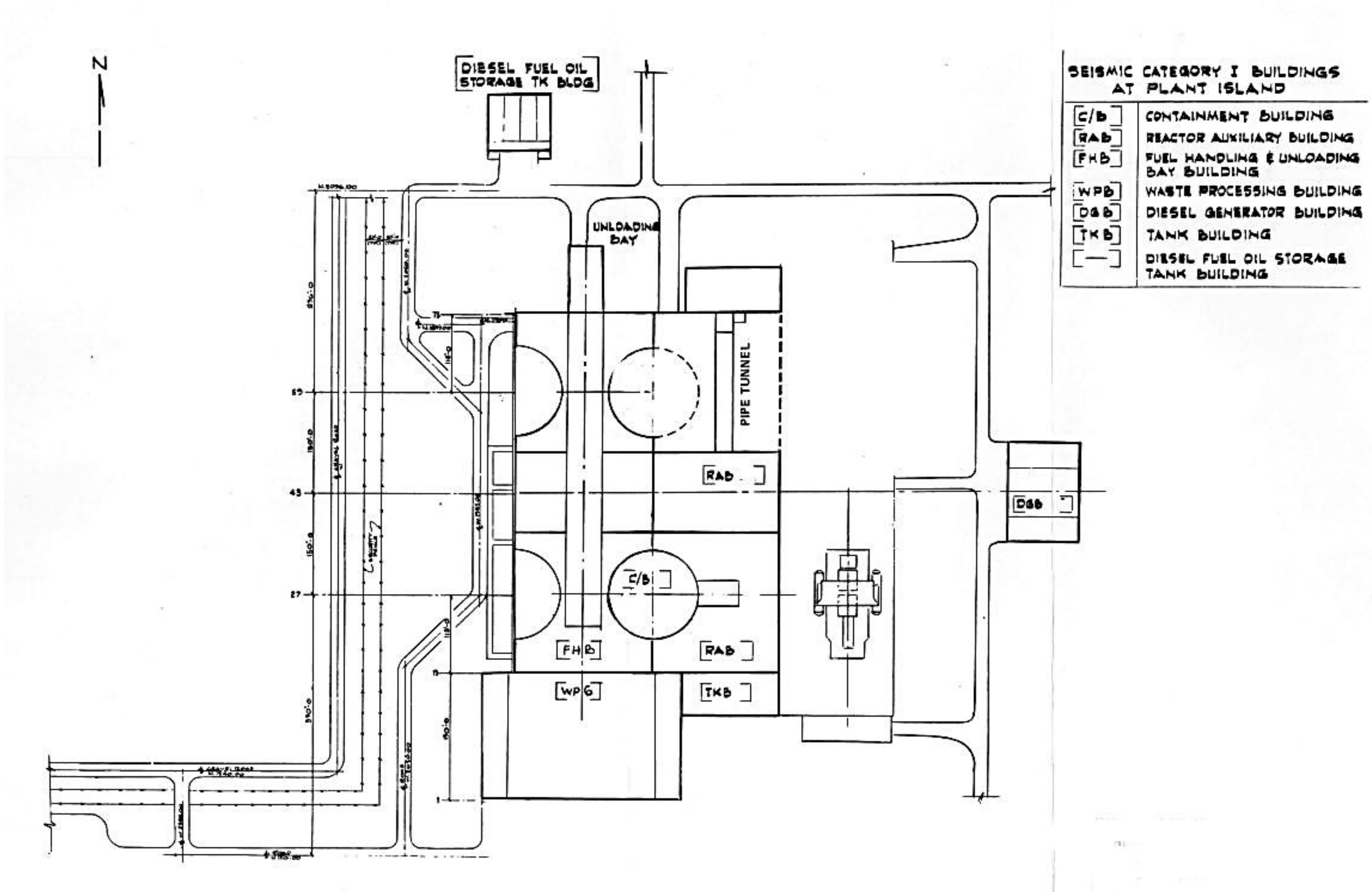


FIGURE 3.8.4-2

STRUCTURAL LAYOUT OF REACTOR AUXILIARY BUILDING – FLOOR PLANS AT EL. 190.0 AND EL. 216.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-3

STRUCTURAL LAYOUT OF REACTOR AUXILIARY BUILDING – FLOOR PLANS AT EL. 236.0 AND EL. 261.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-4

STRUCTURAL LAYOUT OF REACTOR AUXILIARY BUILDING – FLOOR PLANS AT EL. 286.0 AND EL. 305.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-5

STRUCTURAL LAYOUT OF REACTOR AUXILIARY BUILDING – ROOF PLAN

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-6

STRUCTURAL LAYOUT OF REACTOR AUXILIARY BUILDING – CROSS SECTION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-7

STRUCTURAL LAYOUT OF FUEL HANDLING BUILDING – FLOOR PLAN AT EL. 216.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-8

STRUCTURAL LAYOUT OF FUEL HANDLING BUILDING – FLOOR PLAN AT EL. 236.00'

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-9

STRUCTURAL LAYOUT OF FUEL HANDLING BUILDING – FLOOR PLAN AT EL. 261.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-10

STRUCTURAL LAYOUT OF FUEL HANDLING BUILDING – FLOOR PLAN AT EL. 286.00’

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-11

STRUCTURAL LAYOUT OF FUEL HANDLING BUILDING – FLOOR PLANS AT EL. 305.0 AND EL 324.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-12

STRUCTURAL LAYOUT OF FUEL HANDLING BUILDING – ROOF PLAN

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-13

STRUCTURE LAYOUT OF FUEL HANDLING BUILDING – CROSS SECTION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-14

STRUCTURAL LAYOUT OF FUEL HANDLING BUILDING – UNLOADING AREA PLANS AND SECTIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-15

STRUCTURAL LAYOUT OF WASTE PROCESSING BUILDING – FLOOR PLAN AT EL. 211.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-16

STRUCTURAL LAYOUT OF WASTE PROCESSING BUILDING – FLOOR PLAN AT EL. 236.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-17

STRUCTURAL LAYOUT OF WASTE PROCESSING BUILDING – FLOOR PLANS AT EL. 261.0 AND 276.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-18

STRUCTURAL LAYOUT OF WASTE PROCESSING BUILDING FLOOR PLAN AT EL. 291.0

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-19

STRUCTURAL LAYOUT OF WASTE PROCESSING BUILDING – CROSS SECTION

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-20

STRUCTURAL LAYOUT OF DIESEL GENERATOR BUILDING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-21

STRUCTURAL LAYOUT OF TANK BUILDINGS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-22

STRUCTURAL LAYOUT OF DIESEL FUEL OIL STORAGE TANK BUILDING

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-33

EMERGENCY SERVICE WATER SYSTEM RETAINING WALLS – SCREEN STRUCTURE

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-38

AUXILIARY DAM SPILLWAY MASONRY

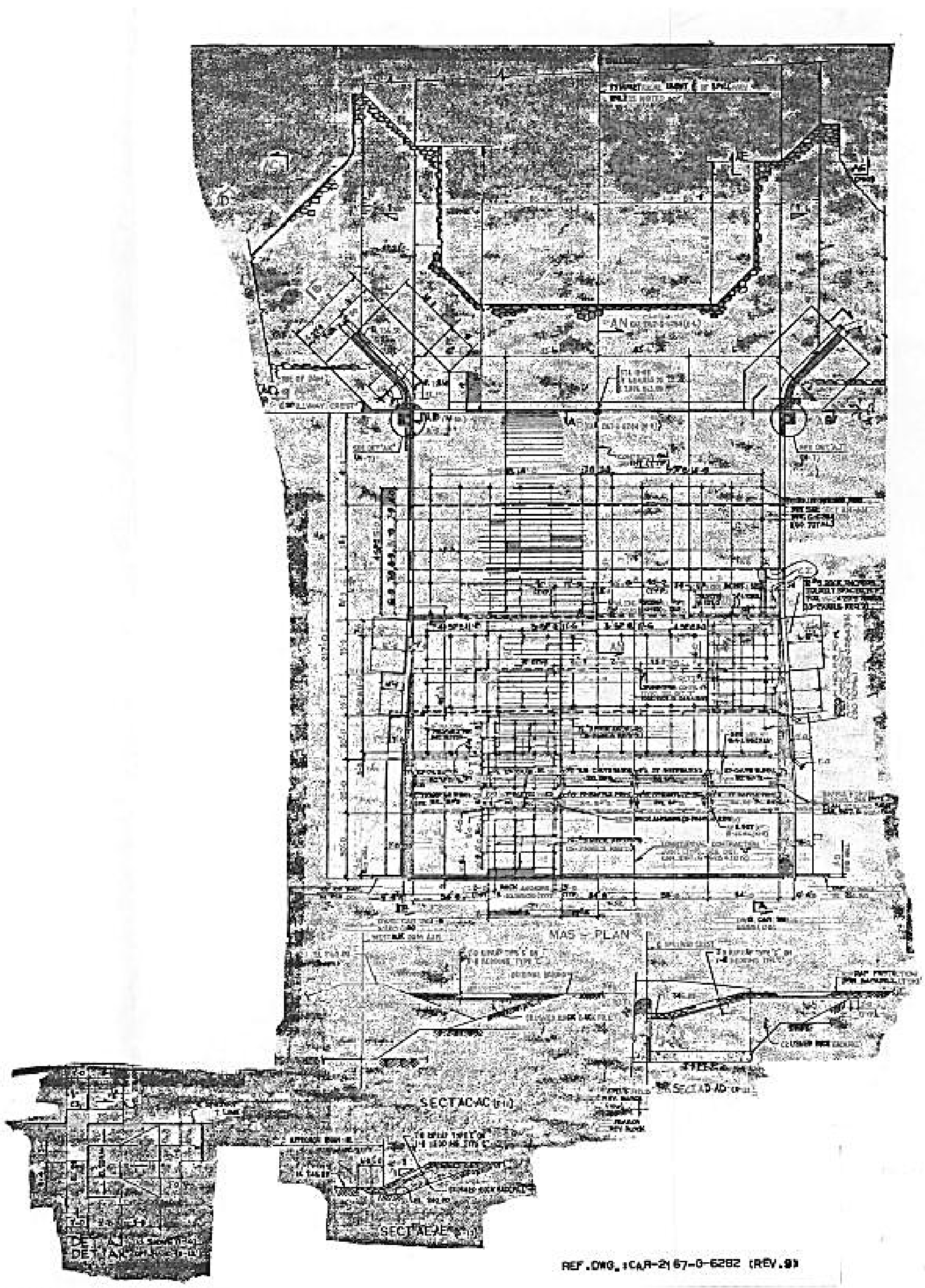
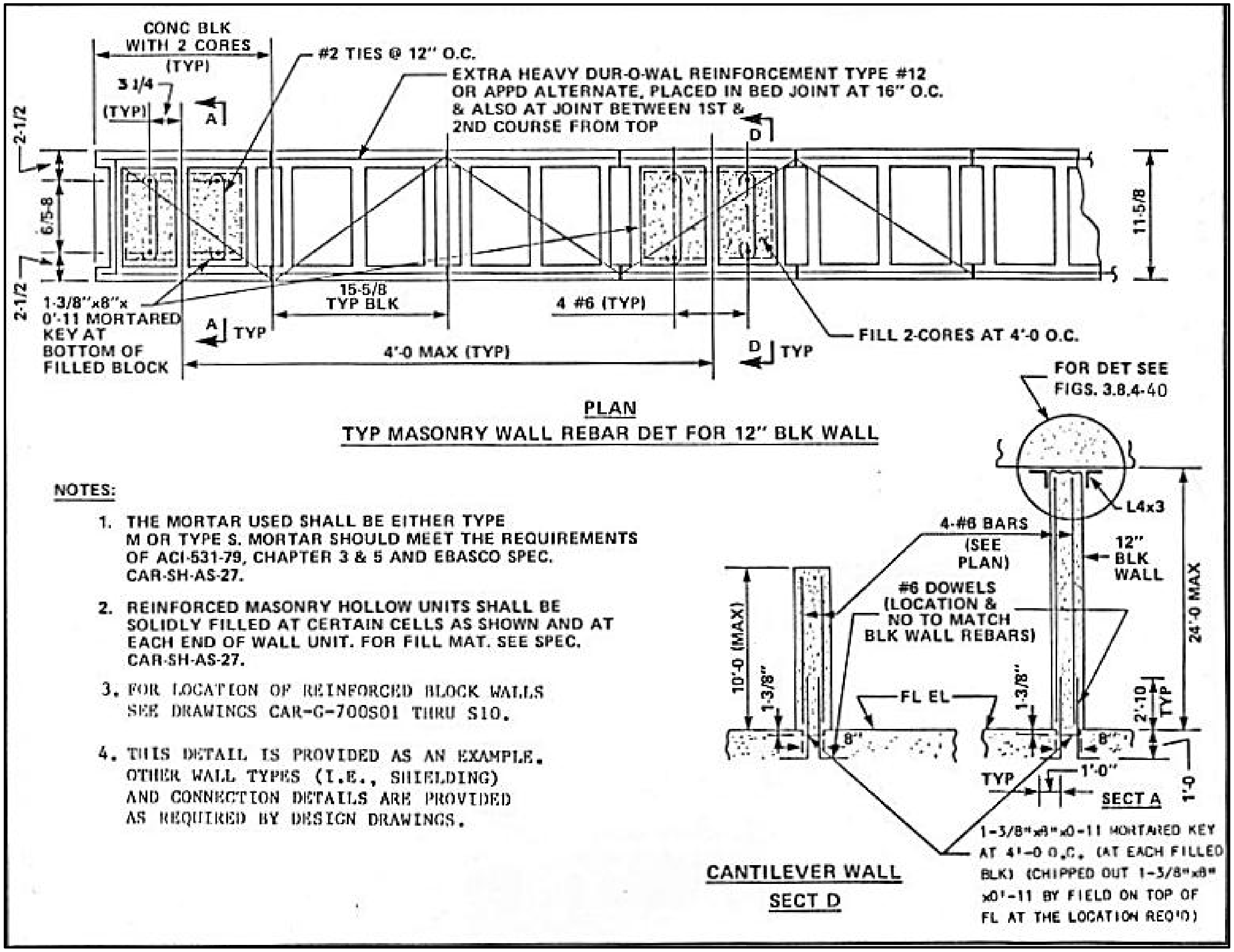


FIGURE 3.8.4-39

TYPICAL DETAILS FOR HOLLOW MASONRY BLOCK WALLS



NOTES:

1. THE MORTAR USED SHALL BE EITHER TYPE M OR TYPE S. MORTAR SHOULD MEET THE REQUIREMENTS OF ACI-531-79, CHAPTER 3 & 5 AND EBASCO SPEC. CAR-SH-AS-27.
2. REINFORCED MASONRY HOLLOW UNITS SHALL BE SOLIDLY FILLED AT CERTAIN CELLS AS SHOWN AND AT EACH END OF WALL UNIT. FOR FILL MAT. SEE SPEC. CAR-SH-AS-27.
3. FOR LOCATION OF REINFORCED BLOCK WALLS SEE DRAWINGS CAR-G-700S01 THRU S10.
4. THIS DETAIL IS PROVIDED AS AN EXAMPLE. OTHER WALL TYPES (I.E., SHIELDING) AND CONNECTION DETAILS ARE PROVIDED AS REQUIRED BY DESIGN DRAWINGS.

FIGURE 3.8.4-40

TYPICAL DETAILS FOR HOLLOW MASONRY BLOCK WALLS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-41

ESW AND CT MAKE UP INTAKE STRUCTURE – FINAL CONFIGURATION

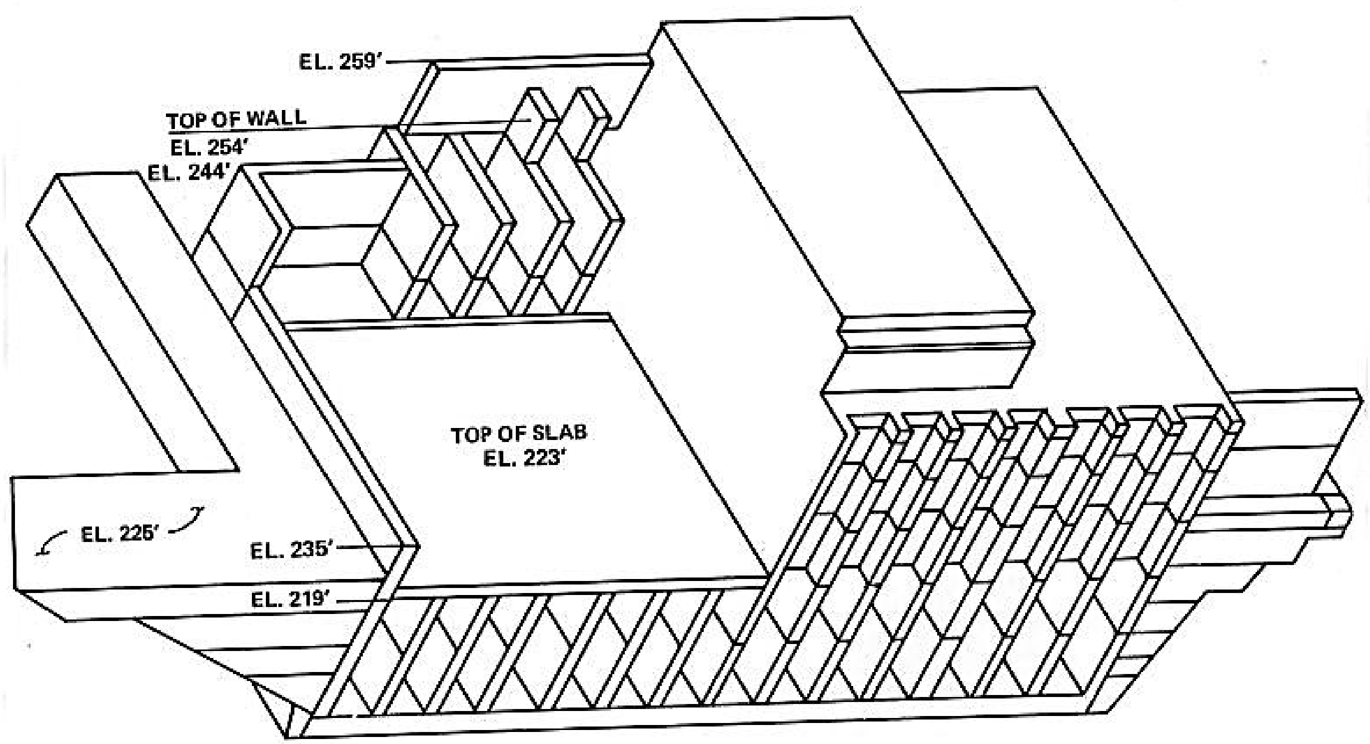


FIGURE 3.8.4-42

FUEL HANDLING BUILDING RETAINING WALL PLAN AND SECTIONS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.4-45

FUEL HANDLING BUILDING RETAINING WALL NORTHEAST SIDE

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.5-1

GENERAL LAYOUT OF SEISMIC CATEGORY I BUILDING FOUNDATION AT PLANT ISLAND

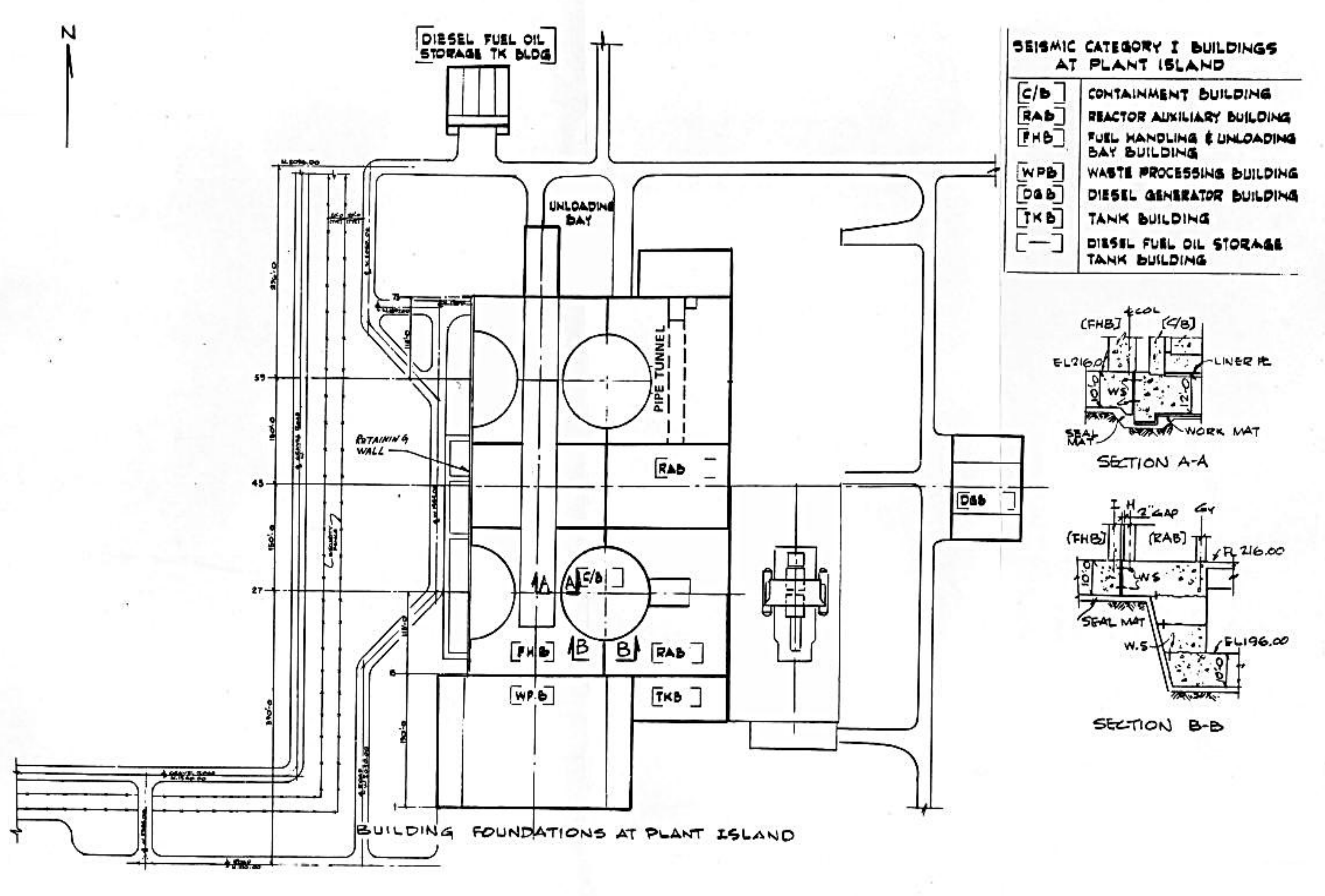


FIGURE 3.8.5-2

GENERAL LAYOUT OF WATERSTOP, WATERPROOFING MEMBRANE FOR THE FOUNDATION MATS AND WALLS

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.5-5

GENERAL LAYOUT OF THE REACTOR AUXILIARY BUILDING – FOUNDATION MAT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.5-6

GENERAL LAYOUT OF THE FUEL HANDLING BUILDING – FOUNDATION MAT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.8.5-7

GENERAL LAYOUT OF THE WASTE PROCESSING BUILDING – FOUNDATION MAT

Security-Related Information - Figure Withheld Under 10 CFR 2.390

FIGURE 3.9.1-1

REACTOR COOLANT LOOP SUPPORTS SYSTEM DYNAMIC STRUCTURAL MODEL

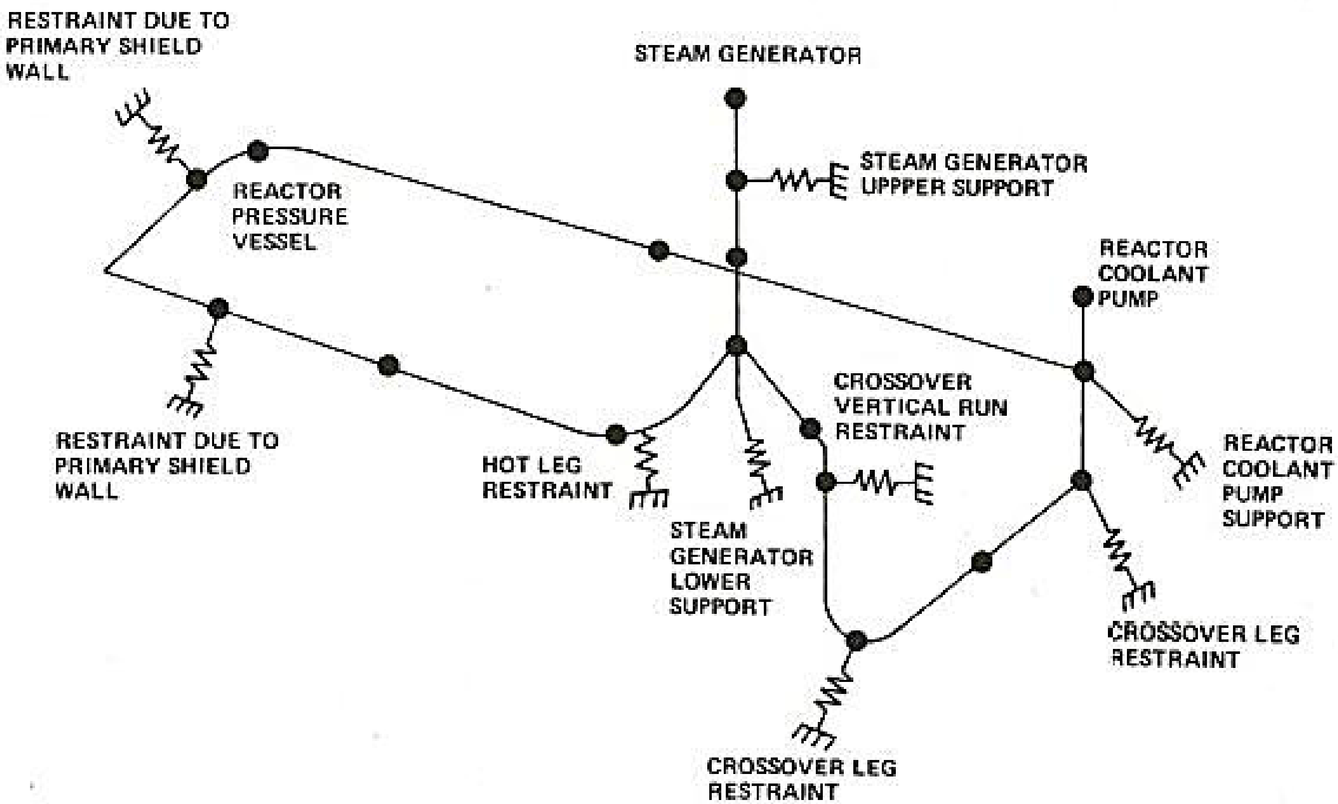


FIGURE 3.9.1-2

THROUGH-WALL THERMAL GRADIENTS

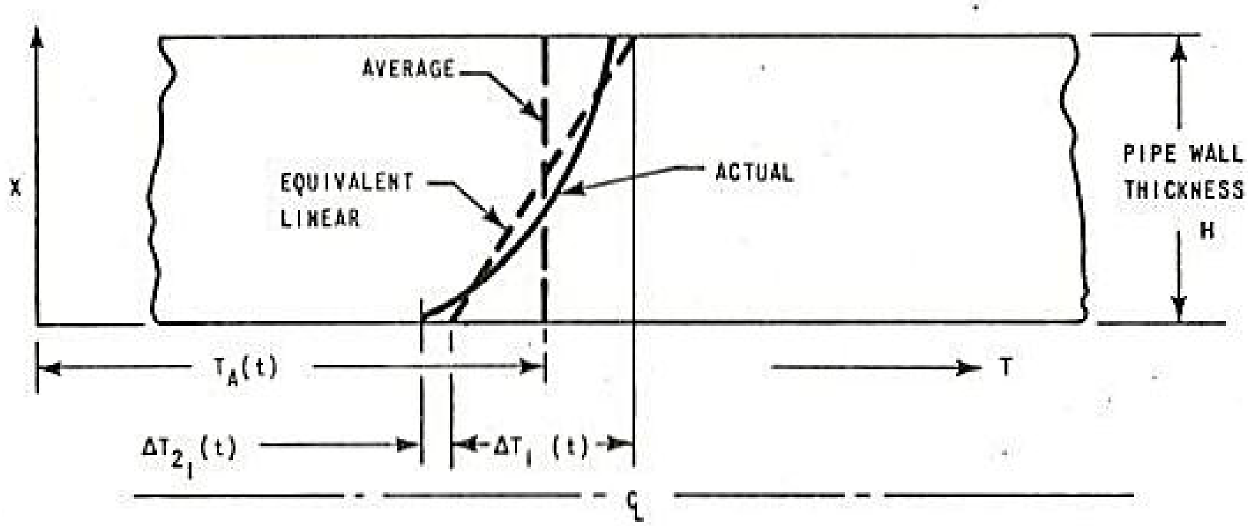
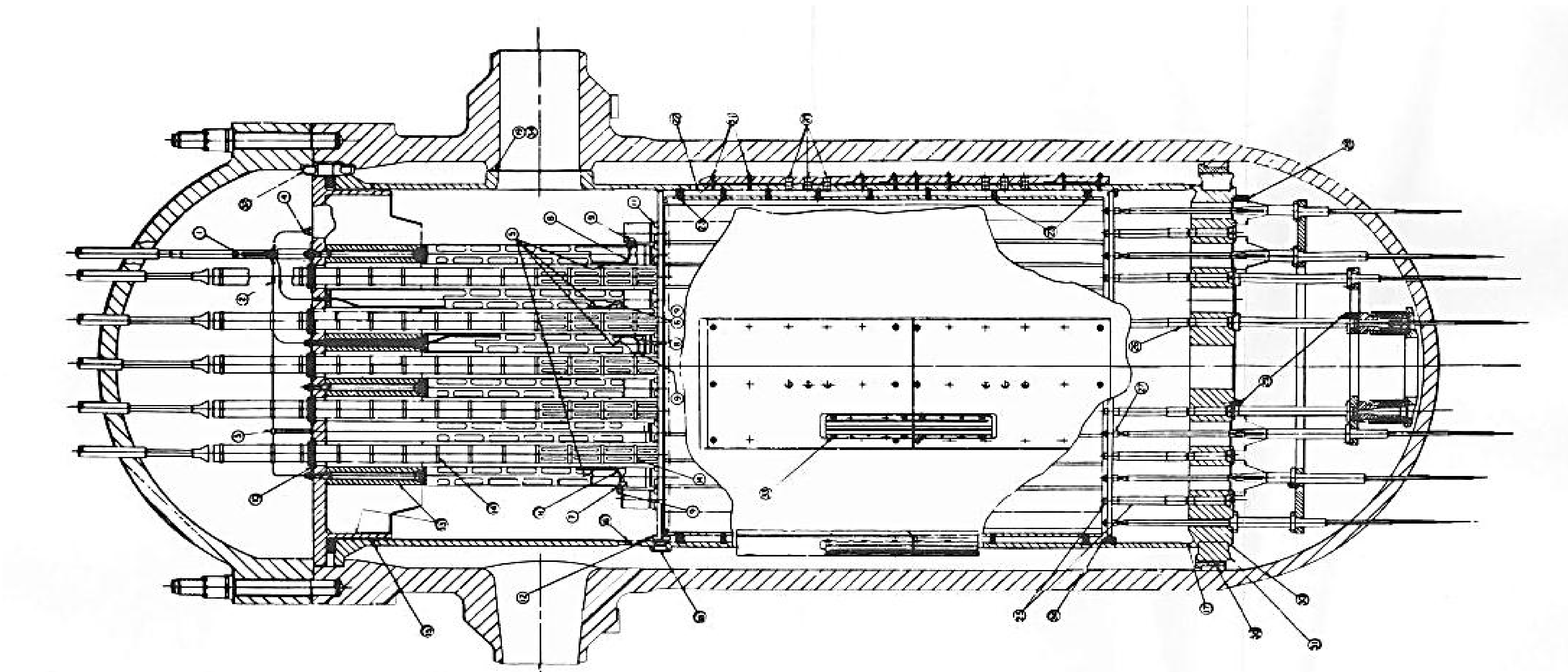


FIGURE 3.9.2-1

VIBRATION CHECKOUT – FUNCTIONAL TEST INSPECTION POINTS



notes		FEATURES TO BE EXAMINED
1	1	THERMOCOUPLE CONDUIT CLAMPS INSIDE THE THERMOCOUPLE COLUMN.
1,8,6	2	CONDUIT SWAGELOK FITTINGS, THEIR BANDINGS, AND THE TAB TYPE LOCKS.
1	3	CLAMP ARRANGEMENTS AT THE MOUNTING BRACKET LOCATIONS.
1	4	PLUG TO CONDUIT WELD AT THE FOUR SUPPORT COLUMNS ADJACENT TO THE THERMOCOUPLE COLUMNS.
1,6,2	5	ACCESSIBLE ANGLE CONDUIT CLAMPS INSIDE THE UPPER SUPPORT COLUMNS.
1	6	ACCESSIBLE WELD JOINTS AT THE THERMOCOUPLE STOP FOR THE SELF INSTRUMENTED COLUMN.
1	7	WELD JOINTS ON ACCESSIBLE SUPPORT COLUMN AND MIXING DEVICE GUSSETS. (THERMOCOUPLE SUPPORT HARDWARE.)
3	8	RIGIDITY OF EXPOSED PORTION OF THERMOCOUPLE CONDUIT RUNS, AT ACCESSIBLE LOCATIONS. (INSIDE SUPPORT COLUMNS LOWER END.)
4	9	RIGIDNESS OF THE ACCESSIBLE PROTRUDING THERMOCOUPLE TIPS.
1,2,5	10	THERMOCOUPLE COLUMN AND GUIDE TUBE SCREW LOCKING DEVICES.
1,2,5	11	ACCESSIBLE SUPPORT COLUMN, MIXING DEVICE, ORIFICE PLATE, AND CORE PLATE INSERT SCREW LOCKING DEVICES.
1,6,7	12	UPPER CORE PLATE INSERTS.
1	13	DEEP BEAM WELDS AT THE SKIRT AND AT THE OUTER HOLLOW ROUNDS.
1	14	ACCESSIBLE GUIDE TUBE WELDS.
1	15	UPPER BARREL TO FLANGE GIRTH WELD.
1	16	UPPER BARREL TO LOWER BARREL GIRTH WELD.
1	17	LOWER BARREL TO CORE SUPPORT GIRTH WELD.

notes		
1,6,7	18	UPPER CORE PLATE ALIGNING PIN WELDS AND BEARING SURFACES.
9	19	INLET NOZZLE INTERFACE SURFACE CONDITION.
1	20	NEUTRON SHIELD PANEL DOWEL PIN COVER PLATE WELDS.
1,2,5	21	NEUTRON SHIELD PANEL SCREW LOCKING DEVICES.
13	22	INTERFACE SURFACES AT THE SPACER PADS ALONG THE TOP AND BOTTOM ENDS OF THE NEUTRON PANELS
1	23	BAFFLE ASSEMBLY SCREW LOCKING ARRANGEMENTS AT THE TWO TOP AND THE TWO BOTTOM FORMER ELEVATIONS.
5	24	LOWER CORE PLATE TO CORE BARREL FLANGE SCREW LOCKING DEVICES ACCESSIBLE AT THE 0°, 90°, 180°, AND 270° AXES.
1,5,6,10	25	CORE SUPPORT COLUMNS AND THEIR SCREW LOCKING DEVICES.
11	26	CORE SUPPORT COLUMN ADJUSTING SLEEVES.
12	27	ACCESSIBLE (2) INSTRUMENTATION GUIDE COLUMN LOCKING COLLARS NEAREST THE MANWAY.
1,6,8,13	28	LOCKING DEVICES AND CONTACT OF THE CRUCIFORM SHAPED BOTTOM INSTRUMENTATION GUIDE COLUMNS WHERE ATTACHED TO THE CORE SUPPORT AND TIE PLATES.
1,2,5	29	LOCKING DEVICES OF THE SECONDARY CORE SUPPORT BUTT COLUMNS AT THE CORE SUPPORT AND AT THE TIE PLATE.
1	30	RADIAL SUPPORT KEY WELDS.
1,6,7	31	RADIAL SUPPORT KEY LOCKING ARRANGEMENTS AND BEARING SURFACES.
1,6,7	32	HEAD AND VESSEL ALIGNING PIN SCREW LOCKING DEVICES AND BEARING SURFACES.
1,2,5	33	IRRADIATION SPECIMEN GUIDE SCREW LOCKING DEVICES AND DOWEL PINS.
9	34	VESSEL NOZZLE INTERFACE SURFACE CONDITION.
1,6,14	35	VESSEL CLEVIS LOCKING ARRANGEMENTS AND BEARING SURFACES.

NOTES:

1. Visually examine welds using 5-10X magnification. No cracks allowed.
2. Verify that clamps are not damaged.
3. Grasp exposed conduit and verify that it is rigid and tightly fastened.
4. Grasp exposed portion of thermocouple tip. Verify that tip does not move freely.
5. Verify that locking devices are crimped and undamaged.
6. Verify that inserts are seated. (.0015 feeler must not pass through interface.) 50% contact required.
7. Visually examine faces for damage using 5-10X magnification.
8. Verify that fittings are tight.
9. Visually examine nozzle surfaces for any evidence of damage.
10. Verify that accessible columns are seated on lower core plate. A .0015 feeler gage must not pass through 90°.
11. Verify that accessible sleeves are seated on columns. A .0015 feeler must not pass through 90°.
12. Verify that locking collars are tight. No movement allowed.
13. Verify seating using a .0015 feeler gage. Feeler must not pass through 90°.
14. Verify seating using a .0015 feeler gage. No gaps allowed.

FIGURE 3.9.4-1

FULL LENGTH CONTROL ROD DRIVE MECHANISM

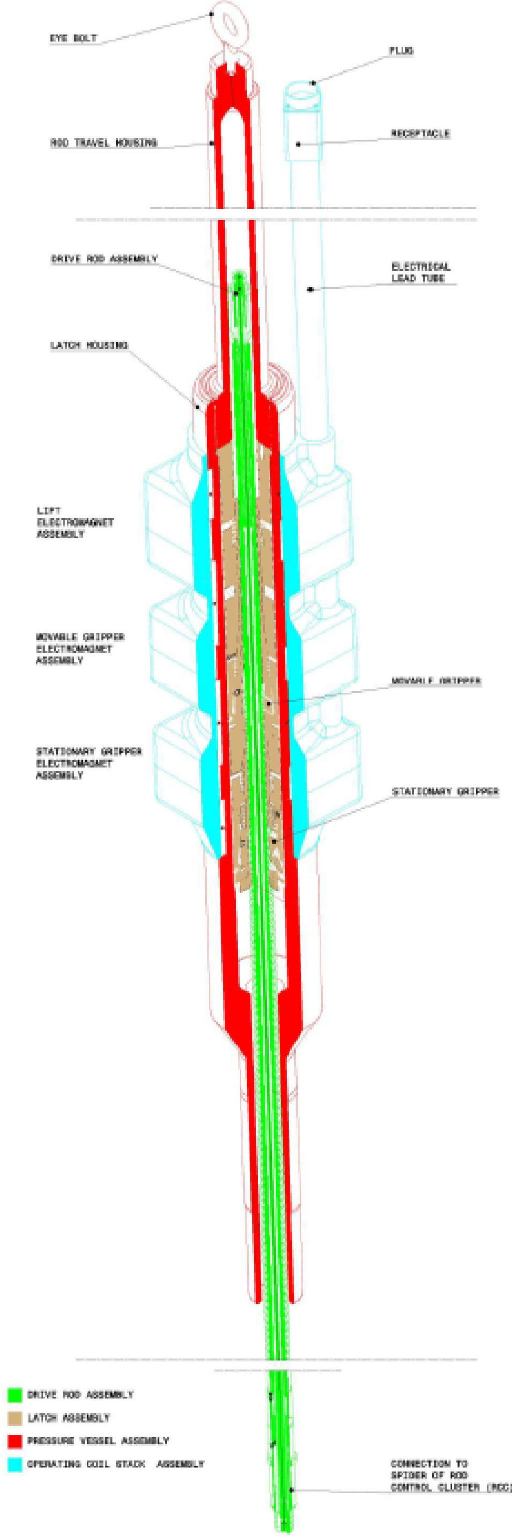


FIGURE 3.9.4-2

CONTROL ROD DRIVE MECHANISM SCHEMATIC

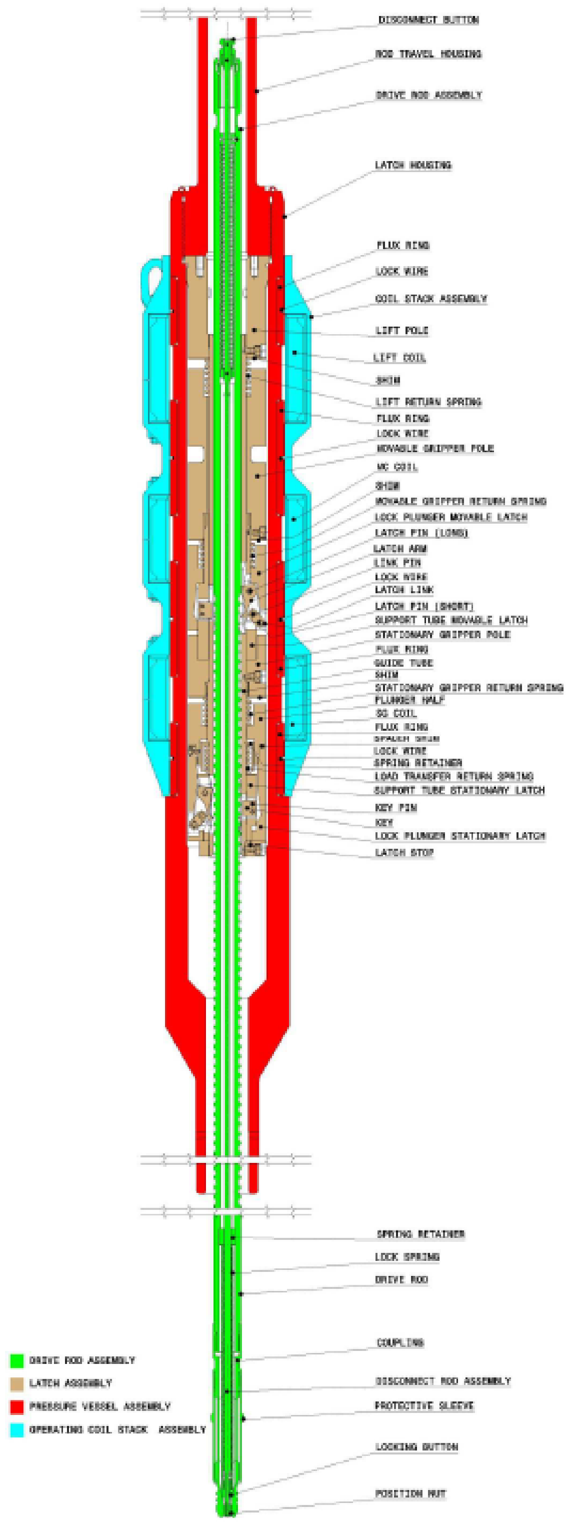
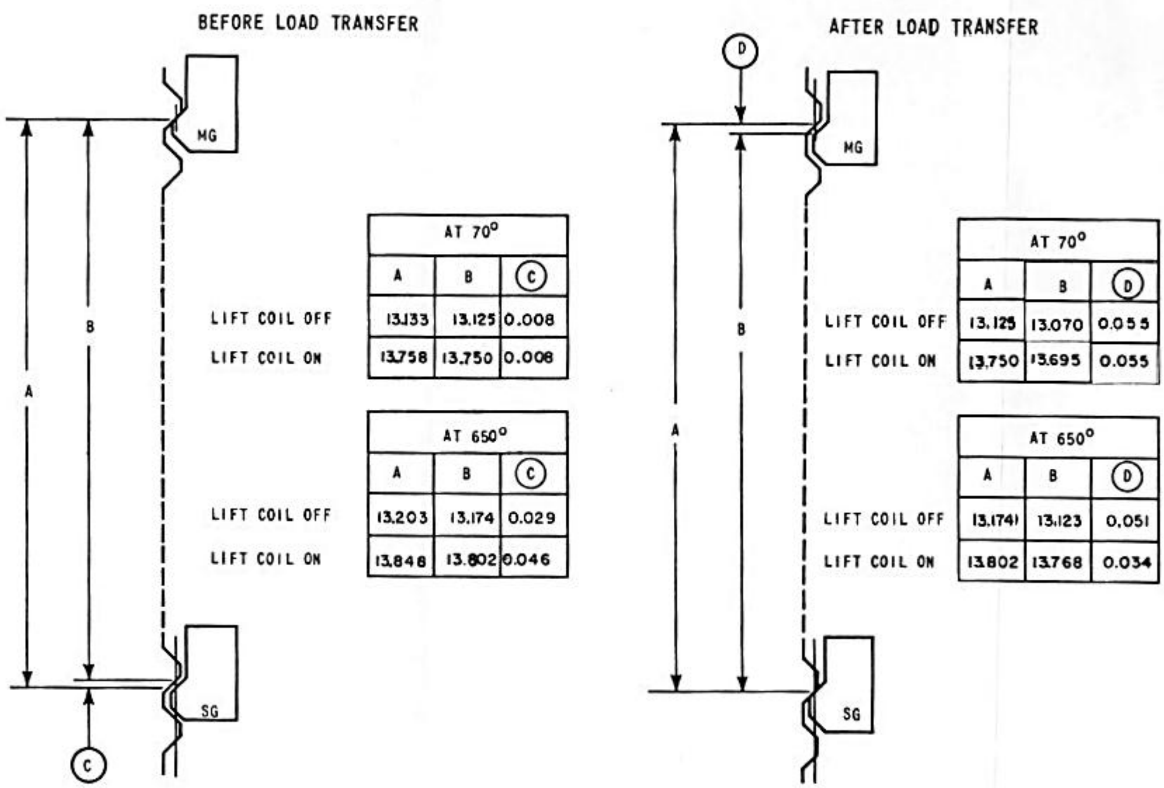


FIGURE 3.9.4-3

NOMINAL LATCH CLEARANCE AT MINIMUM & MAXIMUM TEMPERATURE



BEFORE LOAD TRANSFER

AT 70°	A	B	C
LIFT COIL OFF	13.133	13.125	0.008
LIFT COIL ON	13.758	13.750	0.008

AT 650°	A	B	C
LIFT COIL OFF	13.203	13.174	0.029
LIFT COIL ON	13.848	13.802	0.046

AFTER LOAD TRANSFER

AT 70°	A	B	D
LIFT COIL OFF	13.125	13.070	0.055
LIFT COIL ON	13.750	13.695	0.055

AT 650°	A	B	D
LIFT COIL OFF	13.174	13.123	0.051
LIFT COIL ON	13.802	13.768	0.034

FIGURE 3.9.4-4

CONTROL ROD DRIVE MECHANISM LATCH CLEARANCE THERMAL EFFECT

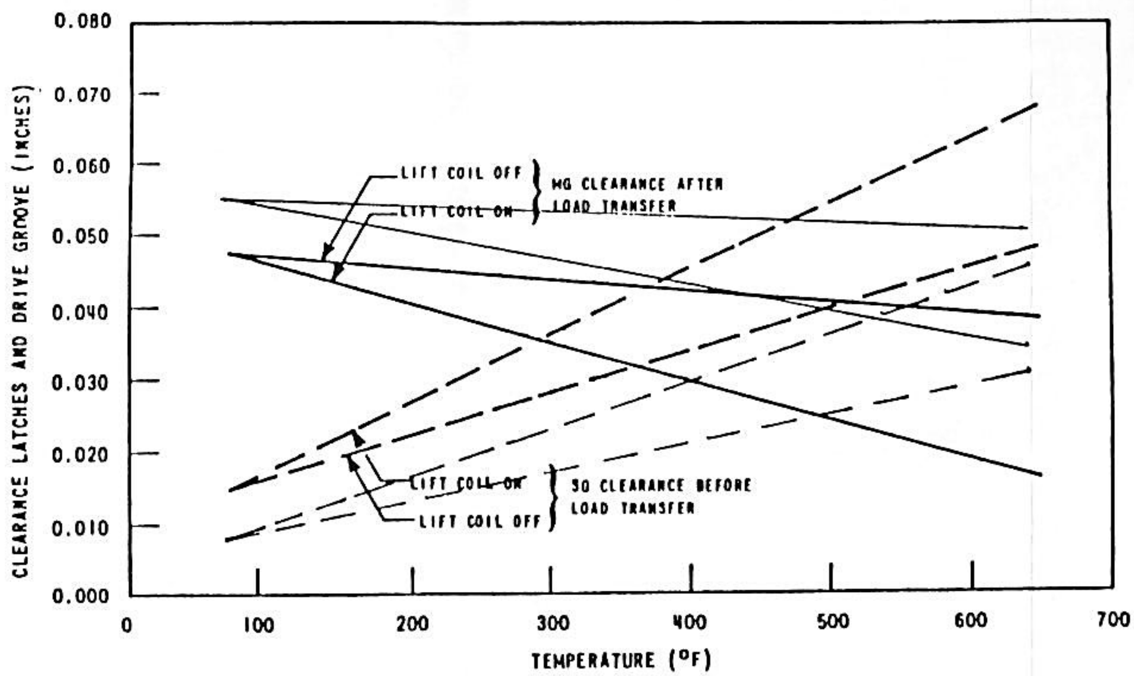


FIGURE 3.9.5-1

LOWER CORE SUPPORT ASSEMBLY (CORE BARREL ASSEMBLY)

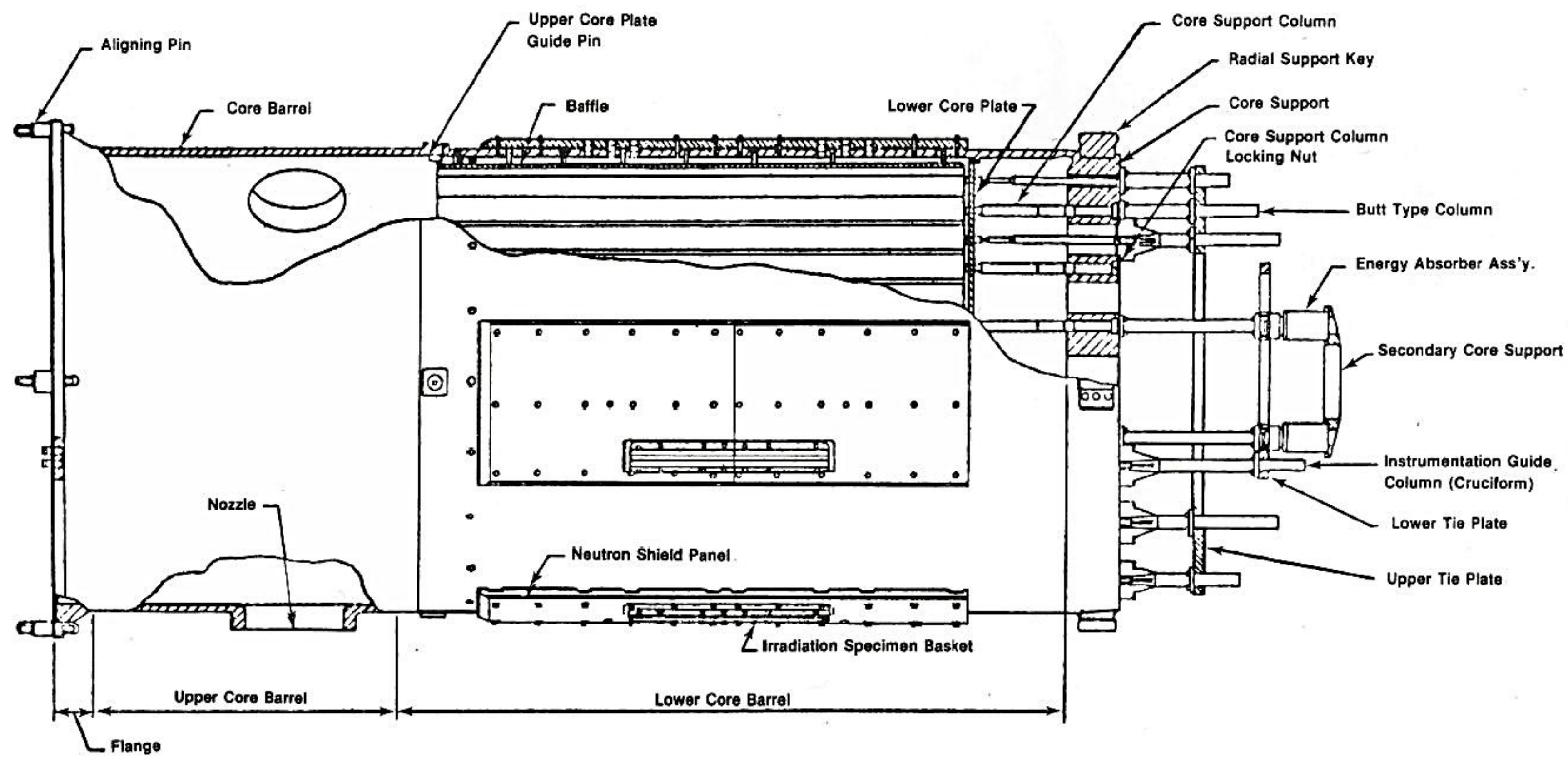


FIGURE 3.9.5-2

UPPER CORE SUPPORT ASSEMBLY

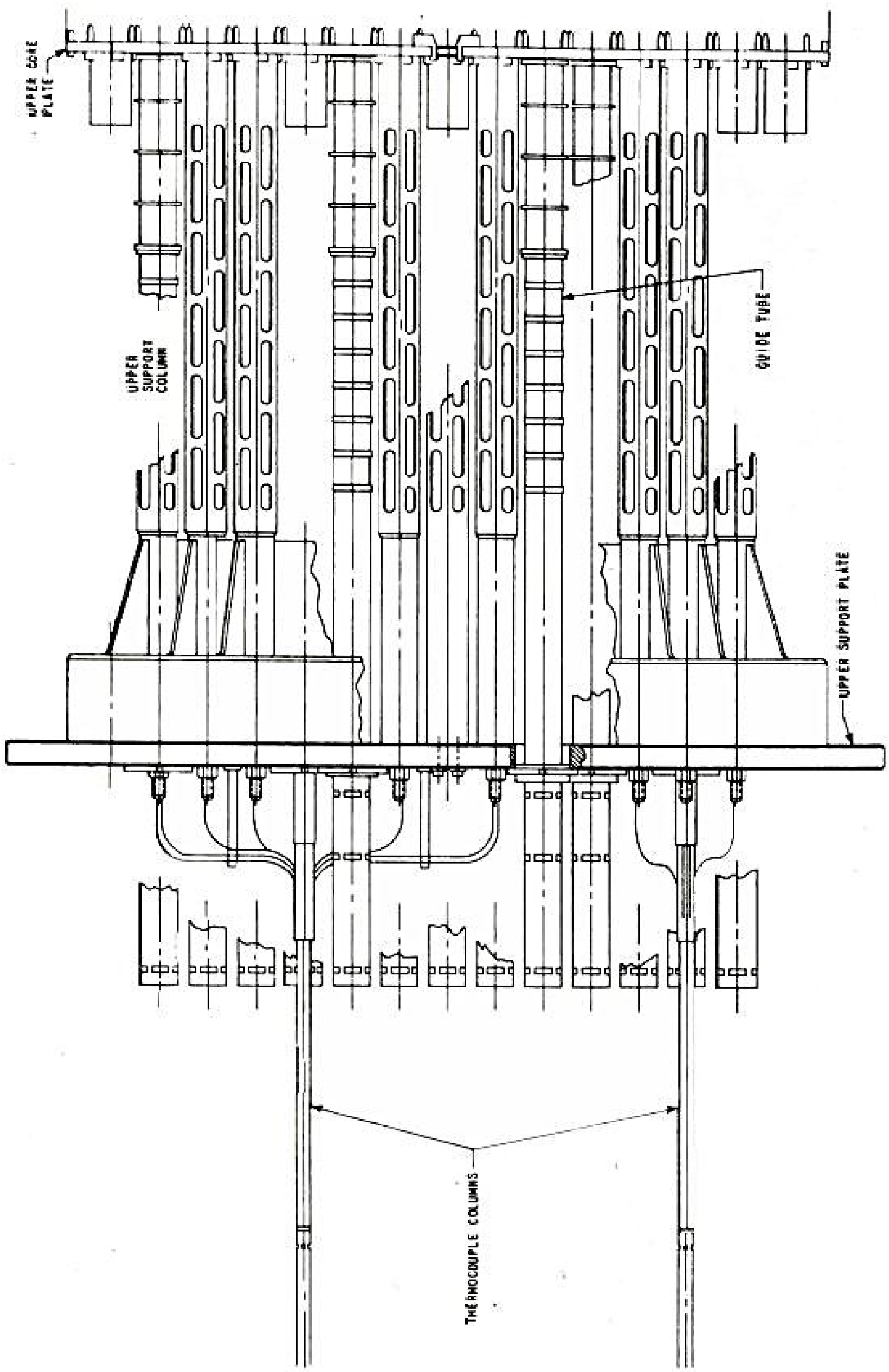


FIGURE 3.9.5-3

PLAN VIEW OF UPPER CORE SUPPORT STRUCTURE

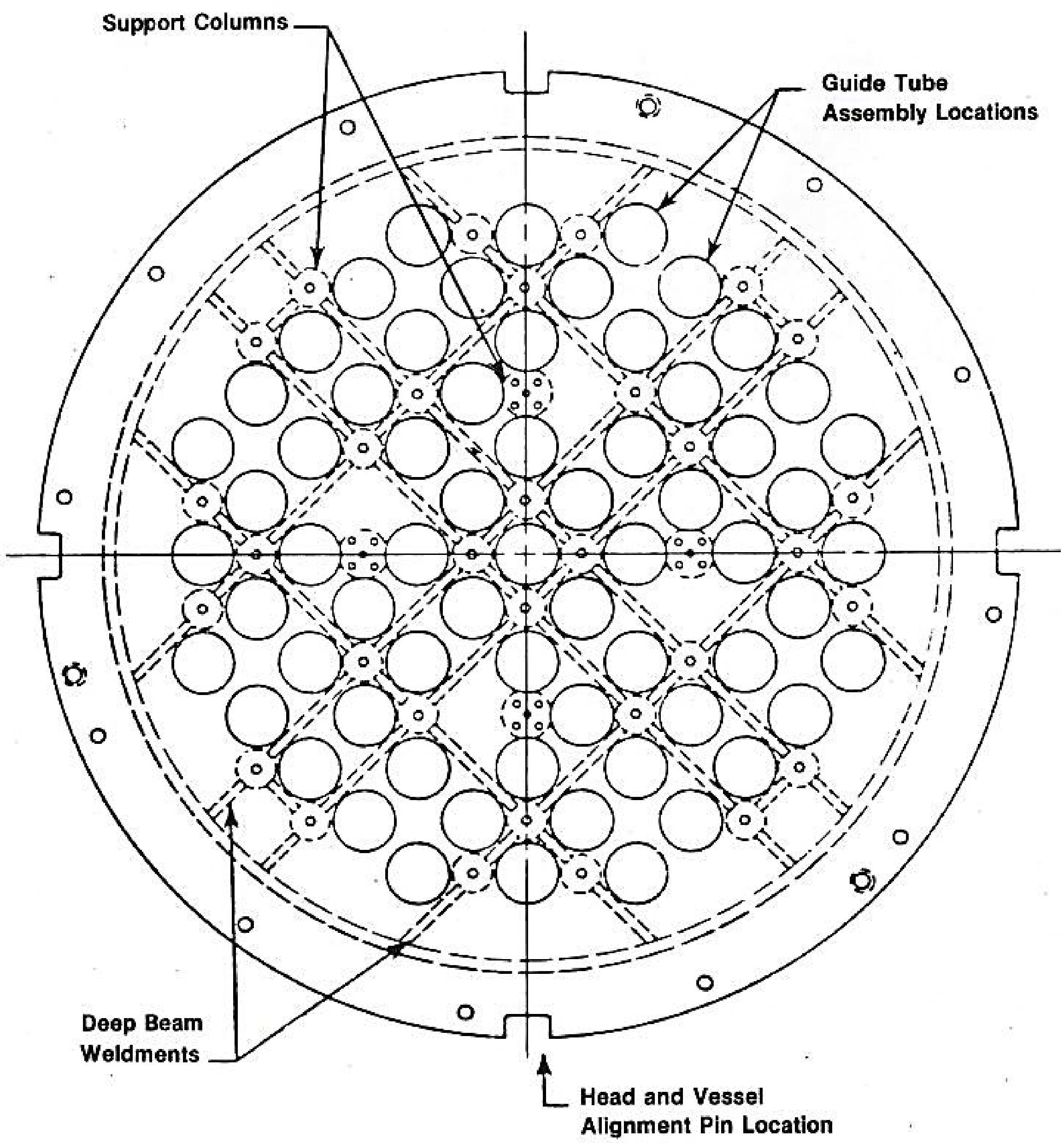


FIGURE 3.10.1-1

SQRT MASTER LIST FORMAT

AE: EBASCO SERVICES INC.
UTILITY: CAROLINA POWER & LIGHT

FACILITY: SHEARON HARRIS NO.1
SEISMIC QUALIFICATION REVIEW TEAM (SQRT)
MASTER LIST

NSSS: WESTINGHOUSE PWR
CAPACITY 900 MW(E)

PAGE NO.

SYSTEM/CODE — SERVICE & FUNCTION	EQUIPMENT: TYPE — COMPON'T	EQUIPMENT: TAG#/REC# — TEST REP#	EQUIPMENT: MAJ.SUPPLR — MANUFACT. — MODEL NO	EQUIPMENT: BLDG — MOUNT — ELEV	EQUIPMENT: PO/ITM — SPEC — SQRT #	QUAL CODES STDS	STATUS Q	STATUS I	LOAD INPUT: ZPA OBE	LOAD INPUT: ZPA SSE	LOAD INPUT: SEISMIC	LOAD INPUT: OTHER DYN	QUALIFICATION METHOD: ANALYSIS S/D	QUALIFICATION METHOD: TEST FREQ	QUALIFICATION METHOD: TEST DIR	LOWEST NATURAL FREQUENCY: F/B	LOWEST NATURAL FREQUENCY: S/S	LOWEST NATURAL FREQUENCY: V	FLOOR RESP. SPECTR	REMARKS

SAMPLE

TAGS INCLUDED:
SORT TYPE:

GLOBAL REV#- DATE

LEGEND: SEE REFERENCE 3.10.1-1

FIGURE 3.11C-1

90, MIL, OKONITE SINGLE CONDUCTOR EQUIPMENT TEMPERATURE VS TIME DURING 102 PERCENT POWER, 1.4 FT^2 MSLB (ASSUMING SINGLE FAILURE OF MFIV)

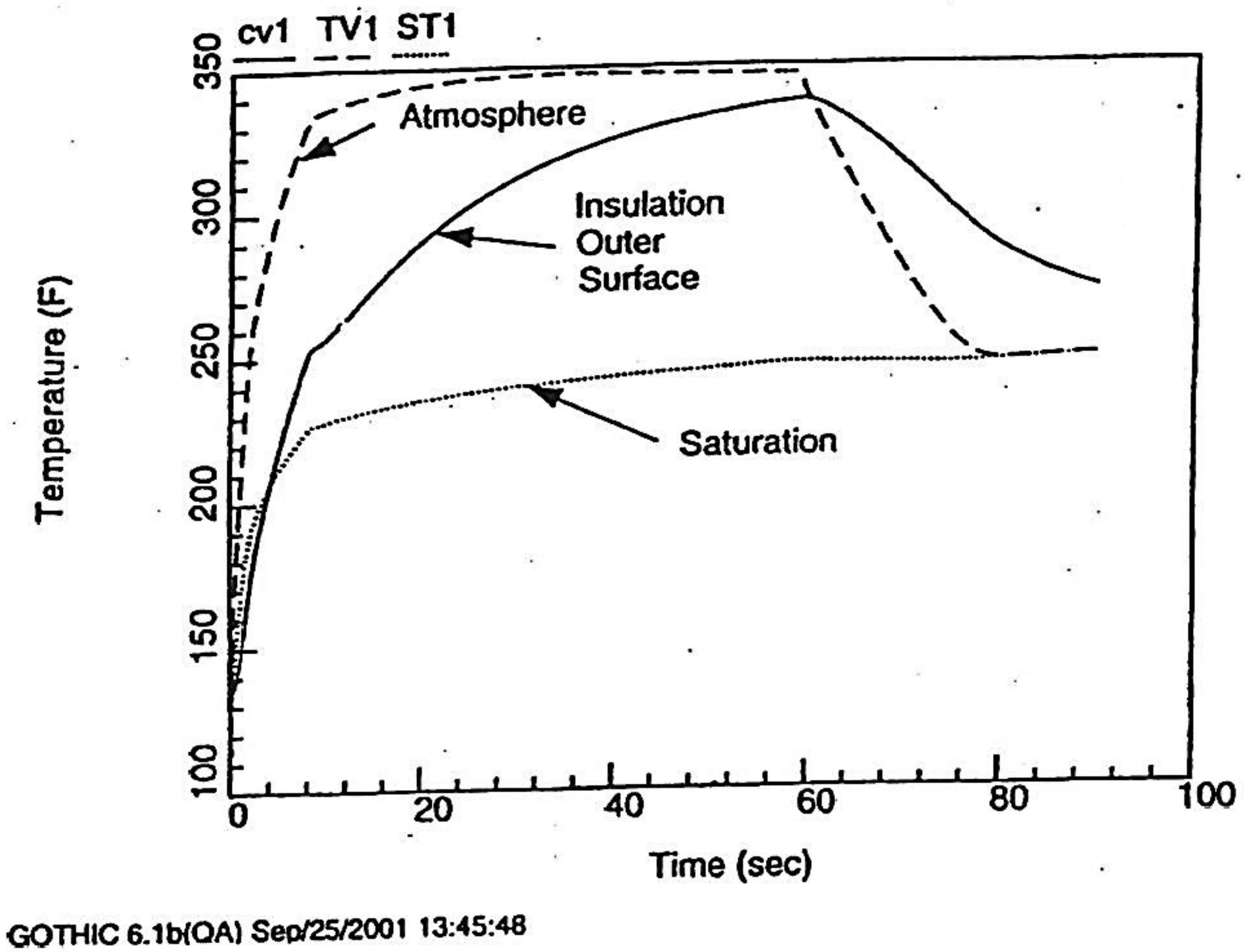


FIGURE 3.11C-2

PA/PG 3200 PRESSURE TRANSMITTER EQUIPMENT TEMPERATURE VS TIME DURING 102 PERCENT POWER, 1.4 FT^2 MSLB (ASSUMING SINGLE FAILURE OF MFIV)

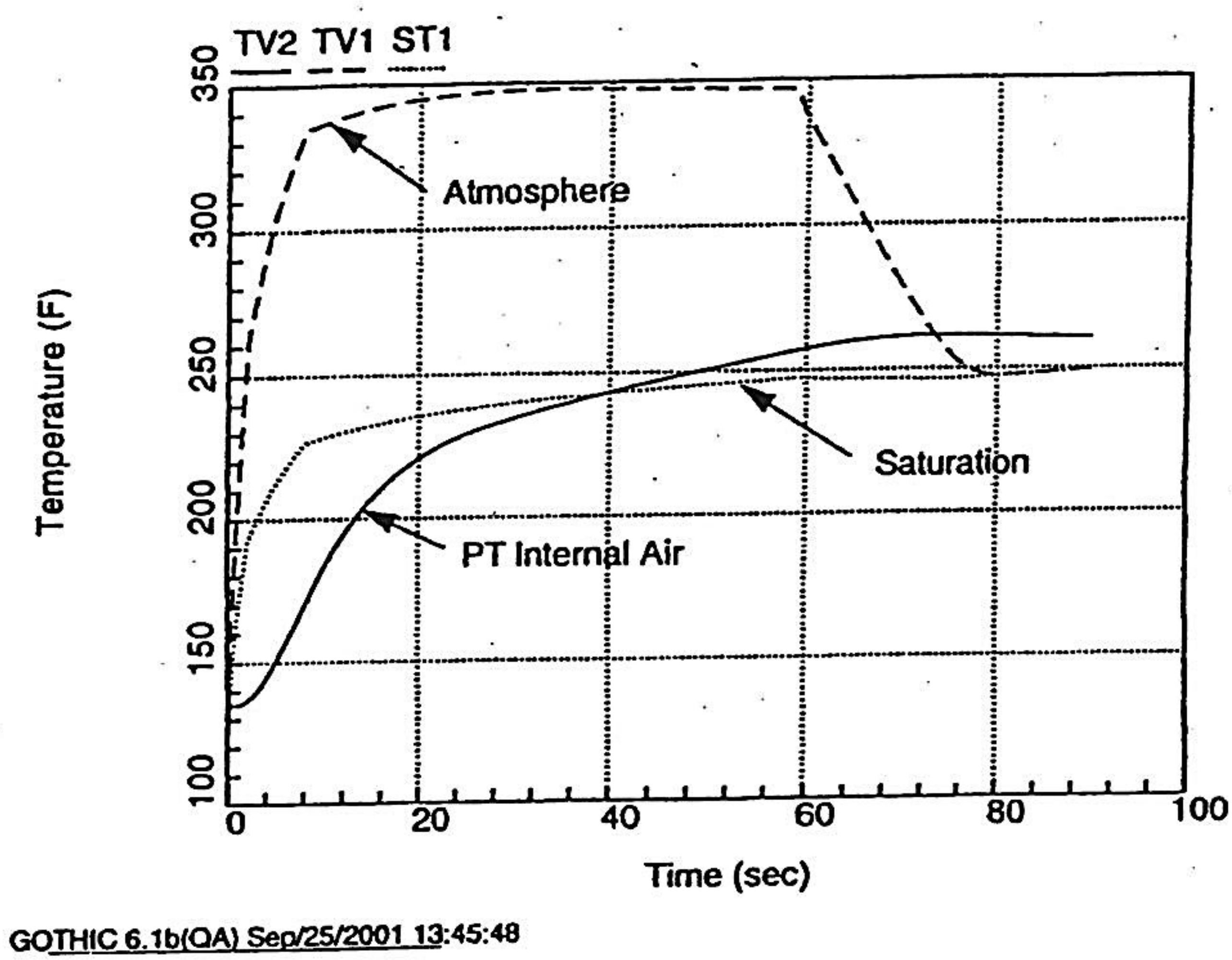


FIGURE 3.11C-3

ROCKBESTOS THERMOCOUPLE EQUIPMENT TEMPERATURE
VS TIME DURING 102 PERCENT POWER, 1.4 FT^2 MSLB
(ASSUMING SINGLE FAILURE OF MFIV)

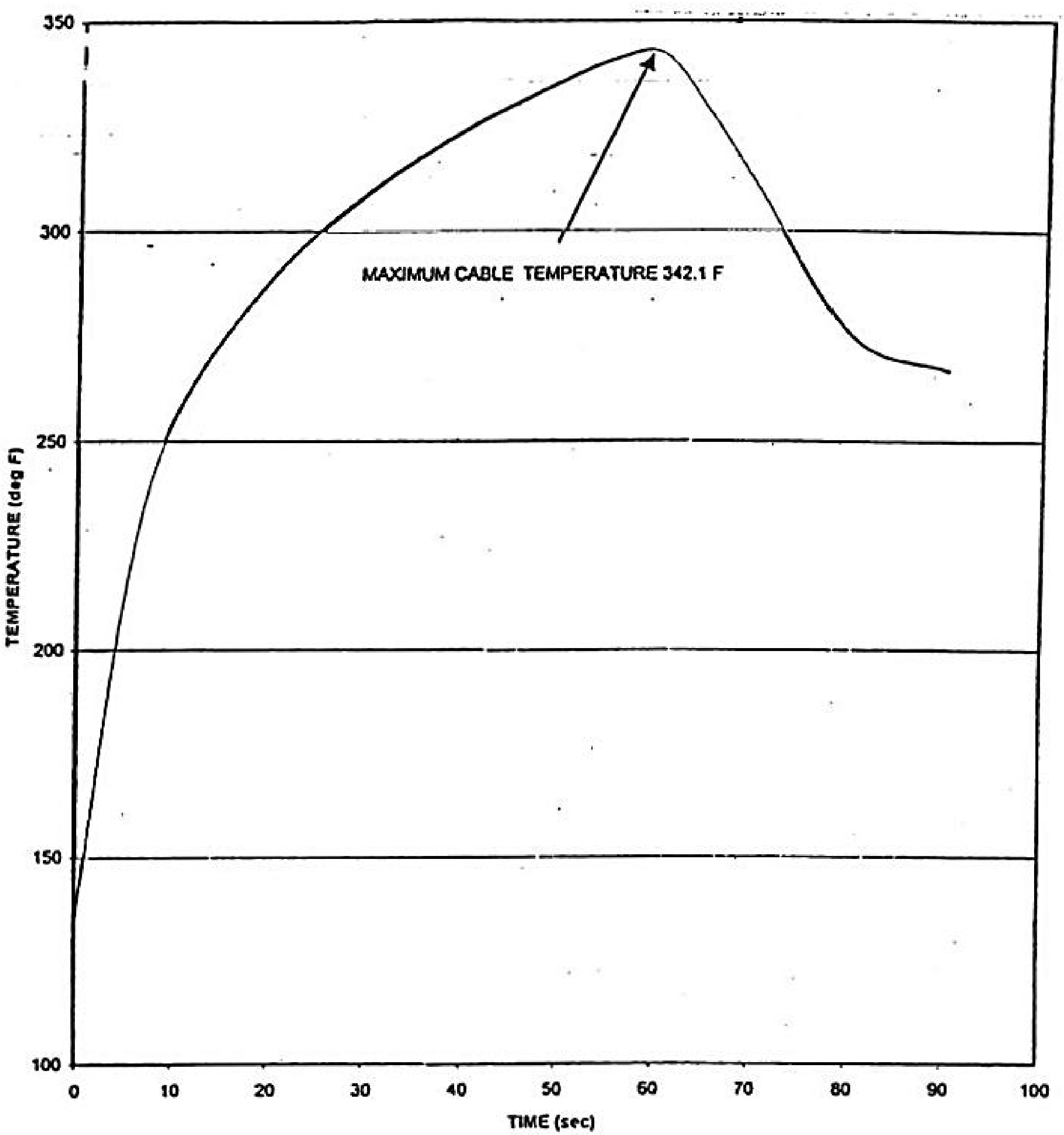


FIGURE 3.11E-1

NAMCO LIMIT SWITCH HEAT TRANSFER COEFFICIENT

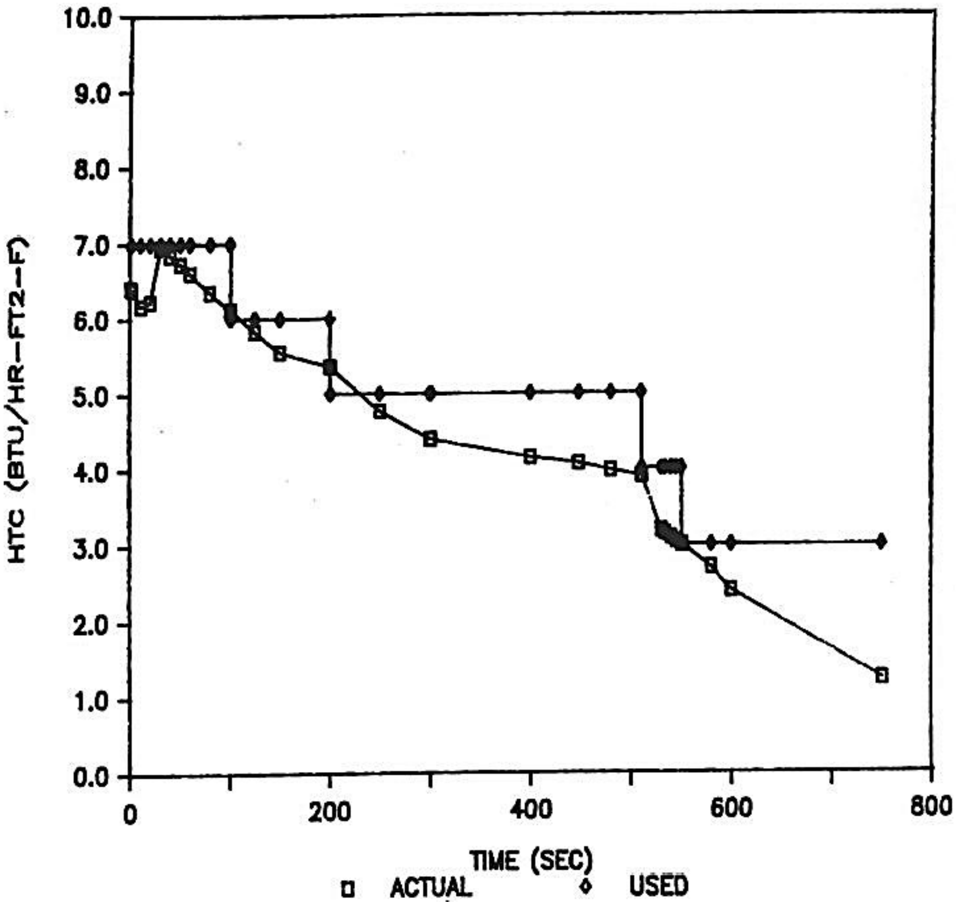


FIGURE 3.11E-2

ASCO SOLENOID VALVE HEAT TRANSFER COEFFICIENT

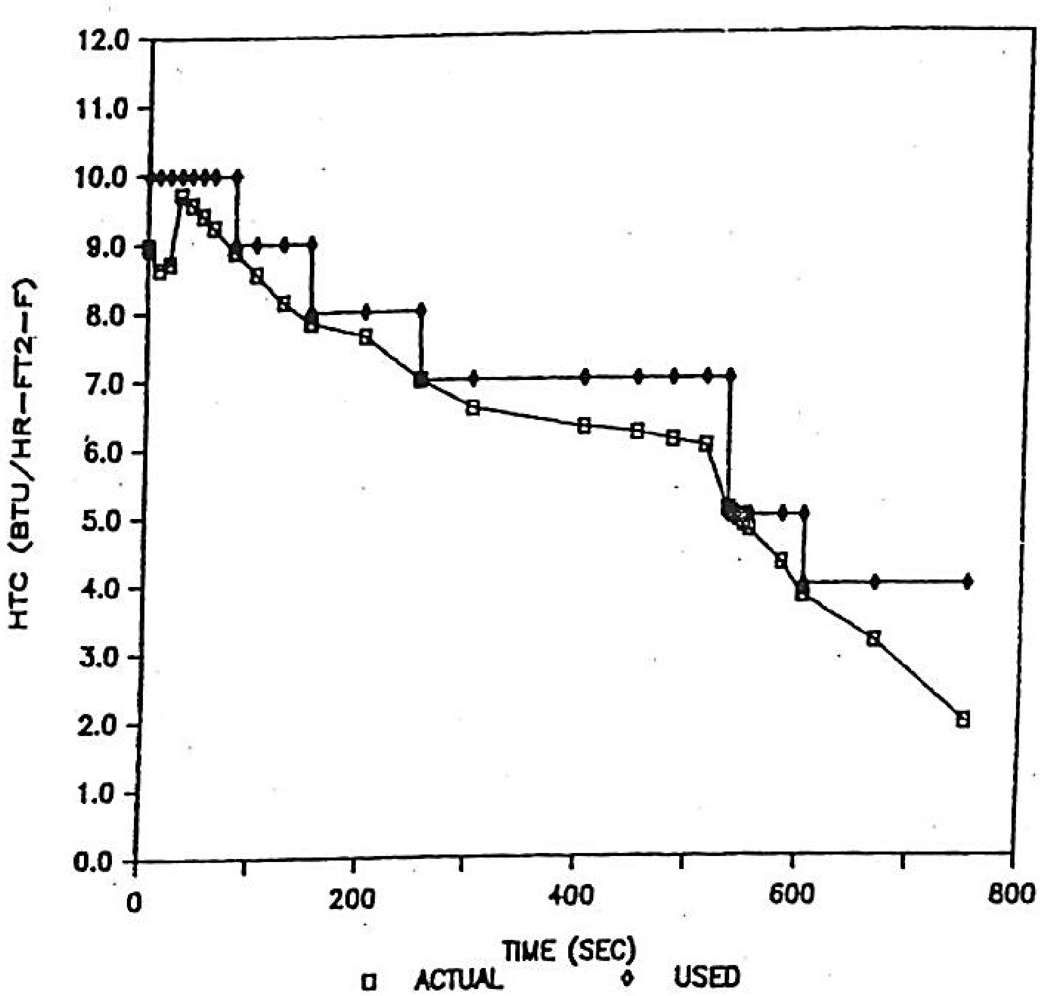


FIGURE 3.11E-3

NAMCO LIMIT SWITCH .5 FT^2 MSLN IN MSLT, 102 % POWER

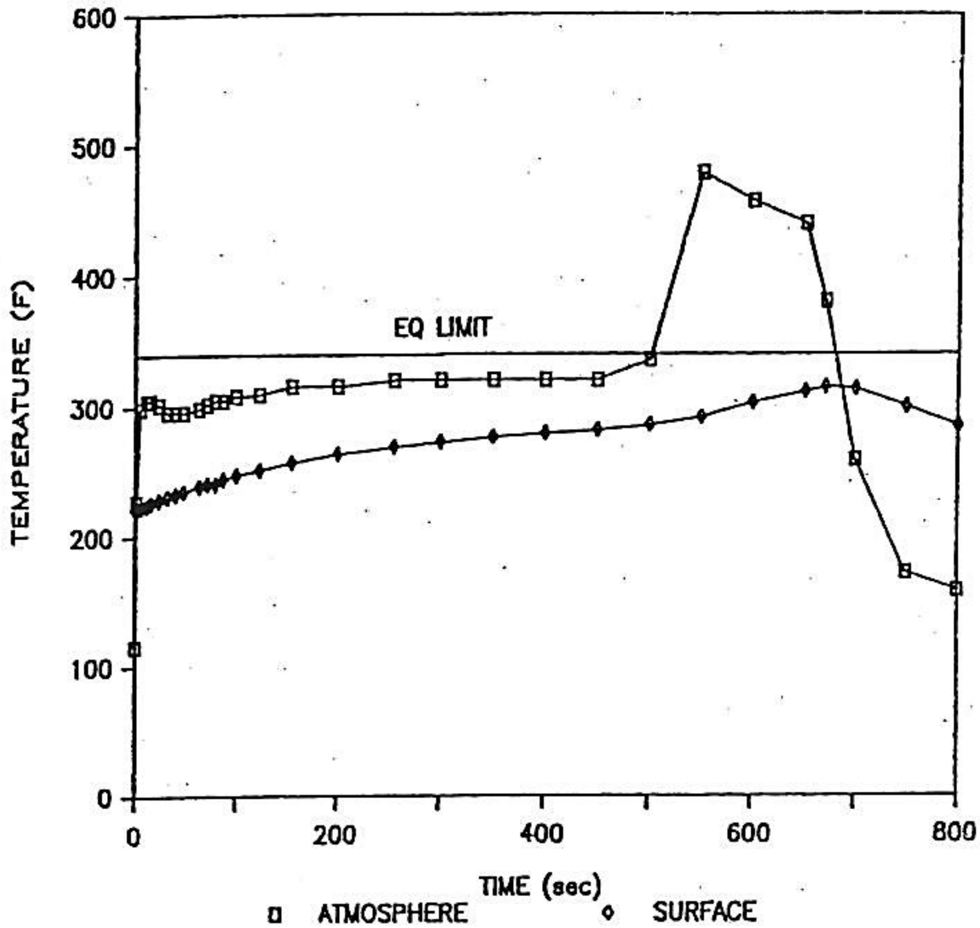


FIGURE 3.11E-4

ASCO SOLENOID VALVE 0.5 FT^2 MSLB IN MSLT, 102% POWER

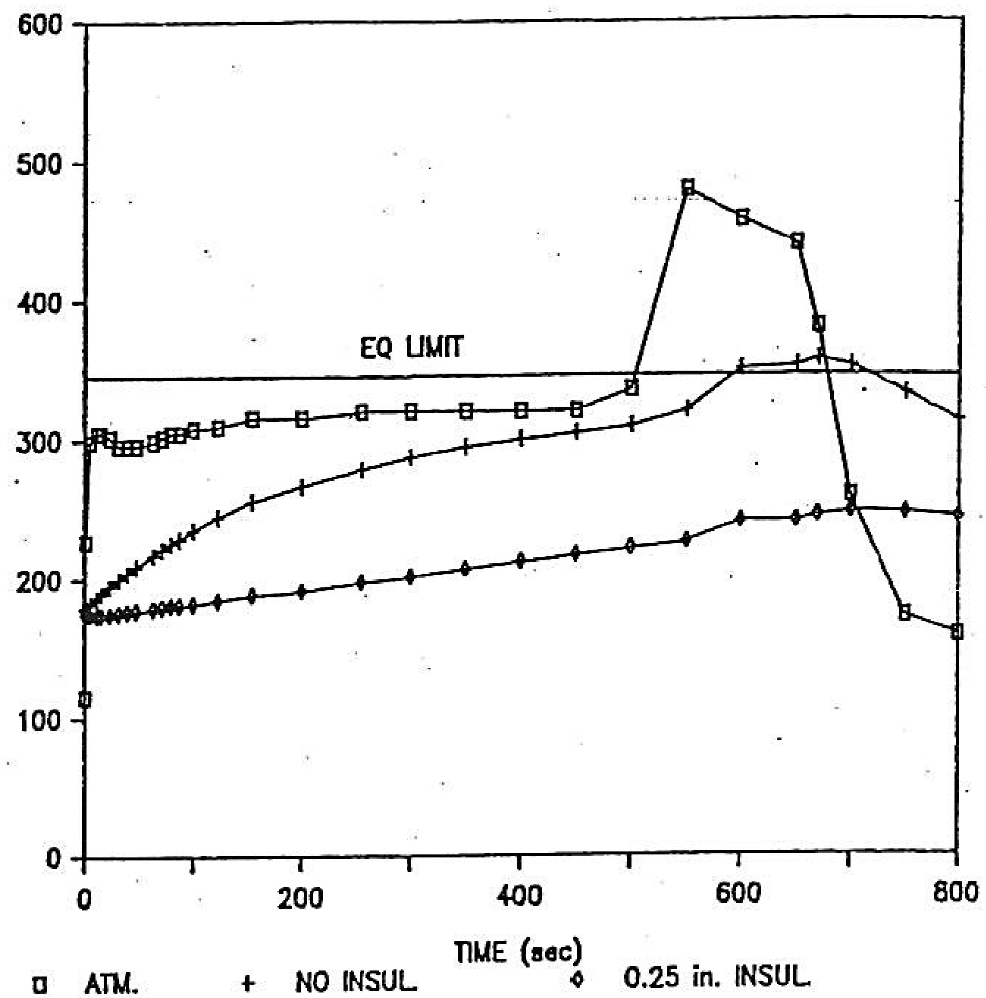


FIGURE 3.11E-5

COMPARISON OF STEAM TUNNEL TEMPERATURE FOR 0.5 FT^2 MSLB WITH AFW ISOLATION