TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Introduction

The Watts Bar Nuclear Power plant was designed to meet the intent of the "Proposed General Design Criteria for Nuclear Power Plant Construction Permits" published in July, 1967. The Watts Bar construction permit was issued in January, 1973. This UFSAR, however, addresses the NRC General Design Criteria (GDC) published as Appendix A to 10 CFR 50 in July, 1971, including Criterion 4 as amended October 27, 1987.

Each criterion is followed by a discussion of the design features and procedures which meet the intent of the criteria. Any exception to the 1971 GDC resulting from the earlier commitments is identified in the discussion of the corresponding criterion. References to other sections of the UFSAR are given for system design details.

3.1.2 WBNP Conformance with GDCs

3.1.2.1 Overall Requirements

Criterion 1 - Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety 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 function. 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.

Compliance

Discussions related to the applicable codes, design criteria and standards used in the design of particular systems are contained in the appropriate SAR sections and in Tables 3.2-1, 3.2-2, 3.2-3, 3.2-4 and 3.2-5.

The Quality Assurance Program conforms to the requirements of 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants." Details of the program are given in Chapter 17.

Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:

1. Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,

2. Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and

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

Compliance

The structures, systems, and components important to safety are designed to either withstand the effects of natural phenomena without loss of capability to perform their safety functions, or to fail in the safest condition. Those structures, systems, and components vital to the shutdown capability of the reactor are designed to withstand the maximum probable natural phenomenon expected at the site, determined from recorded data for the site vicinity, with appropriate margin to account for uncertainties in historical data. Appropriate combinations of normal, accident, and natural phenomena structural loadings are considered in the plant design.

The nature and magnitudes of the natural phenomena considered in the design of the plant are discussed in Sections 2.3, 2.4, and 2.5. Sections 3.2 through 3.10 discuss the design of the plant in relationship to natural events. Seismic and safety classifications, as well as other pertinent standards and information, are given in the sections discussing individual structures and components.

Criterion 3 - Fire Protection

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

Compliance

The plant is designed to minimize the probability of fires and explosions, and in the event of such occurrences to minimize the potential effects of such events to plant safety related equipment and personnel. Prime consideration was given to these requirements throughout the design process by providing for the duplication and physical separation of components in plant design and the use of materials classified as noncombustible and/or fire resistant wherever practical in safety-related areas of the plant. Equipment and facilities for fire protection, including detection, alarm, and extinguishment, are provided to protect both plant-equipment and personnel from fire, explosion, and the resultant release of toxic vapors. Fire-fighting systems are designed to assure that their rupture or inadvertent operation will not significantly impair systems important to safety. Portions of the fire-protection systems necessary to protect safety-related equipment in Class I structures are designed to seismic requirements.

The Fire Protection Systems provided are:
1. High pressure water,
2. Carbon dioxide, and
3. Portable extinguishers.

The Fire Protection System is designed such that a failure of any component of the system or inadvertent operation:
1. Does not cause a nuclear accident or significant release of radioactivity to the environment.
2. Does not impair the ability of equipment to safely shutdown and isolate the reactor or limit the releases of radioactivity to the environment in the event of a postulated accident.

The Fire Protection Systems for the Watts Bar Nuclear Plant are discussed in Section 9.5.1. Protection from fire in the control room is discussed in Section 6.4.

Criterion 4 - Environmental and Missile Design Bases

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

Compliance

This criterion has been implemented as amended and published in the Federal Register, Volume 52, Number 207, October 27, 1987, 41288, which added the following:

> "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."

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The associated environmental parameters are identified and incorporated in the design requirements and specifications. Particular care was given to the extreme environmental conditions associated with major incidents such as LOCAs. Required equipment and instrumentation are identified, environmental conditions such as temperature, pressure, humidity, and irradiation, are calculated, and the effects of the latter on the former were evaluated either analytically or experimentally. The dynamic effects associated with an accident were carefully identified and assurance given that the structures and systems (including engineered safeguards) assumed undamaged in the total assessment of the accident consequences are suitably protected.

Emergency core cooling components are austenitic stainless steel or equivalent corrosion resistant material and hence are compatible with the containment atmosphere over the full range of exposure during the post accident conditions.

Where vital components cannot be located away from potential missiles, protective walls and slabs, local missile shielding, and restraining devices are provided to protect the containment and engineered safety feature components within the containment against damage from missiles generated by the equipment failures associated with the design basis accident (DBA).

The environmental design of safety-related items is discussed in Section 3.8 on the design of structures; Sections 6.2.2 and 6.2.3 on containment heat removal and air purification; and Section 9.4 on ventilation systems. Safety- related systems and components used the input from these sections for design as discussed in Section 3.11. Missile and environmental protection is discussed in Sections 3.5 and 3.11, respectively.

Criterion 5 - Sharing of Structures, Systems, and Components

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

Compliance

The structures important to safety that are shared are the Auxiliary Building (Section 3.8), Control Building (Section 3.8), Diesel Generator Building (Section 3.8), and the intake pumping station (Section 3.8). Shared safety- related systems include the essential raw cooling water (Section 9.2), component cooling water (Section 9.2), fire protection (Section 9.5), spent fuel cooling (Section 9.1), fuel oil storage tanks (Section 9.5), preferred and emergency electric power (Section 8.2 and 8.3, respectively), chemical and volume control (Section 9.3), radioactive waste (Chapter 11), emergency gas treatment system (Sections 6.2 and 6.5), auxiliary control air system (Section 9.3), and control and Auxiliary Building ventilation systems (Section 6.4). The vital dc power system is shared to the extent that a few loads (e.g., the vital inverters) in one nuclear unit are energized by the dc power channels assigned primarily to power loads of the other unit. In no case does the sharing inhibit the safe shutdown of one unit while the other unit is experiencing an accident. All shared systems are sized for all credible initial combinations of normal and accident states for the two units, with appropriate isolation to prevent an accident condition in one unit from carrying into the other.

If the designated equipment configuration is revised to allow system testing or modification, appropriate action will be taken to ensure that the required system availability for accident mitigation is maintained.

3.1.2.2 Protection By Multiple Fission Product Barriers

Criterion 10 - Reactor Design

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

Compliance

The reactor core with its related coolant, control, and protection systems is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The reactor trip system is designed to actuate a reactor trip for any anticipated combination of plant conditions when necessary to ensure that fuel design limits are not exceeded. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of loss of reactor coolant flow, trip of the turbine-generator, loss of normal feedwater and loss of both normal and preferred power sources.

Chapter 4 discusses the design bases and design evaluation of reactor components. Chapter 5 discusses the reactor coolant system. The details of the reactor trip and engineered safety features actuation system design and logic are discussed in Chapter 7. This information supports the accident analyses presented in Chapter 15.

Criterion 11 - Reactor Inherent Protection

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

Compliance

A negative reactivity coefficient is a basic feature of core nuclear design as discussed in Chapter 4.

Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Compliance

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

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

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

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

The stability of the core against xenon-induced power oscillations and the functional requirements of instrumentation for monitoring and measuring core power distribution are discussed in Section 4.3. Details of the instrumentation design and logic are discussed in Chapter 7.

Criterion 13 - Instrumentation and Control

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

Compliance

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, steam and power conversion system, the containment, engineered safety features systems, radiological waste systems and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the control room in proximity with the controls for maintaining the indicated parameter in the proper range.

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

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, or rapidly propagating failure, and of gross rupture.

Compliance

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. In addition to the loads imposed on the piping under operating conditions, consideration is also given to abnormal loadings, such as pipe rupture and seismic loadings as discussed in Sections 3.6 and 3.7, respectively. The piping is protected from overpressure by means of pressure relieving devices as required by applicable codes.

Reactor coolant pressure boundary materials selection and fabrication techniques ensure a low probability of gross rupture or significant leakage.

The materials of construction of the reactor coolant pressure boundary are protected by control of coolant chemistry from corrosion which might otherwise reduce the structural integrity of the boundary during its service lifetime.

The reactor coolant pressure boundary has provisions for inspections, testing and surveillance of critical areas to assess the structural and leak tight integrity. The details are given in Section 5.2. For the reactor vessel, a material surveillance program conforming to applicable codes is provided.

Means are provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary with indication in the control room as discussed in Section 5.2.

Criterion 15 - Reactor Coolant System Design

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

Compliance

Transient analyses presented in Section 5.2 lead to the conclusion that design conditions are not exceeded during normal operation. Protection and control set points are based on these transient analyses.

Additionally, reactor coolant pressure boundary components achieve a large margin of safety by the use of proven ASME materials and design codes, use of proven fabrication techniques, nondestructive shop testing and integrated hydrostatic testing of assembled components.

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

Multiple safety and relief valves are provided for the reactor coolant system. These valves and their set points meet ASME criteria for over-pressure protection. The ASME criteria are satisfactory based on a long history of industry use. Chapter 5 discusses reactor coolant system design.

Criterion 16 - Containment Design

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

Compliance

The reactor containment (Section 6.2) is a freestanding, continuous steel membrane structure housing the reactor and various auxiliary components including the ice condenser. The ice condenser (Section 6.7) limits the initial containment pressure to a value less than design during a large LOCA. A concrete Shield Building surrounding the steel vessel allows for collection of any containment leakage, which is subsequently processed by the emergency gas treatment system (Section 6.5) before release to the environment. The containment also contains a spray system (Section 6.2) which supplements the ice condenser in limiting pressure and which also provides long-term cooling following a LOCA. The design pressure is not exceeded in any pressure transients which result from combining the effects of heat sources with minimal operation of the engineered safety features.

The containment system is designed to provide for protection of the public from the consequences of a LOCA based on a postulated break of the reactor coolant piping up to and including a doubled-ended break of the largest reactor coolant pipe. Periodic containment leak rate measurements ensure that the leaktight barrier is maintained.

Criterion 17 - Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the 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 power sources, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

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

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

Compliance

The capacity and capability of either the onsite or offsite electric power system is sufficient to assure that (1) specified 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.

Offsite Electrical Power System

The offsite electrical power source consists of two physically independent circuits which are energized and available. The offsite sources are two independent 161-kV transmission lines terminating at the 161-kV switchyard, providing power to the plant, on demand, via the common station service transformers to the onsite Class 1E distribution system.

Onsite Electrical Power System

The onsite electrical power system serves both nuclear power units and certain common plant equipment. It consists of two independent diesel generator systems, each system containing two diesel generator units, two redundant Class 1E electric power distribution trains, and four redundant vital instrument and control power channels, each provided with an uninterruptible ac power supply and distribution panel. A plant Class 1E dc power system is provided with four redundant divisions, each consisting of a battery, battery charger, and distribution panel. Each redundant onsite power supply, train, and channel has the capability and capacity to supply the required safety loads assuming the failure of its redundant counterpart.

For a detailed description and analysis of the offsite electrical power system and onsite electrical power system, see Sections 8.2 and 8.3, respectively.

Criterion 18 - Inspection and Testing of Electric Power Systems

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

Compliance

Inspection

In addition to continuous surveillance by visual and audible alarms for any abnormal condition, the onsite power system is designed to permit inspection and checking of wiring, insulation, connections, and switchboards to the extent that personnel safety is not jeopardized, equipment not damaged, and the plant not exposed to accidental tripping.

On-Line Testing

The onsite power system is designed with provision for periodic testing during normal operation with the unit on line, to the extent that the plant is not exposed to accidental tripping and the reliability of the safety system not degraded. These features include provisions for starting and loading of onsite emergency diesel generators, and starting and loading of individual or groups of engineered safeguards to their respective buses. The system is also designed to permit testing of larger integrated segments of the system during planned cooldown of the reactor coolant system.

Off-Line Testing

The onsite power system is designed with facilities for a complete test of the operability of the system as a whole from initiation of protection system, starting and loading of the diesel generators, transfer of power sources and the full operational sequence of engineered safety features.

Inspection and testing of electrical power systems is further described in Sections 8.3.1.1 and 8.3.2.1.

Criterion 19 - Control Room

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

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

Compliance

The plant is provided with a separate structure designated as the Control Building. Within the Control Building are located control rooms, auxiliary instrument room, computer room, battery and dc equipment rooms (including space for instrument motor generator, etc), switchyard relay room, plant communications room and service facilities such as shift engineer office, kitchen, instrument shop, toilet facilities, and mechanical equipment room for heating, ventilating, and air conditioning equipment.

The main control room was provided with unit control panels for each of the two units, the switchyard, electrical recording, dc distribution, operation of the diesel generator system, and for those systems shared by the two units. The unit control panels contain those instruments and controls necessary for operation of the unit, functions such as the reactor and its auxiliary system, turbine generator, and the steam and power conversion systems. Selection of loading from the various plant electrical distribution boards such as the startup boards, common service board, shutdown boards, and motor control centers is accomplished from the unit control panels.

The control room is designed and equipped to minimize the effects of possible events such as fire, high radiation levels, and excessive temperature which might preclude occupancy. The main control room is continuously occupied by qualified operating personnel under all operating and accident conditions except in the case of events such as fire or smoke which could necessitate its evacuation. In the unlikely event that control room occupancy becomes impossible, provisions have been made to bring the reactor units to, and maintain them in, a hot shutdown condition, from a location external to the main control room. By use of appropriate procedures and available equipment, the unit can also be brought to cold shutdown conditions.

Sufficient shielding, distance, and containment integrity are provided to assure that under postulated accident conditions control room personnel shall not be subjected to radiation doses which would exceed 5 rem to the whole body, or its equivalent to any part of the body, including doses received during both ingress and egress. Control room ventilation is provided by a system having a large percentage of recirculated air. After an accident, makeup air is automatically routed through a system of HEPA and charcoal filters.

The design of the control room for occupancy during accidents is discussed in Section 6.4. The heating, ventilation, and air conditioning of the Control Building is discussed in Section 9.4. Radiation doses to control personnel following a LOCA are evaluated in Section 15.5.3. Radiation protection design features are discussed in Section 12.3.

3.1.2.3 Protection and Reactivity Control Systems

Criterion 20 - Protection System Functions

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

Compliance

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 accord with IEEE Standard 279-1971. The reactor trip system automatically initiates a reactor trip when any appropriate monitored variable or combination of variables exceed the normal operating range. Setpoints are chosen to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all the full length rod cluster control assemblies. This will allow the assemblies to free fall into the core, rapidly reducing reactor power output.

The engineered safety features actuation system automatically initiates emergency core cooling, and other safeguards functions, by sensing accident conditions using redundant process protection system channels measuring diverse parameters. Manual actuation of safeguards is relied upon where ample time is available for operator action. The ESF actuation system also provides a reactor trip on manual or automatic safety injection (S) signal generation.

The response and adequacy of the protection systems is analyzed for all conditions specified by the ANS N18.2 standard through Condition IV.

For further discussion of the reactor trip system and engineered safety features actuation system, see Sections 7.2 and 7.3, respectively.

Criterion 21 - Protection System Reliability and Testability

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

Compliance

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

The protection system is designed in accordance with IEEE Standard 279-1971. All safety actuation circuitry is provided with a capability for testing with the reactor at power. The protection systems, including the process protection system, nuclear instrumentation system and the engineered safety features test cabinet comply with Regulatory Guide 1.22 on periodic testing of protection system actuation functions. Under the present design, there are protective functions which are not tested at power. The functions can be tested under shutdown plant conditions, so that they do not interrupt power operation, as allowed by Regulatory Guide 1.22. For those process protection functions that may be tested in bypass, alarms are provided in the control room and at the process rack to indicate the bypassed condition. Additional information on the capability of the process protection system to be tested in the bypassed mode is provided in Section 7.2.2.2, Subsections 10, 11, 12, 13 and 14.

In those cases where equipment cannot be tested at power, it is only the actuation device function which is not tested. The logic associated with the actuation devices has the capability for testing at power. Such testing will disclose failures or reduction in redundancy which may have occurred. Removal from service of any single channel or component does not result in loss of minimum required redundancy. For example, a two-of-three function becomes a one-of-two function when one channel is removed. (Note that this is not true for the logic trains which are effectively a one-out-of-two logic).

Semiautomatic testers are built into each of the two logic trains in a protection system. These testers have the capability of testing the major part of the protection system very rapidly while the reactor is at power. Between tests, a number of internal protection system points including the associated power supplies and fuses are continuously monitored. Outputs of the monitors are logically processed to provide alarms for failures in one train and automatic reactor trip for failures in both trains. Self-testing provision is designed into each tester. Additional details can be found in Sections 7.2 and 7.3.

Criterion 22 - Protection System Independence

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

Compliance

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

Sufficient redundancy and independence is designed into the protection system to assure that no single failure, or removal from service of any component or channel of a system, will result in loss of the protection function. The minimum redundancy is exceeded in each protection function which is active with the reactor at power. Functional diversity and consequential location diversity are designed into the system. For example, loss of one feedwater pump would actuate one pressure reactor trip, one high-level trip, one low-level trip, and two temperature trips. The protective system is discussed in detail in Sections 7.2 and 7.3.

Criterion 23 - Protection System Failure Modes

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

Compliance

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 and engineered safety features actuation channel (except for containment spray and switchover from injection to recirculation) is designed on the deenergize-to-trip principle so loss of power, disconnection, open-channel faults, and the majority of internal channel short-circuit faults cause the channel to go into its tripped mode. The protection system is discussed in Sections 7.2 and 7.3.

Criterion 24 - Separation of Protection and Control Systems

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

Compliance

The protection system is separate and distinct from control systems. Control systems may be dependent on the protection system in that control signals are derived from protection system measurements where applicable. These signals are transferred to the control system by isolation devices which are classified as protection components. The adequacy of system isolation has been verified by testing under conditions of postulated credible faults. The failure or removal 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 discussion of details of compliance, see Chapter 7.

Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

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

Compliance

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 system.

Criterion 26 - Reactivity Control System Redundancy and Capability

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

Compliance

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

During operation the shutdown rod banks are fully withdrawn. The full length 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 full length control banks are designed to shutdown 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 boron system will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCAs are presented in Chapter 4 and the operation is discussed in Chapter 7. The means of controlling the boric acid concentration is described in Chapter 9. Performance analyses under accident conditions are included in Chapter 15.

Criterion 27 - Combined Reactivity Control Systems Capability

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

Compliance

Sufficient shutdown capability is provided to maintain the core subcritical for any anticipated cooldown transient, e.g., accidental opening of a steam bypass or relief valve, or safety valve stuck open. This shutdown capability is achieved by a combination of RCCA insertion and automatic boron addition via the emergency core cooling system with the most reactive control rod assumed to be fully withdrawn. Manually controlled boric acid addition is used to supplement the RCCA in maintaining the shutdown margin for the long-term conditions of xenon decay and plant cooldown. For further discussion, see Sections 4.3 and 7.2.

Criterion 28 - Reactivity Limits

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

Compliance

The maximum reactivity worth of control rods and the maximum rate of reactivity insertion employing control rods and boron removal are limited to values that prevent rupture of the reactor coolant system boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The appropriate reactivity insertion rate for the withdrawal of RCCA and the dilution of the boric acid in the reactor coolant systems are specified in the Technical Specifications for the facility. The specification includes appropriate graphs that show the permissible mutual withdrawal limits and overlap of functions of the several RCCA banks as a function of power. These data on reactivity insertion rates, dilution and withdrawal limits are also discussed in Section 4.3. The capability of the chemical and volume control system to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 9. The relationship of the reactivity insertion rates to plant safety is discussed in Chapter 15.

Criterion 29 - Protection Against Anticipated Operational Occurrences

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

Compliance

The protection and reactivity control systems are designed to assure an extremely high probability of fulfilling their intended functions. The design principles of diversity and redundancy coupled with a rigorous Quality Assurance Program and analyses support accomplishing this probability as does operating experience in plants using the same basic design. Section 4.2.3 and Sections 7.2 and 7.7 describe design bases and system design.

3.1.2.4 Fluid Systems

Criterion 30 - Quality of Reactor Coolant Pressure Boundary

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

Compliance

All reactor coolant system components are designed, fabricated, inspected and tested in conformance with ASME Boiler and Pressure Vessel Code, Section III. Leakage is detected by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The reactor vessel closure joint is provided with a temperature monitored leakoff between double 0-rings.

Leakage inside the reactor containment is drained to the Reactor Building sump where the level is monitored. Leakage is also detected by measuring the airborne activity and humidity of the containment.

See Section 5.2 for compliance of reactor coolant system components with ASME Boiler and Pressure Vessel Code, Section III.

Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

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

Compliance

Close control is maintained over material selection and fabrication for the reactor coolant system to assure that the boundary behaves in a nonbrittle manner. The reactor coolant system materials which are exposed to the coolant are corrosion resistant stainless steel or Inconel. The reference temperature RT_{NDT} of the reactor vessel material samples is established by Charpy V-Notch Tensile and 1/2 T compact tension tests. These tests also insure that materials with proper toughness properties and margins are used.

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

1. A complete independent review of the supplier stress analysis is conducted by Westinghouse on the reactor vessel. Independent stress analysis is conducted in selected areas to ascertain that the design conditions imposed by the Westinghouse specification have been adequately accounted for.

2. The reactor vessel received a complete stress analysis, including analysis for cyclic pressure and temperature operation. The ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Class 1 rules to which these components are designed generally exempt them from cyclic analysis by code Paragraph NB-3222.4 (d).

3. Welding Preheat Requirements - On the reactor vessel, the non-mandatory preheat requirements for P1 and P12 material were performed on all weldments.

4. Reactor Vessel Out-of-Roundness Requirements - To ensure uniform coolant flow, the Westinghouse out-of-roundness requirements on the cylindrical region in the area of the thermal shield are above code. Section III, Class 1 out-of-roundness requirements are stated in Paragraph NB-4221.1 of the code. This referenced paragraph states that the difference in inches between the maximum and minimum inside diameters at any cross section shall not exceed the smaller of (D + 50)/200 and D/100, where D is the nominal inside diameter in inches at the cross section under consideration. Westinghouse required the out-of-roundness to be less than O.5 percent of the diameter in the cylindrical section of the vessel in the region of the thermal shield.

Special requirements were imposed by Westinghouse on the quality control procedure for both the basic materials of construction, and on various sub-assemblies and final assembly for the reactor coolant loop components. These requirements supplemented the rules for quality assurance stated in the applicable design codes. Examples of the special quality assurance requirements for the reactor vessel that are beyond code requirements are:

<u>Ultrasonic Examinations</u>

1. A 100-percent shear wave ultrasonic test of plate material.

2. An ultrasonic test of cladding bond.

3. Weld buildup areas to which the core support pads are attached are examined 100 percent.

4. Selected areas of completed vessel are ultrasonically mapped after hydrotest to provide a base for future in-service inspection.

5. Ultrasonic examination of the entire volume of all full penetration welds and heat affected zones in primary pressure boundary welds. The testing was done during fabrication upon completion of the welding and intermediate heat treatment.

Dye Penetrant Testing

1. Dye penetrant test all cladding surfaces and other vessel and head internal surfaces after hydrotest.

2. Dye penetrant examine the weld between the bottom head and instrumentation tubes, after each layer of weld is deposited.

3. Dye penetrant examine weld between CRDM housing and closure head and vent pipe and closure head after the first layer; each 1/4 inch of weld deposited and final surface.

4. Dye penetrant examine weld between the lower core support pad and the vessel shell, after the first layer and each 1/2 inch of weld metal are deposited.

5. Base metal or weld metal surfaces which are exposed to mechanical operations was dye penetrant or magnetic particle inspected.

Magnetic Particle Testing

1. Magnetic particle examination of all exterior vessel and head surfaces after hydrotest.

2. Magnetic particle examination of welds attaching the vessel supports, closure head lifting lugs, and refueling seal ledge to the reactor vessel, after the first layer and each 1/2 inch of weld metal are deposited.

3. Magnetic particle examination of all closure stud surfaces after threading. Continuous circular and longitudinal magnetization was used.

4. Magnetic particle examination of I.D. surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection was performed after forming and prior to cladding.

The fabrication and quality control techniques used in the fabrication of the reactor coolant system were equivalent to those for the reactor vessel. The inspections of reactor vessel, pressurizer, piping and steam generator were governed by ASME code requirements.

The permissible pressure - temperature relationships for selected heatup and cooldown rates were calculated using the methods of ASME Code Section III Non-mandatory Appendix G. The change in RT_{NDT} due to irradiation during plant life was calculated using conservative methods and will be verified periodically by surveillance program irradiated material test data.

See Section 5.2 for further discussion of compliance.

Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

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

Compliance

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

The RT_{NDT} properties of the reactor vessel core region forging, weldments and associated heat treated zones will be monitored by a surveillance program which is based on ASTM-E-185, Recommended Practice for Surveillance Testing on Structural Materials in Nuclear Reactors. Samples of reactor vessel plate materials will be retained and catalogued in case future engineering development shows the need for further testing.

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

To define permissible operating conditions below RT_{NDT}, a pressure range was established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit was defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected RT_{NDT}, brittle fracture during normal operation is not considered to be a credible mode of failure. Additional details can be found in Section 5.2.

Criterion 33 - Reactor Coolant Makeup

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

Compliance

The chemical and volume control system includes charging pumps and makeup paths that serve the safety function of maintaining reactor coolant inventory during normal operations and in the event of small reactor coolant leakages. The charging pumps can maintain reactor coolant pressure sufficiently high to allow orderly reactor shutdown for small tubing or small pipe breaks. Chapter 5 discusses the reactor coolant system, Section 9.3.4 discusses the chemical and volume control system, and Chapter 15 analyzes charging pump performance and fuel damage in event of postulated accidents. The offsite power system and onsite power system are discussed in Sections 8.2 and 8.3, respectively.

Criterion 34 - Residual Heat Removal (RHR)

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.

Compliance

The RHR system includes two redundant trains of pumps and heat exchangers each having sufficient heat removal capability to ensure fuel protection. The system is Seismic Category I and is provided electric power by either the preferred power system or the diesel generators of the standby power system. The normal steam and power conversion system is used for the first stage cooldown (i.e., above 350°F and 400 psig). The auxiliary feedwater system provides guaranteed backup of the steam and power conversion system in this function. The systems together accommodate the single-failure criterion.

Section 5.5.7 describes the RHR System.

Criterion 35 - Emergency Core Cooling

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

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

Compliance

The Emergency Core Cooling System (ECCS) design and safety analysis is in accordance with the NRC Acceptance Criterion for Emergency Core Cooling System for Light-Water Power Reactors of December 1973 (10 CFR 50.46).

By combining the use of passive accumulators, centrifugal charging pumps, safety injection pumps, and residual heat removal pumps, emergency core cooling is provided even if there should be a failure of any component in any system. The ECCS employs passive system of accumulators which do not require any external signals or source of power. Two independent and redundant pumping systems are also provided to supplement the passive accumulator system. These systems are arranged sothat the single failure of any active component does not prevent meeting the short-term cooling requirements.

The primary function of the ECCS is to deliver borated cooling water to the reactor core in the event of a LOCA. This limits the fuel-clad temperature and thereby ensures that the core will remain intact and in place and fuel damage will not exceed that stipulated as a basis in the safety analysis (Chapter 15). This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical circumferential rupture of a reactor coolant loop,

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

The ECCS is described in Section 6.3. The LOCA, including an evaluation of consequences, is discussed in Chapter 15.

Criterion 36 - Inspection of Emergency Core Cooling System

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

Compliance

Design provisions facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes and valves for visual or nondestructive inspection.

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

Details of the inspection program for the reactor vessel internals are included in Section 5.4. Inspection of the ECCS is discussed in Section 6.3.

Criterion 37 - Testing of Emergency Core Cooling System

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

Compliance

The design provides for periodic testing of both active and passive components of the ECCS.

Proof tests of the components were performed in the manufacturer's shop. Preoperational system hydrostatic and performance tests demonstrate structural and leaktight integrity of components and proper functioning of the system. Thereafter, periodic tests demonstrate that components are functioning properly.

Design provisions include special instrumentation, testing, and sampling lines to perform the tests during plant shutdown to demonstrate proper automatic operation of the ECCS.

Each active component of the ECCS may be individually actuated on the normal power source at any time during plant operation to demonstrate operability. Components are actuated on the emergency power system during preoperational tests and subsequently during plant shutdown per Technical Specifications.

Details of the ECCS are found in Section 6.3, with periodic testing procedures identified in Section 6.3.4. Performance under accident conditions is evaluated in Chapter 15.

Criterion 38 - Containment Heat Removal

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

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

Compliance

Systems are provided to effect post-accident containment heat removal. The systems are classified as engineered safety features and as such incorporate a large degree of redundancy as well as being provided with multiple power supplies.

Containment heat removal is provided by the ice condenser and by containment sprays. The ice condenser is a passive system consisting of energy absorbing ice on which steam is condensed during and immediately after a LOCA. The condensation of steam on the ice limits the pressure and temperature to values less than containment design.

An air return system is used to circulate the containment gaseous inventory through the upper compartment, lower compartment, and ice condenser after the initial blowdown. This maintains proper mixing of the containment air and steam with the heat removal media, spray and ice, for the necessary heat removal.

The containment spray system sprays coolant automatically into the upper compartment containment atmosphere in the event of a large LOCA, thereby removing containment heat. The recirculation mode allows for a long-term heat removal by means of two spray systems, each of which contains redundant components including spray headers. The containment spray system consists of two completely separate trains consisting of pumps, heat exchangers, valves, and headers. The containment spray system is initiated automatically upon containment high pressure and is later manually realigned for proper operation in the recirculation mode. The residual heat removal spray contains two spray headers which are supplied from separate trains of the residual heat removal system by manual diversion of a portion of the low-pressure safety injection system flow during recirculation.

The loss of a single active component was assumed in the design of these systems. Emergency power system arrangements assure the proper functioning of the air return fan system, and the containment spray system and residual heat removal sprays.

The engineered safety features systems are discussed in Chapter 6; the electric power systems in Chapter 8; the protection systems in Chapter 7.

Criterion 39 - Inspection of Containment Heat Removal System

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

Compliance

The ice condenser design includes provisions for visual inspections of the ice bed flow channels, doors, and cooling equipment. The air return fan system provides for visual inspection of the fans and the associated backflow dampers and for duct systems that are not embedded in concrete. The containment spray system and the RHR sprays are designed such that active and passive components can be readily inspected to demonstrate system readiness. Pressure contained systems are inspected for leaks from pump seals, valve packing, flange joints, and relief valves. During operational testing of the containment spray pumps and RHR pumps, the portions of the systems subjected to pressure are inspected for leaks.

System design details are given in Section 6.2.

Criterion 40 - Testing of Containment Heat Removal System

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

Compliance

The containment heat removal systems described in Section 6.2 are designed to permit periodic testing so that proper operation can be assured. In some cases whole systems can be operated for test purposes. In others, individual components are operated for functional tests so that plant operations are not disrupted.

The ice condenser contains no active components, other than the ice condenser doors, which are required to function during an accident condition. Samples of the ice are taken periodically and tested for boron concentration. The lower inlet door opening force is measured when the reactor is in the shutdown condition. The position of the lower inlet doors is monitored at all times.
Top deck door and intermediate deck doors are tested for operability during the shutdown condition. Air return fans and their associated backflow dampers are tested for operability while the reactor is shut down for refueling.

All active components of the containment spray system and the residual heat removal spray system are tested in place after installation. These spray systems receive initial flow tests to assure proper dynamic functioning. Further testing of the active components is conducted after component maintenance and in accordance with technical specifications. Air test lines, located upstream of the spray isolation valves, are provided for testing to assure that spray nozzles are not obstructed. Testing of transfer between normal and emergency power supplies is also conducted.

Criterion 41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality 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.

Compliance

The Shield Building, surrounding the primary containment, serves as a secondary containment. During accident conditions prior to containment isolation, primary and secondary containment purge exhaust is processed by the containment purge system filters prior to release to the atmosphere. The emergency gas treatment system (Section 6.2) maintains this secondary containment at a negative pressure during the entire post-accident period. The emergency gas treatment system also collects and processes the secondary containment atmosphere. After processing, the portion of this processed air necessary to assure a negative pressure is exhausted through the Shield Building exhaust vent. The remainder is recirculated and distributed in the secondary containment.

The Auxiliary Building serves to collect any equipment leakage during the recirculation of containment sump water. The Auxiliary Building ventilation system (Section 9.4) is isolated by the containment Phase A isolation signal. The Auxiliary Building gas treatment system (Section 9.4) then maintains the building at a negative pressure and processes any inleakage prior to release to the environment.

Post-accident hydrogen control within the containment is provided by the hydrogen mitigation system (Section 6.2.5). Distribution of the atmosphere within the containment is provided by the air return fan system (Section 6.8). The air return fan system also takes suction from each compartment to prevent stagnation and excessive accumulation of hydrogen.

Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

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

Compliance

The emergency gas treatment system (Section 6.2) filtration train and fans and the containment purge filters (Section 9.4.6) are located in the Auxiliary Building and are designed to facilitate inspections. The dampers that control recirculation and exhaust of the emergency gas treatment system effluent are located inside the Shield Building and may be inspected during reactor shutdown.

The entire Auxiliary Building gas treatment system (Section 9.4.3) is located in the Auxiliary Building and is designed to facilitate inspection.

Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

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

Compliance

The containment purge system (Section 9.4) is designed to permit testing to assure leaktightness of the filter trains; functional testing to assure operability of containment isolation valves; and performance testing to assure filter efficiency and to demonstrate the isolation valve closure in response to the accident mode isolation signal.

The emergency gas treatment system (Section 6.2) is designed to permit testing to assure leaktightness of the filtration trains; functional testing to assure operability of the fans, dampers, and instrumentation; and performance testing to assure overall operability of the system and to demonstrate the proper alignment of the system to the accident unit.

The Auxiliary Building gas treatment system (section 9.4) is designed to allow testing to assure the pressure and leaktightness of the filters, adsorbers, and the filtration train housing to assure the operability of the fans and dampers; and to assure the operability of the system as a whole. The system design will permit testing of the actuation signals, the isolation of the normal ventilation system, and the proper alignment of dampers.

The hydrogen mitigation system (Section 6.2.5) is designed to allow testing to assure the operability of the manual controls that place the systems in operation. The system is designed to permit, under conditions as close to design as practical, the operability of the system as a whole.

Criterion 44 - Cooling Water

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

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

Compliance

A Seismic Category I Component Cooling System (CCS) (Section 9.2) is provided to transfer heat from the reactor coolant system reactor support equipment and engineered safety equipment to a Seismic Category I Essential Raw Cooling Water (ERCW) system (Section 9.2).

The CCS serves as an intermediate system and thus a barrier between potentially or normally radioactive fluids and the river water which flows in the ERCW system.

The CCS consists of two independent engineered safety trains, each of which is capable of serving all required loads under normal or accident conditions that are important to safety.

In addition to serving as the heat sink for the CCS, the ERCW system is also used as heat sink for the containment through use of the containment spray heat exchangers, and engineered safety equipment through use of compartment and space coolers. The ERCW system consists of two independent trains, each of which is capable of providing all necessary heat sink requirements. The ERCW system transfers heat to the ultimate heat sink (Section 9.2).

Electric power is discussed in Chapter 8.

Criterion 45 - Inspection of Cooling Water System

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

Compliance

The integrity and capability of the component cooling water system (Section 9.2) and essential raw cooling water system (Section 9.2) will be monitored during normal operation by the Surveillance Instruction Program. Nonsafety related systems may be isolated temporarily for inspection. All major components will be visually inspected on a periodic basis.

The component cooling and essential raw cooling water pumps are arranged such that any pump may be isolated for inspection and maintenance while maintaining full plant operational capabilities.

Criterion 46 - Testing of Cooling Water System

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

Compliance

The cooling water systems will be pressurized during plant operations; thus, the structural and leaktight integrity of each system and the operability and performance of their active components will be continuously demonstrated. In addition, normally idle portions of the piping system and idle components will be tested during plant shutdown. The emergency functions of the systems will be periodically tested out to the final actuated device in accordance with the technical specifications.

For details, see the discussions on electric power (Chapter 8), component cooling water (Section 9.2), essential raw cooling water (Section 9.2), and instrumentation and controls (Chapter 7).

3.1.2.5 Reactor Containment

Criterion 50 - Containment Design Basis

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

Compliance

The containment structure, including access openings and penetrations, is designed with sufficient conservatism to accommodate, without exceeding the design leakage rate, the transient peak pressure and temperature associated with a postulated reactor coolant piping break up to and including a double-ended rupture of the largest reactor coolant pipe.

The containment design consists of a freestanding steel containment vessel and a separate outer reinforced concrete shield wall and roof. The ice condenser concept is used for energy absorption during a LOCA. The annular space between the containment vessel and the exterior shield wall forms a double barrier to fission products and is maintained at less than atmospheric pressure. The ice condenser, which is located inside the steel containment and consists of a suitable quantity of borated ice in a cold storage compartment, provides rapid energy absorption to maintain the containment vessel design pressure at a low level and to reduce the peak duration, thus reducing the potential for escape of fission products from the primary containment vessel.

The functional design of the containment is based upon the following assumptions and conditions:

1. A design basis blowdown energy and mass release.

2. Secondary energy released by safety injection.

3. Carryover energy from zirconium-water reaction.

4. Decay heat from the reactor at rated power.

5. The single failure criterion is accommodated.

The internal design pressure of the containment is greater than the peak pressure occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical double-ended severance of the largest reactor coolant pipe. The design pressure is not exceeded during any subsequent long-term pressure transient.

Refer to Section 3.8 for a description of containment, and to Section 6.2 for design basis details.

Criterion 51 - Fracture Prevention of Containment Pressure Boundary

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

Compliance

The containment vessel and its penetration sleeves meet the material, design and technical process requirements of ASME Boiler and Pressure Vessel Code, Section III, Class B. Charpy V-notch impact tests were made of the containment vessel material (ASTM A 516, Grade 70) 5/8 inch and greater, weld deposit, and the base metal weld heat affected zone employing a test temperature at least 30° F below minimum service temperature in accordance with ASME Code, Paragraph N-1210. This test measured the ductile to brittle transition with allowable values for energy absorption given in Tables N-421 and N-422. It insures that the material used will not behave in a brittle manner and that rapidly propagating fracture is minimized. The containment boundary design considered uncertainties in material properties, residual, steady-state and transient stresses, and material flaws along with conservative allowable stress levels for all stressed elements of the containment boundary. All material was examined for flaws that would adversely affect the performance of the material in its intended location. See Section 6.2 for further details.

Criterion 52 - Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Compliance

The reactor containment design permitted overpressure strength testing during construction and permits preoperational integrated leakage rate testing at containment design pressure and at reduced pressure, in accordance with Appendix J, 10 CFR 50. The reactor containment and other equipment which may be subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. All equipment which may be subjected to the test pressure is either vented to the containment, removed from the containment during the test, or designed to withstand the containment design pressure without damage.

The preoperational integrated leak tests at peak pressure verify that the containment, including the isolation valves and the resilient penetration seals, leaks less than the allowable value of 0.25 weight percent per day at peak pressure.

Details concerning the conduct of periodic integrated leakage rate tests are in Section 6.2.

Criterion 53 - Provisions for Containment Testing and Inspection

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

Compliance

The reactor containment and the containment isolation system (Section 6.2) are designed so that:

1. Integrated leak rate tests can be run during plant lifetime (see compliance to Criterion 52).

2. Visual inspections can be made of all important areas, such as penetrations.

3. An appropriate surveillance program can be maintained (see Section 6.2).

4. Periodic testing at containment design pressure of the leaktightness of isolation valves and penetrations which have resilient seals and expansion bellows is possible.

5. The operability of the containment isolation system can be demonstrated periodically.

In testing locally the resilient seals and expansion bellows leakages, the guidelines for Type B tests in Appendix J to 10 CFR 50 will be followed.

Criterion 54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundance, 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.

Compliance

Containment isolation features are classified as Seismic Category I. These components required quality assurance measures which enhance reliability. The containment isolation design provides for a double barrier at the containment penetration in those fluid systems that are not required to function following a design basis event.

All piping systems penetrating the containment, in so far as practical, have been provided with test vents and test connections or have other provisions to allow periodic leak testing as required. Section 6.2.4.4 has further details on testing.

See Section 6.2.4 for general containment isolation details and Section 6.2.4.3 for exceptions to General Design Criteria 54, 55, 56, and 57.

Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

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

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

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

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

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

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

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.

Compliance

The reactor coolant pressure boundary is defined as those piping systems and components which contain reactor coolant at design pressure and temperature. With the exception of the reactor coolant sampling lines, the entire reactor coolant pressure boundary, as defined above, is located entirely within the containment structure. All sampling lines are provided with remotely operated valves for isolation in the event of a failure. These valves also close automatically on a containment isolation signal.

All other piping and components which may contain reactor coolant are low pressure, low temperature systems which would yield minimal environmental doses in the event of failure.

The sampling system and low-pressure systems are described in Section 9.3. An analysis of malfunctions in these systems is included in Chapter 15.

Criterion 56- Primary Containment Isolation

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

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

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

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

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

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

Compliance

At least two barriers are provided between the atmosphere outside the containment and the containment atmosphere, the reactor coolant system, or closed systems which are assumed vulnerable to accident forces.

Redundant valving is provided for piping that is open to the atmosphere and to the containment atmosphere. Additional details can be found in Section 6.2.

Criterion 57 - Closed Systems Isolation Valves

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

Compliance

Those lines that penetrate the containment, that do not communicate with either the reactor coolant pressure boundary or the containment atmosphere, and that are not affected by LOCA forces, are defined as closed systems. All lines penetrating the containment are designed to meet this GDC.

See Section 6.2.4 for a discussion of containment isolation valves.

3.1.2.6 Fuel and Radioactivity Control

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

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Compliance

Liquid, gaseous, and solid radioactive waste processing equipment is provided. The principles of filtration, demineralization, evaporation, solidification and storage for decay are utilized as described in Chapter 11. Process monitoring is provided to control this equipment and regulate releases to the environment as described in Section 11.4.

Criterion 61 - Fuel Storage and Handling and Radioactivity Control

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

Compliance

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

1. Components are designed and located such that appropriate periodic inspection and testing may be performed.

2. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Section 12.1.

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

4. The spent fuel cooling systems provide cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed for testability to permit continued heat removal.

5. The spent fuel pool is designed such that no postulated accident could cause excessive loss of coolant inventory.

Radioactive waste treatment systems are located in the Auxiliary Building, which contains or confines leakage under normal and accident conditions.

The Auxiliary Building Gas Treatment System (ABGTS) includes charcoal filtration which can be used to minimize radioactive material releases associated with a postulated spent fuel handling accident. The ABGTS system is not required to mitigate the consequences of a spent fuel handling accident.

Fuel storage and handling and spent fuel storage at the ISFSI are discussed in section 9.1, and radioactive waste management in Chapter 11.

Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

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

Compliance

The restraints and interlocks provided for safe handling and storage of new or spent fuel are discussed in Section 9.1.

The center-to center distance between adjacent spent fuel assemblies together with the use of fixed Boral neutron absorber panels in the storage racks and burnup credit administrative controls on fuel assembly placement are sufficient to ensure subcriticality, even if unborated water is used to fill the spent fuel storage pool. Credit for borated water is permitted to maintain subcriticality for inadvertent misplacement of a fuel assembly, e.g., loading of a fresh fuel assembly in a storage cell designated for exposed fuel or placement outside of and adjacent to a rack module.

Layout of the fuel handling area is such that the spent fuel casks will never be required to traverse the spent fuel storage pool during removal of the spent fuel assemblies.

Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling area (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.

Compliance

Failure in the spent fuel cooling system will result in control room annunciation and local temperature indication.

High radiation in the spent fuel storage (refuel floor) area will produce the following alarms:

1. Main Control Room alarm from the spent fuel pool area accident monitors.
2. Local and Main Control Room alarms from the refuel floor area monitor.
3. Local alarm from a portable continuous air monitor located on the refuel floor.
4. Main Control Room alarm from the Auxiliary Building Ventilation Radiation Monitor for high airborne radiation in the spent fuel storage area.

High radiation in the waste packaging area will result in the following alarms:

1. Main Control Room and local alarms from the waste packaging area monitor.
2. Local alarms from a portable continuous air monitor located in the area.
3. Main Control Room alarm from the Auxiliary Building Ventilation Radiation Monitor for high airborne radiation in the waste packaging area.

See Sections 9.1 and 12.3 and Chapter 11 for further details.

Criterion 64 - Monitoring Radioactivity Releases

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

Compliance

The facility contains means for monitoring the containment atmosphere and all other important areas during both normal and accident conditions to detect and measure radioactivity which could be released under any conditions. The monitoring system includes area gamma monitors, atmospheric monitors and liquid monitors with full indication in the control room. Alarms are provided to warn of high radioactivity.

Chapter 11 discusses the process and effluent radiation monitoring systems. Chapter 12 discusses the area and airborne radiation monitoring systems.

REFERENCES

None

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 Seismic Classifications

The Watts Bar Nuclear Plant structures, systems, and components which perform a primary safety function have been designed to remain functional in the event of a Safe Shutdown Earthquake (SSE). These structures, systems, and components, designated as Seismic Category I, are those necessary to assure:

1. The integrity of the reactor coolant pressure boundary.

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

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to or in excess of the guideline exposures of 10 CFR Part 100.

These structures, systems, and components are classified in accordance with Regulatory Guide 1.29 unless exception is taken in detailed classification information provided in other sections of the UFSAR, such as Table 3.2-2a, 3.2-6, etc.

Piping, pumps, valves, and other fluid system components which must retain limited structural integrity because their failure could jeopardize to an unacceptable extent the achievement of a primary safety function, because they form an interface between Seismic Category I and non-Seismic Category I plant features, or because they perform a secondary safety function, are designated by TVA as Seismic Category I(L) (i.e., limited requirements). Those fluid containing elements which are included in Seismic Category I(L) are seismically qualified to meet the intent of Position 2 of Regulatory Guide 1.29. For Unit 2 the Seismic Category 1(L) is subdivided into categories 1L(A) Pressure boundary and position retention and 1L(B) position retention.

Where portions of mechanical systems are Category I or I(L) and the remaining portions not seismically classified, the systems have been seismically qualified to a terminating anchor (or other appropriate analysis problem termination) beyond the defined boundary such as a valve, thus meeting Position 3 of Regulatory Guide 1.29.

All Category I safety-related structures, and portions of mechanical and electrical systems and components are listed in Tables 3.2-1, 3.2-2, 3.2-2a, 3.2-2b and 3.2-3. Those Category I(L) portions of mechanical systems are also listed in Table 3.2-2.

3.2.2 System Quality Group Classification

Fluid system components for the Watts Bar Nuclear Plant that perform a primary safety function are identified by TVA Classes A, B, or C (see Section 3.2.2.7 for HVAC Safety Classifications). These piping classes are assigned to fluid systems based on the ANS Safety Classes 1, 2a, and 2b, respectively, which are assigned to nuclear power plant equipment per the August 1970 Draft of ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." Fluid system components whose postulated failure would result in potential offsite doses that exceed 0.5 Rem to the whole body, or its equivalent to any part of the body, are identified as TVA Class D and are based on ANSI N18.2 (Aug., 1970 draft) Safety Class 3 and Regulatory Guide 1.26. The TVA piping classification system for WBNP does not conform strictly to the guidance of Regulatory Guide 1.26 (which was not in effect on the docket date for the Construction Permit). The ANS safety classification of each component has been considered in the various aspects of design, fabrication, construction, and operation.

3.2.2.1 Class A

Class A applies to reactor coolant pressure boundary components whose failure could cause a loss of reactor coolant which would not permit an orderly reactor shutdown and cool down assuming that makeup is only provided by the normal makeup system. Branch piping 3/8 inch inside diameter and smaller, or protected by a 3/8 inch diameter or smaller orifice, is exempted from Class A requirements. The branch piping for the pressurizer steam space instrumentation nozzles (0.83 inch inside diameter, or smaller) is also exempted from Class A requirements.

The components which are within the Reactor Coolant Pressure Boundary (RCPB) and meet all the following requirements may be classified as TVA Class G:

1. Piping and associated components in the RCPB which penetrate containment excluding the actual penetration and its associated components.

2. Piping and associated components which perform no primary safety function.

3. Piping and associated components which are isolated by a normally closed valve off a line in the RCPB that meets the exclusion requirements of 10 CFR Part 50.55a paragraphs (c) (1) and (c) (2). An example would be the ECCS check valve leak test lines.

3.2.2.2 Class B

Safety Class B applies to those components of safety systems necessary to fulfill a system safety function. The classification is specifically applicable to containment and to components of those safety systems, or portions thereof, through which reactor coolant water flows directly from the reactor coolant system or the containment sump.

3.2.2.3 Class C

Class C applies to components of those safety systems that are important to safe operation and shutdown of the reactor but that do not recirculate reactor coolant.

3.2.2.4 Class D

Class D applies to components not in TVA Class A, B, or C whose failure would result in release to the environment of gaseous radioactivity normally held up for radioactive decay. This is being interpreted as those portions of systems whose postulated failure would result in calculated potential offsite doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body.

3.2.2.5 Relationship of Applicable Codes to Safety Classification for Mechanical Components

The applicable codes used for the design, material selection, and inspection of components for the various safety classes are shown in Table 3.2-4. The applicable TVA classification and ANS Safety Classification for each of the fluid systems are tabulated in Table 3.2-2. TVA classifications are also delineated on flow diagrams which have been included as figures in those
sections of the UFSAR where the systems are discussed in detail.

3.2.2.6 Nonnuclear Safety Class (NNS)

Components that are used in Seismic Category I structures whose failure would not result in a release of radioactive products and are not required to function during an accident or malfunction within the reactor coolant pressure boundary have been assigned TVA Classifications G or K. Since these components complement components having a primary safety function during normal operation and may be in close proximity to them, they are seismically qualified as Seismic Category I(L) to the extent necessary to prevent an unacceptable influence on Safety Class equipment during a seismic event. Thus the minimum capability of primary system components is not compromised by the failure of a Class G or K component during a seismic event. Components which are assigned to TVA Class H or L, located inside Seismic Category I structures, are also designed as Seismic Category I(L). The applicable codes, along with the seismic classifications used for the design of the components covered by these classifications, are shown in Table 3.2-5. TVA Class P is assigned to specific sense lines located (in part or totally) in a non-seismic area.

3.2.2.7 Heating, Ventilation and Air Conditioning (HVAC) Safety Classification

Those portions of the HVAC Systems which are safety related have been assigned TVA classifications and have been designed to Seismic Category I and I(L) specifications as applicable. All equipment, components, duct work, etc., in the August 1970 Draft ANSI N18.2 Safety Classes 2a and 2b perform primary safety functions and are designed to Seismic Category I requirements, except as exempted in Table 3.2-6. Portions of systems not performing a safety function may need a degree of seismic qualification because their failure could produce an unacceptable influence on the performance of safety functions. These are designed to Seismic Category I(L) requirements. The applicable codes along with the seismic qualifications used for the design of the HVAC ducting are shown in Table 3.2-6. See Sections 3.7.3.17 and 3.7.3.18 for details of seismic analysis and design of HVAC duct and duct support systems.

3.2.3 Code Cases and Code Editions and Addenda

3.2.3.1 TVA Design and Fabrication

The Code of Record of Section III of the ASME Code applied to systems within TVA's scope is the 1971 Edition with Addenda through Summer 1973. The use of later Edition and Addenda, as permitted by paragraph NA-1140 of the ASME Code, is controlled to ensure the following:

a. Later Edition and Addenda used has been accepted by the NRC through incorporation by reference in 10 CFR 50.55a.

b. Related requirements necessary to support use of later Edition and Addenda are implemented in accordance with NA-1140.

c. Code Cases used have been accepted by the NRC through incorporation by reference in Regulatory Guide 1.84.

d. Additional requirements added by Regulatory Guide 1.84 are implemented.

A listing of Code Cases and provisions of later Code editions and addenda which have been used for design and fabrication is given in Table 3.2-7. A similar listing of Code Cases and provisions of later Code editions and addenda used in analysis of fluid systems is given in Section 3.7.3.8.1. Another similar listing for the RCS is given in Section 5.2.1.4. Code cases and provisions of later Code Editions and Addenda associated with Inservice Inspection and Inservice Testing are found in the Inservice Inspection and Inservice Testing program. Exceptions to the system classification Code requirements associated with Generic Letter 89-09 may be found in notes on the flow diagrams.

3.2.3.2 Purchased Materials and Components

The Code of Record for components ordered by TVA is determined in accordance with 10 CFR 50.55a, footnote 5. Material ordered by TVA and supplied with certification to a later Edition and Addenda is controlled by a comparison of the Edition and Addenda to which it is certified to the Code of Record applicable to the application in which it is used. Deviations from the applicable Code of Record are reconciled prior to use of the material.

Material procured prior to the initiation of the Acceptable Suppliers List (ASL) program (approximately May 1978) has been addressed through an NRC approved alternative to the ASME Code paragraph NA-3451(a)[1]. Material

- procured as ASTM material,
- installed or to be installed in an ASME system,
- whose proof of survey or qualification by TVA of the manufacturer's quality assurance program at the time of procurement cannot be retrieved, and
- whose material specification is identical to the requirements of ASME Section II as stated by the ASME material specification,

is acceptable for use assuming all other attributes of the material and the documentation conform to ASME Code requirements.

REFERENCES

1. Letter from B. D. Liaw, NRC, to O. D. Kingsley, TVA, dated March 15, 1990, "NRC Inspection Report Nos. 50-390/90-02 and 50-391/90-02".

2. Letter from Frederick J. Hebdon, NRR, to Mark O. Medford, TVA, dated February 22, 1993.

WBN

TABLE 3.2-1

CATEGORY I STRUCTURES

1. Reactor Building (Shield Building, Steel Containment Vessel, and Interior Concrete)
2. Auxiliary - Control Building
    a. Auxiliary Building portion
    b. Additional Equipment Building portion
    c. Control bay portion
    d. Waste packaging area
3. Condensate Demineralizer Waste Evaporator Building
4. Class 1E Electrical Systems Structures (Manholes, Handholes, and Conduit Banks)
5. Diesel Generator Building
6. ERCW Pipe Tunnels and RWST Foundations
7. ERCW Structures
8. North Steam Valve Room
9. Intake Pumping Station and Retaining Walls
10. Deleted
11. Refueling Water Storage Tank (RWST).
12. Underground Barrier.
13. ERCW Standpipe Structures I and II and ERCW Discharge Overflow Structure.

WBN
TABLE 3.2-2 (Sheet 1 of 23)
SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Reactor Vessel	W	A	(13)	X	C	X	I
Full Length CRDM Housing	W	A	III-1	X	C	X	I
CRDM Head Adapter Plugs	W	A	III-1	X	C	X	I
Steam Generators (Tube Side)	W	A	III-1	X	C	X	I
(Shell Side)	W	A(17)	III-1	X	C	X	I
Pressurizer	W	A	III-1	X	C	X	I
Reactor Coolant Pipe	W	A	III-1	X	C	X	I
Reactor Coolant Fittings	W	A	III-1	X	C	X	I
Reactor Coolant Fabricated Piping	W	A	III-1	X	C	X	I
Reactor Coolant Crossover Legs	W	A	III-1	X	C	X	I
Reactor Coolant Thermowell	W	A	III-1	X	C	X	I
Thimble Guide Tubing	W	A	III-1	X	C	X	I
Thimble Guide Couplings	W	A	III-1	X	C	X	I
Incore Instrument Thimble Assembly	W	B	III-2	X	C	X	I
Loop Bypass Line	W	A	III-1	X	C	X	I
Pressurizer Safety Valves	W	A	III-1	X	C	X	I
Power Operated Relief Valves	W	A	III-1	X	C	X	I
Pressurizer Relief Tank	W	G	VIII	X	C	P	I(L)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Reactor Coolant Pump	W	A	III-1	X	C	X	I
RCP Casing	W	A	III-1	X	C	X	I
Main Flange	W	A	III-1	X	C	X	I
Thermal Barrier	W	A	III-1	X	C	X	I
Thermal Barrier Heat Exchanger	W	C	(14)	X	C	X	I
No. 1 Seal Housing Bolts	W	A	III-1	X	C	X	I
Upper Seal Housing	W	B	III-1	X	C	X	I
Pressure Retaining Bolting	W	A	III-1	X	C	X	I
RCP Motor	W	2b	NEMA-MG1	X	C	-	I
Motor Rotor	W	2b	NA	X	C	-	I
Motor Shaft	W	2b	NA	X	C	-	I
Shaft Coupling	W	2b	NA	X	C	-	I
Spool Piece	W	2b	NA	X	C	-	I
Flywheel	W	2b	NA	X	C	-	I
Bearing (Motor Upper Thrust)	W	2b	NA	X	C	-	I
Motor Bolting	W	2b	NA	X	C	-	I
Motor Stand	W	2b	NA	X	C	-	I
Motor Frame	W	2b	NA	X	C	-	I

TABLE 3.2-2 (Sheet 3 of 23)

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Upper Oil Reservoir (UOR)	W	2b	NA	X	C	P	I
Upper Oil Cooler	W	2b	(14)	X	C	P	I
Lube Oil Piping	W	2b	(14)	X	C	P	I
Lower Oil Reservoir (LOR)	W	2b	NA	X	C	P	I
LOR Cooling Coil - CC (supports only)	W	2b	NA	X	C	P	I
Motor Air Coolers	W	-	-	X	C	-	I(L)A Note 12
Safety Injection System							
Safety Injection Pumps	W	B	III-2	X	AB	X	I
Accumulators (9)	W	B	III-2	X	C	P	I
Boron Injection Tank	W	B	III-2	X	AB	X	I
Refueling Water Storage Tank	T	B	III-2	X	O	P	I
Residual Heat Removal System							
RHR Pumps	W	B	III-2	X	AB	X	I
RHR Heat Exchangers (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Containment Spray System							
CS Pumps	T	B	III-2	X	AB	X	I
CS Heat Exchangers (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I
CS Nozzles	T	B	-	X	C	P	I
Primary Water Make-Up System							
Pump	T	G	ANSI B31.1	X	AB	-	I(L)
Tank Unit 1	T	G(22)	Note 25	X	O	-	I(L)
Tank Unit 2	T	G(26)	Note 26	X	O	-	I(L)
Chemical and Volume Control System Pumps							
Charging, Centrifugal	W	B	P&V-II	X	AB	X	I
Boric Acid Transfer	W	C	P&V-III	X	AB	-	I
Heat Exchangers							
Regenerative	W	B	III-2	X	C	X	I
Letdown (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I
Excess Letdown (Tube)	W	B	III-2	X	C	X	I
(Shell)	W	B	III-2	X	C	P	I

WBN
TABLE 3.2-2 (Sheet 5 of 23)
SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Seal Water (Tube)	W	B	III-2	X	AB	X	I
(Shell)	W	C	III-3	X	AB	P	I
Tanks							
Volume Control	W	B	III-2	X	AB	X	I
Boric Acid	W	C	III-3	X	AB	P	I
Boric Acid Batching	W	G	VIII	X	AB	-	I(L)
Chemical Mixing	W	G	VIII	X	AB	-	I(L)
Resin Fill	W	G	VIII	X	AB	-	I(L)
Demineralizers							
Mixed Bed	W	D	III-3	X	AB	X	I
Cation	W	D	III-3	X	AB	X	I
Steam Generator Blowdown System							
SG Blowdown Isolation Valves	T	B	III-2	X	AB	P	I
SG Blowdown Heat Exchangers	T	-	VIII	-	TB	P	-
Flash Tank	T	-	VIII	-	TB	P	-
Compressed Air System							
Service & Control Air Subsystem							
Compressors	T	H	-	-	TB	-	-
Receiver Tanks	T	H	VIII	-	TB	-	-
Air Dryers	T	H	VIII	-	TB	-	-
Auxiliary Control Air Subsystem							

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Compressors	T	(19)	-	X	AB	-	I
Receiver Tanks	T	(19)	VIII	X	AB	-	I
Air Dryers	T	C	(14)	X	AB	-	I
Ice Condenser							
Ice Baskets	W	C	-	X	C	-	I
Lower Inlet Doors	W	C	-	X	C	-	I
Lattice Frames	W	C	-	X	C	-	I
Lattice Frame Columns	W	C	-	X	C	-	I
Lower Support Structure	W	C	-	X	C	-	I
Intermediate Deck Doors	W	C	-	X	C	-	I
Wall Panels	W	C	-	X	C	-	I
Floor Structures	W,T	C	-	X	C	-	I
Top Deck Doors	W	C	-	X	C	-	I
Air Handling Unit Supports	W	C	-	X	C	-	I
Top Deck Beams	W	C	-	X	C	-	I
Refrigeration System	W	-	-	X	C,AB	-	I(L)
Ice Machine	W	-	-	X	AB	-	I(L)
Ice Condenser Bridge Crane	W	-	-	X	C	-	I(L)
Floor Drain Gate	W	C	-	X	C	-	I
Containment Isolation System Valves	T	B	III-2	X	C,AB	X,P	I
Air Return Fans	T	(11)	AMCA IEEE	X	C	-	I

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Component Cooling System							
Pumps	T	C	III-3	X	AB	P	I
Heat Exchangers (Tube)	T	C	III-3	X	AB	-	I
(Shell)	T	C	III-3	X	AB	P	I
Surge Tank	T	C	III-3	X	AB	P	I
Valve (Containment Isolation)	T	B	III-2	X	C	-	I
Valves	T	C	III-3	X	AB,C	-	I
Valves	T	G	B31.1	X	AB	-	I(L)
Valves	T	H	B31.1	-	CDWEB	-	
Seal Leakage Return Unit	T	L	-	X	AB	-	I(L)
Radioactive Waste Disposal System							
Tanks							
Laundry & Hot Shower	W	G	VIII	X	AB	X	I(L)
Chemical Drain	W	G	VIII	X	AB	X	I(L)
Reactor Coolant Drain	W	G	VIII	X	C	X	I(L)
Tritiated Drain Collector	W	G	VIII	X	AB	X	I(L)
Waste Condensate (See Note 23)	W	H	VIII	X	AB	X	I(L)
Spent Resin Storage	W	D	III-3	X	AB	X	I
Gas Decay	W	D	III-3	X	AB	X	I
Floor Drain Collector	W	G	VIII	X	AB	X	I(L)
CVCS Monitor	W	G	III	X	AB	P	I(L)
Cask Decontamination Collector	T	G	----	X	AB	P	I(L)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Pumps							
Reactor Coolant Drain Tank Pumps	W	G	B31.1	X	C	X	I(L)
Chemical Drain Tank Pump	W	G	B31.1	X	AB	X	I(L)
Laundry & Hot Shower Tank Pump	W	G	B31.1	X	AB	X	I(L)
Tritiated Equipment Drain Sump Pumps	W	G	B31.1	X	AB	X	I(L)
Waste Condensate Pumps (See Note 23)	W	H	B31.1	X	AB	X	I(L)
Tritiated Drain Collector Tank							
Discharge Pump	W	G	B31.1	X	AB	X	I(L)
Floor Drain Collector Tank Discharge Pump	W	G	B31.1	X	AB	X	I(L)
Aux. Condensate Demin Waste Evap Feed Pump (See Note 23)	W	G	B31.1	X	AB	X	I(L)
CVCS Monitor Tank Pump	W	G	B31.1	X	AB	P	I(L)
Cask Decon Collector Tank Pump	T	G	B31.1	X	AB	P	I(L)
Component							
Containment Pit Sump Pumps	W	G	B31.1	X	C	X	I(L)
AB Floor & Equip Drain Sump Pumps	W	G	B31.1	X	AB	P	I(L)
RB Floor & Equip. Drain Sump Pump	T	G	B31.1	X	C	X	I(L)
RB Floor & Equip Drain Pocket Sump Pump	T	G	B31.1	X	C	X	I(L)
AEB Floor & Equip. Drain Sump Pump	T	G	B31.1	X	AEB	P	I(L)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Miscellaneous							
Waste Gas Compressor Pkg.	W	D	III-3	X	AB	X	I
Waste Gas Filter	T	G	VIII	X	AB	X	I(L)
Nitrogen Manifold	T	G	-	X	AB	-	I(L)
Hydraulic Compactor	W	-	-	-	SB	P	-
Laundry Tank Basket Strainer	W	G	VIII	X	AB	P	I(L)
Cond Demin Waste Evaporator (See Note 23)	T	H	III-3	X	CDWEB	P	-
Fire Protection System							
Valves - Flood Mode & CI	T	C	III-3	X	C,AB,O,B,P	-	I
Valves - Balance of System	T	G	B31.1, UL/FM	X	C,AB,O,B,P,CB DB,SB,CDWEB	-	I(L)
Fire Pumps (vertical turbine)	T	C	III-3	X	O	-	I
Station Ventilation System							
Containment Ventilation							
Containment Purge							
Fans (excluding Inst. Rm. Fan)	T	(11)	AMCA	X	AB	-	I(L)
Filters	T	(11)	-	X	AB	X	I
Dampers	T	(11)	-	X	AB	X	I
Ductwork	T	(11)	SMACNA	X	C,AB	X	I/I(L) (See Note 20)

TABLE 3.2-2 (Sheet 10 of 23)

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Upper Compartment, CRDM & Instrument Room Cooling							
Fan	T	-	AMCA	X	C	P	I(L)
Supply Air Filters	T	(11)	-	X	AB	-	I(L)
Upper Containment Coolers							
Coil Units (Unit 1)	T	-	-	X	C	P	I (See Note 12)
Coil Units (Unit 2)	T	-	-	X	C	P	I(L)A (See Note 12)
CRDM Coolers							
Coil Units (Unit 1)	T	-	-	X	C	P	I(L)A (See Note 12)
Coil Units (Unit 2)	T	-	-	X	C	P	I(L)B (See Note 12)
Instrument Room A/C Water Chiller							
Coil Units (Unit 1)	T	(11)	-	X	AB	P	I(L)A (See Note 12)
Coil Units (Unit 2)	T	-	-	X	AB	P	I(L)B (See Note 12)

TABLE 3.2-2 (Sheet 11 of 23)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Lower Compartment Cooling Units							
Fans	T	(11)	AMCA	X	C	-	I
Motors	T	(11)	IEEE	X	C		I
Coils	T	C	III-3	X	C	P	I (See Note 12)
Backdraft Dampers	T	(11)	ASME	X	C	-	I
Instrumentation/Controls	T	(11)	IEEE	X	C/CB	-	I
Ductwork/Accessories	T	(11)	SMACNA	X	C	P	I
Auxiliary Bldg. Ventilation							
Fan/Coil Units	T	(11)	AMCA	X	AB	-	I(L)
Filters	T	(11)	-	X	AB	-	I(L)
ESF Room Coolers	T	(11)	-	X	AB	P	I
Auxiliary Board Rooms Air-conditioning System	T	(11)	AMCA,ARI	X	AB	-	I
Shutdown Board Rooms Air-conditioning System	T	(11)	AMCA,ARI	X	AB	-	I
Other Air Conditioning Systems	T	(11)	AMCA,ARI	X	AB	-	I(L)
Control Bldg. Ventilation							
Fan	T	(11)	-	X	CB	-	I
Filters	T	(11)	-	X	CB	X	I
Air Conditioning Unit (MCR)	T	(11)	AMCA,ARI	X	CB	-	I
Air Conditioning Unit (Elec. Bd. Rm.)	T	(11)	AMCA,ARI	X	CB	-	I
RB Inst Rm Air Conditioning System	T	(11)	AMCA,ARI	X	RB	-	I(L) (See Note 21)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Diesel Bldg. Ventilation							
Exhaust System	T	(11)	AMCA	X	DB	-	I
Battery Hood Exhaust System	T	(11)	AMCA	X	DB	-	I
Elec Board Room Exhaust System Fans	T	(11)	AMCA	X	DB	-	I
Main Steam System							
Relief Valves	T	B	III-2	X	AB	-	I
Safety Valves	T	B	III-2	X	AB	-	I
MSIVs	T	B	III-2	X	AB	-	I
Isolation Bypass Valves	T	B	III-2	X	AB	-	I
Feedwater System							
MFIVs	T	B	III-2	X	AB	-	I
Auxiliary Feedwater System							
Auxiliary Feedwater Pumps							
Motor Driven	T	C	III-3	X	AB	-	I
Steam Turbine Drive	T	C	III-3	X	AB	-	I
Steam Dump Systems							
Turbine Bypass	W	-	-	-	TB	-	-

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Spent Fuel Pit							
Spent Fuel Pit Heat Exch.							
(Tube)	W	C	III-3	X	AB	X	I
(Shell)	W	C	III-3	X	AB	X	I
Spent Fuel Pit Pump	W	C	III-3	X	AB	X	I
Spent Fuel Pit Filter	W	G	VIII	X	AB	X	I(L)
Spent Fuel Pit Demineralizer	W	G	VIII	X	AB	X	I(L)
Spent Fuel Pit Strainer	W	G	-	X	AB	X	I(L)
Spent Fuel Pit Skimmer Pump	W	G	-	X	AB	X	I(L)
Spent Fuel Pit Skimmer							
Strainer Assembly	W	G	-	X	AB	X	I(L)
Spent Fuel Pit Skimmer Filter	W	G	VIII	X	AB	X	I(L)
Purification Pumps	T	G	ANSI B31.1	X	AB	X	I(L)
Purification Filter	T	G	ANSI B31.1	X	AB	X	I(L)
Fuel Handling System							
Refueling Machine	W	-	-	X	C	-	I(L)
Reactor Vessel Head Lifting Device	W	-	-	X	C	-	-
Reactor Internals Lifting Device	W	-	-	X	C	-	-

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Spent Fuel Pit Bridge & Hoist	W	-	-	X	AB	-	I(L)
Rod Cluster Cont. Handling Fixture	W	-	-	X	-	-	-
Reactor Vessel Stud Tensioner	W	-	-	X	-	-	-
Spent Fuel Handling Tool	W	-	-	X	-	-	-
Fuel Transfer System							
Fuel Transfer Tube & Flange	W	B	-	X	C,AB	P	I
Conveyor System & Controls	W	-	-	X	C,AB	P	-
New Fuel Storage Racks	W	-	-	X	AB	-	I
Spent Fuel Storage Racks	T	-	-	X	AB	X	I
Emergency Diesel Fuel Oil System							
To 7 Day Tanks							
Transfer Pumps	T	G	B31.1	X	DB	-	I(L)
Fuel Oil Tanks (7 Day)	T	I	VIII	X	DB	-	I
Raw Cooling Water System							
Pumps	T	-	-	-	O	-	-
Strainers	T	-	-	-	TB	-	-
Sampling System							
Sample Heat Exchanger	T	-	VIII	-	AB	X	-
Sample Vessel	T	-	VIII	-	AB	X	-
Delay Coil	T	B	III-2	X	C	X	I

TABLE 3.2-2 (Sheet 15 of 23)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Equipment Decontamination System							
Pump	W	-	-	-	AB	-	-
Tank	T	-	D100	-	AB	-	-
Filters							
Reactor Coolant	W	B	III-2	X	AB	X	I
Seal Water Return	W	B	III-2	X	AB	X	I
Seal Water Injection	W	B	III-2	X	AB	X	I
Boric Acid	W	C	III-3	X	AB	-	I
Miscellaneous							
Letdown Orifices	W	B	III-2	X	C	X	I
Boric Acid Blender	W	C	III-3	X	AB	-	I
Boron Recovery System (See Note 24)							
Pumps							
Holdup Tank Recirc.	W	D	P&V-III	X	AB	X	I
Gas Stripper Feed	W	D	P&V-III	X	AB	X	I
Monitor Tank	W	G	VIII	X	AB	P	I(L)
Tanks							
Holdup	T	D	III-3	X	AB	X	I
Monitor	T	G	VIII	X	AB	P	I(L)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Demineralizers							
Evaporator Feed Ion Exch	W	G	VIII	X	AB	X	I(L)
Evaporator Condensate	W	G	VIII	X	AB	X	I(L)
Filters							
Evaporator Feed Ion Exch	W	G	VIII	X	AB	X	I(L)
Evaporator Condensate	W	G	VIII	X	AB	X	I(L)
Concentrates	W	G	VIII	X	AB	X	I(L)
Emergency Gas Treatment System							
Fans	T	(11)	AMCA	X	AB	P	I
Filters	T	(11)	-	X	AB	P	I
Moisture Separator	T	(11)	-	X	AB	P	I
Dampers	T	(11)	-	X	AB,C	P	I
Ducting	T	(11)	-	X	AB,C	P	I
Auxiliary Bldg. Gas Treatment System							
Fans	T	(11)	AMCA	X	AB	P	I
Filters	T	(11)	-	X	AB	P	I
Moisture Separator	T	(11)	-	X	AB	P	I
Dampers	T	(11)	-	X	AB,C	P	I
Ducting	T	(11)	-	X	AB,C	P	I

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Essential Raw Cooling Water System Pumps	T	C	III-3	X	P	-	I
ERCW Pump Motor Bearing							
Cooling Coils	T	(18)	-	X	P	-	I
Containment Isolation Valves	T	B	III-2	X	C	-	I
Valves	T	C	III-3	X	AB	-	I
Valves (yard)	T	C	III-3	X	B	-	I
Valves (Station Air Compressor)	T	H	B31.1	-	TB	-	-
Valves (Auxiliary Air Compressor)	T	C	III-3	X	AB	-	I
Valves (Screen Wash Supply)	T	(15)	B31.1	X	P	-	I(L)(15)
Screen Wash Pumps	T	(15)	B58.1	X	P	-	I(15)
Automatic Backwashing Strainers	T	C	III-3	X	P	-	I
Valves (Discharge Header Air Release & Piping)	T	K	C512/B31.1	X	AB	-	I(L)
Flood Mode Boration Makeup System							
Aux. Boration Makeup Tank	T	C	III-3	X	AB	X	I
Aux. Charging Booster Pumps	T	H	B31.1	X	AB	X	I(L)
Flood Mode Boration Demineralizer	T	H	B31.1	X	AB	X	I(L)
Flood Mode Boration Filters	T	H	B31.1	X	AB	X	I(L)
Aux. Charging Pump	T	H	B31.1	X	AB	X	I(L)
Valves	T	H	B31.1	X	AB	X	I(L)
Valves	T	B	III-2	X	AB	X	I

TABLE 3.2-2 (Sheet 18 of 23)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Scope (1)	TVA/ANS Safety Class (2)	Code (3)	QA Required (4)	Location (5)	Rad Source (6)	Seismic (7)
Valves	T	C	III-3	X	AB	X	I
Flood Mode Boration Filters	T	H	B31.1	X	AB	X	I(L)
Aux. Charging Pump	T	H	B31.1	X	AB	X	I(L)
Valves	T	H	B31.1	X	AB	X	I(L)
Valves	T	B	III-2	X	AB	X	I
Valves	T	C	III-3	X	AB	X	I

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Notes:

(1) T = Tennessee Valley Authority

W = Westinghouse

(2) A = TVA Safety Class A

B = TVA Safety Class B

C = TVA Safety Class C

D = TVA Safety Class D

G = TVA Safety Class G

H = TVA Safety Class H

1, 2a, 2b, or 3 = ANS N18.2 Safety Class

I = Seismic Category I, part of structure

K = TVA Safety Class K

P = TVA Safety Class P

(3) The code class listed for an item is the minimum required. An item may have been obtained to a higher code level than that listed.

III = ASME Boiler and Pressure Vessel Code - Section III

III-1 = ASME Boiler and Pressure Vessel Code - Section III, Class 1

III-2 = ASME Boiler and Pressure Vessel Code - Section III, Class 2

III-3 = ASME Boiler and Pressure Vessel Code - Section III, Class 3

IIIa9 = ASME Boiler and Pressure Vessel Code - Section III, Article 9 "Protection Against Overpressure"

VIII = ASME Boiler and Pressure Vessel Code - Section VIII

P&V-I = ASME Code for Pumps and Valves for Nuclear Power, Class I

P&V-II = ASME Code for Pumps and Valves for Nuclear Power, Class II

P&V-III = ASME Code for Pumps and Valves for Nuclear Power, Class III

D100 = American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs, and Elevated Tanks for Water Storage, AWWA, D100

B31.1 = ANSI B31.1 1973 Edition through summer 1973 Addenda

ACI = American Concrete Institute

AMCA = Air Moving and Conditioning Association

ARI = Air Conditioning and Refrigeration Institute

HIS = Hydraulic Institute Standards

IEEE = Institute of Electrical and Electronics Engineers

NFPA = National Fire Protection Association

B58.1 = ANSI B58.1 Vertical Turbine Pumps

B73.1 = ANSI B73.1M Horizontal end Suction Centrifugal Pumps

UL/FM = Underwriters Laboratory or Factory Mutual

SMACNA = Sheet Metal and Air Conditioning Contractors National Association

C512 = American Waterworks Association, Air-Release, Air/Vacuum, and Combination Air Valves for Waterworks Services ANSI/AWWA, C512.

(4) Quality assurance required:

X = Yes, - = No

(5) C=Containment

AB =Auxiliary Building

AEB = Additional Equipment Building

CB = Control Building

DB = Diesel Generator Building

SB = Service Building

CDWEB = Condensate Demineralizer

TABLE 3.2-2 (Sheet 21 of 23)

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

Waste Evaporator Building

O = Outdoors above ground

B = Buried in ground

P = ERCW Pumping Station

TB = Turbine Building

(6) X = Source of radiation

- = No source of radiation

P = Possible source of radiation

(7) I = Seismically qualified

I(L) = Limited seismic qualification

I(L)A = Limited seismic qualification - pressure boundary integrity and position retention

I(L)B = Limited seismic qualification - position retention

- = Not seismically qualified

(8) AMCA Class III and performance tested in accordance with AMCA Standard air moving devices.

(9) Performance test required.

(10) Deleted by FSAR Amendment 79

(11) Those components of the heating, ventilating, and air conditioning system (HVAC), which are not covered directly by the TVA piping classifications of Subsection 3.2.2, have been designed and constructed to standards and specifications which are equivalent to ANS Safety Class 2b. Coils in ESF Pump Room/Area Coolers located in the Auxiliary Building have been replaced with ASME Section III, Class 3 coils that are qualified to Seismic Category I and TVA Piping Class C (See 47W845 drawing series where the C/C* designation has been removed).

(12) The following HVAC cooling coils perform no safety related cooling function but they are seismically designed or qualified as indicated to provide ERCW pressure boundary integrity to ensure the ERCW can perform its primary safety functions:

- Unit 1 and 2 Lower Compartment Cooler cooling coils — Seismic Class I
- Unit 1 CRDM Cooler cooling coils — Seismic Class I(L)A
- Unit 1 Incore Instrument Room Chiller coils — Seismic Class I(L)A

TABLE 3.2-2 (Sheet 22 of 23)

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

▪ Unit 1 and 2 RCP Motor Air Coolers	Seismic Class I(L)A
▪ Unit 2 Upper Containment Cooler coils	Seismic Class I(L)A
▪ Unit 1 Upper Compartment Cooler cooling coils	Seismic Class I
▪ Unit 2 Upper Compartment Cooler cooling coils	Seismic Class I(L)A

The Unit 2 CRDM Cooler coils and Unit 2 Incore Instrument Room Chiller coils are not required for plant safety and are designed to Seismic Class I(L)B. The differences in Seismic Classification for the HVAC equipment supplied by ERCW are considered in the analyses that demonstrate the acceptable capability of the ERCW system.

(13) Vessel was built to the requirements of ASME code but does not have code stamp.

(14) Acceptable for use within Regulatory Guide 1.26 Quality Group C system (ASME Section III, Class 3.) For the auxiliary air system, see also Note 1 of Table 3.2-2a.

(15) Although the screen wash pumps, piping and valves are required for plant safety, they were not purchased to TVA Class C standards. The pumps are seismically qualified, have limited QA, and were the best commercially available product for the service. The piping and valves are designed to TVA Class G and Seismic Category I(L) for pressure boundary integrity. For this application, this level of qualification meets the intent of TVA Class C. Criteria requires that any future modifications or repair to the ERCW screen wash pumps, piping or valves are made to the requirements of TVA Class C.

(16) This component is actually a system containing many components. Those parts of the system that contain component cooling water are Safety Class C with design code of ASME III, Class 3. The remainder of the system is Safety Class G with design codes as identified in Table 3.2-5.

(17) The secondary chamber of the steam generators (shell side) are built to the ASME B&PV Code Section III, Division 1, Class 1, and applicable code interpretations and/or rulings. Although the shell side of the steam generator functions only dictate a TVA Class B, they were procured to comply with ASME Section III, Class 1. Therefore, repairs, modifications and/or additions shall be in accordance with the original contract specifications and drawing requirements.

(18) The ERCW pump motor bearing cooling coils, required for plant safety, were not purchased to TVA Class C standards. The vendor-supplied cooling coils have been seismically qualified and are considered safety-related and suitable for the intended service. For their application, the level of qualification meets the intent of TVA Class C at the motor interface.

(19) Although not purchased and stamped in accordance with ASME Section III Code Requirements, this equipment meets the highest available commercial quality standards.

SUMMARY OF CRITERIA - MECIANICAL SYSTEM COMPONENTS

(20) All purge air ductwork (supply and exhaust) inside the annulus and exhaust air ductwork from the Shield Building isolation valves 2-FCV-30-61 and -62 to 2–FCV-30-213 and -216 is Seismic Category I. Supply air ductwork from the ABSCE isolation valves 2-FCV-30-294 and -295 to the Shield Building isolation valves 2-FCV-30-2 and -5 and all purge air ductwork (supply and exhaust) inside primary containment up to the inboard containment isolation valves is Seismic Category I(L).

(21) All piping between the containment isolation valves is Seismic Category I. The piping up to the containment isolation valves on each side is Seismic Category I(L).

(22) This tank was procured to ASME III - 3 requirements.

(23) Not used for Unit 2 operation.

(24) The boron recycle (recovery) system is not required for the operation of Unit 2. See FSAR Section 9.3.7. The portions of this system which are used for the operation of Unit 2 are discussed in FSAR Section 9.3.4.

(25) The Primary Water Storage Tank (PWST) meets the ASME Section III, Class 3, design by analysis requirements with Seismic Class I(L) Forcing Functions for atmosphere tanks.

(26) The Unit 2 PWST was initially procured to ASME Section III-3 requirements and was subsequently downgraded to API-650. However, the Unit 2 PWST bottom plate and associated nozzles are Seismic Category I and were procured to the requirements of ANSI N45.2 (Safety Related). The remainder of the Unit 2 PWST is classified as Seismic Category IL(B).

TABLE 3.2-2a (Sheet 1 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Auxiliary Control Air	Portions of the System necessary for containment isolation. (See Note 5)	2a	B	I
	Balance of system. (See Note 1 and Note 5). System boundary is considered to exist to the upstream side of the filters which tie the non-essential control air systems to the auxiliary control air lines.	2b	C	I
Boron Recycle (See Note 12)	Equipment used to provide a ready supply of concentrated boric acid (boric acid tanks, boric acid transfer pumps, boric acid filters, and associated pipes and valves).	2b	C (See Note 11)	I (See Note 11)
	Processing and Waste Holdup Equipment whose failure could result in a site boundary dose of 0.5 rem or more. (See Note 8) (gas stripper feed pumps, holdup tanks and holdup tank recirculation pumps, and associated piping and valves).	3	D	I
	Other equipment which carries minimal or no radioactive wastes and/or has no safety function to perform (monitor tank and pumps, evaporator condensate demineralizers, batching tank, gas stripper and boric acid evaporator packages, evaporator feed ion exchangers, condensate filters, concentrate filters, evaporator feed ion exchange filters, and associated pipes and valving).	-	G	I(L)

TABLE 3.2-2a (Sheet 2 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Chemical and Volume Control	Equipment that circulates reactor coolant normally or during an accident (charging, letdown, excess letdown, seal water return lines; centrifugal charging pumps; volume control tank; and, miscellaneous associated lines and valves). (See Note 8)	2a	B	I
	Equipment necessary for boric acid addition (boric acid tanks, boric acid blender, lines and valves).	2b	C (See Note 11)	I (See Note 11)
	Equipment associated with radwaste cleanup whose failure could result in a 0.5 rem offsite dose (Mixed bed and cation demineralizers, associated piping and valves).	3	D	I
	Balance of equipment (resin fill and chemical mixing tanks, piping and valves).	-	G	I(L)
Component Cooling	Portions of the system necessary for containment isolation.	2a	B	I
	Major pressure boundary components.	2b	C	I
	Equipment inside the CDWE Building. (See Note 13)	-	H	-
	Sample heat exchangers, drains, and vents.	-	G or L	I(L)
Containment Spray	Major pressure boundary components.	2a	B	I

TABLE 3.2-2a (Sheet 3 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Essential Raw Cooling Water	Portions of the system piping necessary for containment isolation.	2a	B	I
	Portions of the system piping required for plant safety. (See Note 4, 10).	2b	C	I
	Portions of the system piping not required for plant safety, but in Seismic Category I structures.	-	G	I(L)
	Portions of system discharge headers located between the two Hydraulic Gradients and the associated Valve Boxes 1 and 2.	-	H	I(L)
	Portions of the system piping not required for safety and not in Seismic Category I structures.	-	H	-
	Auxiliary control air compressors (see Note 1).	-	C	I
	HVAC equipment required for plant safety (see Note 2).	-	-	I
	HVAC equipment not required for plant safety	-	-	See Note 3
	Portions of the strainer backwash/backflush piping (see Note 6).	-	G	I(L)
Feedwater	Downstream of and including the anchors in the valve room exterior walls.	2a	B	I
	Flow transmitter sensing lines	-	P	-
	Upstream of the anchors	-	H	-
Feedwater, Auxiliary	Downstream of and including the first anchor which is immediately up- stream of the check valve closest to and outside of containment.	2a	B	I
	Portions of the system not in SC-2a but required after a seismic event.	2b	C	I
	Condensate supply and other piping not required after a seismic	-	G	I(L)

TABLE 3.2-2a (Sheet 4 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Fire Protection High Pressure (HPFP)	Piping necessary to provide water to AFW system in the event of a flood above plant grade. Equipment necessary to provide makeup to the primary and spent fuel cooling systems in the event of a flood above plant grade.	2b	C	I
	Balance of equipment within Seismic Category I structures.	-	G	I(L)
	Remainder	-	H	-
Flood Mode Boration and Makeup (Auxiliary Charging)	Portion of the system necessary for containment isolation.	2a	B	I
	Piping essential for makeup and boration in the event of a flood above plant grade.	2b	C	I
	Balance of system.	-	H	I(L)
Fuel Oil	Equipment necessary to assure continuous, full power operation of the emergency diesel-generator sets for seven days following a loss of offsite power.	2b	C	I
	Balance of equipment in Seismic Category I structures which performs no safety function but needs to maintain pressure boundary.	-	G or H	I(L)
	Remainder	-	H	-

TABLE 3.2-2a (Sheet 5 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Fuel Pool Cleaning and Cooling	Portions of the system required for containment isolation.	2a	B	I
	Portions of the system required to cool the spent fuel (heat exchangers, pumps, associated piping and valves).	2b	C	I
	Makeup water loop from the RWST through RWP pumps to isolation valve downstream of pumps SFP skimmer piping.	-	G (See Note 7)	I(L)
	Line from isolation valve downstream of RWP pumps to the SFPC loop	2b	C	I
	Balance of system.	-	G	I(L)
Heating, Ventilation, and Air Conditioning (HVAC)	System containment isolation valves and piping between valves.	2a	B	I
	HVAC components, ductwork, and piping located in the Reactor, Auxiliary, Control, and Diesel Generator Buildings that perform safety related air cooling and heating operations or essential air filtration and purification processes or that supply life supporting air (see Note 2).	2b	M,Q, or S	I
	Balance of system (see Note 3)	-	M,Q,S,U,V	I(L)
Hydrogen Analyzer	Portion of H_2 Analyzer system which supplies pure O_2 as reagent gas to H_2 Analyzer Panels and Vacuum Trap Assemblies located on sample tubing low points for H_2 Analyzer System. (see Note 9). (Unit 1 Only)	-	-	I

TABLE 3.2-2a (Sheet 6 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Ice Condenser	Portions of the system required to function during a DBA.	2b	C	I
	Portions of the system that provide a containment isolation function	2a	B	I
	System refrigeration piping	-	N	I(L)
	System refrigeration piping in Reactor Building.	-	M	I(L)
	Balance of system not classified as refrigeration piping (i.e. drains).	-	G or H	I(L)
Main Steam	Upstream of, and including the flued-head anchors in the valve room exterior walls.	2a	B	I
	Turbine impulse pressure transmitter sensing lines	-	P	-
	Downstream of the flued-head anchors.	-	H	-
Reactor Coolant	Equipment within the reactor coolant system boundary, failure of which could cause a Condition III or IV loss-of-coolant accident. (Components downstream of a 3/8 inch or smaller orifice are excluded).	1	A	I
	Portions of the system protected from reactor coolant pressure by a 3/8 inch or smaller orifice, reactor vent head ventilation system, and branch piping for the pressurizer steam space instrumentation nozzles (0.83 inch inside diameter, or smaller).	2a	B	I
	Portions of the system that provide a containment isolation function.	2a	B	I
	Safety and relief valve discharge piping.	-	G	I(L)
	Equipment that does not provide a safety function (pressurizer relief tank (PRT), primary water supply inside containment, nitrogen supply and vent headers to the PRT).	-	G	I(L)

TABLE 3.2-2a (Sheet 7 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

System	System Subsection	Safety Class ANS, N-18.2	TVA Class	Seismic Category
Residual Heat Removal (RHR)	Major pressure boundary components.	2a	B	I
Safety Injection (SI)	Balance of system that recirculates reactor coolant after an accident or prevents leakage of reactor coolant to points external to the system. (See Note 8).	2a	B	I
	Refueling water storage tank and SIS accumulators.	2a	B	I
	Piping from the SIS accumulators to the accumulator isolation valves, and from the RWST to SC-2a valves in the safety injection, RHR, charging pump, and containment spray pump suction lines.	2a	B	I
	Piping to CVCS holdup tanks.	2b	C	I
	Accumulators N_2 fill line.	-	G	I(L)
Steam Generator Blowdown	Piping and valves from the steam generators to and including the containment isolation valves	2a	B	I
	Piping and valves down stream of the containment isolation valves to Column U	-	G	I(L)
	Piping and valves down stream of Column U.	-	H	-
Waste Disposal	Portions of system that provide a containment isolation function.	2a	B	I
	Equipment whose failure could cause a site boundary dose of 0.5 rem or more (per RG 1.26). (See Note 8).	3	D	I
	Balance of system in Seismic Category I structures for which a component failure may cause damage to safety-related equipment.	-	G	I(L)
	Equipment not within Seismic Category I structures.	-	H	-

TABLE 3.2-2a (Sheet 8 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

Note 1: Although not purchased and stamped in accordance with ASME Code Section III requirements, the auxiliary control air system compressors and dryers meet the highest commercial quality standards.

Note 2: Although not purchased under ASME Section III requirements, the HVAC equipment is required for plant safety and does meet the highest commercial quality standards. Safety Class 2b round flexible and triangular duct board ducting installed as part of the ceiling air delivery system in the main control room above the suspended ceiling is qualified to limited seismic requirements, analyzed to ensure that the ducting will remain in place, the physical configuration will be maintained such that flow will not be impeded, the ducting pressure boundary will not be lost, and the ducting is constructed of standard commercial grade materials.

Note 3: The following equipment performs no safety related cooling function but it is seismically designed or qualified as indicated to provide ERCW pressure boundary integrity to ensure the ERCW can perform its primary safety functions:

- Unit 1 and 2 Lower Compartment Cooler cooling coils — Seismic Class I
- Unit 1 CRDM Cooler cooling coils — Seismic Class I(L)A
- Unit 1 Incore Instrument Room Chiller coils — Seismic Class I(L)A
- Unit 1 and 2 RCP Motor Air Coolers — Seismic Class I(L)A
- Unit 1 Upper Containment Cooler cooling coils — Seismic Class I
- Unit 2 Upper Compartment Cooler cooling coils — Seismic Class I(L)A

The Unit 2 CRDM Cooler coils and Unit 2 Incore Instrument Room Chiller coils are not required for plant safety and are designed to Seismic Class I(L)B. The differences in Seismic Classification for the HVAC equipment supplied by ERCW are considered in the analyses that demonstrate the acceptable capability of the ERCW system.

Note 4: Although the screen wash pumps, piping and valves are required for plant safety, they were not purchased to TVA Class C standards. The pumps are seismically qualified, have limited QA, and were the best commercially available product for the service. The piping and valves are designed to TVA Class G and Seismic Category I(L) for pressure boundary integrity. For this application, this level of qualification meets the intent of TVA Class C. Criteria requires that any future modifications or repair to the ERCW screen wash pumps, piping or valves are made to the requirements of TVA Class C.

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

NRC Bulletin 83-06 "Nonconforming Materials Supplied By Tube-Line Corporation" has been evaluated for this system; Carbon Steel and Stainless Steel Program Plans were developed, presented to the NRC, verbally approved, initiated, completed, and reported to the NRC in the NRC Bulletin 83-06 report. The NRC Bulletin 83-06 and NCR GENMEB 8301 were closed and approved by the NRC Inspection Reports 50-390/84-03 and 50-391/84-03. The fittings that were installed and found to be acceptable are identified in Tables 3.1-6 and 3.1-7 of WB-DC-40-36 (Reference DIM WB-DC-40-36-18). The potential effect of unacceptable indications in radiographs of Tube-Line fittings welded with filler material has also been evaluated. The radiographs were supplied by Tube-Line as required by the material specification (ASME SA-403). Piping stress analysis was reviewed for Condition Adverse to Quality Report WBP890546. The review showed that stresses in the fittings are within ASME Section III allowable stresses even if the worst radiographic indications for each size fitting were to be transposed to the highest-stressed fitting. Authorization to use an alternative to the testing requirements of Section III Subsection ND-2000 of the ASME Code was provided by the NRC through a SER dated September 23, 1991.

Note 5: Portions of the control air system were not pneumatic tested to the correct pressure. NRC Inspection Report Nos. 50-390/90-04 and 50-391/90-04 has approved an alternate acceptance to pneumatic test criterion.

Note 6: The strainer backwash/backflush piping has been upgraded from Class G to Class G Seismic Category I(L) for pressure boundary integrity.

Note 7: Class G piping labeled with PBQA is analyzed for Seismic Category I(L) pressure boundary retention and is within the scope of the hydrostatic QA program (the valve seat terminates the PBQA boundary). All remaining Class G piping is seismically supported for position retention only.

Note 8 Manual block valves exist in the discharge piping of the relief valves which provide overpressure protection for the volume control tank, the boron injection tank, and the waste gas compressors. ASME Code, Section III, Subsection NC/ND, Paragraph 7153 prohibits the placement of a block valve in the discharge of pressure relief devices unless the block valve is installed with positive controls and interlocks and means are provided such that the operation of the controls and interlocks can be verified. The following design and administrative features, evaluated and approved by NRC[2], are an acceptable alternative to the ASME code requirements because they provide reasonable assurance that both stop valves would not be left closed during plant operation. Redundant flowpaths exist in the relief valve discharge piping, the locked-open block valves are installed in a controlled access location, and administrative procedures are in place to assure the locked-open position of the block valves.

Note 9 Although not purchased under ASME Section III requirements, the O2 supply bottles, related manifolds and vacuum trap assemblies are required for post-LOCA conditions and meet the highest commercial quality standards, and are qualified to WBNP Seismic Category I classification. (Unit 1 only)

TABLE 3.2-2a (Sheet 10 of 10)

CLASSIFICATION OF SYSTEMS HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION

Note 10 Some welds in the buried portion of the ERCW System did not receive a Section III hydrostatic test visual examination. They did, however, receive a vacuum box examination after welding and the code required NDE. Additionally, a pressure test was performed and held for 1 hour, then a Section XI VT-2 test was performed using 1 psi/min pressure drop or a 2 gal/min loss for 10 min. This was approved by the NRC in Safety Evaluation Report, Supplement 12.

Note 11 Although not originally purchased to ASME Section III and Seismic Category I requirements, the replacement immersion heater assemblies for boric acid tank A have been non-destructive tested and evaluated to be acceptable for the classification.

Note 12 The boron recycle system is not required for operation of Unit 1 or Unit 2. See UFSAR Section 9.3.7. The portions of this system which are used for the operation of Unit 1 or Unit 2 are discussed in UFSAR Section 9.3.4.

Note 13 Not used for Unit 1 or Unit 2 operation.

TABLE 3.2-2b

CLASSIFICATION OF SYSTEMS NOT HAVING MAJOR DESIGN CONCERNS RELATED TO A PRIMARY SAFETY FUNCTION
(See Note 2 below)

System Subsection	Safety Class ANS. N-18.2	TVA Class	Seismic Category
Portion of system necessary for primary containment isolation.	2a	B	I
Portion of system in Seismic Category I structures and not in a higher safety class (except refrigeration piping).	---	G, H or L (See Note 1)	I(L)
Balance of system (except refrigeration piping).	---	H, J, or L	---

Note 1:In special applications, where the code requirements for Class G are not appropriate, Class K may be used.

Note 2:All Entries Above Apply to the Following Systems:

Auxiliary Boiler	Hydrogen Cooling
Carbon Dioxide Storage, Fire Protection, and Purge	Layup Water Treatment
Condensate	Lube Oil
Condenser Circulating Water	Makeup Water
Condenser Tube Cleaning	Potable Water
Demineralized Water	Primary Water
Extraction Steam	Raw Cooling Water
Feedwater Treatment and Secondary Chemical Feed	Raw Service Water
Gland Seal Insulating Oil	Station
Gland Seal Water	Drainage
Heater Drains and Vents	Turbine Drains and Misc. Piping, Vacuum primer

TABLE 3.2-3 (Sheet 1 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
6.9-kV Auxiliary Power System		
6.9-kV Shutdown Boards	2/4	Yes (2)
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
6.9-kV Shutdown Logic Relay Panels	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
6.9-kV/480V Shutdown Board Transformers	6/12	Yes
(2000 kVA) (Unit 1) 1A1-A, 1A-A, 1A2-A, 1B1-B, 1B-B, 1B2-B (Unit 2) 2A1-A, 2A-A, 2A2-A, 2B1-B, 2B-B, 2B2-B		
6.9-kV/480V Pressurizer Heater Backup Group Transformers (500 kVA)	2/4	Yes (3)
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
480V Auxiliary Power System		
480V Shutdown Boards	4/8	Yes
(Unit 1) 1A1-A, 1A2-A, 1B1-B, 1B2-B (Unit 2) 2A1-A, 2A2-A, 2B1-B, 2B2-B		

TABLE 3.2-3 (Sheet 2 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
480V Reactor MOV Boards	4/8	Yes
(Unit 1) 1A1-A, 1A2-A,		
1B1-B, 1B2-B		
(Unit 2) 2A1-A, 2A2-A,		
2B1-B, 2B2-B		
480V Reactor Vent Boards	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
480V Control and Auxiliary		
Bldg. Vent Boards	4/8	Yes
(Unit 1) 1A1-A, 1B1-B,		
1A2-A, 1B2-B		
(Unit 2) 2A1-A, 2B1-B,		
2A2-A, 2B2-B		
480V Diesel Auxiliary Boards	4/8	Yes
(Unit 1) 1A1-A, 1B1-B, 1A2-A, 1B2-B (Unit 2) 2A1-A, 2B1-B, 2A2-A, 2B2-B		
480V Distribution Panelboards for Pressurizer Heater Backup Groups	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		
480V Transfer Switch for Component Cooling System Pump C-S	-/1	Yes

TABLE 3.2-3 (Sheet 3 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
120V AC Vital Plant Control Power System		
Static Inverter System Components	4/12	Yes
a. Auctioneer unit b. A transformer rectifier power supply c. A single phase static inverter with associated equipment for control, voltage, regulation, filtering, and instrumentation d. Regulated transformer bypass source with static and manual bypass switches (Not applicable to spare inverters)		
(Unit 1) 1-I, 1-II, 1-III, 1-IV		
(Unit 2) 2-I, 2-II, 2-III, 2-IV		
(Spare) 0-I, 0-II, 0-III, 0-IV		
120V AC Vital Instrument Power Boards	4/8	Yes
(Unit 1) 1-I, 1-II, 1-III, 1-IV		
(Unit 2) 2-I, 2-II, 2-III, 2-IV		
125V DC Vital Plant Control Power System		
480V AC Vital Transfer Switches	-/4 Note (4)	Yes
Transfer SW I, II, III, IV		
125V DC Vital Battery Chargers	-/9 Note (4) (5)	Yes
Chgrs I, II, III, IV, V		
Chgr 6-S, 7-S, 8-S, and 9-S		
Transfer Devices for Spare 125V DC		
Vital Battery Chargers	-/8 Note (4)	Yes
DC Transfer Switch 68DC1-S & 68DC2-S DC Transfer Switch 79DC1-S &		

TABLE 3.2-3 (Sheet 4 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
79DC2-S AC Transfer Switch 68AC1-S & 68AC2-S AC Transfer Switch 79AC1-S & 79AC2-S		
480V AC Vital Disconnect Panels		
	-/4 Note (4)	Yes
125V DC Vital Batteries	-/4 Note (4)	
Batteries I & II		
Batteries III & IV		
125V DC Vital Battery Boards	-/4 Note (4)	Yes
I, II,		
III, IV		
Electrical Penetrations		
High Voltage Power Penetrations	4/8	Yes
Nuclear Instrument System Penetrations	4/8	Yes
Control Rod Position Indication Penetrations	1/2	Yes
Low Voltage, Power, Control, and Indication Penetrations	41/82	Yes
Thermocouple Penetrations	3/6	Yes
Onsite Electrical Power Source Components		
Diesel Generator Protective Relay Panels	2/4	Yes
(Unit 1) 1A, 1B		
(Unit 2) 2A, 2B		
Diesel Control Panels	2/4	Yes

TABLE 3.2-3 (Sheet 5 of 5)

ELECTRICAL POWER SYSTEM EQUIPMENT DESIGNED TO OPERATE DURING AND AFTER A "SAFE SHUTDOWN EARTHQUAKE"

Equipment	Number /Number Per Unit / In Plant	Qualified in Conformance (1) with IEEE 344-1971
125V Diesel Generator Batteries and Battery Racks	2/4	Yes
DC Distribution Panels	2/4	Yes
125V DC Dual Battery Charger Assemblies	2/4	Yes
Standby Diesel Generators	2/4	Yes
(Unit 1) 1A-A, 1B-B (Unit 2) 2A-A, 2B-B		

1. Those equipment items procured prior to publication of IEEE 344-1971 were purchased under specifications which TVA believes conform to the intent of that document. Equipment procurement, modification, and evaluation activities after September 1, 1974 applied the IEEE 344-1975 standard for seismic qualification.
2. The 6.9-kV shutdown boards are qualified under Section 3.2.2.4.3 of IEEE 344-1971. The test unit withstood higher accelerations than shown on the frequency response spectrum for resonance at 1 percent damping.
3. The 500-kVA transformers were shown analytically to have lower stress under seismic loading conditions than the 2000-kVA transformers which were tested. The 500-kVA transformers are similar in design and construction to the 2000-kVA transformers.
4. The 125-V DC Vital Control Power System is not unitized therefore, numbers shown are on a per plant basis.
5. The discussion of the number of chargers in the vital battery system in certain UFSAR sections (i.e., 1.2.2.7, “Plant Electrical System,” 8.3.2.2, “Analysis of Vital 125V DC Control Power Supply System”) indicate there are eight chargers. The eight chargers are the normal supplies for the four vital channels (I, II, III and IV). The ninth charger is the charger used to maintain Vital Battery V. Additional information regarding this charger is provided in UFSAR Section 8.3.2.2.

TABLE 3.2-4

SUMMARY OF CODES AND STANDARDS
FOR SAFETY CLASS COMPONENTS OF THE WATTS BAR NUCLEAR PLANT

Code Requirements

Safety Class ANS N-18.2	TVA Class	Seismic Category	Code Classification Piping, Pumps, Valves, and Vessels	Remarks
1	A	I	ASME Code, Sec. III, Class 1	Note 1
2a	B	I	ASME Code, Sec. III, Class 2	Note 1
2b	C	I	ASME Code, Sec. III, Class 3	Notes 1,2,3
3	D	I	ASME Code, III, Class 3	Notes 1 & 3

NOTE: 1) Equipment designated "Vendor-Supplied Safety-Related Equipment Packages" on the drawing meet the following requirements:

a. The vendor-supplied equipment packages (component and piping) contained within TVA piping systems classified as A, B, C, or D which do not meet the requirements of ASME Section III are installed and documented using the rules of ASME Section III, and manufacturer's instruction manuals, as requirements, except that the materials and equipment are not certified to Section III, and N-5 data report is not required. 10 CFR 50 Appendix B applies. In some cases there may be portions of these packages which are not safety-related (i.e., drains and vents past the first normally closed isolation valve) and do not require installation to these requirements.

b. Any substitute material used or repairs performed by construction shall be in accordance with the original contract specification and drawing requirements.

c. TVA Class D components are those whose postulated failure would result in potential offsite doses that exceed 0.5 Rem to the whole body, or its equivalent. Class D components do not perform a primary safety function.

d. Exception is taken to Note 1.a above for the auxiliary systems supplied on the diesel generator skid. The fuel oil, engine cooling water (except the ASME Section III, Class 3 heat exchangers), starting air and lubricating oil systems are designed per ANSI B31.1. They are designed to Seismic Category I and are within the 10CFR50 Appendix B QA program. Criteria requires that any modifications to this piping are performed to meet the intent of ASME Section III Class 3 (TVA Class C).

2) ANSI B31.1 code is an acceptable substitute for the ASME code for installation of piping and valves on Class C instrument lines attached to TVA Class M, Q, and S systems. 10 CFR 50 Appendix B applies.

3) Condition Adverse to Quality Report WBP 900336SCA and NRC Violation 390/90-15-02 identified lack of penetration and/or lack of fusion in ASME Code Section III, Class 3 butt welds made prior to September 26, 1990.

TABLE 3.2-5

NON-NUCLEAR SAFETY CLASSIFICATIONS

TVA Class	Seismic Category	Piping	Code Classification Pumps	Valves	Vessels
G	I(L)	ANSI B31.1	Manufac-turers Standards	ANSI B31.1 B16.5, or MSS-SP-66	ASME Code, Sec. VIII, Div. 1
H	Note 1	ANSI B31.1	*	ANSI B31.1, B16.5, or MSS-SP-66	*
J	N/A	*	*	*	*
K	I(L)	*	*	*	*
L	Note 1	*	*	*	*
P	Note 2	*	*	*	*

* Code used is determined by the design requirements of the equipment.

Note 1: Those portions of TVA Class H and L systems located inside Seismic Category I structures and those portions of TVA Class H yard piping located between the two ERCW Hydraulic Gradients and the associated Valve Boxes 1 and 2 are Seismic Category I(L). The balance of these systems are not designed for seismic loading.

Note 2: This class applies to specific sensing lines which meet ASME Code Section III, Class 3 requirements except that inertia effects need not be used for design of lines in non seismic areas and independent verification by an Authorized Nuclear Inspector is not required for fabrication and installation. 10 CFR 50 Appendix B applies. N-5 data report is not required. Portions of sensing lines in Seismic Category I structures meet Seismic Category I(L) pressure boundary requirements.

Note 3 :The Unit 2 Primary Water Storage Tank (PWST) shall be classified as API-650 in lieu of ASME Code Section VIII, Div. 1. The Unit 2 tank bottom plate and nozzles shall be qualified to Seismic Category I requirements and shall meet the requirements of 10 CFR 50, Appendix B; i.e., the Unit 2 PWST bottom plate and associated nozzles were procured to the requirements of ASTM N45.2 (Safety Related). The connecting tank piping located within the ABSCE shall be qualified to Seismic Category 1L(A) (pressure boundary & position retention) requirements. The remainder of the tank and piping shall be qualified to Seismic Category 1L(B) (position retention) requirements.

TABLE 3.2-6

TVA HEATING, VENTILATION AND AIR CONDITIONING CLASSIFICATIONS

TVA Class	ANS Safety Class	Code Jurisdiction	Seismic Category
M	2b*	ANSI B31.5	I or I(L)
N	None	ANSI B31.5	Note 1
Q	2b*	Round Duct, Steel, Spiral or Longitudinal Seam, Locked Seam or Welded (ASTM A211) and SMACNA High Velocity Duct Construction Standards, 2nd Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3). ANSI/ASME N509 (see Note 2).	I or I(L)
S	2b*	Rect. Duct, Steel, Locked Seam or Welded, SMACNA High Velocity Duct Construction Standards, 2nd Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3). ANSI/ASME N509 (see Note 2).	I or (L) Note 4
U	None	Round Duct, Steel, SMACNA Low Velocity Duct Construction Standards, 4th Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3).	I(L)
V	None	Rect. Duct, Steel, SMACNA Low Velocity Duct Construction Standards, 4th Edition, 1969. Criteria requires that future duct constructions after December 21, 1990, are performed in accordance with Specifications G-95, N3M-914, N3C-942 and Design Criteria WB-DC-40-31.8 (see Note 3).	I(L)

*TVA Class M, Q, and S designations are also used on heating, ventilation, and air-conditioning systems which have no ANS safety class requirements if seismic requirements are invoked.

Note 1: Those portions of TVA Class N systems located inside Seismic Category I structures are Seismic Category I(L). The balance of these systems are not designed for seismic loading.

Note 2: Those portions of TVA Classes Q and S Category I duct which are of welded construction, that are fabricated or repaired after January 12, 1987, meet the welding requirements of ANSI/ASME N509 1976. The workmanship samples are not required to have Penetrant Testing (PT) or Magnetic Testing (MT).

Note 3: *Historical Information* - All duct installations prior to December 21, 1990, (except those discussed in Note 4 below) were evaluated and qualified to meet the requirements of WB-DC-40-36.1.

All duct installations after December 21, 1990, shall be in accordance with specifications G-95, N3M-914, N3C-942, and Design Criteria WB-DC-40-31.8.

Note 4 Safety Class 2b round flexible and triangular duct board ducting installed as part of this ceiling air delivery system in the main control room above the suspended ceiling is qualified to Seismic I(L) requirements, analyzed to ensure that the ducting will remain in place, the physical configuration will be maintained such that flow will not be impeded, the ducting pressure boundary will not be lost, and is constructed of standard commercial grade materials.

TABLE 3.2-7 (SHEET 1 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
I.	CODE CASES			
	A. DESIGN/MATERIAL RELATED			
N/A	N-192	Provides rules for use of flexible metal hose	Regulatory Guide 1.84, Rev 26 imposes the following addition to the requirements of Code Case N-192: The applicant should indicate system application, design and operating pressure/temperature rating of the flexible hose. Data to demonstrate compliance of the flexible hose with NC/ND-3649 particularly NC/ND-3649.4(e), are required to be furnished with the application.	Referenced in data report for some of the flexible metal hose assemblies.
N-224-1	N/A	Provides rules for the use of ASTM A-500 Grade B and ASTM A-501 as integrally welded attachments.		
N/A	N-304	Provides for use of other materials not listed in the appendices.	None	Code Case N-304 was originally approved by the NRC in Regulatory Guide 1.84, Rev. 20 dated November 1982. Subsequent revisions of the Code Case have continued to retain unrestricted approval.
NX-2000	N-188-1	Provides rules for using alloy 625 or 825 tubing that is welded without filler metal.	None	Materials to be used for flexible metal hose assemblies
N/A	N-514	Provides alternate analysis for pressure/temperature curves and low temperature overpressure protection (LTOP) system.	None	Pressure and Temperature Limits Report (PTLR)
B.	FABRICATION/EXAMINATION/TESTING RELATED			
N-32-4	NA	Provides for alternative testing of inaccessible or embedded	Regulatory Guide 1.84, Rev. 26, accepts Code Case N-32-4 based on the following clarification	Used in G-29 Process Specification 3.M.9.1

TABLE 3.2-7 (SHEET 2 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		welded joints in piping.	and interpretation. Code Case N-32-4 does not replace paragraph NC/ND-6129, "Provisions For Embedded Or Inaccessible Welded Joints In Piping," of the Code. The intent of the Code Case is to provide additional testing above code requirements and permit liquid penetrant or magnetic particle testing in place of radiographic testing for Class 3 piping with 3/8 inch nominal wall or less. Paragraph 1 contains an additional requirement to the Code. It was, therefore, acceptable but unnecessary to include in the Code Case. Paragraph 2 is a variation in the volumetric examination technique and was acceptable as written. Paragraph 3 contains an additional requirement and is not a relaxation of the Code. It was, therefore, acceptable but unnecessary to include in the Code Case.	
N-127	NA	Provides for an alternative examination requirement for Class 1 and 2 welds made by an automatic welding process.	None	Used on the 47B333 drawings.
N-237-2	NA	Permits acceptance of open ended Class 2 or 3 piping within Class 2 or 3 vessels or tanks or into the gaseous atmosphere of Class MC vessels without hydrostatic testing	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-240	NA	Permits acceptance of open ended piping without hydrostatic testing.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-241	NA	Safety and safety relief valve piping submerged in a suppression pool inside a Class MC or CC vessel may be exempted from hydrostatic	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		testing.		
N-292	NA	Provides rules for addition of weld metal at weld end prep to meet dimensional and minimum wall considerations.	Regulatory Guide 1.84, Rev. 26 imposes the following requirement in addition to the requirements in Code Case N-292: Class 3 piping that is longitudinally welded and that has a weld efficiency factor of 1.0 as selected from Table ND-3613.4-1 should receive a 100% volumetric examination (RT or UT) of the deposited weld metal in accordance with the requirements of ND-5000.	Used to disposition NCR W-4-P.
N-316	NA	Provides for socket weld sizes <1.09 TNOM.	None	Used to disposition NCRs 3702R R0, W-427-P R0, 3555R R0, 4135R R0, 4345R R1, 4739-0, 2217R, 4114, 5305-0, and 5833 R0.
N/A	N-341	Provides for a five year period of certification of Level III NDE personnel.	None	
C.	QUALITY ASSURANCE RELATED			
N-272	NA	Allows information to be cross referenced or depicted on an attached drawing rather than physically attached to the data report.		
N-282	NA	Provides alternatives when Code data plates are removed.	None	Used in disposition of NCRs 1464R R1 and 3750.
II.	CODE EDITION AND ADDENDA			
A.	DESIGN/MATERIAL RELATED			
Sect II SA-36	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-515	74W74	Bend testing made an optional	None	See Interpretation III-1-83-275.

TABLE 3.2-7 (SHEET 4 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		(supplemental) requirement		Used in G-62.
Sect II SA-516	74W74	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-283	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-479	74S76	Decreased the required elongation to 30% and reduction in area to 40%.	None	Used in G-62.
NCA-1140(e)	80W81	Permits the use of material certified to editions and addenda earlier than the Code of Record providing certain requirements are met.	None	Used in G-62
NE-4430	77W78	Deletes the minimum size of attachment pads	None	Used in disposition of NCR 3723R
ND-4435	74ED	Provides for certain minor permanent attachments to be made of uncertified material	None	Used to disposition NCR 5409 R1
NA	80S81	Deleted the paragraph limiting instrument take off size	None	Used to disposition NCR 2001R. Also involved in change number NC/ND3676-1
NE-3213.10	74ED	Provides for acceptance of locally over-stressed conditions	None	Used to disposition NCR 3250 R1
NC/ND-3612.4A	74W76	Provides for use of intervening isolation valves	None	
NA	80S81	Deletes requirements formerly in paragraph NC/ND-3676 affecting lagging of steam instrument lines	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not been used at WBN.
NC/ND-3643.1	77S77	Adds provisions for making branch connections by means of complete penetration pipe to pipe welds.	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not

TABLE 3.2-7 (SHEET 5 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
				been used at WBN.
NC/ND-2311	74W76	Lists materials exempted from impact testing. This provision exists in Section NB 2311 Summer 1972 Addenda which is referenced in Section NC/ND.	None	Requirements of later Code exempting certain materials from impact test for Class 2 and Class 3 systems.
ASME III, Appendix XVII, 2461.1	80ED	Bolting material allowable stress of non-pressure retention parts (i.e. motor actuator mounting screws).	None	For the analysis of non-pressure retention parts of 1-FCV-72-13 and -34 for Sect. increased valve closure thrust loops. SQN calc. SCG-4M-00786 was reviewed for WBN application. Since the 1971 ASME III NC Codes do not contain stress allowables for non-pressure retention parts, thus ASME III NC and Appendix XVII of 1980 Codes, Sect. 2461.1 rules will be used for the analysis. Ref. DCN-S-33722-A and MEB Calc. EPM-CDM-071092 and -071192.
B.	FABRICATION/EXAMINATION/TESTING RELATED			
NX-4436	80W81	Permits limited welding of attachments to a piping system after it has been hydrostatically tested providing certain conditions are met.	None	Used in G-29 Process Specification 3.M.9.1.
NC-6129	77W78	Provides for alternative testing of inaccessible or embedded welding joints in piping.	None	Used in G-29 Process Specification 3.M.9.1.
NB-6128	77S78	Piping systems which serve as spray systems shall be hydrostatically tested to the rules of this section, except that the	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		test may be performed with the spray nozzle attachment connections plugged. The spray nozzles and any connection beyond the run connections need not be hydrostatically tested.		
NX-6211	80S80	Air pockets in components or systems shall be minimized during the conduct of the hydrostatic test by providing vents at high points, or by flushing the system or by providing calculations to show that the entrapped air is dissolved at the pressure/temperature conditions existing during the test.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
NB-4622.1-1 (TBL)	74ED	PWHT holding time is specified to be: For T less than or equal to 1/2 inch, 30 minutes; For T greater than 1/2 inch but less than or equal to 2 inches (P number 1 & 3), 1 hour per inch; For T greater than 2 inches (P number 1 & 3), 2 hours plus 15 minutes per inch over 2 inches; For T greater than 1/2 inch but less than or equal to 5 inches (P number 4,5,6,7,11,11A), 1 hour per inch; For T greater than 5 inches (P number 4,5,6,7), 5 hours plus 15 minutes per inch over 5 inches. For T greater than 5-inches (P Number 11, 11A) one hour per inch.	None	Used in G-29 Process Specification 2.M.1.1.
NB-4623	74ED	For PWHT above 800°F, the rate	None	Used in G-29 Process

TABLE 3.2-7 (SHEET 7 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.		Specification 2.M.1.1.
NB-4624.3	74ED	For PWHT, the minimum width of controlled band at each side of the weld on the face of the greatest weld width shall be the thickness of the weld or 2 inches whichever is less. (2.M.1.1 uses the minimum from 71S73 and the maximum from 74).	None	Used in G-29 Process specification 2.M.1.1. G-29 uses 2 times the thickness or 2 inches whichever is less. This is more conservative than the Code requires.
NC-4623	74ED	For PWHT above 800°F, the rate of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.	None	Used in G-29 Process Specification 2.M.1.1.
NA	74ED	Deleted requirement of Code of Record Paragraph NB-4623.1 that furnace temperature be less than 600°F when the component is placed in it for PWHT.	None	Used in G-29 Process Specification 2.M.1.1.
NX-4453.1	83S83	Relative to examination of defect removal area (adds the following) this examination is not required where defect removal removes the full thickness of the weld and where the backside of the weld is not accessible for removal of examination materials.	None	Used in G-29 Process Specification 1.M.1.2.

TABLE 3.2-7 (SHEET 8 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4231.2	74ED	Deletes Code of Record requirements for MT/PT examination of temporary attachment removal sites on Class 3 components. Permits temporary attachments but deletes the NDE examinations of removal areas.	None	Used in G-29 Process Specification 1.M.1.2.
NC/ND-6322	80S81	(By reference to NC/ND-6222) If the minimum test pressure of NC/ND-6221(a) or (d) is exceeded by 6% at any location, the upper limit shall be established by analysis using all loadings that may exist during the test.	NC/ND-6221(a) NC/ND-6221(d)	Used in G-29 Process Specification 3.M.12.1
NX-5110	74ED	(By reference to Section V T-732.2(B) which refers to T-733.2) Direct or rectified current shall be used at 700/N to 900/N amperes per inch OD up to 5 inches and 500/N to 700/N amperes per inch OD from 5 to 10 inches and 300/N to 500/N amperes per inch OD over 10 inches where N = # of turns.	T-733.2	Used in G-29 Process Specification 3.M.2.1.
NB-5110	74ED	(By reference to Section V T-733.2.) Direct or rectified current shall be used at 700 to 900 amperes per inch OD up to 5 inches and 500 to 700 amperes per inch OD from 5 to 10 inches and 300 to 500 amperes per inch OD over 10 inches.	None	Used in G-29 Process Specification 3.M.2.1.
NX-4427-1 (FIG)	80S80 80W81	Minimum size for socket weld fitting fillet weld (CX) = 1.09 T	None	Used in G-29 Process Specification 3.M.5.1.

TABLE 3.2-7 (SHEET 9 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
	Errata	nominal.		
NX-5110	74S75	(By Reference to Section V T-732.2) For encircling coils, direct or rectified current at 35,000 ampere-turns divided by the sum of 2 plus the L/D ratio of the test part shall be used for magnetization.	T-732.2(A) T-732.2(B)	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74ED	(By reference to Section V T-731.3) Direct or rectified magnetizing current shall be used at a minimum of 100 and a maximum of 125 amperes per inch of prod spacing for sections greater than or equal to 3/4 inch. For sections less than 3/4 inch, amperage shall be 90-110 ampere per inch of prod spacing.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5112	74ED	All NDE performed under this section shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74W74	(By reference to Section V T-630(B)). The penetrant materials are acceptable if the residue does not exceed 0.005 grams or the total sulfur or halogen content shall not exceed 1% of the residue by weight.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5110	74ED	(By reference to Section V T-662). A groove may be machined across the center of each face approximately 1/16	None	Used in G-29 Process Specification 3.M.1.1.

TABLE 3.2-7 (SHEET 10 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
		inch deep and 3/64 inch wide, or some other means should be provided to permit side by side comparison without interfering cross contamination between sides.		
NX-5110	77W79	(By reference to Section V T-660) Permits qualification to be performed by the entire block at each temperature, photographing the results and comparing the photographs.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.7.1.
Section IX, QW-203	74ED	Allows welding procedure qualification in any position to qualify for all position production welds except as noted.	QW-405.2 which requires qualification in the vertical position with upward progression if fracture toughness (impact testing) is a requirement.	Used in G-29 Process Specification 1.M.1.2.
NC/ND-2578	74ED	Unacceptable surface defects may be removed by grinding or machining provided: The remaining thickness is not reduced below the minimum required; the depression after defect elimination, is blended uniformly.	None	Used in G-29 Process Specification 4.M.5.1.

TABLE 3.2-7 (SHEET 11 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4130	74W76	Radiography is not required for welded repairs in material used in components provided that the welds joining these materials are not required to be radiographed, the extent of the welded repair does not exceed 10 square inches of surface area and the magnetic particle or liquid penetrant examination of the repair is made as required by ND-2539.4.	None	Used in G-29 Process Specification 4.M.5.1.
NB-4131 NC-4130	74W74	The time of examination of the weld repairs to weld edge preparations shall be in accordance with NB-5130. (NC-5130).	Since NB/NC-5130d did not exist in summer 1973, these two paragraphs from winter 1974 must be added.	Used in G-29 Process Specification 4.M.5.1.
NX-5110	74ED	(By reference to Section V T-535.1) The primary reference shall be equalized at 50% full screen height.	None	Used in G-29 Process Specification 3.M.7.1.
NB-4620 NB-4651	74ED	Ferritic alloy steel pipe that has been heated for bending shall receive a heat treatment in accordance with NB-4620. Exemptions are listed in Table NB-4622.3-1	None	Used in G-29 Process Specification 4.M.2.1.
Sect V T-263.3	83W83	Provides for not using shims if the radiographic density requirements can be met.	None	Used in G-29 Process Specification 3.M.3.1.
NX-5112	74ED	All nondestructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.3.1.

TABLE 3.2-7 (SHEET 12 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NC/ND-3641.1(a)	71W73	Revise the definition of certain terms in Equation 3 and 4 to permit other means than pipe wall thickness to provide for mechanical strength.		Used in the following Calculations and NCRs: B26850423089, B26850508012, B26850710009, B26850710010, B26850808014, B26860611003, MEB850228009, WBP850211062, NCR4145RRO, NCR2217R. Also involved in Code changes NC-3641(A)-1-3, NC-3641(A)-1-4, NC-3641(A)-1-5, ND-3641(A)-1-3, ND-3641(A)-1-4, and ND-3641(A)-1-6.
NX-4311.3	80W81	Provides for use of capacitor discharge welding of thermocouples and strain gauges.	None	Used in G-29 Process Specification 1.M.4.3.
NC-4244(e)	74ED	Provides alternative for attachment of internally threaded bosses.	None	Used in Calculation B26850613072.
ND-5321	77W77	(By reference to Appendix VI, VI-1132) Provides minimum size for relevant RT indications.	None	Used to disposition CAQR WBP890600.
NX-5342	77ED	Defines nonrelevant condition as any indication with a major dimension of less than 1/16 of an inch. (MT examination).	None	
NB-4622.7	74ED	Provides for exemptions to otherwise mandatory post weld heat treatment based on P number, size, carbon content, and preheat.	None	Also invoked by NC/ND-4000.
TBL NX-4622.3-1	74S76	Provides for substitution of 200°F preheat for PWHT when carbon >0.30% and/or tensile strength >70 ksi for certain material applications.	None	Used in disposition of NCR 1146R and in the 47B333 drawings.
NX-4622.3-1 (TBL)	74ED	Provides additional exemptions to PWHT under conditions as described in Table.	New Paragraph NX-4622.3 - defines term “nominal thickness” as used in Table NX-4622.3-1.	Used on the 47B333 Drawings.

TABLE 3.2-7 (SHEET 13 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-5320 & NC-5320	77W77	(By reference to Appendix IV) Provides relevant size and spacing clarification for rounded indications for RT.	None	Provides acceptance criteria for rounded indications for RT
NB-5250 and NC-5261	77W78	Exempts welds of non-structural attachments from liquid penetrant or magnetic particle examination. Does not exempt the removal area of non-structural or temporary attachments.	None	WBPER910208
NB/NC/ND 6114.2	80S81	Provides exemption to requirement to re-hydro following repair by welding if repair weld is not required to be radiographed per NB/NC/ND-4453.4.	None	Used in G-29 process specification 3.M.9.1
NB-2510(a)	83S83	Seamless pipe, tube, and fittings 1"NPS and less need not be examined by the rules of this subarticle.	None	Use of examination requirements for ASME Section III, Class 1, 1" nominal pipe and smaller from a later edition. Invoke for 1" NPS and less, only.
Table NC-4622.3-1	74S76	Exempts PWHT in Pl materials less than 1 1/2" thick provided 200 degrees F preheat is used.	None	Used to disposition WBP 900419PER
NX-4427	80W82	Allows fillet welds to be undersize 1/16 inch for 10% of length of weld.	None	Weld Project
NCA 1273	80S80	Exempts orifice plates not exceeding 1/2" nominal thickness which are clamped between flanges and used for flow measuring service only from the code.	None	Orifice Plates
NB-2538(a)4	77ED	Areas ground to remove oxide scale or other mechanically caused depressions...need not be PT or MT examined.	None	Provisions for not NDE examining impressions caused by mechanical means.

TABLE 3.2-7 (SHEET 14 of 15)
UNIT 1
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-4622.9	89 ED	Increases the area which can be repaired by temper bead to 100 in2 and up to 1/3 of base material thickness.	None	Steam generator manway repair or other locations as required.
C. QUALITY ASSURANCE RELATED				
NCA-4000	80W81	Provided more detailed QA program requirements.	None	Used in the NCM
NCA-8240	83S83	Provides alternatives when code data plates are removed.	None	Used in disposition of WBP880052, NCR 5577R0, NCR 5611R0, and NCR 5619R0.
NCA-8240(b)	80S81	Provides alternatives when code data plates are removed.	None	Use in disposition of NCR 3951R.
NX-2610b and c	77ED	Makes provision to exempt manufacturers of small products from certain QA program` requirements.	None	Used in Disposition NCR GENMEB8402, NCR 5146-0, WBP880431, WBP889432, WBP880433, WBP880437, and WBP880438.
NCA 3800	89ED	Metallic material manufacturers and material suppliers Quality System program provides for non certificate holders QA Manual revision and date to be placed on the documentation and also provides documentation requirements for 1" bar stock.	None	QA programmatic requirements for vendors supplying materials and components.

TABLE 3.2-7 (SHEET 1 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
I. CODE CASES				
A. DESIGN/MATERIAL RELATED				
N-192	NA	Provides rules for use of flexible metal hose	Regulatory Guide 1.84, Rev 26 imposes the following addition to the requirements of Code Case N-192: The applicant should indicate system application, design and operating pressure/temperature rating of the flexible hose. Data to demonstrate compliance of the flexible hose with NC/ND-3649 particularly NC/ND-3649.4(e), are required to be furnished with the application.	Referenced in data report for some of the flexible metal hose assemblies.
N-224-1	NA	Provides rules for the use of ASTM A-500 Grade B and ASTM A-501 as integrally welded attachments.		
N-304	NA	Provides for use of other materials not listed in the appendices.	None	Code Case N-304 was originally approved by the NRC in Regulatory Guide 1.84, Rev. 20 dated November 1982. Subsequent revisions of the Code Case have continued to retain unrestricted approval.
N-188-1	NA	Provides rules for using alloy 625 or 825 tubing that is welded without filler metal.	None	Materials to be used for flexible metal hose assemblies
N-514	NA	Provides alternate analysis for pressure/temperature curves and low temperature overpressure protection (LTOP) system.	None	Pressure and Temperature Limits Report (PTLR)

TABLE 3.2-7 (SHEET 2 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
B. FABRICATION/EXAMINATION/TESTING RELATED				
N-32-4	NA	Provides for alternative testing of inaccessible or embedded welded joints in piping.	Regulatory Guide 1.84, Rev. 26, accepts Code Case N-32-4 based on the following clarification and interpretation. Code Case N-32-4 does not replace paragraph NC/ND-6129, "Provisions For Embedded Or Inaccessible Welded Joints In Piping," of the Code. The intent of the Code Case is to provide additional testing above code requirements and permit liquid penetrant or magnetic particle testing in place of radiographic testing for Class 3 piping with 3/8 inch nominal wall or less. Paragraph 1 contains an additional requirement to the Code. It was, therefore, acceptable but unnecessary to include in the Code Case. Paragraph 2 is a variation in the volumetric examination technique and was acceptable as written. Paragraph 3 contains an additional requirement and is not a relaxation of the Code. It was, therefore, acceptable but unnecessary to include in the Code Case.	Used in G-29 Process Specification 3.M.9.1
N-127	NA	Provides for an alternative examination requirement for Class 1 and 2 welds made by an automatic welding process.	None	Used on the 47B333 drawings.
N-237-2	NA	Permits acceptance of open ended Class 2 or 3 piping within Class 2 or 3 vessels or tanks or into the gaseous atmosphere of Class MC vessels without hydrostatic testing	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.

TABLE 3.2-7 (SHEET 3 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
N-240	NA	Permits acceptance of open ended piping without hydrostatic testing.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-241	NA	Safety and safety relief valve piping submerged in a suppression pool inside a Class MC or CC vessel may be exempted from hydrostatic testing.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
N-292	NA	Provides rules for addition of weld metal at weld end prep to meet dimensional and minimum wall considerations.	Regulatory Guide 1.84, Rev. 26 imposes the following requirement in addition to the requirements in Code Case N-292: Class 3 piping that is longitudinally welded and that has a weld efficiency factor of 1.0 as selected from Table ND-3613.4-1 should receive a 100% volumetric examination (RT or UT) of the deposited weld metal in accordance with the requirements of ND-5000.	Used to disposition NCR W-4-P.
N-316	NA	Provides for socket weld sizes <1.09 T_{NOM}.	None	Used to disposition NCRs 3702R R0, W-427-P R0, 3555R R0, 4135R R0, 4345R R1, 4739-0, 2217R, 4114, 5305-0, and 5833 R0.
N-341	NA	Provides for a five year period of certification of Level III NDE personnel.	None	
C.	QUALITY ASSURANCE RELATED			
N-272	NA	Allows information to be cross referenced or depicted on an attached drawing rather than physically attached to the data report.		
N-282	NA	Provides alternatives when Code data plates are removed.	None	Used in disposition of NCRs 1464R R1 and 3750.

TABLE 3.2-7 (SHEET 4 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
N-520-3	NA	Under the provisions Watts Bar Unit 2 will complete documentation and Code Data report and Stamping for those portions of the plant which are suitable for use		Provide documentation to the N-Certificate Holder having overall responsibility to support WBN2 Completion.
II.	CODE EDITION AND ADDENDA			
A.	DESIGN/MATERIAL RELATED			
Sect II SA-36	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-515	74W74	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-516	74W74	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-283	74S76	Bend testing made an optional (supplemental) requirement	None	See Interpretation III-1-83-275. Used in G-62.
Sect II SA-479	74S76	Decreased the required elongation to 30% and reduction in area to 40%.	None	Used in G-62.
NCA-1140(a)	80W81	The use of the provisions of NCA-1140(a) from 1980 Edition with Addenda through Winter 1981 will allow the development of design specifications for various components without requiring these components to be in accordance with latest ASME Code Edition. This will allow for consistency in specifying the Code of Record for these components to be as required by the existing design specifications..	None	Preparation of Certified Design Specifications for Watts Bar Unit 2.

TABLE 3.2-7 (SHEET 5 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NCA-1140(e)	80W81	Permits the use of material certified to editions and addenda earlier than the Code of Record providing certain requirements are met.		Used in G-62
NE-4430	80S81	Deletes the minimum size of attachment pads	None	Used in disposition of NCR 3723R
ND-4435	74ED	Provides for certain minor permanent attachments to be made of uncertified material	None	Used to disposition NCR 5409 R1
NA	80S81	Deleted the paragraph limiting instrument take off size from NC/ND-3676.3	None	Used to disposition NCR 2001R.
NE-3213.10	74ED	Provides for acceptance of locally over-stressed conditions	None	Used to disposition NCR 3250 R1
NC/ND-3612.4A	74W76	Provides for use of intervening isolation valves	None	
NA	80S80	Deletes requirements formerly in paragraph NC/ND-3676 affecting lagging of steam instrument lines	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not been used at WBN.
NC/ND-3643.1	77S78	Adds provisions for making branch connections by means of complete penetration pipe to pipe welds.	None	This request has been generated to cover situations in which it is not possible to state with certainty that some provisions of later Codes have or have not been used at WBN.

TABLE 3.2-7 (SHEET 6 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NC/ND-2311	74S75	Lists materials exempted from impact testing. This provision exists in Section NB 2311 Summer 1972 Addenda which is referenced in Section NC/ND.	None	Requirements of later Code exempting certain materials from impact test for Class 2 and Class 3 systems.
ASME III, Appendix XVII, 2461.1	80ED	Bolting material allowable stress of non-pressure retention parts (i.e. motor actuator mounting screws).	None	For the analysis of non-pressure retention parts of 1-FCV-72-13 and -34 for Sect. increased valve closure thrust loops. SQN calc. SCG-4M-00786 was reviewed for WBN application. Since the 1971 ASME III NC Codes do not contain stress allowables for non-pressure retention parts, thus ASME III NC and Appendix XVII of 1980 Codes, Sect. 2461.1 rules will be used for the analysis. Ref. DCN-S-33722-A and MEB Calc. EPM-CDM-071092 and -071192.
B.		FABRICATION/EXAMINATION/TESTING RELATED		
NB-4121.3	98ED/ 99AD	The use of provisions of NB-4121.3 is desired to support restoration of various Unit 2 components.	None	Completion of Watts Bar Unit 2
NX-4436	80W81	Permits limited welding of attachments to a piping system after it has been hydrostatically tested providing certain conditions are met.	None	Used in G-29 Process Specification 3.M.9.1.
NC-6129	77W78	Provides for alternative testing of inaccessible or embedded welding joints in piping.	None	Used in G-29 Process Specification 3.M.9.1.

TABLE 3.2-7 (SHEET 7 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-6128	77S78	Piping systems which serve as spray systems shall be hydrostatically tested to the rules of this section, except that the test may be performed with the spray nozzle attachment connections plugged. The spray nozzles and any connection beyond the run connections need not be hydrostatically tested.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
NX-6211	80S80	Air pockets in components or systems shall be minimized during the conduct of the hydrostatic test by providing vents at high points, or by flushing the system or by providing calculations to show that the entrapped air is dissolved at the pressure/temperature conditions existing during the test.	None	Used in G-29 Process Specification 3.M.9.1. Also invoked by NC/ND-6000.
NB-4622.1-1 (TBL)	74ED	PWHT holding time is specified to be: For T less than or equal to 1/2 inch, 30 minutes; For T greater than 1/2 inch but less than or equal to 2 inches (P number 1 & 3), 1 hour per inch; For T greater than 2 inches (P number 1 & 3), 2 hours plus 15 minutes per inch over 2 inches; For T greater than 1/2 inch but less than or equal to 5 inches (P number 4,5,6,7,11,11A), 1 hour per inch; For T greater than 5 inches (P number 4,5,6,7), 5 hours plus 15 minutes per inch over 5 inches. For T greater than 5-inches(P Number 11, 11A) one hour per inch.	None	Used in G-29 Process Specification 2.M.1.1.

TABLE 3.2-7 (SHEET 8 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-4623	74ED	For PWHT above 800°F, the rate of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.	None	Used in G-29 Process Specification 2.M.1.1.
NB-4624.3	74ED	For PWHT, the minimum width of controlled band at each side of the weld on the face of the greatest weld width shall be the thickness of the weld or 2 inches whichever is less. (2.M.1.1 uses the minimum from 71S73 and the maximum from 74).	None	Used in G-29 Process specification 2.M.1.1. G-29 uses 2 times the thickness or 2 inches whichever is less. This is more conservative than the Code requires.
NC-4623	74ED	For PWHT above 800°F, the rate of heating & cooling shall not exceed a rate of 400° per hour divided by the maximum thickness in inches of the material being heat treated but in no case more than 400°F per hour.	None	Used in G-29 Process Specification 2.M.1.1.
NA	74ED	Deleted requirement of Code of Record Paragraph NB-4623.1 that furnace temperature be less than 600°F when the component is placed in it for PWHT.	None	Used in G-29 Process Specification 2.M.1.1.
NX-4453.1	83S83	Relative to examination of defect removal area (adds the following) this examination is not required where defect removal removes the full thickness of the weld and where the backside of the weld is not accessible for removal of examination materials.	None	Used in G-29 Process Specification 1.M.1.2.

TABLE 3.2-7 (SHEET 9 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4231.2	74ED	Deletes Code of Record requirements for MT/PT examination of temporary attachment removal sites on Class 3 components. Permits temporary attachments but deletes the NDE examinations of removal areas.	None	Used in G-29 Process Specification 1.M.1.2.
NC/ND-6322	80S81	(By reference to NC/ND-6222) If the minimum test pressure of NC/ND-6221(a) or (d) is exceeded by 6% at any location, the upper limit shall be established by analysis using all loadings that may exist during the test.	NC/ND-6221(a) NC/ND-6221(d)	Used in G-29 Process Specification 3.M.12.1
NX-5110	74ED	(By reference to Section V T-732.2(B) which refers to T-733.2) Direct or rectified current shall be used at 700/N to 900/N amperes per inch OD up to 5 inches and 500/N to 700/N amperes per inch OD from 5 to 10 inches and 300/N to 500/N amperes per inch OD over 10 inches where N = # of turns.	T-733.2	Used in G-29 Process Specification 3.M.2.1.
NB-5110	74ED	(By reference to Section V T-733.2.) Direct or rectified current shall be used at 700 to 900 amperes per inch OD up to 5 inches and 500 to 700 amperes per inch OD from 5 to 10 inches and 300 to 500 amperes per inch OD over 10 inches.	None	Used in G-29 Process Specification 3.M.2.1.
NX-4427-1 (FIG)	80S80 80W81 Errata	Minimum size for socket weld fitting fillet weld (CX) = 1.09 T nominal.	None	Used in G-29 Process Specification 3.M.5.1.

TABLE 3.2-7 (SHEET 10 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NX-5110	74S75	(By Reference to Section V T-732.2) For encircling coils, direct or rectified current at 35,000 ampere-turns divided by the sum of 2 plus the L/D ratio of the test part shall be used for magnetization.	T-732.2(A) T-732.2(B)	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74ED	(By reference to Section V T-731.3) Direct or rectified magnetizing current shall be used at a minimum of 100 and a maximum of 125 amperes per inch of prod spacing for sections greater than or equal to 3/4 inch. For sections less than 3/4 inch, amperage shall be 90-110 ampere per inch of prod spacing.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5112	74ED	All NDE performed under this section shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.2.1.
NX-5110	74W74	(By reference to Section V T-630(B)). The penetrant materials are acceptable if the residue does not exceed 0.005 grams or the total sulfur or halogen content shall not exceed 1% of the residue by weight.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5110	74ED	(By reference to Section V T-662). A groove may be machined across the center of each face approximately 1/16 inch deep and 3/64 inch wide, or some other means should be provided to permit side by side comparison without interfering cross contamination between sides.	None	Used in G-29 Process Specification 3.M.1.1.

TABLE 3.2-7 (SHEET 11 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NX-5110	77W79	(By reference to Section V T-660) Permits qualification to be performed by the entire block at each temperature, photographing the results and comparing the photographs.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.1.1.
NX-5112	74ED	All non-destructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.7.1.
Section IX, QW-203	74ED	Allows welding procedure qualification in any position to qualify for all position production welds except as noted.	QW-405.2 which requires qualification in the vertical position with upward progression if fracture toughness (impact testing) is a requirement.	Used in G-29 Process Specification 1.M.1.2.
NC/ND-2578	74ED	Unacceptable surface defects may be removed by grinding or machining provided: The remaining thickness is not reduced below the minimum required; the depression after defect elimination, is blended uniformly.	None	Used in G-29 Process Specification 4.M.5.1.

TABLE 3.2-7 (SHEET 12 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
ND-4130	74W76	Radiography is not required for welded repairs in material used in components provided that the welds joining these materials are not required to be radiographed, the extent of the welded repair does not exceed 10 square inches of surface area and the magnetic particle or liquid penetrant examination of the repair is made as required by ND-2539.4.	None	Used in G-29 Process Specification 4.M.5.1.
NB-4131 NC-4130	74W74	The time of examination of the weld repairs to weld edge preparations shall be in accordance with NB-5130. (NC-5130).	Since NB/NC-5130d did not exist in summer 1973, these two paragraphs from winter 1974 must be added.	Used in G-29 Process Specification 4.M.5.1.
NX-5110	74ED	(By reference to Section V T-535.1) The primary reference shall be equalized at 50% full screen height.	None	Used in G-29 Process Specification 3.M.7.1.
NB-4620 NB-4651	74ED	Ferritic alloy steel pipe that has been heated for bending shall receive a heat treatment in accordance with NB-4620. Exemptions are listed in Table NB-4622.3-1	None	Used in G-29 Process Specification 4.M.2.1.
Sect V T-263.3	83W83	Provides for not using shims if the radiographic density requirements can be met.	None	Used in G-29 Process Specification 3.M.3.1.
NX-5112	74ED	All nondestructive examinations shall be executed in accordance with detailed written procedures which have been proven by actual demonstration to the satisfaction of the inspector.	None	Used in G-29 Process Specification 3.M.3.1.

TABLE 3.2-7 (SHEET 13 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NC/ND-3641.1(a)	71W73	Revise the definition of certain terms in Equation 3 and 4 to permit other means than pipe wall thickness to provide for mechanical strength.		Used in the following Calculations and NCRs: B26850423089, B26850508012, B26850710009, B26850710010, B26850808014, B26860611003, MEB850228009, WBP850211062, NCR4145RRO, NCR2217R. Also involved in Code changes NC-3641(A)-1-3, NC-3641(A)-1-4, NC-3641(A)-1-5, ND-3641(A)-1-3, ND-3641(A)-1-4, and ND-3641(A)-1-6.
NX-4311.3	80W81	Provides for use of capacitor discharge welding of thermocouples and strain gauges.	None	Used in G-29 Process Specification 1.M.4.3.
NC-4244(e)	74ED	Provides alternative for attachment of internally threaded bosses.	None	Used in Calculation B26850613072.
ND-5321	77W77	(By reference to Appendix VI, VI-1132) Provides minimum size for relevant RT indications.	None	Used to disposition CAQR WBP890600.
NX-5342	77ED	Defines nonrelevant condition as any indication with a major dimension of less than 1/16 of an inch. (MT examination).	None	

TABLE 3.2-7 (SHEET 14 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-4622.7	74ED	Provides for exemptions to otherwise mandatory post weld heat treatment based on P number, size, carbon content, and preheat.	None	Also invoked by NC/ND-4000.
TBL NX-4622.1-1	74S76	Provides for substitution of 200°F preheat for PWHT when carbon >0.30% and/or tensile strength >70 ksi for certain material applications.	None	Used in disposition of NCR 1146R and in the 47B333 drawings.
NX-4622.3-1 (TBL)	74ED	Provides additional exemptions to PWHT under conditions as described in Table.	New Paragraph NX-4622.3 - defines term “nominal thickness” as used in Table NX-4622.3-1.	Used on the 47B333 Drawings.
NB-5320 & NC-5320	77W77	(By reference to Appendix IV) Provides relevant size and spacing clarification for rounded indications for RT.	None	Provides acceptance criteria for rounded indications for RT
NB/NC/ND-4452	01ED / 03AD	The use of NB/NC/ND-4452 allows defects detected by visual or volumetric method and located on an interior surface need only by reexamined by the method which initially detected the defect when the interior surface is inaccessible for surface examination.		Used in disposition of PER 934436
NB-5250 and NC-5261	77W78	Exempts welds of non-structural attachments from liquid penetrant or magnetic particle examination. Does not exempt the removal area of non-structural or temporary attachments.	None	WBPER910208
NB/NC/ND 6114.2	80S81	Provides exemption to requirement to re-hydro following repair by welding if repair weld is not required to be radiographed per NB/NC/ND-4453.4.	None	Used in G-29 process specification 3.M.9.1

TABLE 3.2-7 (SHEET 15 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NB-2510(a)	83S83	Seamless pipe, tube, and fittings 1"NPS and less need not be examined by the rules of this subarticle.	None	Use of examination requirements for ASME Section III, Class 1, 1" nominal pipe and smaller from a later edition. Invoke for 1" NPS and less, only.
Table NC-4622.3-1	74S76	Exempts PWHT in Pl materials less than 1 1/2" thick provided 200 degrees F preheat is used.	None	Used to disposition WBP 900419PER
NX-4427	80W82	Allows fillet welds to be undersize 1/16 inch for 10% of length of weld.	None	Weld Project
NCA 1273	80S80	Exempts orifice plates not exceeding 1/2" nominal thickness which are clamped between flanges and used for flow measuring service only from the code.	None	Orifice Plates
NB-2538.4	77S78	Areas ground to remove oxide scale or other mechanically caused depressions...need not be PT or MT examined.	None	Provisions for not NDE examining impressions caused by mechanical means.
NB-4622.9	89ED	Increases the area which can be repaired by temper bead to 100 in2 and up to 1/3 of base material thickness.	None	Steam generator manway repair or other locations as required.
C.	QUALITY ASSURANCE RELATED			
NCA-4000	80W81	Provided more detailed QA program requirements.	None	Used in the NCM
NCA-8240	83S83	Provides alternatives when code data plates are removed.	None	Used in disposition of WBP880052, NCR 5577R0, NCR 5611R0, and NCR 5619R0.

TABLE 3.2-7 (SHEET 16 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
NCA-8240(b)	80S81	Provides alternatives when code data plates are removed.	None	Use in disposition of NCR 3951R.
NX-2610b and c	77ED	Makes provision to exempt manufacturers of small products from certain QA program` requirements.	None	Used in Disposition NCR GENMEB8402, NCR 5146-0, WBP880431, WBP889432, WBP880433, WBP880437, and WBP880438.
NCA 3800	89ED	Metallic material manufacturers and material suppliers Quality System program provides for non certificate holders QA Manual revision and date to be placed on the documentation and also provides documentation requirements for 1" bar stock.	None	QA programmatic requirements for vendors supplying materials and components.
N-801	NA	Rules for repair of N-stamped Class 1, 2 and 3 components by organization other than the N-certificate holder that originally stamped the component being repaired Section III, Division 1.	None	This code case will allow N-certificate holders at WBN Unit 2 to perform repairs of N-stamped components in support of N-5 code data report completion. These components were installed between 1971 to the present. Many of the older components require refurbishment to meet or exceed design or code requirements. Refurbishment may require replacement of code and non-code material or repair of pressure boundary material. Because the original N-certificate holder that stamped the component may no longer exist or maintain an N-certificate, relief is necessary to perform the required work. Code Case N-801 will be applied at WBN Unit 2 to the repairs accomplished by the N-certificate holders at WBN Unit 2.

TABLE 3.2-7 (SHEET 17 of 17)
UNIT 2
CODE CASES AND PROVISIONS OF LATER CODE EDITIONS AND ADDENDA
USED BY TVA FOR DESIGN AND FABRICATION

Source Document	New Source	Provisions of Later Code	Related Requirements and Regulatory Guide Requirements	Examples When Used
N-802	NA	Rules for repair of stamped components by the N-certificate holder that originally stamped the component Section III, Division 1.	None	This code case will allow N-certificate holders that stamped components to perform repairs of these components in support of N-5 code data report completion. These components were installed between 1971 to the present. The original N-certificate holder may need to refurbish these components to meet or exceed design or code requirements. Refurbishment may require replacement of code and non-code material or repair of pressure boundary material. Code Case N-802 will be applied at WBN Unit 2 to the repairs accomplished by the N-certificate holder that originally stamped the component. This includes stamped components shipped to the N-certificate holder's facility or those repaired on site if the holder's certificate extends to the WBN Unit 2 site.

3.3 Wind and Tornado Loading

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

The Seismic Category I structures are designed for a 95-mile per hour wind, 30 feet above grade, with a 100 year recurrence interval. The wind was determined from Figure 1, ASCE paper 3269, "Wind Forces on Structures."[3] ANSI A58.1-1972, 'Building Code Requirements for Minimum Design Loads in Building and Other Structures'[4] is an acceptable alternative for determining design wind velocities and forces.
The wind was applied for the full height of the structure. A gust factor of 1.1 is included for all wind loads and combinations of loads where wind is involved as recommended in ASCE paper 3269.[3]

3.3.1.2 Determination of Applied Force

The pressure and pressure distribution of wind loads on structures were determined by the methods described in ASCE Paper 3269.[3] The dynamic wind pressure, q, is defined as $q = 0.00256V^2$, where q is in psf and V is in mph. A gust factor of 1.1 is applied which redefines q as $q = 0.00256\ (1.1V)^2 = 0.00310V^2$. The wind pressure, p, in psf, is defined as $p = Cq$, where C is the pressure distribution coefficient (Cpe or Cpi) or the shape coefficient (C_D) determined from Table 4 in ASCE Paper 3269.[3]

For the analysis of box-shaped structures, a shape coefficient (C_D) of 1.3 is used which defines the wind pressure as p = 1.3q. Of the total pressure (p = 1.3q), 0.8q is applied to the windward wall, and 0.5q is applied to the leeward wall. Concurrently the end walls receive 0.7q negative pressure and the roof receives 0.5q uplift.

For the analysis of cylindrical structures, such as the Shield Buildings and storage tanks, the shape coefficients and pressure distribution coefficients are obtained from Table 4(f) of ASCE Paper 3269.[3]

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

All Category I structures except for the additional Diesel Generator Building are designed for a "funnel" of wind moving with a translational velocity of 60 miles per hour and having a rotational velocity of 300 miles per hour. Category I structures are also designed for an external depressurization of 3 psi occurring in 3 seconds.

The tornado loading for the Additional Diesel Generator Building and structures initiated after July 1979 is discussed in Section 2.3.1.

Information about the spectrum and pertinent characteristics of tornado-generated missiles is in Section 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

The pressures and pressure distribution of wind forces on Category I structures due to tornado wind loadings were determined by following the recommendations of ASCE Paper 3269, 'Wind Forces on Structures'.[3] ANSI A.58.1-1972, 'Building Code Requirements for Minimum Design Loads in Building and Other Structures'[4] is used to provide an alternate method to determine tornado wind loads. The provisions for gust factors and variations of wind velocity with height are not applied. The dynamic wind pressure, q, is defined as $q = 0.00256V^2$, where q is in psf and V is in mph. The wind pressure, p, in psf, is defined as $p = Cq$, where C is the shape coefficient (C_D).

A 1.3 shape coefficient is included for box-shaped structures with vertical walls normal to the wind direction. The dynamic pressure load, $p = 1.3q$, due to tornadoes is applied to the structure walls and roof in the same manner as the wind loads in Section 3.3.1.

Cylindrical structures and tanks have the same shape coefficients applied as for wind loads in Section 3.3.1. The pressures are applied over the structures as shown in Table 4(f) of ASCE Paper 3269.[3]

The loadings of the wind force and the depressurization are considered to act concurrently. Coincident wind velocities and pressure drops for the design tornado are shown in Figure 3.3-1. The relationship between wind velocity and pressure in the design tornado shown in Figure 3.3-1 was developed based on Hoecker's studies of the Dallas tornado of 1957.[1,2]

Venting, when used as a design procedure for reducing the tornado-generated differential pressure, is accomplished by using blowoff panels that fail at a lower differential pressure. Upon relief of the differential pressure by the blowoff panel from the exterior wall of a room, the interior walls and slabs of these rooms are designed for the 3 psi pressure differential.

The effective loads on Category I structures due to tornado-generated missiles were determined using the procedures described in Section 3.5.3.

The effect of various combinations of tornado loadings were studied with respect to each Category I structure. The most adverse combination was selected individually for the design basis of each structure.

The tornado loadings are not considered to be coincident with accident or earthquake loadings.

Venting is utilized to reduce the effective tornado-generated differential pressure in portions of the Auxiliary Building. Four hundred square feet of relief panel area are provided in the roof over the spent fuel pool room and cask loading room at Elevation 814.75 for venting purposes during the tornado. The relief panels are held in place by gravity. An upward pressure of 0.25 psi is sufficient to offset the weight of the panels and cause them to be lifted from their nominal positions. Two corners of each panel are chained to the roof to prevent the panel from becoming a missile after it relieves.

The shutdown board room and, in general, the area between columns q and u at Elevation 757.0 is not part of that portion of the Auxiliary Building vented by design; however, the remainder of the building is considered to depressurize due to the vent area provided by the air intake openings and through ventilation penetrations. In addition, the Diesel Generator Building and the Intake Pumping Station are designed to depressurize due to the vent areas provided by the ventilation openings in those buildings.

The roof and exterior walls of the spent fuel pool room and cask loading area were evaluated for the effective tornado-generated pressure differential and were found to be within allowable stress limits. Air velocity induced by venting is expected to be high at the vent opening, but decrease rapidly within a few feet of the opening. No hazard to equipment is foreseen since the vents are located in the Auxiliary Building roof, well away from any essential equipment.

No hazard to equipment in these areas is foreseen due to the small pressure differential and low air velocities. Walls, ceilings, and floors separating areas experiencing depressurization during a tornado from areas not experiencing depressurization are designed to withstand the total tornado-generated pressure differential of 3 psi.

The analytical model employed in determining the effective differential pressures utilizes isentropic, perfect gas relations in a step-wise, steady-state first law analysis. The analysis determined pressure and temperature variations within the structure induced by the design base tornado defined in Section 3.3.2.1.

Pressure differentials and assorted air velocities are expected in all areas which depressurize due to the vented design of the building. In these areas, the partition walls have been checked for the differential pressure from depressurization. In the room(s) where the differential pressure exceeds the wall design, administrative operating instructions ensure that the doors will remain open during a tornado event to reduce the differential pressure to an acceptable value.

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

An investigation of the effect of tornado loading on the Turbine Building was made to determine the extent of failure of the structure as to collapse or to the possibility of generating missiles that could damage Category I structures and impair their ability to perform their intended design function.

The following information was determined:

1. The metal siding panels will fail at loads considerably below the design tornado loading and will become missiles that could affect the Control Building. The siding will fail before the main girts are overloaded enough to cause failure. The failure of the parapet girts is likely, resulting in the release of 6WF15.5 in 4-foot lengths, 8C11.5 in 8-foot lengths, 18-inch x 3/8-inch plate in varying lengths, and 4ST8.5 in 7-foot lengths.

   The roof of the Control Building was investigated for the above missiles and found to be adequately designed to resist the missiles.

2. Following the failure of the siding, the structural steel framing of the building will be exposed to tornado forces acting upon the steel structure, equipment, piping, and other items of wind resistance. The resistance of the structure at this point will be sufficient to prevent collapse onto the Control Building.

3. The turbine room cranes, if not anchored, could possibly be blown from the crane girders, either falling on the operating floor or out the end of the building onto the Control Building roof.

   To preclude the occurrence of this event, the cranes will be anchored to stops at one end of the runway during tornado alerts, watches, and tornadoes.

4. The potable water tanks and gland seal water tanks at Elevation 796.0 floor could be blown to the Control Building roof along with air intake hoods, auxiliary boiler stack, and heating and vent equipment on the Elevation 796.0 floor.

   The Control Building roof was determined to be adequately designed to resist the described events.

The methods used to convert the tornado wind loadings into loads acting on the structures, as well as their distributions across the structures, were determined by following the recommendations of ASCE Paper 3269.[3]

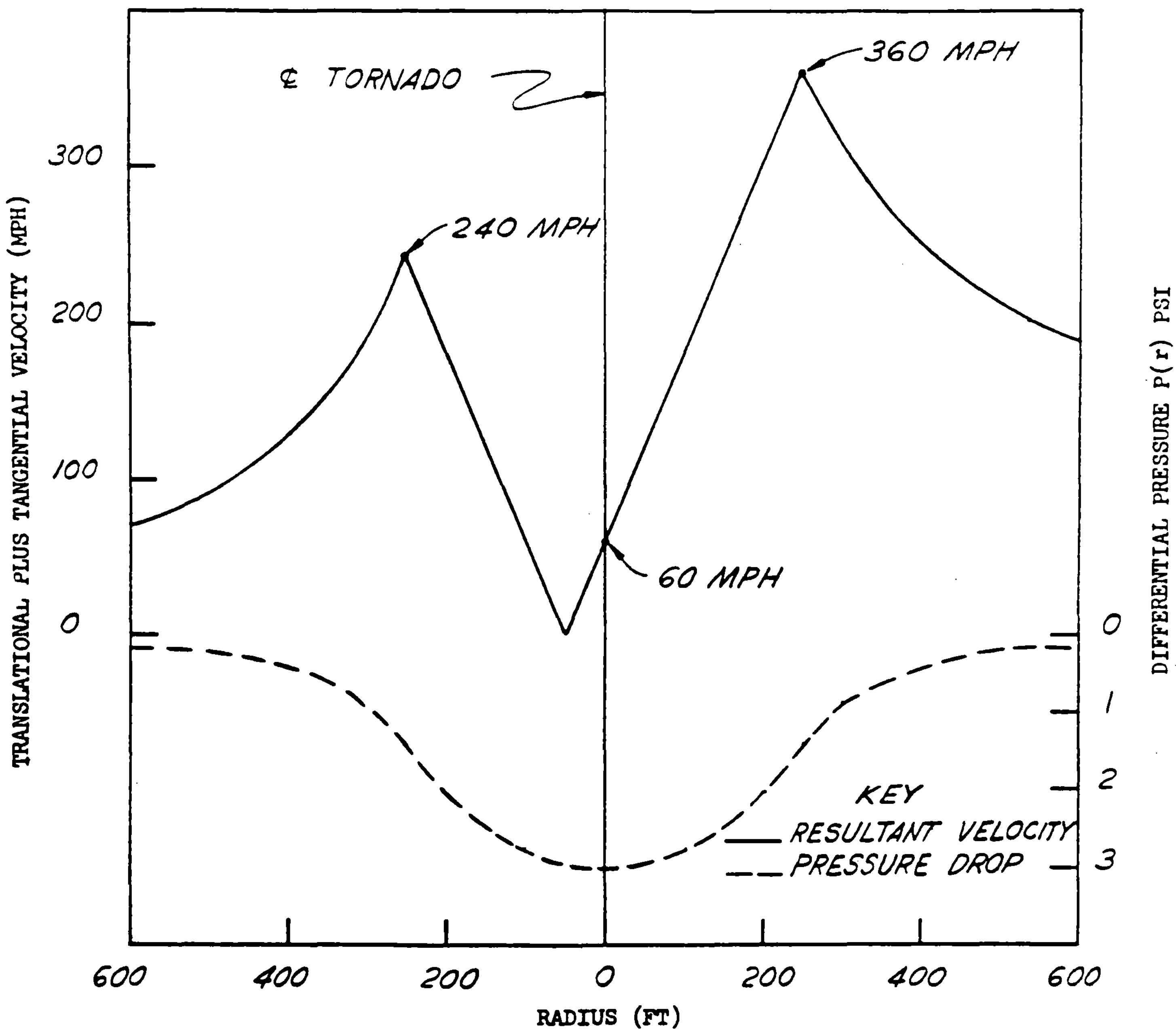
REFERENCES

1. Hoecker, W. H., 'Wind Speed and Air Flow Patterns in the Dallas Tornado and Some Resultant Implications', Monthly Weather Review, May 1960.

2. Hoecker, W. H., 'Three Dimensional Pressure Pattern of the Dallas Tornado and Some Resultant Implications,' Monthly Weather Review, December 1961.

3. 'Wind Forces on Structures', Final Report, Task Committee On Wind Forces, Committee on Loads and Stresses, Structural Division, Transactions, American Society of Civil Engineers, Publication Number 3269, Volume 126, Part II, (1961).

4. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Building and Other Structures," Committee A58.1, American National Standards Institute, 1972.



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Variations of Differential Pressure and Tangential Plus Translational Velocity as a Function of the Distance from the Center of a Tornado

FIGURE 3.3-1

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

The flood protection requirements and provisions for Category I structures are discussed in Sections 2.4.1.1, 2.4.2.2, 2.4.2.3, and 2.4.10.

3.4.2 Analysis Procedure

The methods and procedures by which the static and dynamic effects of the design basis flood conditions are applied to Category I structures are discussed in Sections 2.4.3.6 and 2.4.13.5.

REFERENCES

None.

3.5 MISSILE PROTECTION

Category I structures have been analyzed and designed to be protected against a wide spectrum of credible missiles. Failure of certain rotating or pressurized components of equipment is credible and would presumably lead to generation of missiles. In addition, noncredible missiles are identified and justification is given for their not being a credible source of missiles. Tornado-generated missiles and missiles resulting from activities peculiar to the site are also discussed in this section. It is shown that the missile protection criteria to which the plant has been analyzed and protected comply with the intent of Criterion 4 of 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants.

A very basic premise for protection is to design components and equipment so that they will have a low potential for generation of missiles. In general, the design that results in reduction of missile generation potential promotes the long life and usability of a component, and is well within permissible limits of accepted codes and standards. The following general methods are used in the design, manufacture, and inspection of equipment:

1. Pressurized equipment and sections of piping that from time to time may become isolated under pressure have been provided with pressure relief valves. (Relief valves are in accordance with ASME Section III or the appropriate industry standards.) These valves are present to ensure that no pressure buildup in equipment or piping sections will exceed the design limits of the materials involved.

2. Components and equipment of the various systems have been designed and built to the standards established by the ASME or other equivalent industrial standard. A stringent quality control program has been enforced during manufacture, testing, and installation.

3. Volumetric and ultrasonic testing where required by code, coupled with periodic inservice inspections of materials used in components and equipment, adds further assurance that any material flaws that could permit the generation of missiles will be detected.

The design bases to which the plant has been designed in order to meet the intent of the criterion are listed below.

Design Bases

1. Protection shall be provided against potential missiles that could cause a loss-of-coolant accident (LOCA).

2. Protection shall be provided against potential missiles that could result in the loss of ability to control the consequences of a LOCA, including both the necessity for core cooling and for retention of containment integrity.

3. Protection shall be provided against potential missiles that could jeopardize functions necessary to bring the reactor to a safe shutdown condition during normal or abnormal conditions.

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

The structures that are to be protected against damage from internally-generated missiles outside containment are listed in Table 3.2-1. The systems and components that are to be protected against damage from internally generated missiles outside containment include the ANS Safety Class equipment listed in Tables 3.2-2a, 3.2-2b, and 3.2-3.

A discussion of the individual structures and the systems and components inside these structures is provided below. In general, the systems and components important to safety located in the structures of Table 3.2-1 rely on redundancy and separation for protection from internal missiles generated by failure of high pressure system components. Missiles which result from pipe breaks are not considered. Missiles that are associated with overspeed failures of rotating components are considered a greater safety hazard than those generated by failure of high-pressure system components and are evaluated in more detail.

3.5.1.1.1 Shield Building

No rotating components which might generate missiles are installed between the primary containment and the Shield Building. No high-pressure system components whose failure could generate missiles are located in this area.

3.5.1.1.2 North and South Steam Valve Rooms

No rotating components are installed in these rooms. The barriers provided to limit pipe whip and jet impingement in these rooms also provide protection from any potential missiles generated by failure of any high-pressure system components. The main steam isolation valve stems and the main feedwater isolation valve stems are not postulated as credible missiles in these rooms as explained below.

Numerous features of the main steam isolation valve (MSIV) design and construction serve to prevent the valve stem from being ejected as a missile. The MSIV's are 32 inch wye, bi-directional globe valves that are air-opened, spring-closed, and qualified to ASME Section III, Class 2. Several components of the valve and operator would have to fail concurrently before the valve stem could be ejected. First, the pilot poppet is fastened onto the bottom end of the valve stem and backseated against the poppet cap. The lower segment of the valve stem has a larger diameter than the opening provided in the bonnet, the bottom spring seat, and the bottom bead of the air cylinder that the upper segment of the valve stem and the piston rod normally operate through. The worst case failure tending to cause MSIV stem ejection would be the complete severance of the valve stem itself at the valve bonnet. This failure would eliminate the components and design features described thus far as barriers to valve stem ejection. The internal area of the stem, tending to eject it. However, the valve operator is designed to withstand the transmitted forces acting through the stem during the normal operation of the valve. These thrust forces are at least as great as those that would act on a broken stem. Thus, several additional components of the valve and operator would still have to fail before even this fractured segment of the valve stem could be ejected. First, the top of the valve stem is fastened into the bottom spring seat cap which is in turn bolted to the bottom spring seat. The air cylinder piston rod is located directly over the valve on the same axis and is inserted through the opening in the air cylinder bottom bead. The air cylinder piston and top head (each 3 inch plate) are also located over the valve stem, perpendicular to its axis. The combination of these valve and operator features precludes the MSIV stem (or a segment of it) from being postulated as a credible missile.

Similarly, various features of the main feedwater isolation valve (MFWIV) design and construction serve to prevent its valve stem from being ejected as a missile. The MFWIV's are 16 inch bolted bonnet, wedge gate valves with motor operators and are qualified to ASME Section III, Class 2. Several components of the valve and operator would have to fail concurrently before the stem could be ejected. First, the wedge gate holds the bottom end of the valve stem in a close-fitting ball-and-socket type arrangement. The lower segment of the valve stem itself is backseated against the valve bonnet. This segment has a larger diameter than the openings provided in the bonnet, the upper and lower yoke plates, and the motor operator that the upper segment of the valve stem normally operates through. The worst case failure tending to cause MFWIV stem ejection would be the severance of the stem itself above the backseat segment just described. For the same reasons as given previously for the MSIV, additional components of the MFWIV operator would still have to fail before even this fractured segment of the valve stem could be ejected. First, the upper stem segment is geared into the motor-driven worm shaft of the operator. Then, a pipe cap is bolted on top of the operator unit over the valve stem and perpendicular to its axis. This combination of valve and operator features precludes the MFWIV stem (or a segment of it) from being postulated as a credible missile.

3.5.1.1.3 Auxiliary Building

The only rotating component which was considered for overspeed condition is the auxiliary feedwater steam-driven pump. All other pumps are electrically driven and incapable of achieving an overspeed condition. The manufacturer of the steam turbine (Terry Turbine) has indicated that they have tested their solid wheel turbine under overspeed conditions and no missiles are generated. The pump itself (manufactured by Ingersoll-Rand) may develop missiles under overspeed conditions but its potential for damage is small because of the small size of any missiles postulated. The room containing the pump is oriented in such a manner as to minimize the potential for damage caused by postulated pump missiles.

Consideration of missiles associated with failure of high-pressure system components is handled by redundancy and separation of safety-related systems.

The internal walls and floors of the Auxiliary Building are constructed of reinforced concrete which limit the range of any potential internal missiles. In particular, the spent fuel pool is protected by at least one wall or floor of reinforced concrete from internal missiles generated in other parts of the Auxiliary Building.

The portions of the CVCS and SIS outside of containment are physically separated and protected by concrete barriers of sufficient strength to contain any postulated internally generated missiles.

3.5.1.1.4 Control Building

There are no credible potential internal missiles in this building. There are no rotating components which could have an overspeed failure and no high-pressure systems. The carbon dioxide fire protection system inside the Control Building is not pressurized until it is actuated.

3.5.1.1.5 ERCW Structures

At the Intake Pumping Station the essential raw cooling water (ERCW) pump motors are exposed to the atmosphere. A structural steel grillage system, discussed in Section 3.8.4, provides protection to the pumps from tornado missiles. A concrete shield wall separates the four motors of Train A from those of Train B. These components are arranged in a straight line over a distance of about 100 feet. An overspeed failure is not postulated for these pumps. Even if a failure were postulated, no credible trajectory of any resultant missile could damage enough components to reduce the number available to less than four. No credible failure of any high-pressure component could create a missile which could reduce the availability of pumps on the opposite power train.

No credible, potential internal missile sources are installed in the remainder of the ERCW structure.

3.5.1.1.6 ERCW Pipe Tunnels and RWST Foundations

No credible potential internal missile sources are installed in these structures.

3.5.1.1.7 Diesel Generator Building

Four emergency diesel generators, which are required to supply emergency power to certain engineered safety features, are each located inside a separate room in the Diesel Generator Building. Interior walls of reinforced concrete separate these generators.

There is a mechanical governor on the diesel engine of each diesel-generator unit which is designed to assume control of the engine when there is a tendency to overspeed. In addition, the diesel generators have an overspeed trip which cuts off fuel to the diesel engine upon an overspeed condition. Consequently, no missiles are postulated for overspeed conditions of the generator. The diesel generator units are protected from the effects of a postulated failure of the carbon dioxide storage tank by an 18-inch thick reinforced concrete wall. Therefore, any missiles or pressure build-up generated by a rupture of the carbon dioxide storage tank would not damage essential equipment.

The vent path for the carbon dioxide storage tank compartment is through one set of standard double doors into a stairwell. If additional pressure relief is required, the vent path is through another set of standard double doors which open to the atmosphere from the stairwell.

3.5.1.2 Internally Generated Missiles (Inside Containment)

3.5.1.2.1 Missile Selection

Catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings, and piping leading to generation of missiles is not considered credible. Massive and rapid failure of these components is not postulated because of the material characteristics and 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 as a source of missiles for the reasons discussed in Section 5.2.6.

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

1. Control rod drive mechanism housing plug, drive shaft, and the drive shaft and drive mechanisms latched together.

2. Certain valves.

3. Temperature sensor assemblies.

4. Pressurizer instrument well and heaters.

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

1. Full length control rod drive mechanisms are shop tested at 4105 psig.

2. The mechanism housings are individually hydro-tested to 3107 psig as they are installed on the head adapters of the reactor vessel and they are checked during the hydro-test of the completed reactor coolant system.

3. Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops.

4. The mechanism housings are made of type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

However, it is postulated that the top plug on the control rod drive mechanism will become loose and be forced upward by the water jet. The following sequence of events is assumed: The drive shaft and control rod cluster are forced out of the core by the differential pressure of 2500 psi across the drive shaft; the drive shaft and control rod cluster, latched together, are assumed fully inserted when the accident starts; after approximately 12 feet of travel, the rod cluster control spider hits the underside of the upper support plate; upon impact, the flexure arms in the coupling joining the drive shaft and control cluster fracture, completely freeing the drive shaft from the control rod cluster. The control cluster would be completely stopped by the upper support plate; however, the drive shaft would continue to be accelerated upward to hit the missile shield provided. This analysis is summarized in Table 3.5-1. For a description of the missile shield see Section 3.5.1.2.6.

Valve stems are considered to be extremely unlikely sources of missiles because of the design, construction qualifications, and testing of the valves. The isolation valves installed in the reactor coolant system have stems with a back seat. This effectively eliminates the possibility of ejecting valve stems even if the stem threads fail. Analysis shows that the back seat or the upset end would not penetrate the bonnet. Additional interference is encountered with air and motor operated valves.

Valves with nominal diameter larger than 2 inches have been designed against bonnet body connection failure and subsequent bonnet ejection by means of:

1. Using the design practice of ASME Section VIII which limits the allowable stress of bolting material to less than 20% of its yield strength;

2. Using the design practice of ASME Section VIII for flange design; and

3. By controlling the load during the bonnet body connection stud tightening process.

The pressure containing parts are designed per code Class 1 requirements established by the ASME Section III Code.

The proper stud torquing procedures and the use of a torque wrench, with indication of the applied torque, limit the stress of the studs to the allowable limits established in the ASME Code.

This stress level is far below the material yield. The complete valves are hydrotested per the ASME Section III Code. The stainless steel bodies and bonnets are volumetrically and surface tested to verify soundness.

Valves with a nominal pipe size of 2 inches or smaller are forged and have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads are designed to withstand the hydrostatic end force. The pressure containing parts are designed per criteria established by the ASME III Code specification.

Whereas valve missiles are not generally postulated as outlined in the above discussion, it was decided to conservatively review valves as though their bonnets or stems could become missiles. Representative of these are the valves in the region where the pressurizer extends above the operating deck. Valves in this region are the pressurizer safety valves, the motor-operated isolation valves in the relief line, the air-operated relief valves, and the air-operated spray valves. Although failure of these valves is unlikely, provisions are made to assure protection of safety-related equipment, including the containment liner, from hypothetical missiles from these valves (see Table 3.5-2).

The potential for other valve missiles in the lower compartment to damage safety equipment is assessed to be extremely limited. The same measures taken to prevent damage from other postulated non-valve missiles will be effective against damage from these hypothetical valve missiles. These measures include layout of the basic plant arrangement utilizing the design philosophy of physical separation of equipment and components through distance or by barriers. In consideration of the postulation that valve bonnet fragmentation may occur resulting in the ejection of a valve stem, valves are oriented such that any missile will strike such a barrier.

The only other postulated jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system is that represented by the temperature sensor assemblies, as listed in Table 3.5-3. The resistance temperature sensor assemblies are of two types: 'with well' and 'without well'. Two rupture locations have been postulated: around the weld (or thread) between the temperature element assembly and the boss for the 'without well' element, and the weld (or thread) between the well and the boss for the 'with well' element.

A temperature sensor is installed on the reactor coolant pumps close to the radial bearing assembly. 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 weld 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. Adequate barriers are provided for the missiles above to protect safety-related equipment.

3.5.1.2.2 Missile Description

The postulated control rod drive mechanism (CRDM) missiles are summarized in Table 3.5-1. The velocities of the missiles have been calculated using the method shown in Appendix 3.5A. The reactor coolant discharge rate from the break has been calculated using the Burnell equation.[1] The coolant pressure has been assumed constant at the initial value. No spreading of the water jet has been assumed.

The missile characteristics of the bonnets of the typical valves in the region where the pressurizer extends above the operating deck are given in Table 3.5-2.

The missile characteristics of the postulated piping temperature sensor assemblies are given in Table 3.5-3. 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 of Table 3.5-3.

The characteristics of other missiles postulated within reactor containment are given in Table 3.5-4. A 10-degree expansion half-angle water jet has been assumed.

3.5.1.2.3 Electrical Cables

Electrical cables are not protected against damage from internal missiles. However, separation and redundancy of vital cables are such that any single failure within the protection system will not prevent proper protective action at the system level when required.

3.5.1.2.4 Upper Compartment

The generation of internal missiles inside the upper compartment is not postulated. Piping in this area is not used during normal operation. The upper compartment is protected from missiles generated in the lower compartment by the steel reinforced divider deck and missile shield and by the reinforced concrete walls of the upper portions of the steam generator and pressurizer compartments.

3.5.1.2.5 Ice Condenser Compartment

The generation of internal missiles inside the ice condenser is not postulated. Low energy refrigeration piping is not considered to be a potential missile source. Missiles generated in portions of the lower compartment will be prevented from entering the ice condenser compartment by the crane wall. The trajectory of missiles generated within the bottom regions of the lower compartment is such that the missiles will not pass through the inlet door openings in the lower crane wall except by ricochet. The potential for damage of such a ricocheting missile is considered negligible. This situation is shown in Figure 3.5-1. As can be seen in the figure, the location of the main portions of the reactor coolant system and of the other systems which connect to it are below Elevation 718 feet-0 inches, whereas the openings for the ice condenser lower inlet doors are between Elevation 746 feet-5 inches and Elevation 753 feet-9inches.

3.5.1.2.6 Lower Compartment

The spectrum of missiles generated within the lower compartment is discussed in Sections 3.5.1.2.1 and 3.5.1.2.2. These missiles will not cause failure of vital systems inside the lower compartment. Particular attention was paid to the potential missile damage to the steel containment structure, the emergency core cooling system, and the containment isolation system.

Any missile generated within the lower compartment will not impair the integrity of the steel containment structure. Protection against the postulated missiles in the lower compartment was accomplished by locating a reinforced concrete wall (crane wall), a steel-reinforced concrete slab (divider deck), and steel-reinforced concrete removable blocks (control rod drive mechanism missile shield) between the primary reactor coolant system and the containment structure. Additionally, since there are openings in the crane wall, protection for the containment structure is enhanced by orienting potential missile sources, especially valve components, so that their anticipated trajectory will not permit them to pass through these openings.

The control rod drive mechanism missile shield has been located above the reactor vessel and will prevent the postulated missiles of the control rod drive system from striking the inside surface of the containment structure or the containment spray headers (see Table 3.5-1).

The accumulator tanks and associated check valves and piping are not credible sources of missiles for the containment structure. Components are prevented from becoming a source of damaging missiles by orienting the components so that anticipated missile trajectories are away from the containment structure. Some other high-pressure system components are located in the space between the crane wall and the steel containment structure. Protection is accomplished by orienting components so that the anticipated missile trajectories are away from the containment structure.

The emergency core cooling system (Section 6.3) includes four accumulator tanks which are located in separate rooms between the crane wall and the containment structure. The crane wall protects these tanks and their associated valves and piping from the postulated missiles generated within the lower compartment, and the Shield Building protects them from external missiles. The active components of the system (pumps, motors, and heat exchangers) are located in separate rooms in the Auxiliary Building. Therefore, these active components are protected from the postulated missiles generated within the lower compartment.

Isolation valves of the containment isolation system (Section 6.2.4) are located in three regions: 1) inside the containment structure, 2) between the containment structure and the Shield Building, and 3) outside the Shield Building. The isolation valves which are located inside the containment structure are protected from the postulated missiles generated in the lower compartment by the crane wall, and are protected from tornado-generated missiles by the Shield Building.

Even though the preceding methods have been used to protect the containment structure, emergency core cooling system, and the containment isolation system from potential internal missiles, the basic approach was to assure design adequacy against generation of missiles rather than to allow a missile to be generated and then try to contain the effects.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

The Watts Bar turbine-generators were originally manufactured by Westinghouse Electric Corporation (now Siemens Energy, Inc.). The Watts Bar turbine-generator unit consists of a double-flow high pressure turbine and three double-flow low pressure turbines with extraction nozzles arranged for seven stages of feedwater heating. The Unit 1 turbine utilizes a digital electrohydraulic control (DEH) system for control of both speed and load. The DEH system is composed of redundant filed modules coupled through suitable electrohydraulic transducers to a high-pressure hydraulic fluid system, provides control of the main stop, governing, intercept, and reheat stop valves of the turbine. Equipment and functions are provided in the control system to protect the turbine from dangerous operating conditions by tripping or unloading the turbine as appropriate and indicate the condition and action to the operator. Any tripping action from this system results in removal of hydraulic fluid pressure resulting in rapid closure of all turbine inlet valves. Turbine trip inputs are configured such that a single DEH component failure will not initiate a spurious turbine trip, nor disable the turbine protection function.

Emergency speed protection is provided by the independent overspeed protection system (IOPS). The IOPS will trip the turbine independently of the DEH system. The IOPS uses passive speed probes to monitor turbine speed that are independent of the speed probes that are used by DEH. As with the DEH system, a single component failure of the IOPS will not initiate a spurious turbine trip, nor disable the turbine protection function.

Additional turbine protection is provided that will trip the turbine on evidence of low condenser vacuum, abnormal thrust bearing wear, or low bearing oil pressure (See Section 10.2.4 for a complete list). The DEH system is equipped with solenoid-operated trip devices, which provide a means to initiate direct tripping of the turbine upon receipt of appropriate electrical signals, as shown in figure 10.2-1. Also, when a turbine trip is initiated, the extraction system nonreturn valves are tripped to close by means of a pilot dump valve connected to the turbine trip system. For overpressure protection of the turbine exhaust hoods and the condenser, four rupture diaphragms which rupture at approximately 5 psig are provided on each turbine exhaust hood. Additional protective devices include exhaust hood high temperature alarm and manual trip. Each stop, governing, reheating stop, and intercept valve is spring closed; therefore, it is necessary only to dump the high pressure fluid from under the servoactuators to close the valves.

For additional details on the turbine, see Section 10.2.

The unit 2 turbine utilizes a Westinghouse designed electrohydraulic control (EHC) system for control of both speed and load. The EHC system, composed of solid state electronic devices coupled through suitable electrohydraulic transducers to a high-pressure hydraulic fluid system, provides control of the main stop, governing, intercept, and reheat stop valves of the turbine. Emergency speed protection is provided by a mechanical overspeed trip mechanism, backed up by an electrical overspeed trip circuit.

3.5.1.3.2 Potential Missile Sources and Missile Characteristics

The most significant source of turbine missile is a burst-type failure of one or more bladed shrunk-on disks of the low-pressure (LP) rotors.

Failures of the high-pressure (HP) and generator rotors would be contained by relatively massive and strong casings, even if failure occurred at maximum conceivable overspeed of the unit. Evaluations of existing and new retrofit HP rotors were performed considering various failure modes to assess the potential for generating a missile [23]. Ductile burst would require rotational speed beyond terminal speed as explained. Failure due to high cycle fatigue has not occurred in the past and the retrofit rotors have improved design safety factors. Failure due to low cycle fatigue (LCF) is unlikely since rotor LCF is significantly greater than 10,000 start cycles for original and retrofit rotors. Based on the successful operating history of nuclear HP rotors and the results of this evaluation, HP rotors of integral construction od no need to considered when assessing missile generation probability of nuclear turbines.

There is a remote possibility that some minor missiles could result from the failure of couplings or portions of rotors which extend outside the casings. These missiles would be much less hazardous than the LP disk missiles, due to low mass and energy and therefore, have not been considered.

The probability of a turbine missile generation (P1) is evaluated by conservatively considering two distinct types of LP shrunk-on disk failures, namely:

1. Failure at normal operating speed up to 120% of the rated speed and
2. Failure due to run-away overspeed greater than 120% of the rated speed for all disks.

Normal operating speed conditions are expected to occur one or more times per year of operation, but run-away overspeed conditions are expected to occur rarely. Missiles resulting from normal operating speed failures are the result of the brittle fracture of turbine blade wheels or portions of the turbine rotor itself. Failures of this type can occur during startup or normal

operation. Missiles resulting from run-away overspeed failures would be generated if the overspeed protection system malfunctioned and if the turbine speed increased to a point at which the low-pressure wheels or rotor would undergo ductile failure.

The probability of normal operating speed reaching the value up to 120% of the rated speed is assumed to be 1.0, conservatively. It is also conservatively assumed that, given the overspeed protection system fails, the probability of a disk burst and that of casing penetration of the burst fragments is 1.0 for all disks. The overspeed probability is directly correlated to the inspection interval as well as the test frequency of the overspeed protection system.

Missiles from a turbine failure can be divided into the following two groups:

1. "low-trajectory" or "direct" missiles, which are ejected from the turbine casing directly toward an important to safety SSC, and
2. "high-trajectory" missiles, which are ejected upward through the turbine casing and may cause damage if the falling missile strikes an important to safety SSC.

Low Pressure Turbine Construction and Design

Each unit has three (3) LPs. On Unit 1, LPA and LPB are Westinghouse BB281 10-disc design rotors. LPC is a Siemens 13.9m^2 damped blade, advanced disc design (6-disc) rotor. All three LPs of Unit 2 have the advanced disc design rotor with 13.9m^2 free standing blades.

The double flow low pressure turbine incorporates high deficiency blading, diffuser type exhaust and liberal exhaust hood design. The low pressure turbine cylinder is fabricated from steal plate to provide uniform wall thickness thus reducing thermal distortion to a minimum. The entire outer casing is subjected to low temperature exhaust steam.

For LPA and LPB (Unit 1), the temperature drop of the steam from its inlet to the LP turbine to its exhaust from the last rotating blades is taken across three walls: an inner cylinder No. 1, a thermal shield, and an inner cylinder No. 2. For LPC (Unit 1) and the Unit 2 LPs, the temperature drop is across two walls: an inner casing and a thermal shield. This precludes a large temperature drop across any one wall, except the thermal shield which is not a structural element, thereby virtually eliminating thermal distortion. For LPA and LPB (Unit 1), the fabricated inner cylinder No. 2 (inner casing for LPC (Unit 1) and Unit 2 LPS) is supported by the outer casing at the horizontal centerline and is fixed transversely at the top and bottom and axially at the centerline of the steam inlets, thus allowing freedom of expansion independent of the outer casing. For LPA and LPB (Unit 1), inner cylinder No. 1 is, in turn, supported by inner cylinder No. 2 at the horizontal centerline and fixed transversely at the top and bottom and axially at the center line of the steam inlets, thus allowing freedom of expansion independent of inner cylinder No. 2. For LPA and LPB, inner cylinder No. 1 is surrounded by the thermal shield. For LPC (Unit 1) and the Unit 2 LPs, the inner casing is surrounded by the thermal shield. The steam leaving the last row of blades flow into the diffuser where the velocity energy is converted to pressure energy.

For LPA and LPB (Unit 1), the outer cylinder and the two inner cylinders are fabricated mainly of ASTM 515-GR65, or equivalent material. For LPC (Unit 1) and Unit 2 LPs, the outer cylinder and inner casing are fabricated mainly of ASTM 515-GR65, or equivalent material.

The low-pressure rotors are made of NiCrMoV alloy steel.

The shrunk-on discs are made of NiCrMoV alloy steel. There are ten (for LPA and LPB, Unit 1) and six (for LPC (Unit 1) and Unit 2 LPs) discs shrunk on the shaft with five per flow (for LPA and LPB, Unit 1) and three per flow for LPC (Unit 1) and Unit 2 LPs. These discs experience

different degrees of stress when in operation. For LPA and LPB (Unit 1), Disc No. 2, starting from the transverse centerline, experiences the highest stress, while Disc No. 5 experiences the lowest. For LPC (Unit 1) and Unit 2 LPs, Disc No. 3 starting from the transverse centerline, experiences the highest stress, while Disc No. 1 experiences the lowest.

The minimum specified mechanical properties for the cylinders, rotor, and discs are shown in Section 10.2.3.1.

3.5.1.3.3 Primary Safety-Related Equipment Installations and Structures

The primary safety-related equipment installations and structures at the plant are those whose loss could lead to conditions in excess of the guidelines specified in 10 CFR 100. Items in this category are those in which a single strike by a potential turbine missile could result in a loss of the capability to function in the manner needed to meet these guidelines. At the Watts Bar Nuclear Plant, these are the a) Reactor Building, b) main control room, c) spent fuel pool, d) main steam valve rooms, and e) ERCW electrical conduits from manhole Nos. 1-3.

Nuclear power plants should protect their essential SSCs against both high-trajectory and low-trajectory turbine missiles resulting from the failure of main turbine-generator sets. Plants can protect essential SSCs against turbine missiles by any one of a number of approaches; however, the preferred option of Regulatory Guide (RG) 1.115 is as follows:

> For favorably oriented turbines, appropriately placing and orienting the turbine units such that essential SSCs are located outside the low-trajectory missile strike zone defined in RG 1.115 and limiting the calculated turbine missile generation frequency for high-trajectory missiles to a value less than 10^{-4} per year.

Each Unit's turbine-generator is located south of the Reactor Building with its shaft oriented north-south and is considered to be favorably oriented. The turbine placement and orientation are shown in Figure 3.5-4. Favorably oriented turbine generators are located such that the containment and all, or almost all, safety-related structures, systems and components (SSCs) outside containment are excluded form the low-trajectory hazard zone described in RG 1.115. With the exception of the ERCW conduit, all safety-related structures, systems and components are located outside the low-trajectory missile strike zone. For the ERCW conduit target, the strike probability due to a low-trajectory turbine missile is considered to be zero since the rotation of the turbine will preclude a tangential missile from directly impacting the ERCW conduit as the turbine pedestal and Turbine Building structure provide barriers to the trajectory. Thus, postulated low-trajectory missiles cannot directly strike safety-related SSCs and are not considered a threat.

In addition, the safety-related SSCs are protected from high-trajectory turbine missiles by controlling the turbine missile generation frequency (P1) to be less than 10^{-4} per year (see Section 3.5.1.3.5).

3.5.1.3.4 Turbine Missile Protection Criterion

The turbine missile protection criterion utilized in the design of the Watts Bar Nuclear Plant was that the probability of unacceptable damage (P4) should not be significant. In this instance, an event having a probability of causing unacceptable damage on the order of about 10^{-7} per year per reactor unit at the plant is not considered significant. Therefore, for the two-unit Watts Bar Nuclear Plant, an event having a probability of occurrence less than 2×10^{-7} will fulfill this criterion.

3.5.1.3.5 Turbine Missile Hazard Evaluation

The probability of unacceptable damage resulting from turbine missiles, P4, is expressed as the product of the following three items:

1. the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing, P1,
2. the probability of ejected missiles perforating intervening barriers and striking essential SSCs, P2, and
3. the probability of essential SSCs that are struck failing to perform their safety functions, P3.

Stated in mathematical terms, $P4 = P1 \times P2 \times P3$. As noted in Section 3.5.1.3.4, P4 is limited to less than 10^{-7} per year, which the NRC staff considers to be an acceptable risk rate for the loss of an essential SSC from a single event.

In the past [11], analyses assumed the probability of turbine missile generation (P1) to be approximately 10^{-4} per turbine year for a favorably oriented unit, based on the historical failure rate. The strike probability (P2) was estimated on the basis of postulated missile sizes, shapes and energies and on available plant specific information such as turbine placement and orientation, number and type of intervening barriers, target geometry, and potential missile trajectories. The damage probability (P3) was generally assumed to be 1.0. The overall probability of unacceptable damage to safety-related systems (P4), which the sum over all targets of the product of these probabilities, was then evaluated for compliance with the NRC safety objective. This logic placed the regulatory emphasis on the strike probability, that is, it necessitated that P2 be made less than or equal to 10^{-3}, and disregarded all the plant specific factors that determine the actual P1 and its unique time dependency.

Although the calculation of strike probability was not difficult in principle, for the most part being not more than a straightforward ballistics analysis, it presented a problem in practice. The problem stemmed from the fact that numerous modeling approximations and simplifying assumptions were required to make tractable the incorporation into acceptable models of available data on the (1) properties of missiles, (2) interactions of missiles with barriers and obstacles, (3) trajectories of missiles as they interact with and perforate (or are deflected by) barriers, and (4) identification and location of safety-related targets. The particular approximations and assumptions made tended to have a significant effect on the resulting value of P2. Similarly, a reasonably accurate specification of the damage probability (P3) was not a simple matter because of the difficulty in defining the missile impact energy required to render given safety-related systems unavailable to perform their safety functions and the difficulty in postulating sequences of events that would follow a missile-producing turbine failure.

Because of the uncertainties associated with calculating P2 and P3, the staff concludes that such analyses are "order of magnitude" calculations only. On the basis of simple estimates for a variety of plant layouts, the strike and damage probability product can be reasonably assumed to fall in a range that depends on the gross features of turbine generator orientation. For favorably oriented turbine generators, such as Watts Bar, the product of P2 and P3 tends to be in the range of 10^{-4} to 10^{-3} per year. For these reasons, in the evaluation of P4, the probability of unacceptable damage to safety-related systems from potential turbine missile, the staff is giving credit for the product of the strike and damage probabilities of 10^{-3} (P2xP3) for a favorably oriented turbine and is discouraging the elaborate calculation of these values. For these reasons, strike and damage calculations were not performed for Watts Bar instead the missile analysis [15] was updated for the new upgraded 13.9m² rotors as documented in the current missile reports [13].

Therefore, the NRC staff has shifted emphasis in the reviews of the turbine missile issue from the strike and damage probability (P2xP3) to the missile generation probability (P1) and, in the process, has attempted to integrate the various aspects of the issue into a single, coherent evaluation. The staff believes that maintaining an initial small value of P1 through turbine testing and inspection is a reliable means of ensuring that the objectives precluding turbine missiles and unacceptable damage to safety-related structures, systems, and components can be met. It simplifies and improves procedures for evaluation of turbine missile risks and ensures that the public health and safety is maintained.

Unit 1

A missile probability analysis was performed for the Unit 1 BB281-13.9m² damped blade rotor with advanced disc design shrunk-on discs. Based on conservative assumptions, the probability of turbine missile generation for unit speeds up to 120% of rated speed is 5.54×10^{-5}. It is observed that LPC, being a Siemens 13.9m² advanced disc rotor design, complies with the NRC limits up to 100,000 operating hours between successive disc inspections provided that no cracks are detected in those inspections. LPA and LPB rotors, being of Westinghouse design, remain unchanged and their recommended inspection interval of 60 months between LP disc inspections remains conservatively unchanged.

As documented in WCAP-16501-P and LTR-RAM-I-07-039 [21, 22], Unit 1 has elected to perform turbine valve test intervals at a frequency of every 6 months. This results in a probability of overspeed of 1.93×10^{-7} per year [13a].

Unit 2

A missile probability analysis was performed for the Watts Bar 2 BB281-13.9m² free standing rotors with advanced disc design shrunk-on discs. Based on conservative assumptions, the probability of an external missile for speeds up to 120% of rated speed is 2.13×10^{-6} for a disc inspection interval of 100,000 operating hours.

As documented in WCAP-16501-P and LTR-RAM-I-10-040 [21, 22], Unit 2 has elected to perform turbine valve test intervals at a frequency of every 6 months. This results in a probability of overspeed of 1.39×10^{-7} per year [13b].

As summarized in the missile reports [13], these probabilities (P1) for both Units 1 and 2 are well below the NRC limit of 10^{-4} per year for a favorably oriented unit. Therefore, it is concluded that the safety-related SSCs are protected from high-trajectory turbine missiles. The missile analysis methodology used was submitted to the NRC and approved in March 2004 [12].

Based on the calculated probabilities of turbine missile generation ($P1<10^{-4}$) and the NRC approved value of the strike and damage probability ($P2xP3=10^{-3}$) for favorably oriented turbines, the probability of unacceptable damage resulting from turbine missiles, P4, is less than 10^{-7} per year [15]. Therefore, for the two-unit plant, an event having a probability of occurrence less than 2×10^{-7} fulfills the criterion.

3.5.1.3.6 Turbine Missile Selection

Analyses described above indicate that the hazard from turbine missiles at the Watts Bar Nuclear Plant is not significant. In addition, the plant uses turbines designed, manufactured, installed, and operated in accordance with standards that minimize the possibility of an accident that may produce dangerous missiles. The unit has its essential, safety-related equipment installations and structures positioned to minimize the strike probability on these items.

Such findings indicate that the potential for turbine generated missiles at the Watts Bar Nuclear Plant is credible, but not significant. Therefore, turbine missile hazards need not be considered in the design of the Watts Bar Plant.

3.5.1.4 Missiles Generated By Natural Phenomena

Category I structures at Watts Bar Nuclear Plant are designed for tornado- generated missiles based on the following:

1. Spectrum A (see Table 3.5-7) was used in the design of:

   a. Manholes and protective slabs over manholes for Class 1E electric systems.

   b. Protection for Class 1E conduit duct runs.

   c. Hatch assemblies for personnel access openings to pipe tunnels A & B.

   d. Slabs supporting the ERCW piping at the Intake Pumping Station.

   e. Pipe encasement at the Diesel Generator Building.

   f. Refueling Water Storage Tank foundation.

   g. Roofs and walls of Category I structures except as noted for Spectrums B, C & D.

   h. Protection for diesel generator exhaust stacks.

   i. Protection for fuel oil storage tank vent lines in Diesel Generator Building.

2. Spectrum B (see Table 3.5-8) was used in the design of the equipment doors and bulkheads on the Diesel Generator Building.

3. Spectrum C (see Table 3.5-9) was used in the design of the structures not covered by Items 1, 2 or 4 (Intake pumping station structural steel roof, ERCW standpipe encasement, ERCW discharge overflow structure, and ERCW valve covers).

4. Spectrum D (See Table 3.5-17) was used in the design of the Additional Diesel Generator Building and any additional Category I Structures after July 1979.

3.5.1.5 Missiles Generated by Events Near the Site.

There are no postulated accidental explosions in the vicinity of the site (See Section 2.2.3). The only significant nearby industrial activity is the Watts Bar Steam Plant. Turbine missiles from this plant are treated in Section 3.5.1.3. No other missiles are considered significant. Therefore, the Watts Bar Nuclear Plant need not be designed for protection against missiles generated by explosions of trucks, trains, ships, barges, industrial facilities, hydrogen storage tanks, pipelines and military facilities.

3.5.1.6 Aircraft Hazards

There is one federal airway passing within two miles of the nuclear facility. The probability per year of an aircraft crashing into the plant (P_{FA})[17] is estimated in the following manner:

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

where:

C = inflight crash rate per mile for aircraft using airway.

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

N = number of flights per year along the airway, and

A = effective area of plant in square miles.

For the Watts Bar site, C = 4.0 x 10^{-10}, w = 8 miles, N = 2 x 2200/year, A = 0.01 mi^2. The value for N, per Section 2.2.2.5, was doubled to account for increases in aircraft traffic. Therefore, P_{FA} = 2.2 x 10^{-9}/year. This conservative upper bound probability is less than the Standard Review Plan range of "1 x 10^{-7}" and indicates that further consideration of hazard from this federal airway is unnecessary.

There are no airports located within five miles of the site. There are no airports with projected operations greater than 500 d^2 movements per year located within 10 miles of the site or greater than 1,000 d^2 outside 10 miles, where d is the distance in miles from the site.

There are no military installations or any airspace use that might present a hazard to the site. Therefore, aircraft hazards need not be considered in the design of the Watts Bar Nuclear Plant.

3.5.2 Systems to Be Protected

Systems whose failure could lead to offsite radiological consequences or which are required for reactor shutdown and cooldown under normal and/or design basis conditions are listed in Table 3.2-1.

These include 1) the fluid handling equipment in ANS Safety Classes 1, 2a, 2b or 3 listed by system in Tables 3.2-2a and 3.2-2b, 2) Class 1E electrical systems and components listed in Table 3.2-3, and 3) the Heating, Ventilating, and Air Conditioning components in TVA Classes B, M, Q, and S as described in Table 3.2-2a. A more detailed discussion of required equipment and its safety-related implications may be found in the UFSAR sections covering individual system.

It is important to note that all of the equipment referenced above is required for some safety-related function but not all at once. The list of required equipment for a particular missile event depends on the nature of the missile (whether it is associated with or can cause a LOCA, whether or not the missile is being generated from a safety-related piece of equipment, etc). Hence, much of the missile damage consideration outside containment can be reduced to looking at those systems and components required for reactor shutdown, coolant system makeup, and for decay heat removal to the ultimate heat sink.

Table 3.5-14 lists outdoor features, including air intakes and exhausts, which may be required to perform a safety-related function coincident with or following the occurrence of a tornado.

For the Watts Bar Nuclear Plant the layout of structures and equipment is such that there are no systems or components which rely upon redundancy and separation alone for protection against externally generated missiles. There are however certain systems and components which, due to their location and/or separation of trains, are inherently protected against specific types of missiles. This protection by separation and location concept has been addressed where applicable in the missile analyses of Section 3.5.1.

3.5.3 Barrier Design Procedures

Historical Information - To arrive at a formula to use in computing penetration into concrete walls, a comparison was made of formulas listed in ORNL-NSIC-22, "Missile Generation and Protection in Lightwater-Cooled Power Reactor Plants." Four equations were studied in ORNL-NSIC-22 in connection with penetration in concrete. Two of these, the Army Corps of Engineers formula and the National Defense Research Committee formula, do not apply for impact velocities under 500 ft/sec and thus are not applicable here (velocity of 300 mph = 440 ft/sec). The remaining two equations are the modified Petry formula and the Ballistic Research Laboratory formula. These two formulas were compared for a 6-inch diameter missile of 100 pounds and a 16-inch diameter missile of 2,500 pounds with velocities in the range from 0 to 500 ft/sec. As seen in Figures 3.5-6 and 3.5-7, the Petry formula is the most conservative for velocities greater than 150 to 200 mph.

The following describes the barrier procedures utilized for concrete barriers. The depth to which a missile penetrated a concrete wall was estimated by use of the modified Petry formula. [6]

$$D' = 12\,KAV'\,[1 + e^{-4(a-2)}]$$

where

D' = depth of penetration

V = impact velocity

K = A material constant

A = weight of missile/impact area of missile

$$V' = \log_{10}\left[1 + \frac{V^2}{215{,}000}\right]$$

$$a = \frac{T}{12KAV'}$$

T = wall thickness

The results are given in Figures 3.5-8 and 3.5-9. According to C. V. Moore,[7] spalling on the inside face of a wall does not occur for penetrations less than two-thirds the wall thickness.

Conservatism was assured by assuming nondeformable missiles in the penetration analysis using the modified Petry formula.

None of the postulated missiles described in Section 3.5.1, internal or external, will impair the capability of the engineered safety features to shut down the reactor or to maintain the reactor in a safe shutdown mode indefinitely. For portions of the engineered safety features located within the containment structure, protection against missiles generated inside containment is accomplished with the basic approach of assuring design adequacy against generation of credible missiles rather than to allow missile formation and try to contain the subsequent effects. Further, valves are oriented so that the trajectory of missiles will not likely pass through openings in the crane wall and the valve bonnets and stems will not penetrate the containment shell should they strike it. For these same engineered safety features, protection against tornado-generated missiles is provided by the Shield Building. If one of the pressurizer heaters in the bottom of the pressurizer should become loose and become a jet-propelled missile, it would move downward and could strike the pressurizer surge line beneath the pressurizer. The line will not be perforated and will not jeopardize the capability to bring the nuclear facility to a safe shutdown.

For those portions of the engineered safety features located outside the Shield Building and required for shutdown of the reactor and/or indefinite maintenance of the reactor in the safe shutdown mode, protection is provided against tornado-generated missiles. Protection is provided by locating these features within structures which have been designed to withstand damage by the spectrum of credible tornado-generated missiles.

The postulated missiles inside the containment as defined in Tables 3.5-1, 3.5-2, 3.5-3, and 3.5-4 have been investigated to determine their penetration characteristics. Penetration depths, or minimum thickness to just perforate, have been calculated based upon three commonly used equations. They are:

1. The Stanford Equation.

2. The Ballistics Research Laboratory Equation.

3. The Recht and Ipson Equation.

The minimum thicknesses to just perforate a plate having the characteristics of SA-516-GR70 carbon steel represent the largest values obtained from the three above-mentioned equations. The worst case involves a penetration depth that is 45% of the actual containment thickness. Based upon the analysis, it is reasoned that none of the postulated missiles pose a threat to the integrity of the containment.

Tornado missile impact loads, where required, were calculated based upon several applicable techniques. Impact loads for all missiles of Spectrum A (Table 3.5-7) except missile A3, the 4,000-pound automobile at 50 miles per hour, were determined by the relationships presented in Reference [9].

Missile A3 loads were based on actual test results and analysis technique. Time histories of decelerations were obtained from the National Highway Safety Bureau for automobile crash tests early in a crash safety program. Their time histories were converted to shock spectra by the usual methods and dynamic load factors were plotted against period of structures or elements of structures. An envelope was then constructed enveloping all spectra.

To determine the automobile impact load on a structure or element of a structure, the natural period of the item was determined and the appropriate loading obtained from the shock spectra. This technique yields the maximum load irrespective of time.

Values obtained from this technique have been corroborated with subsequent reports by the National Highway Safety Bureau. In Reference [5], time histories of forces are presented for several automobile crash tests which are closely confirmatory. The impact loads obtained by the previously described methods were then applied to the structures and the structures were analyzed for the effect of the loads by conventional analytical methods. Impact loads from the missiles of Spectrum C (Table 3.5-9) were calculated using the procedures of Reference [8]. See Section 3.5.1.4 for a discussion of structures designed for Spectrum C.

The structural steel grillage roof system for the Intake Pumping Station was designed for impact from the tornado missiles listed in FSAR Table 3.5-9. The EPRI testing results[18] were used to determine the possible maximum impact force for the steel grillage roof system. The force-time history of the end impact loading was further refined to account for the stiffness of the missile and the target. The impulse-momentum principle[19, 20] was used to analyze the midspan and end impacts. For the midspan impact, the response of the elastic-plastic single-degree system was considered along with a maximum ductility ration of μ = 20 for bending deformation[8].

Tornado missile protection for all safety-related buried piping is provided by one of the four protective schemes described below.

1. 10 feet of compacted fine-grained soil.

2. 7 feet of compacted crushed stone.

3. 18 inches of conventional unreinforced concrete.

4. 18 inches of roller-compacted unreinforced concrete.

In each scheme, a 12-inch cushion of either compacted sand or fine-grained earthfill is required over the top of the pipe.

The acceptability of each scheme has been verified by a full-scale test program[16] in which missiles from the NRC spectrum were dropped from a helicopter into test pits of crushed stone or earthfill and onto concrete slabs. The missiles used in the testing were:

1. a 1,500-pound utility pole,

2. a 12-inch diameter schedule 40 steel pipe,

3. a 1-inch diameter steel rod,

4. a 3-inch diameter schedule 40 steel pipe, and

5. a 6-inch diameter schedule 40 steel pipe.

Of these missiles, the 12-inch pipe and utility pole caused the greatest penetration depths. Impact velocities of 200-215 ft/s were achieved for both the utility pole and 12-inch pipe which equals or exceeds the design velocities for those missiles as listed in Tables 3.5-7 and 3.5-9. The protective thicknesses listed above are based on the maximum thicknesses observed in the test program and are, therefore, conservatively chosen.

It is concluded that the missile protection criteria to which the plant has been analyzed and protected against comply with Criteria 2 and 4 of 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants.

3.5.3.1 Additional Diesel Generator Building (And Other Category I Structures Added After July 1979)

The openings in the walls and roof for access, ventilation, air intakes, and exhaust discharge, are designed to withstand the effects from the tornado missiles listed in Spectrum D of Table 3.5-17.

The 480V auxiliary board room (ABR) ventilation air intake vent is the only primary safety-related equipment located outside the Additional Diesel Generator Building[1] not protected against tornado missiles. The roof opening for the ABR vent is protected against tornado missile entry by a missile shield installed inside the roof structure. In-lieu of protecting the ABR vent from tornado missiles, operator actions are specified in the event of a tornado warning to restore ventilation cooling to the ABR. Overall structural response evaluation of concrete barriers to tornado missile impact was performed using the general requirements of Appendix C, ACI 349-76, "Code Requirements For Nuclear Safety-Related Concrete Structures." Minimum concrete thickness required to resist penetration, perforation or backface scabbing from these tornado missiles are given in Table 3.5-18.

REFERENCES

1. Burnell, J., 'The Flow of Boiling Water Through Nozzles, Orifices, and Pipes.' The Institute of Engineers, Australia, March 1946.

2. Deleted

3. Deleted

4. Deleted

5. DeLeys, Norman J., et al., 'Full Scale Crash Tests of Rigid Simulated Heavy Vehicle Underride Guard,' Cornell Aeronautical Laboratory, Inc., Buffalo, New York, March 1970. Report prepared for Department of Transportation, Federal Highway Administration, National Highway Safety Bureau, Washington, DC.

6. Amirikan, A., 'Design of Protective Structures,' NP-3726, Bureau of Yards and Docks, Department of the Navy, Washington, DC, August 1950.

7. Moore, C. V., 'The Design and Barricades for Hazardous Pressure Systems,' Nuclear Engineering and Design, Vol. 5, 1967, pp. 85-86. (Applicable to Unit 1 and Unit 2)

8. 'Design of Structures for Missile Impact', BC-TOP-9A, Revision 2 Topical Report, Bechtel Power Corporation, San Francisco, California, September 1974.

9. Westinghouse Systems Standard Design Criteria, "Protection Criteria Against Dynamic Effects Resulting from Pipe Rupture" (Identification No. 1.19).

10. Deleted

11. Letter from Mr. Herbert N. Berkow, NRC Director, to Mr. Stan Dembkoski, SWPC Director, dated March 30, 2004, Subject: Final Safety Evaluation Regarding Referencing the Siemens Technical Report No. CT-27332, Revision 2, "Missile Probability Analysis for the Siemens 13.9m2 Retrofit Design of Low-Pressure Turbines by Siemens AG", TAC No. MB7964.

12. Topical Report TP-04124-NP-A: "Missile Probability Analysis for the Siemens 13.9m2 Retrofit Design of Low-Pressure Turbine by Siemens AG", Submitted to the Nuclear Regulatory Commission, For Public Record, June 7, 2004, Siemens Westinghouse Power Corporation.

    Note: Topical Report TP-04108-P-A: "Missile Probability Analysis for the Siemens 13.9m² Retrofit Design of Low-Pressure Turbine by Siemens AG", Submitted to the Nuclear Regulatory Commission, Not For Public Record, June 7, 2004, Siemens Westinghouse Power Corporation. This is the proprietary version of the same report referenced above.

13a. CT-27547 Rev. 1, Missile Probability Analysis Report, TVA Watts Bar 1, BB281-13.9m² Damped Blades, October 14, 2016, Siemens Energy, Inc. (Applicable to Unit 1)

13b. CT-27467, Missile Report, TVA Watts Bar 2, BB281-13.9m2, October 6, 2009, Siemens Energy, Inc. (Applicable to Unit 2 only)

14. Deleted

15. Calculation TI-521, "Probability for Turbine Missile Strike Damage at Watts Bar Nuclear Plant."

16. CEB Report CEB-81-20, "Testing of Protective Cover for Essential Buried Structures."

17. U.S. Nuclear Regulatory Commission NUREG-0800, "Standard Review Plan," Section 3.5.1.6.III.2.

18. Stephenson, A. E., and G. E. Sliter, “Full-Scale Tornado-Missile Impact Tests,” Electric Power Research Institute Final Report, EPRI NP-440, July 1977.

19. Civil Design Guide DG-C1.5.7, Rev. 1, “Design of Slabs for Missile Impact.”

20. Topical Report TVA-TR78-4, “Design of Structures to Resist Missile Impact,” TVA Office of Engineering Design and Construction (B41 851204 002).

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

22a. LTR-RAM-I-07-039, “Review of Watts Bar Unit 1 for Applicability of WCAP-16501-P”, September 2007 (Applicable to Unit 1 only, attached to WAT-D-11600)

22b. LTR-RAM-I-10-040, “Review of Watts Bar Unit 2 for Applicability of WCAP-16501-P”, September 2010 (Applicable to Unit 2 only, attached to 25402-011-V1A-MG00-03478-001)

23. EC-02262, "Missile Generation Risk Assessment for Original and Retrofit Nuclear HP Rotors", December 17, 2002, Siemens Westinghouse Power Corporation.

WBN

TABLE 3.5-1

SUMMARY OF POSTULATED CRDM MISSILE ANALYSIS

POSTULATED MISSILES		WEIGHT(5) (LB)	THRUST AREA (IN^2)	IMPACT AREA (IN^2)	IMPACT(1) VELOCITY(4) (Ft/SEC)	KINETIC(1) ENERGY (Ft-Lbs)
1.	Mechanism Housing Plug	11	5.94(2)	7.07	450	34,600
2.	Drive Shaft Assembly	136	2.41	2.41	179	69,800
3.	Mechanism Housing(3) Plug And Drive Shaft Impacting On Same Missile Shield Spot	-	-	-	-	-
4.	Drive Shaft Latched To Mechanism	1500	2.41	11.04	34	26,400

NOTES:

(1) Velocities and kinetic energies for a distance of 4.389(ft) between top of CRDM Housing and Missile Shield.

(2) Flow discharge area (equal to thrust area-no expansion of jet assumed).

(3) Assume drive shaft further pushes housing plug into shield.

(4) For the calculational methods, see Appendix 3.5A.

(5) Dry weight

TABLE 3.5-2

TYPICAL POSTULATED VALVE MISSILE CHARACTERISTICS

Missile Description	Weight (lb)	Flow Discharge Area (in^2)	Thrust Area (in^2)	Impact Area (in^2)	Weight to Impact Area Ratio (psi)	Terminal[(1)] velocity (fps)
Safety Relief Valve Bonnet (3" x 6" or 6" x 6")	350	2.86	80	24	14.5	110
3 Inch Motor Operated Isolation Valve Bonnet (plus motor and stem) (3")	400	5.5	113	28	14.1	135
2 Inch Air Operated Relief Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
3 Inch Air Operated Spray Valve Bonnet (plus stem)	120	5.5	50	50	2.4	190
4 Inch Air Operated Spray Valve	200	9.3	50	50	4	190

NOTES:

1. For the calculational methods, see Appendix 3.5A

TABLE 3.5-3

POSTULATED PIPING TEMPERATURE ELEMENT ASSEMBLY MISSILE CHARACTERISTICS

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

Characteristics	"without well"	"with well"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	7.1 in^2	9.6 in^2
Missile Weight	11.0 lb	15.2 lb
Area of Impact	3.14 in^2	3.14 in^2
$\frac{\text{Missile Weight}}{\text{Impact Area}}$	3.5 psi	4.84 psi
Velocity [(1)]	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
$\frac{\text{Missile Weight}}{\text{Impact Area}}$	1.27 psi	1.94 psi
Velocity	75 ft/sec	120 ft/sec

NOTES:
1. For the calculational methods, see Appendix 3.5A

TABLE 3.5-4

CHARACTERISTICS OF OTHER MISSILES
POSTULATED WITHIN REACTOR 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
Missile Weight / Impact Area	0.5 psi	4.1 psi	6.25 psi
Velocity[(1)]	260 ft/sec	100 ft/sec	55 ft/sec

NOTES:

1. For the calculational procedures, see Appendix 3.5A

TABLE 3.5-5

DELETED

(See Section 10.2.3.1, historical information)

TABLE 3.5-6

DELETED

TABLE 3.5-7

TORNADO MISSILE SPECTRUM A FOR CATEGORY I STRUCTURES[1]

Missile[2]	Description	Design Velocity Exterior Wall ft/s (mi/h)	Design Velocity Roof System ft/s (mi/h)
A1	Wood plank, 2 in. x 4 in. x 12-ft long, weight 27 lbs	440 (300)	--
A2	Cross tie, 7 in. x 9 in. x 8.5 ft long, weight 186 lbs	440 (300)	--
A3	Automobile, weight 4000 lbs, up to 25 ft above grade at structure	73 (50)	--
A4	Steel pipe, 2-in. diameter, 7-ft long, weight 26 lbs	147 (100)	--
A5	Steel rod, 1-in. diameter x 3-ft long, weight 8 lbs	210 (143)	168 (115)
A6	Utility pole, 13.5 in. diameter x 35-ft long, weight 1490 lbs, up to 30 feet above grade	200 (136)	160 (109)

Notes:

(1) See Section 3.5.1.4.

(2) Missiles A1 through A4 were considered in original design. Missiles A5 and A6 were based on the structural adequacy of as-designed structures.

TABLE 3.5-8

TORNADO MISSILE SPECTRUM B
DIESEL GENERATOR BUILDING EQUIPMENT DOORS[1]

Missile	Description	Design Velocity ft/sec (mph)
B1	100-lb missile with 4-in. diameter for impact area	147 (100)
B2	10-ft length of 2 in. standard pipe impact endwise (weight = 36.5 lbs.)	147 (100)
B3	10-ft length of 1/2-in. standard pipe impacting endwise (weight 8.5 lbs.)	147 (100)
B4	Wood plank, 2 in. x 4 in. x 12 ft long, weight 27 lbs.	440 (300)
B5	Cross tie, 7 in. x 9 in. x 8.5 ft long, weight 186 lbs.	440 (300)
B6	Steel pipe, 2-in. diameter by 7 ft long, weight 26 lbs.	147 (100)

Missiles B1, B2, and B3 were considered in the design of the equipment doors. Additional protection is provided for missiles B4, B5, and B6.

Note:

1. See Section 3.5.1.4.

TABLE 3.5-9

TORNADO MISSILE SPECTRUM C
FOR CATEGORY I STRUCTURES[1]

		Design Velocity	
		Exterior Wall ft/s (mi/h)	Roof System ft/s (mi/h)
C1	Wood plank, 4 in. x 12 in. x 12 ft., weight 200 lbs.	368 (251)	294 (200)
C2	Steel pipe, 3-in. diameter, 10-ft long, weight 78 lbs	268 (183)	215 (147)
C3	Steel rod, 1-in. diameter, 3-ft long, weight 8 lbs	259 (177)	207 (141)
C4	Steel pipe, 6-in. diameter, 15-ft long, weight 285 lbs	230 (157)	184 (125)
C5	Steel pipe, 12-in. diameter, 15-ft long, weight 743 lbs	205 (140)	165 (112)
C6	Utility Pole, 13-1/2-in. diameter, 35-ft long, weight 1490 lbs.	241 (164)	205 (140)
C7	Automobile, frontal area 20 ft^2, 4000 lbs, up to 30 ft above grade	100 (70)	80 (56)

NOTE:

(1) See Section 3.5.1.4.

WBN-1

TABLE 3.5-10

DELETED

TABLE 3.5-11

DELETED

TABLE 3.5-12

DELETED

TABLE 3.5-13

DELETED

TABLE 3.5-14
(Sheet 1 of 2)

OUTDOOR SAFETY-RELATED FEATURES(1)
(Including Air Intakes and Exhausts)

Feature	UFSAR Figure	Tornado Protection
1. ERCW pumps, traveling screens, and ancillary features located on the deck of the intake pumping station.	3.8.4-50	Steel grillage roof system consisting of a series of wide flange beams (W21 x 49) spaced 9 inches on center and rotated 45 degrees. Walls of 15 inch reinforced concrete.
2. ERCW Standpipes	2.5-225	Reinforced concrete enclosure (15-inch clear cover)
3. ERCW Overflow Box	3.8.4-46a	Reinforced concrete cover (2-feet thickness)
4. ERCW yard piping	9.2-40	Buried with one of the following minimum covers: 18 inches of concrete, 7 feet of crushed stone, or 11 feet of earthfill.
5. RWST	3.8.4-35	3-1/2 feet high concrete wall with earth backfill surrounding tank to preserve necessary volume of water
6. Pipe tunnels	3.8.4-35	Buried reinforced concrete cover (2-feet thickness)
7. Diesel generator ventilation intakes and exhausts	3.8.4-26	Reinforced concrete canopy (1-foot thickness)
8. Diesel generator engine combustion air exhausts	3.8.4-26	Reinforced concrete curb, 3 feet high and 18 inches thick around the exhaust
9. Diesel generator electric board room air intakes (4 total)	3.8.4-26	Steel canopy with barrier protection (steel frame with 1 inch cover plate).
10. 480V transformer room, ventilation intakes and exhausts	1.2-1	Suspended steel grating (2-1/2 inch x 3/8 inch main bars and 1/2 in diameter connecting bars).
11. Vital battery room exhausts	1.2-1	Same as for Item 10 except the fifth Vital Battery Room is same as Item 9.
12. 480V board room condensing unit air intake and exhaust (train A)	1.2-1	Reinforced concrete roof parapet (a minimum of 12 inches thick and 3 feet high)
13. 480V board room condensing unit air intake (train B)	1.2-1	Reinforced concrete canopy (18 inches thick)

NOTE:

(1) This tabulation consists of these outdoor features which are safety-related in the event of a tornado and as defined in Branch Technical Position ASB3-2 of the Standard Review Plan for Section 3.5.1-4 "Missiles Generated by Natural Phenomena."

TABLE 3.5-14
(Sheet 2 of 2)

OUTDOOR SAFETY-RELATED FEATURES
(Including Air Intakes and Exhausts)

Feature	UFSAR Figure	Tornado Protection
14. 480V board room condensing unit air exhaust (train B)	1.2-1	On roof of mechanical equipment rooms.
15. Class 1E duct banks and manholes	3.8.4-37	Buried with concrete protection (9 inch clear concrete cover for conduits; 12 inches total concrete thickness of manholes).
16. Fuel oil 7-day tank vent line (portion above roof)	3.8.4-26	Encased in reinforced concrete.
17. IPS El. 728' terrace roof openings which expose Cat. I electrical cable trays	3.8.4-50	Steel grating 4'-10" x 4' -10" Borden type C Style 18 size C A36, structural steel shroud (3/4" thickness), 18" thick precast concrete panels.
18. DGB fuel oil 1-day tank vent lines (portions protruding from exterior walls)	3.8.4-27	Steel plate shrouds (2" cover plate thickness)
19. Vacuum Relief Ducts in Aux. Bldg. Roof, Units 1&2	1.2-1	Steel plate canopy (1/2" thickness) over existing duct opening
20. Diesel Generator Bldg. roof access hatches (two)	3.8.4-26	Structural steel plate shrouds under DGB roof. Steel plate thickness varies from 1/4" to 1/2", as required
21. Emergency Pressurizer Fan opening on Cont. Bldg. roof, column lines C10-C11 & n-q	1.2-1	Steel plate canopy (1/2" min. thick.) over opening
22. Toilet and locker room exhaust fan opening on Cont. Bldg. roof, column lines C2-C3 & n-q	1.2-1	Steel plate canopy (1/2" min. thick.) over opening
23. Battery room exhaust vent in Cont. Bldg. roof, column lines C3-C4 & n-q	1.2-1	Steel plate canopy (1/2" min. thick.) over opening

TABLE 3.5-15
UNIT 1 ONLY

DELETED

TABLE 3.5-16
UNIT 1 ONLY

DELETED

TABLE 3.5-17

TORNADO MISSILE SPECTRUM D[1]

Missile Description	Weight (lb)	Cross Section	Length (ft)	Horizontal Velocity[2] (ft/sec)
Wooden Plank	115	4" x 12"	12	272
Steel Rod	9	1" dia	3	167
6" Schedule 40 Pipe	287	6" dia	15	171
12" Schedule 40 Pipe	743	12" dia	15	154
Utility Pole	1124	13-1/2" dia	35	180
Automobile	4000	6.5' x 4.3'	16.5	194

Notes:

(1) For Additional Diesel Generator Building and additional Category I structures after July 1979

(2) Vertical velocities of 70% of the postulated horizontal velocities are acceptable except for the 1 inch steel rod which shall have a vertical velocity equal to its horizontal velocity (167 ft/sec). These missiles are capable of striking in any horizontal or downward direction and at all elevations.

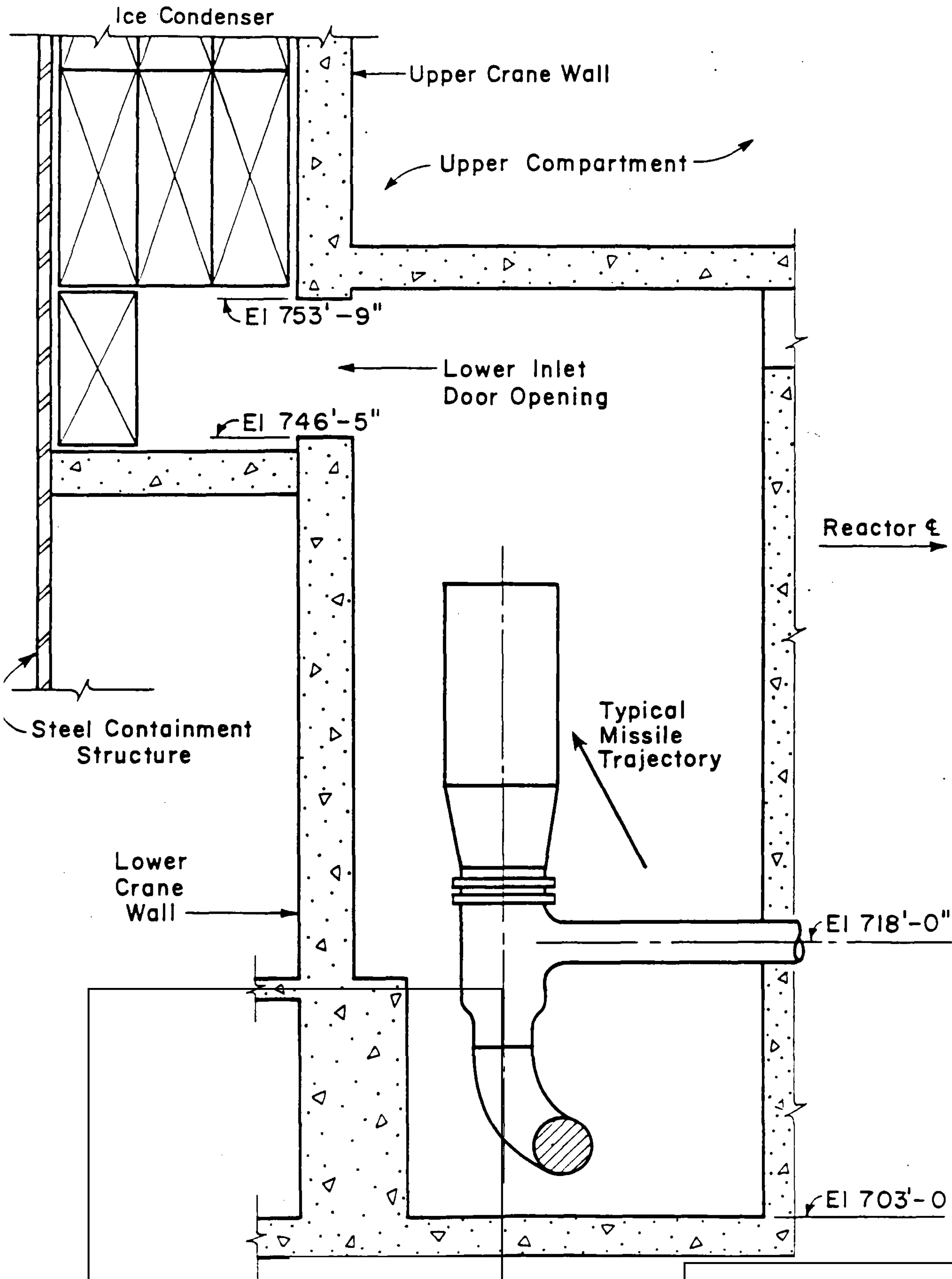
TABLE 3.5-18

Minimum Wall and Roof Thickness Requirements
To Resist the Effects of Tornado Missile Impact[(1)]

Tornado Intensity Region	28-Day Concrete Strength (PSI)	Wall Thickness (Inches)	Roof Thickness (Inches)
Region I	3000	23	18
	4000	20	16
	5000	18	14

NOTE:

(1) For the Additional Diesel Generator Building and additional Category I structures added after July 1979.



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Ice Condenser Lower Inlet Door Opening, Typical Missile Trajectory Orientation

FIGURE 3.5-1

FIGURE 3.5-2

DELETED

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Physical Dimensions of
Important Potential
Turbine Missiles

FIGURE 3.5-2

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.5-3

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Physical Dimensions of Important Potential Turbine Missiles FIGURE 3.5-3

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.5-4

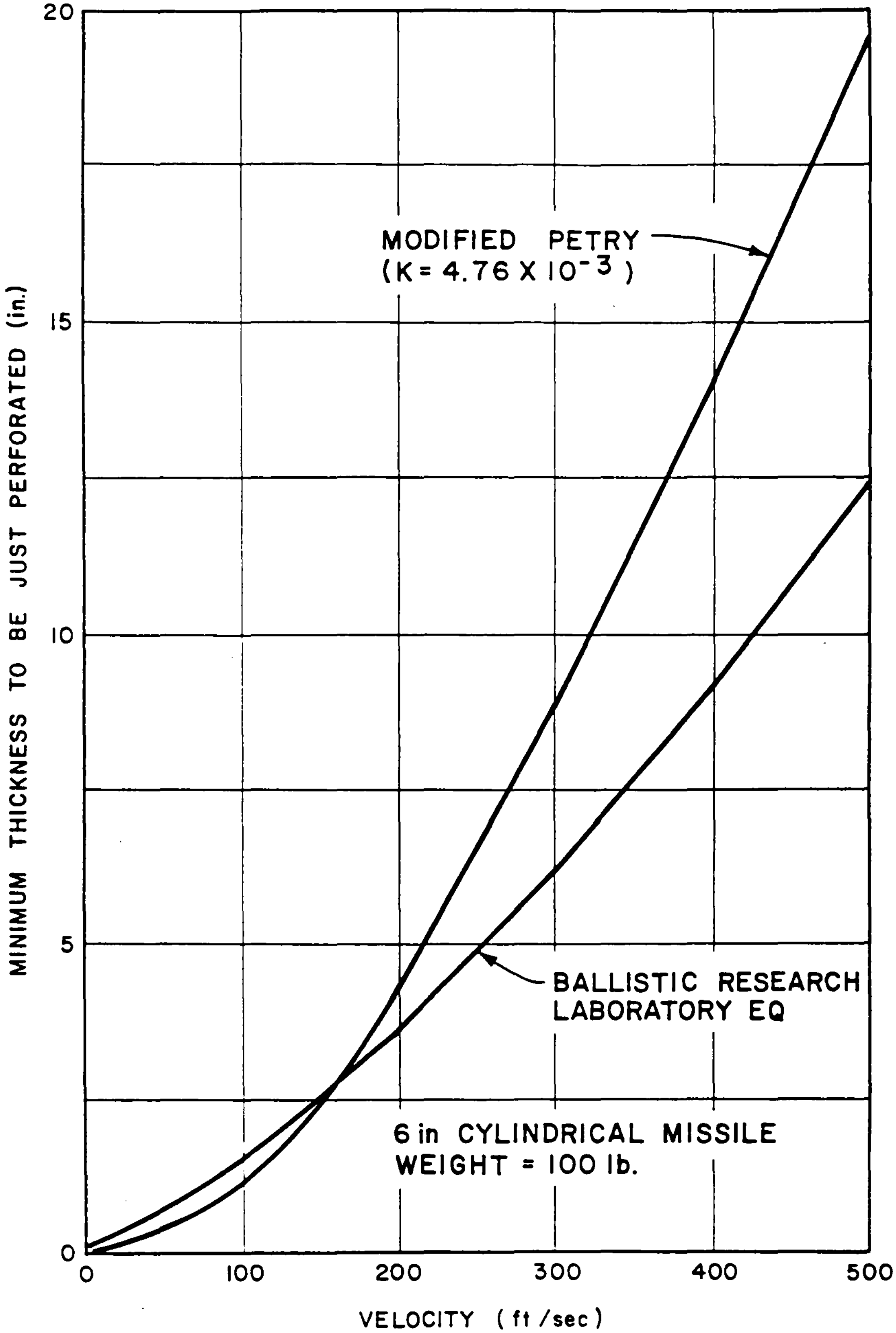
FIGURE 3.5-5

DELETED

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Cross-Sectional Analysis of
Susceptibility of Critical
Components to Upward Turbine Missile
Trajectories

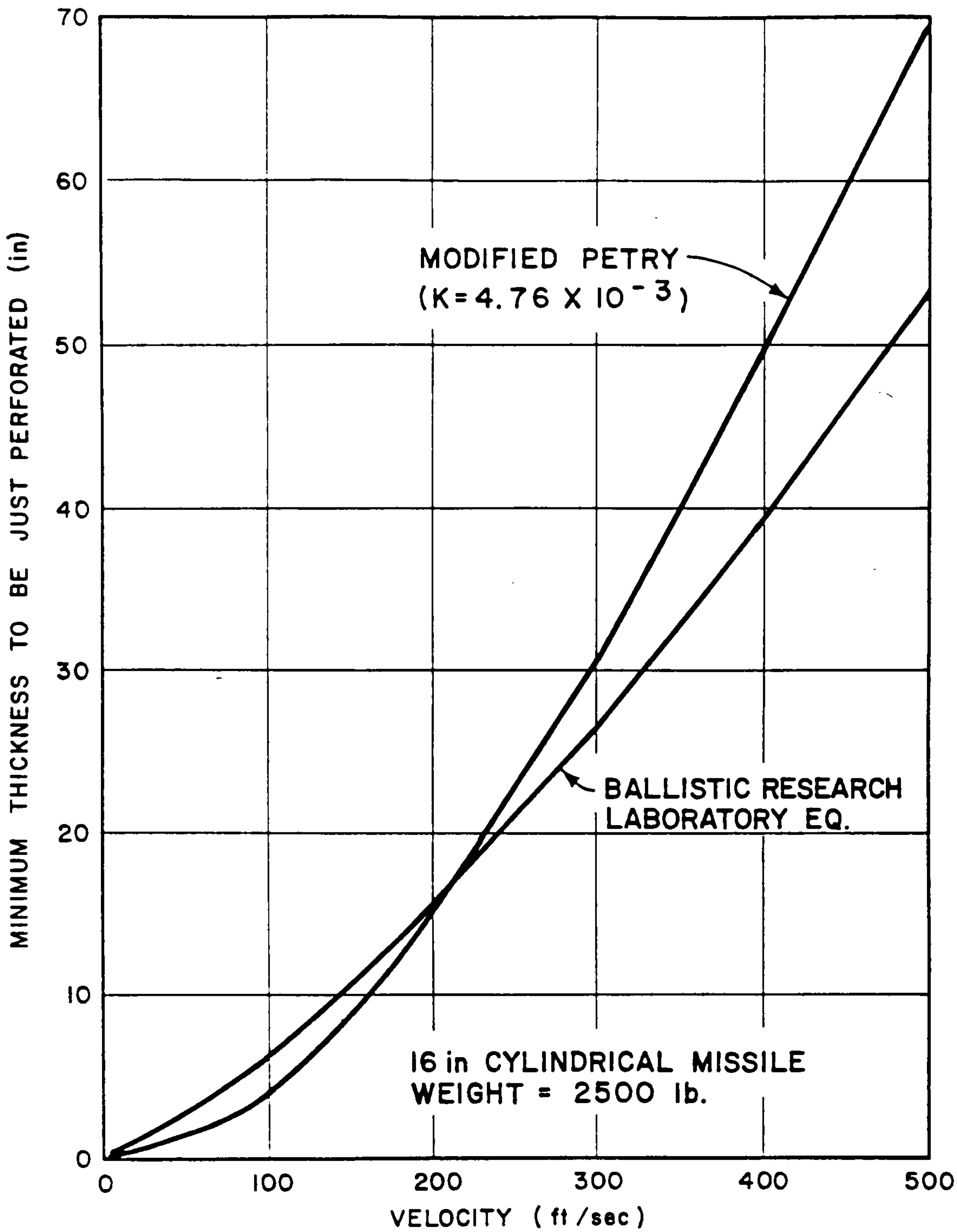
FIGURE 3.5-5



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Comparison of Missile Formulas

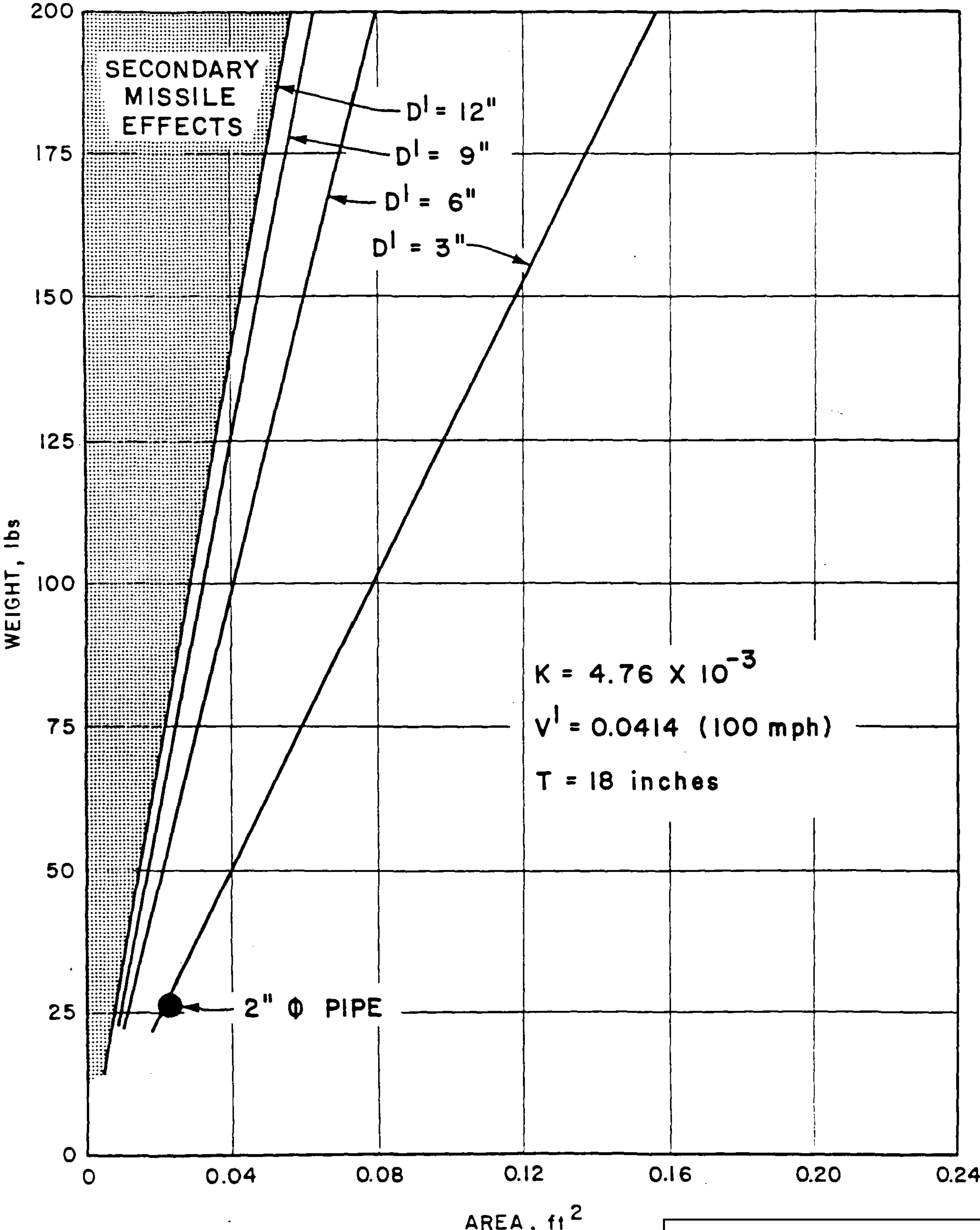
FIGURE 3.5-6



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Comparison of Missile Formulas

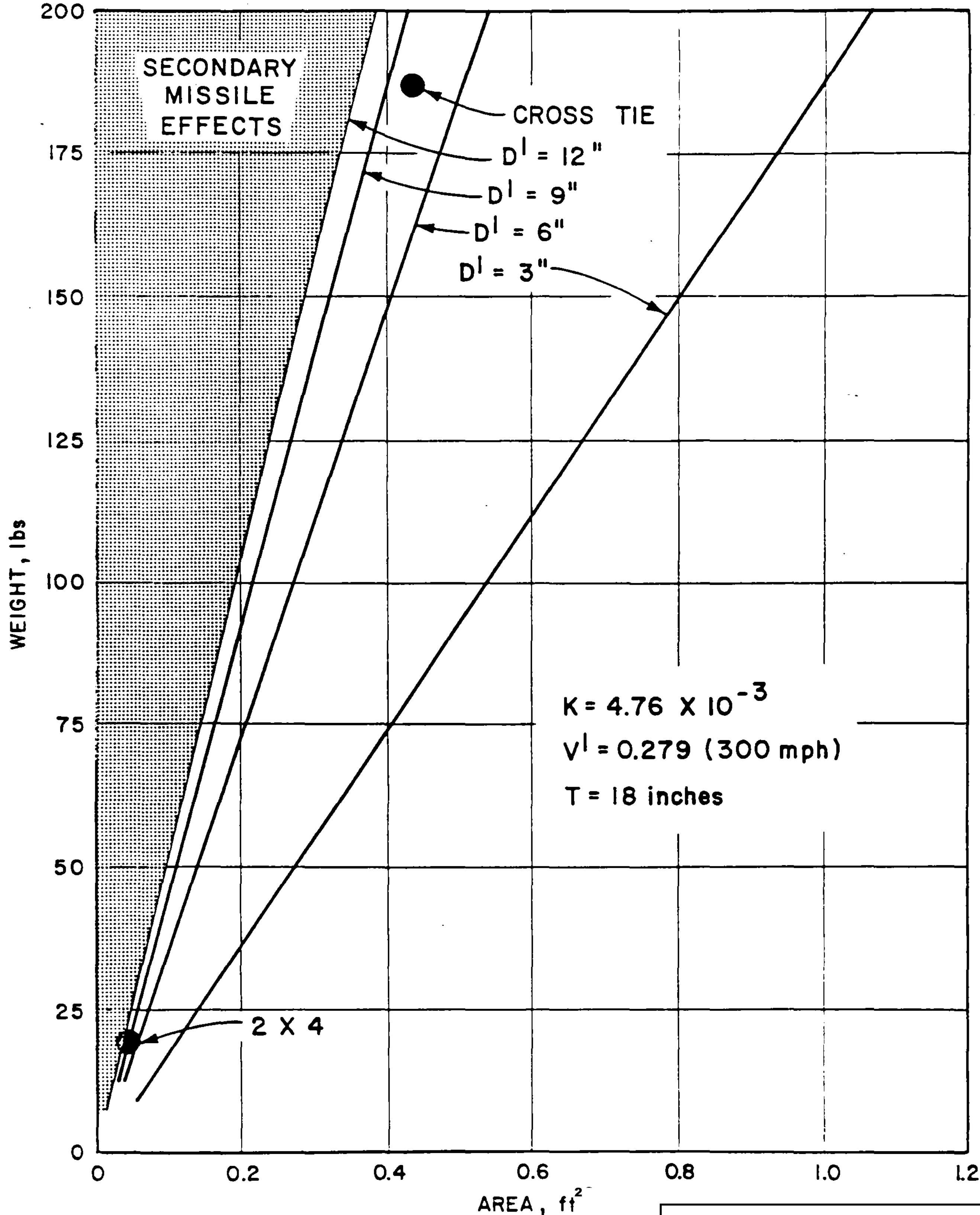
FIGURE 3.5-7



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Depth of Missile Penetration
for Tornado

FIGURE 3.5-8



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Depth of Missile Penetration for Tornado

FIGURE 3.5-9

FIGURE 3.5-10

DELETED

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LP Disc Missiles

FIGURE 3.5-10

WBN-1

FIGURE 3.5-11

DELETED

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

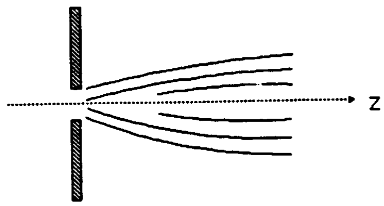
LP Cylinder & Blade Ring Fragments

FIGURE 3.5-11

APPENDIX 3.5A

ESTIMATES OF VELOCITIES OF JET PROPELLED MISSILES

A. Jet Stream Relations



For steady flow, assuming bulk properties across a cross section of a "free jet," the following conservation relations hold between the orifice and any downstream position.

Continuity in axial direction:

1) $W = G_o A_o = GA = \rho V_z A$

From conservation of axial momentum:

2) $$T_j = P_o A_o + \frac{\dot{W}}{g} V_o = PA + \frac{\dot{W}}{g} V_z = (P_o + \frac{G_o V_o}{g}) A_o$$

Where,

W = mass discharge rate

A = area of jet stream

V_z = axial fluid velocity

P = fluid pressure

G = mass flux rate

ρ = fluid density

T_j = total jet thrust

$$G_o = 0.61\sqrt{2\rho g(P_r - P_o)}$$

$$V_o = \sqrt{\frac{2g(P_r - P_o}{\rho}}$$

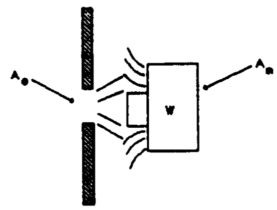
P_o = orifice pressure

P_r = reservoir pressure

Subscript o refers to orifice conditions.

These relations indicate that mass flow rate and axial jet force are constant at any downstream plane even though pressure, area, density, and fluid velocity change from one downstream position to another.

B. Missile Acceleration



Assuming that fluid impinging on missile imparts all of its axial momentum to the missile and splashes radially out of the stream, the force balance on the missile is:

3) $$\frac{W}{g}\frac{dV_m}{dt} = f(z)[P + \frac{G}{g}(V_z - V_m)]A\,2$$

Where,

f(z) = represents the fraction of the jet impinging

W = missile weight

V_m = missile velocity

For no expansion of jet, f(z) = 1, $A = A_o$ ($A_m \geq A_o$ is assumed)

C. Effect on Jet Radial Expansion

For the general case, the missile would not receive the full thrust of the jet throughout its travel, and an estimate of f(z) with downstream distance is needed. For a first approximation, it is assumed that the jet expands with a constant half angle and that the thrust fraction, f(z), impinging on the missile is:

$f(z) = 1$ for $A_m \geq A(z)$

$f(z) = A_m/A(z)$ for $A_m \leq A(z)$

Where,

$A(z) = \pi(R_o + Z \tan \beta)^2$

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

General Design Criterion 4 of Appendix A to 10 CFR 50 requires that structures, systems, and components important to plant safety be protected from the dynamic effects of a pipe rupture. This section of the UFSAR describes the design measures necessary to ensure compliance with this requirement. This section is subdivided into Part A and Part B. To be consistent with the standard format, all sections and subsection numbers are suffixed with either A or B.

Part A (3.6A) includes all piping systems inside and outside containment except the reactor coolant loop piping. The reactor coolant branch lines, however, are within the scope of this part. Also, jet impingement considerations of the reactor coolant loop on components other than those associated with the primary loop are within the scope of this report.

Part B (3.6B) includes the reactor coolant loop system except as stated in 3.6A.

Table 3.6-1
Unit 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
MAIN STEAM LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-2	1-MS-1	MS1-B0-1	32116	37800
		MS1-B0-2	29055	37800
		MS1-B0-3	16925	37800
		MS1-B0-4N	*	*
		MS1-B0-5N	*	*
		MS1-B0-6	30092	37800
3.6-3	1-MS-2	MS2-B0-1	45990	37800
		MS2-B0-2	29432	37800
		MS2-B0-3	26230	37800
		MS2-B0-4N	*	*
		MS2-B0-5N	*	*
		MS2-B0-6	25175	37800
3.6-4	1-MS-3	MS3-B0-1	29145	37800
		MS3-B0-2	26407	37800
		MS3-B0-3N	*	*
		MS3-B0-4N	*	*
		MS3-B0-5	27766	37800
		MS3-B0-6	26658	37800
3.6-5	1-MS-4	MS4-B0-1	31076	37800
		MS4-B0-2	30605	37800
		MS4-B0-3	14347	37800
		MS4-B0-4N	*	*
		MS4-B0-5N	*	*
		MS4-B0-6	25726	37800

Note: All breaks are circumferential ruptures.

* Branch connection to be supported in accordance with alternate analysis criteria. Stresses are not available.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL

Table 3.6-1
Unit 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
MAIN STEAM LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-2	1-MS-1	MS1-B0-1**	32116	37800
		MS1-B0-2**	29055	37800
		MS1-B0-3	16925	37800
		MS1-B0-4N	*	*
		MS1-B0-5N	*	*
		MS1-B0-6	30092	37800
3.6-3	1-MS-2	MS2-B0-1	45990	37800
		MS2-B0-2**	29432	37800
		MS2-B0-3**	26230	37800
		MS2-B0-4N	*	*
		MS2-B0-5N	*	*
		MS2-B0-6	25175	37800
3.6-4	1-MS-3	MS3-B0-1	29145	37800
		MS3-B0-2**	26407	37800
		MS3-B0-3N	*	*
		MS3-B0-4N	*	*
		MS3-B0-5	27766	37800
		MS3-B0-6**	26658	37800
3.6-5	1-MS-4	MS4-B0-1**	31076	37800
		MS4-B0-2**	30605	37800
		MS4-B0-3	14347	37800
		MS4-B0-4N	*	*
		MS4-B0-5N	*	*
		MS4-B0-6	25726	37800

Note: All breaks are circumferential ruptures.

* Branch connection stresses were not available.

** Breaks selected to satisfy the minimum of intermediate breaks are no longer required.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL

Table 3.6-2
UNIT 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
FEEDWATER LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-6	1-FW-1	FW1-B0-1	9651	32400
		FW1-B0-2	16919	32400
		FW1-B0-3N	*	*
		FW1-B0-4N	*	*
		FW1-B0-5N	*	*
		FW1-B0-6	24434	32400
		FW1-B0-7	23499	32400
		FW9-B0-1N	*	*
		FW9-B0-4N	*	*
3.6-7	1-FW-2	FW2-B0-1	24519	32400
		FW10-B0-1N	*	*
		FW2-B0-2	16306	32400
		FW2-B0-3N	*	*
		FW10-B0-4N	*	*
		FW2-B0-5N	*	*
		FW2-B0-6	24748	32400
		FW2-B0-7	23130	32400
3.6-8	1-FW-3	FW3-B0-1	22658	32400
		FW3-B0-2	15181	32400
		FW3-B0-3	13407	32400
		FW3-B0-4N	*	*
		FW11-B0-4N	*	*
		FW3-B0-6N	*	*
		FW11-B0-1N	*	*
		FW3-B0-7	19775	32400
3.9-9	1-FW-4	FW4-B0-1	9787	32400
		FW4-B0-2	12898	32400
		FW4-B0-3N	*	*
		FW4-B0-4N	*	*
		FW4-B0-5N	*	*
		FW4-B0-6	18915	32400
		FW4-B0-7	18203	32400
		FW12-B0-1N	*	*
		FW12-B0-4N	*	*

Note: All breaks are circumferential ruptures.

* Branch connection to be supported in accordance with alternate analysis criteria. Stresses are not available.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF VA’S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

Table 3.6-2
UNIT 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
FEEDWATER LINES

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)
3.6-6	1-FW-1	FW1-B0-1	9651	32400
		FW1-B0-2**	16919	32400
		FW1-B0-3N	*	*
		FW1-B0-4N	*	*
		FW1-B0-5N	*	*
		FW1-B0-6**	24434	32400
		FW1-B0-7	23499	32400
		FW9-B0-1N	*	*
		FW9-B0-4N	*	*
3.6-7	1-FW-2	FW2-B0-1	24519	32400
		FW10-B0-1N	*	*
		FW2-B0-2**	16306	32400
		FW2-B0-3N	*	*
		FW10-B0-4N	*	*
		FW2-B0-5N	*	*
		FW2-B0-6	24748	32400
		FW2-B0-7	23130	32400
3.6-8	1-FW-3	FW3-B0-1	22658	32400
		FW3-B0-2**	15181	32400
		FW3-B0-3**	13407	32400
		FW3-B0-4N	*	*
		FW11-B0-4N	*	*
		FW3-B0-6N	*	*
		FW11-B0-1N	*	*
		FW3-B0-7	19775	32400
3.6-9	1-FW-4	FW4-B0-1	9787	32400
		FW4-B0-2**	12898	32400
		FW4-B0-3N	*	*
		FW4-B0-4N	*	*
		FW4-B0-5N	*	*
		FW4-B0-6	18915	32400
		FW4-B0-7	18203	32400
		FW12-B0-1N	*	*
		FW12-B0-4N	*	*

Note: All breaks are circumferential ruptures.

* Branch connection stresses were not available.

** Breaks selected to satisfy the minimum of intermediate breaks are no longer required.

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

TABLE 3.6-3
UNIT 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
AUXILIARY FEEDWATER SYSTEM STEAM SUPPLY LINE

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE 1)
3.6-10	1-AFD-8	AFD8-BO-1	18146	32400	C
(Unit 1)	1-AFD-7	AFD7-BO-1	19515	32400	C
	1-AFD-7	AFD7-B1-2X	(Note 2)		C
	1-AFD-9	721	36751	32400	C,L
	1-AFD-9	719	35073	32400	C,L
	1-AFD-9	40	20305	32400	C
	1-AFD-9	1	6642	32400	C
	-	LMM	22438	32400	C
	-	L81 L82	23087	32400	C
	-		23000	32400	C

Notes: 1. C = Circumferential, L = Longitudinal Split
2. Not required to be postulated

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

TABLE 3.6-3
UNIT 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
AUXILIARY FEEDWATER SYSTEM STEAM SUPPLY LINE

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE 1)
3.6-10	1-AFD-8	AFD8-BO-1	18146	32400	C
(Unit 1)	1-AFD-7	AFD7-BO-1	19515	32400	C
	1-AFD-7	AFD7-B1-2X	(Note 2)		C
	1-AFD-9	721	36751	32400	C,L
	1-AFD-9	719	35073	32400	C,L
	1-AFD-9	40**	20305	32400	C
	1-AFD-9	1	6642	32400	C
	-	LMM	22438	32400	C
	-	L81**	23087	32400	C
	-	L82**	23000	32400	C

Notes: 1. C = Circumferential, L = Longitudinal Split
2. Not required to be postulated

** Breaks selected to satisfy the minimum number of intermediate breaks are no longer required.

Reflects Analysis Which Was Current At Time of Amendment 51 Submittal

WBN

TABLE 3.6-3A

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS
FOR AUXILIARY FEEDWATER SYSTEM STEAM SUPPLY LINE

Figure No.	Line No.	Break No.	Combined Stress (psi)	Rupture Stress 0.8 (1.2 S_h + S_a) (psi)
3.6-10	2-AFD-8	AFD8-B0-1	21426	32400
(Unit 2)	2-AFD-8	AFD8-B2-1**	-	32400
	2-AFD-7	AFD7-B0-1	17243	32400
	2-AFD-9	AFD7-B2-2**	16104	32400
	-	2	2199	32400
	-	91	25427	32400
	-	92**	18723	32400
		97**	17702	32400

Note: All breaks are circumferential.

** Breaks selected to satisfy the minimum number of intermediate breaks are no longer required.

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

Table 3.6-4
UNIT 1

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS - SI COLD LEG INJECTION

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE)
3.6-11 (Loop 1)	1-SI-5	SI5-B0-1N	30046	39448	C
		SI5-B0-2N	44038	39448	C,L
		SI5-B0-3	21902	38172	C
	1-SI-506	SI506-B0-1N	31669	37244	C
		SI506-B0-2N	35242	37244	C
		SI506-B0-3N	9354	38172	C
3.6-12 (Loop 4)	1-SI-4	SI4-B0-1N	23807	39448	C
		SI4-B0-2N	41325	39448	C,L
		SI4-B0-3N	27237	38172	C
	1-SI-511	SI511-B0-1N	14013	38172	C
		SI511-B0-2N	42275	37244	C
		SI511-B0-3N	32244	37244	C
3.6-13 (Loop 2)	1-SI-9	SI9-B0-1N	8827	39448	C
		SI9-B0-2	8950	39448	C,L,X
		SI9-B0-3	35707	38172	C
		SI9-B2-5	11998	39448	C,L
	1-SI-507	SI507-B0-1N	2878	37244	C
		SI507-B0-2N	25818	37244	C
		SI507-B0-3N	25815	37244	C
3.6-14 (Loop 3)	1-SI-10	SI10-B0-1N	13377	39448	C
		SI10-B0-3	12237	39448	C,L,X
		SI10-B0-4	15800	39448	C,L
		SI10-B0-5	43670	38172	C
		SI10-B1-6	16297	39448	C,L,X
		SI10-B1-7	11652	39448	C,L,X
	1-SI-510	SI510-B0-1N	45985 (61970)	37244	C
		SI510-B0-2N	43727 (57446)	37244	C
		SI510-B0-3N	41913 (56095)	37244	C

Note: C = Circumferential,
L = Longitudinal split
X = Break not required to be postulated

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

Table 3.6-4
UNIT 2

SUMMARY OF COMBINED STRESSES AT BREAK LOCATIONS - SI COLD LEG INJECTION

FIGURE NO.	LINE NO.	BREAK NO.	COMBINED STRESS (psi)	ALLOWABLE PIPE RUPTURE STRESS $0.8(1.2S_h + S_a)$ (psi)	BREAK TYPE (NOTE)
3.6-11 (Loop 1)	1-SI-5	SI5-B0-1N	30046	39448	C
		SI5-B0-2N	44038	39448	C,L
		SI5-B0-3**	21902	38172	C
	1-SI-506	SI506-B0-1N**	31669	37244	C
		SI506-B0-2N**	35242	37244	C
		SI506-B0-3N**	9354	38172	C
3.6-12 (Loop 4)	1-SI-4	SI4-B0-1N	23807	39448	C
		SI4-B0-2N	41325	39448	C,L
		SI4-B0-3N**	27237	38172	C
	1-SI-511	SI511-B0-1N**	14013	38172	C
		SI511-B0-2N**	42275	37244	C
		SI511-B0-3N**	32244	37244	C
3.6-13 (Loop 2)	1-SI-9	SI9-B0-1N	8827	39448	C
		SI9-B0-2	8950	39448	C,L,X
		SI9-B0-3**	35707	38172	C
		SI9-B2-5	11998	39448	C,L
	1-SI-507	SI507-B0-1N**	2878	37244	C
		SI507-B0-2N**	25818	37244	C
		SI507-B0-3N**	25815	37244	C
3.6-14 (Loop 3)	1-SI-10	SI10-B0-1N	13377	39448	C
		SI10-B0-3	12237	39448	C,L,X
		SI10-B0-4	15800	39448	C,L
		SI10-B0-5**	43670	38172	C
		SI10-B1-6	16297	39448	C,L,X
		SI10-B1-7	11652	39448	C,L,X
	1-SI-510	SI510-B0-1N**	45985 (61970)	37244	C
		SI510-B0-2N**	43727 (57446)	37244	C
		SI510-B0-3N**	41913 (56095)	37244	C

Note: C = Circumferential, L = Longitudinal split X = Break not required to be postulated
Stress shown in parenthesis is Unit 2 stress at the same location

Breaks selected to satisfy the minimum number of intermediate breaks are no longer required. Note that longitudinal splits, although indicated above, were not required to be considered at intermediate locations where the criteria for a minimum number of break locations was applied.

** Class 1 stresses are not shown. Note that in this historic analysis the Class 1 piping was designed to the requirements of subsection C.

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal.

WBN

TABLE 3.6-5
UNIT 1

SUMMARY OF STRESSES AT BREAK LOCATIONS
RHR/SI HOT LEG RECIRCULATION, LOOP 4

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR, U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-15	1-RHR-6	RHR6-B0-1N	59159	9242	24065	0.011	C	A
		RHR6-B0-2N	89977	10427	20308	0.872	C,L	C
		RHR6-B0-3N	93475	14639	10414	0.672	C,L	C
		RHR6-B0-4	91510	14148	14188	0.763	C,L	C
		RHR6-B0-5	77232	14128	20308	0.402	C,L	C
		RHR6-B0-6N	72556	4273	20554	0.302	C,L	C
		RHR6-B0-7N	78263	3018	26597	0.303	C,L	C
		RHR6-B0-8N	77616	15111	13909	0.173	C,L	C
	1-RHR-7	RHR7-B0-1	86735	10297	25388	0.555	C,L	C
		RHR7-B0-2N	93711	23140	24120	0.841	C,L	C
	1-SI-14	SI14-B0-1N	88393	21576	25922	0.432	C	A
		SI14-B0-2N	56917	8574	15679	0.0	C	B

Allowable stress intensity values: $3S_m$ = 58800 psi, $2.4S_m$ = 47040 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III.
2) U is based on NB-3653.5 if $S_n < 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n < 3S_m$, and U > 0.1; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or $S > 2.4S_m$, or U > 0.1
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per Equation 10.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

WBN

TABLE 3.6-5
UNIT 2

SUMMARY OF STRESSES AT BREAK LOCATIONS
RHR/SI HOT LEG RECIRCULATION, LOOP 4

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR, U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-15	1-RHR-6	RHR6-B0-1N	59159	9242	24065	0.011	C	A
		RHR6-B0-2N	89977	10427	20308	0.872	C,L	B
		RHR6-B0-3N	93475	14639	10414	0.672	C,L	B
		RHR6-B0-4	91510	14148	14188	0.763	C,L	B
		RHR6-B0-5	77232	14128	20308	0.402	C,L	B
		RHR6-B0-6N	72556	4273	20554	0.302	C,L	B
		RHR6-B0-7N	78263	3018	26597	0.303	C,L	B
		RHR6-B0-8N	77616	15111	13909	0.173	C,L	B
	1-RHR-7	RHR7-B0-1	86735	10297	25388	0.555	C,L	B
		RHR7-B0-2N	93711	23140	24120	0.841	C,L	B
	1-SI-14	SI14-B0-1N	88393	21576	25922	0.432	C	A
		SI14-B0-2N	56917	8574	15679	0.0	C	C

Allowable stress intensity values: $3S_m$ = 48996-58950 psi, $2.4S_m$ = 39197-47160 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III.
2) U is based on NB-3653.5 if $S_n < 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n < 3S_m$, and U > 0.1; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or S > $2.4S_m$, or U > 0.1
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per Equation 10. (This is no longer required).

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

TABLE 3.6-6
UNIT 1

SUMMARY OF STRESSES AT BREAK LOCATIONS
SI HOT LEG RECIRCULATION < LOOPS 1, 2, AND 3

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-16	1-SI-15	SI15-B0-1N	45624	1788	20504	0.0	C	A
3.6-17	1-RHR-4	RHR4-B0-1	41998	14358	19733	0.0	C	A
	1-RHR-5	RHR5-B0-1	62824	33251	16372	0.006	C	A
		RHR5-B0-2N	58135	33855	15368	0.0	C	C
		RHR5-B0-3	59832	32948	15489	0.0	C	C

Allowable stress intensity values: $3S_m$ = 50592 psi, $2.4S_m$ = 40474 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III
2) U is based on NB-3653.5 if $S_n < 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n < 3S_m$, and U > 0.1; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or $S > 2.4S_m$, or U > 0.1
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per equation 10

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF TVA'S LETTER TO NRC DATED MAY 9, 1984 (FSAR AMENDMENT 51)

TABLE 3.6-6
UNIT 2

SUMMARY OF STRESSES AT BREAK LOCATIONS
SI HOT LEG RECIRCULATION < LOOPS 1, 2, AND 3

FIGURE NO.	LINE NO.	BREAK NO.	EQUATION 10[1] S_n (psi)	EQUATION 12[1] S_e (psi)	EQUATION 13[1] S (psi)	USAGE FACTOR U[2]	BREAK TYPE[3]	LOCATION CRITERIA[4]
3.6-16	1-SI-15	SI15-B0-1N	45624	1788	20504	0.0	C	A
3.6-17	1-RHR-4	RHR4-B0-1	41998	14358	19733	0.0	C	A
	1-RHR-5	RHR5-B0-1	62824	33251	16372	0.006	C	A
		RHR5-B0-2N	58135	33855	15368	0.0	C	C
		RHR5-B0-3	59832	32948	15489	0.0	C	C

Allowable stress intensity values: $3S_m$ = 50592 psi, $2.4S_m$ = 40474 psi

Notes: 1) For equation number, refer to NB-3600 of the ASME Code Section III
2) U is based on NB-3653.5 if $S_n \leq 3S_m$; or NB-3653.6 if $S_n > 3S_m$
3) C = Circumferential break; L = Longitudinal split
4) Location criteria: A - Terminal end
B - $S_n \leq 3S_m$, and U > 0.1; or $S_n > 3S_m$ and $S_e > 2.4S_m$, or $S > 2.4S_m$, or U > 0.1
C - Break selected to satisfy the minimum number of intermediate breaks based on highest stress intensity per equation 10, (this is no longer required).

Reflects Analysis Which Was Current at Time of Amendment 51 Submittal

TABLE 3.6-7

AND

TABLE 3.6-8

DELETED

TABLE 3.6-9 (Sheet 1 of 3)

SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4]

MAIN STEAM

PIPING SYSTEM Main Steam PIPING NOMINAL DIA. 36 inch PIPING SCHEDULE 1.307-inch wall

BREAK LOCATION	BREAK TYPE	THRUST[1] DIRECTION	WHIP FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
201 301 203 303	C	Downstream	Yes	Pipe whip into refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
				Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
205, 305, 207, 307	C	Downstream	Yes	Pipe whip into refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
				Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
		Upstream	Yes	Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
209, 309	C	Downstream	Yes	Pipe whip into refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
				Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
		Upstream	Yes	Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
211, 311	C	Downstream	Yes	Pipe whip damage to A1 wall of Auxiliary Building	Unacceptable, wall fails, environmental damage to essential components will result from steam entering Auxiliary Building	Restraints H31W, H21W
		Upstream	Yes	Jet impingement on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank
217, 317	C	Downstream	Yes	Pipe whip damage to A1 wall of Auxiliary Building	Unacceptable, wall fails, environmental damage to essential components will result from steam entering Auxiliary Building	Restraints H31W, H21W
		Upstream	Yes	Pipe impact on refueling water storage tank	Unacceptable, results in loss of emergency core cooling water supply	Provide a bunker to prevent unacceptable damage to tank

See Sheet 3 for Notes

TABLE 3.6-9 (Sheet 2 of 3)
SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4] (cont'd)

MAIN STEAM

PIPING SYSTEM Main Steam PIPING NOMINAL DIA. 36 inch PIPING SCHEDULE 1.307-inch wall

BREAK LOCATION	BREAK TYPE[1]	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
217, 317	L	Up	Yes	Jet impingement on ceiling of counting room and radio-chemical laboratory (unit 1 only)	Unacceptable, ceiling fails, environmental damage to essential components due to steam entering elevation 713 of the Auxiliary Building	Sleeves S217 S317
	L	Down	Yes	Pipe impact on ceiling of counting room and radio-chemical laboratory (unit 1 only)	Unacceptable, ceiling fails, environmental damage to essential components due to steam entering elevation 729 of the Auxiliary Building	Sleeves S217 S317
		Left	Yes	Pipe whip into Auxiliary Building HVAC intake	Unacceptable, environmental damage to essential components due to steam entering elevation 737 of the Auxiliary Building	Sleeves S217, S317
		Right	Yes	Jet impingement on Auxiliary Building HVAC intake	Unacceptable, environmental damage to essential components due to steam entering elevation 737 of the Auxiliary Building	Sleeves S217, S317
218, 318	C	Upstream	Yes	Pipe whip damage to south wall of south steam valve room	Unacceptable, damage to main steam and feedwater isolation valves located in valve room	Restraints G22W, G32W
418, 118, 218, 318	L	Right	Yes	Jet impingement on spreading room exhaust duct in C11-wall (unit 2 only)	Unacceptable, loss of Control Building habitability due to steam environment	HVAC to have 3 psi backdraft damper installed to prevent steam from entering control building
419, 119, 219, 319	C	Upstream	Yes	Pipe impact on elevation 755, which supports control room HVAC equipment	Unacceptable, floor fails and results in environmental damage to control room	Restraints L42D, L12D, L22D, L32D
420, 120, 220, 320	C	Upstream	Yes	Pipe impact on elevation 755, which supports control room HVAC equipment	Unacceptable, floor fails and results in environmental damage to control room	Restraints L42D, L12D, L22D, L32D
423 124	L	Up	Yes	Pipe impact on elevation 729 floor of Turbine Building adjacent to doors to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125
423, 124	L	Down	Yes	Jet Impingement on elevation 729 floor of Turbine Building adjacent to doors to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125

See Sheet 3 for Notes

TABLE 3.6-9 (Sheet 3 of 3)
SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4] (cont'd)

MAIN STEAM

PIPING SYSTEM Main Steam   PIPING NOMINAL DIA. 36 inch   PIPING SCHEDULE 1.307-inch wall

BREAK LOCATION	BREAK TYPE	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
424, 125, 225, 325	C	Upstream	Yes	Pipe impact on the N-wall of Control Building	Unacceptable, wall fails, environmental damage to essential components within Control Building	Restraints M42S, M32S, M22S, M12S
424, 125	L	Up	Yes	Pipe impact on elevation 729 floor of Turbine Building adjacent to door to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125
424, 125	L	Down	Yes	Jet impingement on elevation 729 floor of Turbine Building adjacent to door to Control Building	Unacceptable, floor fails, possible damage to essential components within Control Building due to jet/missile impingement on Control Building doors	Sleeves S424, S125
425, 126, 226, 326	C	Upstream	Yes	Pipe impact on N-wall of Control Building	Unacceptable, wall fails, environmental damage to essential components in Control Building	Restraints M42S, M12S, M22S, M32S
	Through-Wall Leakage Cracks					
	Through-wall leakage crack break below Control Building HVAC exhaust ducting at elevation 755 on Q-wall (unit 2 only)			Through-wall leakage crack break would fill control room HVAC with steam	Unacceptable, loss of habitability of control room	HVAC to have 3 psi backdraft damper installed to prevent steam from entering control room
	Through-wall leakage crack break below Auxiliary Building HVAC intake canopy at elevation 743 on A1-wall			Through-wall leakage crack break would fill Auxiliary Building with steam	Unacceptable, environmental damage to essential components in Auxiliary Building	HVAC to have temperature sensors installed which control intake fans, preventing steam from entering Auxiliary Building

In all other cases effects of through-wall leakage crack breaks are acceptable.

Notes:

1) Direction of thrust on pipe. Jet load is opposite.

For circumferential (C) breaks consider upstream thrust on the upstream pipe and downstream thrust on the downstream pipe.
For longitudinal (L) breaks consider up, down, lateral left, and lateral right thrust (facing downstream).

2) Whip trajectory is governed by hinge mechanism and direction of vector thrust of break force. Maximum 180° rotation about any plastic hinge. Sweep of jet is governed by pipe motion.

3) Type of effect (jet, whip, environment, etc.) and components affected.

4) This applies to unit 1. Unit 2 is opposite hand unless otherwise noted.

TABLE 3.6-10 (Sheet 1 of 2)

SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4]

FEEDWATER

PIPING SYSTEM Feedwater PIPING NOMINAL Dia. 18 inch PIPING SCHEDULE 80

BREAK LOCATION	BREAK[1] TYPE	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
418, 118, 218, 318	L	Left	No	Jet impingement on spreading room exhaust duct in C11-wall (unit 2 only)	Unacceptable loss of Control Building ventilation	Provide 3 psi backdraft damper
420, 120, 220, 320	L	Down	Yes	Pipe impact on elevation 708 floor of Control Building	Unacceptable, failure of floor allows pipe whip into electrical board room air handling units below. Environmental damage to essential components and loss of control room habitability may result.	Restraints J42U, J12U, J22U, J32U
421, 121, 221, 321	C	Downstream	Yes	Pipe impact on elevation 708 floor of Control Building	Unacceptable, failure of floor allows pipe whip into electrical board room air handling units below. Environmental damage to essential components and loss of control room habitability may result.	Restraints J42U, J12U, J22U, J32U
422, 122, 222, 322	C	Downstream	Yes	Pipe impact on elevation 708 floor of Control Building	Unacceptable, failure of floor allows pipe whip into electrical board room air handling units below. Environmental damage to essential components & loss of control room habitability may result.	Restraints J42U, J12U, J22U, J32U
423, 123, 223, 323	C	Upstream	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W, K12W, K22W, K32W
424, 124, 224, 324	C	Upstream	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W, K12W, K22W, K32W
425, 125, 225, 325	C	Downstream	Yes	Pipe impact on elevation 755 floor	Unacceptable, floor fails and results in environmental damage to control room.	Restraints K42W, K12W, K22W, K32W
	L	Down	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W, K12W, K22W, K32W

See Notes on Sheet 2

TABLE 3.6-10 (Sheet 2 of 2)

SUMMARY OF PROTECTION REQUIREMENTS - OUTSIDE CONTAINMENT[4]

FEEDWATER

PIPING SYSTEM Feedwater PIPING NOMINAL Dia. 18 inch PIPING SCHEDULE 80

BREAK LOCATION	BREAK[1] TYPE	THRUST[1] DIRECTION	WHIP[2] FORMED	EFFECT ON REQUIRED[3] COMPONENTS	ACCEPTABLE/UNACCEPTABLE	REQUIRED FIX
426, 126, 226, 326	C	Downstream	Yes	Pipe impact on elevation 755 floor	Unacceptable, floor fails and results in environmental damage to control room.	Restraints K42W, K12W, K22W, K32W
	L	Down	Yes	Pipe impact on C-3 wall of Control Building	Unacceptable, failure of wall allows pipe whip into spreading room of Control Building. Environmental damage to essential components and loss of control room habitability may result.	Restraints K42W. K12W, K22W, K32W
428, 128, 228, 328	C	Downstream	Yes	Pipe impact on elevation 755 floor	Unacceptable, floor fails and results in environmental damage to Control Building.	Restraints K42W, K12W, K22W, K32W
	Through-Wall Leakage					
	Through-wall leakage crack break below Control Building HVAC exhaust ducting at elevation 755 on Q-wall (unit 2 only)			Through-wall leakage crack break would fill control room HVAC with steam	Unacceptable, loss of habitability of control room	HVAC to have 3 psi backdraft damper installed to prevent steam entering control room
	Through-wall leakage crack break below Auxiliary Building HVAC intake canopy at elevation 743 on A1-wall			Through-wall leakage crack break would fill Auxiliary Building HVAC with steam	Unacceptable, environmental damage to essential components in Auxiliary Building	HVAC to have temperature sensors installed, which control intake fans, preventing steam from entering Auxiliary Bldg.
	In all other cases, effects of through-wall leakage crack breaks are acceptable.					

Notes:

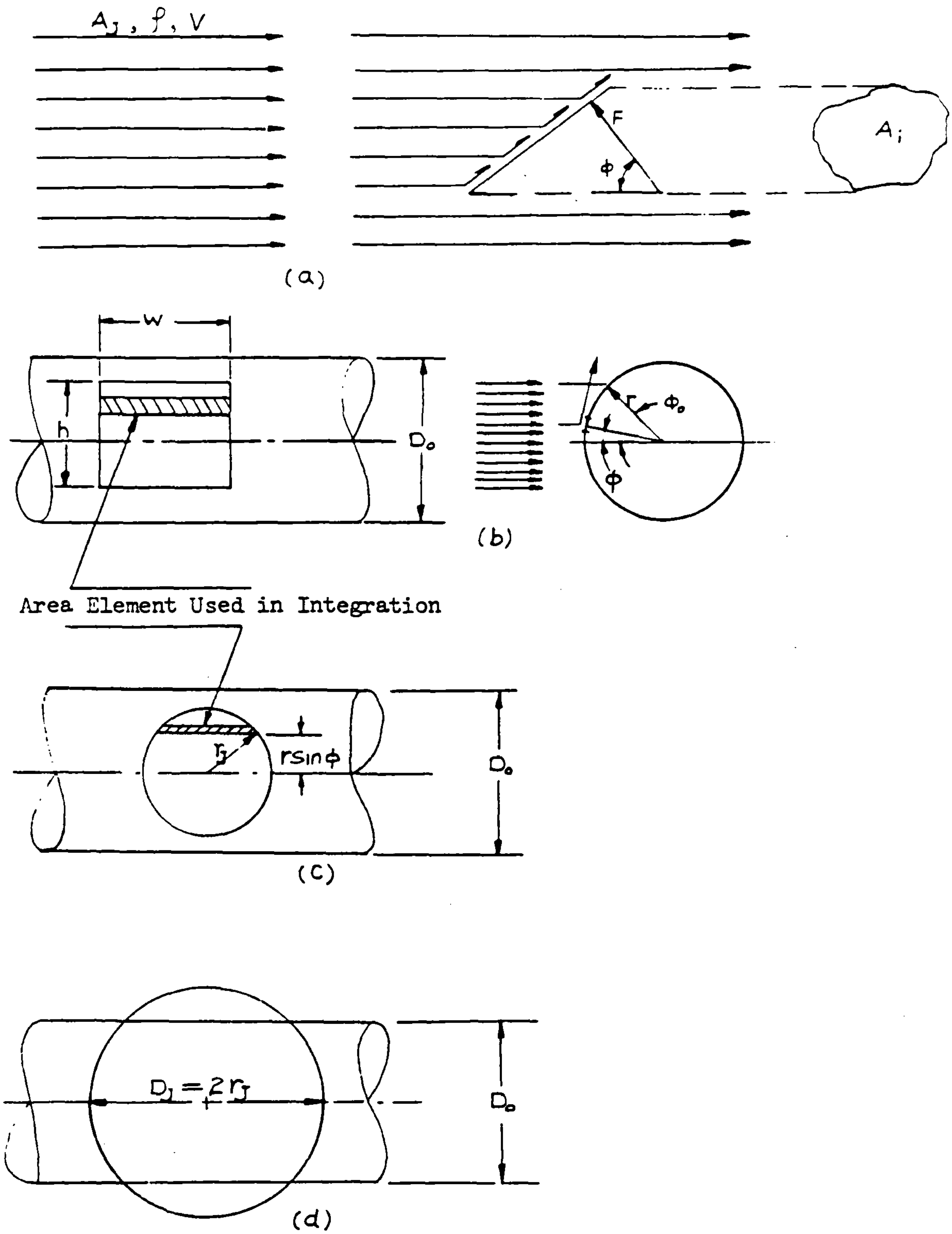
1) Direction of thrust on pipe. Jet load is opposite.
For circumferential (C) breaks consider upstream thrust on the upstream pipe and downstream thrust on the downstream pipe.
For longitudinal (L) breaks consider up, down, lateral left, and lateral right thrust (facing downstream).

2 Whip trajectory is governed by hinge mechanism and direction of vector thrust of break force. Maximum 180° rotation about any plastic hinge. Sweep of jet is governed by pipe motion.

3) Type of effect (jet, whip, environment, etc.) and components affected.

4) This applies to Unit 1. Unit 2 is opposite hand unless otherwise noted.

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL

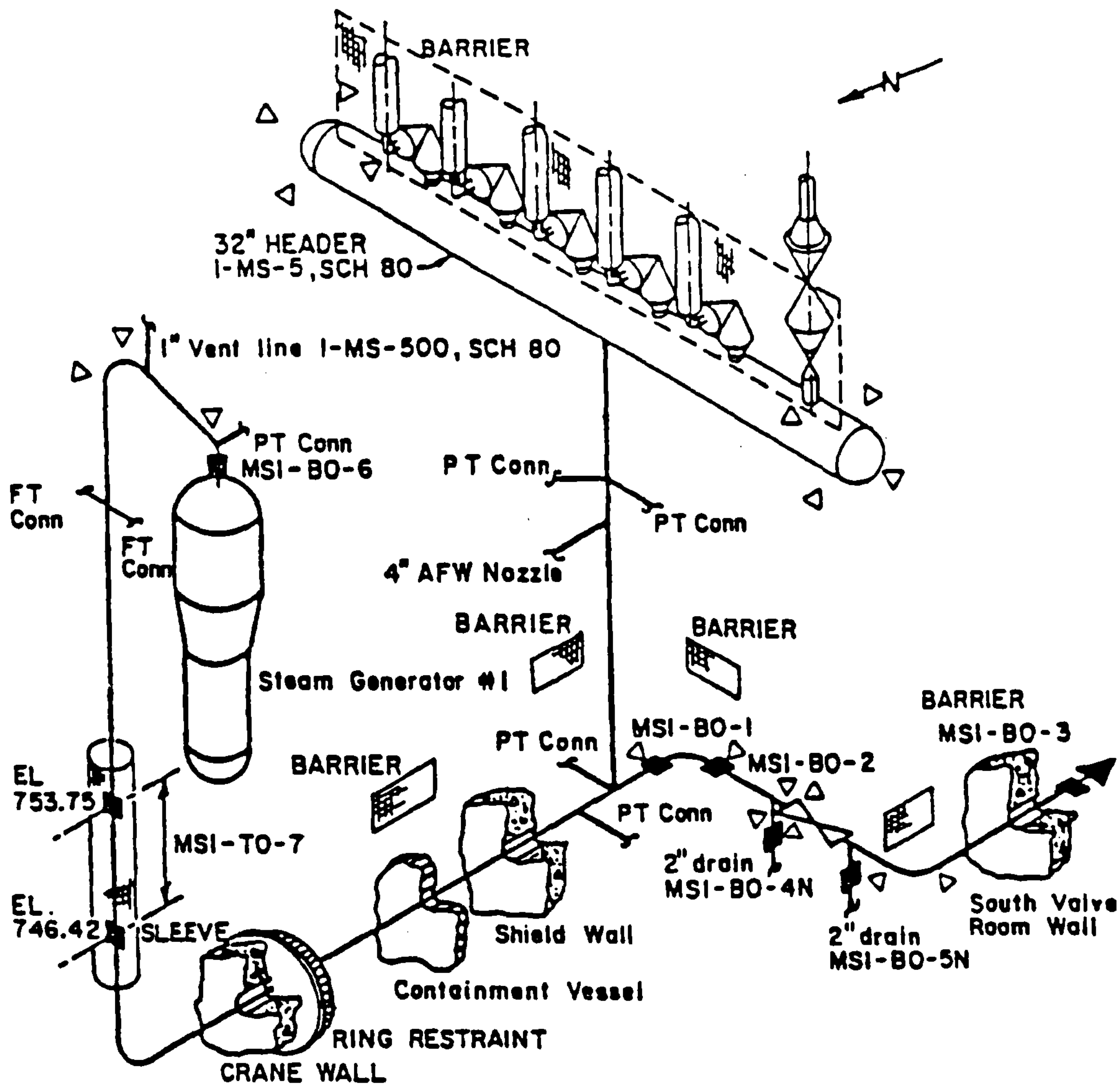


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Shape Factors

FIGURE 3.6-1

MAIN STEAM LINE FROM
STEAM GENERATOR #1



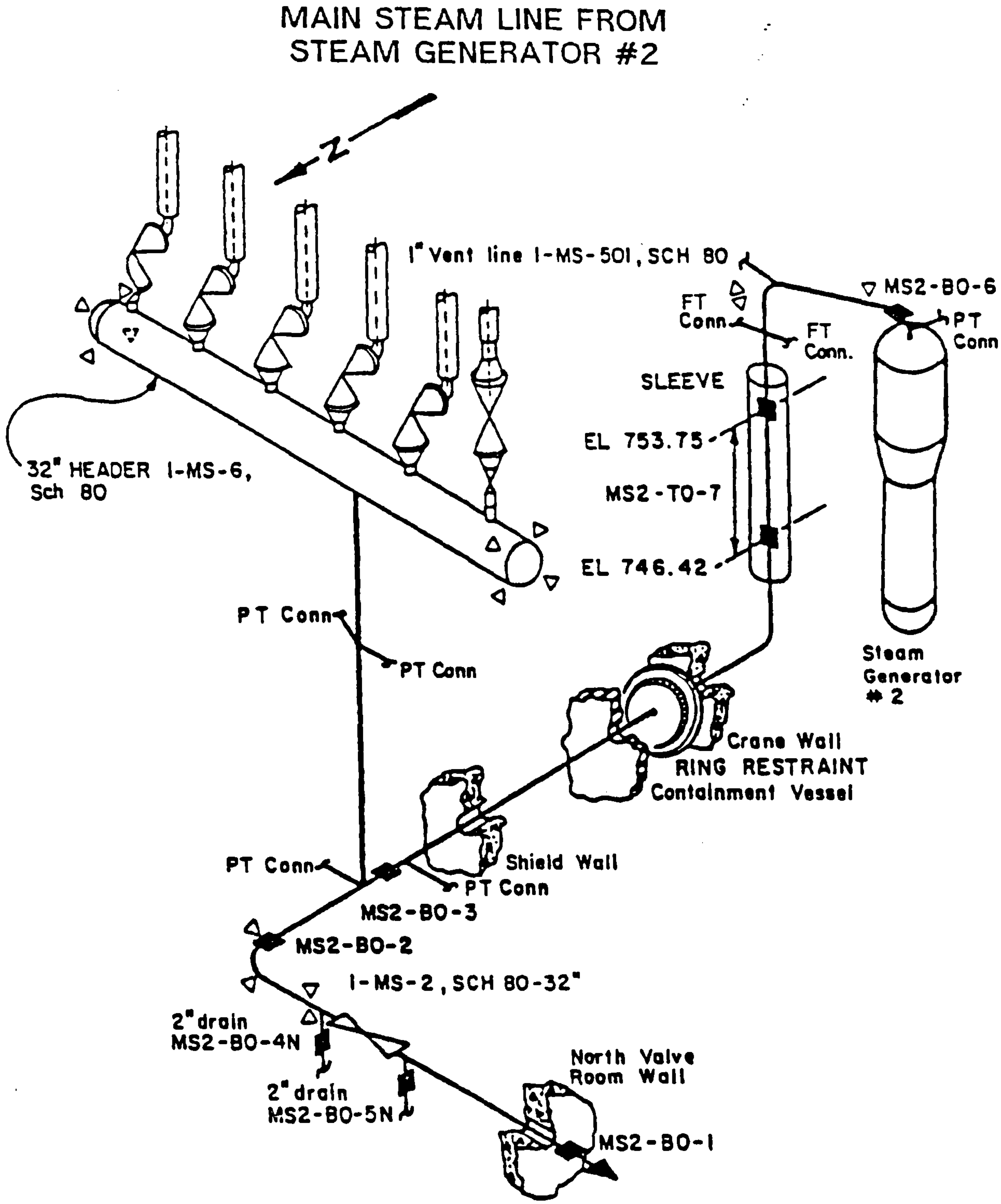
▷ DENOTES DIRECTION & LOCATION OF PROTECTIVE DEVICE RUPTURE RESTRAINTS, TYPICAL ALL FIGURES

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

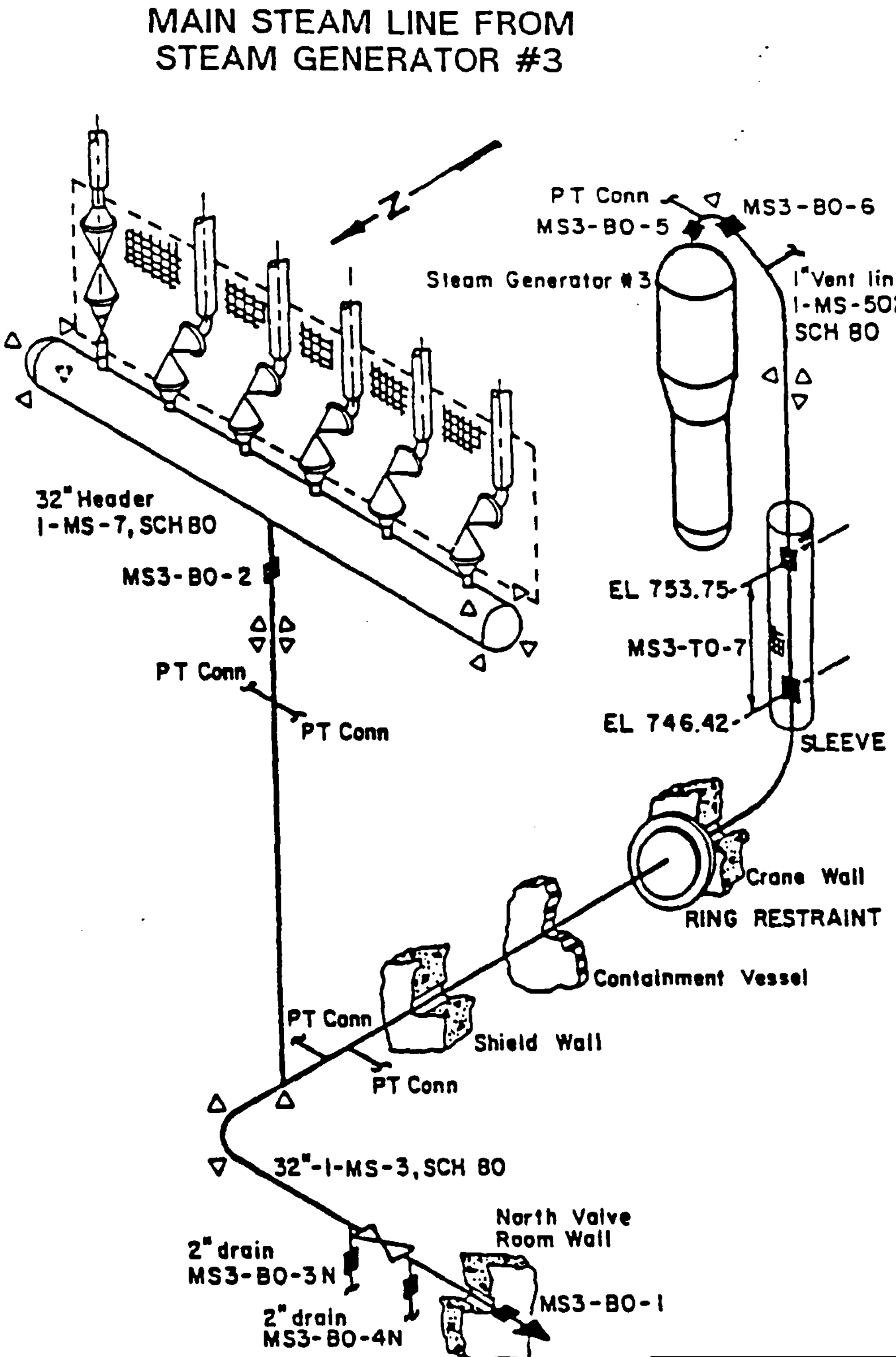
Isometric of Postulated
Break Locations

FIGURE 3.6-2



REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-3



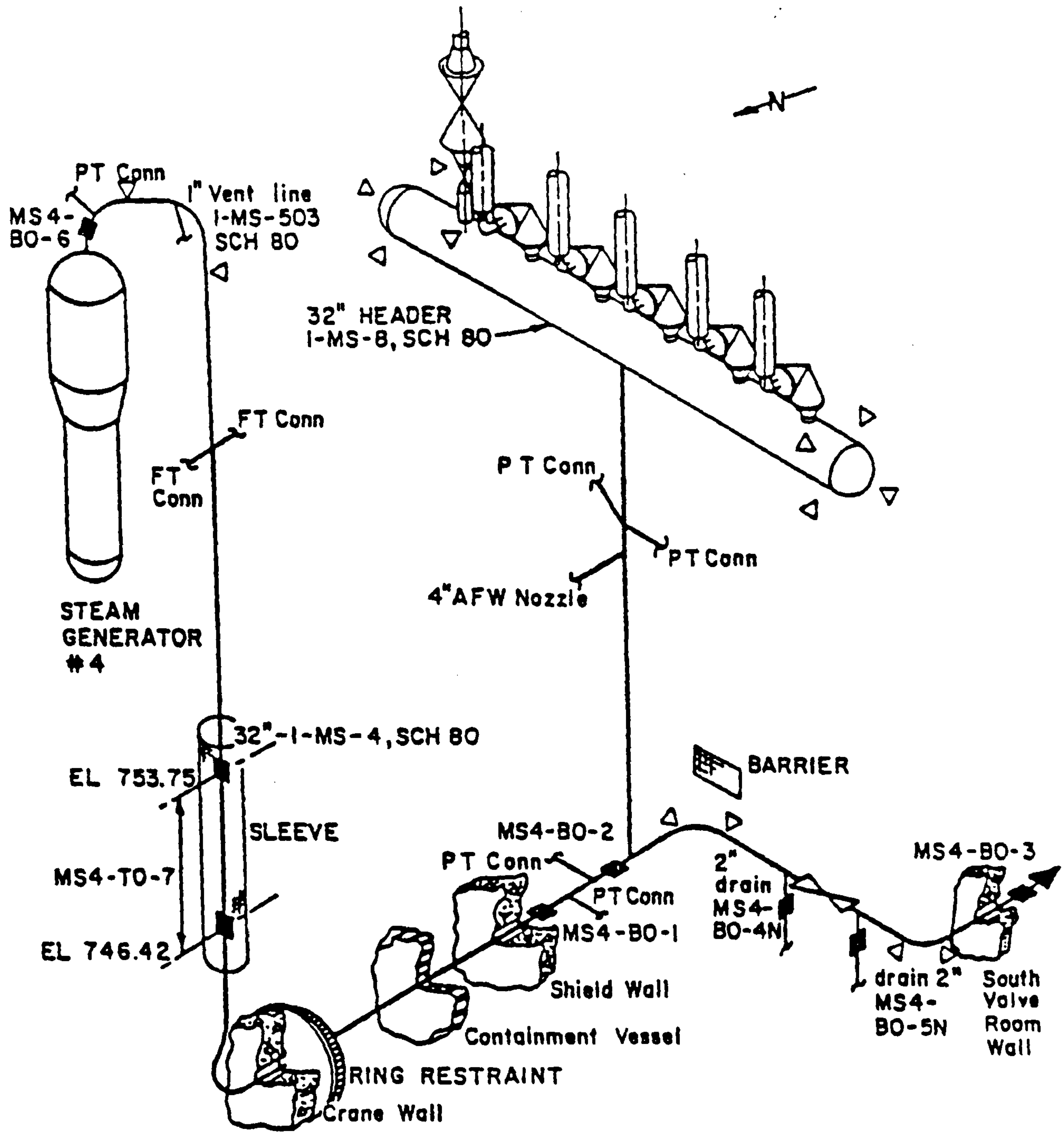
REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Isometric of Postulated
Break Locations

FIGURE 3.6-4

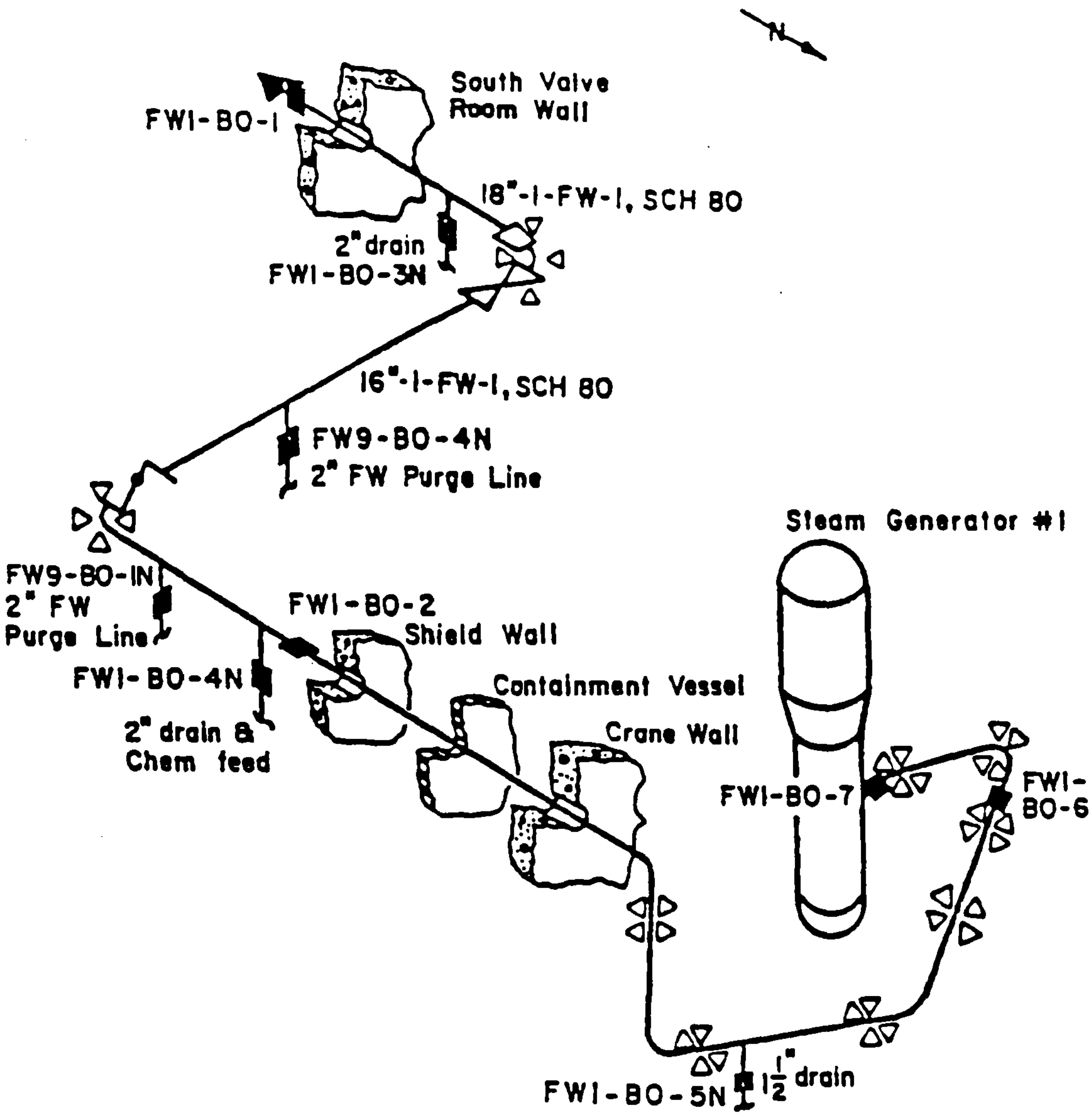
MAIN STEAM LINE FROM STEAM GENERATOR #4



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-5

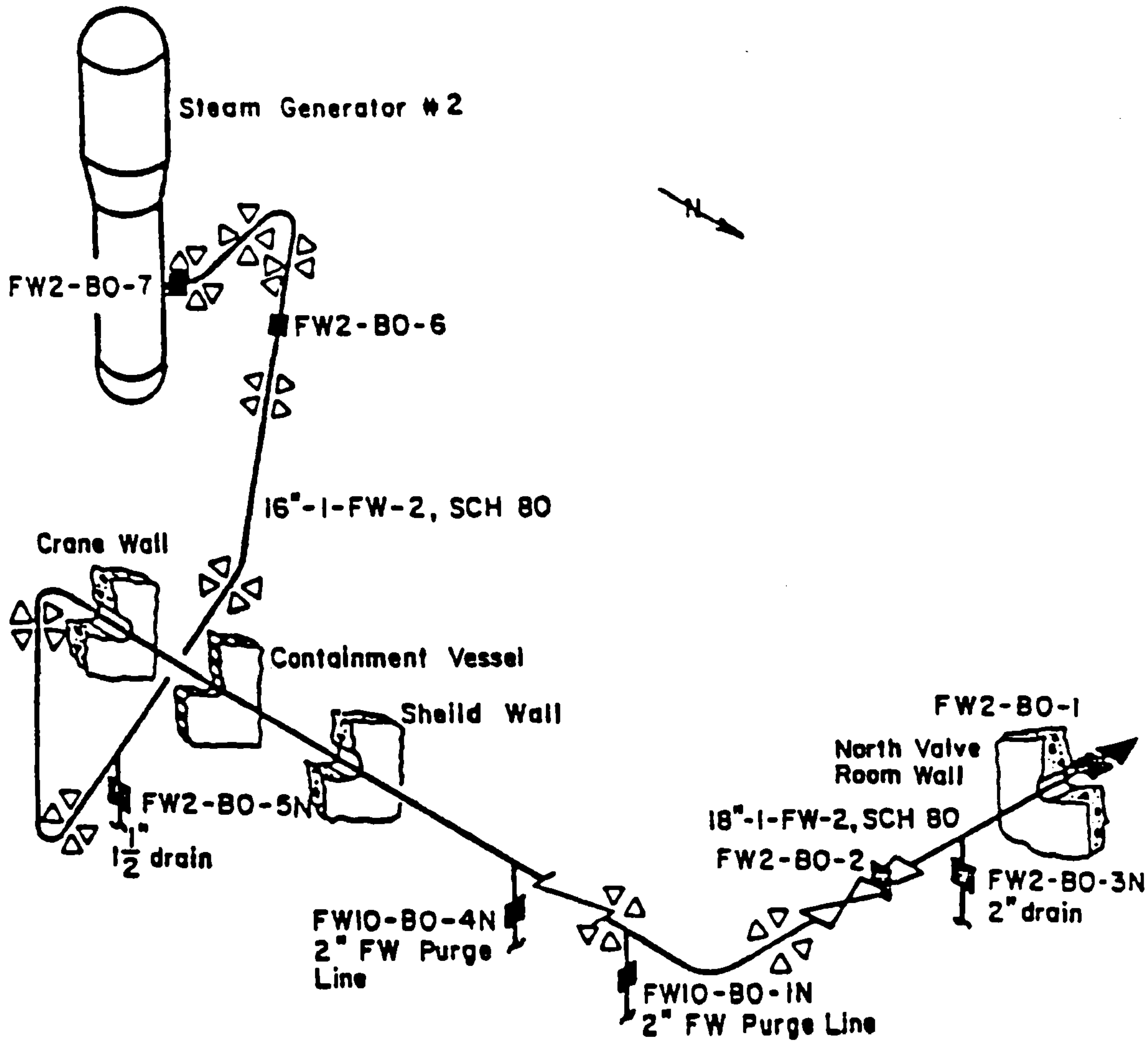
FEEDWATER LINE TO STEAM GENERATOR #1



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-6

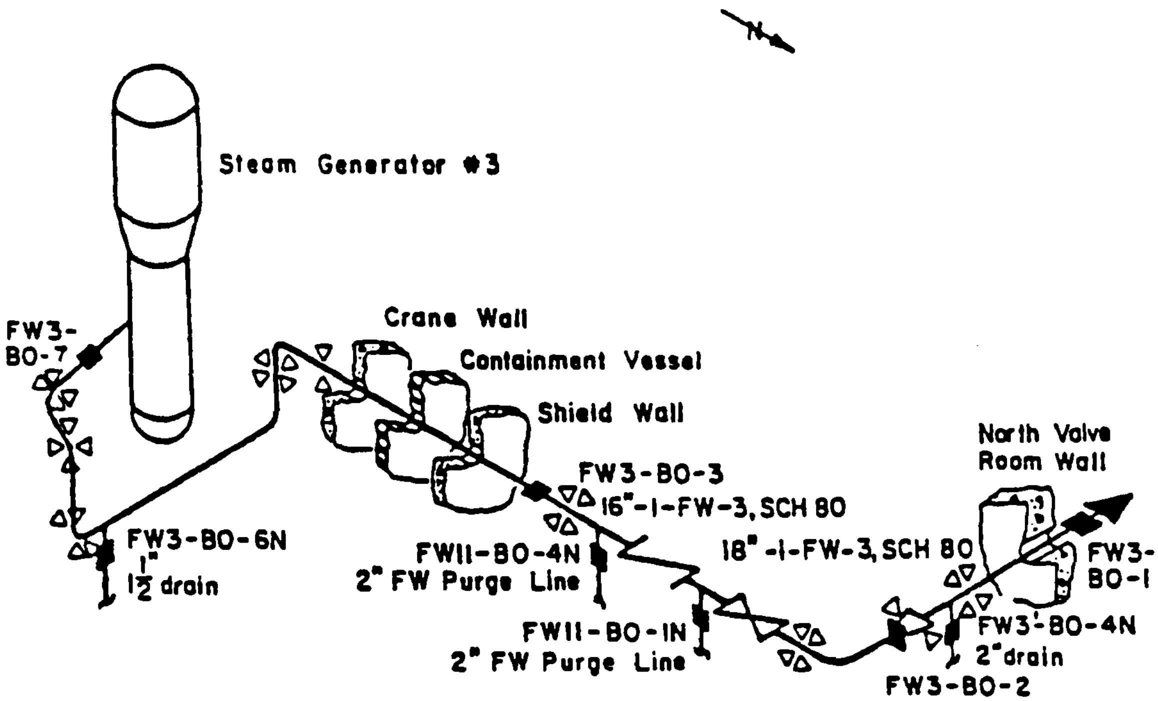
FEEDWATER LINE TO STEAM GENERATOR #2



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

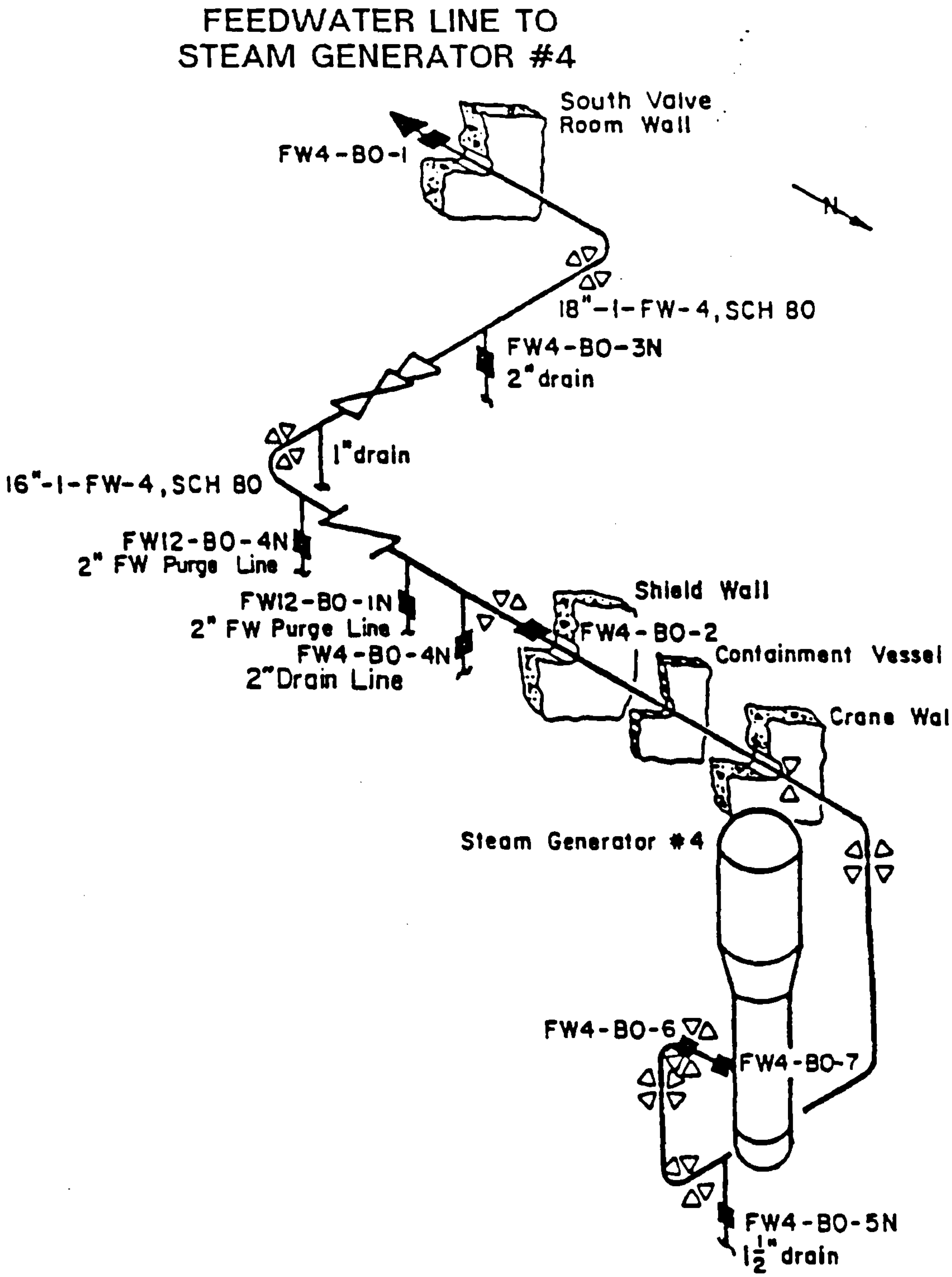
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-7

FEEDWATER LINE TO STEAM GENERATOR #3



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

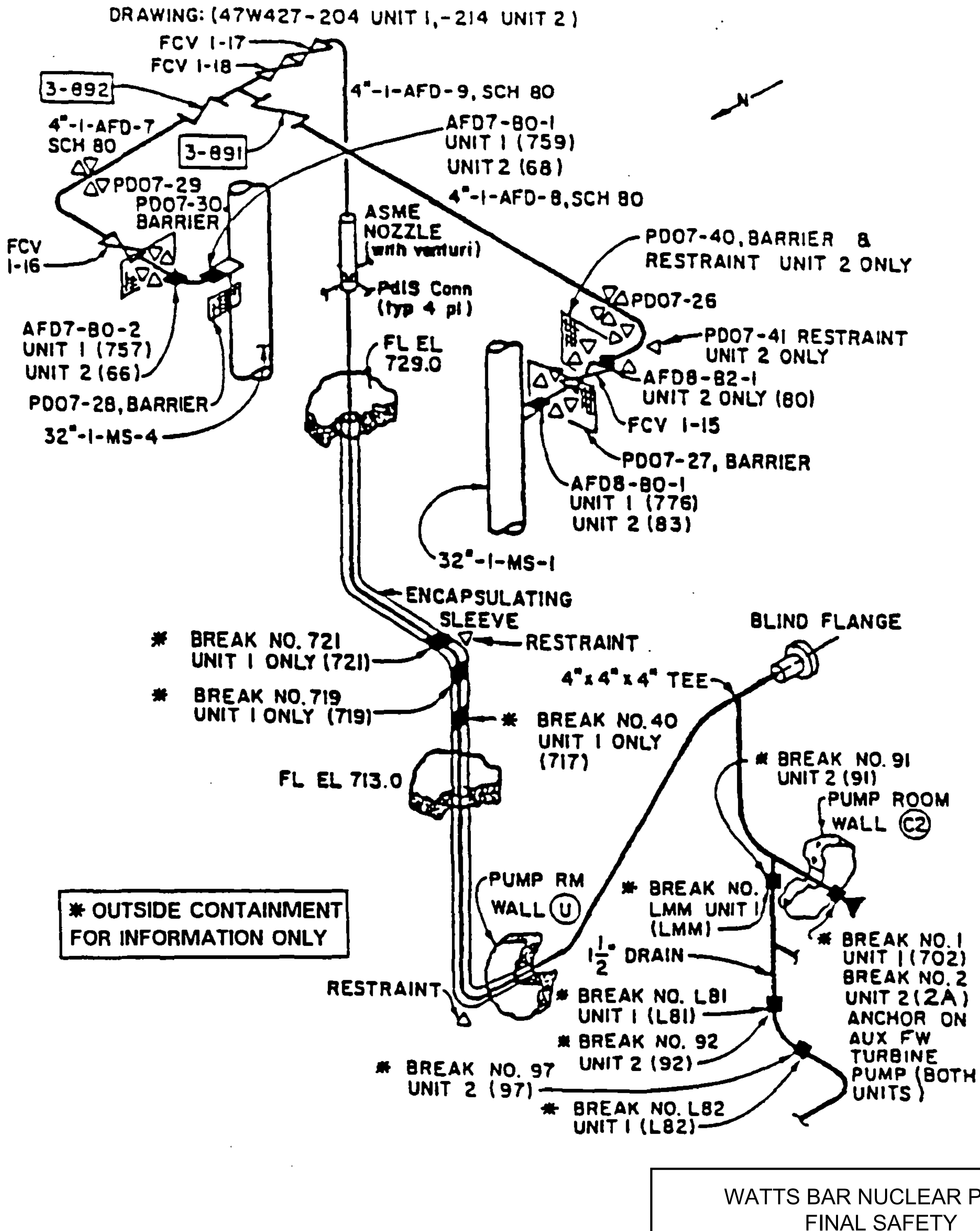
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-8



REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-9

AUXILIARY FEEDWATER STEAM SUPPLY LINES

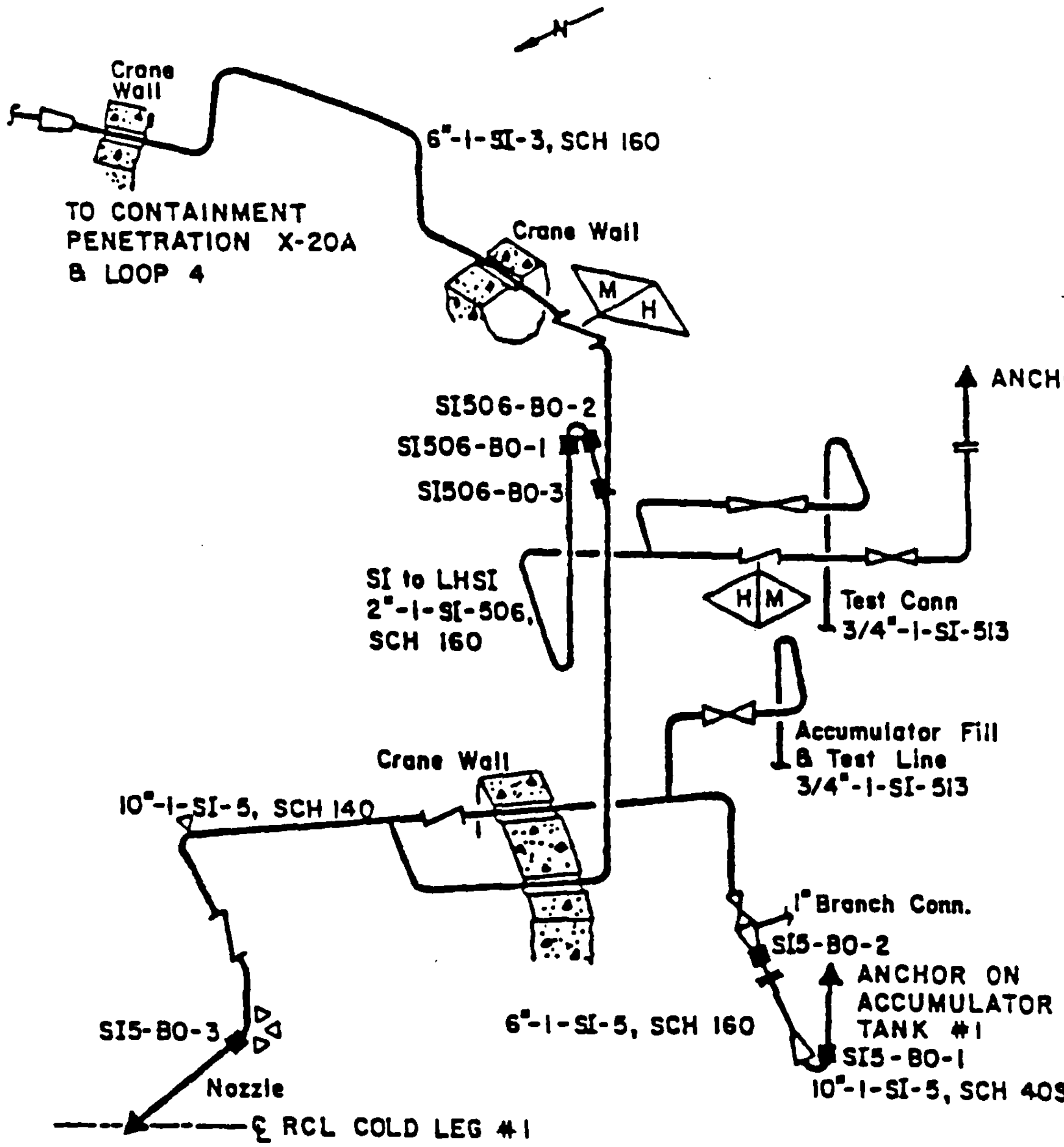


REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 62 SUBMITTAL.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FIGURE 3.6-10

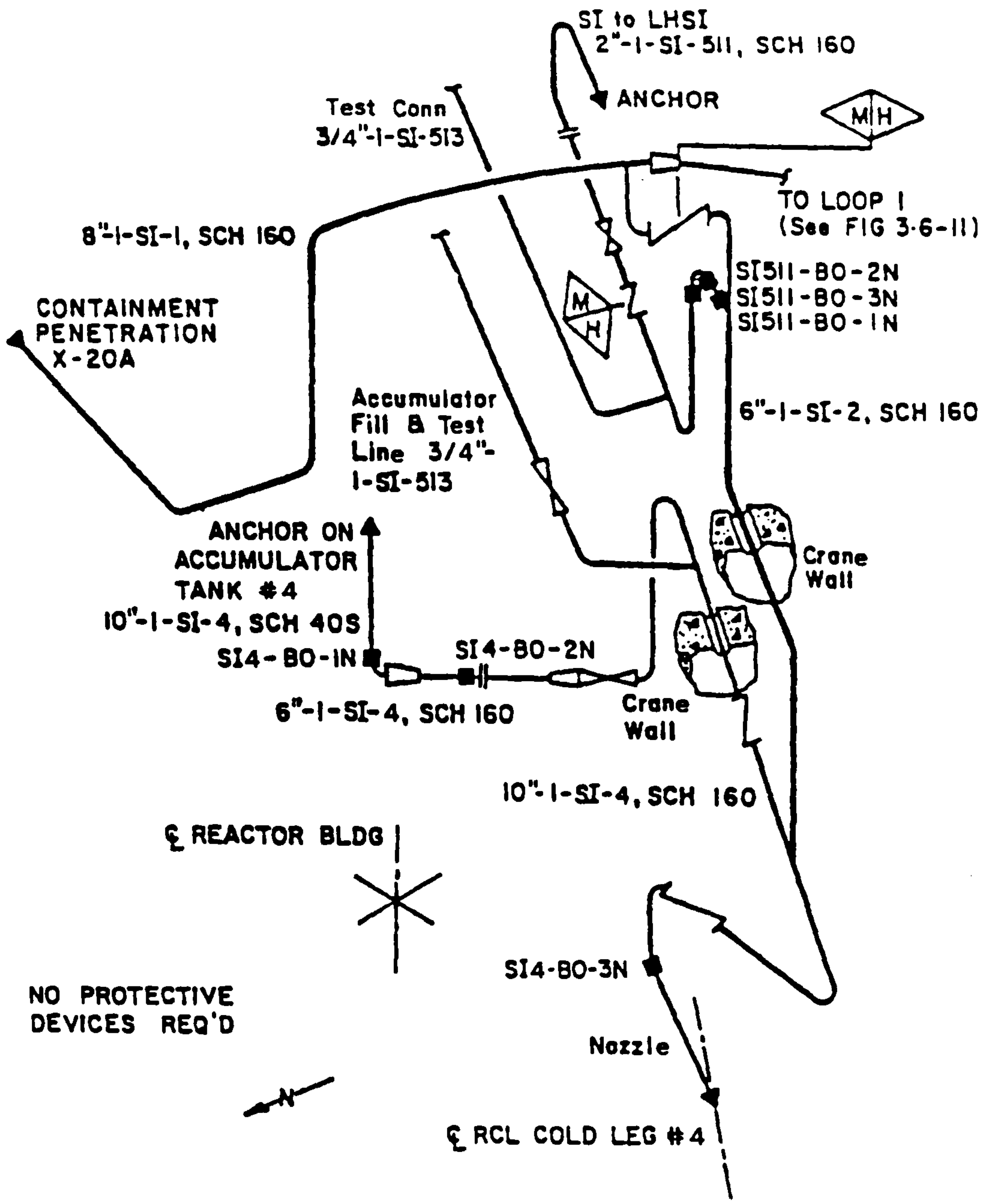
SI COLD LEG INJECTION
LOOP 1



REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
FIGURE 3.6-11

SI COLD LEG INJECTION
LOOP 4

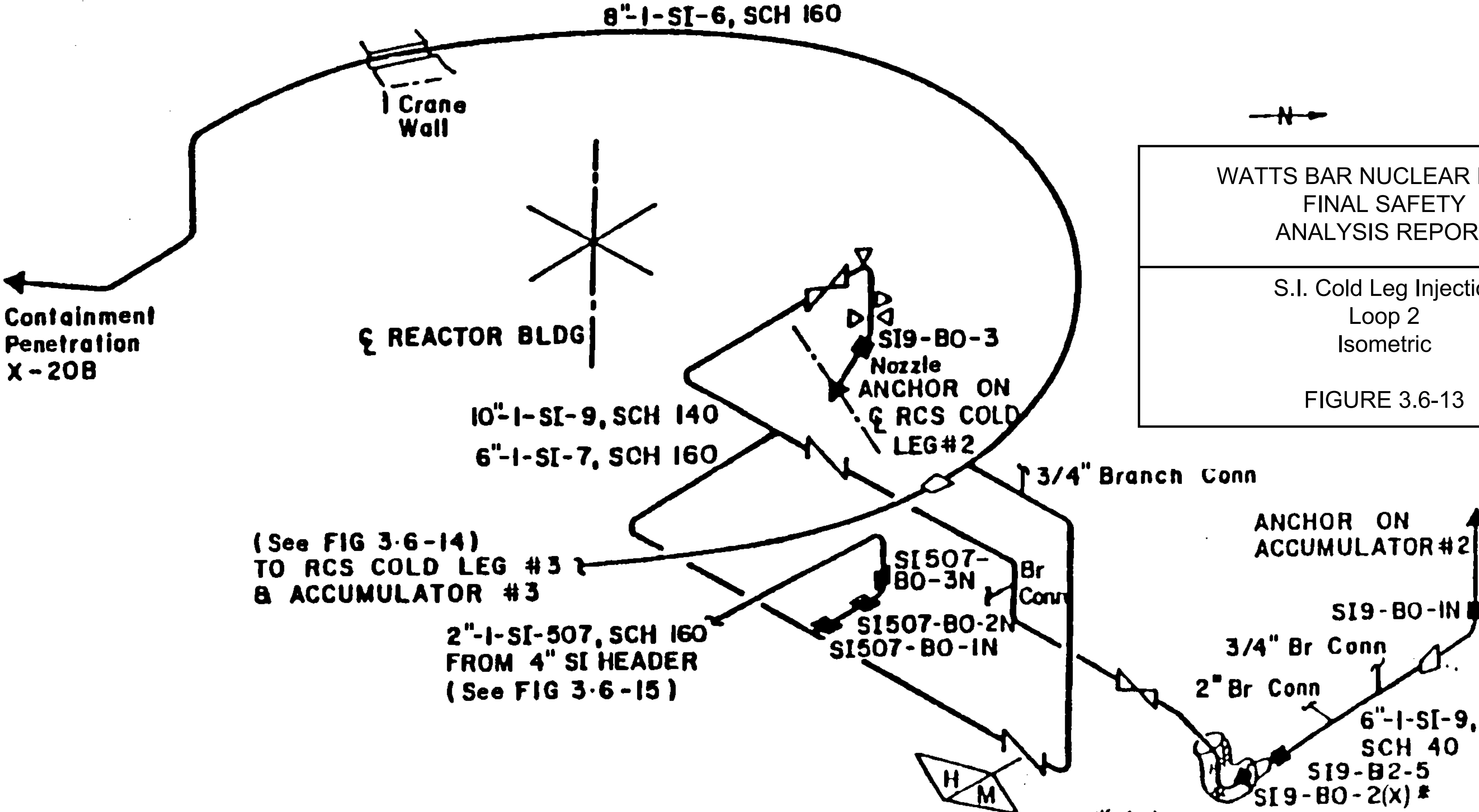


REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

FIGURE 3.6-12

SI COLD LEG INJECTION
LOOP 2

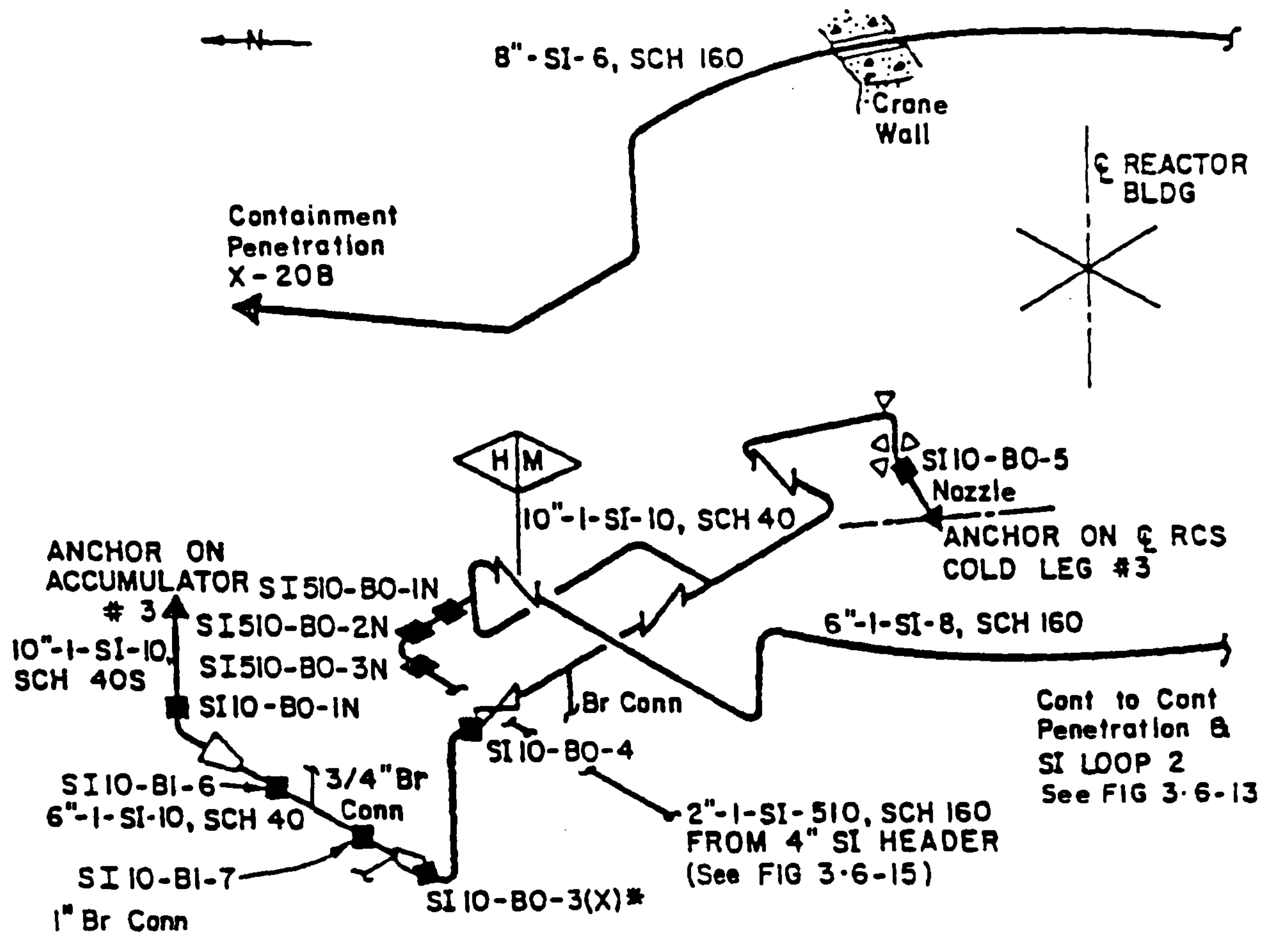


*(X)-DESIGNATES THIS BREAK NOT REQUIRED FOR LICENSING

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
S.I. Cold Leg Injection Loop 2 Isometric FIGURE 3.6-13

SI COLD LEG INJECTION
LOOP 3

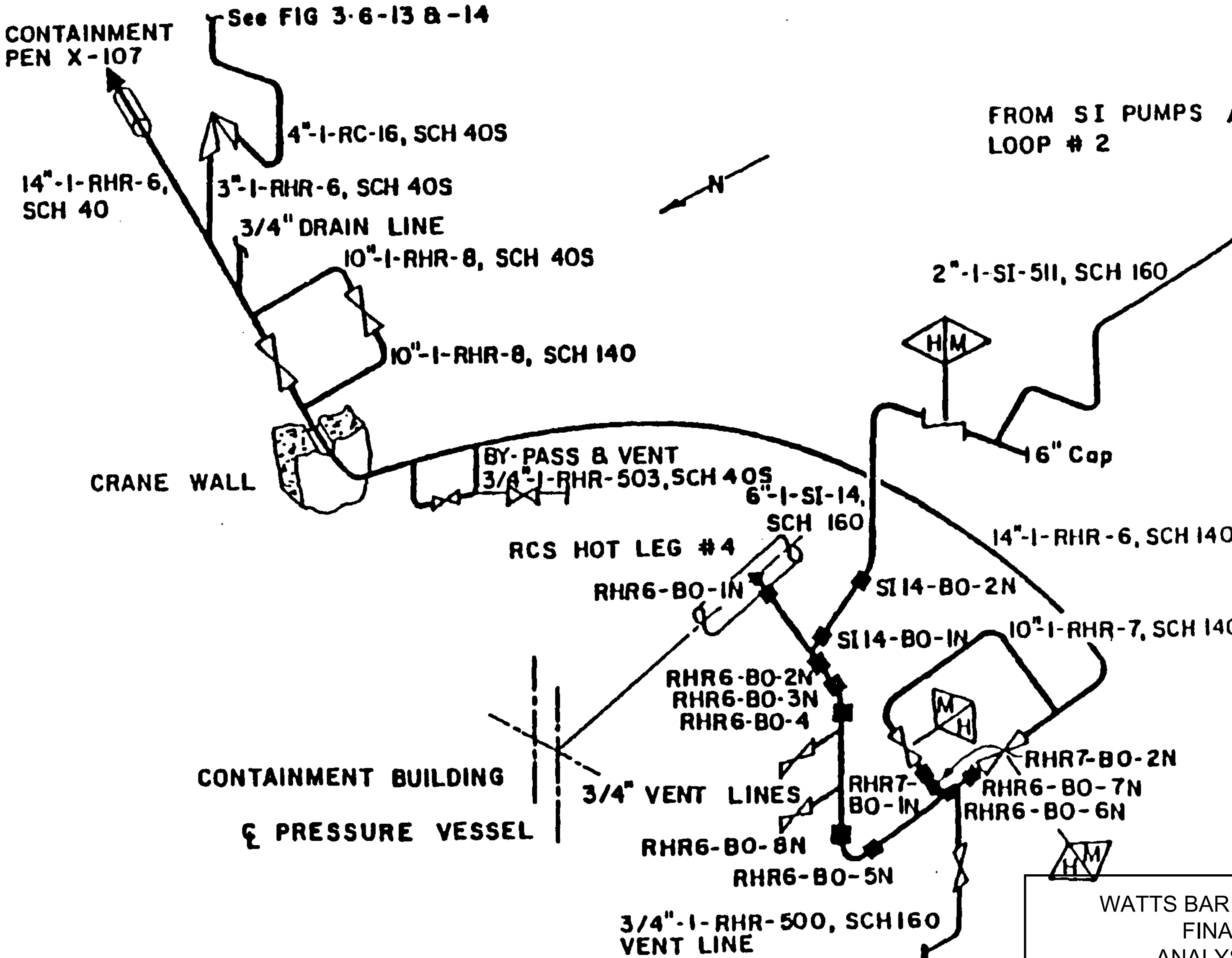


*(X)- DESIGNATES THIS BREAK NOT REQUIRED FOR LICENSING

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Isometric of Postulated Break Locations FIGURE 3.6-14

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 61 SUBMITTAL



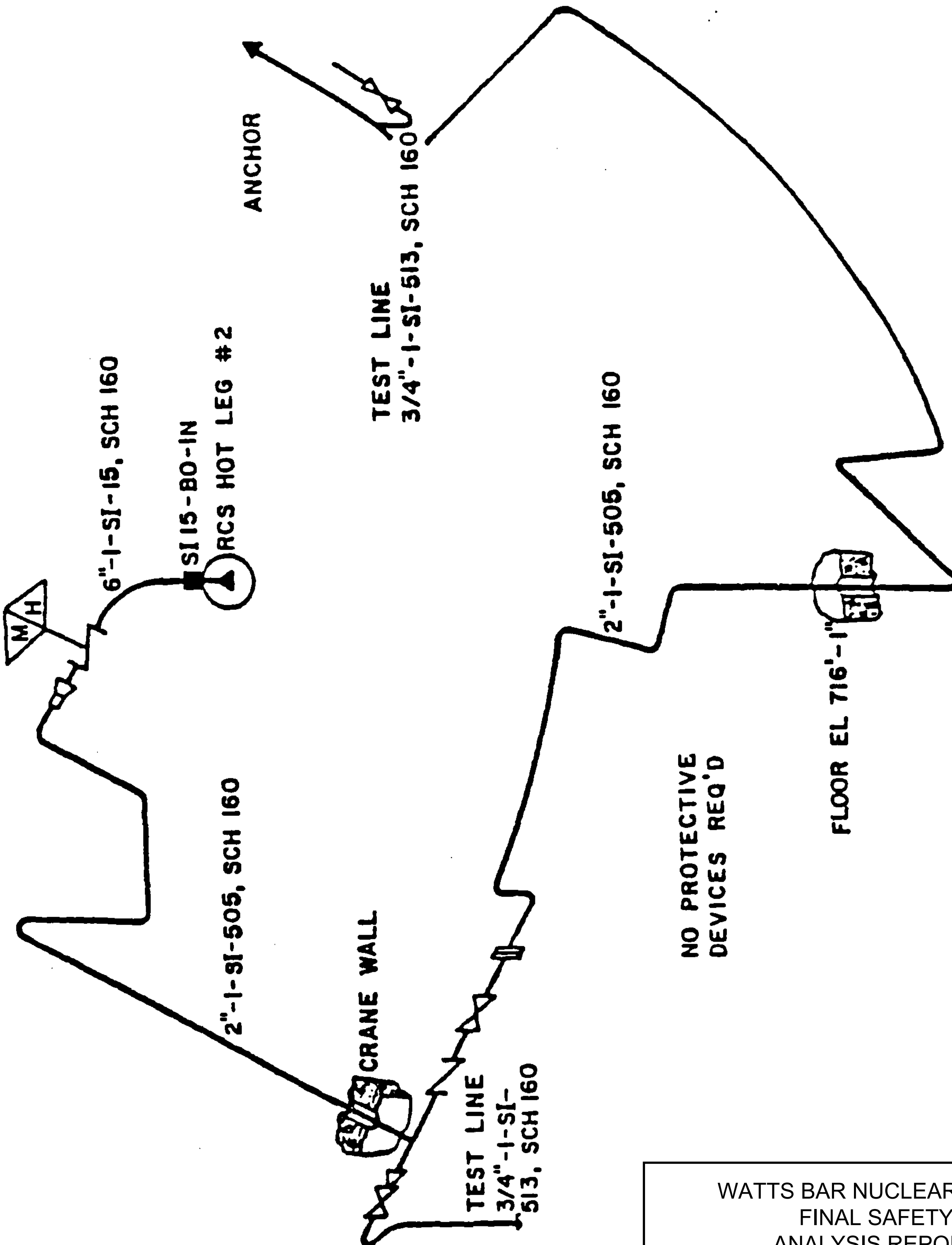
RHR/SI HOT LEG RECIRCULATION, LOOP 4

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RHR/S.I. Hot Leg Recirculation
Loop 4
Isometric

FIGURE 3.6-15

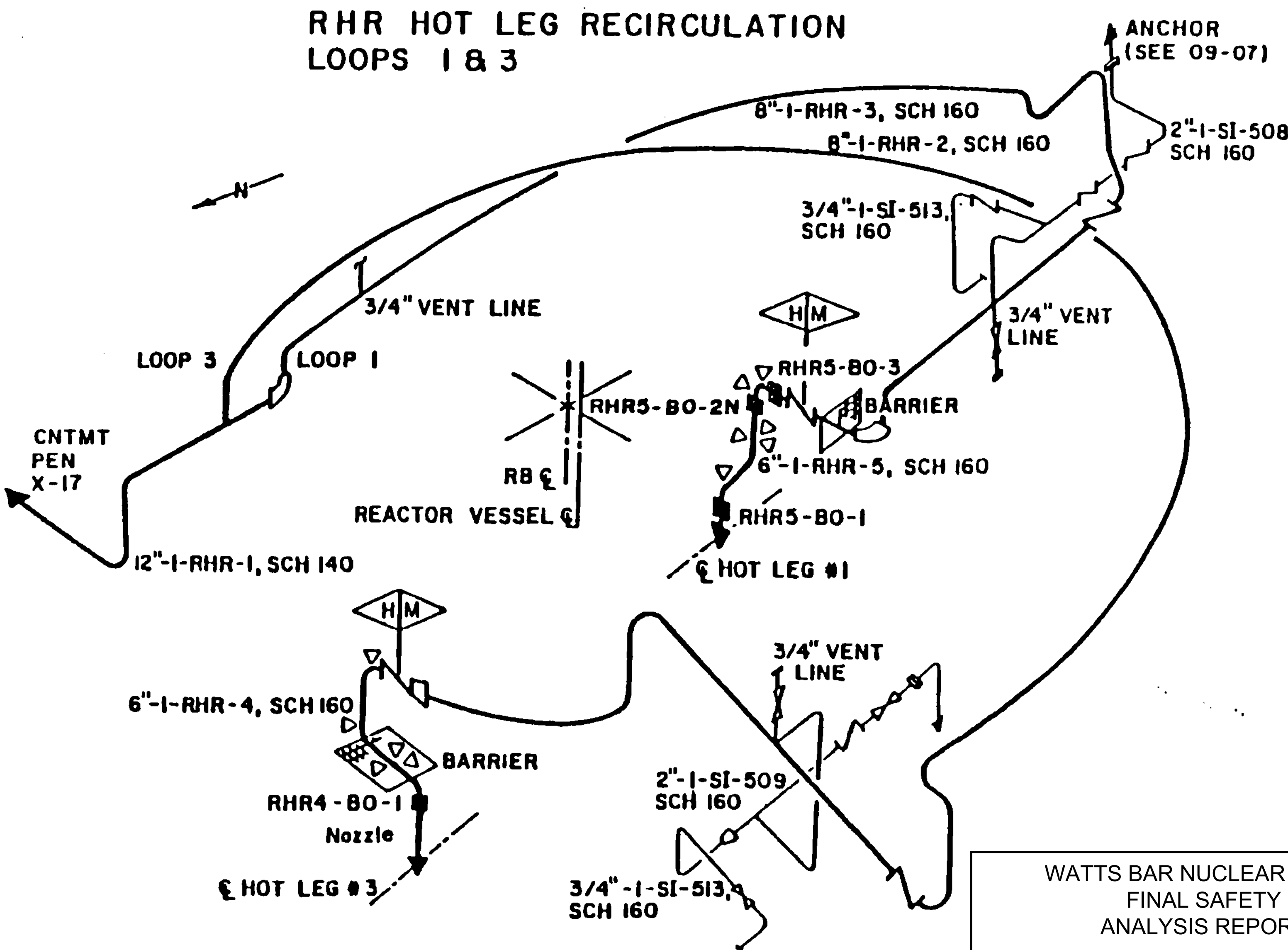
SI HOT LEG RECIRCULATION
LOOP 2



REFLECTS ANALYSIS WHICH WAS
CURRENT AT TIME OF AMENDMENT
51 SUBMITTAL.

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
S.I. Hot Leg Recirculation Loop 2 Isometric FIGURE 3.6-16

REFLECTS ANALYSIS WHICH WAS CURRENT AT TIME OF AMENDMENT 61 SUBMITTAL.



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

RHR Hot Recirculation
Loops 1 and 3
Isometric
FIGURE 3.6-17

FIGURE 3.6-18 THRU

FIGURE 3.6-20

DELETED

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.6-21

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.6-22

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.6-23

APPENDIX 3.6A

PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING (EXCLUDING REACTOR COOLANT SYSTEM PIPING)

Criteria presented herein regarding break size, shape, orientation, and location are in accordance with the guidelines transmitted to TVA by the NRC in letter, dated December 1972, and subsequent amendments for outside containment, and NRC Regulatory Guide 1.46 for inside containment. These criteria also include considerations which are further clarified in the NRC Branch Technical Positions ASB 3-1 and MEB 3-1 where appropriate. Arbitrary intermediate breaks (AIBs) postulated in accordance with the documents noted above are eliminated by NRC Generic Letter 87-11[4].

The final routing of field routed systems was not completed until late in the plant construction schedule. Field-routed piping generally possesses very little potential, insofar as their functions are concerned, toward affecting plant shutdown. Their failure can, however, cause damage to other components and equipment, especially electrical, which may be required for shutdown of the plant. Field-routed and field-located items such as electrical conduit, cable trays, instrument and control lines, and junction and terminal boxes, etc., are protected as required for plant shutdown. Where field routing was required, guidance was provided to minimize the number of unacceptable interactions. A followup field review and evaluation, for identifying unacceptable interactions and ensuring implementation of corrections was performed.

The following definitions and assumptions are applicable to this section:

DEFINITIONS

1. Acceptable Interaction

   A pipe rupture interaction for which, from a systems standpoint, the net required safety functions for a particular rupture are not impaired when assuming a single active component

2. Active Component

   Any component which must perform a mechanical motion or change of state during the course of accomplishing a primary safety function.

3. Double-Ended Rupture

   A circumferential pipe rupture where flow is sustained from both ends of the break.

4. Environmental Effects

   The wetting, pressure, temperature, flammable, radiation, etc., conditions within the 'zone of influence' (Definition 28) of a pipe rupture.

5. <u>Essential Systems and Components</u>

Systems and components required to shutdown the reactor and/or mitigate the consequences of a postulated pipe failure without offsite power. The seismic classification of essential components and systems is in accordance with Regulatory Guide 1.29.

6. <u>High Energy Fluid Systems</u>

Fluid systems that, during normal plant conditions, satisfy the following:

a. Maximum operating temperature exceeds 200 °F, and

b. Maximum operating pressure exceeds 275 psig.

Systems may be classified as moderate energy (see Definition 13) if the total time that the above conditions are exceeded is less than either of the following:

a. One percent of the normal operating life span of the plant.

b. Two percent of the time period required for the system to accomplish its design function.

7. <u>Inside Containment</u>

Inside containment is defined for pipe rupture evaluation purposes to include all piping inside the Shield Building and the main steam valve rooms. The actual containment boundary for integrity purposes is normally taken at the second isolation valve.

8. <u>Jet Impingement Force</u>

The jet force on an object resulting from a ruptured pipe. The magnitude of this force depends on such parameters as the thermodynamic conditions of the fluid in the pipe, distance of the pipe rupture from the target and the shape of the target.

9. <u>Jet Thrust</u>

That reactive dynamic force on a ruptured pipe due to a fluid being accelerated out of a break.

10. <u>Line-Mounted Valves</u>

Valves located in a line and supported by the line.

11. <u>Loss-of-Coolant Accident (LOCA)</u>

LOCA is defined as a net loss of reactor coolant inventory when makeup is provided only by the normal makeup system and an orderly shutdown of the plant is prevented. Normal makeup is sized to maintain a constant reactor coolant system (RCS) inventory with a rupture equivalent to a 3/8-inch diameter hole. Therefore, a rupture is considered a LOCA when the flow rate is greater than the equivalent flow from a 3/8-inch diameter hole.

12. <u>LOCA Boundary</u>

For piping extended from the RCS, the boundary of postulated pipe rupture which cannot be isolated when assuming a single active failure shall be defined as follows:

a. First locked closed or administratively closed isolation valve (pressurizer safety valves are examples). The valves forming the Class 1 boundary in all drain lines are considered as administratively closed.

b. Second of two normally open, remotely operable, independent isolation valves capable of automatic closure and verification that they will close.

c. First normally closed check valve capable of verification that it is closed and capable of providing isolation from a reactor coolant source.

d. Second of two normally open check valves capable of verification that they will close and capable of providing isolation from a reactor coolant source. (Verification that a check valve will close should be interpreted as meaning 'capable of periodic test that will verify its capability of closure, such as during a refueling outage.')

e. First normally open and remotely operable automatic isolation valve following a normally open check valve (capable of providing isolation from a reactor coolant source) if both are capable of verification that they will close.

If a pipe failure beyond the above defined boundary of possible isolation could result in a normally open boundary valve failing to close, then a LOCA may exist beyond that boundary.

13. <u>Moderate Energy Fluid Systems</u>

Fluid systems that, during normal plant conditions, satisfy either of the following:

a. Maximum operating temperature is 200°F or less or

b. Maximum operating pressure is 275 psig or less.

Other systems which may be classified as moderate energy are discussed in Definition 6.

14. Normal Plant Conditions

    Plant operating conditions during reactor startup, refueling, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

15. Outside Containment

    Outside containment includes all of those regions not included in the definition of 'Inside Containment' (Definition 7)

16. Pipe Whip

    The movement of a pipe caused by jet thrust resulting from a pipe failure. Pipe whip is assumed to occur in the plane defined by piping geometry and configuration unless limited by structural members, pipe restraints, or pipe stiffness.

17. Primary Safety Function

    The passive or active function of a structure, system, or component which must remain functional to assure directly: (1) the integrity of the RCPB, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures in excess of the guideline exposure of 10 CFR 100.

18. Postulated Piping Failures

    Longitudinal splits, circumferental ruptures, or through-wall leakage cracks.

19. Protective Structures or Compartments

    Structural units provided to separate or enclose redundant trains of safety-related systems or enclose high and moderate energy lines. (These structures are designed as Seismic Category I)

20. Reactor Coolant Pressure Boundary (RCPB)

    Those pressure containing components such as pressure vessels, piping, pumps, and valves, which are:

    a. Part of the reactor coolant system or

    b. Connected to the reactor coolant system, up to and including any of the following:

        1. The outermost containment isolation valve in system piping which penetrates the containment.

2. The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the containment.

3. The reactor coolant system safety and relief valves.

21. Safety Related

Those plant features which are important to safety because they perform either a primary safety function or a secondary safety function.

22. Secondary Safety Function

The function of a portion of a structure, systems or component which must retain limited structural integrity because its failure could jeopardize the achievement of a primary safety function or because it forms an interface between Seismic Category I and Seismic Category I(L) or non-seismic plant features.

23. Seismic Category I

Those structures, systems, or components which perform primary safety functions are designated as Seismic Category I and are designed and constructed so as to assure achievement of their primary safety functions at all times including a concurrent safe shutdown earthquake (SSE).

24. Seismic Category I(L)

Those portions of structures, systems, or components which perform secondary safety functions and are designed and constructed so as to assure achievement of their secondary safety functions at all times including a concurrent safe shutdown earthquake (SSE).

25. Shutdown Logic Diagram

A logic diagram identifies safety related systems and safety functions and actions required for shutdown to safe conditions.

26. Single Active Component Failure

A single active failure is the failure of an active component to complete its intended function upon demand. The failure of an active component of a fluid system is considered to be a failure of the component to perform its function not the loss of structural integrity. The direct consequences of a single active failure are evaluated. (A single active failure is postulated to occur simultaneously with the pipe failure; passive failures are not postulated.)

27. Terminal Ends

Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, etc), or pipe anchors that act as rigid constraints to piping thermal expansion. A branch connection to a main piping run may be considered as a terminal end of the branch run unless each of the following conditions are met:

a. That branch is modeled with the main piping run.

b. A rigorous ASME, Class 1, 2, or 3 analysis is conducted.

c. The nominal size of the branch line, in the vicinity of the branch connection, is greater than or equal to one-half the nominal size of the run.

28. Zone of Influence

The maximum physical range of the direct effects of pipe whip, jet impingement, and/or the environmental effects resulting from a pipe failure.

ASSUMPTIONS

In analyzing the effects of postulated piping failures, the following assumptions shall be made relative to plant and system operation before and after a pipe failure.

1. Operating Mode

All normal plant operating modes (see Definition 14) shall be investigated when evaluating the effects of a postulated pipe failure.

2. Single Active Component Failure

A single active failure is assumed in systems used to mitigate consequences of the postulated piping failure and to shutdown the reactor. The single active failure is assumed to occur in addition to and concurrent with the postulated piping failure and any consequences of the piping failure.

3. Available Systems

All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account shall be taken of the postulated failure and its consequences such as unit trip and loss of offsite power and of the assumed single active component failure and its consequences. The feasibility of carrying out operator actions shall be judged on the basis of ample time and adequate access to equipment being available for the proposed actions. No operator action is assumed to be initiated for at least 10 minutes after pipe failure.

4. Offsite Power

In general, if it is the worst case, offsite power shall be assumed to be unavailable during a portion of or throughout the sequence of events that follow a pipe failure. This loss of offsite power shall be assumed to act concurrently with the postulated pipe failure and the single active failure. If it can be shown that the loss of offsite power is not a consequence of the pipe failure, then a loss of offsite power is not assumed.

5. Unintended Operation of Equipment

The performance of an unintended active function by equipment not within the zone of influence of a pipe failure shall not be postulated. Unintended operation of equipment within the zone of influence of the pipe failure may occur if caused by the pipe failure, provided the unintended operation is a credible postulation. Unintended operation will not be considered to place equipment in any operating mode other than those modes for which it is normally required to function.

6. Operator Response

It shall be assumed that a proper sequence of events is initiated by the operator to bring the plant to a safe condition, with the capability of going to a cold shutdown if required. However, it shall be assumed that no operator action is initiated for at least 10 minutes after pipe failure. Additional time will be allocated for actions outside the main control room.

3.6A.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.6A.1.1 Design Bases

3.6A.1.1.1 List of Potential Targets

Safety-related systems or components that are located proximate to and are susceptible to the consequences of failures of piping systems are discussed in Section 3.11.

3.6A.1.1.2 Interaction Criteria

The following criteria define how interactions are evaluated:

1. Pipe Whip Interaction

A whipping pipe is not considered to inflict unacceptable damage to other pipes and associated supports of equal or greater size and wall thickness. A whipping pipe is considered capable of only developing through-wall leakage cracks in other pipes of equal or greater size with smaller wall thickness.

Any active component (electrical, mechanical, and instrumentation and control) shall be assumed incapable of performing its active function following impact by any whipping pipe unless an analysis or test is conducted to show otherwise. Active components in pipe lines which are allowed to whip are assumed to be incapable of performing their active functions unless the line is sufficiently restrained to control the motion of the components to limits for which they have been qualified.

Structural components shall be assumed to fail upon experiencing pipe impact loads that exceed the allowable limits. Plastic action of steel, yield line methods etc., may be used to determine the allowable limits where applicable.

2. Jet Impingement Interactions

Jet impingement force from a pipe is not considered to inflict unacceptable damage to other pipes and associated supports of equal or greater size and wall thickness. The jet impingement force is considered capable of only developing through-wall leakage cracks in other pipes of equal or greater size with smaller wall thickness.

Active components (electrical, mechanical, and instrumentation and control) shall be assumed incapable of performing their function when subjected to a jet unless the active component is enclosed in a qualified spray-proof enclosure (such as one qualified to the NEMA IV, Hosedown Test Standard), the component is known to be insensitive to such an environment, or unless justified that the active function will not be impaired.

When the jet consists of steam or subcooled liquid that flashes at the break, unprotected components located at a distance greater than 10 diameters (ID) from the break or equivalent diameter of the crack shall be assumed undamaged by the jet without further analysis. The basis for this criterion is contained in Reference [5].

Concrete erosion that may result from jet impingement shall be assumed to be of insufficient magnitude to jeopardize structural integrity.

3. Environmental Interaction

An active component (electrical, mechanical, and instrumentation and control) shall be assumed incapable of performing its active function upon experiencing environmental conditions exceeding any of its environmental ratings. However, credit for the component may be taken if sufficient time is available for accomplishing its function before environmental ratings are exceeded.

3.6A.1.1.3 Acceptability Criteria

1. Systems

The capability to eventually achieve a cold shutdown condition shall not be jeopardized even if the pipe failure is followed by a single active failure. The system requirements and available redundancy shall be that shown on a shutdown logic diagram, as supplemented by current system descriptions and equipment lists, for mitigating the effects of the postulated failure.

Repair of failures may be considered to assure achievement of the cold shutdown condition where such repairs can be shown to be practicable and timely, and provided the unit can be held in a safe state during the time required for the repair.

2. Protective Structures

The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, should not preclude habitability of the control room or access to surrounding areas required for safe control of reactor operations that are needed to cope with the consequences of the piping failure.

For piping systems that are enclosed in suitably designed structures or compartments to protect other structures, systems, and components important to safety, pipe breaks shall be postulated according to section 3.6A.2 and the resulting jet thrust loading effects determined. "Worst case" breaks may be postulated in a piping component within the protective structure or compartment at locations which result in the maximum loading from the impact of the postulated ruptured pipe and jet discharge force on each wall, floor, and roof of the structure or compartment, including internal pressurization.

3.6A.1.1.4 Protective Measures

Where physical separation of source and target and relocation or rerouting are not feasible, the following protective devices will be provided to mitigate the unacceptable consequences of the postulated ruptures.

1. Pipe Whip Restraints: An engineered structure which permits limited pipe motion and rotation but limits or prevents unrestricted pipe whip. Crushable material may be used with certain restraints to absorb the kinetic energy of the ruptured pipe, and to limit the loads on the restraint structure.

2. Jet Deflector: A barrier which shields a target from the forces and environmental conditions within a jet.

3. Impact Barrier: An engineered structure located to limit pipe motion and designed to withstand the impact of a whipping pipe.

4. Pipe Sleeve: A metal sleeve that encloses a portion of a process pipe and is designed to restrict and redirect jet forces.

Welding for protective structures designed to the requirements of AISC (see Section 3.8.1.2, Item 2) was in accordance with the American Welding Society, "Structural Welding Code," AWS D1.1 (see Section 3.8.1.2, Item 4). Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 (see Section 3.8.1.2, Item 12) may be used after June 26, 1985, to evaluate weldments that were designed and fabricated to the requirements of AISC/AWS.

3.6A.1.2 Description of Piping System Arrangement

Separation was the primary consideration in the piping system layout and arrangement. Where physical separation is not feasible, protective devices shall be provided as required. Protection shall be provided such that the environmental design limits of mechanical and electrical equipment required for safe shutdown are not exceeded. Habitability is discussed in Section 6.4.

3.6A.1.3 Safety Evaluation

Safety functions shall be identified for initiating event by means of shutdown logic diagrams (SLD). The SLD shall identify at least one success path from each postulated event to each protective function required to prevent the event's potentially unacceptable results. Each SLD shall include the set of all safety systems necessary to provide the protective function specified at the end of the success path. Shutdown logic diagrams may be supplemented by current system descriptions and equipment lists.

For each postulated pipe rupture, credible unacceptable interaction shall be evaluated.

Possible interactions shall be evaluated to determine their credibility, damage potential, and acceptability from the standpoint of a safe shutdown capability.

In establishing system requirements for each postulated break, it is assumed that a single active component failure occurs concurrently with the postulated rupture.

3.6A.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6A.2.1 Criteria Used to Define Break and Crack Location and Configuration

3.6A.2.1.1 Pipe Failure Type, Size, and Orientation

1. Circumferential Rupture

   The break area is equal to the effective cross-sectional flow area of the pipe at the break location. The plane of the break is normal to the pipe flow axis. Flow may be out of each of the broken ends (double ended rupture) of the pipe, depending upon reverse flow capability.

This break is applicable to high energy piping and branch runs whose diameter is greater than 1-inch nominal pipe size. Circumferential ruptures are assumed to result in a lateral offset of one pipe diameter unless mitigating devices, structure members, or the inherent pipe stiffness can be specifically shown to limit this offset.

2. <u>Longitudinal Split</u>

The break area is assumed to be equal to the effective pipe cross-sectional flow area at the break location. If the break occurs at a transition from a smaller pipe to a larger pipe, the flow area is defined as one-half the sum of the upstream and downstream cross-sectional flow areas. The length of the break is two pipe inside diameters and is parallel with the pipe flow axis. As an alternate analysis procedure, fluid flow may be assumed to be from a circular opening equal to the effective cross-sectional flow area of the pipe. In the absence of a detailed analysis, the break is assumed at any location around the circumference of the pipe. Alternatively, a single split may be assumed at the point on the circumference of highest tensile stress as determined by a detailed stress analysis. This break is applicable to high energy pipe that has a nominal pipe size of 4 inches or larger.

3. <u>Through-Wall Leakage Crack</u>

The crack area may be based on a circular opening with an area equal to an equivalent rectangular opening of one-half the piping inside diameter in length and one-half the wall thickness in width and can be oriented in any direction.

3.6A.2.1.2 <u>Break Location</u>

1. <u>High Energy Fluid System</u>

A. ASME Section III Class 1 Piping Runs

Circumferential ruptures and longitudinal splits, in accordance with Sections 3.6A.2.1.1 (Item 1) and 3.6A.2.1.1 (Item 2), are postulated to occur at the following locations in ASME Section III Class 1 piping:

1. The terminal ends of piping or branch runs (circumferential ruptures only).

2. At intermediate locations per either one of the following [method a or method b]:

   a. At each location of potential high stress and fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, and flanges, or

b. At all locations where either one of the following are met.

1) $S_n < 2.4\ S_m$* (Equation 10) and $U > 0.I$ (U calculated according to NB-3653.5); or

2) $S_n > 2.4\ S_m$* (Equation 10) and $S_e > 2.4\ S_m$ (Equation 12), or

$S > 2.4\ S_m$ (Equation 13),

or $U > 0.I$ (U calculated according to NB-3653.6)

*For stress qualification to Summer 1973 Code, use 3.0 S_m
For stress qualification to Winter 1982 Code, use 2.4 S_m

Where:

S_n = primary-plus-secondary stress-intensity range, as calculated from Equation 10 in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III for normal and upset plant condition loads with the upset plant condition loads defined as: sustained loads + all system operating transients associated with upset condition + OBE.

S_m = allowable design stress-intensity value, as defined in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

U = the cumulative usage factor, as calculated in accordance with Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

S_e = Nominal value of expansion stress as defined in equation (12) of NB-3653.6 of ASME Code, Section III.

S = The range of primary plus secondary membrane plus bending stress intensity as defined in equation (13) of NB-3653.6 of ASME Code, Section III.

Longitudinal splits need not be postulated in Class 1 piping at terminal ends or branch connections.

Through-wall leakage cracks are postulated in all high energy pipe outside containment whose diameter is greater than one inch nominal pipe size. Through-wall leakage cracks are not postulated in high energy piping inside containment whose diameter is greater than 1 inch nominal pipe size. However, through-wall leakage cracks are postulated in the main steam and feedwater lines inside containment where impingement could occur on the ice condenser doors. Also, through-wall leakage cracks, may be postulated in other high-energy lines in particularly susceptible areas.

B. Break Locations in ASME Section III Class 2 and 3 Piping Runs

Circumferential ruptures and longitudinal splits, in accordance with Sections 3.6A.2.1.1 (Item 1) and 3.6A.2.1.1 (Item 2), are postulated to occur at the following locations in ASME Section III Class 2 and 3 piping:

1. The terminal ends of piping or branch runs (circumferential ruptures only).

2. At intermediate locations selected by either one of the following method a) or method b):

   a. At each location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves and flanges, or

   b. At all locations where the stress, S, exceeds 0.8 (1.2 S_h + S_a)

where:

S = stresses under the combination of loadings associated with the normal and upset plant condition plus OBE loadings, as calculated from the sum of Equations (9) and (10) in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

S_h and S_a = Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, for Class 2 and 3 piping as defined in Subarticle NC-3600 of ASME Code, Section III.

Through-wall leakage cracks are postulated as indicated in Section 3.6A.2.1.2 (Item 1A).

C. Exceptions for Longitudinal Splits and Circumferential Ruptures

The following exceptions are applicable to high-energy Class 1, 2 and 3 piping and to high-energy non-safety class piping for which a Class 2 or 3 analysis is conducted.

1. Longitudinal splits need not be postulated at terminal ends or branch connections.

2. When values defined in 3.6A.2.1.2 are exceeded for Class 1 piping or the stresses exceed 0.8 (1.2 S_h + S_a), for Class 2 and 3 piping longitudinal splits need not be postulated if the stress in the axial direction is greater than or equal to 1.5 times the stress in the circumferential direction; and circumferential ruptures need not be postulated if the stress in the circumferential direction is greater than or equal to 1.5 times the stress in the axial direction.

D. High Energy Non-Safety-Class and Field Routed Fluid Systems

Circumferential ruptures and longitudinal splits in high energy non-safety-class and high energy field routed piping components are postulated to occur at terminal ends and at intermediate pipe fittings, flanges, and valves. Through-wall leakage cracks are postulated as indicated in Section 3.6A.2.1.2, Item 1A.

Break locations in high energy non-safety-class systems, which are analyzed to the same requirements as Class 2 or 3 piping, (these cases will be fully coordinated and documented) may be postulated according to the requirements of Section 3.6A.2.1.2, Item 1B.

2. Moderate Energy Fluid Systems

Circumferential ruptures and longitudinal splits are not postulated in any moderate energy lines. Through-wall leakage cracks are postulated in moderate energy piping which exceed a nominal pipe size of 1 inch, but may be excluded where either of the following rules apply.

A. Piping systems are located in areas containing systems and/or components important to safety enveloped by previously postulated high energy breaks in the same region.

B. Where the maximum stress, S, as defined in Section 3.6.A.2.1.2 (Item 1B) is less than or equal to 0.4 ($1.2\ S_h + S_a$) for Class 2 and 3 piping or where S_n by equation 10 is less than or equal to 1.2 S_m for Class 1 piping.

The cracks should be postulated to occur individually at locations that result in the maximum effects from fluid spraying and flooding. It shall be at any location on the pipe circumference or along the surface of the pipe.

3. High/Moderate Energy Interfaces

Line supported valves sometimes form the interface between high energy lines and moderate energy lines. In this case, the fixity as implied in the word, 'terminal,' does not exist at the line supported valve. This condition is treated as if there were no terminal.

3.6A.2.1.3 Failure Consequences

The failure interactions that must be evaluated to determine the consequences of failure are dependent upon the energy level of the pipe considered. They are as follows:

1. High Energy Piping

Circumferential ruptures and longitudinal splits

a. Pipe whip.
b. Jet impingement.
c. Environmental effects.

Through-wall leakage cracks

a. Jet impingement
b. Environmental effects

2. Moderate Energy Piping

Through-wall leakage cracks

a. Environmental effects.

In particularly susceptible areas, the jet impingement load associated with a through-wall leakage crack in moderate energy piping with the pressure exceeding 275 psig shall also be considered.

3.6A.2.1.4 Flooding

Flooding consequences are also considered in addition to the local effects listed above in Section 3.6A.2.1.3 from piping failures. Additional environmental concerns are addressed in Section 3.11.2.

1. High Energy Line Breaks (HELBs)

For the purposes of flooding evaluations, fluid systems that, during normal plant conditions are either in operation or maintained pressurized under conditions where maximum operating temperature exceeds 200°F are conservatively classified as high energy. This is bounding since for a given line, the flow from a high energy break emanates from a larger break area than flow from a moderate energy crack. The circumferential rupture is the bounding break for HELB flooding analyses.

Systems classified as high energy are re-classified as moderate energy if the total time that the above conditions are exceeded is less than either of the following:

a. 1% of the normal operating life span of the plant, or
b. 2% of the time required for the system to accomplish its system design function.

The systems evaluated for high energy break flooding include the reactor coolant, main steam, feedwater, auxiliary boiler, auxiliary feedwater steam supply, and chemical and volume control system.

2. Moderate Energy Line Breaks (MELBs)

For the purposes of flooding evaluations, fluid systems are classified as moderate energy that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where: (1) The maximum operating temperature is 200°F or less or (2) the 1 or 2% exclusion rules described above are applicable. The through-wall leakage crack is the postulated break for the MELB flooding analysis. Flood levels are calculated for the plant on an area basis. Both submergence and structural loading are addressed in the flooding studies.

HELB and MELB flooding effects are evaluated on all essential equipment on a case by case basis. If it is determined that an essential component is not qualified or cannot be demonstrated to operate under the adverse flood conditions, then the essential component is protected. Protection is accomplished by relocating the component or by installing a barrier or curb. Safe shutdown is ensured for design basis HELB/MELB flooding events through these actions.

3.6A.2.1.5 Leak-Before-Break Application

The application of leak-before-break as applied to the primary loop piping is discussed in Section 3.6B.1.

In addition, leak-before-break technology has been applied to the pressurizer surge line to eliminate the dynamic effects of a pressurizer surge line rupture as a design basis for Watts Bar Nuclear Plant. This is in accordance with the final rule change to General Design Criteria 4.[12] Authorization for their elimination is discussed in Reference [9] and is based on fracture mechanics results presented in References [10] and [11].

3.6A.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6A.2.2.1 Assumptions

1. The thrust load acting on the pipe due to a blowdown jet is equal and opposite to the jet.
2. The discharge coefficient is equal to 1.0.
3. The break opens to its defined size in 1 millisecond.
4. For the purpose of estimating jet forces, the blowdown is to an infinite volume at standard conditions.
5. The initial fluid condition within the pipe prior to rupture is that for normal plant operating condition.
6. The jet profile expansion half angle is 20 degrees.

3.6A.2.2.2 Blowdown Thrust Loads

The thrust force at any time, T (t) is given by

$$T(t) = \left(\frac{\rho_E V_E^2}{g_c} + \left[P_E - P_A \right] \right) A_{jE}$$

where:

ρ_E = fluid density at break at time t

V_E = fluid velocity at break at time t
A_{jE} = pipe break exit area
P_E = control volume pressure at break at time t
P_A = ambient pressure
g_C = gravitation constant

A simplified analysis may be conducted by assuming that the fluid is blowing down in a steady-state condition with frictionless flow from a reservoir at fixed absolute pressure Po. (Po is the initial line pressure.) When the fluid is subcooled, nonflashing liquid, the flow will not be critical at the break area so that:

$$P_E = P_A$$

and

$$V_E = \sqrt{2g_c(P_O - P_A)/\rho_E}$$

If $P_A << P_o$ the thrust force may be conservatively approximated by:

$$T = 2P_0 A_{jE}$$

When the fluid is saturated, flashing or super-heated vapor, the fluid can be assumed to be a perfect gas. The velocity for critical flow at the break area is given by:

$$V_E = \sqrt{Kg_c P_E/\rho_E}$$

And

$$P_E = P_O\left[\frac{2}{K+1}\right]^{\frac{k}{K-1}}$$

where

$K = C_p/C_v$ is a ratio of specific heats
C_p = specific heat at constant pressure
C_v = specific heat at constant volume

A value of K = 1.3 is justified for steam as being conservative. If $P_E >> P_A$, the thrust force may be conservatively approximated by:

$$T = 1.26\ P_o\ A_{jE}$$

3.6A.2.2.3 Jet Impingement Loads

The loads on an object exposed to the jet from a pipe break can be determined from the blowdown thrust and the profile of the impinged object.
where

$$Y_j = T \cdot \frac{A_i}{A_j} \cdot S_F \cdot D_{LF} \cos\phi$$

Y_j =Normal load applied to a target by the jet

A_i =Cross-sectional area of jet intercepted by target structure

A_j =Total cross-sectional area of jet at the target structure

S_F =Shape factor

D_{LF} =Dynamic load factor

T =Total blowdown thrust at break as calculated in Section 3.6A.2.2.2

ϕ =Angle between jet axis and a line perpendicular to the target.

The ratio A_i/A_j represents the proportion of the total mass flow from the jet which is intercepted by target structure. A dynamic load factor of 2.0 shall be used in the absence of an analysis justifying a lower value. The following shape factors are recommended.

Jet impinging on a slab [Figure 3.6-1 sector (a)]

$$S_F = 1$$

Rectangular jet impinging on a pipe larger than jet [Figure 3.6-1 sector (b)]

$$S_F = 1 - \frac{h}{2D_o}$$

Rectangular jet impinging on a pipe with h greater than D_o [Figure 3.6-1 sector (b)]

$$S_F = ½$$

Circular jet impinging on pipe with jet diameter ($D_j = 2r_j$) less than pipe diameter [Figure 3.6-1 sector (c)]

$$S_F = 1 - 0.424\frac{D_j}{D_o}$$

Circular jet impinging on pipe with jet diameter greater than pipe diameter [Figure 3.6-1 sector (d)]

$$S_F = 0.576$$

These are the most common cases that will occur in the pipe rupture evaluation. Other shape factors may be obtained by idealizing the surface as infinitesimal planes and performing an integration over the area impinged upon by the jet.

3.6A.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6A.2.3.1 General Criteria for Pipe Whip Evaluation

1. The dynamic nature of the piping thrust load shall be considered. In the absence of analytical justification to the contrary, a dynamic load factor of 2.0 may be applied in determining piping system response.

2. Nonlinear (elastic-plastic strain hardening) pipe and restraint material properties may be considered as applicable.

3. Pipe whip shall be considered to result in unrestrained motion of the pipe along a path governed by the hinge mechanism and the direction of the vector thrust of the break force. A maximum of 180° rotation may take place about any hinge.

4. The effect of rapid strain rate of material properties may be considered. A 10% increase in yield strength may be used to account for strain rate effects.

3.6A.2.3.2 Main Reactor Coolant Loop Piping System

The dynamic analyses applicable to the reactor coolant loop piping are discussed in Section 3.6B.

3.6A.2.3.3 Other Piping Systems

The pressure time history, jet impingement load on targets, and the thrust resulting from the blowdown of postulated ruptures in piping systems shall be determined by thermal and hydraulic analyses or a conservative simplified analyses.

In general, the loading that may result from a break in piping will be determined using either a dynamic blowdown or a conservative static blowdown analysis. The method for analyzing the interaction effects of a whipping pipe with a restraint will be one of the following:

1. Equivalent static method

2. Lumped parameter method

3. Energy balance method.

In the cases where time history or energy balance method is not used, a conservative static analyses model will be assumed. The loading factors to be used for the static model are discussed in Section 3.6A.2.3.5.

The lumped parameter method is carried out by utilizing a lumped mass model. Lumped mass points are interconnected by springs to take into account inertia and stiffness properties of the system. A dynamic forcing function or equivalent static loads may be applied at each hypothesized break point with unacceptable pipe whip interactions. Clearances and inelastic effects will be considered in the analyses.

The energy balance method is based on the principle of conservation of energy. The kinetic energy of the pipe generated during the first quarter cycle of movement will be assumed to be converted into equivalent strain energy, which will be distributed to the pipe or the support. The strain in the restraint shall be limited to 50% of the ultimate uniform strain.

3.6A.2.3.4 Simplified Pipe Whip Analysis

A conservative method may be used to determine for a given rupture whether pipe whip takes place. This method is based on calculation of the minimum internal forces necessary to form a plastic hinge in the pipe, and the number of hinges required for a pipe whip mechanism.

Occurrence of a pipe whip is dependent on formation of a sufficient number of hinges to develop a mechanism. Two commonly encountered examples are:

A. Cantilever pipe with end load

$$T_{whip} = \frac{M_{ult}}{L}$$

B. Continuous pipe supported at both ends with lateral load

$$\mathrm{T_{whip}} = 2\mathrm{M_{ult}}\left(\frac{1}{\mathrm{L_1}} + \frac{1}{\mathrm{L_2}}\right)$$

Where

L, L_1, L_2 = Distance from support to load.

M_{ult} = The ultimate moment

T_{whip} = The thrust load at which pipe whip will occur.

The applied thrust load shall consider a dynamic amplification factor of 2.0 unless an analysis is performed to justify a lesser value.

3.6A.2.3.5 Pipe Whip Restraint Design

The design limits which shall be used in the design of pipe whip restraints are shown in the following table:

Type of Design	Plastic	Elastic
Loading Combination	$D+L+T_a+P_a+Y_r+Y_j+Y_m$	$D+L+T_a+P_a+Y_r+Y_j+Y_m$
Stress/strain limits	50% uniform Ultimate strain	1.5 S_m or 1.2 S_y, but not to exceed 0.7 S_u

Note: Earthquake and pipe rupture are not assumed to exist concurrently when evaluating the pipe whip restraints.

Where:

D	=	Dead load
L	=	Live
T_a	=	Thermal load resulting from postulated break
P_a	=	Pressure load resulting from postulated break
Y_r	=	Pipe restraint reactions resulting from postulated break
Y_j	=	Jet impingement load generated by postulated break
Y_m	=	Pipe whip impact load resulting from postulated break
Sm	=	Design stress - intensity
Sy	=	Yield stress
Su	=	Ultimate tensile stress

Dynamic response amplification was accounted for by multiplication of loads by appropriate dynamic factors or through use of dynamic analysis. The following dynamic load factors were used for the local structure components design.

1. For piping system with no gaps at the restraint, a dynamic load factor of 2.0 was applied regardless of pipe size.

2. For piping system with gaps not exceeding 1 inch at the restraint, a dynamic load factor of 3.0 may be applied.

3. A linear interpolation for gaps between zero and 1 inch may be made. The above dynamic factors in items 1 and 2 are applicable to small line (6-inch nominal diameter or less) without subsequent analyses. Items 2 and 3 may also be applied to large lines (larger than 6-inch nominal diameter) providing sufficient analyses are performed to show that the dynamic factor has not been exceeded.

4. For gaps in excess of 1 inch, dynamic load factors shall be justified by analyses.

3.6A.2.3.6 Energy Absorbing Materials

An energy absorbing material (crushable honeycomb) is sometimes used to absorb the kinetic energy of the ruptured pipe and to limit the loads on the restraint structure. For systems where the energy balance method of analysis is used, the kinetic energy of the pipe generated during the first quarter cycle of movement will be assumed to be converted into equivalent strain energy, which will be distributed on the pipe or the support. The actual crush shall not exceed 90% of the available crush depth.

3.6A.2.4 Guard Pipe Assembly Design Criteria

Guard pipes for penetrations are classified as TVA Class K. The chemical and mechanical tests and nondestructive examinations shall be in accordance with the ASME Material Specification. Markings and certified mill tests shall be in accordance with the requirements for process pipe. All welding shall be made in accordance with ASME Code, Section III, NC-4000. All girth butt welds shall be magnetic particle or liquid penetrant inspected in accordance with Appendix IX of ASME Code, Section III. Acceptance standards shall be in accordance with NE-5000.

The guard pipe shall be designed for the same temperature and pressure as the process pipe. However, the allowable stresses shall be 90% of yield strength (0.2% offset) at design temperature.

The guard pipe shall be designed to have its lowest natural frequency greater than 33 Hz where possible to allow the zero period acceleration to be used. Where 33 Hz is not practical, the actual frequencies expanded by 10%, shall be used in conjunction with the appropriate floor response spectra, to determine the design acceleration. The seismic loading shall be that which results from input accelerations of 1.5 g horizontal and 1 g vertical for the operating basis earthquake and twice these values for the safe shutdown earthquake.

Inservice inspections and accessibility requirements are discussed in Section 5.2.8 for ASME Class 1 systems, Section 6.6 for ASME Class 2 and 3 systems, and Section 3.8.2.7.9 for ASME Class MC and metallic liners of Class CC components. Penetration assemblies to be used for piping penetrations of containment areas are discussed in Section 3.8.2.

If circumferential ruptures or longitudinal splits are postulated in the process pipe (in accordance with Section 3.6A.2.1.2) at locations enclosed by the guard pile, the guard pipe shall be capable of mitigating the consequences of the break. If no circumferential ruptures or longitudinal splits are postulated in the process pipe, arbitrary through-wall leakage cracks shall be assumed and the guard pipe shall be capable of mitigating the consequences of the cracks.

3.6A.2.5 Summary of Dynamic Analysis Results

A letter from J. E. Gilleland to Mr. Giambusso dated May 16, 1974, submitted CEB Report No. 72-22, "Evaluation of the Effects of Postulated Pipe Failures Outside of Containment for the Sequoyah Nuclear Plant Units 1 and 2." In this letter it was stated that this report is also applicable to the Watts Bar Nuclear Plant and upon completion of the Watts Bar piping outside the containment, any differences between the Sequoyah and Watts Bar designs would be addressed in the Watts Bar UFSAR. The major differences between the Watts Bar and Sequoyah designs outside containment are the main steam and feedwater routing in the open bay area of the Control Building.

3.6A.2.5.1 Stress Summary and Isometrics - Inside Containment

The stress summary for each of the postulated break locations for the following systems larger than 4 inches in nominal size are presented in Tables 3.6-1 through 3.6-6;

Table No.	System Description
3.6-1	Main Steam Lines
3.6-2	Main Feedwater Lines
3.6-3	Auxiliary Feedwater Steam Supply Lines - Unit 1
3.6-3A	Auxiliary Feedwater Steam Supply Lines - Unit 2
3.6-4	SI Cold Leg Injection
3.6-5	RHR/SI Hot Leg Recirculation, Loop 4
3.6-6	SI Hot Leg Recirculation, Loops 1,2, and 3

Isometrics showing break type locations, protective device locations and constrained directions of the above systems are presented in Figures 3.6-2 through 3.6-17.

Inside containment the isometrics for Unit 2 are generally opposite hand to Unit 1. The stress summaries and isometrics are based on the analysis current at the time of the amendment submittal indicated on the figures and tables. This information is considered representative and presents typical historical results for Units 1 and 2.

3.6A.2.5.2 Summary of Protection Requirements and Isometrics Outside Containment

A summary of protection requirements including break types and locations for main steam and main feedwater lines are presented in Tables 3.6-9 and 3.6-10, respectively. Isometrics showing break types, locations. protective device locations and constrained directions for these lines are shown in Figures 3.6-21 through 3.6-24.

Outside containment, the break types, break locations, isometrics, protective device locations and constrained directions for Unit 2 are generally opposite hand to Unit 1. The isometrics and protection requirements reflect the analysis current at the time of amendment submittal indicated on the figures and tables. This information is considered representative and presents typical historical results for Units 1 and 2.

APPENDIX 3.6B

PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6B.1 Break Locations And Dynamic Effects Associated With Postulated Primary Loop Pipe Rupture

The dynamic effects of postulated double-ended pipe ruptures in the reactor coolant loop piping have been eliminated from the design basis of the Watts Bar Nuclear Plant by the application of leak before break technology in accordance with the final rule change to General Design Criterion 4 (Reference 12). Authorization for their elimination is provided in Reference [6] and is based on fracture mechanics analysis results presented in References [7] and [8].

The plant design bases was revised in several areas to take advantage of the elimination of reactor coolant loop (RCL) pipe breaks. The protective measures taken to mitigate the dynamic effects of these breaks remain in place. However, these protective devices no longer perform a pipe whip restraint function. See UFSAR Section 5.5 and Figures 5.5-11, 5.5-12, and 5.5-13.

In other areas, design basis analyses have been conducted based on the original postulated double-ended breaks. Even with the elimination of these dynamic effects, these analyses continue to demonstrate the adequacy and acceptability of the plant design. These analyses shall remain the analyses of record unless indicated otherwise in this safety analysis report.

Leak-before-break has also been applied to the pressurizer surge line as discussed in Section 3.6A.2.1.5.

As stipulated in the final rule change to GDC-4, a non-mechanistic double-ended rupture of the largest pipe in the reactor coolant system is still postulated for the purposes of containment design, ECCS design, and environmental qualification of electrical and mechanical equipment.

Previously postulated breaks in branch lines (except the pressurizer surge line) attached to the reactor coolant loops remain unaffected.

3.6B.2 Analytical Methods to Define Forcing Function and Response Models

The reactor coolant loop breaks used in determining the forcing functions (discussed below) and in calculating the resulting hydraulic transients and loadings have been eliminated as noted in Section 3.6B.1. However, these analyses envelope the effects of any remaining breaks, e.g., in branch lines at the loop attachment points, and as such continue to demonstrate the adequacy of the design for these loadings.

Following is a summary of the methods used to determine the dynamic response of the reactor coolant loop associated with postulated pipe breaks in the loop piping. Detailed descriptions of the methods are given in Reference [1].

In order to determine the thrust and reactive force loads to be applied to the reactor coolant loop during the postulated loss of coolant accident (LOCA), it is necessary to have a detailed description of the hydraulic transient.

Hydraulic forcing functions are calculated for the ruptured and intact reactor coolant loops as a result of a postulated LOCA. These forces result from the transient flow and pressure histories in the reactor coolant system. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flow rates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates and calculates the time history of forces at appropriate locations in the reactor coolant loops.

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

The blowdown hydraulic analysis is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid through out the reactor system. The MULTIFLEX 3.0[17] computer code was developed with this capability, which is an enhancement and extension of MULTIFLEX 1.0[2], NRC reviewed and approved computer code developed for the same space-time dependent analysis of nuclear power plants. The MULTIFLEX 3.0 features which differ from MULTIFLEX 1.0 are primarily related to vessel forces. The loop forcing functions do not differ significantly from those generated using the NRC approved MULTIFLEX 1.0 model. MULTIFLEX 3.0 has been accepted by NRC for several applications[13],[14],[15],[16] and has been extensively used for the LOCA analyses of various 2, 3 and 4 loop nuclear plants.

MULTIFLEX is a digital computer program for calculation of pressure, velocity, and force transients in reactor primary coolant systems during the subcooled, transition, and the early saturation portion of blowdown caused by LOCA. During this phase of the accident, large amplitude rarefaction waves are propagated through the system with the velocity of sound causing large differences in local pressures. As local pressures drop below saturation, causing formation of steam, the amplitudes and velocities of these waves drastically decrease. Therefore, the largest forces across the loop piping due to wave propagation occur during the subcooled portions of the blowdown transient. MULTIFLEX includes mechanical structure models and their interaction with the thermal-hydraulic system, although these features are only involved in the vessel and steam generator modeling.

The THRUST computer program was developed to compute the transient (blowdown) hydraulic loads resulting from a LOCA.

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

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

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

F = Force, lb_f
A = Actual calculated break flow area, ft^2
P = System pressure, psia
$\dot{m}$ = Mass flow rate, lb_m/sec
ρ = Density, lb_m/ft^3
g = Gravitational constant = 32.174 ft/sec^2
A_m= Mass flow area, ft^2

In the model to compute forcing functions, the reactor coolant loop system is represented by a similar model as employed in the blowdown analysis. The entire loop layout is described in a coordinate system. Each node is fully described by: 1) blowdown hydraulic information, and 2) the orientation of the streamlines of the force nodes in the system, which includes flow areas, and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

The THRUST Code is described in Reference [3].

3.6B.3 Dynamic Analysis of the Reactor Coolant Loop Piping Equipment Supports and Pipe Whip Restraints

The dynamic analysis of the reactor coolant loop piping for the LOCA loadings is described in Section 5.2.1.10.

Section 5.2 defines the loading combinations, associated with the reactor piping systems, considered to assure the integrity of vital components and engineered safety features.

REFERENCES

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

2. WCAP-8708-PA (Westinghouse Proprietary), WCAP-8709-A (Non-Proprietary), “MULTIFLEX A FORTRAN Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics,” Takeuchi, K., et al., September 1977.

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

4. 'Relaxation in Arbitrary Intermediate Pipe Rupture Requirements', NRC Generic Letter 87-11, June 19, 1987.

5. 'Two Phase Jet Loads' NUREG/CR-2913, January 1983.

6. NRC Letter to TVA, dated May 17, 1990, "Safety Evaluation of Primary Loop Piping."

7. TVA Letter to NRC dated April 17, 1989, "Elimination of Primary Loop Pipe Breaks."

8. "Technical Justification For Eliminating Large Primary Loop Pipe Rupture As The Structural Design Basis For Watts Bar Units 1 & 2", WCAP- 11984 (Non-Proprietary) and WCAP-11985 (Proprietary), November 1988.

9. NRC letter to TVA, dated April 28, 1993, "Leak-Before-Break Evaluation of the Pressurizer Surge Line".

10. TVA Letter to the NRC dated June 22, 1992 transmitting enclosures, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for WBN Units 1 and 2", WCAP-12773 (proprietary) and WCAP-12774 (nonproprietary), both dated December 1990.

11. TVA letter to the NRC dated March 26, 1993 transmitting supplemental information to the reference 10 letter.

12. Federal Register, Volume 52, Number 207, October 27, 1987, 41288.

13. WCAP-15029-P-A/WCAP-15030-NP-A, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distribution Under Faulted Load Conditions," R. E. Schwirian, et al., January 1999.

14. WCAP-15245 (Proprietary)/WCAP-15246 (Non-Proprietary), "Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook Units 1 and 2," J. A. Barsic, D. C. Garner, Y. C. Lee, K. B. Neubert, C. Yu, February 28, 1999.

15. WCAP-11004-P/WCAP-11005 (NP), "Comparison of DATA for Beaver Valley Power Station Unit 2 with WCAP-9735 Data, Prepared for NRC Review in Conjunction with Review of WCAP-9735, Docket No. 50-412," D. R. Bhandari, K. Takeuchi, M. E. Wills, November 1985.

16. WCAP-11522 (Proprietary)/WCAP-11523 (Non-Proprietary), "Response to NRC Questions on the LOCA Hydraulic Forces Analysis of the Beaver Valley Power Station Unit 2, Prepared for NRC Review in Conjunction with Review of WCAP-9735, Docket No. 50-412," D. C. Garner, M. P. Kachmar, M. R. Wengerd, June 1987.

17. K. Takeuchi, et al., "MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model," WCAP-9735, Revision 2, Proprietary, and WCAP-9736, Revision 1, Non-Proprietary, February 1998.

3.7 SEISMIC DESIGN

The original analyses of Category I structures were performed using methodologies that were prevalent prior to issuance of the Standard Review Plan (SRP) (NUREG-0800, Rev. 1). Throughout this section, the bases for these analyses are called the "Original Seismic Analysis Criteria" and analysis results (Amplified Response Spectra (ARS), forces, displacements, etc.) using these criteria are termed Set A. The plant's design basis is Set A criteria.

As a result of various seismic analysis issues identified during 1987-1989, reanalysis of some structures was necessary. The intent of the reanalysis was to demonstrate, by addressing these issues, the seismic design adequacy of structures, systems and components. Evaluations of the adequacy of existing hardware are based on SRP compatible criteria and current practices. This criteria, called the "Evaluation Seismic Analysis Criteria," includes the Site Specific Response Spectra (SSRS) developed for WBN, three-dimensional seismic models, and SRP compatible damping values. Evaluation criteria analysis results are termed Set B criteria.

In order to develop seismic input for future designs and modifications of existing designs, the Category I structures analyzed for Set B criteria were also reanalyzed using the original criteria with current modeling techniques, including soil-structure interaction. These analyses results are termed Set C.

The SRP 1981, Revision 1 formed the basis for Set B and Set C analyses, updated to the provisions of SRP, 1989, Revision 2. Specific evaluations were performed for the following:

a. The requirement of varying the soil shear modulus by +100%, -50% from the best-estimate (mean), and the best estimate soil shear modulus.

b. The limitation of hysteretic soil damping ratio to the maximum of 15%.

The seismic responses (ARS, accelerations, displacements, forces, and moments) defined by the envelope of Set B and Set C (Set B+C) are for use in new designs and modifications. New designs and modifications initiated after October 1, 1989, are based on Set B+C responses.

Underground electrical conduit banks were evaluated using Set B criteria. Conduit banks were reevaluated because the original seismic analysis was not retrievable, and the design criteria had been revised to incorporate the design requirement to consider axial loads in the analysis of conduit banks. Set B and Set C analysis were not performed for the Waste Packaging Area, (WPA), and Condensate Demineralizer Waste Evaporator Structure, (CDWE), since these two structures do not house any safety-related systems and components. Furthermore, Set B and Set C analyses were not performed for the essential raw cooling water system (ERCW) retaining walls, miscellaneous yard structures and Class 1E electrical system manholes and handholes because the seismic design input for these features is the ground motion; thus, the generation of ARS are not necessary, and there are no outstanding issues which necessitate a reevaluation. If a reevaluation of such features to resolve CAQ's etc., is required, Set B ground motion will be used in the reevaluation.

3.7.1 Seismic Input

3.7.1.1 Ground Response Spectra

Vibratory ground motions are defined by two sets of site seismic design response spectra: The Modified Newmark Ground Response Spectra or Original Site Design Response Spectra for Set A and Set C analyses and the Site Specific Ground Response Spectra for Set B (Evaluation) analyses.

3.7.1.1.1 Original Site Ground Response Spectra (Set A and Set C)

The original site seismic design response spectra which define the vibratory ground motion of the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) for rock-supported structures are shown in Figures 2.5-236a and 2.5-236b. The maximum rock acceleration for the SSE is 0.18g for horizontal motion and 0.12g for vertical motion. The OBE is equal to one-half the SSE, as outlined in Section 2.5.2.7 (historical information), with maximum horizontal and vertical rock accelerations of 0.09g and 0.06g, respectively.

3.7.1.1.2 Site Specific Ground Response Spectra (Set B)

Seismic input motions for the evaluation of existing structures, systems, and components are defined by the top-of-rock SSRS shown in Figures 3.7-4a through 3.7-4r. Peak SSE and OBE top-of-rock accelerations are 0.215g (horizontal SSE), 0.15g (vertical SSE), 0.09g (horizontal OBE), and 0.06g (vertical OBE).

3.7.1.2 Design Time Histories

3.7.1.2.1 Time Histories for Original Site Ground Response Spectra (Set A and Set C)

For time history analyses, four artificial acceleration time histories were developed so that the response spectra produced by the arithmetic average of the response spectra of each individual record envelope the site seismic design response spectra. Figures 3.7-1 through 3.7-4 show the comparison, for the various damping ratios, of these averaged response spectra and the site seismic design response spectra for the SSE. Table 3.7-1 lists the system period intervals at which the response spectra are calculated.

3.7.1.2.2 Time Histories for Site Specific Ground Response Spectra (Set B)

Set B analyses utilize three statistically independent acceleration time histories. The response spectra for these three statistically independent time histories are shown in Figures 3.7-4a through 3.7-4r. These time histories satisfy the SRP design spectra enveloping requirements.

The power spectral density function (PSDF) enveloping criteria of NUREG/CR-5347 were used to ensure adequate energy content of the artificial time histories. The PSDF enveloping criteria are that the PSDFs of artificial time histories whose response spectra envelope the 84th-percentile target response spectra should generally envelope the "minimum required" target PSDF for the corresponding non-exceedance probability level to ensure adequate motion energy contents of artificial time histories. The minimum required target PSDF is defined as the 80% of the target PSDF. The minimum required horizontal and vertical 84th-percentile target PSDFs for the Watts Bar site-specific ground motions were calculated and compared with the corresponding PSDFs of the artificial time histories as shown in Figures 3.7-4s, 3.7-4t, and 3.7-4u.

As can be seen from Figures 3.7-4s and 3.7-4t, the PSDFs of the horizontal artificial time histories envelope the corresponding minimum required 84th-percentile target PSDFs in the frequency range of 0.7 cps to 25 cps. The PSDF of the artificial time history H2 dip slightly below the horizontal minimum required 84th-percentile target PSDF in the small frequency range of 0.5 cps to 0.7 cps. This slight dip is considered inconsequential because the response spectral values of H2 time history envelope the site-specific response spectra in this frequency range and no structural frequencies of Category I structures exist in this low frequency range. Thus, the horizontal SSRS-compatible artificial time histories have adequate motion energy contents and their PSDFs satisfy the PSDF enveloping criteria proposed in NUREG/CR-5347 in the frequency range of 0.7 cps to 25 cps.

Similarly, as can be seen from Figure 3.7-4u, the PSDF of the vertical artificial time history envelope the corresponding minimum required, 84th-percentile target PSDF in the frequency range from 1.6 to 25 cps. The PSDF of the artificial time history has very slight dips below the vertical minimum required 84th-percentile target PSDF in the small frequency ranges of 0.40 to 0.42 cps, and 1.2 to 1.6 cps. These slight dips are considered inconsequential because the response spectral values of the vertical time history envelope the site-specific response spectra in these frequency ranges and no structural frequencies of Category I structure exist in this low frequency range. Thus, the vertical SSRS-compatible artificial time history has adequate motion energy contents and its PSDF satisfy the PSDF enveloping criteria proposed in NUREG/CR-5347 in the frequency range of 1.6 to 25 cps.

3.7.1.3 Critical Damping Values

The specific percentages of critical damping values used for Category I structures, systems, and components are provided in Table 3.7-2 for Sets A, B, and C.

3.7.1.4 Supporting Media for Seismic Category I Structures

A complete description of the supporting media for each Seismic Category I structure is provided in Section 2.5.4 (historical information). Pertinent data concerning the supporting media for Set A, B, and C analyses of each Seismic Category I structure is also given in Table 3.7-3.

3.7.2 Seismic System Analysis

This section describes the seismic analysis performed for Category I structures.

3.7.2.1 Seismic Analysis Methods

The seismic methods of analysis used for the Category I structures listed in Section 3.2.1 are described in the following sections.

3.7.2.1.1 Category I Rock-Supported Structures - Original Analyses (Set A)

The seismic analyses of Category I structures were based upon dynamic analyses using the lumped mass normal mode method with idealized mathematical models. The inertial properties of the models were characterized by the mass, eccentricity, and mass moment of inertia of each mass point. Mass points were located at floor slabs, changes in geometry, and at intermediate points to adequately model the structure. The stiffness properties were characterized by the moment of inertia, area, shear shape factor, torsion constant, Young's modulus, and shear modulus. Significant modes of vibration were considered in determining the total response. For structures with built-in asymmetry and open structures which have low torsional resistance, coupled translation and torsion were included in the dynamic analyses. Torsional effects for the closed structures with small eccentricities have insignificant effect on the responses. To demonstrate this, a dynamic analysis study of the steel containment vessel, including an accidental eccentricity of 5% of the diameter, showed that the induced torsion had a negligible effect on the acceleration response spectra. Structural response was calculated in both the east-west and north-south directions except where symmetry justifies analyses in one direction. The effect of the vertical component of earthquake motions on the structural response was included.

For structures surrounded by soil, the effect of the soil stiffness on the structural response was determined by replacing the soil with springs of equivalent stiffness. Due to seismic motion, the soil pressure against structures was increased above the static soil pressure. The magnitude of this increase was determined by using the shaking table experiments performed for the design of TVA's Kentucky Hydro Project.[1] For a ground acceleration of 0.18g the static soil pressure was increased by 46% for a dry fill and 22% for a saturated fill. This incremental increase was combined with the static pressure as a triangle of pressure whose apex is at the rock surface and maximum ordinate is at the ground surface. In addition to the soil pressure increase as described above for a saturated fill, the hydrostatic pressure of water within the fill was increased 22%. This incremental increase was combined with the static water pressure as a triangle of pressure whose apex is at the water surface and maximum ordinate is at the rock surface or bottom of structure. Calculations using the shaking table experiment results have been confirmed using information in Reference [2]. A more detailed description of the seismic analyses of Category I rock-supported structures is discussed below.

The in situ measured shear wave velocity of the bedrock upon which the structures are founded has an average value of 5,900 feet per second (Section 2.5.4.8, historical information). Therefore, the effect of structure-foundation interaction was investigated for the major structures. The results of the investigation are discussed below as one of the parameters associated with the analysis of those structures.

The structural response was computed using the response spectrum modal analysis method. The techniques used to account for the three components of motion and the method of combining modal responses when computing the structural response of a structure are explained in Sections 3.7.2.6 and 3.7.2.7, respectively.

Response spectra were produced by the time history-modal analysis method using the four artificial accelerograms discussed in Section 3.7.1.2 and the techniques of Sections 3.7.2.5 and 3.7.2.9.

When torsion is considered, accelerations and deflections were calculated at the farthest points on the structure from the shear center, on the axis perpendicular to the direction of motion. The moment and shear due to earthquake motion were used in combination with other appropriate loads to determine overturning moments.

The response was calculated for both the OBE and the SSE, except when the same percentages of critical structural damping were specified for both earthquake levels, in which case the response was calculated for the OBE only (the SSE results are twice the OBE results). For applicable stress criteria, see Section 3.8.

The damping ratios used in the dynamic analyses of the structures are given in Table 3.7-2.

To ensure that the results of the seismic analysis of the structures were used in the design, the analyses became part of the nuclear plant design criteria and were submitted to the design sections responsible for design and to the principal engineer. For more detailed procedures and criteria of design control measures, see Section 3.8.

<u>Shield Building</u>

Two separate, distinct analyses were performed on the reinforced concrete structure to determine the response of the structure to horizontal motion when modeled as a cantilever beam and the response of the dome to vertical motion when modeled as a shell.

The Watts Bar Shield Building is identical to the Shield Building at TVA's Sequoyah Nuclear Plant. The building has been assumed to have identical structural properties in both the east-west and north-south directions. A sketch of the lumped mass model is shown in Figure 3.7-5 and the structural properties are listed in Table 3.7-4. The dome was considered a rigid body and its weight added at mass point 25. The dynamic analyses in both the horizontal and vertical directions was done by the normal mode response spectrum method. Although no structural eccentricities exist in the building, an accidental eccentricity of 5% of the diameter was assumed in the design. Periods for the normal modes of vibration are listed in Table 3.7-5.

Since torsion is considered, the maximum structural accelerations and deflections will not occur at the center of mass but rather at the point on the structure farthest from the shear center. For the Shield Building the shear center is located at the geometric center. Accordingly, all structural accelerations and deflections as well as the floor response spectra have been computed at a point located on the shell wall. Structural responses were calculated for both the OBE and SSE using structural damping of 2 and 5 percent, respectively.

Foundation-structure interaction studies were performed to determine the response characteristics of the Shield Building steel containment-interior concrete system to rocking-type motion. These analyses were performed considering lumped-mass models of the structure coupled with foundation springs. These springs were calculated as detailed by Whitman.[3] The results of these investigations indicated that the Shield Building accelerations would increase by less than 15% compared to the accelerations of a rigid base, single structure system. As a result, all spectra used to compute structural response and all accelerograms used to compute floor response spectra were multiplied by a factor of 1.15. The site response spectra for structures without rocking have previously been shown in Figures 2.5-236a and 2.5-236b. The effects of the soil which partially surround the building were investigated for the Sequoyah Nuclear Plant[4] and the effects are negligible.

Floor response spectra were computed for four individual artificial earthquakes (increased in amplitude by 15%) and the result found by taking the arithmetic mean of the four analyses. Spectra were computed for damping ratios of 0.005, 0.01, and 0.02 for the OBE and 0.005, 0.01, 0.02, 0.030, 0.040, and 0.050 for the SSE.

Vertical modes of vibration were calculated for comparison with the results for the dome as a shell. The rigid-body simulation of the dome as performed in the analysis of the cantilever beam model does not provide an accurate representation of the response of the dome to vertical earthquake excitation. Thus, an analogy was developed using shell theory to determine the earthquake moments and forces in the dome.

Figure 3.7-6 illustrates the logic performed in the analysis. The shell model is shown in Figure 3.7-7.

A flexibility matrix was developed using the shell model and the analysis performed using the response spectrum modal analysis techniques. The modes involving primarily deformation of the cylinder as computed in this analysis (modes 1 and 5) compare favorably with modes 1 and 2 of the vertical lumped mass cantilever beam analysis as shown in Table 3.7-5 (periods agree within 3%). Modes 2, 3, and 4 are primarily modes of vibration involving the dome. Also, the total meridional force at the base of the building as calculated by this method compares closely with the total force at the base in the cantilever beam analysis. This indicated the appropriateness of the analogy.

The structural response for the shell model was calculated for both the OBE and the SSE using structural damping of 2% and 5%, respectively.

Interior Concrete Structure

The idealized lumped mass model of the reinforced concrete structure used in the dynamic earthquake analysis is shown in Figure 3.7-8. Element properties are given in Table 3.7-6 and mass point properties in Table 3.7-7. The foundation structure interaction analysis of the Shield Building interior concrete-steel containment system discussed above for the Shield Building analysis revealed no significant change in the response of the interior concrete structure as compared to the response assuming a fixed base. Therefore, the dynamic analysis was performed using a fixed base model.

The dynamic earthquake analysis was performed by the response spectrum modal analysis technique. The results were computed for both the OBE and SSE conditions with structural damping of 2% and 5% respectively. The effects of torsion and longitudinal motion were considered. Periods for the normal modes of vibrations are listed in Table 3.7-8.

Response spectra were produced for damping values of 0.005, 0.01, and 0.02 for the OBE at mass points 1, 2, 3, 4, 6, 8, 10, 12, 13, and 14 for motion in both the east-west and north-south directions. Response spectra for vertical motion were obtained at ground and at mass point 14, and linear interpolation (Section 3.7.2.5) was used to produce vertical spectra at intermediate mass points. Response spectra were produced for damping values of 0.005, 0.010, 0.020, 0.050, 0.100, and 0.150 for the SSE at mass points 3, 5, 6, 8, 9, 10, 11, and 14 for motion in both the east-west and north-south directions. Response spectra for vertical motion were produced at mass point 14 and ground. Linear interpolation (see Section 3.7.2.5) was used to obtain vertical response spectra at intermediate points.

Auxiliary/Control Building

The idealized lumped mass model of the reinforced concrete structure is shown in Figure 3.7-9. Foundation-structure interaction was investigated by using a lumped mass-rock spring model, as discussed in Section 3.7.2.4. The results verified that a fixed base analysis may be used with no loss in accuracy. The dynamic analysis was performed by the response spectrum modal analysis technique. The results were computed for the OBE condition, and results for the SSE were obtained by doubling the values from the OBE. Element properties for the fixed base model are given in Table 3.7-9 and mass point properties in Table 3.7-10. Contributory weights to account for the soil contained within the wing walls at the north end of the structure were included in the total weights of the appropriate mass points. The effects of torsion and longitudinal motion were considered. Periods for the normal modes of vibrations are listed in Table 3.7-11.

Steel Containment Vessel

The containment vessel dynamic seismic analyses were performed using a lumped mass beam model. Structural and equipment masses were included, and structural properties were computed by hand calculations. The beam model and its properties are shown in Figure 3.7-7B.

Maximum overturning moments, shears, deflections, and shell stresses were computed by the response spectrum method. The site seismic design response spectra for 1% damping described in Sections 2.5.2.6 and 2.5.2.7 (both historical information) were utilized. The analyses were performed by CBI proprietary computer program 1017 described in Appendix 3.8C. Total response was computed by taking the absolute sum of modal responses.

A time-history analysis using the model in Figure 3.7-7B was performed in order to develop response spectra for equipment attached to the containment vessel. Four artificial earthquakes having an averaged response spectrum greater than the design response spectrum provided the seismic input. Using each of the artificial earthquakes individually, the beam model was analyzed and histories of acceleration were generated. For each of the acceleration histories, response spectra for various mass points and values of assumed damping were generated. The design spectra were the envelopes of the spectra generated from the four earthquakes and were used to design the vessel and the vessel's appurtenances in the scope of CBI, the vessel designer, fabricator, and erector. These calculations were performed by CBI computer programs 1017, 1044, and 1668, all of which are described in Appendix 3.8C.

As part of the review process and to provide response spectra for the design of equipment, piping and subsystems attached to and or supported by the containment vessel not supplied by the CBI, TVA performed an independent dynamic seismic analysis of the containment vessel. The ground motion input used to generate the floor response spectra consisted of the same four accelerograms of artificial earthquakes used by CBI.

The containment was idealized as a beam-type model consisting of lumped masses connected by massless elastic members. This lumped mass model is shown in Figure 3.7-7C. The element properties and inertial properties which were used in the analysis are shown in Table 3.7-5A and Table 3.7-5B, respectively.

<u>North Steam Valve Room</u>

The idealized lumped mass model of the reinforced concrete structure is shown in Figure 3.7-10. The structure is founded on bedrock and partially imbedded in soil. The effect of the soil restraint on the seismic response of the structure was included in the lumped mass model as soil spring restraints. The soil springs were calculated in accordance with the methodology given in Section 3.7.2.1.3. Element properties are shown in Table 3.7-12 and mass point properties in Table 3.7-13.

The dynamic analysis was performed using the response spectrum modal analysis technique. Response spectra were produced for selected elevations within the structure by the time history modal analysis method. Results were computed for the OBE with results for the SSE obtained by doubling those for the OBE. The frequencies for those modes considered important to the response of the structure are listed in Table 3.7-14.

Essential Raw Cooling Water Intake Pumping Station

The idealized lumped mass model of the reinforced concrete structure used in the analysis is shown in Figure 3.7-11. The dynamic analysis was performed by the response spectrum modal analysis technique. The results were computed for both the OBE and SSE with an assumed structural damping of 5% for each earthquake. Element properties are given in Table 3.7-15 and mass point properties in Table 3.7-16. The effects of torsion and soil restraint were considered. Periods for normal modes of vibration are listed in Table 3.7-17.

In addition, the effect on the building response of various water levels inside the pump wells was studied. The results of this study showed that the natural period of vibration was affected by variations in water level. Therefore, the structural responses used for design were those for the "worst-case" conditions. The amplitude of the response spectra peaks was not significantly affected by the water level variations. Only the location of the peak changed as the natural periods changed in response to water level variations. Accordingly, the response spectra peaks were broadened to account for the range of variations in natural period.

The response spectra for horizontal motion were produced for damping ratios of 0.005, 0.01, and 0.02 for both the OBE and SSE at mass points 6, 7, 8, and 10. Response spectra for vertical motion were produced for the base and mass points 8 and 10. Vertical spectra for intermediate mass points were developed using linear interpolation, as outlined in Section 3.7.2.5.

Essential Raw Cooling Water Intake Pumping Station-Retaining Walls

The reinforced concrete retaining walls were designed as a rigid structure subjected to the top of rock acceleration. Dynamic soil pressures on the retaining wall was determined in accordance with Reference [1].

3.7.2.1.2 Category I Rock - Supported Structures - Evaluation and New Design or Modification Analyses (Set B and Set B+C)

Analysis methodologies used in the original analyses (Set A) of Category I rock-supported structures were used in Evaluation (Set B) and New Design/Modification (Set B plus Set C) analyses except as noted in the remainder of this subsection. These exceptions provide for a seismic modeling approach which is consistent with current SRP Subsection 3.7.2 (NUREG 0800, Rev. 1) requirements.

Structures were represented as three-dimensional lumped mass stick models in the analyses except when coupling effects from omitted degrees of freedom were not significant. Actual centers of rigidity and actual mass eccentricities were modeled. Sufficient numbers of modes were included in the response to assure participation of at least 90% of the total mass. The simultaneous effects of three components of seismic input were considered by combining the co-directional responses resulting from the three components of input either by algebraic summation (for simultaneous inputs) or by the square-root-of-the-sum-of-squares (SRSS) method.

For design of structural elements, calculated seismic torsional moments were increased to account for accidental torsion. This increase is determined by multiplying the story shear force by the accidental eccentricity (defined as $\pm$ 5% of the structure dimension perpendicular to the direction of excitation.)

Rock-supported structures (ACB, IPS) were modeled as fixed base structures except where rock-structure interaction (Reactor Building) or structure-surrounding soil interaction (NSVR) effects were important. In these cases, three-dimensional finite element analysis was used to account for the interaction effects. The analyses were based on a foundation rock shear wave velocity of 5900 fps (Section 2.5.4.8, historical information). For rock-supported structures deeply embedded in soil, the effect of soil-structure interaction was considered and, where significant, included in the analysis model. The soil stiffness was determined as discussed in Subsection 3.7.2.1.4.

For WBN Unit 1 to support replacement of the steam generators, the same basic coupled model analytical approach used previously was adopted, while applying Replacement Steam Generator (RSG) parameters in the NSSS Reactor Coolant Loop (RCL) model. There was no change in the RCL seismic support arrangement. The RCL model was adjusted to account for an increase in the RSG mass and a lower center of gravity. The accuracy of the model and the Set B time history input motion was improved some, consistent with SRP guidance.

To support replacement of the WBN Unit 1 steam generators, the WBN Unit 1NSSS RCL was reanalyzed using the basemat Set B+C response spectra. Also, piping attached to the RCL was reanalyzed or evaluated for RCL Set B+C Amplified Response Spectra (ARS) and displacements. No piping or pipe support modifications were identified due to ARS or displacement changes. A comparison of the original Set B and B+C ARS with the new ARS was performed to determine whether or not reanalysis was required for other structures, systems or components in the Reactor Building. It determined that no other reanalysis was required due to ARS or displacement changes.

Structural damping values used in Set B and Set C analyses are given in Table 3.7-2. Where necessary, element associated damping was converted to modal damping using the strain energy composite modal damping approach. Damping values used in the new RCL analysis supporting the WBN Unit 1 steam generator replacementare consistent with the SRP, Regulatory Guide 1.61, and Table 3.7-2. The new RCL analysis damping values are for Set B+C instead of Set B, because replacement of the steam generators constitutes a significant modification.

Reactor Building Rock-Structure Interaction Analysis

The Reactor Building consists of the Shield Building (SB), Steel Containment Vessel (SCV), and interior concrete structure (ICS) including the NSSS piping, equipment, and components. The Set B and Set C Reactor Building model is a three branch, three dimensional (3-D) lumped mass model with branches representing the ICS, SB, and SCV. The ICS model was developed from a finite element analyses whereas the SB and SCV models were Set A models updated to include, in the vertical model, the fundamental vertical drumming mode of the dome for each of these structures. The ICS model used for the analyses also includes the NSSS model.

The Reactor Building is partially embedded in soils below the finished grade at Elevation 728.0 and in foundation rock below Elevation 702.78. In order to take into account the embedment effect on seismic responses, rock-structure interaction analyses were performed for the Reactor Building using the 3-D SSI analysis-computer program SASSI.

For the seismic response analysis, the input ground motion input was prescribed at the surface of the rock foundation (Elevation 702.78). This is the elevation of the top of the Reactor Building basemat where base fixity was provided for the structural models used in the original design basis seismic analysis (Set A). For rock-structure interaction analyses, the structural models for SB, SCV, and ICS were coupled together through the common Reactor Building basemat. The embedment of the Reactor Building basemat in the rock foundation was considered.

Using the SASSI computer program, time history response analyses for the Reactor Building were performed in the frequency domain using the Fast Fourier Transform (FFT) method.

Shield Building (SB)

The Set B and Set C dynamic model for the axisymmetric Shield Building structure is represented by a 3-D lumped mass single stick (Figure 3.7-5A) having the center of mass coincident with the center of rigidity for each lumped mass elevation. The model consists of 25 lumped masses interconnected with 25 elastic beam elements and a single-degree-of-freedom (SDOF) system located at the dome spring line elevation (Elevation 852.0) for simulating the fundamental vertical drumming mode of the dome. Except for the vertical SDOF system for the dome and the concrete modulus, the model configuration, lumped masses, and elastic beam element properties are the same as those used in the original design basis seismic analyses (Set A analysis). The vertical SDOF system for representing the fundamental vertical drumming mode of the dome was developed by matching the frequency and effective modal mass of the SDOF system with those of the fundamental vertical drumming mode of the dome obtained from a separate finite element modal analysis for the dome. The model geometry, lumped masses, and elastic beam element properties for SB used for Set B and Set C analyses are summarized in Table 3.7-4A.

Interior Concrete Structure (ICS)

The ICS consists of a complex assemblage of curved walls, columns and slabs which have some cross sections with significant asymmetry. In order to develop a seismic model, static, 3-D, finite element analyses were performed to determine the equivalent beam properties that simulate the seismic responses of the ICS. Consistency of equivalent stick model properties and response transfer functions with those of the finite element model demonstrated the adequacy of the 3-D equivalent stick model.

Since the equivalent beam model results in center-of-rigidity locations for axial and bending deformations different from those for shear and torsional deformations, the 3-D stick model for the ICS was represented by a combination of two sticks. One stick consists of elements with only axial areas of the structure located at the centers of rigidity for axial and bending deformations and another stick consists of elements with all other beam element properties, except the axial area, located at the centers of rigidity for shear and torsional deformations. The final configuration of the 3-D stick model for the ICS is shown in Figures 3.7-8A and 3.7-8B.

Mass and member element properties are summarized in Tables 3.7-6A and 3.7-6B. Mass

properties are unchanged from those of the original analysis (See Table 3.7-7).

Steel Containment Vessel (SCV)

The dynamic model for the SCV Set B and Set C analyses is represented by a 3-D lumped mass, concentric single stick model as shown in Figure 3.7-7A. The model consists of 23 lumped masses interconnected with 23 elastic beam elements and a vertical SDOF system located at the dome spring line elevation (Elevation 814.5) to represent the fundamental vertical mode of the dome. Mass and member element properties are defined in Table 3.7-5C. Except for the mass eccentricities and the SDOF vertical model, the model configuration, lumped masses, and elastic beam element properties are the same as those used in the original (Set A) design basis seismic analyses described in Section 3.7.2.1.1.

During the analysis of Set B and Set C, it was determined that the 5% accidental eccentricity will yield much higher eccentric responses than from the actual eccentricities which were used in Set A analysis. Therefore, the actual eccentricities were neglected in Set B and Set C analyses. However, 5% accidental eccentricity was used to calculate torsional moments. The SDOF vertical dome model for SCV was developed by matching the frequency and effective modal mass of the SDOF system with those of the fundamental vertical mode of the dome obtained from a separate finite element modal analysis.

Nuclear Steam Supply System (NSSS) Components

For Set B and Set C analyses, the dynamic model for the NSSS components is coupled with the Interior Concrete Structure (ICS) model, and the coupled model is used for seismic response analyses. The dynamic model for the NSSS components included in the coupled model for the ICS consists of the models for the Reactor Pressure Vessel (RPV), four primary reactor coolant loop piping (hot legs, cold legs, and cross-over legs), the steam generator (SG), and the reactor coolant pump (RCP) associated with each loop, as shown in Figures 3.7-8C through 3.7-8G. In coupling the NSSS model to the ICS stick model, the RCL attachment points are connected to the ICS model at the appropriate elevations of the attachment points through rigid links. The dynamic model data for the NSSS components are obtained from Westinghouse Electric Corporation.

Due to the presence of gaps and tension-only tie rods at the NSSS supports, these supports exhibit nonlinear behavior under dynamic loading conditions. For the purpose of linear response analysis, four linearized NSSS analysis cases, each with a unique set of linearized NSSS support stiffness, are used to represent the nonlinear behavior under various dynamic loading conditions. For each NSSS analysis case, a specific set of NSSS supports with their specified orientations are activated for a particular loading condition and linear support stiffnesses are developed and provided by Westinghouse Electric Corporation to represent the active supports for application to the particular analysis case.

The final acceleration response spectra and movements values are the envelope values resulting from the different NSSS cases. Furthermore, the ARS and movement values at the corresponding locations of the four loops are enveloped to obtain the enveloped ARS and movements applicable to all four loops.

The seismic analysis of the NSSS components which was performed by Westinghouse is discussed in Section 5.2.1.10.

Auxiliary Control Building (ACB)

The Set B and Set C three-dimensional lumped parameter fixed-base model of the Auxiliary Control Building is shown in Figure 3.7-9A. The centers of mass and centers of rigidity were modeled at their actual geometric locations as defined in Table 3.7-9A. The element properties and masses are unchanged from the original analysis, except for the concrete shear modulus, and are listed in Tables 3.7-9 and 3.7-10.

The dynamic analysis was performed by the time-history modal analysis technique. Structural responses were computed and floor ARS were generated for the same elevations as Set A. For Set C, since the structure damping ratios for OBE and SSE are the same (5%), the OBE responses were computed and the SSE responses were obtained by doubling the OBE responses. Separate OBE and site-specific SSE analyses were performed for Set B using structure damping ratios of 4% for OBE and 7% for site-specific SSE.

Essential Raw Cooling Water (ERCW) Intake Pumping Station (IPS)

The ERCW IPS original analysis model is updated to consider torsional effects. It incorporates rotatory inertia and the eccentricities between the centers of mass and centers of rigidity. No lateral soil springs were included as these had been determined from previous analyses to produce a negligible soil-structure interaction effect. The highest water level was used for both the Set C SSE and OBE earthquakes and Set B site-specific SSE and OBE earthquakes, since this condition yields the lowest frequency and hence would produce the highest response levels. The Set B and Set C IPS model is shown in Figure 3.7-11A. Table 3.7-15A presents the element properties. Tables 3.7-16A and 3.7-16B define the weight properties and coordinates of centers of mass and centers of rotation, respectively.

North Steam Valve Room (NSVR)

To account for the soil-structure interaction effects due to the presence of backfill surrounding the foundation walls, soil-structure interaction (SSI) analyses were performed for the NSVR. The methodology used for SSI analysis is the same as that used for Category I soil-supported structures described in Section 3.7.2.1.4.

The three-dimensional lumped mass model used in the seismic analysis of the NSVR superstructure is shown in Figures 3.7-10A and 3.7-10B, and the model properties are given in Tables 3.7-13A and 3.7-13B.

3.7.2.1.3 Category I Soil-Supported Structures - Original Analysis (Set A)

For structures founded on soil, the acceleration at top of rock was amplified or attenuated through the soil deposit using the techniques outlined in Section 3.7.2.4. The soil-supported structures were analyzed using lumped-mass and soil spring modeling techniques. A typical

model is shown in Figure 3.7-12.

Table 3.7-3 contains a tabulation of Seismic Category I soil-supported structures for the plant (small miscellaneous structures are not included in the table). Details of the supporting media and foundation characteristics are presented in Table 3.7-3 and Section 3.7.1.4. The horizontal and vertical translational soil springs and the rocking soil spring included in the lumped-mass model to simulate soil structure interaction are calculated using the procedures outlined by Whitman.[3] The damping ratio used for soil-supported structures depends on the predominant type of motion, as explained by Richard,[5] but is not permitted to exceed 10% in any case.

Embedment effects are accounted for by constructing a translational soil spring using Whitman's vertical spring expressions and attaching it to appropriate point or points on the structure. Specific features associated with the seismic analysis of the Category I soil-supported structures are discussed below.

<u>Diesel Generator Building</u>

The idealized lumped mass model of the reinforced concrete structure used in the analysis is shown in Figure 3.7-13. Element properties are given in Table 3.7-18 and mass point properties in Table 3.7-19. The effects of horizontal translation and rocking of the base were considered.

The soils investigation of Section 2.5.4 (historical information) revealed a soils profile from bedrock consisting of a firm silty gravel overlain by lean clays, silt of low plasticity, and sandy silt. In order to assure a firm foundation for the structure, the material between the top of firm gravel and the grade slab (a depth of approximately 17 feet) was excavated and replaced with compacted granular fill.

The Diesel Generator Building is founded on granular fill overlying firm gravel (see Figure 2.5-226). The shear wave velocity for the foundation material was determined to be 1650 ft/s and was used to calculate the value of the soil springs for the lumped-mass soil-structure interaction model. A parametric study was conducted to investigate the effects on building response of varying the shear wave velocity of the foundation material from 1150 ft/s to 2150 ft/s. The parametric study resulted in the structure being designed for earthquake loads from the peak of the amplified response spectrum for surface motion.

The predominant motion of the structure was a translatory rigid body motion. Motion of this type results in large damping; therefore, a damping ratio of 0.10 was used for the analysis. Longitudinal motion was also considered. Periods for the normal modes of vibrations are listed in Table 3.7-20.

Response spectra were produced for damping ratios of 0.005, 0.010, 0.020, 0.050, and 0.070 for mass points 1, 3, and 6 for motion in both east-west and north-south directions.

<u>Waste Packaging Area (WPA)</u>

The following two paragraphs describe the original design basis analysis for using Set A criteria performed for the WPA.

The idealized lumped mass model of the reinforced concrete structure is shown in Figure 3.7-14. Element properties are given in Table 3.7-21 and mass point properties in Table 3.7-22. The analysis indicated that the primary motion of the structure was in rocking and translation of

the base. Motion of this type results in large damping; therefore, a damping ratio of .10 was used in the analysis. Longitudinal motion was also considered. Periods for the normal modes of vibration are listed in Table 3.7-23.

Due to the extent of excavation for the Auxiliary Building and the results of the investigation for the Diesel Generator Building, all in situ material down to the top of rock was excavated and replaced with compacted granular fill for the WPA (see Figure 2.5-225). The shear wave velocity for the material was determined to be 1650 ft/s, and was used to calculate the value of the soil springs for the lump-mass soil-structure interaction model. A parametric study was conducted to investigate the effects on building response of varying the shear wave velocity from 1150 ft/s to 2150 ft/s. The parametric study resulted in the structure being designed for earthquake loads from the peak of the amplified response spectrum for surface motion. Additional studies beyond those described above have been performed to determine relative displacements between the WPA and the Auxiliary and Control Buildings.

Refueling Water Tanks and ERCW Pipe Tunnels

The refueling water tank and foundations were designed for seismic loads determined from the basic procedure outlined above. Soil property variations were considered in order to define conservative design loads, and ten-percent damping was used because of predominant translational soil spring motion. The adequacy of the design was later verified by more exact analytical techniques for soil-structure and fluid-structure interaction.

Pipe tunnels are analyzed as discussed under "Underground Electrical Concrete Conduit Banks" except axial loads are not considered due to the segmented configuration of the tunnel. Dynamic soil pressures on the walls are determined in accordance with Reference [1].

Underground Electrical Concrete Conduit Banks

The underground electrical concrete conduit banks which lead from the Auxiliary Building to the Diesel Generator Building and the Intake Pumping Station were seismically analyzed.

Utilizing the average values for the soil shear wave velocity and density, the ground deformation pattern in terms of wave length and amplitude is determined. The buried conduit banks are assumed to deform along with the surrounding soil layers. The average shear wave velocity of a single layer representation of a multi-layered soil system may be determined by:

$$V_{ST} = \sum \frac{V_s \, h'}{h}$$

Where,

V_{ST} = Average shear velocity in the soil, ft/sec
V_S = Shear velocity in each layer of soil, ft/sec
h' = Depth of each layer of soil, ft
h = Total depth of soil, ft

The fundamental period of the single layer is calculated from the following equation:

$$T = \frac{4h}{V_{ST}}(\text{seconds})$$

If the depth of the soil layer varies over the distance traversed by the buried conduit bank, both cases, for maximum and minimum depths, are considered.

The maximum amplitude of the sine wave which represents the maximum displacement of the conduit bank is:

$$A = \text{Displacement} = \left(\frac{T}{2\pi}\right)^2 * (\text{Accel})$$

Where,

T = Fundamental period, sec
Accel = Amplified soil acceleration value, in/sec^2

The wave length, L, is calculated as:

$$L = V_{ST}\, T$$

The bending moment resulting from the seismic disturbance, assuming the conduit bank follows the soil and deforms as a sine wave, is given by:

$$M = \frac{\pi^2 EIA}{(L/2)^2}$$

where,

M = Maximum bending moment, in-lb
E =Modulus of the conduit bank, psi
I = Moment of inertia of the conduit bank, in^4
A =Maximum amplitude, in
L =Wave length, in

The axial strain experienced by the conduit banks due to deformation of the soil is also evaluated. The axial strain due to seismic propagating waves is computed following the methods of Newmark, [15] [16] Yeh,[18] and Kuesel[17] which assume the soil is linearly elastic and homogenous, the conduit bank behaves as a slender beam, and the buried member deforms with the surrounding soil (this implies the strain in the soil equals the strain in the member).

The effect of soil strain from a seismic event on conduit bank turns in a buried system must be analyzed in greater detail than just calculating the axial strain. The effect of these strains on turns is more complex due to the turn trying to resist the strain. The complexity is a function of the conduit bank and backfill soil properties.

The basis for determining the effect of the strains on the conduit bank turns is described by Shah and Chu.[19] The Shah and Chu theory has been developed into an analysis procedure by Goodling.[20,21,22] The committee on Seismic Analysis of the ASCE Structural Committee on Nuclear Structures and Materials prepared a report "Seismic Response of Buried Pipes and Structural Components"[23] which explains and amplifies the referenced methodology[19] and analysis procedure.[20,21,22] These references shall be used for analysis of the effects of axial strain on buried conduit bank turns.

The magnitude of friction acting on the conduit banks to use in the analysis depends on several factors, such as surface condition, contact pressure, soil strengths, etc. The friction force acting on the conduit banks is determined in accordance with Reference [24].

Differential movement due to soil settlement or displacement during a seismic event was also evaluated in accordance with criteria given in Sections 2.5.4.10 and 2.5.4.8 (both historical information), respectively.

The conduit banks were evaluated for settlement due to the potential liquefaction of the underlying soil as discussed in Section 2.5.4.8 (historical information) (see Figures 2.5-576 through 2.5-578 for the potential settlement values). The banks were evaluated for potential settlements between manholes and at building/conduit interfaces. The only area of potential structural inadequacy was at the Intake Pumping Station (IPS). The conduit banks in this area (see Figure 3.8.4-46) required modification to accommodate the potential settlements. This modification consists of cutting 10 grooves on all 4 sides of the banks. The 2 inch wide by 2 inch deep grooves on top and sides and 3 inch wide by 2 inch deep grooves on the bottom begin 76 feet from the IPS and are spaced at 8 inch between centers for a distance of 6 feet along each bank. Settlement of the conduit banks will cause plastic hinges to develop at the grooves and at the pile supports farthest from the IPS. This results in a structural mechanism which will allow the conduit bank to settle without compromising the intended function of the encased conduits.

Class 1E Electrical Systems Manholes

These manholes are rigid structures which have the same motion as the soil deposits in which they are located. The soil deposits were analyzed as explained in Section 3.7.2.4. The accelerations obtained for the soil deposit at the level of the manholes were used to determine the inertia force on the structures and to calculate the increase in the static soil pressure using the shaking table experiments performed for the design of TVA's Kentucky Hydro Project,[1] as discussed in Section 3.7.2.1.1.

Miscellaneous Yard Structures

The ERCW discharge overflow structure, ERCW standpipe structures I and II, and other miscellaneous yard structures are normally rigid structures. These structures are designed for a rigid body acceleration. Dynamic soil pressures on the walls, if appropriate, are determined in accordance with Reference [1].

Structure Interaction Analysis - WPA, CDWE, and ACB

In the WPA Original Analysis (Set A) a decoupled, two-stage SSI analysis was used to determine conservative structural responses. An analysis, using the Set A Criteria and revised soil properties, confirmed that there is sufficient gap between the WPA and ACB to preclude

impact during a seismic event. For the CDWE, the Set A analysis was based on engineering judgments relating to the modeling of the supporting piles and on the assumption of full contact between the building's mat foundation and underlying soil. Additional analysis was performed to more accurately consider the stiffness of the pile groups and the postulated gap between the slab and soil. Results of this analysis confirmed that the gap between the buildings is sufficient for seismic separation and the design of the structure and piles is adequate.

3.7.2.1.4 Category I Soil-Supported Structures - Evaluation and New Design/Modification Analysis (Set B and Set B+C)

For Category I structures founded upon soil, the top-of-rock motions were considered to be amplified (or attenuated) through the soil. The value of amplification and the change in frequency content of the excitation were determined by a soil column analysis that incorporates strain-dependent soil properties. The soil properties were varied by the amount given in Tables 2.5-17A through 2.5-17D to obtain different soil surface motion time histories associated with mean, upper bound and lower bound shear moduli and bulk modulus for the horizontal and vertical analyses, respectively. For vertical motion, strain-compatible soil properties determined from the horizontal analysis were used. Using these surface motions as control motions, OBE and SSE and site-specific SSE and OBE Soil-Structure Interaction (SSI) analyses were performed and structural responses including floor acceleration time histories were obtained. The SSI analyses were performed using a 3-D flexible-volume substructuring technique and Fast Fourier Transform (FFT) method. From the floor acceleration time histories, ARS were developed. For Set C, the responses obtained from the four time history analyses were averaged. For Set B, the co-directional responses from the three component earthquake excitations were combined using the SRSS method for each of the three soil property cases. Responses from these three soil property cases were enveloped for Set B and C.

Details of the supporting media and foundation characteristics to be used in Set B and Set C analysis of Category I soil-supported structures are discussed in Section 2.5 (historical information). Additional details of seismic analyses specific to each of the Category I soil-supported structures are described in the following paragraphs.

Diesel Generator Building

The 3-D lumped parameter model used for the Diesel Generator Building is shown in Figures 3.7-13A and 3.7-13B, and the associated model properties are given in Tables 3.7-19A and 3.7-19B.

Refueling Water Storage Tank

The hydrodynamic effects were modeled considering the effects of tank flexibility. The 3-D lumped parameter model of the refueling water storage tank is shown in Figure 3.7-13C, and the associated model properties are given in Table 3.7-19C.

Waste Packaging Area

The waste packaging area does not house any safety systems and components. Therefore, Set B and Set C analyses were not performed.

ERCW Pipe Tunnels

Since the tunnels are embedded in soil, their response follows the response of the surrounding soil medium. Therefore, the ARS for the tunnels were obtained as the envelope of the ARS at the tunnel elevation from the soil column analyses considering mean, upper bound and lower bound shear moduli. For Set C, the ARS from the four time history analyses were averaged prior to enveloping. The horizontal ARS and the vertical ARS were determined from analysis of the appropriate soil column. The seismic analyses methodology used for the pipe tunnels is described in Section 3.7.2.1.3.

3.7.2.1.5 Category I Pile-Supported Structures - Original Analysis (Set A)

For structures founded on piles, the acceleration at top of rock was considered to be amplified through the soil as discussed in Section 3.7.2.4. The translational and rocking foundation springs included in the lumped mass model of the structure to characterize soil-structure interaction were calculated using Reference [3]. The damping ratio used for soil-supported structures depended upon the predominant type of motion as explained in Reference [5].

A more detailed description of the seismic analysis of Category I pile-supported structures is discussed below.

Additional Diesel Generator Building

Refer to Section 3.7.2.1.6.

Condensate Demineralizer Waste Evaporator Building (CDWE)

The CDWE Building is a pile supported, reinforced concrete structure. The building consists of two stories and is approximately 54 feet-9 inches by 41 feet-9 inches in plan and 59 feet high. The pile group supporting the CDWE Building consists of 104 vertical and 46 batter piles driven through 30 feet of soil to refusal in sound rock.

The seismic analysis of the CDWE Building was comprised of both a normal mode analysis using lumped mass models and a plane strain analysis using 2-dimensional models. The normal mode analysis was conducted for the north-south, east-west, and vertical directions. The plane strain analysis was conducted for the east-west and vertical directions assuming a unit depth in the north-south direction.

In the normal mode analysis, a model of the soil deposit was used to determine the acceleration time history at the top of ground from the specified bedrock acceleration records. The top of ground acceleration records were then used as input to a lumped mass model of the CDWE Building through a set of translational and rotational springs representing the pile group. The lumped mass models for the normal mode analysis are shown in Figure 3.7-15A.

The earthquake motion used in the analysis was determined by amplifying the four artificial earthquake input at top of rock through the supporting soil. The maximum top of rock horizontal accelerations for these earthquakes are 0.09g and 0.18g for the OBE and the SSE, respectively. The vertical motions are two-thirds of the horizontal.

The amplification of these earthquakes through the soil is performed by considering the soil as an elastic medium and making a dynamic analysis of a slice of unit thickness considering only the horizontal resistance of the soil. The soil deposit was divided into layers which would permit transmission of vibrational frequencies up to 30 Hz. An average value of the shear modulus was determined for each layer based on the effective vertical stress in the layer and then an

average for the entire deposit was calculated. To account for uncertainties in the soil properties, three soil profiles were considered in the normal mode analysis. The three profiles correspond to soil deposits having the calculated average value of shear modulus and variations of ± 50% in the shear modulus. Only the average profile was considered in the plan strain analysis. The values of shear modulus and corresponding shear wave velocities for the three soil profiles are shown in Table 3.7-23A. A damping ratio of 10% is used for the soil. From this analysis, four corresponding top of rock earthquake motions are obtained for use as input to the structural model. The vertical motion at top of ground is assumed to be two-thirds of the horizontal motion.

The lumped mass model of the building for the normal mode analysis consists of four mass points and four elements, the mass and inertia of the base, and translational and rotational springs representing the pile group. The mass points, elements, and spring properties are given in Table 3.7-23A.

The pile group is composed of 104 vertical and 46 batter piles. The pile group was modeled by equivalent translation and rocking springs in both horizontal directions and a vertical spring.

Once a set of spring constants were determined, the lateral and rocking springs were both modified by the same factor to produce a natural period for the structure of 0.15 second in each horizontal direction to correspond to the peak in the top of ground acceleration response spectrum. The spring constants representing the pile group are shown in Table 3.7-23A.

A normal mode time history analysis of the lumped mass model was conducted. A damping factor of 5% of critical was used in this step of the analysis for both soil springs and structural elements. The loads thus compared were considered to be overly conservative, and since the top of ground horizontal accelerations were approximately doubled by the base springs, the horizontal loads in the building were reduced by one-half. A plane strain analysis of the soil-structure system was then conducted for the SSE in the E-W and vertical directions to verify the reduction in the horizontal loads computed by the normal mode analysis. The input accelerations for the latter analysis were the top of rock acceleration records specified for the Watts Bar Nuclear Plant.

The plane strain analysis was conducted using a 2-dimensional model of the soil-structure system in order to verify reducing the results obtained in the normal mode analysis. The model included soil-structure interaction effects, and cases were run with and without the pile group stiffness included in the soil properties. Damping factors of 10% of critical for the soil elements and 5% of critical for the base mat and CDWE Building elements were used in the plane strain analysis. The soil properties are linear and elastic.

The time history accelerations specified for top of rock were applied at the base of the model, and the free field top of ground acceleration was compared to the lumped mass model top of ground motion. The plane strain analysis indicated the horizontal acceleration amplification through the soil and base springs in the lumped mass analysis was excessive and a reduction of the horizontal loads in the building by a factor of one-half was justified.

3.7.2.1.6 <u>Category I Pile-Supported Structures - Evaluation and New Design/Modification Analyses (Set B and Set B+C)</u>

<u>Additional Diesel Generator Building (ADGB)</u>

The original criteria for the ADGB design basis seismic analysis was based on NUREG-0800 and Regulatory Guide 1.60 ground design spectra. These criteria were incorporated into the UFSAR after the issuance of NUREG-0847, WBNP Safety Evaluation Report, Supplement 2,

1984. In order to bring the ADGB in line with the other Category I structure, the structure has been reanalyzed in accordance with Set B and Set C criteria. The seismic responses (ARS, accelerations, displacements, forces and moments) defined by Set B and the envelope of Set B and Set C (Set B + C) are used in evaluating the adequacy of existing structures, as well as new designs and modifications.

The 3-D lumped parameter model used for the ADGB is shown in Figures 3.7-15B and 3.7-15C, and the associated model properties are given in Tables 3.7-23B and 3.7-23C.

Condensate Demineralizer Water Evaporator Building (CDWE)

The CDWE Building does not house any safety-related systems and components. Therefore, Set B and Set C analyses were not performed.

3.7.2.2 Natural Frequencies and Response Loads for NSSS

The natural frequencies of Westinghouse supplied components are considered in the system seismic analysis. The natural frequencies are listed in detail in the component stress reports.

3.7.2.3 Procedures Used for Modeling

3.7.2.3.1 Other than NSSS

The procedures used to formulate original analysis mathematical models of each Category I structure have been discussed in Sections 3.7.2.1.1 and 3.7.2.1.3. The mass of supported equipment was considered in the lumped masses at the points of support. The stiffness of supported equipment was not considered in the lumped mass model of the structure.

For evaluation and new design or modification analyses, the stiffness and mass of a subsystem (supported equipment, a system, or a component) are included in the model if either Criteria 1 or 2 given below apply:

1) $0.01 \leq R_m \leq 0.10$ and $0.8 \leq R_f \leq 1.25$

2) $R_m \geq 0.1$

where,

$$R_m = \frac{\text{total mass of subsystem}}{\text{total mass of structure}}$$

$$R_f = \frac{\text{fundamental frequency of subsystem}}{\text{dominant frequency of structure}}$$

When the criteria given above for the inclusion of both stiffness and mass are not met the mass of a subsystem is included in the model if the subsystem is comparatively rigid in relation to the supporting structure and rigidly connected to the supporting structure.

3.7.2.3.2 For NSSS Analysis

The first step in any dynamic analysis for a system or component supplied by Westinghouse is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dash pots suitable for mathematical analysis. Essentially, the

procedure is to select mass points so that the displacements obtained will be a good representation of the motion of the system or component. Stated differently, the true inertia forces are not altered so as to appreciably affect the internal stresses in the structure or component. The mathematical model used for the dynamic analysis of the reactor coolant system is shown in Figure 5.2-1. Figure 5.2-2 shows the mathematical model of the reactor pressure vessel.

The determination as to whether the structure or component is analyzed as part of a system analysis or independently as a subsystem, is justified on a case by case basis.

3.7.2.4 Soil/Structure Interaction

3.7.2.4.1 Original Analysis (Set A)

For Category I structures founded upon soils, the rock motion was amplified to obtain the ground surface motion by considering the soil deposit as an elastic medium and making a dynamic analysis of a slice of unit thickness using only the horizontal shearing resistance of the soil. The four artificial earthquakes mentioned in Section 3.7.1.2 were considered as the input motion at top of rock. Once the time history of surface accelerations was known, a response spectrum was produced for the analysis of the soil-supported structure. The vertical surface motion was considered as two-thirds of the horizontal surface motion.

The soil amplification analysis is affected by the variations of onsite soil measurements, slanted soil layers, soil density, and depth of the soil deposit. Therefore, for structures supported on a soil deposit, the parameters of the soil deposit beneath the structure were varied to obtain a series of ground motion spectra. An envelope was drawn from these spectra resulting in the final ground motion spectrum used in analyzing the structure.

By following the procedure outlined, the maximum amplification of the ground response was obtained and the peak width of the ground response spectrum was wide enough to allow for variations in the frequencies of the structure due to variations in soil parameters.

3.7.2.4.2 Evaluation and New Design or Modification Analyses

For Category I structures founded upon soil, the top-of-rock motions were considered to be amplified (or attenuated) through the soil. The value of amplification and the change in frequency content of the excitation were determined by a soil column dynamic analysis that incorporates strain-dependent soil properties. Therefore, the soil properties beneath the structure were varied by the amounts given in Tables 2.5-17A through 2.5-17D to obtain different soil surface motion time histories.

For Set B analyses, the top-of-rock input motions are those defined by the Evaluation Site Design Response Spectra and Evaluation Site Design Time Histories of Section 3.7.1. For Set C analyses, the input motions are defined by the Original Site Design Response Spectra and the Original Site Design Time Histories (Section 3.7.1).

3.7.2.5 Development of Floor Response Spectra

3.7.2.5.1 Original Analysis

Response spectra for use in computing the response of structural appurtenances, or of equipment attached to Category I structures were produced by the time-history modal analysis technique. The four artificially produced accelerograms (Section 3.7.1.2) were the input motion

at top of rock. To obtain a set of response spectra for one mass point for one direction of motion, the procedure outlined in Figure 3.7-37 was used.

Spectral values were computed for the periods using the distributions shown in Table 3.7-1, in addition to the natural frequencies of the structure. In all time-history calculations a time interval of 0.010 second was used.

Response spectra were computed for percentages of critical equipment damping of 0.5, 1.0, 2.0, 3.0, 4.0, 5.0 and 7.0. Response spectra were calculated for both the OBE and SSE; except, for those instances when the same percentage of critical structural damping was specified for both earthquakes, response was calculated for the OBE or SSE only (the SSE results equal twice the OBE).

Horizontal response spectra were produced at ground level, at major floors, and at other points of interest within the structure for both east-west and north-south directions, except where symmetry justifies the use of one direction.

For a direction in which torsion is considered, the time histories of accelerations used to produce the spectra will be computed where the maximum accelerations occur at that level (the farthest points on the structure from the shear center, on the axis perpendicular to motion).

Unless otherwise noted, vertical response spectra were produced at ground and at major floor elevations. The response spectra for ground was used throughout that portion of the structure where no structural amplification occurred. For other points, values were interpolated linearly between adjacent floors.

3.7.2.5.2 Evaluation and New Design or Modification Analysis

Response spectra for Set B and Set C analyses are produced by the time history modal analysis technique. For evaluation (Set B) analyses, the co-directional time history responses are either computed directly by simultaneous application of the directional seismic inputs or by the SRSS method. Set C co-directional responses are combined by the SRSS method only.

OBE (Set B) and OBE (Set C) constant damping response spectra are computed for damping ratios of 1, 2, 4, 5 and 7%. Site-specific SSE and SSE spectra are computed for 2, 3, 5 and 7%. Site-specific SSE, OBE (Set B), OBE and SSE variable damping response spectra are also computed for both Set B and Set C in accordance with ASME Code Case N411.

The ARS values were generated for the standard 75 spectral frequencies specified in Table 3.7.1-1 of the SRP plus the significant structure natural frequencies that are below the frequency limit of 33 Hz. For the ACB, UFSAR Table 3.7-1 spectral frequencies were used for Set C analyses. A study comparing the spectra obtained from the use of SRP and UFSAR frequencies concluded that the use of UFSAR frequencies for ACB Set C analyses is adequate.

Two solution methods were used to generate floor response spectra. These were time domain method of analysis and frequency domain method of analysis. For the time domain method of analysis, a time interval of 0.005 second was used for structural analysis, and time intervals of 0.005 and 0.0025 seconds were used for generation of floor response spectra. For the frequency domain method of analysis, a time interval of 0.01 second was used for structural analysis, and a time interval ranging from 0.01 to 0.0025 seconds was used for generation of floor response spectra.

The final Set B and Set C ARS include ± 15% and ± 10% peak broadening, respectively, for structures other than the ERCW tunnels. For Set C analyses, because of identical OBE and SSE structural damping, OBE ARS accelerations are one-half the corresponding SSE values. New design/modification ARS are defined by the envelope of Set B and Set C ARS.

The ERCW pipe tunnels are embedded in soil and their response follows the motion of the surrounding medium. The ARS at tunnel elevations were obtained from an envelope of the ARS generated from soil column analyses using the mean, upper, and lower bound soil shear moduli.

Vertical response spectra are calculated at the building extremities for the basemat and for all major floor elevations.

3.7.2.6 Three Components of Earthquake Motion

3.7.2.6.1 Original Analysis (Set A)

The seismic responses of Category I structures were computed by assuming the vertical earthquake to occur simultaneously with each of the two major horizontal directions separately. The derivation of the site response spectra and the design time histories for horizontal and vertical motion has been detailed in Sections 3.7.1.1 and 3.7.1.2, respectively.

3.7.2.6.2 Evaluation and New Design/Modification Analyses (Set B and Set C)

The seismic responses of the Category I structures are determined assuming that the three components of the earthquake occur simultaneously.

When the response spectrum method is used for seismic analysis of structures, the maximum structural response due to each of the three components of earthquake motion is combined by the SRSS of the maximum co-directional responses caused by each of the three components of earthquake motion at a particular point of the structure.

When the time history analysis method is used for Set B analysis of structures, the co-directional responses from each of the three components of earthquake motions are either combined algebraically at each time step or the maximum responses from each earthquake are combined by the SRSS method. For Set C time history analyses, only the SRSS method is used to combine co-directional responses.

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 Other Than NSSS

3.7.2.7.1.1 Original Analysis (Set A)

The responses of all Category I structures were computed by the response spectrum modal analysis method. The responses were calculated in each component mode. The total response was then calculated by determining the square root of the sum of the squares (SRSS) of the modal responses. For example, the total acceleration in any direction was calculated as:

$$a_T = \sqrt{a_1^2 + a_2^2 + ... + a_n^2}$$

Similar expressions exist for the other responses.

When the frequencies of two or more modes are found to be closely spaced (modes whose frequencies are within 10% of each other), the responses of these modes were combined in an absolute sum manner. The resulting total was treated as that of a pseudo-mode and combined with the remaining modes by the SRSS method.

The stresses in the structures were calculated assuming the vertical earthquake to occur simultaneously with either horizontal earthquake. For example, a typical expression for the stress x, caused by a horizontal earthquake in the x-direction and a vertical earthquake in the y-direction, would be:

$$\sigma_x = \pm\sigma_{xx} + \sigma_{xy}$$

3.7.2.7.1.2 Evaluation and New Design or Modification Analyses

The response spectrum method was used to determine the seismic responses for the Category I structures. The most probable response is obtained as the square root of the sum of the squares from the individual modes.

For Set B and Set C analyses, either the response spectrum or time history analysis methods were used to determine the seismic responses of Category I structures. When the response spectrum method was used, modal responses were combined in accordance with NRC Regulatory Guide 1.92, Rev. 1. Modal responses computed by the time history method were combined algebraically at each time step. For either analysis method, a sufficient number of modes were investigated to assure participation of all significant modes.

3.7.2.7.2 NSSS System

The total seismic response of systems and major components within Westinghouse scope of responsibility is obtained by combining the individual modal responses utilizing the SRSS method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10% of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root 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 $\varepsilon_{K\ell}$. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{K=M_j}^{N_j-1} \sum_{\ell=K+1}^{N_j} R_K R_\ell \varepsilon_{K\ell} \qquad (1)$$

Where,

RT	=	total response
Ri	=	absolute value of response of mode i
N	=	total number of modes considered
S	=	number of groups of closely spaced modes
Mj	=	lowest modal number associated with group j of closely spaced modes
Nj	=	highest modal number associated with group j of closely spaced modes
$\varepsilon_{\kappa l}$	=	coupling factor with

$$\varepsilon_{K\ell} = \left(1 + \left[\frac{\omega_{K'} - \omega_{\ell'}}{(\beta_{K'}\omega_K + \beta_{\ell'}\omega_\ell)}\right]^2\right)^{-1} \tag{2}$$

$$\beta'_j = \beta_j + \frac{2}{\omega_j t_d} \tag{3}$$

$$\omega'_j = \omega_j[1 - (\beta'_j)^2]^{1/2} \tag{4}$$

Where,

ϖj	=	frequency of closely spaced mode j (rad/sec)
βj	=	fraction of critical damping in closely spaced mode j
t_d	=	duration of the earthquake (sec.)

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Frequency	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20

There are two groups of closely spaced modes, namely with modes 2, 3, 4 and 6, 7. Therefore,

S	=	2 number of groups of closely spaced modes
M_1	=	2 lowest modal number associated with group 1
N_1	=	4 highest modal number associated with group 1
M_2	=	6 highest modal number associated with group 2
N_2	=	7 highest modal number associated with group 2
N	=	8 total number of modes considered

The total response for this system is, as derived from the expansion of Equation (1):

$$R_T^2 = [R_1^2 + R_2^2 + R_3^2 + ... + R_8^2] + 2R_2 \; R_3 \; \in_{23} + 2 \; R_2 \; R_4 \; \in_{24} + 2 \; R_3 \; R_4 \; \in_{24} + 2 \; R_6 \; R_7 \; \in_{67} \tag{5}$$

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

All interfaces between Category I and non-Category I structures were designed to withstand the displacement and/or dynamic loads produced by both the Category I and non-Category I structures and equipment. The Turbine Building, Service Building and Old Steam Generator Storage Facility (OSGSF) are the only non-Category I structures for which this section applies.

The Turbine and Service Buildings were analyzed for a total lateral base shear computed as the product of the mass of the structure and the ground acceleration for the SSE. The total lateral shear was distributed in the height of the structure according to the provisions of the Uniform Building Code.

The OSGSF seismic design is in accordance with the 2000 edition of the International Building Code.[31]

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

To account for variations in structural frequencies owing to variations in material properties of the structure and soil and to approximations in modeling techniques used in seismic analyses, the computed floor response spectra are smoothed and peaks associated with the structural frequencies are broadened $\pm$ 10% for Set A and Set C and $\pm$ 15% for Set B.

For the soil-supported structures in which floor response spectra were produced, the soil properties were varied to account for variations in soil properties. Soil-structure interaction was considered as discussed in Sections 3.7.2.1 and 3.7.2.4.

3.7.2.10 Use of Constant Vertical Load Factors

3.7.2.10.1 Other Than NSSS

3.7.2.10.1.1 Original Analysis (Set A)

A vertical lumped mass dynamic analysis using the techniques outlined in Section 3.7.2.1.1 was performed for all of the Category I structures to determine the vertical loads. The results for each horizontal earthquake analysis were separately added on an absolute basis to the loads from the vertical earthquake analysis. Static vertical load factors were not used unless the dynamic analysis indicated the structure behaved as a rigid body in the vertical direction.

3.7.2.10.1.2 Evaluation and New Design or Modification Analyses

The Category I structures, when analyzed for vertical motion, used lumped-mass dynamic techniques as discussed in Section 3.7.2.1.2. For Evaluation Analyses (Set B), the co-directional time history responses are either computed by simultaneous application of the seismic input in three directions or by the SRSS method. For Set C, co-directional responses are combined by SRSS only. For systems and components the appropriate floor response spectra was used in the analysis. Static load factors were not used for either Set B or Set C analysis.

3.7.2.10.2 For NSSS

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

3.7.2.11 Methods Used to Account for Torsional Effects

The dynamic analysis of structures is discussed in Section 3.7.2.1. In original or Set A analyses, torsional effects were considered by using a lumped-mass cantilever beam model to represent stiffness and inertial characteristics. The torsional moment of inertia, eccentricity, and mass moment of inertia were included in the analyses.

In the process of preparing lumped-mass mathematical models for the Set A analyses, the location of both the center of rotation and center of mass for each floor were computed. Accelerations and deflections were calculated where their maximum values occurred (at the farthest points on the structure from the shear center, on the axis perpendicular to the direction of motion).

For Set B and Set C analyses, modeling of torsional effects was refined by three-dimensional modeling.

The models described above were subjected to seismic excitations and the resultant responses in the form of frequencies, mode shapes, moments, and forces were obtained.

3.7.2.12 Comparison of Responses - Set A versus Set B

The comparison of Set A and Set B responses showed that, in general, Set A responses were higher. In making the ARS comparisons, the applicable damping ratios of Set A and Set B were used. In certain frequency ranges, Set B responses were higher than Set A responses. An evaluation was performed on a building by building basis to assess the impact of Set B response. Adequacy of structures, systems, and components for Set B effects has been documented in calculations.

As a sample comparison of Set A and Set B responses, the ARS comparisons for Auxiliary Control Building, which is a rock-supported structure, and for the Diesel Generator Building, which is a soil-supported structure, are presented. The ARS for north-south, east-west and vertical directions are compared. The comparison at Elevation 692.0 and Elevation 814.25 of the Auxiliary Control Building are presented in Figures 3.7-15D through 3.7-15I.

3.7.2.13 Methods for Seismic Analysis of Dams

Since no dams are utilized to impound bodies of water to serve as heat sinks, this section is not applicable to this site.

3.7.2.14 Determination of Category I Structure Overturning Moments

3.7.2.14.1 Original Analysis

From the dynamic analyses of the structures, the seismic moments, shears, and vertical loads were determined at the base of the structure. These loads were used in combination with other appropriate loads in determining total overturning effects as discussed in Section 3.8.

3.7.2.14.2 Evaluation and New Design or Modification Analysis

From the dynamic earthquake, analyses total moments, shears, and vertical loads were computed.

The earthquake moment, shear, and vertical load were used in combination with other appropriate loads in determining total overturning effects as discussed in Section 3.8.

3.7.2.15 Analysis Procedure for Damping

The damping values used in the dynamic earthquake analyses of Category I structures are given in Table 3.7-2.
For Set A analysis, the Category I structural models were not coupled together, therefore, the structural damping values used in the seismic analyses are as shown in Tables 3.7-2 and 3.7-24.

For Set B and Set C analyses, either composite modal damping or structural damping were used in the seismic analyses of Category I structures. The damping values used for the various structures and components are given in Tables 3.7-2 and 3.7-24. The damping used in the seismic analysis of systems and components are also given in Tables 3.7-2 and 3.7-24.

The new Unit 1 RCL analysis damping values (from Table 3.7-2) are for Set B+C instead of Set B. Under the Westinghouse standard scope of supply and analysis, the lowest damping value associated with each element of the system is used for all modes.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods for Other Than NSSS

The seismic analysis of Category I piping systems is described in detail in Section 3.7.3.8.

In the analysis of piping subsystems there are two distinct approaches to seismic analysis. A detailed analysis is discussed in Section 3.7.3.8.2 and a simplified analysis is discussed in Section 3.7.3.8.3.

The general seismic analysis of Category I equipment and components is discussed in Section 3.7.3.16. Additional details applicable for simplified analysis are discussed in Sections 3.7.3.5 and 3.7.3.10.

The seismic analyses of HVAC and conduit/cable tray subsystems are discussed in Sections 3.7.3.17 and 3.10.3, respectively.

The detailed seismic analyses of Category I subsystems is based upon dynamic analyses using the lumped mass normal mode method with idealized mathematical models. The inertial properties of the models are characterized by mass, eccentricity, and mass moment of inertia of each mass point. Mass points are located at carefully selected points in order to accurately model the subsystem as described in Section 3.7.3.3.1. The stiffness properties are characterized by the moment of inertia, area, torsion constant, Young's modulus, and shear modulus.

The response of Category I subsystems are computed by the response spectrum modal analysis method for designs. All significant modes of vibration are considered in determining the total response. Subsystem response is calculated in three orthogonal directions.

Seismic responses of the Category I subsystems, equipment, and components are determined and combined in accordance with Sections 3.7.3.6 and 3.7.3.7. The damping ratios used in the dynamic analyses of the structures, subsystems, and equipment/components are shown in Table 3.7-2.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Category I Systems and Components Other Than NSSS

During the design life of the plant (40 years), two earthquakes of OBE magnitude and one SSE are postulated to occur. This was based upon a study of seismic history in the Southern Appalachian Province over a 100-year period. Based on this study, each occurrence is conservatively assumed to have a time duration of 15 seconds of strong excitation.

For Class A Category I components, an evaluation of predominant frequencies revealed that the most significant response of components is conservatively considered using an average frequency of 20 Hz. Therefore, the total number of cycles considered for the OBE and SSE are 600 and 300, respectively.

The seismic qualification testing of Category I equipment considers the number of events and durations described above in accordance with IEEE 344-1975.

ASME Section III Class 1 Piping Analysis - Since the piping in this scope has been reanalyzed in accordance with SRP requirements, the piping analysis has assumed the occurrence of 5 OBEs and 1 SSE. The number of peak stress cycles
may be obtained from the synthetic time history used for the analysis (with a minimum duration of 10 seconds), or a minimum of 10 peak stress cycles per event assumed.

3.7.3.2.2 NSSS System

Where fatigue analysis of mechanical systems and components is required, Westinghouse specifies in the equipment specification that 20 occurrences of OBE having 20 cycles of maximum response for each occurrence, be analyzed. The fatigue analyses are performed as part of the stress report.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Other Than NSSS

3.7.3.3.1.1 Modeling of Piping Systems for Detailed Rigorous Analysis

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped at the nodes connected by weightless elastic members, representing the physical properties of each segment. The pipe lengths between mass points are such that the adequate simulation of the dynamic characteristics of the piping system is ensured. All concentrated weights on the piping system such as main valves, relief valves, pumps, motors, and effects of support mass on piping system when found to be significant are modeled as lumped masses unless isolated from the system by positive anchorage. The torsional effects of the valve operators and the other line-mounted equipment with offset center of gravity with respect to center line of the pipe, is included in the analytical model.

3.7.3.3.1.2 Modeling of Equipment

For seismic analysis, Seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

1. The number of modes of a dynamic system is controlled by the number of masses used. The number of masses is chosen so that all significant modes are included. The modes are considered as potentially significant if the corresponding natural frequencies are less than 33 Hz. For modes greater than 33 Hz the rigid response contribution is considered.

2. Mass is lumped at points where significant concentrated weight and continuous mass are located.

3.7.3.3.1.3 Modeling of HVAC, Conduit, and Cable Tray Subsystems

Runs of HVAC, conduit, and cable tray subsystems (including supports) are modeled by continuous or discrete mass models with the interconnecting elements represented by their effective stiffness properties. Additional lumped masses are applied at or near significant concentrated weights such as from fittings or other in-line or attached commodities. Significant concentrated weights are those which cannot be adequately represented by smearing their effect as part of the overall uniform mass. Mass eccentricities and torsional stiffnesses are considered. Where models are truncated, at least one span and the next support on either side of the contiguous span(s) and support(s) of interest for evaluation are modeled. Alternately, the contiguous span(s) and support(s) of interest are evaluated with one half of the adjacent spans on either side modeled with symmetry boundary conditions such that no artificial stiffening is introduced.

A sufficient number of masses (or degrees of freedom) are modeled such that additional masses would not increase the predicted responses by more than 10%. Alternately, the number of masses are modeled to be at least twice as many as the number of modes with frequencies less than 33 Hz. The dynamic analysis considers all modes with significant mass participation such that inclusion of additional modes would not increase the predicted responses by more than 10%. Alternately, the dynamic analysis considers all modes up to 33 Hz and includes an additional check for any missing mass.

3.7.3.3.2 Modeling of NSSS Subsystems

The criteria and procedures used for modeling of NSSS subsystems is given in Section 3.7.2.3.

3.7.3.4 Basis for Selection of Frequencies

3.7.3.4.1 Other Than NSSS

The method used to analyze systems for dynamic loading is the modal response spectrum method.

Frequencies of the subsystems are selected such that all significant modes of vibration are included in the analysis. Frequencies of simplified analysis models are determined by solutions of closed form expressions. Frequencies of detailed analysis models are determined by computerized solutions.

The subsystem or component model is subjected to loadings in the form of accelerations that represent the seismic environment of its supports. Since the response spectrum employed is representative of the building elevation at the equipment/system location considered, structural amplifications are reflected in the spectra. Therefore, the input acceleration values taken from the building response spectra and utilized as input to the dynamic analysis of the subsystem or component assures the model is loaded in a representative manner and the proper amplifications determined. The subsystem or component was analyzed and designed for the amplified loading.

3.7.3.4.2 NSSS Basis for Selection of Forcing Frequencies

The analysis of equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports based upon the mass and stiffness characteristics of the system, will determine the seismic accelerations which the equipment must withstand.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low-period region of the floor response spectra.

2. If the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant and the equipment behaves as though supported on the ground.

3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

In addition, an equipment/support system is considered to be rigid if the fundamental natural frequency is greater than 33 Hz.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

3.7.3.5.1 Other Than NSSS

For discussion of the equivalent static load method as applied to equipment/components, see Sections 3.7.3.10.1, 3.7.3.16.1, 3.7.3.16.2, and 3.7.3.16.3.

For other Category I subsystems, the following discussion applies:

Simplified seismic analysis by the equivalent static load method may be used as an alternative to detailed computer analysis when the subsystem being analyzed is adequately represented by an effective one degree-of-freedom system with multi-mode effects accommodated by the use of a multi-mode factor. A modal participation factor of 1.0 is used for the equivalent static load method. If the subsystem is determined to be rigid (fundamental frequency $\geq$ 33 Hz), then the acceleration of the building at the elevation of the subsystem attachment (floor zero period acceleration) is used with a multi-mode factor of 1.0; i.e., the subsystem is evaluated for rigid-body response. When no frequency evaluation of the subsystem is made, the peak acceleration of the applicable floor response spectrum is used multiplied by a multi-mode factor of 1.5 except where a lower factor is justified. When a frequency evaluation is made and the subsystem is determined to be flexible, the highest acceleration at or above the determined frequency is used for evaluation multiplied by a multi-mode factor of 1.5 except where a lower factor is justified. For HVAC, conduit, and cable tray subsystems a multi-mode factor of 1.2 has been justified.

3.7.3.5.2 Use of Equivalent Static Load Method of Analysis for NSSS

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient, which is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as a single-degree-of-freedom system are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems, which may be in the resonance region of the amplified response spectra curves, are increased by 50 percent to account conservatively for the increased modal participation.

3.7.3.6 Three Components of Earthquake Motion

Seismic responses of Category I subsystems, equipment, and components are analytically computed or simulated by qualification tests for the applicable Set A, B, and C seismic inputs in three orthogonal directions. The Set A, B, and C inputs for original analysis/qualification, evaluation, and new design/modification are described in Section 3.7.2.

3.7.3.6.1 Piping Subsystems

The seismic responses of Category I piping subsystems are determined assuming that the three components of the earthquake motion occur simultaneously. The maximum response due to each of the three components of earthquake motion is combined by SRSS of the maximum directional responses caused by each of the three components of earthquake motion.

3.7.3.6.2 HVAC Ducting, Conduit, and Cable Tray Subsystems

The seismic responses of HVAC ducting, cable tray, and conduit subsystems are determined by two dimensional seismic analysis and associated testing of representative duct, cable tray, and conduit spans. Seismic input in each major horizontal direction is applied separately but simultaneously with vertical input. Horizontal and vertical responses are analytically combined by absolute summation.

3.7.3.6.3 Other Than NSSS Equipment and Components

The seismic responses of Category I equipment and components were determined by analysis or test in accordance with the guidelines of IEEE 344-1971 for procurements initiated prior to September 1, 1974. After that date procurement, evaluation, and modification activities applied the guidance of IEEE 344-1975 to determine the seismic responses.

Floor or wall mounted equipment and components and their supports and anchorage are seismically analyzed or tested by application of the required seismic response spectra described in Section 3.7.2.5, in a two-dimensional manner. Seismic input in each major horizontal direction is applied separately but simultaneously with vertical input. Horizontal and vertical responses are analytically combined by absolute summation.

Seismic responses of line-mounted equipment and components are determined by device analysis or testing techniques from IEEE 344-1971 or IEEE 344-1975, as applicable. These techniques are applied in a two-dimensional manner relative to the three orthogonal local axes of the line-mounted equipment and component. Calculated seismic response of the subsystem at the equipment and component location is maintained at a level which is less than or equal to the device seismic qualification level.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Other Than NSSS

Modal responses of the piping subsystems are combined in accordance with Regulatory Guide 1.92, Revision 1. Modal responses of other subsystems are analytically combined by the techniques described in Section 3.7.2.7.1 for structures.

Category I equipment and components are seismically analyzed or tested by IEEE Standard 344-1971 or -1975 techniques, as described in Section 3.7.3.6. In accordance with these standards, modal responses are analytically combined by SRSS techniques except for closely-spaced modes whose responses are combined by absolute summation.

3.7.3.7.2 Combination of Modal Responses of NSSS

For the NSSS procedure for the combination of modal responses see Section 3.7.2.7.2.

3.7.3.8 Analytical Procedures for Piping Other Than NSSS

3.7.3.8.1 General

The analysis of classified fluid system components other than the reactor coolant system considers both static and dynamic loadings. The loading combinations considered and the allowable stress limits are discussed in Section 3.9.3.1. Thermal expansion, dead load, and normal operational stresses due to system pressurization for Category I piping systems are analyzed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1 Nuclear Power Plant Components, 1971 Edition up to and including the Summer 1973 Addenda. Non-nuclear safety classes of pipe are analyzed in conformance with ANSI B31.1, Power Piping Code, 1973 Edition up to and including Summer 1973 Addenda as shown in Table 3.2-5. In addition, TVA Class M (chilled water) piping conforms to ANSI B31.5, 1974. Stresses due to all loadings are appropriately combined with the seismic stresses in accordance with Code requirements.

As permitted by NA-1140 of applicable ASME Code, the following sections of more recent editions and addenda of the ASME Boiler and Pressure Vessel Code and ASME Code Cases are used. All related requirements were met.

A. CODE EDITIONS AND ADDENDA

1. Stress Intensification Factors

   a. 1974 Code; used for Stress Intensification Factors for Class 2 and 3 piping.

2. Nozzle Dimensions

   a. Figure NB-3686.1-1 for nozzle dimensions from the Summer 1975 Addenda.

3 Material Properties

   a. 1980 Edition - including Summer 1980 Addenda, Appendix I, Table I-4.0; for thermal conductivity and thermal diffusivity of materials.

   b. 1983 Edition - including Winter 1983 Addenda, Appendix I, Table I-5.0; for coefficient of Thermal Expansion of materials which are not available in the Code of Record.

   c. 1983 Edition - including Summer 1985 Addenda, Appendix I, Table I-6.0; for Modulus of Elasticity of materials which are not available in the Code of Record.

d. 1983 Edition - including Summer 1985 Addenda, Appendix I, Tables I-1.1, I-1.2, I-1.3, I-2.1, I-2.2, I-3.1, I-3.2, I-7.1, I-7.2, I-7.3, and I-9.1 for materials which are not available in the Code of Record.

4. <u>Stress Qualification</u>

a. 1980 Edition - up to and including Winter 1982 Addenda, Section III, Subsection NB; May be used for the stress qualification of Class 1 piping (NB-3600).

b. 1974 Edition - Summer 1976 Addenda, Section III, Paragraph NB-3630 (d); used for Class 1 piping which can be analyzed per requirements of Subsection NC.

c. 1974 Edition - Winter 1976 Addenda, Section III, Paragraph NC/ND-3611.2.

d. 1977 Edition - Section III, Paragraph NC/ND-3652.3.

e. 1974 Edition - Summer 1975 Addenda, Section III, paragraph NC/ND-3651.

f. 1974 Edition - Section III, Paragraph NC/ND-3652.4.

5. <u>Welded Attachments</u>

a. 1980 Edition - Winter 1980 Addenda, Section III, Paragraph NB-4433 which permitted the use of continuous fillet or partial penetration welds for welded structural attachments (Lugs) to the pipe.

6. <u>Flange Qualification</u>

a. 1983 Edition - up to and including Winter 1983 Addenda, Section III; Used for Class 1 Flange qualification per NB-3658; Used for Class 2 and 3 Flange qualification per NC-3658 and ND-3658.

7 <u>Relief and Safety Valve Thrust</u>

a. 1977 Edition - Winter 1978 Addenda, Section III, Paragraph NC/ND-3622.5 and Appendix O.

B. CODE CASES

1. Half-Coupling Branch Connections

   a. Code Case N-313, November 28, 1986, Alternate Rules for Half Coupling Branch Connections, Section III, Division 1, Class 2.

2. Response Spectra

   a. Code Case N-411-1, February 20, 1986, Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping Systems, Section III, Division 1, may be used.

3. Stress Qualification

   a. Code Case 1606-1, December 16, 1974, Stress Criteria, Section III, Classes 2 and 3 Piping Subject to Upset, Emergency, and Faulted Operating Conditions.

   b. Code Case N-319, July 13, 1984, Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping. (Applicable to Unit 1 Only)

   c. Code Case N-319, July 13, 1981, Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping. (Applicable to Unit 2 Only)

4. Welded Attachments

   a. Code Case N-122, January 21, 1988, Stress Indices for Integral Structural Attachments, Section III, Division 1, Class 1. (Applicable to Unit 1 Only)

   b. Code Case N-318-3, September 5, 1985, Procedure for Evaluation of the Design of Rectangular Cross Section Welded Attachments on Class 2 or 3 Piping, Section III, Division 1.

   c. Code Case N-391, November 28, 1983, Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1.

   d. Code Case N-392, November 28, 1983, Procedure for Evaluation of the Design of the Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping, Section III, Division 1.

   e. Code Case N-122, January 21, 1982, Stress Indices for Integral Structural Attachments, Section III, Division 1, Class 1. (Applicable to Unit 2 Only)

Category I piping is classified into two analytical categories. These categories are defined below.

Rigorous Analysis (Detailed Seismic Analysis)--A comprehensive computer-aided analysis of the piping system to ensure that the system design meets all the ASME Section III requirements for stress in the piping.

Alternative (Simplified) Analysis--A conservative method for locating supports and determining support loads, using computer generated data, hand calculations and/or computer aided analysis to ensure that the ASME Section III code requirements are met.

Systems Rigorously Analyzed

TVA evaluates the necessity of performing a Rigorous Analysis on all piping systems and identifies the limits of the analysis using the following guidelines:

1. Class A piping systems not analyzed by the NSSS vendor.

2. TVA Class B, C and D lines 2-1/2inches in diameter and larger.

3. Piping in Category I structures larger than 1-inch diameter that has a maximum operating temperature of 200°F or greater and a maximum operating pressure of 275 psig or greater unless it is determined that there is not a potential for unacceptable pipe rupture interactions.

4. Piping which, due to high temperature or other extraordinary loading conditions, cannot be supported using alternate analysis methods.

Systems Analyzed by Alternate (Simplified) Methods

Piping requiring seismic qualification, but not requiring rigorous analysis as outlined above, may be evaluated according to the alternate methods.

3.7.3.8.2 Detailed Seismic Analysis (Rigorous) for Piping Systems

A detailed seismic analysis is performed on applicable piping systems by the response spectrum method. Each pipe run is idealized as a mathematical model consisting of lumped masses connected by weightless elastic members. Lumped masses are located at carefully selected points in order to adequately represent the dynamic and elastic characteristics of the pipe system. Using the elastic properties of the pipe, the flexibility matrix for the pipe is determined. The flexibility calculations include the effects of the torsional, bending, shear, and axial deformations. The stiffness of curved members, valves, branch connections, etc., is also taken into consideration.

Once the flexibility and mass matrices of the mathematical model are determined, the frequencies and mode shapes for all significant modes of vibration are determined. All modes having a period greater than 0.0303 seconds (natural frequencies < 33 Hz) are used in the analysis. The mode shapes and frequencies are solved in accordance with the following equation:

$$(K - w_n^2 M)\ \phi_n = 0$$

where: K = Square stiffness matrix of the pipe loop

M = Mass matrix for the pipe loop

w_n = Frequency for the nth mode

$ø_n$ = Mode shape matrix of the nth mode

After the frequency is determined for each mode, the participation factors can be calculated by the following equation:

$$\Gamma_{njk} = \frac{\phi_n^T M \gamma_{jk}}{\phi_n^T M \phi_n}$$

Where:

Γ_{njk} = Participation factor for mode n in the jth direction of support zone k.

γ_{jk} = Displacement matrix of all nodes due to a unit displacement of the jth direction restrained degrees of freedom in support zone k.

Support zone = A set of restrained nodes which move together during a dynamic event.

Using these results and the corresponding spectral accelerations of the mode for the direction and support zone being excited, the response for each mode is determined by the following equation:

$$(V_{in})_{jk} = \frac{\Gamma_{njk} \phi_{in} S_{anjk}}{W_n^2}$$

Where:

$(V_{in})_{jk}$ = Displacement of mass for mode n for an earthquake in the jth direction of support zone k.

$ø_{in}$ = Value associated with mass i in $ø_n$

S_{anjk} =Spectral acceleration for mode n for an earthquake in the jth direction of support zone k.

Using these results, the maximum displacements for each mode are calculated for each mass point in accordance with the following equation:

$$(V_{in})_j = \sum_{k-1}^{NZ} \left| (V_{in})_{jk} \right|$$

where:

$(V_{in})_j$ = Displacement of mass i for mode n for an earthquake in the jth direction

NZ = Number of support zones used for the pipe loop. However, if ASME Code Case N-411 damping values are used then all supports are in a single support zone.

The maximum displacements for each mode are calculated as follows:

$$V_{in} = \sqrt{\Sigma(V_{in}^2)_j}$$

where:

$(V_{in}$ = maximum displacement of mass i for mode n.

j = x, y, and z

The maximum displacement for each mass is determined by combining the maximum deflection for each mode by the method described in Section 3.7.3.7. The contribution from higher frequency modes (period less than 0.0303 seconds) are combined with lower frequency modes by the SRSS rule.

With the displacements known, the associated member forces/moments can be obtained by standard structural techniques. The forces for each mode and each earthquake direction will be combined using the conventions described above.

3.7.3.8.3 Alternate (Simplified) Analysis for Piping Systems

Section 3.7.3.8.1 defines alternate analysis and specifies the piping for which it may be applied. Various methods are used to perform alternate analysis. These methods may involve the use of simple beam equations, computer generated data and/or computer assisted analysis. For each method, the following general requirements are observed.

1. Deadweight

   Supports are located such that adequate rigidity is assured and pipe sagging is minimized.

2. Seismic

   Seismic effects are approximated using accelerations from the applicable building response spectra. Response spectra accelerations at the frequency computed for the piping system are used except that if the computed frequency is below the frequency corresponding to the peak of the response spectra, the peak accelerations are used. The response spectra accelerations are increased by at least 50 percent to account for multimode response, unless justification is provided for using a lesser increase.

3. Thermal Expansion and Anchor Movement

Thermal expansion and anchor movement are evaluated using conventional hand calculation methods, the results of computer analysis of typical configurations and/or computer aided thermal flexibility analysis.

4. Pipe Stress

Pipe stress resulting from applicable load sources are evaluated and combined in accordance with applicable code requirements. Details of load combinations and stress limits are provided in Section 3.9.3.1.2.

5. Support Loads

Support loads resulting from applicable load sources are evaluated and combined as specified in Section 3.9.3.4.2.

3.7.3.8.4 Seismic Analysis of Piping Systems That Span Two or More Seismic Support Zones Such as Buildings, Portions of Buildings, or Primary Components

Each building, portion of building, or primary component may be considered a separate support zone. The worst enveloped response spectrum for which any portion of the pipe located in that zone is subjected is used to represent the input motion in that zone.

For the evaluation of relative support motions in the seismic analysis of piping systems interconnecting two or more seismic support zones, the maximum relative movement between component supports is assumed and the piping system is subjected to movements through the piping system supports and restraints. Separate cases for each of the three orthogonal directions are considered. Support movements are based on the maximum of the floor movements immediately above and below the support location.

3.7.3.9 Multiple Supported Equipment and Components with Distinct Inputs

3.7.3.9.1 Other Than NSSS

The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs are similar to those described for piping in Section 3.7.3.8.4. When the equipment is supported at two or more points located at different elevations in the building, the response spectrum for the most severe single point of attachment is chosen as the design spectra.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the additional stresses due to support displacements.

3.7.3.9.2 Multiple Supported NSSS Equipment and Components with Distinct Inputs

When response spectrum methods are used to evaluate reactor coolant system primary components interconnected between floors, the procedures of the following paragraphs are used. There are no components in Westinghouse scope of analysis which are interconnected between buildings. The primary components of the reactor coolant system are supported at no more than two floor elevations.

A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis is the worst floor response spectra. Any deviation from this position will be subject for NRC review on a case-by-case basis. The reactor coolant loop analysis described below only includes response spectra at the basemat.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis.

Per ASME Code rules, this stress caused by differential seismic motion is clearly secondary for piping (NB 3650) and component supports (NF 3231). For components, the differential motion will be evaluated as a free end displacement, since, per NB 3213.19, examples of a free end displacement are motions 'that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping'. The effect of the differential motion is to impose a rotation on the component from the building. This motion, then, being a free end displacement and being similar to thermal expansion loads, will cause stresses which will be evaluated with ASME Code methods including the rules of NB 3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses. Since the reactor coolant loops and primary supports are coupled with the reactor building interior concrete structure in the reactor coolant loop seismic analysis, there is no separate consideration of seismic anchor motions.

3.7.3.10 Use of Constant Vertical Load Factors

3.7.3.10.1 Use of Constant Load Factors for Equipment Other Than NSSS

With respect to equipment, static analysis for seismic loading is recognized as an acceptable approach with restrictions as follows:

1. The analysis method is consistent with the 'static coefficient method' as prescribed in IEEE 344-1975, Paragraph 5.3. The peak acceleration values of the applicable floor response spectra are multiplied by a factor of 1.5 if natural frequencies are not determined. The increased acceleration values are used as equivalent static load factors applied to the entire mass of the equipment being evaluated. Lower multiplication factors (between 1.0 and 1.5) are only used as justified by frequency analysis.

2. The static coefficient analysis method is used only for the evaluation of structural integrity of equipment. It is recognized that the static analysis method alone is not sufficient for the qualification of safety-related active equipment where the demonstration of operability is required.

3.7.3.10.2 Use of Constant Vertical Load Factors for NSSS

Constant vertical load factors are not used as the vertical floor response load for the seismic design of NSSS safety-related systems and components.

3.7.3.11 Torsional Effects of Eccentric Masses

3.7.3.11.1 Piping Other Than NSSS

The torsional effects of eccentric masses such as valve operators are modeled in the piping mathematical model as lumped masses at the free end of cantilevered rods with a length equal to the distance from the center of gravity of the mass to the pipe flow axis. The stiffness of the rod is used to simulate the valve extended structure flexibility.

3.7.3.11.2 Torsional Effects of Eccentric Masses of NSSS

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

3.7.3.12 Buried Seismic Category I Piping Systems

Buried piping complies with the ASME Boiler and Pressure Vessel Code, Section III and is analyzed seismically as follows:

The soil is considered to be a horizontal 1-layer system which responds to the earthquake by moving in a continuous sinusoidal plane wave and supported by a second layer or base material. The top layer is assumed to pick up accelerations from the base material.

Utilizing the average values for the shear wave velocity and density for the top layers, the ground deformation pattern in terms of wave length and amplitude is determined. The buried pipes are assumed to deform along with the surrounding soil layers.

The average shear wave velocity of a single layer representation of a multi-layered soil system may be determined by:

$$V_{ST} = \frac{\Sigma V_S h'}{h}$$

where,

V_{ST} = Average shear velocity in the top layers of soil, ft/sec
V_S = Shear velocity in each layer of soil, ft/sec
h' = Depth of each layer of soil, ft

h = Total depth of top layers of soil, ft

The fundamental period of the single layer is calculated from the following equation:

$$T = \frac{4h}{V_{ST}} \quad \text{(seconds)}$$

If the depth of the soil layer varies over the distance traversed by the buried pipe, both cases, for maximum and minimum depths, are considered.

The maximum amplitude of the sine wave which represents the maximum displacement of the pipe is:

$$A = \text{Displacement} = \left(\frac{T}{2\pi}\right)^2 * (Accel)$$

Where:

T = Fundamental period, sec
Accel = Amplified soil acceleration value, in/sec^2

The wave length, L, is calculated as:

$$L = V_{ST} T$$

The bending moment resulting from the seismic disturbance, assuming the pipe follows the soil and deforms as a sine wave, is given by

$$M = \frac{\pi^2 EIA}{(L/2)^2}$$

Where:

M = Maximum bending moment, in-lb
E = Modulus of the pipe, psi
I = Moment of inertia of the pipe, in4
A = Maximum amplitude, in.
L = Wave length, in.

The corresponding bending stress is obtained by dividing the moment by the section modulus of the pipe. The above bending stress is combined with bending stresses due to other loads according to the applicable loading combinations.

The axial strain experienced by the pipe due to deformation of the soils is also evaluated. The axial strain due to seismic propagating waves is computed following the methods of Newmark,[15] and [16] Yeh,[18] and Keusel,[17] which assume the soil is linearly elastic and homogenous, the pipe behaves as a slender beam, and the buried member deforms with the surrounding soil (this implies the strain in the soil equals the strain in the member).

The effect of soil strain from a seismic event on elbows or turns in a buried pipe system must be analyzed in greater detail than just calculating the axial strain. The effect of these strains on elbows/turns is more complex due to the pipe elbow/turn trying to resist the strain. The complexity is a function of the pipe and backfill soil properties.

The basis for determining the effect of the strains on the piping elbows/turns is described by Shah and Chu.[19] The Shah and Chu theory has been developed into an analysis procedure by Goodling.[20], [21], [22] The committee on Seismic Analysis of the ASCE Structural Committee on Nuclear Structures and Materials prepared a report "Seismic Response of Buried Pipes and Structural Components,"[23] which explains and amplifies the referenced methodology[19] and analysis procedure.[20], [21],[22] These references shall be used for analysis of the effects of axial strain on buried piping.

The magnitude of friction acting on the pipe used in the analysis depends on several factors, such as pipe surface conditions, contact pressure, soil strengths, etc. The friction force acting on the pipe is determined in accordance with Reference [24].

Differential Movement

Differential movement between the piping and a structure/feature occurs from two sources. The first is vertical, which can be caused by differential soil consolidation below the pipe or structure/feature. The second source is horizontal movement due to differential movement during a seismic event.

Where practical, seismic classed buried piping is routed to avoid areas of weak soils. Where weak soils are encountered, the bad material is removed and replaced by backfill. The backfill is placed to standards that ensure suitable bearing conditions; therefore, the transition from one material to another, i.e., in situ soil to backfill, should not be a problem. In lieu of the above, in some cases an analysis is performed to show that the pipe has sufficient strength to bridge the discontinuity and support the soil above the pipe without exceeding the allowable stress of the piping material.

Category I buried piping which penetrates structures where fill settlement or seismic movements are expected to be high is protected from differential movement of the soil and structure by Category I concrete slabs or encasements. The slab or encasement is supported by a bracket on the structure on one end and on undisturbed or Class A backfill at the other end. Bearing piles are used if required to support the slab. The encased pipes are insulated to prevent bonding between the pipes and concrete. For details of the slab at the intake pumping station and the encasement at the Diesel Generator Building, refer to Section 3.8.4.4.

For seismic classed buried piping that penetrates structures in areas where very little fill is involved and seismic movements are low, protection from differential movement of the soil and structure is provided by an oversized opening in the structure. The annular space between the pipe and opening is filled with a resilient material. The first support inside the structure is located to allow for relative movement of the pipe and structure. The soil- structure interface is treated as an anchor, and stresses are limited to code allowables.

Soil consolidation is determined in conformance with criteria given in Section 2.5.4.10 (static settlement) and 2.5.4.8 (dynamic settlement - soil liquefaction).

The ERCW piping was evaluated for potential settlement due to soil liquefaction as discussed in Section 2.5.4.8. The potential settlements used for the evaluation were determined in the liquefaction evaluation using the strain criteria specified by the NRC staff which are shown on Figures 2.5-571 through 2.5-575. The effect of these potential settlements was evaluated for the entire length of pipe and also at all building interfaces. The evaluation of the effect of these potential settlements was done in two phases.

The first phase was a preliminary screening which involved calculations to identify areas of the pipe which may undergo excessive settlement. In the preliminary screening, the boundaries of the pipe system, the pipe sizes, and pipe materials were determined. Because of the size and length of pipe involved, a 60 foot length was chosen as sufficient to model the system. A fixed-fixed end model was assumed to describe the piping for the initial calculations. Using the standard equation for maximum deflection for a fixed-fixed end model:

$$Y_{max} = \frac{ML^2}{32EI}$$

M = Resultant moment
L = Span length
E = Young's modulus
I = Moment of inertia

The settlement can be determined if the resulting-moment were known. ASME Code Section III (1977 edition) states that the effects of any single nonrepeated anchor movement is governed by Equation 10A:

$$\frac{iM}{Z} \leq 3.0S_c$$

i = Stress intensification factor
Z = Section modulus
S_c= Allowable stress at room temperature

To expand this equation to include thermal effects (assuming Mc = 0) would involve adding it to Equation 11 (1971 ASME III Code, Summer 1973 Addenda, NC-3652.3) thus;

$$\frac{iM}{Z} \leq 3.0\ S_c + S_A$$

S_A = Allowable stress for expansion

Since the pipe sizes and materials are known, and the stress intensification factor can be calculated, the resultant moment at any point on the pipe can be determined. Thus the potential settlement can be found by using the standard equation for the fixed-fixed end model. The results from these preliminary screening calculations were used in conjunction with the potential settlement evaluation, Section 2.5.4.8, to identify potential areas of excessive settlement, either at the buildings or along the pipeline.

The second phase of the evaluation consisted of making rigorous piping analyses at the potential areas of excessive settlement. There were three areas along the pipeline with apparent problems that were modeled into the TPIPE piping analysis program. These areas were modeled for a distance on both sides of the potential high settlement area. The areas that were modeled were: (1) from the intake pump station to boring SS-131; (2) from boring SS-141 to boring SS-90; and (3) from boring SS-163 to boring SS-159.

At these areas the potential settlements were used as input in the phase II analysis to give the most conservative results. In all cases, the stress levels are below the ASME Code allowable for settlement induced loads (Reference 1977 ASME Code).

Cement-mortar lined carbon steel pipe is used in the buried portion of the ERCW yard piping system. The reason for the mortar lining is given in Section 9.2.1.6. The seismic qualification of the cement-mortar lining is provided by testing. This testing is described below.

A full-scale testing program consisting of laboratory tests, field tests, and vibration measurements was conducted for seismic qualification of the cement-mortar lined carbon steel pipes. A total of 100 feet of 30-inch diameter pipe, 20 feet of 18-inch diameter pipe, and a 90-degree elbow of 30-inch diameter were lined. Pipe sections tested were: one 30-foot pipe of 30-inch diameter, one 40-foot pipe of 30-inch diameter, one 90-degree elbow of 30-inch diameter with a 5-foot pipe welded to each end, 14 two-foot sections of 30-inch diameter, and 10 two-foot sections of 18-inch diameter. Cement-mortar samples were taken from the mixer before lining application began. Density and moisture content tests were performed on the compacted backfill material surrounding the pipe for field tests. Lining materials and procedures were conforming to American Water Works Association Standard C602-76, 'Cement-Mortar Lining of Water Pipelines - 4 Inches and Larger - In Place.'

Cement-mortar specimens were tested for compressive, tensile, and flexural strength, modulus of elasticity, and density. The two-foot pipe sections were subjected to three-edge-bearing, cyclic loading, torsion, drop, and impact tests. The 30-foot pipe was subjected to bending, cyclic loading, and drop tests. The 90-degree elbow was subjected to bending tests. The 40-foot pipe was installed in a trench and after backfilling it was subjected to a dynamic loading of 36,000 pounds at 28 hertz (Hz) from a vibratory roller with a smooth drum of 60-inch diameter by 84-inch width. Two accelerometers were mounted on two of the 30 inch pipes to monitor vibrations experienced by the pipes during the 100-mile trip from the Phipps Bend construction site near Kingsport, Tennessee, to Singleton Materials Engineering Laboratory near Knoxville, Tennessee. The vibrations of the 30-foot pipe (bottom) and a two-foot section (top) were measured and recorded on tape for later analyses. It was expected that the difference in dimension and difference in physical location of the pipes would result in different vibration magnitudes and frequency contents. Comparison between the recorded vibrations and the design earthquake was also made.

The acceleration time histories and their corresponding Fourier amplitude spectra at certain high acceleration locations on the record were processed. The acceleration time histories are

recorded data and the Fourier amplitude spectra are calculated from the recorded data. This transform of data from time domain to frequency domain reveals the frequency content of the vibration data. The maximum acceleration experienced by the bottom pipe (30 feet long) was 0.6g and that experienced by the top pipe (two-foot section) was 2.1g. Both values are higher than the SSE accelerations for the design of TVA nuclear plants. The recorded maximum peak-to-peak accelerations were 1.2 g and 3.8 g, respectively. Dominant frequencies ranged from 15 to 70 Hz, mostly concentrated in the range of 15 to 50 Hz.

For most large earthquakes the dominant frequencies are in the range of 0.5 to 10 Hz. Lower frequencies indicate that a buried pipe would experience less number of cycles of vibration during real earthquakes. Since a pipe has to move with its surrounding soil, vibration amplification due to structure properties is minimal.

No crack due to vibration was found in any of the lining after unloading. It is concluded that the linings had experienced more severe vibrations than any recorded earthquakes in terms of magnitude and number of cycles. The vibration measurements were considered as effective as shaking table tests.

The three-edge-bearing tests showed that the cement-mortar linings were flexible. The lining underwent considerable cracking prior to separation and falling of the linings. Linings only fell after the formation of the plastic hinges in the steel.

The testing program covers a much broader range in types of loading than earthquake loadings. They simulated dead load (loading from roller without vibration, three-edge-bearing test, torsion, and bending tests), low frequency load (cycle tests), large dynamic load at 28 Hz (loading from roller with vibration), large acceleration load with a major frequency content of 0-100 Hz (vibration measurements during shipping), line load with very short duration (drop test), and point load with very short duration (impact test).

From these tests, it is concluded that the test loadings applied to the cement-mortar lining were much more severe and broad-ranged than the design seismic loadings. Therefore, the cement-mortar lining in the underground ERCW pipes is seismically qualified.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

The analysis of a Category I piping system may be terminated at the interface of a nonnuclear safety class piping run by either of the following methods.

1. Terminate the analysis at an in-line anchor designed to prevent transfer of rotations and deflections. The design of the anchor will be sufficient to accommodate reactions from all adjacent piping runs.

2. Extend the analysis and support of the Category I system far enough into the nonnuclear safety class system to ensure that the effects of this adjacent system have been imposed on the Category I system.

Normally, a valve serves as a seismic-nonseismic boundary in a fluid system. The valve capability to maintain a pressure boundary in the event of a seismic event is assured by seismically designing piping on the nonclassified side as described above.

3.7.3.14 Seismic Analyses for Fuel Elements, Control Rod Assemblies, Control Rod

Drives, and Reactor Internals

Fuel assembly component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling. The time history floor response based on a standard seismic time history normalized to SSE levels is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in References [7] and [9].

The Control Rod Drive Mechanisms (CRDM) are seismically analyzed to confirm that system stresses under seismic conditions do not exceed allowable levels as defined by the ASME Boiler and Pressure Vessel Code Section III for 'upset' and 'faulted' conditions. Based on these stress criteria, the allowable seismic stresses in terms of bending moments in the structure are determined. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation, and the resultant seismic bending moments along the length of the CRDM are calculated. These values are then compared to the allowable seismic bending moments for the equipment, to ensure adequacy of the design.

The seismic qualification of Watts Bar reactor vessel internals is demonstrated using a generic basis for a four loop plant. The generic basis or analysis consists of generic design response spectra and generic reactor vessel supports which envelope the analogous specific Watts Bar values.

The generic seismic analysis of the reactor internals is conducted in accordance with the guidelines specified in Regulatory Guide 1.92. The seismic analysis determines the response of the reactor internals to OBE and SSE vertical and horizontal seismic shock components. The horizontal and vertical seismic analysis use the modal response spectrum method and the WECAN general purpose finite element program to determine the internals response. The method used to obtain the combined response of the modal spectral responses is square-root-of-the-sum-of-the squares (SRSS).

The effect of closely spaced modes is considered using the Ten Percent Method (Regulatory Guide 1.92, Paragraph 1.2.2); however, the effect has been shown to be insignificant. The maximum or total seismic response value of the reactor internals is obtained by taking the SRSS of the maximum values of the co-directional responses due to the three components of earthquake motion. In general, this combination is made in the Stress Analysis section of the particular structural component.

When appropriate (e.g., simple beam analysis) LOCA and SSE loads are combined on a reactor internals structural component basis per the SRSS method, the resultant stress intensities calculated. For more complex structural geometries (e.g., core barrel shell) the stress components due to LOCA and SSE are combined either by absolute sum or SRSS, preserving the appropriate signs. These stress components are used to determine the stress intensity for the structural component. For the LOCA, the maximum stresses from the time history response are used. Since the seismic stresses are calculated using response spectrum techniques, the responses are unsigned; therefore, when the LOCA and SSE stresses are combined, the most unfavorable sign convention for the SSE is assumed. The horizontal and vertical seismic models contain 118 and 27 active dynamic degrees of freedom, respectively. Results from the modal analysis of the horizontal and vertical systems indicates, in general, 17 and 3 modes

present with frequencies less than 33 Hz.

In developing the seismic model of the reactor vessel and internals, a systematic approach was used to ensure that basic fundamental frequencies, i.e., both component and system frequencies are described and inherent in the mathematical models. The approach used to verify the mathematical modeling of reactor vessel and internals was to compare and require that the system frequencies and mode shapes from the mathematical models to be in agreement with plant test and scale model test data.

In determining the seismic response of the reactor system due to the excitation of unidirectional shock spectrum, those modes contributing to the first 80-90% of total system mass was considered in the solution.

Hydrodynamic mass effects, for both horizontal and vertical directions, was included in the reactor vessel-internals system models. The numerical values for the various hydrodynamic masses effects within the reactor system is based on scale model and plant tests and applicable analytical expressions, e.g., Fritz, Fritz & Kiss, etc.

The effect of significant nonlinearities in the reactor system, i.e., gaps between reactor vessel and internals on the seismic response is considered in the system analysis. The nonlinearities due to the gaps are included by determining an effective stiffness at the gap location. The validity of this approach has been investigated and found to be conservative for the frequency response range of the reactor internals.

The structural damping values used in the system seismic analysis are in accordance with Regulatory Guide 1.61; i.e., 2 and 4 percent for OBE and SSE, respectively.

In addition, the stiffness of the primary piping and the stiffness of reactor vessel supports are considered in the analysis. Coupling effects between the horizontal and vertical directions are insignificant and are not considered in the analysis.

The frequency response for the Watts Bar reactor vessel internals system is enveloped by the frequency response of the four loop reactor internals which uses the generic vessel support stiffness. The generic frequency response of four loop reactor internals results in acceleration values on the generic response spectra curve. The generic spectra envelopes the specific Watts Bar spectra by a considerable margin and therefore, the loads for the four loop generic analysis envelope the loads for Watts Bar. Consequently, seismic qualification of the Watts Bar reactor internals is demonstrated since the four loop reactor internals have been qualified on a generic basis.

3.7.3.15 Analysis Procedure for Damping

The specific percentages of critical damping value used for Category I structures, systems, and components are provided in Tables 3.7-2 and 3.7-24.

3.7.3.16 Seismic Analysis and Qualification of Category I Equipment Other Than NSSS

All seismic Category I floor or wall-mounted mechanical and electrical equipment was analyzed or tested and designed to withstand seismic loadings in the horizontal and vertical directions. The floor response spectra obtained from the analysis of structures were used in the analyses. Each procurement specification for equipment contained the particular floor response spectra curve for the floor on which the equipment is located. Depending on the relative rigidity and/or the complexity of the equipment being analyzed, the vendor could use one of the following four methods to qualify the equipment:

1. Dynamic analysis method,
2. Simplified dynamic analysis method,
3. Equivalent static load method,
4. Testing method.

The basis used for selection of the appropriate accelerations used in the above paragraph is described in further detail in Section 3.7.3.16.2. Table 3.7-25 identifies how each Seismic Category I item was qualified.

Equipment is considered to be rigid for seismic design if the first natural frequency is equal to or more than 33 cycles per second.

The Watts Bar Category I electrical and mechanical equipment seismic qualification program is consistent with the guidance provided by the NRC Standard Review Plan (NUREG-0800), Revision 2, July 1981, Section 3.10, acceptance criteria for plants with Construction Permit applications docketed before October 27, 1972. The equipment has been seismically qualified either in direct compliance with IEEE Std. 344-1975/Regulatory Guide 1.100 (equipment procured after September 1, 1974), or in accordance with a program which provided as a minimum, qualification to the requirements of IEEE 344-1971 and in addition addressed the guidelines of SRP 3.10.

3.7.3.16.1 Dynamic Analysis Method For Equipment and Components

Equipment that is rigid and rigidly attached to its support structure was analyzed for a g-loading equal to the acceleration of the supporting structure at the appropriate elevation.

For nonrigid, structurally simple equipment, the dynamic model consisted of one mass and one spring. Keeping the values of the mass and the spring constant, the natural period of the equipment was determined. The natural period, together with the appropriate damping value, was used to enter the appropriate acceleration response spectrum to obtain the equipment acceleration in units of g's. The corresponding inertia force was obtained by multiplying the weight times the acceleration.

If the equipment is structurally complex to the extent that a single-degree-of-freedom-system model does not adequately represent the action of the structure to dynamic loads, then a multi-degree-of-freedom model was used with a complete multi-degree-of-freedom analysis. Enough modes were considered to adequately represent the response of the equipment.

3.7.3.16.2 Simplified Dynamic Analysis Method For Equipment and Components

In the simplified dynamic analysis method, the acceleration value corresponding to the maximum shown on the response spectrum curve is used in qualifying the equipment. The forces on the equipment are determined by multiplying the equipment weight times the acceleration. This provides an acceptable method of analysis providing one of the following criteria is met:

1. The item of equipment is simple enough to be adequately modeled by a simple one-degree-of-freedom spring-mass system.

2. The item of equipment is not simple but its fundamental frequency is greater than the rigid frequency. The rigid frequency is defined as that frequency of the floor response spectrum above which there is no acceleration amplification.

3. The item of equipment is not simple and its fundamental frequency is lower than the rigid frequency but its other frequencies are higher than the rigid frequency.

If the equipment can be shown to meet one of these criteria, any amplification due to internal dynamics will not cause stresses greater than those obtained by using the peak value of the floor response spectrum. All of the equipment listed in Table 3.7-25 as having been analyzed by the simplified dynamic analysis method has been reviewed to verify that it meets one of these criteria.

The method described above is conservative since the maximum acceleration, regardless of the frequency of the equipment, is used.

3.7.3.16.3 Equivalent Static Load Method

The description of equivalent load method and its applicability are detailed in Section 3.7.3.5.

3.7.3.16.4 Testing Method

Equipment that did not lend itself to mathematical modeling and structural analysis to determine no loss of function was evaluated by actual vibration testing. The seismic qualification of mechanical equipment, instrumentation and electric equipment are described in Sections 3.9 and 3.10, respectively.

3.7.3.16.5 Equipment and Component Mounting Considerations

Seismic loads for vendor-supplied floor or wall mounted Category I equipment and fluid system component (equipment/component) assemblies and their TVA-designed supports and/or anchorages are determined with consideration of the damping values and stiffness of each. Damping values for these equipment/component assemblies and their bolted or welded structural steel supports and/or anchorages are as indicated in Table 3.7-2. Most of the TVA-designed supports and/or anchorages are effectively rigid; e.g., they do not result in significant amplification of the building structure seismic input. When a TVA-designed support and/or anchorage is not effectively rigid a coupled analysis of the equipment and/or component assembly and its support and/or anchorage is performed using composite modal damping response spectrum analysis techniques.

Examples of vendor-supplied floor or wall mounted mechanical equipment/ component assemblies include: tanks, heat exchangers, diesel generator sets, air handling units, chiller units, compressor assemblies, fan assemblies, and pumps. Electrical equipment assemblies include: transformers, battery racks, instruments and control (I/C) cabinets, I/C panels, and I/C racks.

Seismic loads for line-mounted Category I equipment/components and their mountings are determined from analysis of the subsystems on which they are mounted. The line-mounted equipment/component is tested or analyzed using device qualification techniques as described in Section 3.7.3.6.3. Mass and stiffness characteristics of the equipment/components are included in the subsystem analysis when significant to its seismic response. For example, Section 3.7.3.11.1 describes the modeling of valves in Category I piping subsystems. The subsystem response at the equipment/component location is kept below the device qualification level of the equipment/component. Local mounting brackets for line-mounted equipment/components are seismically qualified with the equipment/component (as part of the device) or they are designed to be effectively rigid. In this case, effectively rigid means the local mounting brackets do not result in significant amplification of the seismic input from the subsystem.

Examples of line-mounted mechanical and electrical equipment/components include: valves, HVAC dampers, and locally-mounted I/C devices of all types.

The techniques described in this section ensure compatibility of the seismic loads for qualification of the Category I equipment/components and the predicted seismic responses of structures and subsystems to which they are mounted.

3.7.3.17 <u>Seismic Analysis and Design of HVAC Duct and Duct Support Systems</u>

This section addresses the analysis and design of Category I and I(L) (see Sections 3.2.1 and 3.2.2.7) HVAC duct and duct support subsystems.

3.7.3.17.1 <u>Description of HVAC Duct and Duct Support Subsystems</u>

HVAC duct and duct support subsystems consist of continuous runs of round and rectangular sheet metal ducts multiple supported along their lengths by structural steel support frames or rod hangers. Scheduled pipe and pipe supports functionally used for an HVAC purpose are treated as piping subsystems in accordance with Section 3.9.

For purpose of analysis, an HVAC duct and duct support subsystem is regarded as any continuous portion of a total duct run and its supports which may be conservatively modeled for evaluation of the loads and stresses within the portion of interest. Significant mass and mass eccentricities of in-line or attached mechanical and electrical components are accounted for in the subsystem model to represent their effects in structural qualifications of the ducts and duct supports in accordance with Sections 3.7.3.17.2 through 3.7.3.17.6. Qualification of the in-line or attached Category I mechanical or electrical equipment and components are in accordance with Sections 3.7.3.6, 3.7.3.16.5, 3.9, and 3.10.

3.7.3.17.2 <u>Applicable Codes, Standards, and Specifications</u>

The following codes, standards, and specifications are applicable to various portions of the HVAC duct and duct support subsystems:

1) SMACNA High Velocity Duct Construction Standards, 2nd Edition, 1969

2) ANSI/ASME N509 Standard, "Nuclear Power Plant Air Cleaning Units and Components," 1976

3) ASTM Standards

4) AISI Specifications for the Design of Cold-Formed Steel Structural Members, 1986 Edition

5) AISC Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th and 8th Editions except welded construction is in accordance with Item 7 below.

6) Manufacturer's Standardization Society of the Valve and Fittings Industry, Standard Practice MSS-SP-58, "Pipe Hangers and Supports - Materials and Design," 1967 Edition

7) American Welding Society, AWS D1.1 Structural Welding Code (See Section 3.8.1.2, Item 4)

8) American Welding Society, AWS D1.3 Structural Welding Code for Sheet Metal

9) American Welding Society, AWS D9.1 Specifications for Welding Code for Sheet Metal

10) NRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2.

3.7.3.17.3 Loads and Load Combinations

HVAC duct and duct support subsystems are designed for the following loads:

DL - Dead loads

OBE - Operating basis earthquake loads

SSE - Safe shutdown earthquake loads

To - Thermal effects and loads during normal operating or shutdown conditions based on the most critical transient or steady-state conditions

Ta - Time varying thermal loads under conditions generated by the design basis accident condition and including To

Note: The maximum value of Ta need not be considered simultaneously with the DBA if time phasing evaluation shows that less than Ta maximum occurs during the DBA transient.

Po - Operating pressure in the duct

Pj - Accident pressure external to the duct due to jet impingement loads from a pipe break. The ducts shall be protected against possible Pj loadings; therefore, this load need not be considered.

Pa - Compartmental pressure loads resulting from a design basis accident.

DBA - Design basis accident dynamic loads due to pressure transient response

F - Airflow induced dynamic loads acting on turning vanes inside the ducts (dependent on the mean airflow velocity). These loads are considered in the following combinations for the duct and duct support elements of the subsystems:

Ducts	Duct Supports
(1) DL + Po + F + OBE	(1) DL + OBE
(2) DL + Po + To + F + OBE	(2) DL + To + OBE
(3) DL + Po + To + F + SSE	(3) DL + To + SSE
(4) DL + Po + Ta + F + OBE + DBA + Pa	(4) DL + Ta + OBE + DBA
(5) DL + Po + Ta + F + SSE + DBA + Pa	(5) DL + Ta + SSE + DBA

3.7.3.17.4 Analysis and Design Procedures

Existing HVAC duct and duct support subsystems that were originally analyzed and designed to Set A seismic response spectra are reevaluated to Set B response spectra as the basis for their qualification. New designs and modification designs to existing subsystems are based on the envelope of Set B+C response spectra.

3.7.3.17.5 Structural Acceptance Criteria

The various elements of the HVAC duct and duct support subsystems are qualified for structural acceptance based on allowable stress criteria.

Allowable stresses for the duct supports are specified in Table 3.7-26.

Allowable stresses for the ducts involve a number of specialized considerations to address both overall and local stresses. Overall stress allowables for duct plate (membrane) elements are developed based on AISI equations. These equations are modified where necessary to adjust for large height-to-thickness and width-to-thickness ratios beyond the normal AISI limits. These adjustments are based on correlations to results of testing, large displacement finite element analyses, and/or industry literature. Additional specialized considerations are made for local stress evaluations. Stress evaluations of the duct stiffeners (including companion-angles) and bolting between these stiffeners are based on AISC allowables. Stress evaluations of the tinners rivets connecting the companion-angles to the duct plate are based on correlation to test results.

In general, unfactored duct stress allowables are used in evaluations of loading combination (1). These stress allowables are multiplied by 1.5 in evaluations of loading combinations (2), (3), (4), and (5). Critical elements of the duct necessary to maintain overall cross section stability are limited to 0.90 Fy except shear is limited to 0.52 Fy and buckling is limited to 0.90 Fcr. Local plate stresses are maintained within 0.90 Fy for mid-plane membrane stresses although surface stresses may exceed yield. The effective cross section of a duct is evaluated based on the post-buckled membrane strength of the duct panels between stiffeners.

3.7.3.17.6 Materials and Quality Control

Some HVAC sheet metal materials installed prior to March 1990 were not always specified and controlled sufficiently to assure known mechanical properties. Samples of these materials were taken from the installed ducts and tested to determine their mechanical properties. The following mechanical properties are used for designs with these materials:

Duct Construction Type	Yield Strength, Fy	Tensile Strength, Fu
SMACNA rectangular (ASTM A525/A527 galvanized sheet)	33 ksi	45 ksi
Specially formed round or rectangular welded (ASTM A570 sheet)	30 ksi	49 ksi
Spiral-welded pipe (ASTM A211)	30 ksi	40 ksi
SMACNA round spiral-lock or longitudinal-lock	20 ksi	37 ksi

HVAC duct sheet metal materials specified after March 1990 and the associated mechanical properties used for designs are as follows:

Specified Material	Yield Strength, Fy	Tensile Strength, Fu
ASTM A527 galvanized steel sheet with ASTM A446 Grade A(minimum) base metal	33 ksi	45 ksi
ASTM A570 Grade 30 (minimum) steel sheet (also used for ASTM A211 spiral-welded pipe)	30 ksi	49 ksi

HVAC structural steel supports are fabricated of ASTM A36 or equivalent or stronger material and are evaluated as having mechanical properties of Fy=36 ksi and Fu=58 ksi.

All steel materials used in the fabrication of HVAC ducts and duct supports are evaluated with a Young's Modulus of E=29 x 10^3 ksi except for those areas within the Reactor Building where reductions must be taken due to extreme accident thermal conditions.

3.7.3.18 Seismic Qualification of Main Control Room Suspended Ceiling and Air Delivery Components

Flexible ducting, triangular ducts, and air bar linear diffusers deliver air flow from the sheet metal ducts located above the Main Control Room (MCR) suspended ceiling to the air space below the ceiling. These air delivery components have been seismically qualified to ensure position retention and structural integrity such that pressure boundary and air flow delivery is maintained during and after the Safe Shutdown Earthquake (SSE).

Seismic qualification of the suspended ceiling and the air delivery components has been accomplished by rigorous time history analysis using the ANSYS computer code. The analysis models non-linear response due to gaps, friction, ceiling support wires, and geometric effects of the ceiling grid work. The seismic time histories correspond to the Control Building response to the Set B SSE at the floor elevation above the suspended ceiling. The combined time histories were then adjusted to account for ±15 percent frequency uncertainty. A factor of safety of at least 1.3 for seismic qualification of the ceiling and air delivery components was demonstrated by increasing the time history motions by 30 percent and verifying that the seismic demand is less than the capacity of the ceiling grid members (including air bars), support wires, and flexible and triangular ducts. The ceiling grid member and support wire capacities are based on classical structural analysis formulas. The flexible and triangular duct capacities were based on analysis for potential failure modes, industry precedents, and the analytical determination that the ceiling grid work remains stable. Other suspended ceiling components, including luminous panels, were shown to retain their position during and after the SSE.

3.7.4 Seismic Instrumentation Program

Seismic instrumentation is provided in order to assess the effects on the plant of earthquakes which may cause exceedance of the Operating Basis Earthquake (OBE = 0.09g horizontal and 0.06g vertical ground acceleration). The seismic monitoring system (SMS) is not safety-related, nor does it have any effect on safety-related systems or components. The components of the SMS are selected to emphasize accuracy and reliability. The instrumentation program is described in the following subsections.

3.7.4.1 Comparison with Regulatory Guide 1.12

The instrumentation is described in Section 3.7.4.2 below and meets the requirements of Regulatory Guide 1.12.

3.7.4.2 Location and Description of Instrumentation

The seismic instrumentation locations are shown in Figures 3.7-39 through 3.7-45.

Instrumentation consists of the following:

1. A strong motion triaxial accelerometer at each of the following locations:

   a. Elevation 702.78, Unit 1 Reactor Building, on the floor slab in the annulus between the Shield Building and the Steel Containment Vessel as shown in Figure 3.7-39.

   b. Elevation 756.63, Unit 1 Reactor Building, on the floor slab as shown in Figure 3.7-40.

   c. Elevation 742.0, Diesel Generator Building, on the base slab as shown in Figure 3.7-41.

   These accelerometers are connected to digital recorders (See Item 3). The recording system is located in the Control Building. The full scale range of the transducers is 0 to 1.0g with a bandwidth of 0 Hz to 50 Hz and a temperature effect of less than 2% per 100°F change.

2. A triaxial strong motion accelerograph with a range of 0g to 2g at Elevation 757 in the Auxiliary Building contains an internal battery backup and is capable of digitally recording a minimum of 25 minutes of data with a minimum of 3 seconds of pre-event memory. An internal seismic trigger with a bandwidth of 0.1 to 12.5 Hz actuates the recording system when a threshold acceleration level is sensed.
3. A seismic instrumentation panel board located at Elevation 708 in the Control Building as shown in Figure 3.7-42. The panel board houses a centralized SMS consisting of a recorder panel, a central controller assembly, a display panel, an alarm panel, and a printer panel. A description of each item mounted on the panel board is given below.

   a) Two recorder panels containing a total of three digital recorders capable of 18-bit resolution. The three strong motion accelerometers of Items 1a, 1b, and 1c above provide input to the recorders. Each digital recorder contains three channels and is capable of recording a minimum of 25 minutes of data with a minimum of 3 seconds of pre-event memory. Each recorder has an internal trigger with a bandwidth of 0.1 to 12.5 Hz which constantly monitors its interconnected triaxial accelerometer. When one of the recorders senses a seismic event, an interconnected network causes the other recorders to trigger and record data at the same time to ensure time-synchronized event-data files. The trigger threshold is set to initiate recording when the acceleration at the containment foundation exceeds 0.01g. A signal is also sent to the alarm panel to indicate that the system is recording (See Item 3c). The recorders can operate for up to 36 hours on internal batteries.

   b) A central controller consisting of an industrial computer and custom software which provides a user interface in a multi-task operating system that supports simultaneous acquisition and interrogation. The controller is powered by 120V AC power.

The central controller retrieves data files from the digital recorders after an event and performs automatic analysis on the data. The event analysis capabilities include calculation of the spectral content of the recorded data and comparison to the site OBE design basis response spectrum. The results of the analysis are displayed on the LCD display panel, sent to a printer, and saved to disk for later off-line analysis. The central controller's software capabilities also include automatic event alarm and annunciation, as well as configurable built in tests of the components comprising the centralized system.

c) An alarm panel containing visual alarms to locally indicate that a seismic event has been recorded, that the OBE site design response spectrum has been exceeded in a damaging frequency range, and to indicate either loss of AC or DC power. The seismic event alarm is triggered by the recorder panels; while the OBE exceedance alarm (See Item 4) is triggered by the central controller. Activation of either the event alarm or exceedance alarm also causes corresponding windows on an annunciator panel in the Main Control Room to illuminate.

d) A display panel to provide a visual display for operation of the centralized system.

e) A printer panel to provide a permanent copy of operational data and event analysis results.

4. Annunciator lights mounted on a window box located on Panel 1-M-15, Main Control Room, Control Building, as shown by reference in Figure 3.7-43. The messages displayed on the annunciator windows in the Main Control Room are 'Seismic Recording Initiated,' 'OBE Spectra Exceeded,' and 'Seismic Instrumentation Loss of Power.'

The basis for the selection of the Reactor Building for installation of seismic instrumentation is that it is the rock-supported building most important to safety. The basis for the selection of the Diesel Generator Building is that it is the soil-supported building most important to safety. The basis for the selection of the Auxiliary Building is that it is a rock-supported structure outside containment.

Steps for utilization of the data recorded by the above described instrumentation are provided in Sections 3.7.4.4 and 3.7.4.5 below.

3.7.4.3 Control Room Operator Notification

The operator receives three annunciation signals in the Main Control Room.These annunciations are independent of each other. The first annunciation is ‘Seismic Instrumentation Loss of Power,’ which serves to provide warning of equipment operability problems under normal conditions as well as following a seismic event. The next annunciation is provided by the recorder panel described in Item 3a, Section 3.7.4.2, which informs the operator that a seismic event is being recorded. This annunciation indicates that one of the triggers for the digital recorders sensed seismic motion in excess of 0.01g.

The final annunciation signal (‘OBE Spectra Exceeded’) is received later and is provided by the central controller described in Item 3b, Section 3.7.4.2, and is only received if the event-analysis software indicates that the site OBE site design response spectrum has been exceeded in a potentially damaging frequency range, i.e., at any frequency between 2 to 10 Hz, or the design response spectral velocity has been exceeded between 1 to 2 Hz.

The basis for establishing the OBE design response spectrum for the levels at which control room operator notification is required is that the design of structures, systems, and components for loading combinations, which include OBE, is to code allowable stress levels which are well within the elastic limit of the materials.

3.7.4.4 Controlled Shutdown Logic

The operator will utilize input from multiple sources to determine the need for a controlled shutdown following the seismic event. The decision for a controlled shutdown will be based primarily on an assessment of the actual damage potential of the event. The event analysis data from the SMS will be reviewed to confirm the 'OBE Spectra Exceeded' alarm. The operator may also confirm that ground motion was sensed by plant personnel and/or confirm the occurrence of the seismic event with the National Earthquake Center. Walkdowns of key plant structures, systems, and components will be performed following the seismic event. The walkdowns will be performed using the guidance of Reference [26], and will include checks of the neutron flux monitoring sensors and containment isolation system. If the 'OBE Spectra Exceeded' alarm is confirmed by analysis and the event is confirmed by plant personnel, data from these other sources will be used to determine the best manner in which to proceed with plant shutdown. If a seismic event occurs which does not result in an OBE exceedance (as determined either by annunciation or subsequent analysis), a plant walkdown may be performed to confirm plant condition, however plant shutdown will not be required unless it is determined to be necessary by the operator based on consideration of available information.

The assessment of the damage potential will be made using the OBE Exceedance Criteria developed by the Electric Power Research Institute (EPRI).[25], [26], [27], [28], [29], and [30] As noted above, the indication of damage potential will be provided by event analysis software installed on the centralized SMS described in Section 3.7.4.2. The analysis will be performed for the uncorrected accelerograms recorded from the strong motion triaxial accelerometer located on the base slab in the annulus of the Unit 1 Reactor Building (Item 1a of Section 3.7.4.2). Use of the uncorrected accelerograms is known to be conservative. The basis for use of the seismic motion on the base slab of this structure is that the site OBE design response spectrum is defined at top-of-rock, which corresponds to the base slab location. An engineer will confirm the event analysis results from the SMS.

The EPRI OBE Exceedance Criteria uses two indicators of damage potential. The first indicator of damage potential is specified as the cumulative absolute velocity (CAV), of the accelegram. A meaningful usage of the CAV requires that the recorded data be obtained by an accelerometer mounted in the free-field. As noted above, the OBE design spectrum for WBN is defined as occurring at top-of-rock (i.e., foundation level of the rock-supported structures); whereas, free-field is defined as top-of-soil at sufficient distance from nearby structures to preclude interference/interaction effects. The SMS does not have a free-field accelerometer. Therefore, the shutdown logic adopted will concede CAV exceedance and base the decision on the need for a controlled shutdown solely on the second indicator, as discussed below.

In the absence of data from a free-field accelerometer, the second indicator is an evaluation of the frequency at which the OBE spectrum is exceeded. This criterion is based on research indicating that exceedances above a frequency of 10 Hz are not damaging to nuclear plant structures, systems and components. Two measures of damage potential are used for this second indicator. The OBE design response spectrum is considered exceeded if the 5% damped response spectra generated for any one of the three components of the uncorrected accelerograms from the Containment Building base slab is larger than:

1. The corresponding OBE design response spectral acceleration in a frequency range between 2 - 10 Hz, or,

2. The corresponding OBE design response spectral velocity for frequencies between 1 - 2 Hz.

Basing shutdown logic on the actual damage potential reduces shutdown risk by avoidance of unnecessary shutdowns while ensuring that the operator has the information on plant status necessary to make an informed shutdown decision.

3.7.4.5 Comparison of Measured and Predicted Responses

The steps to be followed after the initiation of a controlled shutdown due to OBE exceedance are discussed in the following sections.

3.7.4.5.1 Retrieval of Data

The digital records for the Reactor Building and Diesel Generator Building accelerometers and the strong motion accelerograph in the Auxiliary Building will be retrieved. The accelerometers and the accelerograph will be recalibrated to confirm the accuracy of the recorded area.

3.7.4.5.2 Evaluation of Recorded Earthquake

Corrected accelerograms and corresponding response spectra, will be prepared for event data recorded by the aforementioned accelerometers and accelerograph. The response spectra for the recorded motion at Elevation 757 in the Reactor Building, Elevation 745 in the Diesel Generator Building, and Elevation 757 in the Auxiliary Building will be compared to the corresponding design spectra for the OBE.

The structural response of these buildings to the recorded earthquake will be compared with the OBE design structural response, and if less, no further analysis will be required. If the structural response of these buildings to the recorded earthquake is greater than the OBE design structural response, then floor response spectra, for the same mass points in these buildings as used in the equipment design, will be produced for use in evaluation of mechanical and electrical equipment response.

REFERENCES

1. 'Dynamic Effects of Earthquake on Engineering Structures,' Tennessee Valley Authority, Report No. 8-194, August 1939.

2. Hayashi, Satoshi, 'Analysis and Design of Earth Structures and Foundations,' Syllabus for Earthquake Engineering Fundamentals, August 22 through September 2, 1966, Engineering Extension, Department of Engineering, University of California, Los Angeles.

3. Whitman, R. V. 'Analysis of Foundation Vibrations, 'Vibration in Civil Engineering, 1966, Buttersworth, London.

4. Sequoyah Nuclear Plant Final Safety Analysis Report, Tennessee Valley Authority, Docket Numbers 50-327 and 50-328.

5. Richard, F. E., Jr., J. R. Hall, Jr., R. D. Woods, Vibrations of Soils and Foundations, Prentice-Hall, Incorporated, 1970, New Jersey.

6. N. M. Newmark, 'Design Criteria for Nuclear Reactors Subjected to Earthquake Hazards,' Proceedings, IAEA Panel on Seismic Design and Testing of Nuclear Facilities, Japan Earthquake Engineering Promotion Society, Tokyo, May 1967.

7 T. L. Gesinski, 'Fuel Assembly Safety Analysis For Combined Seismic and Loss-of-Coolant Accident,' WCAP-7950, July 1972.

8. E. L. Vogeding, 'Seismic Testing of Electrical and Control Equipment,' WCAP-7817, and Supplement I, December 1971.

9. Gessinki, L. and D. Chaing, 'Safety Analysis of the 17xl7 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident', WCAP-8288, January 1974.

10. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.60, December 1973.

11. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.61, October 1973.

12. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.92, February 1976.

13. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.84, Revision 25, May 1988.

14. Deleted

15. Newmark, N.M., "Problems in Wave Propagation in Soil and Rock," Proceeding International Symposium on Wave Propagation and Dynamic Properties of Earth Materials, Albuquerque, New Mexico, 1968.

16. Newmark, N.M., "Earthquake Response Analysis of Reactor Structures," Nuclear Engineering and Design, Volume 20, 1972, pages 303-322.

17. Kuesel, T.T., "Earthquake Design Criteria for Subways," ASCE, Journal of the Structural Division, Vol. 95, No. ST6, Proceeding Paper 6616, June 1969, pages 1213-1231.

18. Yeh, G.C.K., "Seismic Analysis of Slender Buried Beams," Bulletin of the Seismological Society of America, Volume 64, No. 5, October 1974, pages 1551-1562.

19. Shah, H.H., and Chu, S.L., "Seismic Analysis of Underground Structural Elements," ASCE, Journal of the Power Division, Vol. 100, No. P01, Proceeding Paper 10648, July 1974, pages 53-62.

20. Goodling, E.C., "Flexibility Analysis of Buried Piping," Joint ASME/CSME Pressure Vessels and Piping Conference, Montreal, Canada, June 25-30, 1978.

21. Goodling, E.C., "Buried Piping - An Analysis Procedure Update," International Symposium on Lifeline Earthquake Engineering, Fourth U. S. National Conference on Pressure Vessels and Piping Technology, ASME, Portland, OR, 1983.

22. Goodling, E.C., "Seismic Stresses in Buried Elbows," ASCE National Convention, Boston, MA, 1979.

23. "Seismic Response of Buried Pipes and Structural Components," ASCE Structural Division Committee on Nuclear Structures and Materials, ASCE, New York, NY, 1983.

24. Iqbal, M.A., and Goodling, E.C., "Seismic Design of Buried Piping," Second ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, New Orleans, LA, 1975.

25. EPRI Report NP-5930, “A Criterion for Determining Exceedance of the Operating Basis Earthquake,” July 1988.

26. EPRI Report NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989.

27. EPRI Report TR-100082, "Standardization of the Cumulative Absolute Velocity," December 1991.

28. EPRI Report TR-104239, "Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guideline for Implementation," June 1994.

29. NRC Memorandum from Stuart A. Treby, OGC, to Goutam Bagchi, NRR, concerning "Interpretaion of Part 100, Appendix A Regarding: Proposed Guidelines for Determining when Operating Basis Earthquake is Exceeded," dated May 3, 1988.

30. Nuclear Regulatory Commission Regulatory Guide 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions," March 1997.

31. International Building Code, International Code Council, Inc., March 2000.

TABLE 3.7-1

PERIODS FOR SPECTRAL VALUES[1]

SET A

Range of Periods, T (sec)	Increment, T (sec)
0.03 to 0.10	0.005
0.11 to 0.30	0.010
0.32 to 0.50	0.020
0.55 to 1.0	0.050

SET B AND SET C[2]

Frequency Range (hertz)	Increment (hertz)
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

NOTES:

(1) Spectral values were computed for the periods/frequencies shown above in addition to the natural frequencies of the structure.

(2) Except for the Auxiliary-Control Building where Set A periods were used in Set C analysis.

TABLE 3.7-2

STRUCTURAL DAMPING RATIOS USED IN

ANALYSIS OF CATEGORY I STRUCTURES, SYSTEMS AND COMPONENTS

	Set A		Set B[8]		Set C	
CATEGORY I STRUCTURES	OBE	SSE	OBE	SSE	OBE	SSE
Reactor Building -						
Interior Concrete Structure	2	5[1]	4	7	2	5[1]
Steel Containment Vessel	1	1	2	4	1	1
Shield Building	2	5[1]	4	7	2	5[1]
Additional Diesel Generator Bldg	N/A	N/A	4	7	5	5
Other Concrete Structures	5	5[1]	4	7	5	5[1]
Refueling Water Storage Tank	2	2	2	4	2	2
Other Welded Steel Structures[4]	2	2[2]	2	4	2	2[2]
Other Bolted Steel Structures[4]	5	5[1]	4	7	5	5[1]

	Set A		SET B[8]		Set B+C	
CATEGORY I SYSTEMS AND COMPONENTS	OBE	SSE	OBE	SSE	OBE	SSE
Piping -						
12" or Larger	0.5	1	2	3	2	3
Less than 12"	0.5	1	1	2	1	2
Optional (Code Case)	N/A	N/A	Note 7	Note 7	Note 7	Note 7
Cable Tray	4	5	4	7	4	7
Conduit	Note 5	2	4	7	4	7
HVAC -						
Companion Angle	Note 6	7	4	7	4	7
Pocket Lock	Note 6	7	7	7	7	7
Welded Duct	Note 3	Note 3	2	4	2	4
Equipment/Components	2	3	2	3	2	3

NOTES:

(1) Damping value of 7% may be used when stress levels are at or near yield.
(2) Damping value of 5% may be used when stress levels are at or near yield.
(3) Not addressed.
(4) Includes TVA-designed supports and anchorage for equipment and component assemblies.
(5) Design is based on SSE only.
(6) OBE loads are assumed to be ½ of SSE loads.
(7) N-411-1--Damping values from ASME Code Case N-411-1.
(8) For Set B, OBE and SSE are site-specific OBE and SSE

TABLE 3.7-3

SUPPORTING MEDIA FOR CATEGORY I STRUCTURES

Rock-Supported Structures (Set A, Set B, and Set C Analyses)

Structure	Shear Wave Velocity of Bedrock, fps
Shield Building	5900
Interior Concrete Structure	5900
Auxiliary-Control Building	5900
Steel Containment Vessel	5900
North Steam Valve Room	5900
ERCW Intake Pumping Station	5900

Soil-Supported Structures

Structure	Shear Wave Velocities (fps)[1] Set A Analysis	Set B and Set C Analyses
Diesel Generator Building	1650	Note 2
Waste-Packaging Area	1650	N/A
Refueling Water Storage Tank	1008	Note 2
ERCW Pipe Tunnels	1150	Note 2

Pile-Supported Structures

Structure	Shear Wave Velocities (fps)[1] Set A Analysis	Set B and Set C Analyses
Condensate Demineralizer Waste Evaporator Building	761	N/A
Additional Diesel Generator Building	N/A	Note 2

NOTES:

(1) Shear wave velocities are defined at zero shear strain.
(2) Shear wave velocities for Set B and Set C analyses are related to the soil layer, overburden, shear strain, etc. See Section 2.5 (historical information) for a description of the supporting media dynamic soil properties.

TABLE 3.7-4

SHIELD BUILDING STRUCTURAL PROPERTIES (SET A)

$E = 545,000\ K/FT^2$ $G = 218,000\ K/FT^2$

Element No.	Length Ft	Area Ft^2	Moment of Inertia. Ft^4	Mass Pt. No.	Weight Kips
1	6.67	1194	2435×10^3	1	789.83
2	2.27	1194	2435×10^3	2	556.11
3	4.06	1194	2435×10^3	3	710.40
4	4.06	1174	2435×10^3	4	704.40
5	4.06	1174	2398×10^3	5	710.40
6	4.06	1194	2398×10^6	6	974.30
7	6.92	1194	2435×10^3	7	1590.40
8	10.88	1194	2435×10^3	8	1298.50
9	3.62	1194	2435×10^3	9	648.34
10	3.62	1194	2435×10^3	10	649.24
11	3.63	1194	2435×10^3	11	608.03
12	3.33	1133	2202×10^3	12	565.93
13	3.33	1133	2202×10^3	13	566.78
14	3.43	1133	2202×10^3	14	570.53
15	3.42	1148	2250×10^3	15	573.43
16	3.42	1148	2250×10^3	16	574.29
17	3.43	1148	2250×10^3	17	622.79
18	4.28	1194	2435×10^3	18	750.43
19	4.28	1194	2435×10^3	19	750.43
20	4.25	1194	2435×10^3	20	1500.90
21	12.57	1194	2435×10^3	21	2251.30
22	12.57	1194	2435×10^3	22	2252.20
23	12.58	1194	2435×10^3	23	2253.10
24	12.58	1194	2435×10^3	24	2253.10
25	12.58	1194	2435×10^3	25	6893.50

TABLE 3.7-4A (Sheet 1 of 3)

LUMPED-MASS MODEL PROPERTIES
OF SHIELD BUILDING MODEL (SET B AND SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)	Mass Moment of Inertias (10^4 x $k\text{-}ft\text{-}sec^2$) J_z	Axial Area (ft^2) A	Shear Areas (ft^2) $A_x = A_y$	Moment of Inertias (10^4 x ft^4) J	$I_{xx} = I_{yy}$
852.1	(Note 1)	42.52				
			1194	597	487	243.5
839.5	69.97	28.52				
			1194	597	487	243.5
826.9	69.97	28.52				
			1194	597	487	243.5
814.3	69.94	28.52				
			1194	597	487	243.5
802.1	69.92	28.52				
			1194	597	487	243.5
789.8	46.61	19.01				
			1194	597	487	243.5
785.6	23.31	9.51				
			1194	597	487	243.5
781.3	23.31	9.51				
			1194	597	487	243.5
777.0	19.34	8.39				
			1148	574	450	225
773.6	17.84	7.26				
			1148	574	450	225

Note 1: Horizontal Mass = 214.1
Vertical Mass = 134.3

TABLE 3.7-4A (Sheet 2 of 3)

LUMPED-MASS MODEL PROPERTIES
OF SHIELD BUILDING MODEL (SET B AND SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)	Mass Moment of Inertias ($10^4 \times k\text{-}ft\text{-}sec^2$) J_z	Axial Area (ft^2) A	Shear Areas (ft^2) $A_x = A_y$	Moment of Inertias ($10^4 \times ft^4$) J	$I_{xx} = I_{yy}$
770.1	17.81	7.26				
			1148	574	450	225
766.7	17.72	7.23				
			1133	567	440.4	220.2
763.3	17.60	7.16				
			1133	567	440.4	220.2
760.0	17.58	7.16				
			1133	567	440.4	220.0
756.6	18.88	7.70				
			1194	597	487	243.5
753.0	20.16	8.21				
			1194	597	487	243.5
749.4	20.13	8.21				
			1194	597	487	243.5
745.8	40.33	16.45				
			1194	597	487	243.5
734.9	49.39	20.15				
			1194	597	487	243.5
728.0	30.26	12.34				
			1194	597	487	243.5

TABLE 3.7-4A (Sheet 3 of 3)

LUMPED-MASS MODEL PROPERTIES
OF SHIELD BUILDING MODEL (SET B AND SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)	Mass Moment of Inertias (10^4 x $k\text{-}ft\text{-}sec^2$) J_z	Axial Area (ft^2) A	Shear Areas (ft^2) $A_x = A_y$	Moment of Inertias (10^4 x ft^4) J	$I_{xx} = I_{yy}$
723.9	22.06	9.01				
			1174	587	479.6	239.8
719.8	21.88	8.94				
			1174	587	479.6	239.8
715.8	22.06	9.01				
			1194	597	487.6	243.5
711.7	17.27	7.04				
			1194	597	487	243.5
709.5	24.53	10.00				
			1194	597	487	243.5
702.8	18.30	7.46				

Dome Vertical SDOF Oscillator

Mass = 79.81 ($k\text{-}sec^2/ft$)
Spring Stiffness = 806 x 10^3 (k/ft)

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

WBN

TABLE 3.7-5

SHIELD BUILDING NAUTRAL PERIODS

	CYLINDRICAL SHELL				DOME	
	Translation Motion		Vertical Motion		Vertical Motion	
Mode No.	Period (Seconds)	Participation Factor	Period (Seconds)	Participation Factor	Period (Seconds)	Participation Factor
1	0.1868	1.326	0.0671	1.232	0.063	2.237
2	0.0951	0.046			0.040	-2.213
3	0.0552	0.580			0.033	1.207
4	0.0313	0.008			0.026	-0.676
5					0.020	1.281

WBN

TABLE 3.7-5A

STEEL CONTAINMENT VESSEL
ELEMENT PROPERTIES

$E = 4{,}176{,}000\ K/Ft^2$ $G = 1{,}670{,}400\ K/Ft^2$

Element No.	Length, Ft	Area (Ft^2)	Torsion Constant (Ft^4)	North-South Motion: Moment of Inertia (Ft^4)	North-South Motion: Shear Factor	East-West Motion: Moment of Inertia (Ft^4)	East-West Motion: Shear Factor
1	1.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
2	6.22	45.55	152×10^3	75.3×10^3	2	75.3×10^3	2
3	6.50	45.55	152×10^3	75.3×10^3	2	75.3×10^3	2
4	8.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
5	9.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
6	11.00	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
7	9.50	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
8	3.50	41.55	137×10^3	68.7×10^3	2	68.7×10^3	2
9	6.00	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
10	4.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
11	5.00	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
12	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
13	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
14	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
15	9.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
16	3.50	45.16	149×10^3	74.7×10^3	2	74.7×10^3	2
17	2.50	41.30	137×10^3	68.3×10^3	2	68.3×10^3	2
18	12.00	41.30	129×10^3	64.5×10^3	2	64.5×10^3	2
19	12.46	41.30	108×10^3	54.0×10^3	2	54.0×10^3	2
20	9.54	24.40	456×10^2	22.8×10^3	2	22.8×10^3	2
21	9.00	24.50	268×10^2	13.4×10^3	2	13.4×10^3	2
22	9.00	24.46	81×10^2	40.5×10^2	2	40.5×10^2	2
23	3.00	28.23	36×10	18.2×10	2	18.2×10	2

TABLE 3.7-5B

STEEL CONTAINMENT VESSEL MASS POINT PROPERTIES

Elevations, Ft	Total Horizontal Weight, Kips	Total Vertical Weight, Kips	Weight of Inertia WR^2 $K\text{-}ft^2$	Eccentricity Used in Dynamic Analysis, Ft
703.78	91.87	91.87	305×10^3	0.0
710.00	147.60	147.60	491×10^3	2.43
716.50	227.64	227.64	754×10^3	0.995
724.50	393.44	393.44	$1{,}301 \times 10^3$	0.0
733.50	335.23	335.23	$1{,}108 \times 10^3$	-0.033
744.50	424.10	409.28	$1{,}402 \times 10^3$	0.57
754.00	220.28	190.94	728×10^3	-1.53
757.50	158.12	137.75	523×10^3	-0.82
763.50	310.98	288.44	$1{,}028 \times 10^3$	0.99
768.00	145.52	125.07	481×10^3	1.23
773.00	222.08	191.02	734×10^3	0.25
782.50	407.87	367.35	$1{,}349 \times 10^3$	-0.13
792.00	295.51	254.97	977×10^3	-0.105
801.50	318.18	283.31	$1{,}052 \times 10^3$	-0.052
811.50	216.89	205.01	717×10^3	-0.075
814.50	69.60	64.17	229×10^3	-0.036
817.00	192.21	185.28	630×10^3	0.0
829.00	302.84	302.84	933×10^3	0.0
841.46	183.10	183.10	485×10^3	0.0
851.00	114.66	114.66	227×10^3	0.0
860.00	155.67	155.67	192×10^3	0.0
869.00	84.00	84.00	386×10^2	0.0
872.00	23.25	23.25	186×10	0.0

TABLE 3.7-5C (Sheet 1 of 3)

LUMPED-MASS MODEL PROPERTIES OF STEEL CONTAINMENT VESSEL MODEL

Elevation (ft)	Masses ($K\text{-}sec^2/ft$)		Mass Moment of Inertias (10^2 k-ft-sec^2)	Axial Area (ft^2)	Shear Areas ($10^2 \times ft^2$)	Moment of Inertias ($10^2 \times ft^4$)	
	$M_x = M_y$	M_z	J_z	A	$A_x = A_y$	J	$I_{yy} = I_{xx}$
872.0	0.72	0	0.58				
				28.23	14.11	3.64	1.82
869.0	2.61	0	11.99				
				24.46	12.23	80.98	40.49
860.0	4.83	0	59.54				
				24.50	12.25	268.0	134.0
851.0	3.56	0	70.41				
				24.4	12.2	456.0	228.0
841.5	5.69	0	150.70				
				41.30	20.65	1080.0	540.0
829.0	9.40	0	289.67				
				41.30	20.65	1290.0	645.0
817.0	5.96	0	195.89				
				41.30	20.65	1356.0	682.5
814.5	2.16	28.49	71.02				
				45.16	22.58	1439.0	746.6
811.0	6.74	6.37	222.70				
				45.16	22.58	1439.0	746.6
801.50	9.88	8.80	326.71				
				45.16	22.58	1439.0	746.6

TABLE 3.7-5C (Sheet 2 of 3)

LUMPED-MASS MODEL PROPERTIES OF STEEL CONTAINMENT VESSEL MODEL

Elevation (ft)	Masses ($K\text{-}sec^2/ft$)		Mass Moment of Inertias (10^2 k-ft-sec^2)	Axial Area (ft^2)	Shear Areas ($10^2 \times ft^2$)	Moment of Inertias ($10^2 \times ft^4$)	
	$M_x = M_y$	M_z	J_z	A	$A_x = A_y$	J	$I_{yy} = I_{xx}$
792.0	9.18	7.92	303.43				
				45.16	22.58	1439.0	746.6
782.5	12.67	11.41	418.82				
				45.16	22.58	1439.0	746.6
773.0	6.83	5.93	228.03				
				45.16	22.58	1439.0	746.6
768.0	4.52	3.88	149.43				
				45.16	22.58	1439.0	746.6
763.5	9.66	8.96	319.32				
				45.16	22.58	1493.0	746.6
757.5	4.91	4.28	162.36				
				41.55	20.77	1373.7	686.8
754.0	6.84	5.93	226.20				
				41.55	20.77	1373.7	686.8
744.5	13.17	12.71	435.47				
				41.55	22.77	1373.7	686.8
733.5	10.41	10.41	344.19				
				41.55	20.77	1373.7	686.8
724.5	12.22	12.22	403.98				
				41.55	20.77	1373.7	686.8

TABLE 3.7-5C (Sheet 3 of 3)

LUMPED-MASS MODEL PROPERTIES OF STEEL CONTAINMENT VESSEL MODEL (cont'd)

Elevation (ft)	Masses ($K\text{-}sec^2/ft$) $M_x = M_y$	M_z	Mass Moment of Inertias (10^2 k-ft-sec^2) J_z	Axial Area (ft^2) A	Shear Areas ($10^2 \times ft^2$) $A_x = A_y$	Moment of Inertias ($10^2 \times ft^4$) J	$I_{yy} = I_{xx}$
716.5	7.07	7.07	234.22				
				45.55	22.77	1516.0	752.9
710.0	4.56	4.56	152.55				
				45.55	22.77	1516.0	752.9
703.8	2.85	2.85	94.81				
				41.55	20.77	1374.0	686.8

Dome Vertical SDOF Oscillator

Mass = 6.74 ($k\text{-}sec^2/ft$)
Spring Stiffness = 287×10^3 (k/ft)

Steel Properties

Modulus of Elasticity = 4,176,000 k/ft^2
Poisson's Ratio = 0.25

+X = EAST
+Y = NORTH

WBN

TABLE 3.7-6

INTERIOR CONCRETE ELEMENT PROPERTIES

E_C = 720000 K/Ft^2 G_C = 288000 K/Ft^2

Element No.	Length, Ft	Area, Ft^2	Torsion Constant, Ft^4	East-West Motion: Moment of Inertia, Ft^4	East-West Motion: Shear Factor	North-South Motion: Moment of Inertia, Ft^4	North-South Motion: Shear Factor
1	12.22	1779	1840 x 10^3	1024 x 10^3	1.76	1021 x 10^3	1.79
2	10.00	2107	1700 x 10^3	1849 x 10^3	1.70	1281 x 10^3	2.27
3	9.96	1796	1610 x 10^3	1829 x 10^3	1.49	1271 x 10^3	2.20
4	9.96	1796	1610 x 10^3	1829 x 10^3	1.49	1271 x 10^3	2.20
5	5.36	880	249 x 10^3	990 x 10^3	1.07	320 x 10^3	1.75
6	5.35	880	249 x 10^3	990 x 10^3	1.07	320 x 10^3	1.75
7	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
8	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
9	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
10	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
11	6.73	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
12	6.72	1154	151 x 10^3	707 x 10^3	2.02	1047 x 10^3	1.98
13	11.82	816	1510 x 10^3	755 x 10^3	2.00	755 x 10^3	2.00
14	11.82	816	1510 x 10^3	755 x 10^3	2.00	755 x 10^3	2.00

TABLE 3.7-6A (Sheet 1 of 2)

LUMPED-MASS MODEL PROPERTIES
OF INTERIOR CONCRETE STRUCTURE-HORIZONTAL MODEL - SET B AND SET C

Elevation (ft)	Masses (K-sec^2/ft)	Mass Moment of Inertias (10^4 k-ft-sec^2)	Shear Center (ft)		Shear Areas (ft^2)		Moment of Inertias (10^3 x ft^4)		
		J_z	e_x	e_y	A_x	A_y	I_{xx}	I_{yy}	J
819.0	38.9	4.2	0.0	0.0					
					335	408	755	625	1510
807.8	45.5	8.3	0.0	0.0					
					335	408	755	625	1510
796.0	123.7	13.0	-33.67	3.06					
					445	200	1070	655	840
789.3	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
782.6	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
775.8	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
769.1	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
762.4	58.4	8.0	-33.67	3.06					
					445	200	1070	655	840
755.6	89.8	18.4	28.21	0.90					
					575	540	555	800	250
750.3	40.0	4.4	28.21	0.90					
					575	540	555	800	250

TABLE 3.7-6A (Sheet 2 of 2)

LUMPED-MASS MODEL PROPERTIES
OF INTERIOR CONCRETE STRUCTURE-HORIZONTAL MODEL - SET B AND SET C

Elevation (ft)	Masses ($K\text{-}sec^2/ft$)	Mass Moment of Inertias (10^4 k-ft-sec^2) J_z	Shear Center (ft) e_x	e_y	Shear Areas (ft^2) A_x	A_y	Moment of Inertias (10^3 x ft^4) I_{xx}	I_{yy}	J
744.9	135.2	14.7	-0.25	0.85					
					915	580	1140	1460	1650
735	110	12.6	-0.25	0.85					
					915	580	1140	1460	1650
725	114.5	15.2	3.63	1.76					
					1075	835	1170	1455	1830
715	160.2	23.2	0.54	0.13					
					940	1185	965	825	1815
702.8	1160	230	0.0	0.0					

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

TABLE 3.7-6B (Sheet 1 of 2)

LUMPED-MASS MODEL PROPERTIES OF
INTERIOR CONCRETE STRUCTURE-VERTICAL MODEL - SET B AND SET C

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)	Mass Moment of Inertia ($10^4 k\text{-}ft\text{-}sec^2$)	Location of Centroid (ft)		Areas (ft^2)
		J	d_x	d_y	A
819.6	38.9	4.2			
			0.0	0.0	816
807.8	45.5	8.3			
			0.0	0.0	816
796.0	123.7	13.0			
			-4.2	0.61	1154
789.3	58.4	8.0			
			-4.2	0.61	1154
782.6	58.4	8.0			
			-4.2	0.61	1154
775.8	58.4	8.0			
			-4.2	0.61	1154
769.1	58.4	8.0			
			-4.2	0.61	1154
762.4	58.4	8.0			
			-4.2	0.61	1154
755.6	89.8	18.4			
			16.60	0.29	880
750.3	40.0	4.4			
			16.60	0.29	880

TABLE 3.7-6B (Sheet 2 of 2)

LUMPED-MASS MODEL PROPERTIES OF
INTERIOR CONCRETE STRUCTURE-VERTICAL MODEL - SET B AND SET C

Elevation (ft)	Masses (k-sec^2/ft)	Mass Moment of Inertia (10^4k-ft-sec^2) J	Location of Centroid Areas (ft)(ft^2)		
			d_x	d_y	A
744.9	135.2	14.7			
			8.56	0.60	1796
735	110	12.6			
			8.56	0.60	1796
725	114.5	15.2			
			6.52	0.98	2107
715	160.2	23.2			
			0.25	0.08	1779
702.8	1160	230			

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

WBN

TABLE 3.7-7

INTERIOR CONCRETE STRUCTURE - MASS POINT PROPERTIES (SET A)

Point No.	Total Wt. Kips	Equip. Wt. Kips	WR^2 $K\text{-}Ft^2$	Eccentrictiy, Ft E-W Motion	Eccentrictiy, Ft N-S Motion
1	8203	3588	7.48×10^6	0.0	3.8
2	4539	1619	4.89×10^6	0.0	7.2
3	3574	894	4.07×10^6	0.0	11.0
4	4352	1211	4.74×10^6	0.0	- 3.4
5	1288	578	1.41×10^6	0.0	-12.6
6	5451	3397	5.92×10^6	0.0	21.6
7	1879	714	2.56×10^6	0.0	43.7
8	1879	714	2.56×10^6	0.0	43.7
9	1879	714	2.56×10^6	0.0	43.7
10	1879	714	2.56×10^6	0.0	43.7
11	1879	714	2.56×10^6	0.0	43.7
12	3983	1734	4.16×10^6	0.0	17.7
13	1464	14	2.67×10^6	0.0	0.0
14	1253	588	1.34×10^6	0.0	0.0

WBN

TABLE 3.7-8

INTERIOR CONCRETE STRUCTURE - NORMAL MODES OF VIBRATION (SET A)

	East-West Motion		North-South Motion		Vertical Motion	
Mode No.	Frequency, cps (Period, sec)	Participation Factor	Frequency, cps (Period, sec)	Participation Factor	Frequency, cps (Period, sec)	Participation Factor
1	8.81 (0.114)	1.665	4.96 (0.202)	1.600	22.68 (0.044)	1.408
2	23.82 (0.042)	-0.950	9.64 (0.104)	2.329		
3			14.88 (0.067)	0.032		
4			22.91 (0.044)	-0.852		
5			24.88 (0.040)	0.887		
6			32.34 (0.031)	1.760		

WBN

TABLE 3.7-9

AUXILIARY BUILDING ELEMENT PROPERTIES (SET A, SET B, AND SET C)

E_C =590,000 k/ft^2; G_C = 236,000 k/ft^2 (For Set A)

E_C = 590,000 k/ft^2; G_C = 252,800 k/ft^2 (For Set B and Set C)

				North-South Motion		East-West Motion	
Elevation	Length (ft)	Area (ft^2)	Torsion Constant (ft^4)	Moment of Inertia (ft^4)	Shear Area (ft^2)	Moment of Inertia (ft^4)	Shear Area (ft^2)
692.00	7.62	11,172	2,893x10^4	11,914x10^4	4,968	5,728x10^4	4,860
699.62	7.63	11,172	2,893x10^4	11,914x10^4	4,968	5,728x10^4	4,860
707.25	4.25	11,172	2,893x10^4	11,914x10^4	4,968	5,728x10^4	4,860
711.50	8.38	8,410	2,125x10^4	8,570x10^4	2,244	5,708x10^4	3,592
719.88	8.37	8,410	2,125x10^4	8,570x10^4	2,244	5,708x10^4	3,592
728.25	8.25	7,902	2,302x10^4	8,178x10^4	2,108	4,801x10^4	3,340
736.50	9.25	7,340	2,640x10^4	8,052x10^4	2,174	4,645x10^4	2,867
745.75	9.75	7,340	2,640x10^4	8,052x10^4	2,174	4,645x10^4	2,867
755.50	16.00	5,609	2,746x10^4	5,961x10^4	1,503	4,469x10^4	2,310
771.50	10.00	4,269	1,820x10^4	3,782x10^4	1,242	2,460x10^4	1,609
781.50	4.00	4,269	1,820x10^4	3,782x10^4	1,242	2,460x10^4	1,609
800.50	15.00	1,495	286x10^4	1,037x10^4	432	672x10^4	570
814.25	13.75	781	233x10^4	601x10^4	319	86x10^4	201

WBN

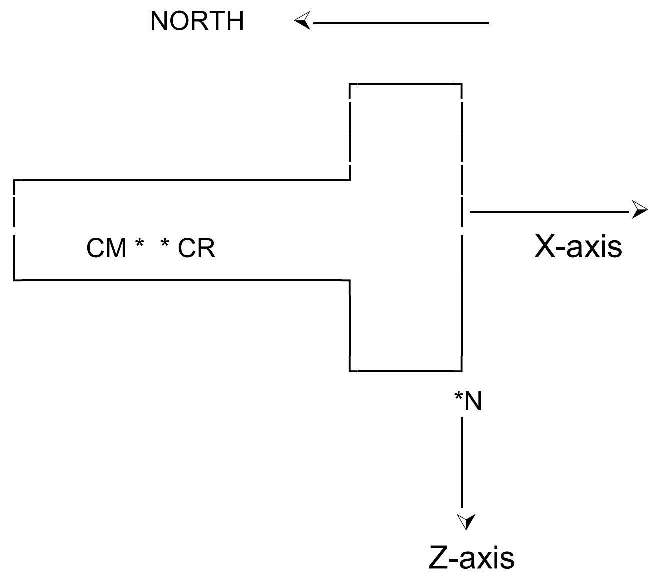
TABLE 3.7-9A

AUXILIARY BUILDING NODAL COORDINATES (SET B AND SET C)

Legend

CM - Center of mass

CR - Center of rigidity



Elev. Y-Coord	CM Mode X-Coord	CR Mode X-Coordinate H-Model	CR Mode X-Coordinate V-Model	CM&CR Mode Z-Coord
814.25	-213.75			0.00
		-230.28	-218.36	0.00
800.50	-207.42			0.00
		-178.72	-205.68	0.00
785.50	-135.51			0.00
		-84.47	-139.18	0.00
781.50	-154.65			0.00
		-84.47	-139.18	0.00
771.50	-96.66			0.00
		-67.32	-123.15	0.00
755.50	-140.46			0.00
		-79.42	-158.35	0.00
745.75	-156.75			0.00
		-79.42	-158.35	0.00
736.50	-145.28			0.00
		-84.25	-160.35	0.00
728.50	-155.67			0.00
		-85.92	-162.52	0.00
719.88	-149.55			0.00
		-85.92	-162.52	0.00
711.50	-137.02			0.00
		-65.17	-161.35	0.00
707.25	-125.61			0.00
		-65.17	-161.35	0.00
699.62	-158.00			0.00
		-65.17	-161.35	0.00
692.00	Fixed base	-65.17	-161.35	0.00

WBN

TABLE 3.7-10

AUXILIARY BUILDING MASS POINT PROPERTIES (SET A, SET B, SET C)

	EAST-WEST MOTION		NORTH-SOUTH MOTION		VERTICAL MOTION		
Elevation	Total Weight (kips)	Equip. & Added Soil Weight (kips)	Total Weight kips	Equip. & Added Soil Weight (kips)	Total Weight kips	Equip. & Added Soil Weight (kips)	WR^2 ($K\text{-}Ft^{2}$)
699.62	18209	2471	20320	4582	22791	7053	2.22×10^8
707.25	18036	1925	19681	3570	21605	5494	2.64×10^8
711.50	29461	4466	30450	5455	32496	7501	2.90×10^8
719.88	18620	3534	21718	6632	24431	9347	2.11×10^8
728.25	23473	2886	24559	3972	25510	4923	3.02×10^8
736.50	21840	895	21840	895	21840	895	2.52×10^8
745.75	13131	905	13131	905	13131	905	1.91×10^8
755.50	25921	273	25921	273	25921	273	3.85×10^8
771.50	20797	146	20797	146	20797	146	3.36×10^8
781.50	7311	0	7311	0	7311	0	0.97×10^8
785.50	7870	399	7870	399	7870	399	0.90×10^8
800.50	4676	41	4676	41	4676	41	0.45×10^8
814.25	5023	352	5023	352	5023	352	0.30×10^8

TABLE 3.7-11

AUXILIARY BUILDING NATURAL PERIODS (SET A)

	North-South Motion		East-West Motion		Vertical Motion	
Model No.	Frequency (cycles/sec.)	Mass Participation Factor	Frequency (cycles/sec.)	Mass Participation Factor	Frequency (cycles/sec.)	Mass Participation Factor
1	8.17	-2.157×10^3	6.05	1.324×10^3	23.25	-2.300×10^3
2	17.60	0.897×10^3	10.11	1.762×10^3		
3	24.84	0.747×10^3	16.00	0.645×10^3		
4			18.77	-0.696×10^3		

WBN

TABLE 3.7-12

NORTH STEAM VALVE ROOM ELEMENT PROPERTIES

$E_c = 720000\ k/Ft^2$ $G_c = 300000\ K/Ft^2$

Element No.	Length (Ft)	Area (Ft^2)	Torsion Constant (Ft^4)	North-South Direction Moment of Inertia (Ft^4)	North-South Direction Shear Factor	East-West Direction Moment of Inertia (Ft^4)	East-West Direction Shear Factor
1	9.375	830	4423	102117	1.23	289089	5.32
2	9.375	830	4423	102117	1.23	289089	5.32
3	9.375	830	4423	102117	1.23	289089	5.32
4	9.375	830	4423	102117	1.23	289089	5.32
5	7.000	1960	422400	235500	1.00	542700	1.00
6	8.580	317	1352	20587	1.61	139759	2.64
7	8.420	318	1263	19471	1.49	151177	3.03
8	7.000	373	1717	22339	1.39	159900	3.55
9	5.000	593	6160	33462	1.97	216100	2.03
10	5.000	593	6160	33462	1.97	216100	2.03
11	8.000	400	1861	28528	1.33	184401	4.05
12	5.250	230	422	17895	1.83	110963	2.21

WBN

TABLE 3.7-13

NORTH STEAM VALVE ROOM MASS POINT PROPERTIES

Mass Point No.	Total Weight (Kips)* N-S Direction	Total Weight (Kips)* E-W Direction
1	1976	2796
2	1976	2796
3	1976	2796
4	2723	3443
5	1149	1369
6	658	658
7	475	475
8	477	477
9	552	552
10	473	473
11	331	331
12	330	330

*Includes the weight of contained fill material for mass points 1-4.

WBN

TABLE 3.7-13A

LUMPED-MASS MODEL PROPERTIES
OF UNIT 1 NORTH STEAM VALVE ROOM (NSVR) - HORIZONTAL MODEL (SET B, Set C)

		Mass Moment of Inertia (10^3 k-ft-Sec^2)		Shear Center (ft)		Shear Areas (ft^2)		Moment of Inertias (10^3 x ft^4)		
Elevation (ft)	Masses (K-sec^2/ft)	J_K	J_Y	e_x	e_y	A_X	A_Y	J	I_{x-x}	I_{y-y}
777	15.39	0.60	3.75							
				8.72	4.14	106	257	43.8	21.5	102.3
763	28.4	1.69	10.87							
				8.68	3.69	319	295	60.8	38.2	177.6
753	30.75	1.52	10.43							
				6.18	3.83	110	259	54.9	18.7	132.4
738	27.82	1.16	8.54							
				1.49	-1.32	130	228	89.1	14.6	71.9
728	9.9	0.48	3.25							

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

WBN

TABLE 3.7-13B

LUMPED-MASS MODEL PROPERTIES OF
UNIT 1 NORTH STEAM VALVE ROOM (NSVR) - VERTICAL MODEL (SET B, SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)	Mass Moment of Inertia (10^3 $k\text{-}ft\text{-}sec^2$) J_z	Location of Centroid (ft)		Areas (ft^2) A
			d_x	d_y	
777	15.39	7.37			
			7.19	-8.68	363
763	28.4	13.34			
			1.20	-5.91	613
753	30.75	12.78			
			2.54	-8.75	369
738	27.82	10.72			
			-.28	-9.62	358
728	9.9	3.74			

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2

Poisson's Ratio = 0.15

+X = EAST

+Y = NORTH

WBN

TABLE 3.7-14

NORTH STEAM VALVE ROOM NAUTRAL FREQUENCIES

		North-South Motion		East-West Motion		Vertical Motion	
Case	Mode No.	Frequency (Hz)	Participation Factor	Frequency (Hz)	Participation Factor	Frequency (Hz)	Participation Factor
	1	2.63	-0.0465	2.62	-0.2666		
	2	6.76	0.2428	6.04	-0.9882		
	3	9.03	2.6640	9.59	2.4961		
	4	11.28	-0.6918	11.30	1.7919		
0.5 G	5	16.25	0.4464	16.12	0.6909		
	6	22.22	0.2433	20.04	-0.9738		
	7	24.33	1.5204	22.32	0.3151		
	8	25.88	-1.3297	25.43	-0.0209		
	1	2.64	-0.0476	2.62	-0.2543	34.37	1.7273
	2	6.79	0.2278	6.10	-0.9356	---	----
G	3	9.19	2.6917	9.96	2.7548	---	----
	4	11.29	-0.7362	11.36	2.2774	---	----
	5	16.26	0.4458	16.14	0.7316	---	----
	6	22.25	0.2179	20.25	-1.0336	---	----
	7	24.56	1.6430	22.33	0.3650	---	----
	8	25.98	-1.5517	25.43	-0.0218	---	----
	1	2.65	-0.0487	2.63	-0.2444		
	2	6.81	0.2154	6.14	-0.8933		
1.5 G	3	9.33	2.7216	10.28	-3.3180		
	4	11.29	-0.7817	11.43	2.8117		
	5	16.27	0.4452	16.17	0.7708		
	6	22.25	0.1981	20.47	-1.0920		
	7	24.76	1.6990	22.33	0.4278		
	8	26.12	-1.7493	25.43	-0.0229		

WBN

TABLE 3.7-15

PUMPING STATION ELEMENT PROPERTIES

E_C = 590,000 k/ft^2; G_C = 246,000 k/ft^2

			Motion about X-X		Motion about Y-Y	
Element No.	Length Ft	Area Ft^2	Moment of Inertia Ft^4	Shear Factor	Moment of Inertia Ft^4	Shear Factor
1	10.25	2338	1410×10^3	1.565	3061×10^3	2.769
2	8.00	2825	1833×10^3	1.865	3248×10^3	2.155
3	10.00	2825	1833×10^3	1.865	3248×10^3	2.155
4	10.00	2825	1833×10^3	1.865	3248×10^3	2.155
5	10.00	2825	1833×10^3	1.865	3248×10^3	2.155
6	9.50	2825	1833×10^3	1.865	3248×10^3	2.155
7	11.00	2747	1779×10^3	2.005	3389×10^3	1.995
8	5.75	2602	1774×10^3	2.124	3305×10^3	1.889
9	6.50	1932	497×10^3	1.807	2710×10^3	2.238
10	6.50	1932	497×10^3	1.807	2710×10^3	2.238

WBN

TABLE 3.7-15A

INTAKE PUMPING STATION BEAM ELEMENT PROPERTIES (SET B, SET C)

				Moment of Inertia		
Elev.	A_x	A_y	A_z	J_{xx}	I_{yy}	I_{zz}
	←	ft^2	→	←	ft^4	→
754.00						
	1217	541.6	675.4	$0.512x10^6$	1337353	294476
739.50						
	2167	847.3	1319.7	$1.13x10^6$	3158004	572896
733.00						
	2167	847.3	1319.7	$1.13x10^6$	3158004	572896
726.50						
	2772	1297.3	1474.7	$2.15x10^6$	3730181	1745654
720.75						
	3105	1639.4	1465.6	$2.15x10^6$	4070156	1763585
709.75						
	3148	1520.5	1627.5	$2.05x10^6$	3903497	1904018
700.25						
	3000	1302.0	1698.0	$2.05x10^6$	3674971	1896620
690.25						
	3000	1302.0	1698.0	$2.05x10^6$	3674971	1896620
680.25						
	2958	1260.1	1697.9	$2.05x10^6$	3601451	1889580
670.25						
	2945	1248.7	1696.3	$2.05x10^6$	3654683	1860185
662.25						
	2204	894.8	1309.2	$1.89x10^6$	2981336	1344534
652.00						

Notes: 1. x, y and z are local coordinate axes, i.e.,
x:vertical (global Z)
y:transverse (global Y)
z:longitudinal (global X)

2. A_x is the cross-sectional area, and A_y and A_z are the shear areas in the transverse and longitudinal directions respectively. J_{xx} is torsional moment of inertia, and I_{yy}, and I_{zz} are the bending moments of inertia about the transverse and longitudinal axes respectively.

3. E_c for beam elements: 590000 k/ft^2

WBN

TABLE 3.7-16

PUMPING STATION MASS POINT PROPERTIES

	1/2 SSE PROPERTIES		SSE PROPERTIES	
Mass Point No.	Total Wt Kips	Equip. & Water Wt Kips	Total Wt Kips	Equip. & Water Wt Kips
1	7871	4378	7871	4378
2	7804	3990	7804	3990
3	7448	3210	7448	3210
4	7448	3210	7454	3216
5	7262	3130	7467	3335
6	8637	4357	8306	4026
7	6856	3467	6384	2995
8	3217	1153	3217	1153
9	1884	0	1884	0
10	4593	3652	4593	3652

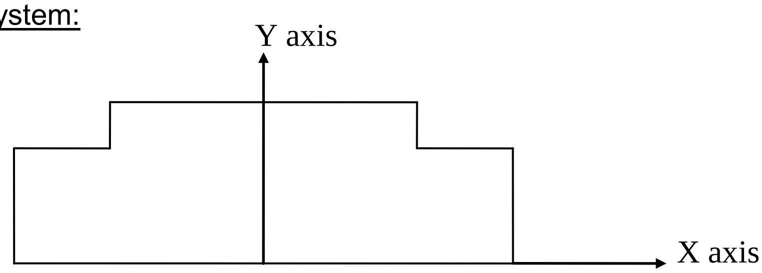
TABLE 3.7-16A

INTAKE PUMPING STATION NODAL WEIGHT PROPERTIES (SET B and SET C)

WEIGHTS

Elev.	WEIGHTS W_x	W_y	W_z	MOMENTS OF INERTIA W_{xx}	W_{yy}	W_{zz}
feet	<----------- kips		------------->	<------------ 10^6 kips-ft^2		------------>
754.00	1696	1696	1696	0.38878	1.8942	2.2830
739.50	5046	5046	5046	1.0626	6.5340	7.5965
733.00	2160	2160	2160	0.60628	3.1329	3.7392
726.50	3595	3595	3368	1.9182	3.7848	5.7031
720.75	6206	6206	5123	2.3578	7.1056	9.4634
709.75	7709	7709	5571	3.2726	6.5889	9.8615
700.25	7856	7856	4493	2.7792	5.5375	8.3167
690.25	7994	7994	4499	2.8449	5.5125	8.3574
680.25	7977	7977	4469	2.8397	5.4573	8.2970
670.25	7543	7543	5108	2.8676	6.2701	9.1378
662.25	7562	7562	4564	2.5089	5.8541	8.3630

Coordinate System:



X axis - Longitudinal direction
Y axis - Transverse direction
Z axis - Vertical direction

WBN

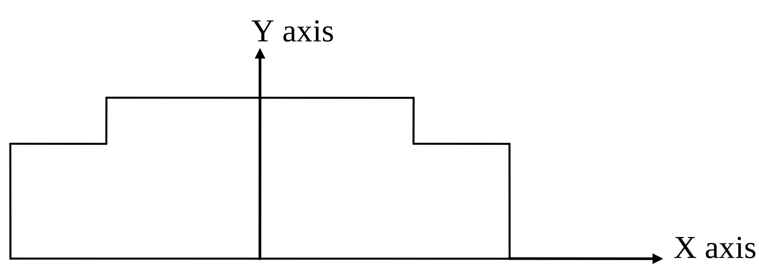
TABLE 3.7-16B

INTAKE PUMPING STATION NODAL COORDINATES (SET B AND SET C)
(Feet Units)

Elev.	CM Node		CR Node	
Z-Coord	X-Coord	Y-Coord	X-Coord	Y-Coord
754.00	0.38	23.15		
			0.00	24.02
739.50	0.06	23.31		
			0.00	25.41
733.00	0.74	21.55		
			0.00	25.41
726.50	0.03	36.61		
			0.00	42.40
720.75	-0.01	31.20		
			0.00	42.97
709.75	0.00	36.24		
			0.00	44.71
700.25	0.00	33.21		
			0.00	45.54
690.25	0.00	33.04		
			0.00	45.54
680.25	0.00	33.13		
			0.00	46.00
670.25	0.00	34.30		
			0.00	45.87
662.25	0.00	33.13		
			0.00	38.01
652.00			0.00	38.01
648.00	Fixed Base==>		0.00	43.00

Coordinate System:

CM - Center of Mass
CR - Center of Rigidity



X axis - Longitudinal direction
Y axis - Transverse direction
Z axis - Vertical direction

TABLE 3.7-17

PUMPING STATION NATURAL PERIODS

1/2 SAFE SHUTDOWN EARTHQUAKE

Mode No.	Motion About X-X Period (Sec)	Motion About X-X Participation Factor	Motion About Y-Y Period (Sec)	Motion About Y-Y Participation Factor	Vertical Motion Period (Sec)	Vertical Motion Participation Factor
1	0.1085	1.4529	0.1091	1.3786	0.0420	1.2995
2	0.0353	-0.6714	0.0374	-0.5608	0.01454	-0.4914
3	0.0254	0.3298	0.0222	0.2693	0.00939	0.2946

SAFE SHUTDOWN EARTHQUAKE

Mode No.	Motion About X-X Period (Sec)	Motion About X-X Participation Factor	Motion About Y-Y Period (Sec)	Motion About Y-Y Participation Factor	Vertical Motion Period (Sec)	Vertical Motion Participation Factor
1	0.1075	1.4579	0.1082	1.3830	0.0417	1.3035
2	0.0353	-0.6750	0.0374	-0.5652	0.0145	-0.4958
3	0.0205	0.3259	0.0222	0.2671	0.00938	0.2924

TABLE 3.7-18

DIESEL-GENERATOR BUILDING ELEMENT PROPERTIES

E_C = 590,000 K/FT^2 G_C = 236,000 K/FT^2

			North-South Motion		East-West Motion	
Element No.	Length, Ft.	Area, Ft^2	Moment of Inertia, Ft^4	Shear Factor	Moment of Inertia, Ft^4	Shear Factor
1	6.00	1060	119 x 10^4	1.77	235 x 10^4	2.30
2	6.00	1060	119 x 10^4	1.77	235 x 10^4	2.30
3	5.75	1162	137 x 10^4	1.91	252 x 10^4	3.10
4	3.75	1259	117 x 10^4	2.10	225 x 10^4	3.00
5	5.00	992	64 x 10^4	1.65	190 x 10^4	6.52
6	2.75	1259	117 x 10^4	2.10	225 x 10^4	3.00

WBN

TABLE 3.7-19

DIESEL-GENERATOR BUILDING MASS POINT PROPERTIES

Mass Point No.	Total Weight (Kips)	Equipment Weight (Kips)	Weight Moment of Inertia ($K\text{-}Ft^2$)	
			N-S Motion	E-W Motion
Base	14,800	650	178×10^5	273×10^5
1	960	-	107×10^4	212×10^4
2	980	-	113×10^4	215×10^4
3	3,250	205	223×10^4	472×10^4
4	920	-	57×10^4	135×10^4
5	800	-	48×10^4	118×10^4
6	2,250	-	100×10^4	223×10^4

TABLE 3.7-19A

LUMPED-MASS MODEL PROPERTIES
OF DIESEL GENERATOR BUILDING - HORIZONTAL MODEL (SET B and SET C)

Elevation (ft)	Masses (K-sec^2/ft)	Mass Moment of Inertias (10^4 k-ft-sec^2) J_z	Shear Center (ft) e_x	Shear Center (ft) e_y	Shear Areas (ft^2) A_x	Shear Areas (ft^2) A_y	Moment of Inertias (10^3 x ft^4) J	Moment of Inertias (10^3 x ft^4) I_{y-y}	Moment of Inertias (10^3 x ft^4) I_{y-y}
773.5	107.58	27.85							
			0	10.49	430.8	645.3	1295	868	1445
768.5	-	-							
			0	2.27	182.7	635.9	996	711.5	1003
763.5	-	-							
			0	10.08	434.7	642.1	1295	926.5	1333
759.75	188.89	49.40							
			0	4.51	562	589.6	1290	765.5	1653
754	-	-							
			0	-17.1	697.05	715.97	1707	1522	2013
748	-	-							
			0	-19.41	729.1	765.8	1765	1731	2098
742	43.73	15.67							
			0	0	Rigid Link				
Basemat	540.87	130.59*							

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2

Poisson's Ratio = 0.15

+X = EAST

+Y = NORTH

*Rocking Mass Moment of Inertia: I_{X-X} = 85.6 x 10^4 k-ft-S^2; I_{Y-Y} = 137.92 x 10^4 k-ft-S^2

WBN

TABLE 3.7-19B

LUMPED-MASS MODEL PROPERTIES OF
DIESEL GENERATOR BUILDING - VERTICAL MODEL (SET B and SET C)

Elevation (ft)	Masses (K-sec^2/ft)	Mass Moment of Inertia (10^4k-ft-sec^2) J_z	Location of Centroid (ft) d_x	d_y	Areas (ft^2) A
773.5	107.58	27.85			
			0	10.14	1086.03
768.5	-	-			
			0	7.90	818.6
763.5	-	-			
			0	9.48	1076.8
759.75	188.89	49.40			
			0	7.91	1151.6
754	-	-			
			0	-3.54	1413.02
748	-	-			
			0	-6.27	1494.9
742	43.73	15.67			
Base	540.87	130.59*			

Concrete Properties

Modulus of Elasticity = 576,000 k/ft^2
Poisson's Ratio = 0.15

+X = EAST
+Y = NORTH

*Rocking Mass Moment of Inertia:

$I_{X\text{-}X}$ = 85.6 x 10^4 k-ft-S^2; $I_{Y\text{-}Y}$ = 137.92 x 10^4 k-ft-S^2

TABLE 3.7-19C

LUMPED-MASS MODEL PROPERTIES
OF REFUELING WATER STORAGE TANK - SEISMIC MODEL (SET B and SET C)

Elevation (ft)	Masses ($k\text{-}sec^2/ft$)		Mass Moment of Inertias (10^4 $k\text{-}ft\text{-}sec^2$)		Axial Area A (ft)	Shear Areas (ft^2)		Moment of Inertia (ft^4)		
	M_X=M_Y	M_Z	I_X=I_Y	J		A_x	A_y	J	I_{x-x}	I_{y-y}
767.20	0.96	0.96	-	-						
					3.551	1.884	1.884	1678	839	839
763.20	3.94	0.31	-	-						
					3.551	1.884	1.884	1678	839	839
759.66	7.86	.34	-	-						
					4.233	2.246	2.246	2000	1000	1000
755.85	8.17	0.37	-	-						
					4.233	2.246	2.246	2000	1000	1000
752.04	8.17 28.32*	0.37	-	-						
					4.643	2.463	2.463	2192	1096	1096
748.23	8.17	0.37	-							
					4.643	2.463	2.463	2192	1096	1096
744.42	8.23	0.43	-	-						
					6.007	3.187	3.187	2836	1418	1418
740.61	8.26	0.47	-	-						
					6.007	3.187	3.187	2836	1418	1418
736.80	8.29	0.50	-	-						
					7.507	3.982	3.982	3542	1771	1771
733.0	8.32	0.56	-	-						
					7.507	3.982	3.982	3542	1771	1771
729.20	4.16	28.98 69.22**	1.24	2.48						

Base

Young's Modulus E = 30,000 ksi

Shear Modulus G = 11,540 ksi

*Sloshing-induced horizontal mass, $M_X = M_Y = 28.32$ $k\text{-}sec^2/ft$

associated horizontal spring, $K_X = K_Y = 76.2$ k/ft

**Seismic-induced vertical effective mass, $M_Z = 69.22$ $k\text{-}sec^2/ft$

associated vertical spring $K_Z = 246120$ k/ft

WBN

TABLE 3.7-20

DIESEL-GENERATOR BUILDING NATURAL PERIODS

V_S = 1150 FPS

Mode No.	N-S Motion $K_T = 147 \times 10^4$ K/Ft $K_R = 300 \times 10^7$ K-Ft RAD Period, Second	E-W Motion $K_T = 141 \times 10^4$ K/Ft $K_R = 425 \times 10^7$ K-Ft RAD Period, Second
1	0.154	0.156
2	0.103	0.111
3	0.029	0.035

V_S = 1650 FPS

Mode No.	N-S Motion $K_T = 308 \times 10^4$ K/Ft $K_R = 614 \times 10^7$ K-Ft RAD Period, Second	E-W Motion $K_T = 294 \times 10^4$ K/Ft $K_R = 887 \times 10^7$ K-Ft RAD Period, Second
1	0.108	0.110
2	0.072	0.077
3	0.028	0.034

V_S = 2150 FPS

Mode No.	N-S Motion $K_T = 517 \times 10^4$ K/Ft $K_R = 1031 \times 10^7$ K-Ft RAD Period, Second	E-W Motion $K_T = 493 \times 10^4$ K/Ft $K_R = 1490 \times 10^7$ K-Ft RAD Period, Second
1	0.085	0.087
2	0.056	0.059
3	0.028	0.033

WBN

TABLE 3.7-21

WASTE-PACKAGING AREA ELEMENT PROPERTIES

E_C = 590,000 K/FT^2 G_C = 236,000 K/FT^2

Element No.	Length (Ft)	Area (Ft^2)	North-South Motion: Moment of Inertia (Ft^4)	North-South Motion: Shear Factor	East-West Motion: Moment of Inertia, Ft^4	East-West Motion: Shear Factor
1	9.00	573.2	184100	2.36	556500	1.63
2	6.50	573.2	184100	2.36	556500	1.63
3	6.50	573.2	184100	2.36	556500	1.63
4	5.75	573.2	184100	2.36	556500	1.63
5	5.75	319.2	78630	1.59	393500	2.56
6	5.75	237.7	72700	1.95	267000	1.91
7	5.75	156.4	66728	3.89	136000	1.25

TABLE 3.7-22

WASTE-PACKAGING AREA MASS POINT PROPERTIES

Mass Point Mo.	Total Weight Kips	Weight Moment of Inertia, K-ft^2	
		N-S Motion	E-W Motion
Base	3108	6.04×10^5	2.08×10^6
1	971	1.86×10^5	5.63×10^5
2	629	1.79×10^5	5.43×10^5
3	512	1.69×10^5	5.11×10^5
4	418	1.13×10^5	4.10×10^5
5	597	0.65×10^5	2.85×10^5
6	535	0.60×10^5	1.74×10^5
7	444	0.29×10^5	0.59×10^5

WBN

TABLE 3.7-23

WASTE-PACKAGING AREA NATURAL PERIODS

	1/2 SSE	
	N-S Motion	E-W Motion
	$K_T = 8.54 \times 10^5$ K/Ft	$K_T = 8.30 \times 10^5$ K/Ft
Mode No.	$K_R = 4.24 \times 10^8$ K-Ft/Rad Period, Second	$K_R = 1.65 \times 10^9$ K-Ft/Rad Period, Second
1	0.143	0.116
2	0.072	0.065

	SSE	
	N-S Motion	E-W Motion
	$K_T = 5.34 \times 10^5$ K/Ft	$K_T = 8.30 \times 10^5$ K/Ft
Mode No.	$K_R = 6.02 \times 10^7$ K-Ft/Rad Period, Second	$K_R = 1.65 \times 10^9$ K-Ft/Rad Period, Second
1	0.313	0.116
2	0.108	0.065

Table 3.7-23A

CDWE BUILDING

SOIL DEPOSIT SHEAR MODULI AND SHEAR WAVE VELOCITIES

Soil Profile	Shear Modulus (ksf)	Shear Wave Velocity (f/s)
Average	2409	761
-50% Variation	1205	538
+50% Variation	3613	932

SPRING CONSTANTS FOR PILE GROUP

Direction	Spring Constant
N-S Translation	5.61×10^5 k/f
E-W Translation	5.31×10^5 k/f
Rocking About E-W Axis	1.61×10^9 k-f/rad
Rocking About N-S Axis	2.57×10^9 k-f/rad
Vertical	3.41×10^6 k/f

MASS POINT PROPERTIES - LUMPED MASS MODEL

Mass Point	Weight (kips)	Center of Gravity (f)		Mass Moment About CG (k-f)		Mass Moment About Geometric Center (k-f)		
		X	Y	X	Y	X	Y	Z
Base	2669.5	-1.054	-0.829	386095	659047	370909	621259	955811
1	742.3	0.0	0.0	220746	331660	220746	331660	530274
2	2031.9	-1.850	0.050	357010	612045	357455	611943	932527
3	742.3	0.0	0.0	220746	331660	220746	331660	530274
4	1120.3	0.600	0.0	227799	346548	227126	346548	570708

ELEMENT PROPERTIES - LUMPED MASS MODEL

Element	Length (f)	Area (f^2)	Moment of Inertia (f^4)		Torsion Constant (f^4)	Shape Factor	
			X	Y		X	Y
1	14.75	370.0	104513	159798	5718000	1.69	2.22
2	14.0	370.0	104513	159798	5718000	1.69	2.22
3	13.0	370.0	104513	159798	5718000	1.69	2.22
4	14.0	370.0	104513	159798	5718000	1.69	2.22

E = 720000 ksf
E/G = 2.50

TABLE 3.7-24

DAMPING RATIOS FOR FLUID SYSTEM PIPING AND THEIR SUPPORTS
ANALYZED BY NSSS VENDOR (SET A)

Item	Damping Ratio Percentage of Critical Viscous Damping	
	OBE	SSE
Reactor Coolant Loop	1	1
Auxiliary Piping Systems(2)	0.5	1
Welded Steel Structures	1	1-2(1)
Bolted Steel Structures	2	2-5(1)

Notes:

(1) Damping value used when stress levels are at or near yield.

(2) An as option, for some cases or piping response spectrum seismic analysis, variable damping of 5% to 10 hertz decreasing linearly to 2% at 20 hertz and remaining at 2% to 33 hertz was used for both OBE and SSE as described in ASME Code Case N-411.

TABLE 3.7-25 (Sheet 1 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Reactor Coolant System						
Reactor Vessel	X				See Section 5.2	
Full-length CRDM housing		X			"	
Part-length CRDM housing		X			"	
Reactor coolant pump		X			"	
Steam generator		X			"	
Pressurizer		X			"	
Reactor coolant piping to pressure boundary		X			"	
RC system supports		X			"	
Surge pipe and fittings		X			"	
Bypass manifold	X					
RC Thermowells				X	See Section 5.2	
Safety valves	X				"	
Relief valves	X				"	
Valves to RC system boundary	X				"	

TABLE 3.7-25 (Sheet 2 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
CRDM head adapter plugs		X			See Section 5.2	
Chemical and Volume Control System						
Regenerative HX		X			See Section 3.9	
Letdown HX		X			"	
Mixed-bed demineralizer		X			"	
Cation bed demineralizer		X			"	
Reactor coolant filter		X			"	
Volume control tank		X			"	
Charging/high head safety injection pump	X			X	See Section 3.9	Tests were run to determine natural frequency of the foundation system to meet seismic criteria.
Seal water injection filter		X			"	
Excess letdown HX		X			"	
Seal water return filter		X			"	
Seal water HX		X			"	
Boric acid tanks		X			"	
Boric acid filter		X			"	
Boric acid transfer pump	X				"	
Boric acid blender		X			"	

TABLE 3.7-25 (Sheet 3 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Chemical and Volume Control System						
Reactor makeup water storage tank		X			w/o exceeding 90% of yield stresses and/or loss of function	
Emergency Core Cooling System						
Accumulators		X			"	
Boron injection tank		X			"	
BIT recirculation pump	X				"	
Boron injection surge Tank		X			See Section 3.9	
Residual Heat Removal System						
Residual heat removal/low head safety injection pump	X				"	
Residual heat exchanger		X			"	
Containment Spray System						
Spray additive tank		X			"	
Containment spray pump	X				"	
Containment Isolation System						
Valves	X				See Section 3.9	

TABLE 3.7-25 (Sheet 4 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Containment Cooling System						
Fans		X			See Section 3.9	
Heat Exchanger		X			"	
Component Cooling System						
Pumps	X				"	
Heat exchangers		X			"	
Surge Tank		X			"	
Spent Fuel Pool Cooling System						
Spent fuel pool heat exchanger		X			"	
Spent fuel pool pump	X				"	
Boron Thermal Regeneration Subsystem						
Moderating HX		X			See Section 3.9	
Letdown chiller HX		X			"	
Letdown reheat HX		X			"	
Thermal regeneration demineralizer		X			"	

TABLE 3.7-25 (Sheet 5 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Liquid Recycle and Waste Subsystem						
Recycle holdup tank		X			See Section 3.9	Per API 650
Recycle evaporator feed pump	X				"	
Recycle evaporator feed demineralizer		X			"	
Recycle evaporator feed filter	X				"	
Recycle evaporator	X				"	
R.C. drain tank HX		X			"	
Waste holdup tank		X			"	
Waste evaporator feed filter	X				"	
Waste evaporator	X				"	
Spent resin storage tank		X			"	
Spent resin sluice pump	X				"	
Spent resin sluice filter	X				"	
Floor drain tank		X			"	
ES room sump pump	X				"	

TABLE 3.7-25 (Sheet 6 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Gas Handling Subsystem						
Gas compressor	X			X	See Section 3.9	Vibration tests were conducted to determine seismic capability
Gas decay tanks		X			"	
Emergency Diesel Fuel Oil System						
Transfer pumps		X			"	
Fuel oil tanks	X				"	
Service Water System						
Pumps		X			"	
Fuel Handling System						
Fuel manipulator crane		X			See Section 3.9	
Fuel transfer tube	X				"	
Underwater fuel conveyor car and rail system	X				"	
Fuel pool bridge crane		X			"	

TABLE 3.7-25 (Sheet 7 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Fuel Handling System						
Polar crane	X				See Section 3.9	
Crane supports	X				"	
Refueling Water System						
Storage tank		X			w/o exceeding 90% of yield stresses and/or loss of function	
Auxiliary Building Ventilation System						
ES air cooling units: Heat exchanger		X			w/o exceeding 90% of yield stresses and/or loss of function	
Fan		X			"	
Penetration Room Filtration System						
Fans		X			"	
Filters (HEPA and charcoal)		X			"	
Control Room Ventilation System						
Fans		X			"	
Filters		X			"	

TABLE 3.7-25 (Sheet 8 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Control Room Ventilation System						
Air Conditioning unit		X			See Section 7.1	
Heat exchanger		X			"	
Diesel Building Ventilation System						
Fans				X	w/o exceeding 90% of yield stresses and/or loss of function	
Filters				X	"	
Main Steam System						
Isolation valves	X				"	
Auxiliary Feedwater System						
Auxiliary feedwater pumps motordriven, steam turbine driven		X			"	
Condensate storage tank		X			"	
Steam Dump Systems						
Relief Valves	X				"	
Safety Valves	X				"	

TABLE 3.7-25 (Sheet 9 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Electrical Components and Systems						
6.9 kV shutdown boards (engineered safe-guard buses)				X	w/o exceeding 90% of yield stresses and/or loss of function	
6.9 kV to 480 V transformers (associated with engineered safeguard systems)		X			"	
480 V shutdown boards (engineered safeguard systems buses)				X	"	Test on prototype
480-V motor-control centers (associated with engineered safeguard systems)				X	"	Test on prototype
125-Vdc vital batteries				X	"	Test on prototype
120-Vac vital inverters, (associated with vital instrument buses)				X	See Section 7.1	
125-Vdc battery boards				X	See Section 7.1	Tests on two panels selected at random
120-Vac vital instrument power boards				X	"	

TABLE 3.7-25 (Sheet 10 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Electrical Components and Systems						
125-v dc switchgear				X	See Section 7.1	Tests on prototype
125-v dc battery chargers				X	"	Test on one charger
Solid-state protection system cabinets				X	"	
Reactor trip switchgear				X	"	
Nuclear instrumentation system cabinets				X	"	
Process protection and control system cabinets				X	"	
Cable tray supports (associated with engineered safeguard system)		X			"	
Auxiliary relay racks				X	"	
Containment penetration assemblies				X	w/o exceeding 90% of yield stresses and/or loss of function	Test on one medium voltage penetration assembly plus test on a composite assembly comprised of 1 1000-v dc power and 600-v control and instrument cables

TABLE 3.7-25 (Sheet 11 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Electrical Components and Systems						
Emergency power board		X		X	w/o exceeding 90% of yield stresses and/or loss of function	Instruments and switches are tested
Direct-current emergency lighting				X	"	Test on prototype
Diesel generators		X			"	
Diesel generator control panels				X	"	
Diesel generator sequencers				X	"	Test on one panel
Boric acid heat-tracing equipment		X			"	
Balance of plant instrument cabinets and equipment contained therein				X	w/o loss of function	
Equipment contained within balance of plant instrument cabinets		X		X	w/o loss of function	
Containment purge radiation monitors		X		X	w/o loss of function	
Fuel handling area radiation monitors		X		X	w/o loss of function	

TABLE 3.7-25 (Sheet 12 of 12)

METHODS USED FOR SEISMIC ANALYSES
OF CATEGORY I SYSTEMS AND COMPONENTS

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Sampling System						
1. Cabinet		X			w/o exceeding 90% of yield stresses and w/o loss of function	
2. Tubing, valves, coolers, sample vessels	X				w/o loss of function	
Electrical Components and Systems						
Balance of plant field mounted instruments	X			X	w/o loss of function	
Instrument valves for field mounted instruments	X				w/o loss of function	
Instrument lines for field mounted instruments	X				w/o exceeding code allowable stresses	

TABLE 3.7-26

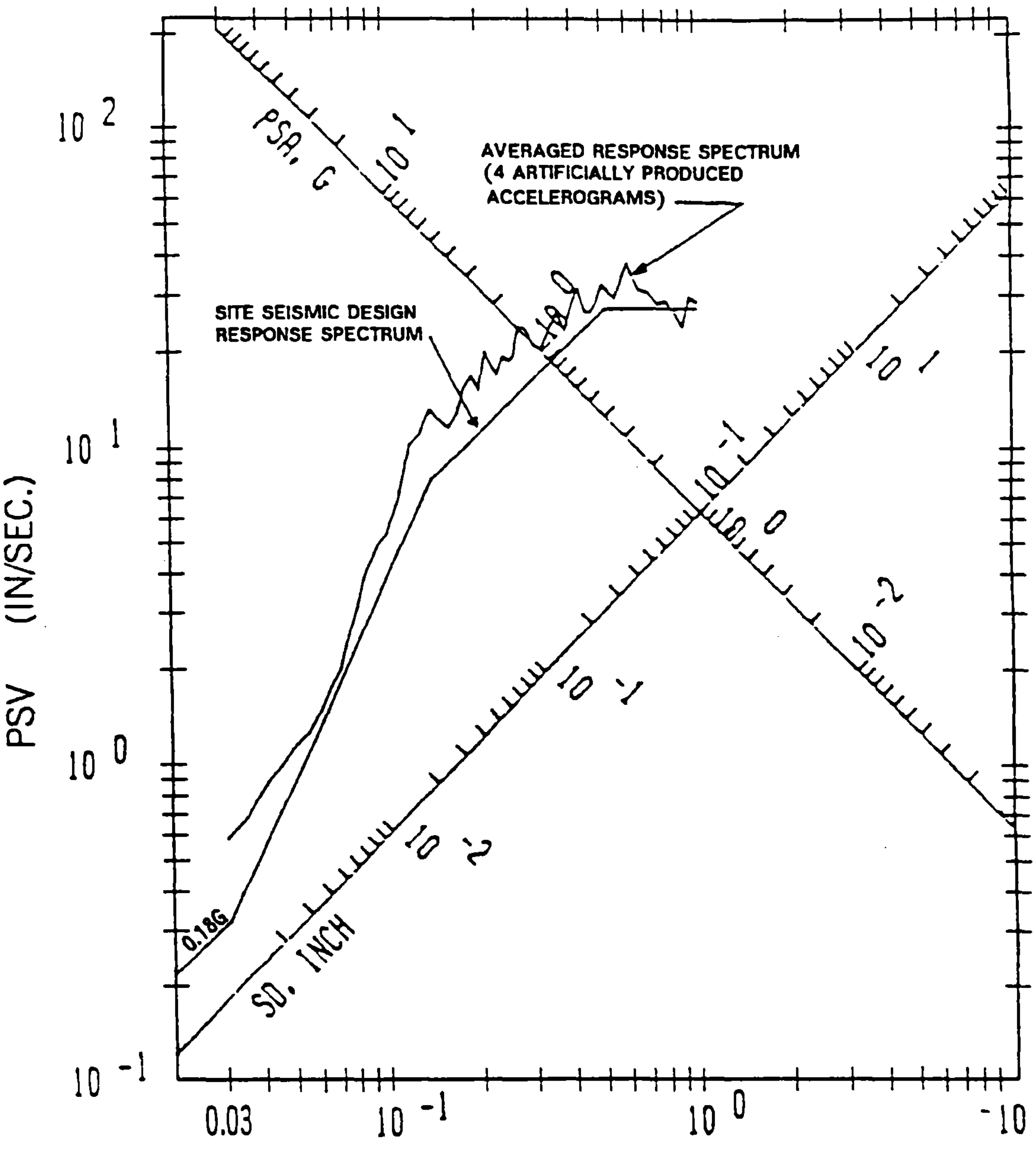
ALLOWABLE STRESSES FOR DUCT SUPPORTS

Elements	Load Combination[1]	Allowables
Component Standard Supports	(1) and (2)	Factor of Safety of 5 against ultimate strength
	(3), (4), and (5)	0.90 F_y (F_y = minimum specified yield stress) or a minimum factor of safety of 2.5 against ultimate strength
Steel Structural Members and Connecting Welds (Linear Supports)	(1)	AISC allowables
	(2)	1.5 x AISC allowables but less than 0.90 F_y[2]
	(3) and (4)	1.6 x AISC allowables but less than 0.90 F_y[2]
	(5)	1.7 x AISC allowables but less than 0.90 F_y[2]
Anchorage in Hardened Concrete		
Expansion Anchors	(1), (2), (3), (4), and (5)	Factor of Safety on minimum anchor ultimate tensile capacity
(a) Shell Types (SSD & SDI)		5
(b) Other Types (Wedge)		4

NOTES:

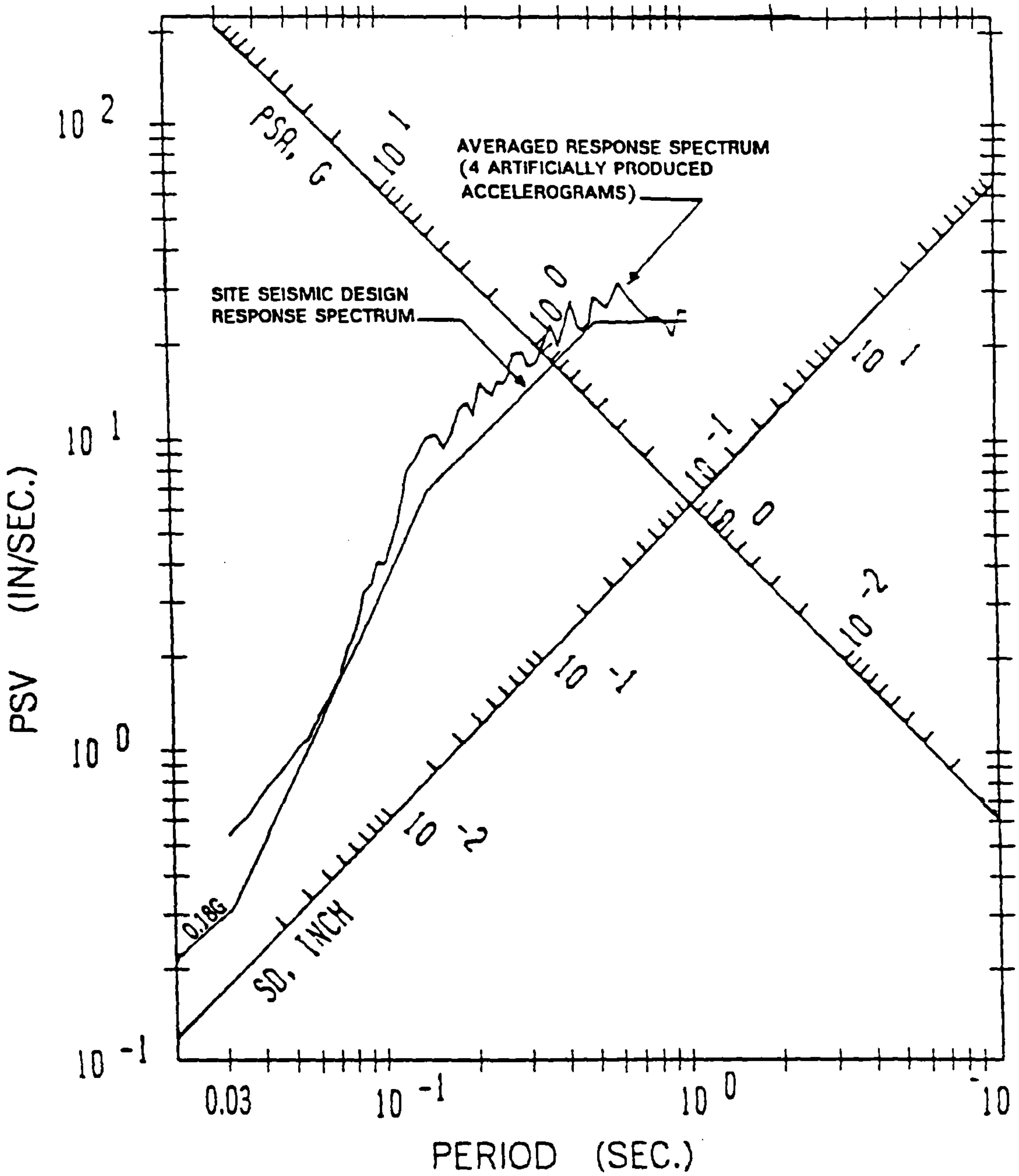
(1) In applying the above load combinations for design, dead load and thermal effects may be combined directly, accounting for their signs. Seismic loads are reversing and their effects must be combined without sign with the other loads. The latter is also true for DBA loads. (See Section 3.7.3.17.3 for definition of loads and their combinations.)

(2) But less than 0.52 F_y for shear stresses, and less than 0.90 F_{cr} for critical buckling stresses.



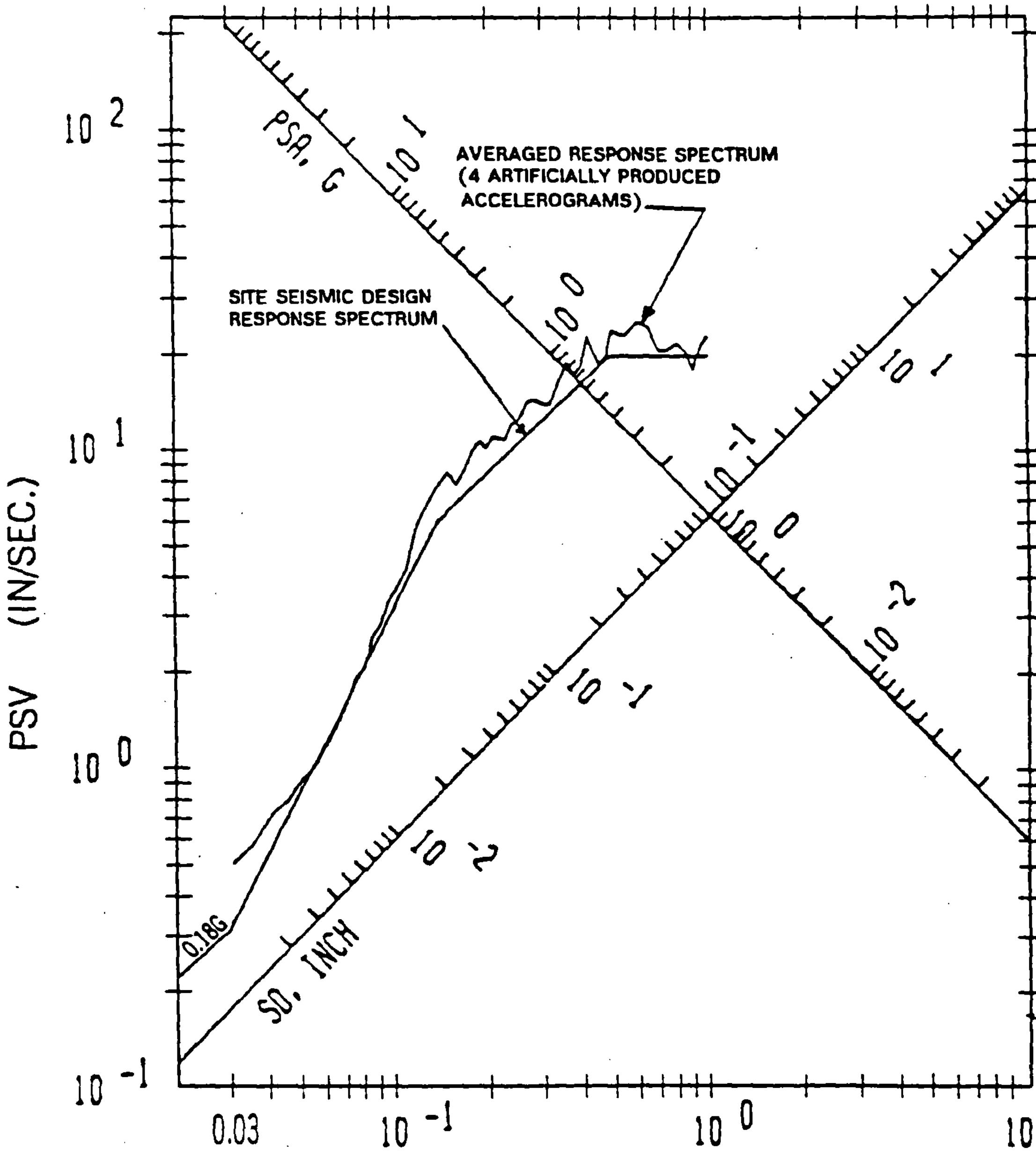
SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
1/2% DAMPING

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Response Spectra FIGURE 3.7-1



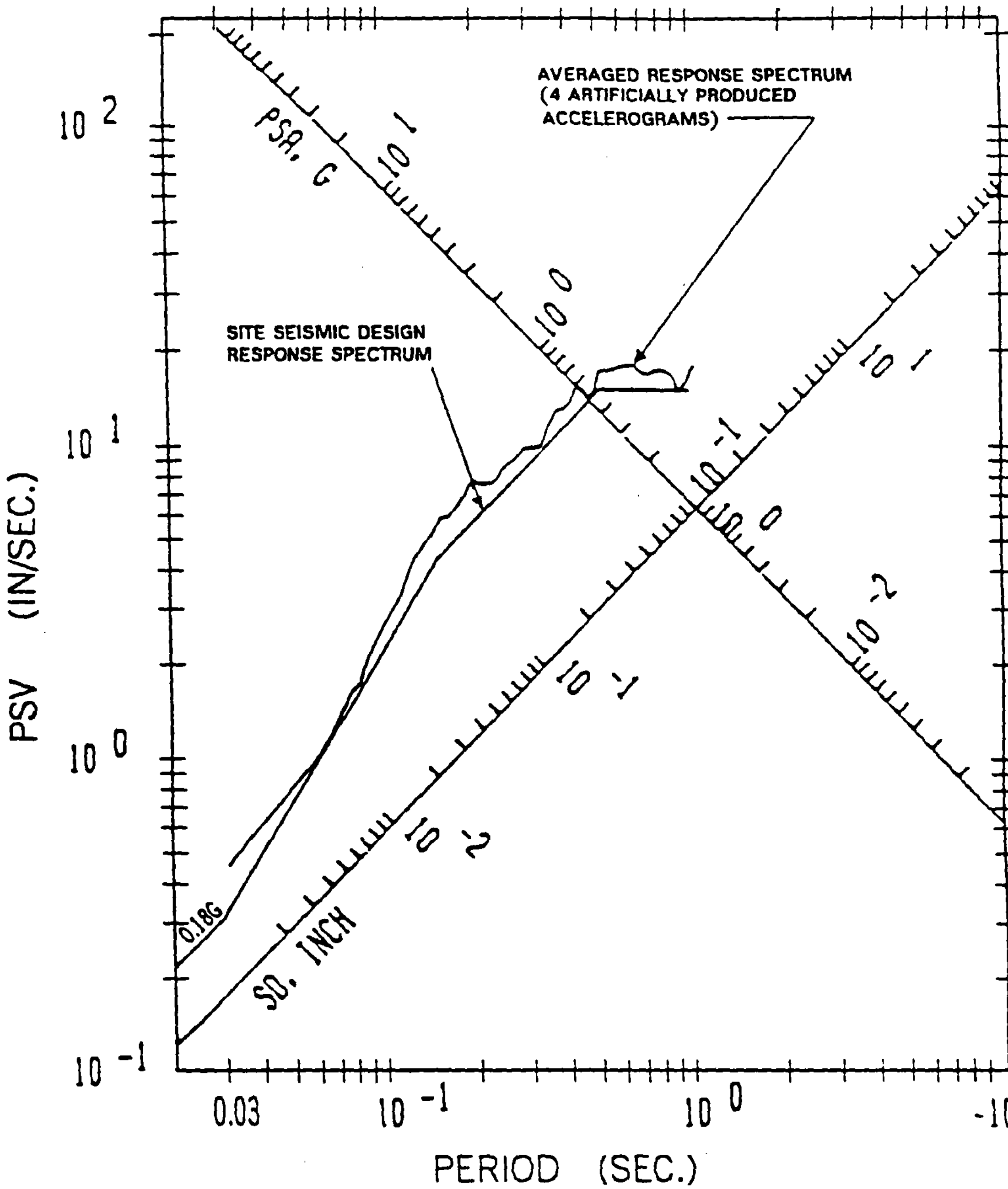
SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
1% DAMPING

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Response Spectra FIGURE 3.7-2



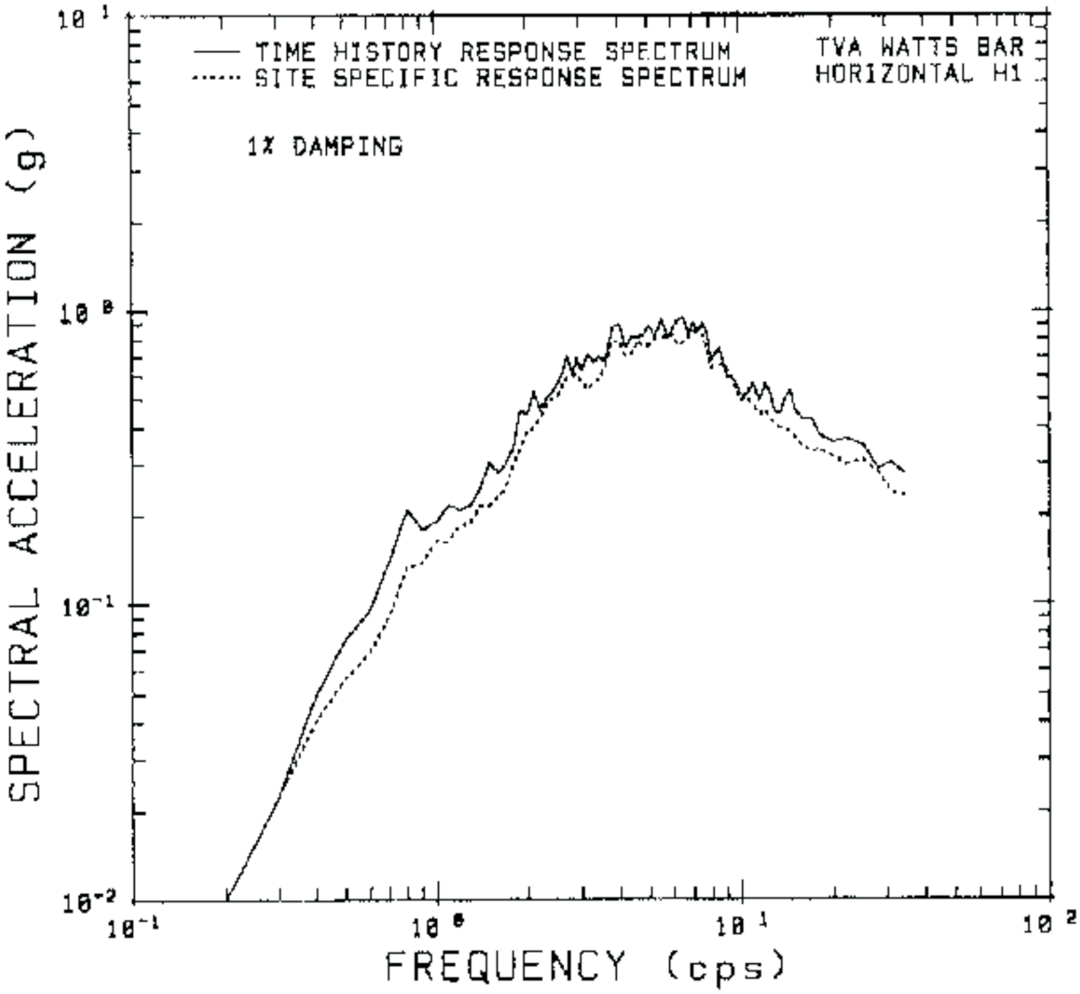
SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
2 % DAMPING

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Response Spectra FIGURE 3.7-3



SET A AND SET C SITE DESIGN RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
ROCK-SUPPORTED STRUCTURES
5% DAMPING

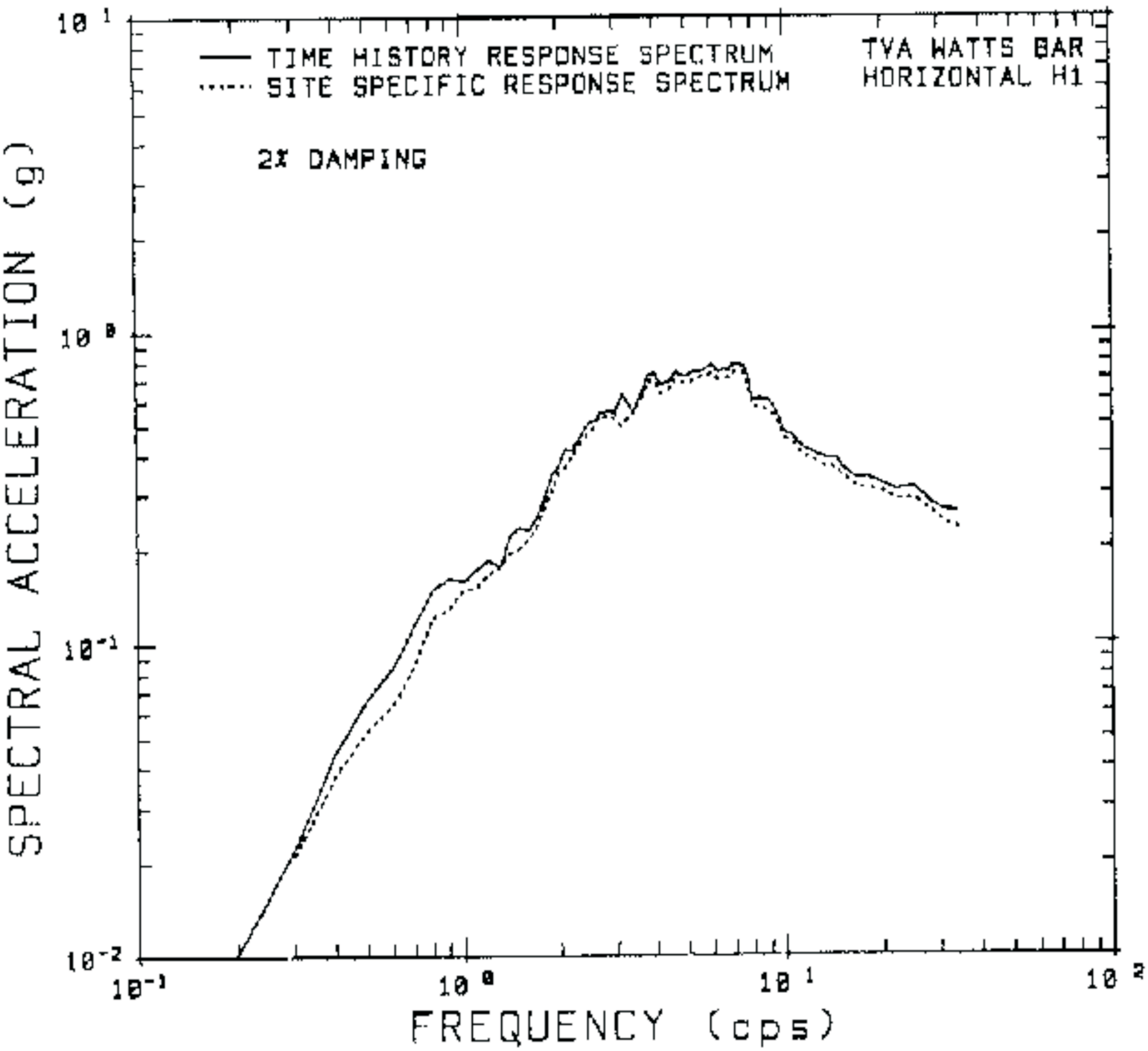
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Response Spectra FIGURE 3.7-4



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
1% DAMPING

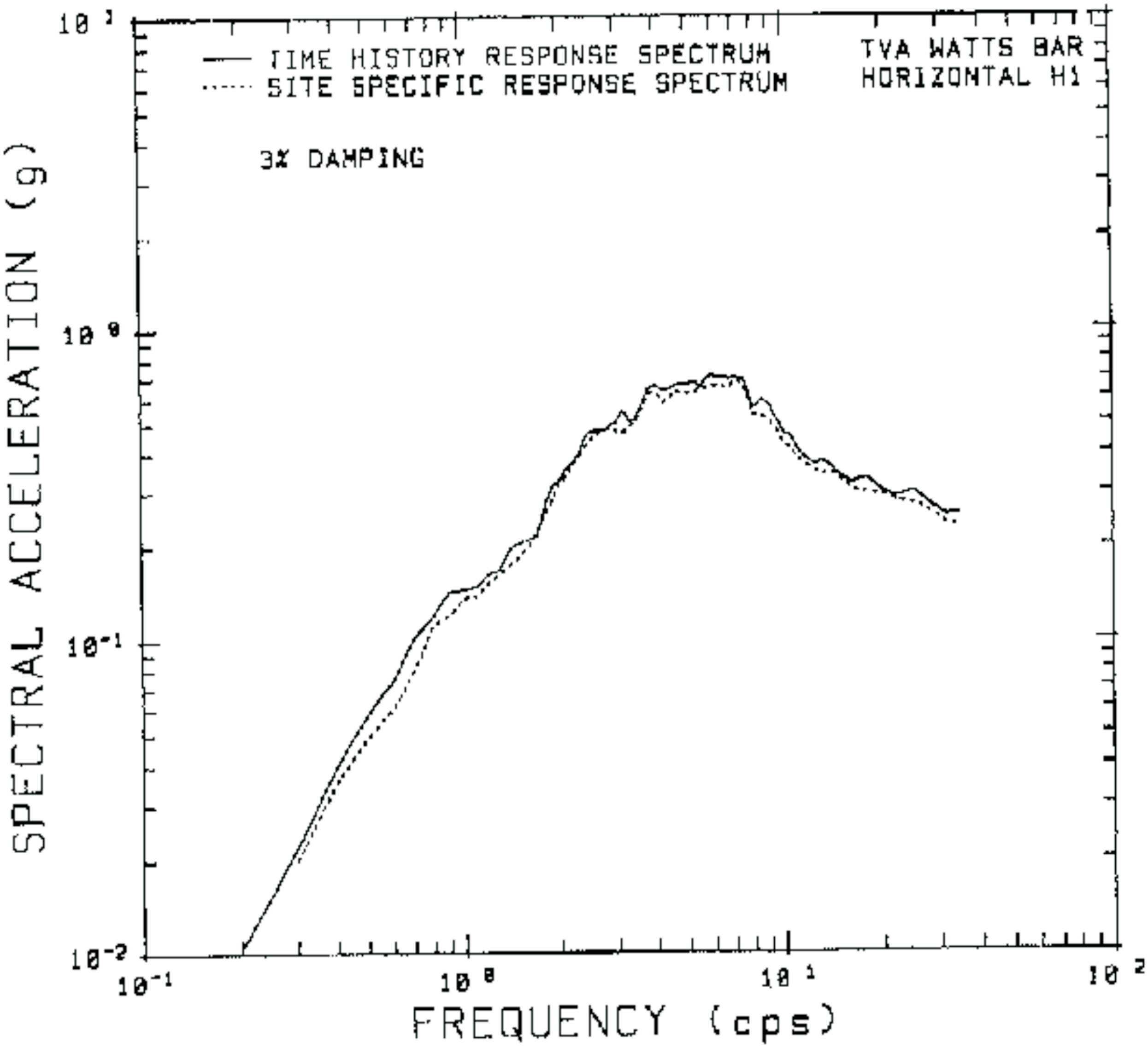
FIGURE 3.7-4a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
2% DAMPING

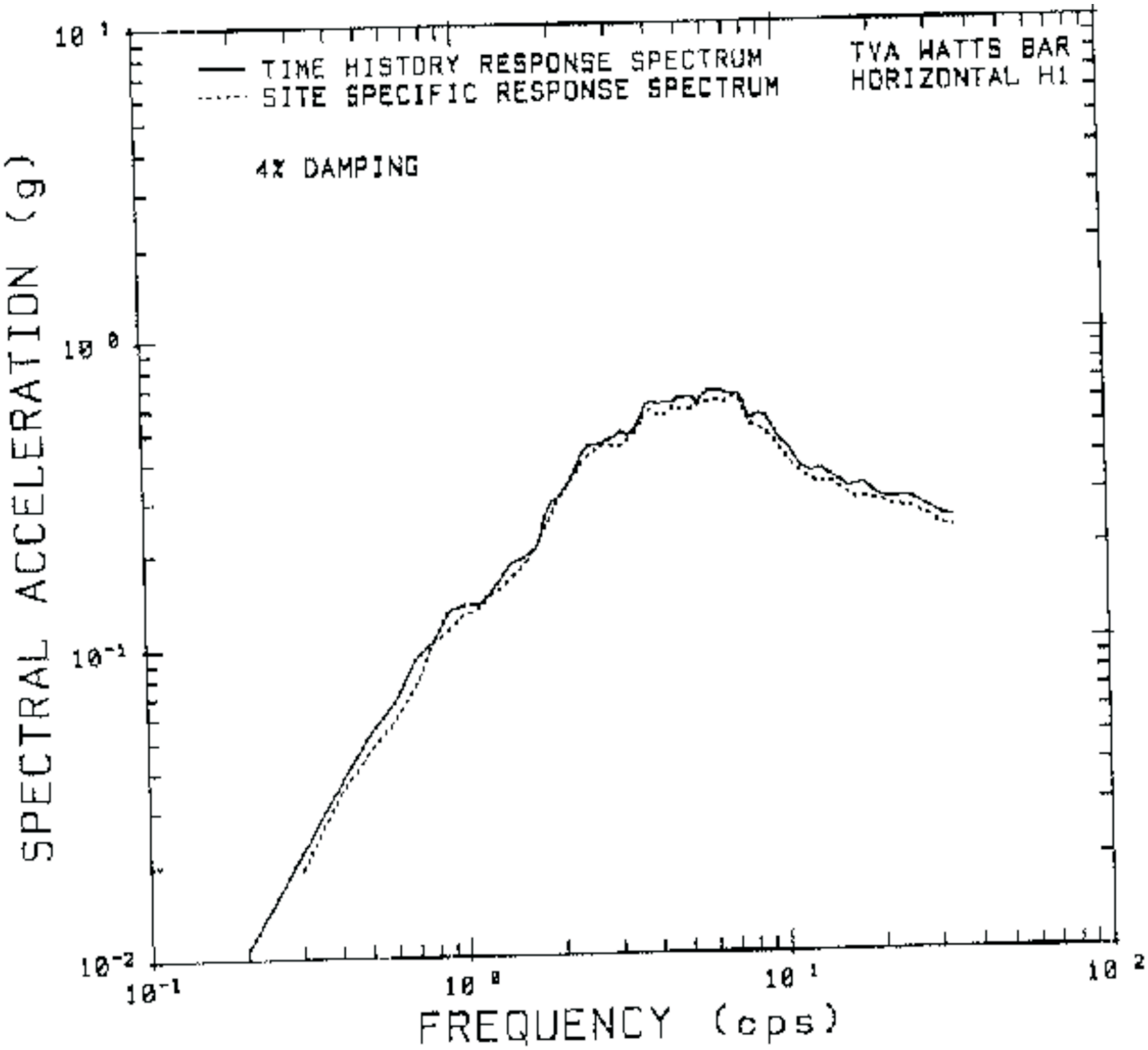
FIGURE 3.7-4b



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
3% DAMPING

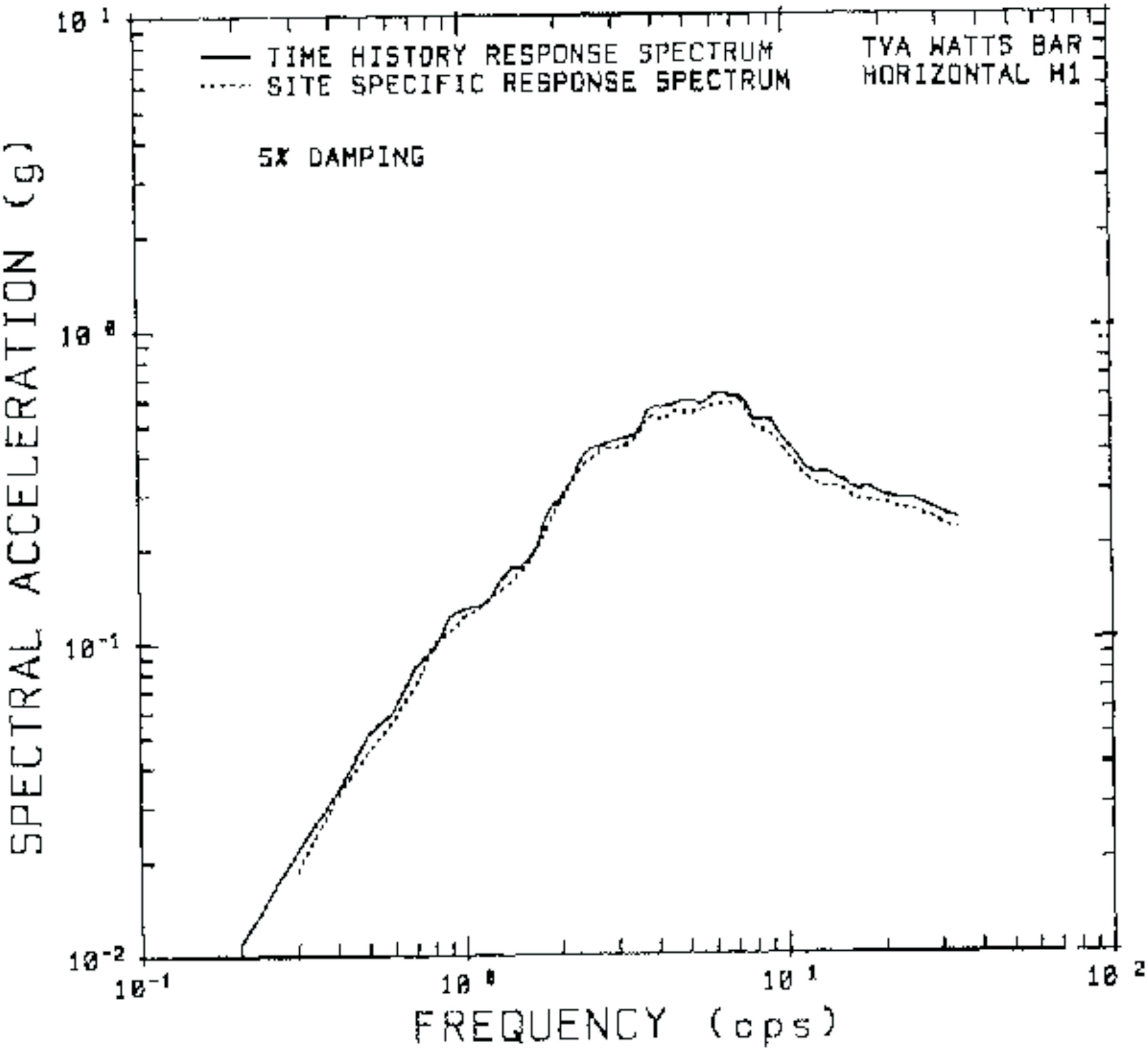
FIGURE 3.7-4c



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM SAFE SHUTDOWN EARTHQUAKE (N-S) ROCK SUPPORTED STRUCTURES 4% DAMPING

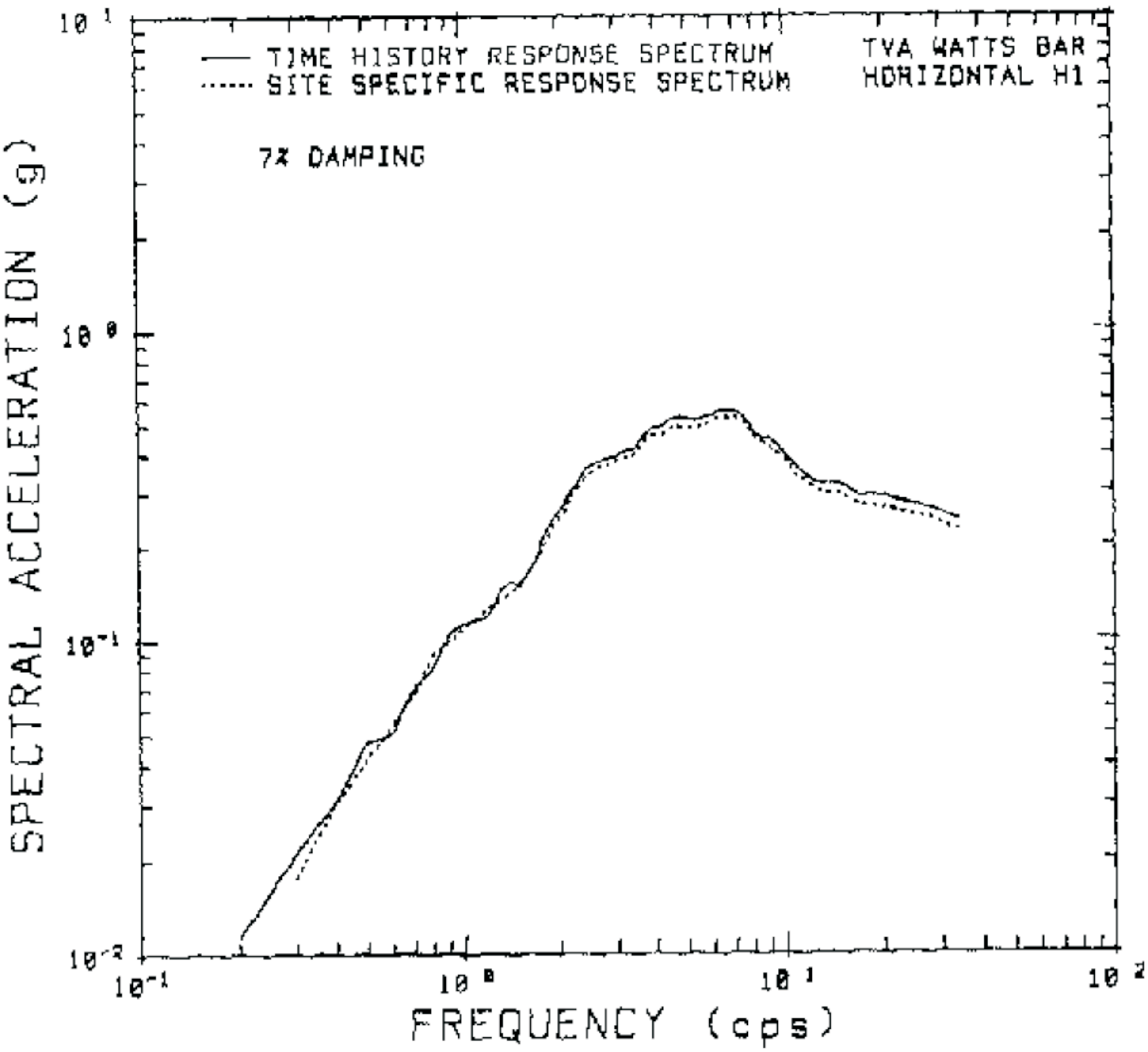
FIGURE 3.7-4d



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
5% DAMPING

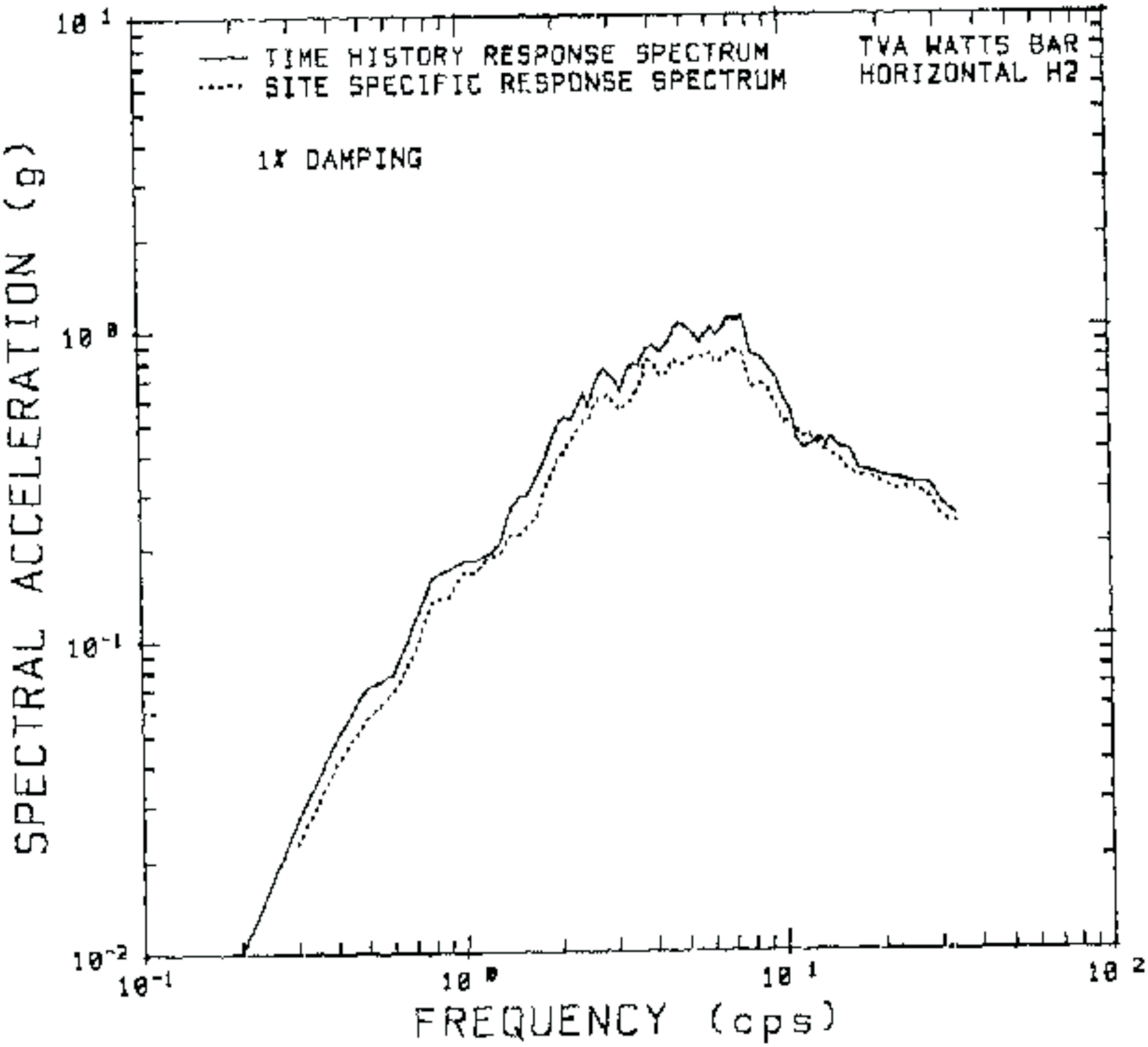
FIGURE 3.7-4e



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (N-S)
ROCK SUPPORTED STRUCTURES
7% DAMPING

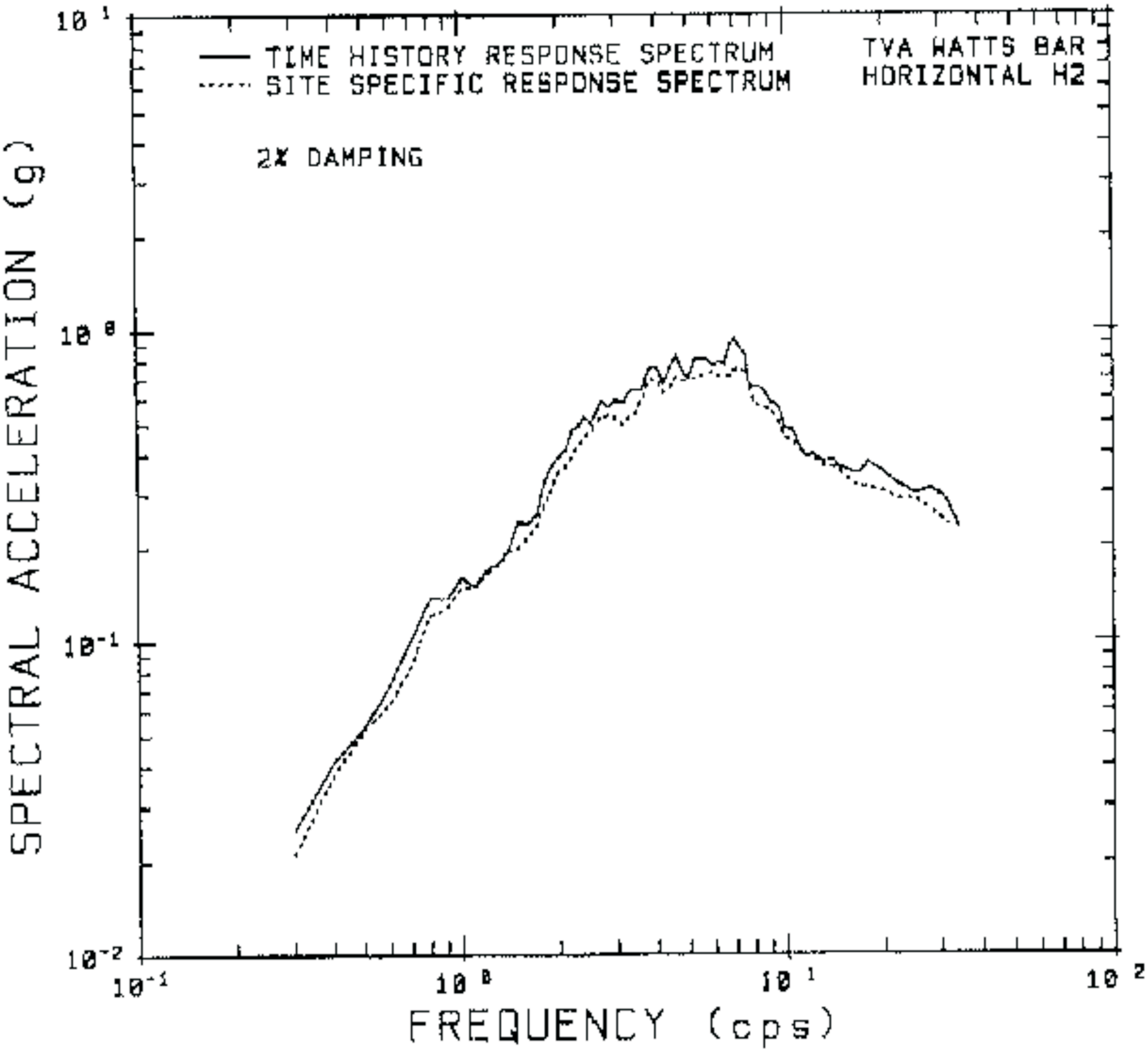
FIGURE 3.7-4f



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
1% DAMPING

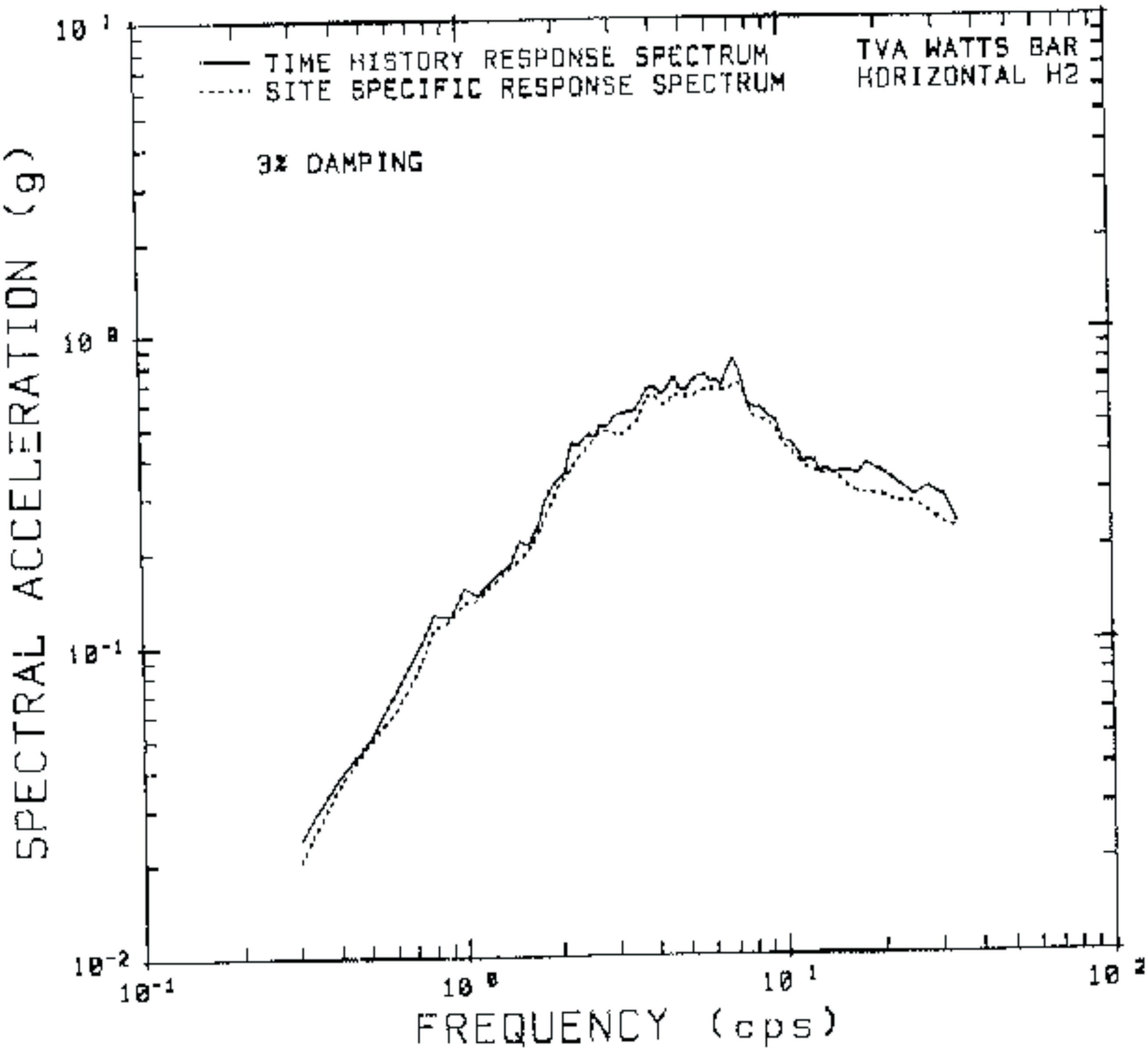
FIGURE 3.7-4g



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
2% DAMPING

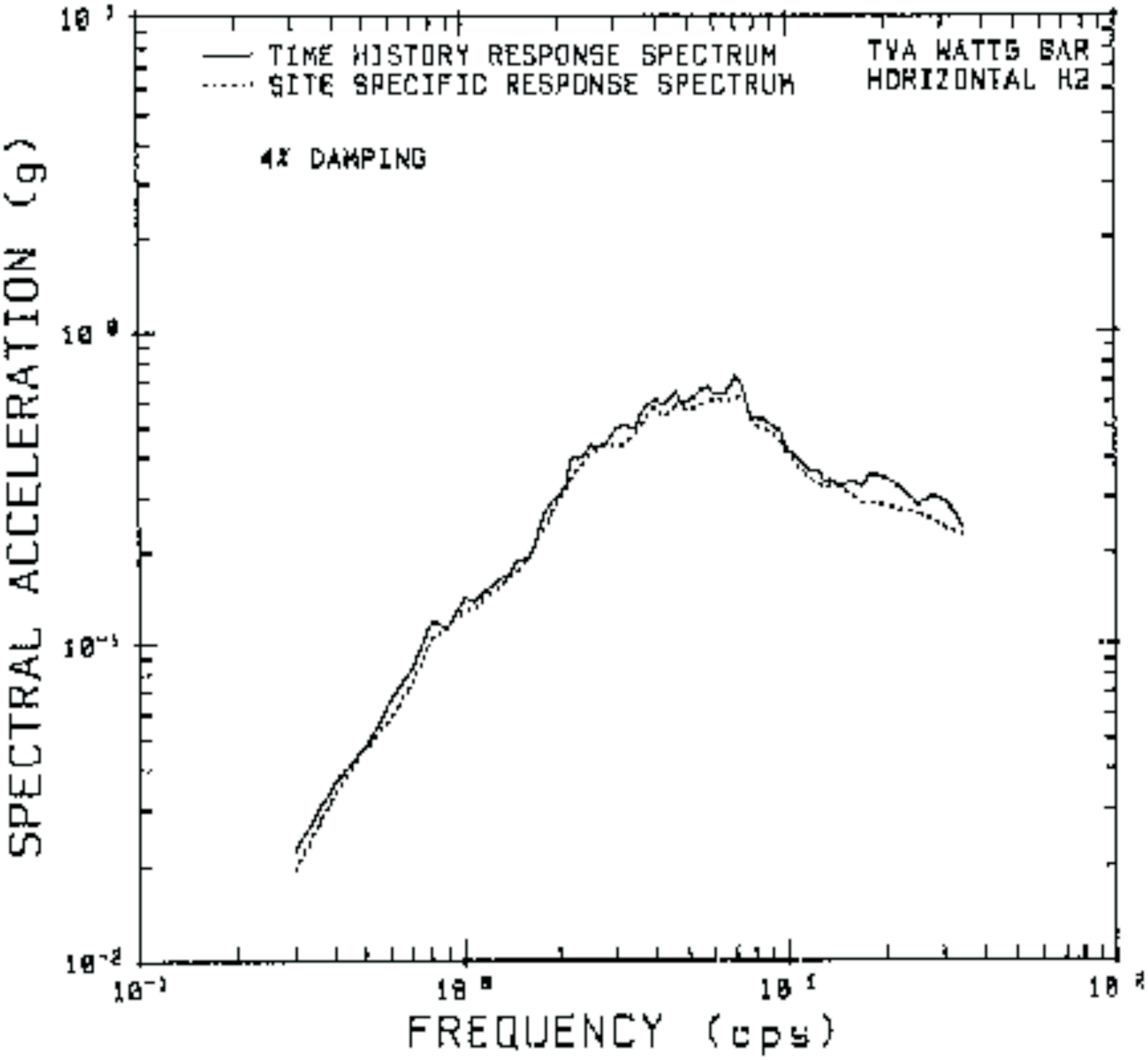
FIGURE 3.7-4h



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN
RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
3% DAMPING

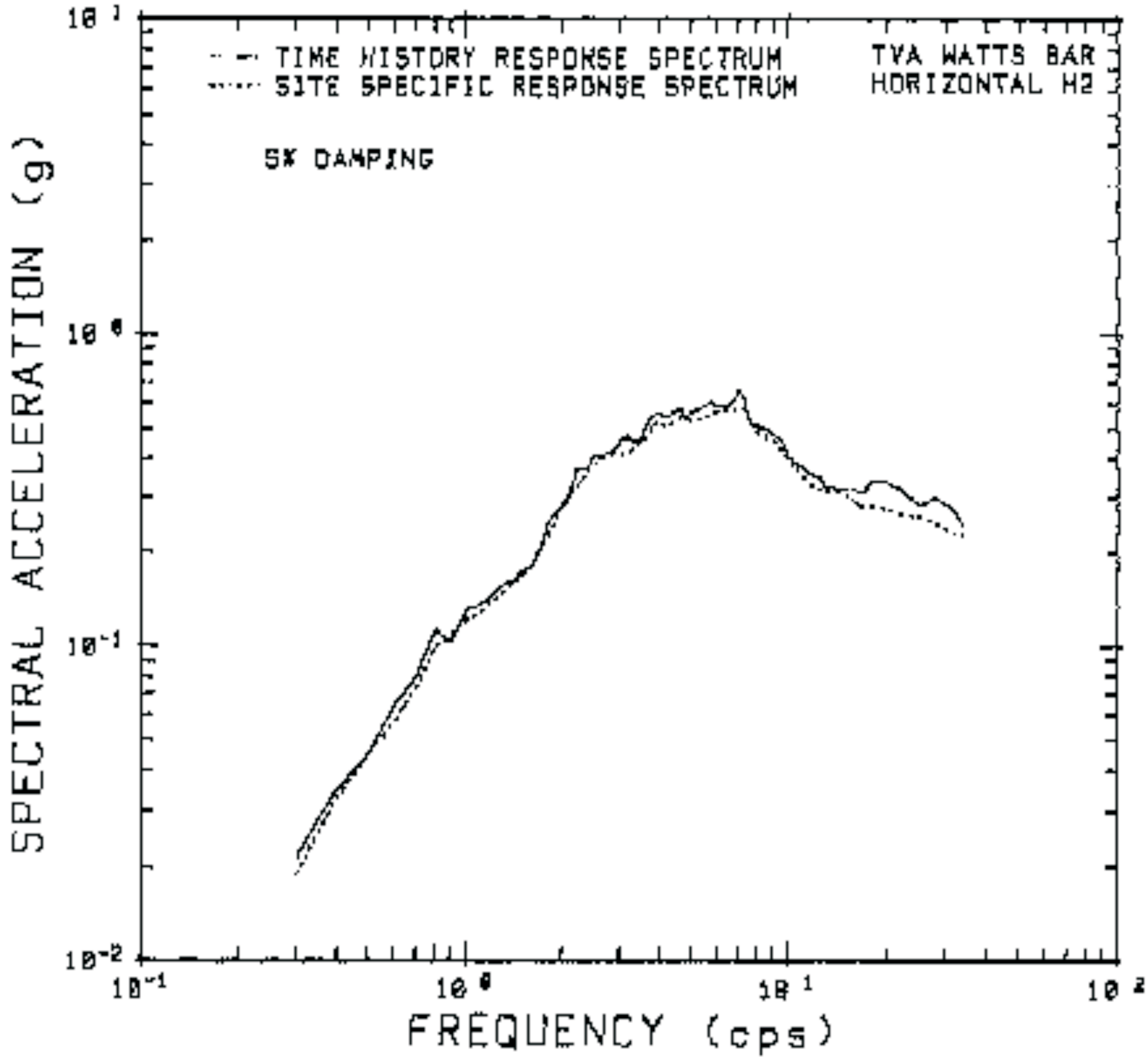
FIGURE 3.7-4i



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
4% DAMPING

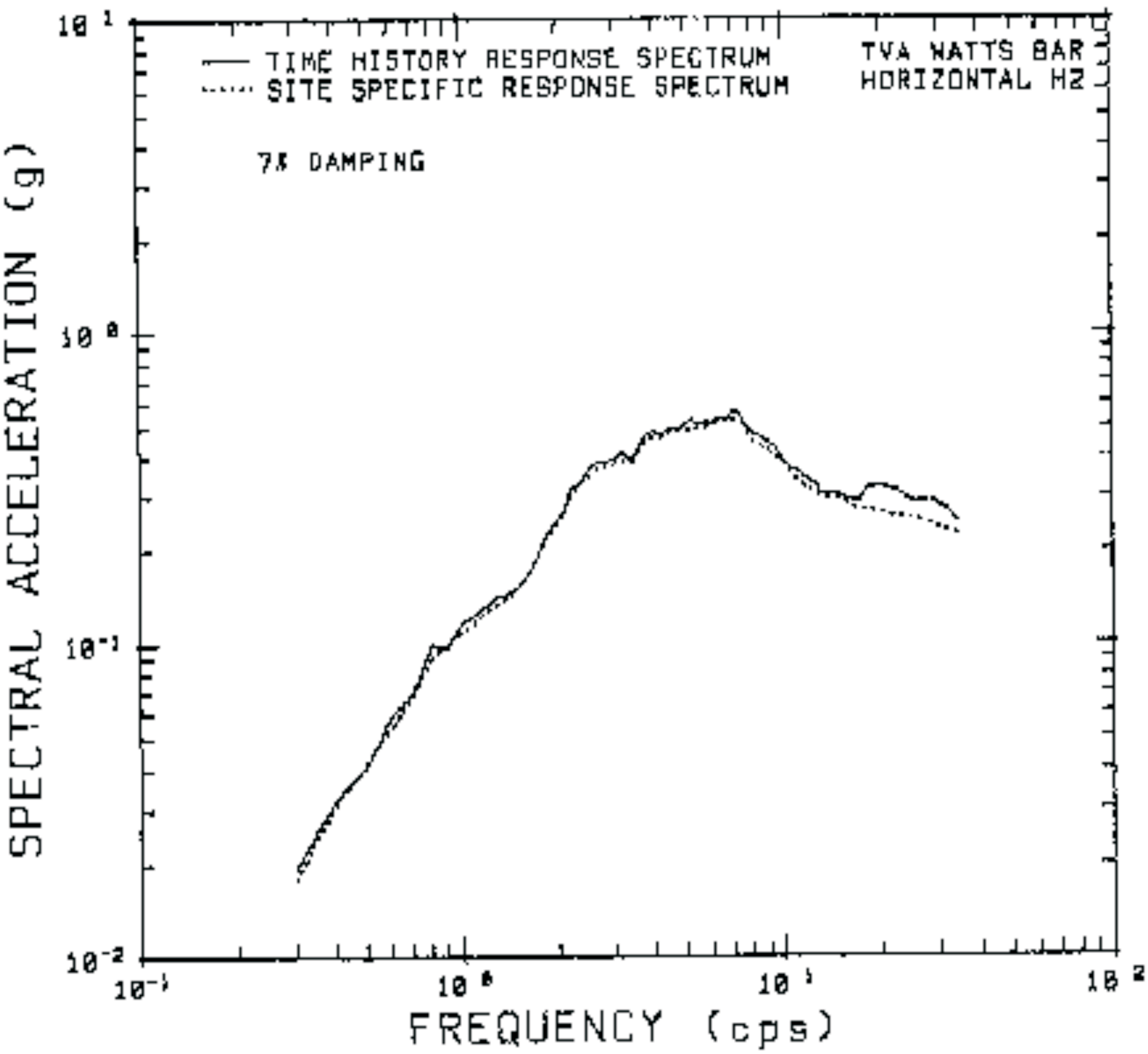
FIGURE 3.7-4j



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
5% DAMPING

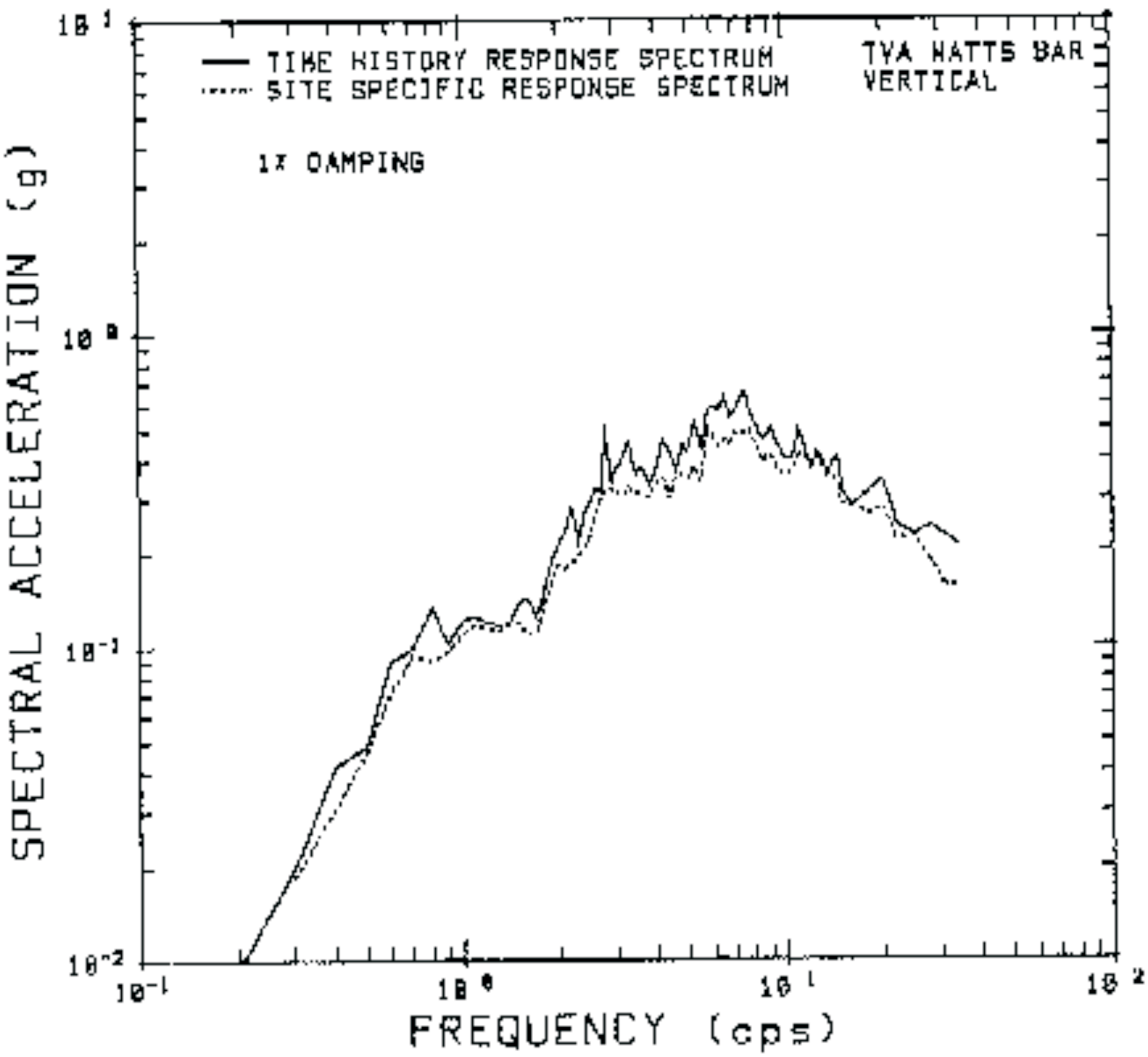
FIGURE 3.7-4k



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (E-W)
ROCK SUPPORTED STRUCTURES
7% DAMPING

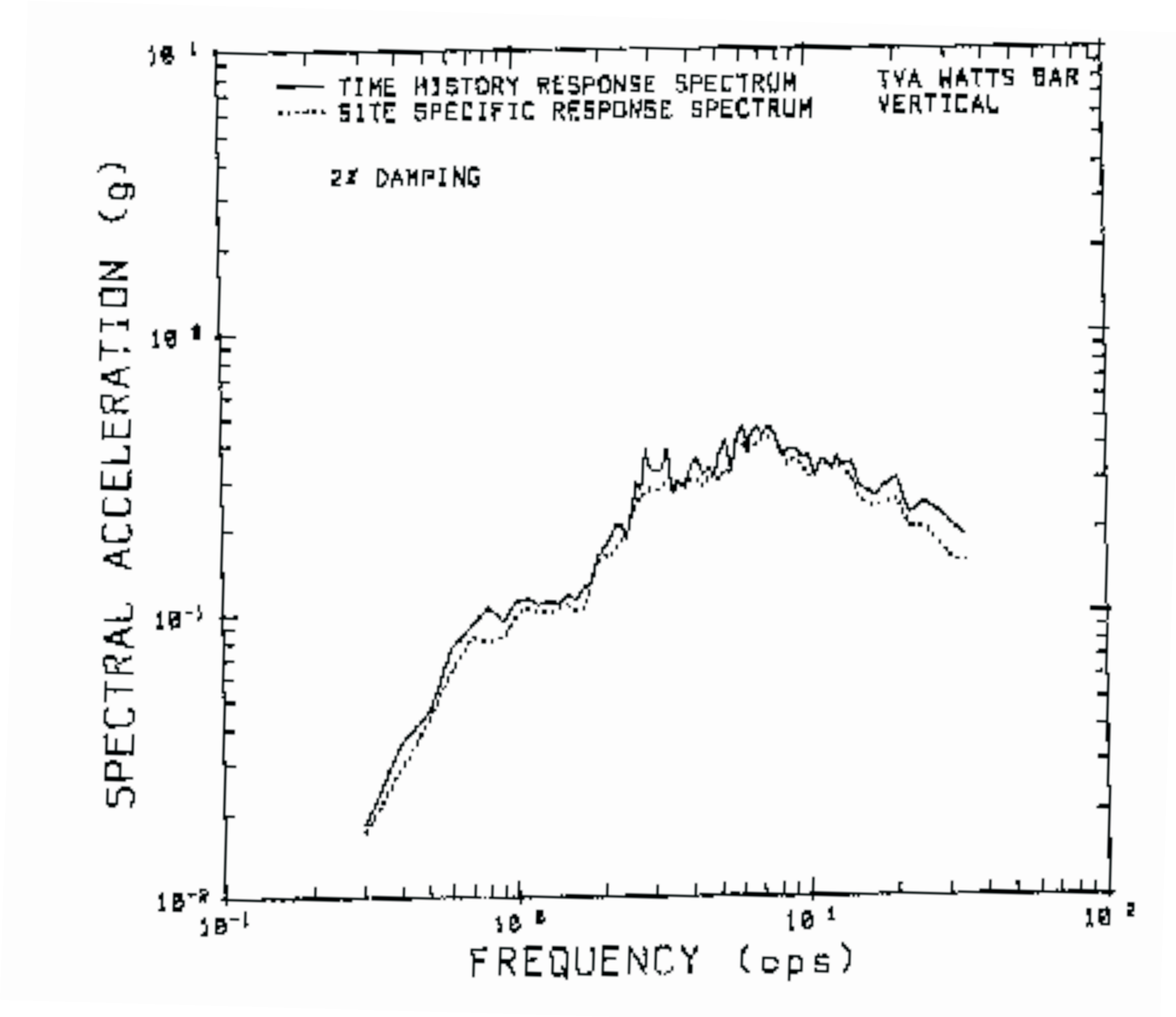
FIGURE 3.7-4l



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
1% DAMPING

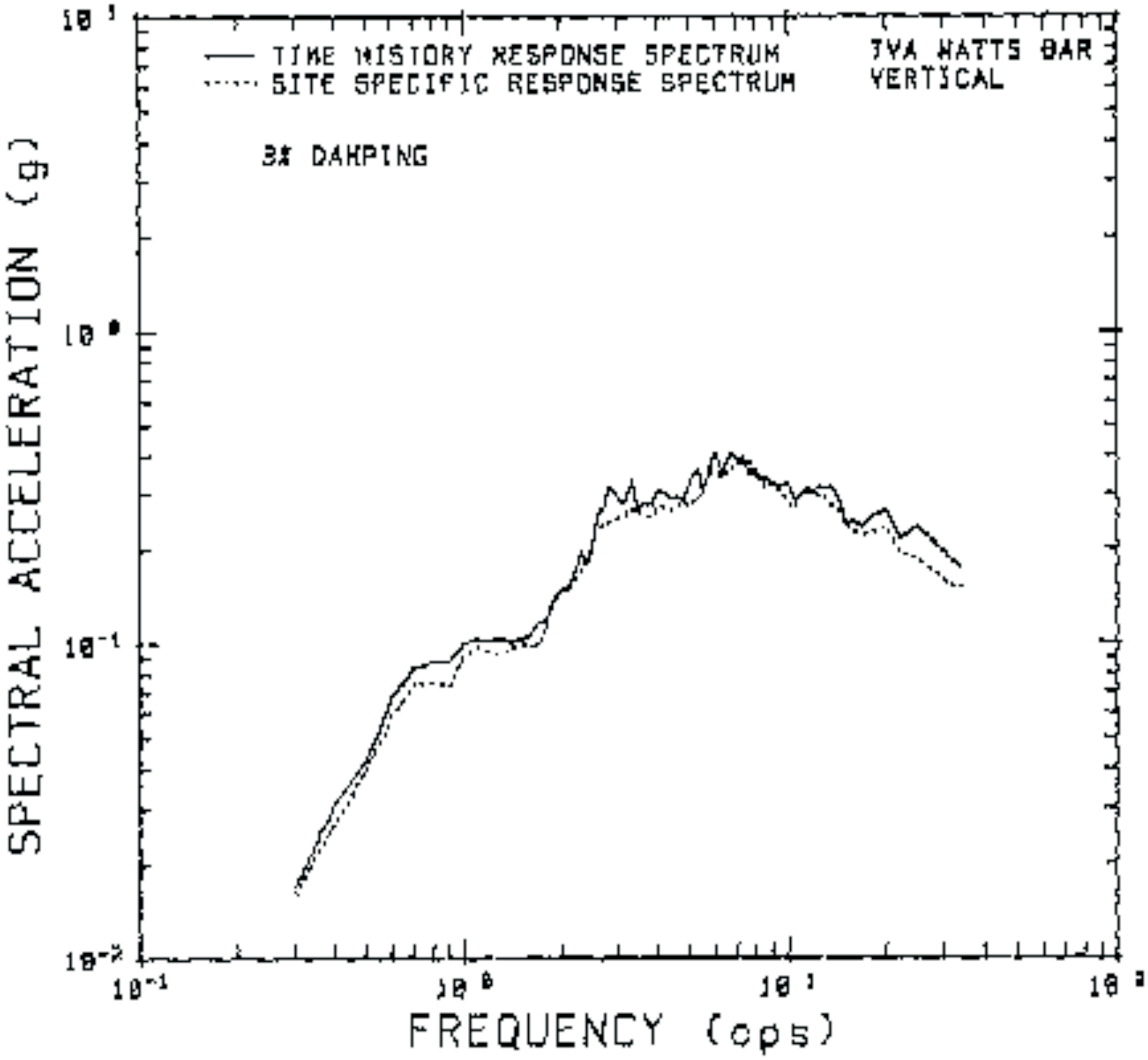
FIGURE 3.7-4m



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
2% DAMPING

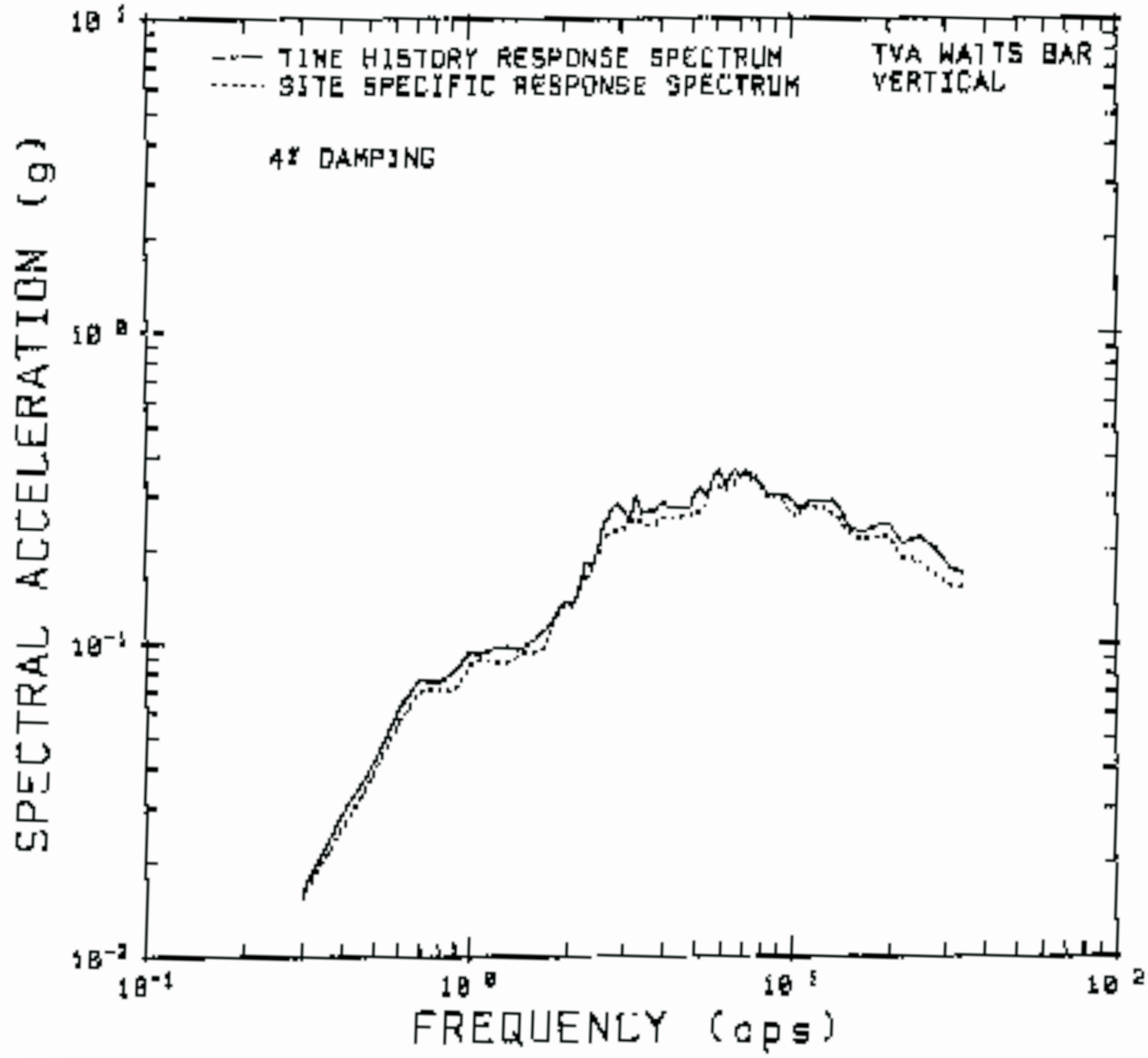
FIGURE 3.7-4n



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
3% DAMPING

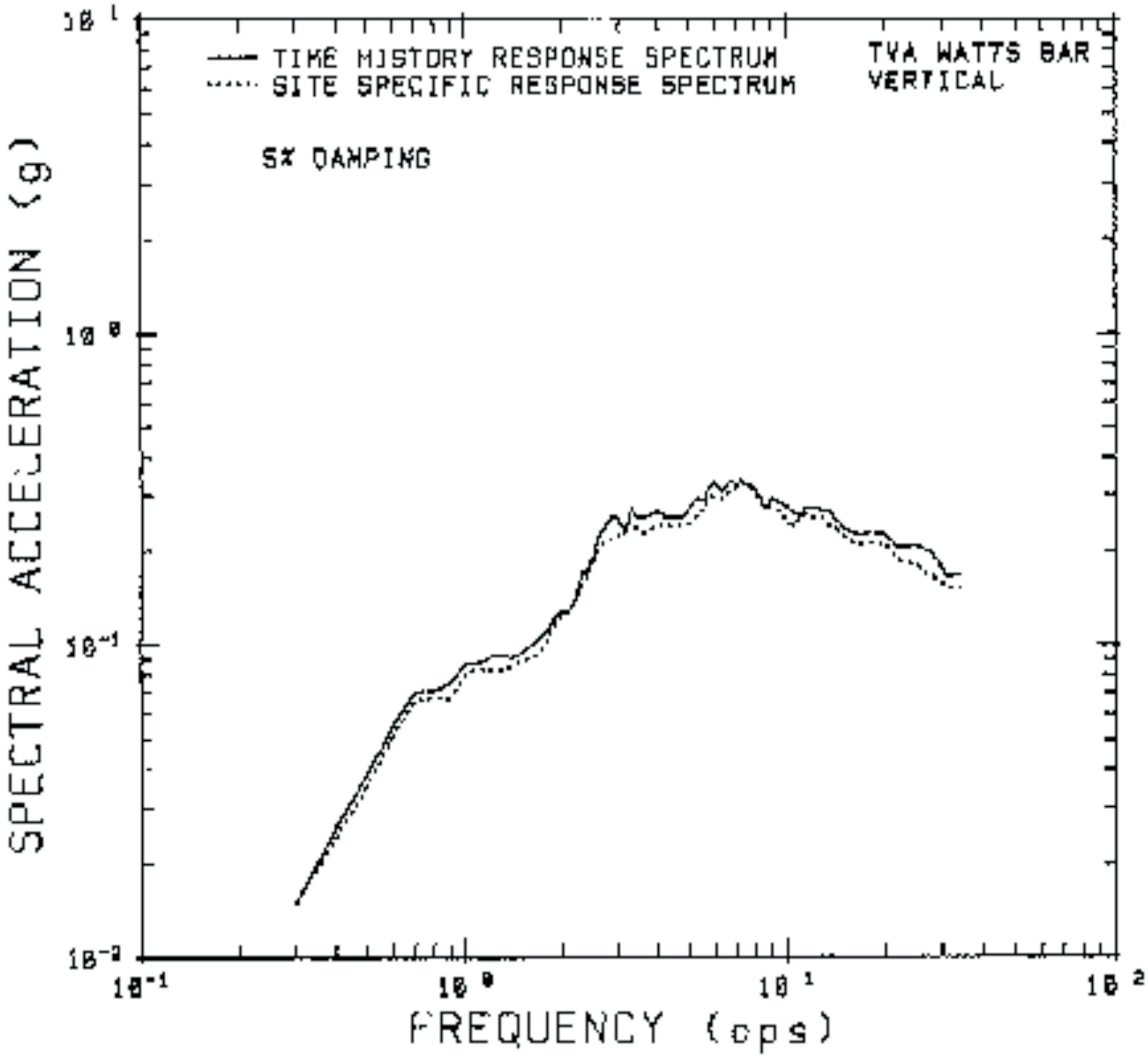
FIGURE 3.7-4o



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
4% DAMPING

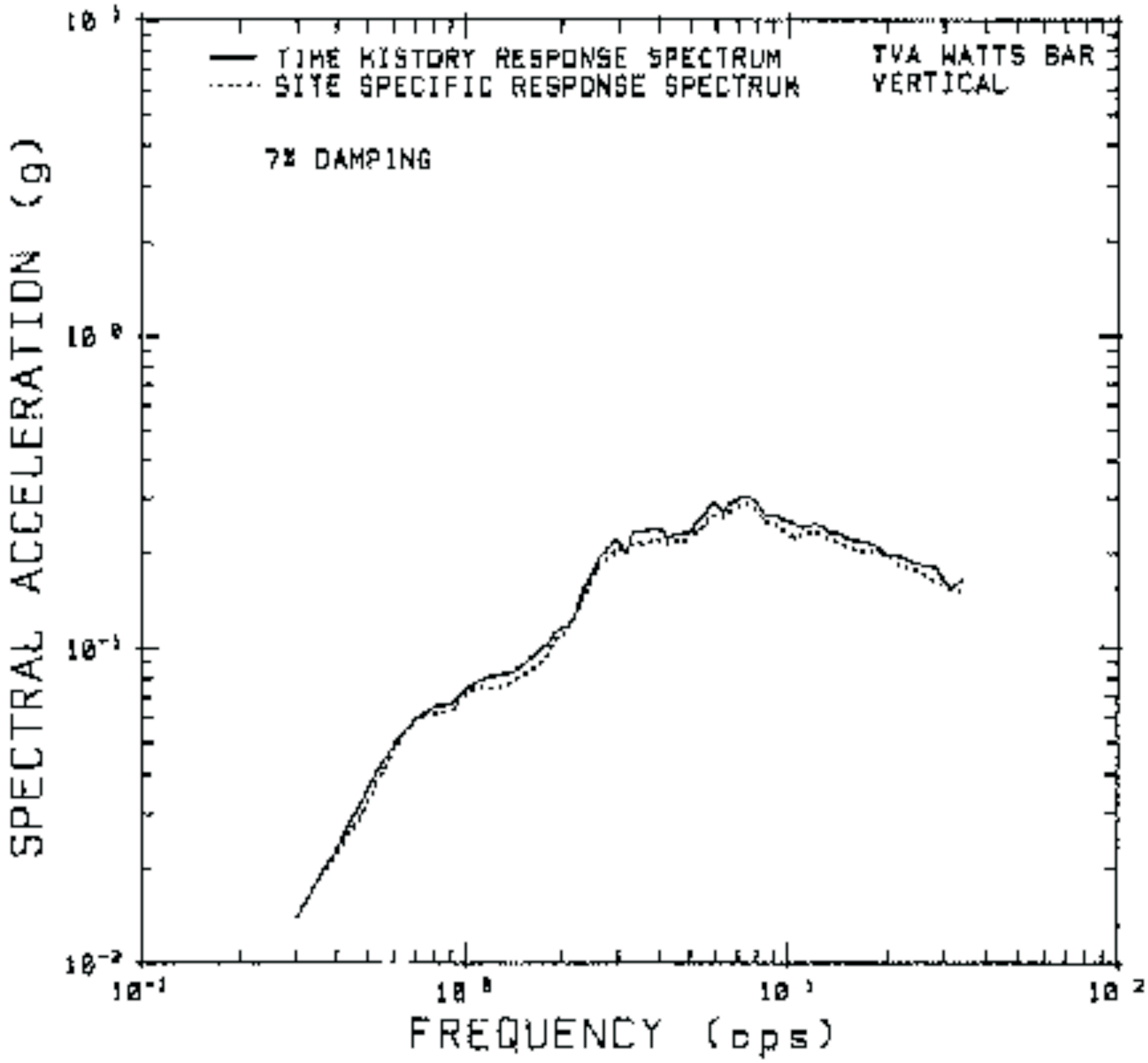
FIGURE 3.7-4p



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
5% DAMPING

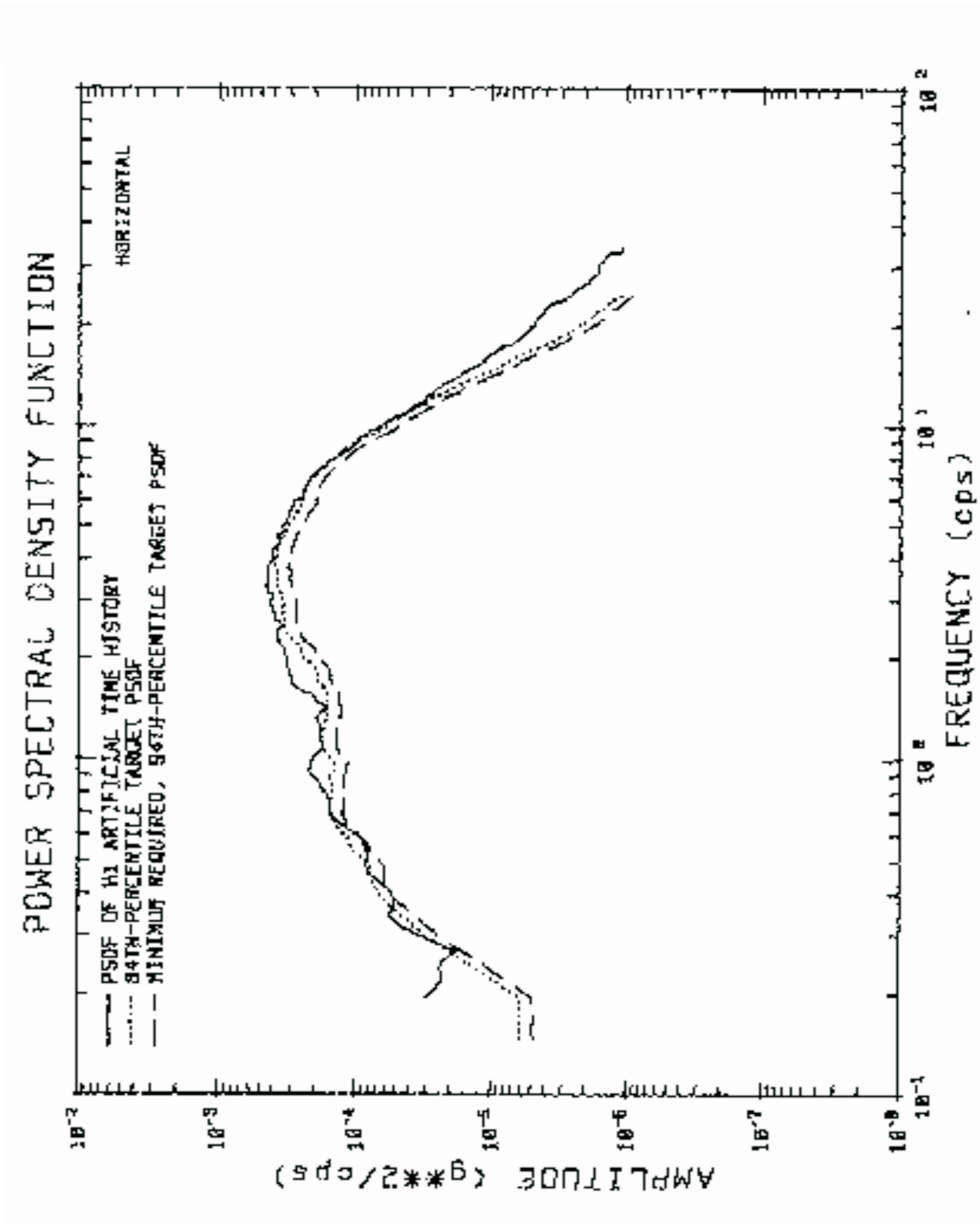
FIGURE 3.7-4q



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SET B SITE SPECIFIC DESIGN RESPONSE SPECTRUM
SAFE SHUTDOWN EARTHQUAKE (VERTICAL)
ROCK SUPPORTED STRUCTURES
7% DAMPING

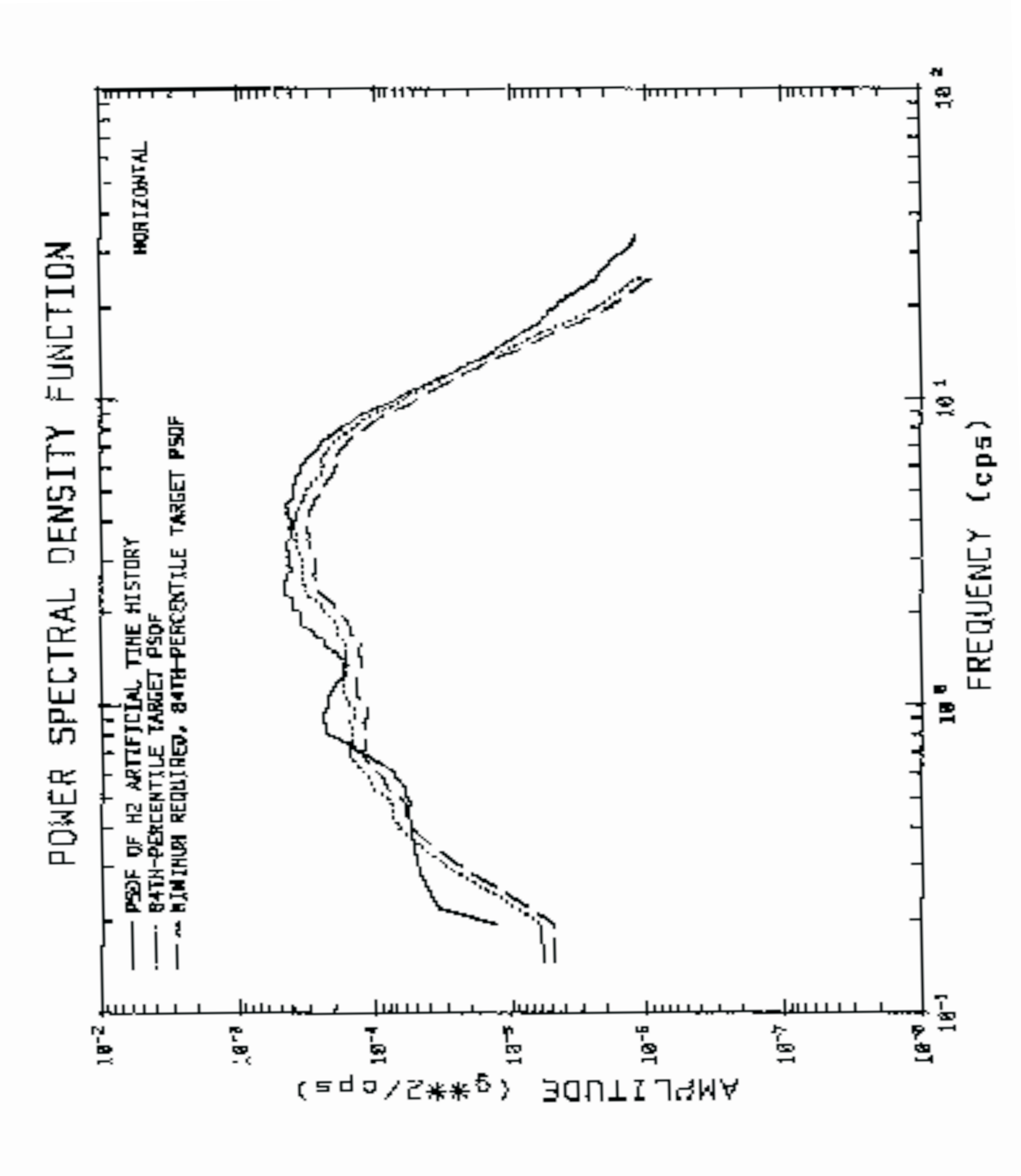
FIGURE 3.7-4r



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPARISONS OF H1 ARTIFICIAL TIME HISTORY PSDF WITH HORIZONTAL, 84TH-PERCENTILE, AND MINIMUM REQUIRED, 84TH-PERCENTILE TARGET PSDFs

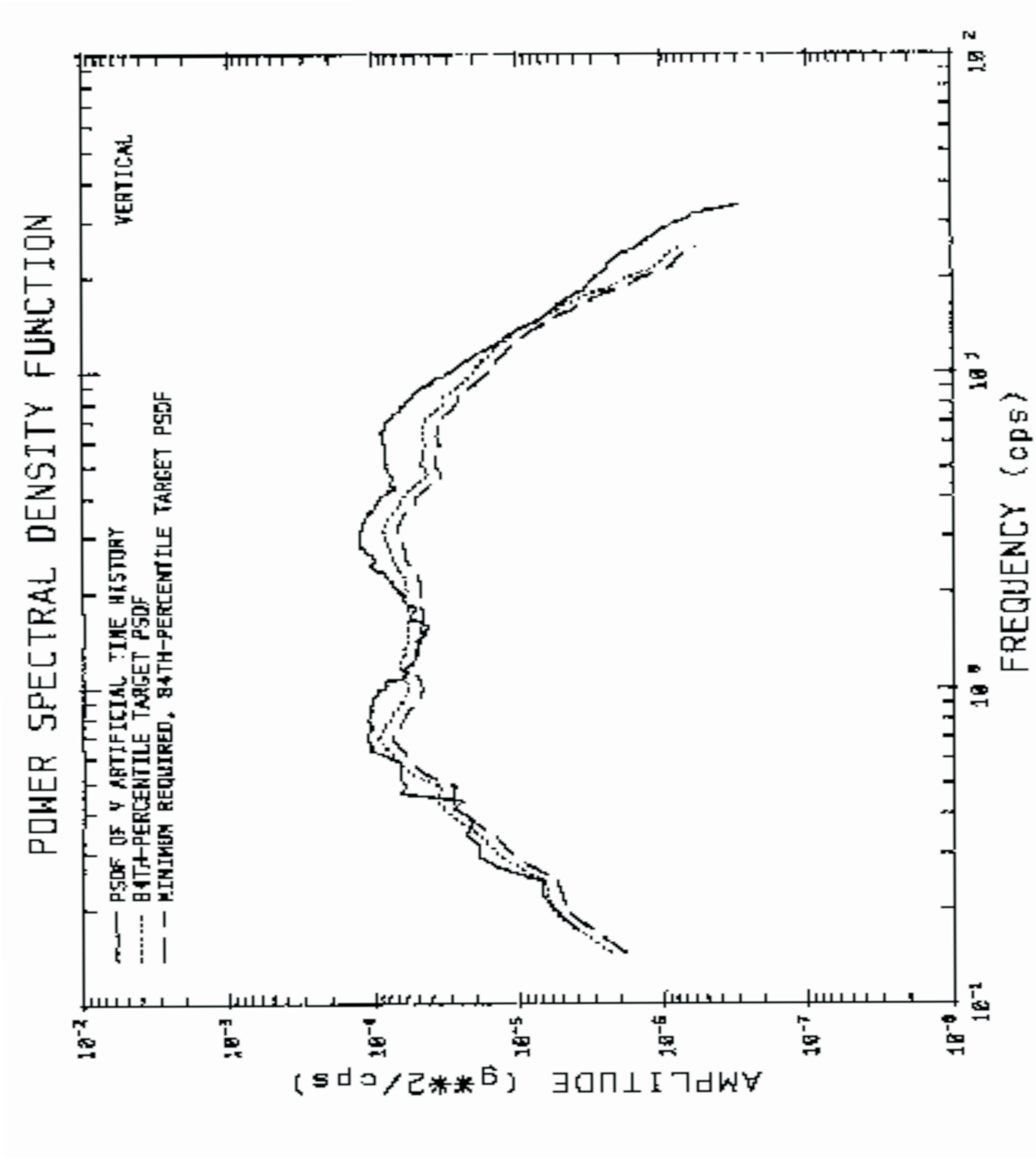
FIGURE 3.7-4s



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPARISONS OF H2 ARTIFICIAL TIME HISTORY PSDF WITH HORIZONTAL, 84TH-PERCENTILE, AND MINIMUM REQUIRED, 84TH-PERCENTILE TARGET PSDFs

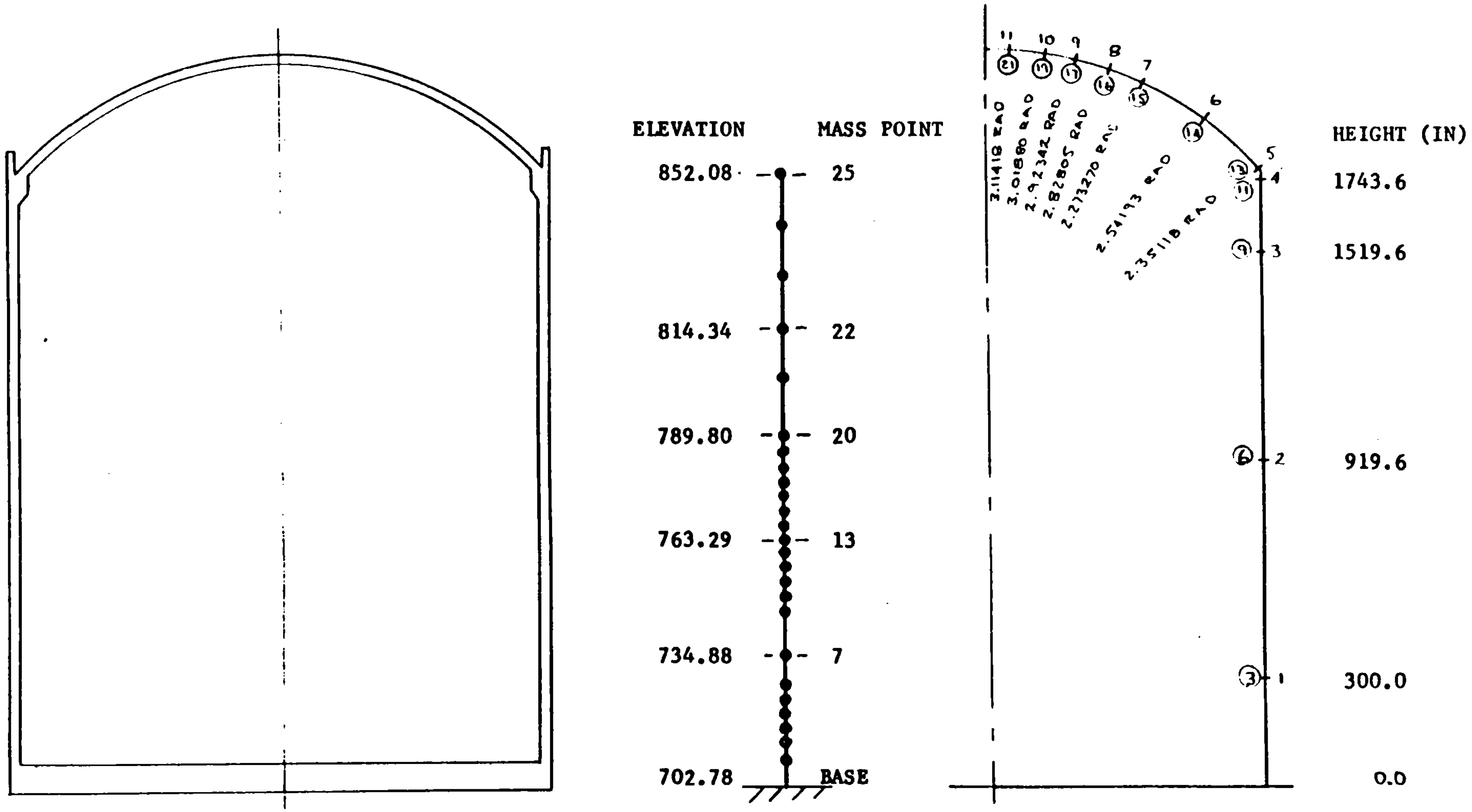
FIGURE 3.7-4t



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

COMPARISONS OF V ARTIFICIAL TIME HISTORY PSDF WITH HORIZONTAL, 84TH-PERCENTILE, AND MINIMUM REQUIRED, 84TH-PERCENTILE TARGET PSDFs

FIGURE 3.7-4u



SECTION THRU REACTOR SHIELD BUILDING

LUMPED MASS MODEL FOR ANALYSIS OF CYLINDIRCAL SHELL

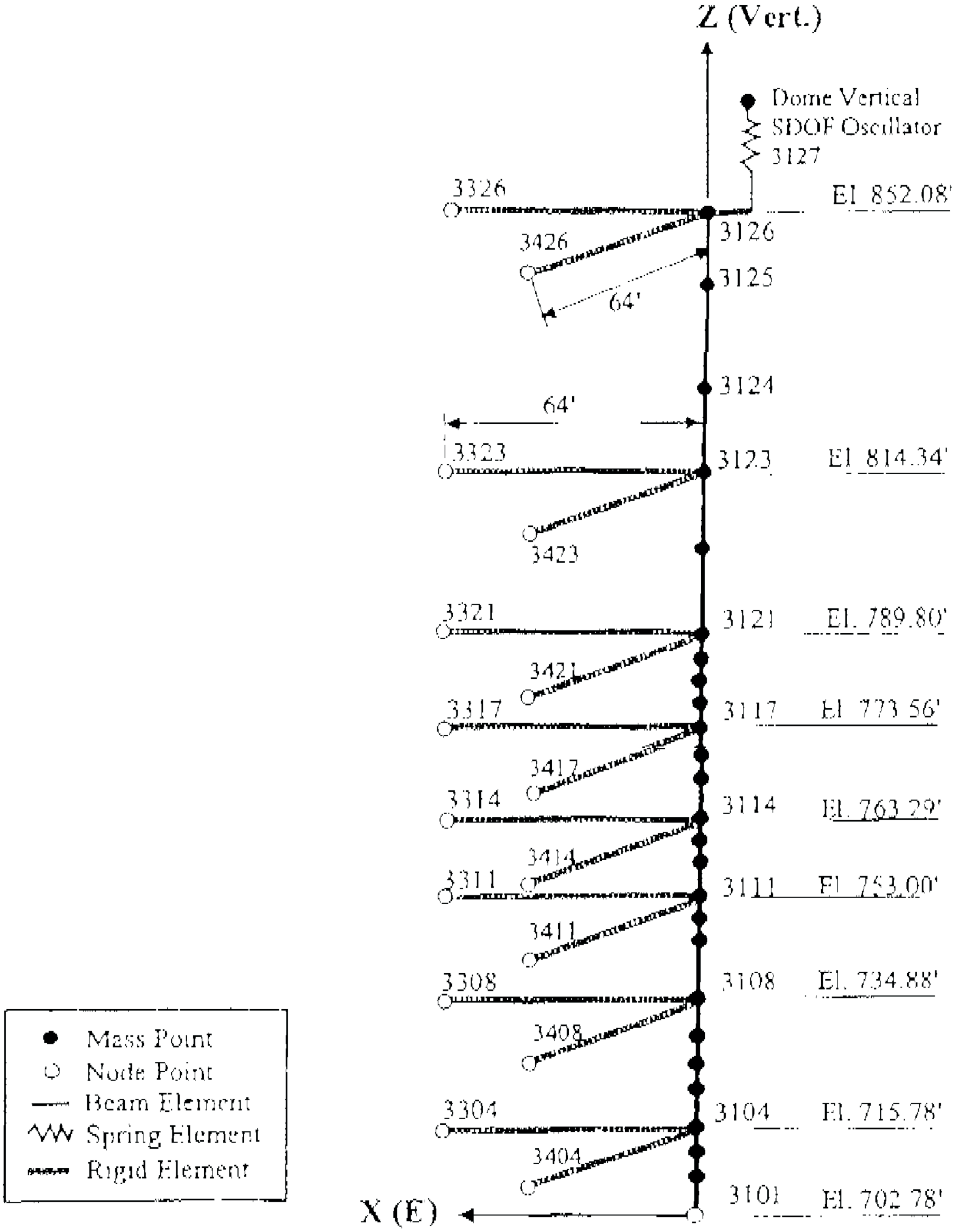
LUMPED MASS MODEL FOR ANALYSIS OF DOME

× - MASS POINT NOS.

⊗ - NODE NOS. FROM GENSHL-2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

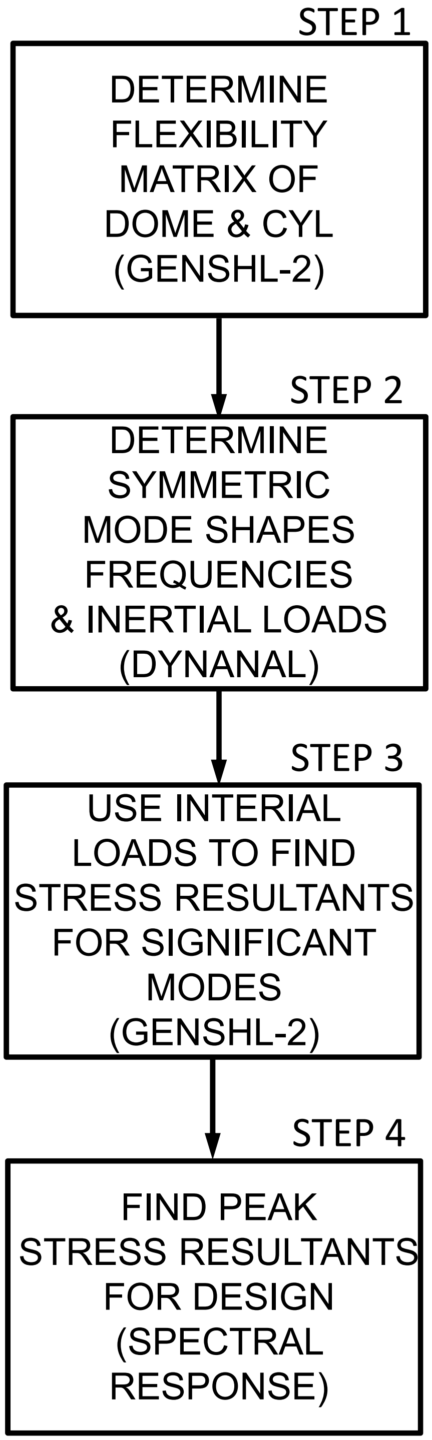
Lumped Mass Model
FIGURE 3.7-5



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Concentric Single-Stick Model Used for Unit-1 Shield Building

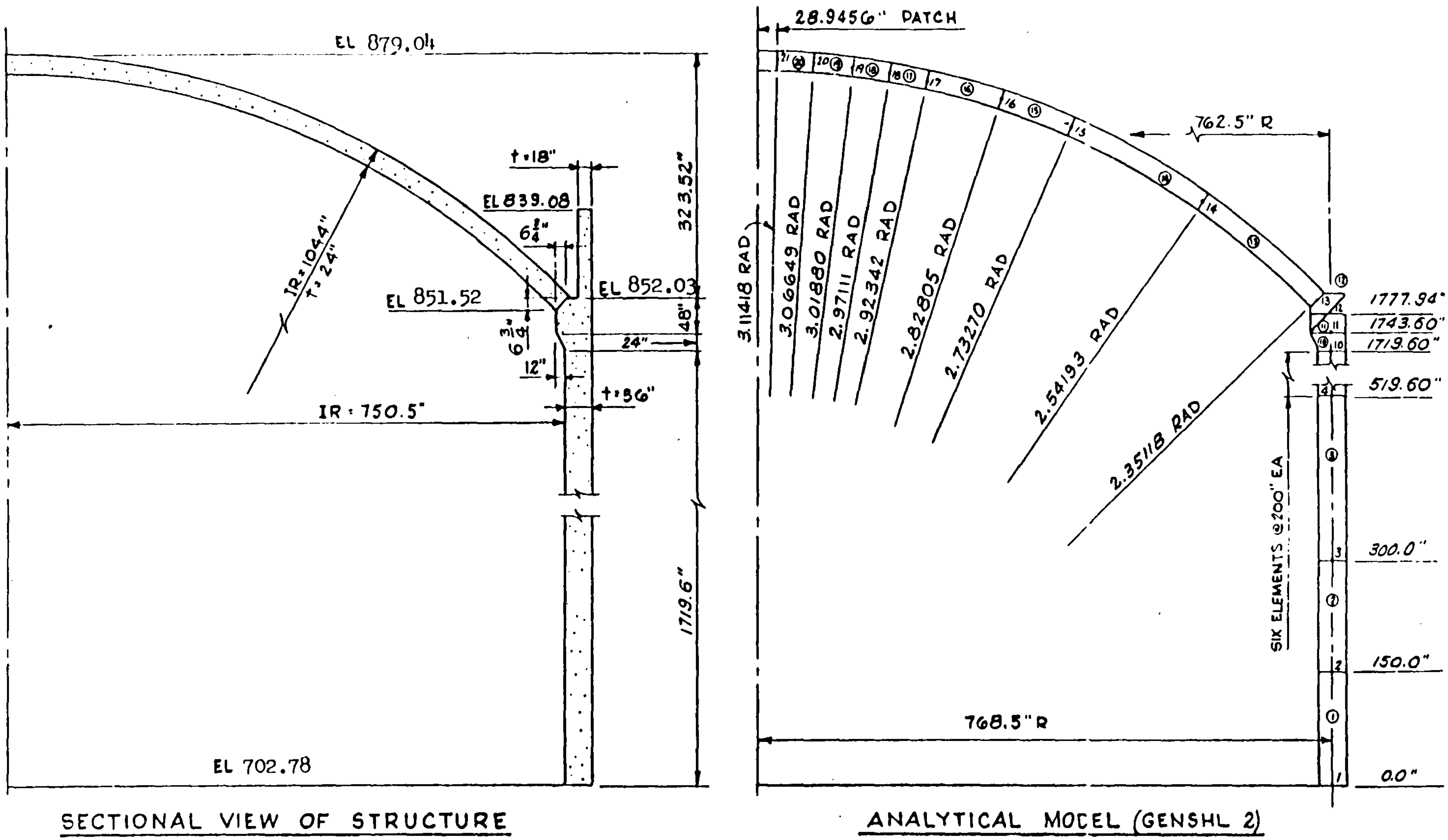
FIGURE 3.7-5a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Flow Chart of Operations
For Response of the Dome

FIGURE 3.7-6

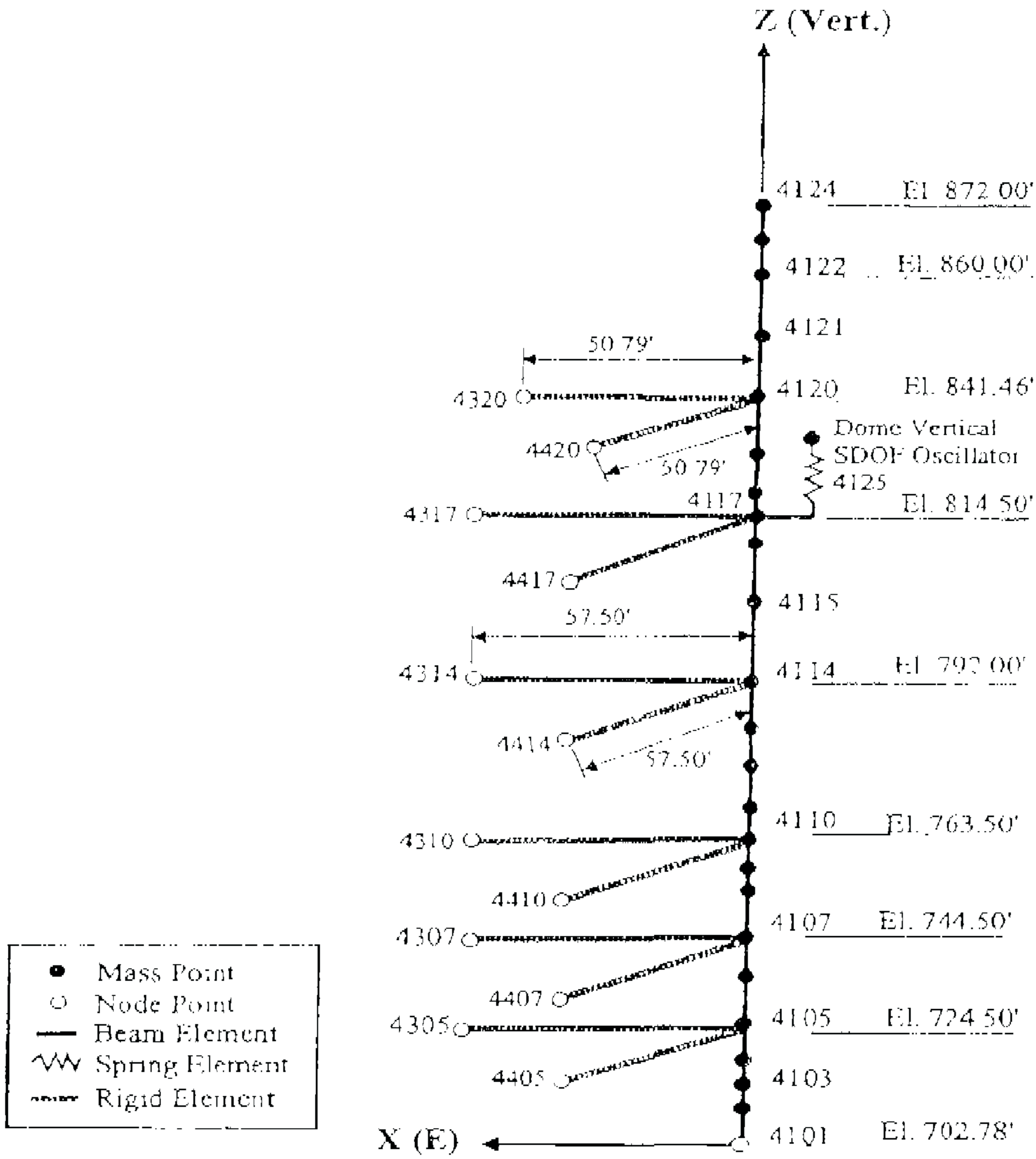


SHELL MODEL FOR DOME ANALYSIS-SHIELD BUILDING

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Shell Model for Dome Analysis

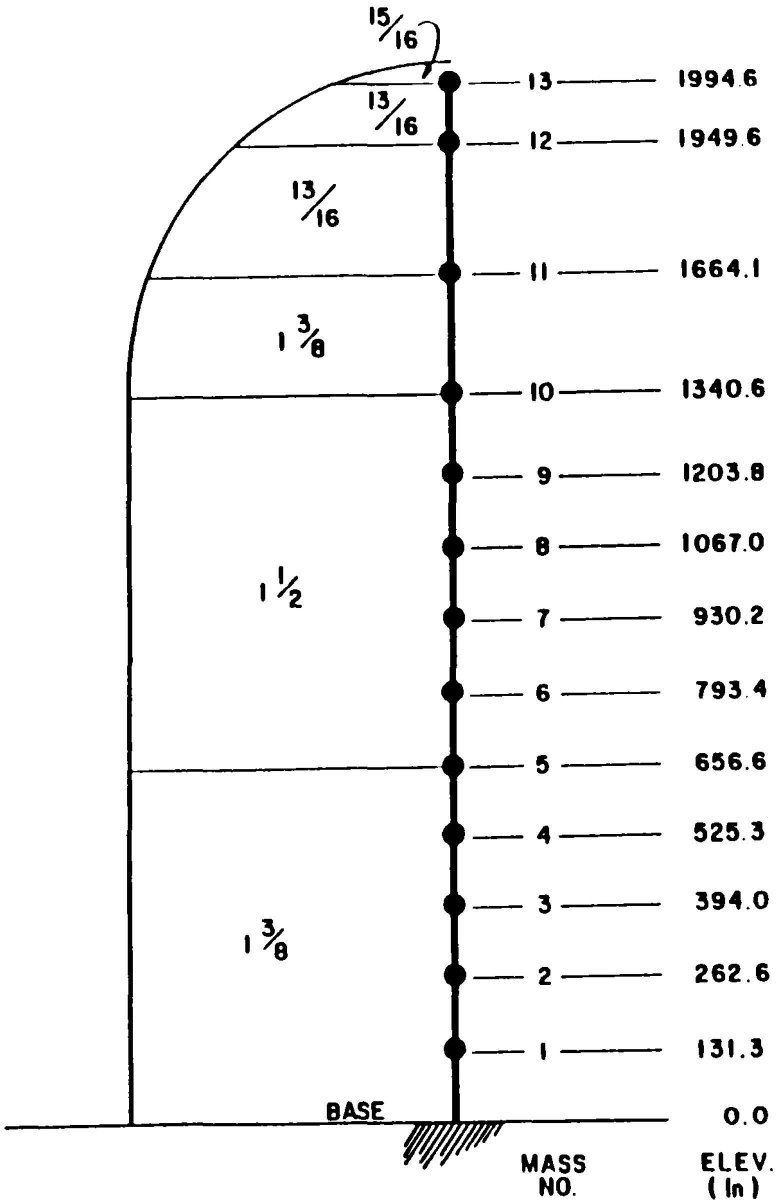
FIGURE 3.7-7



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Concentric Single-Stick Model Used for Unit-1 Steel Containment Vessel

FIGURE 3.7-7a

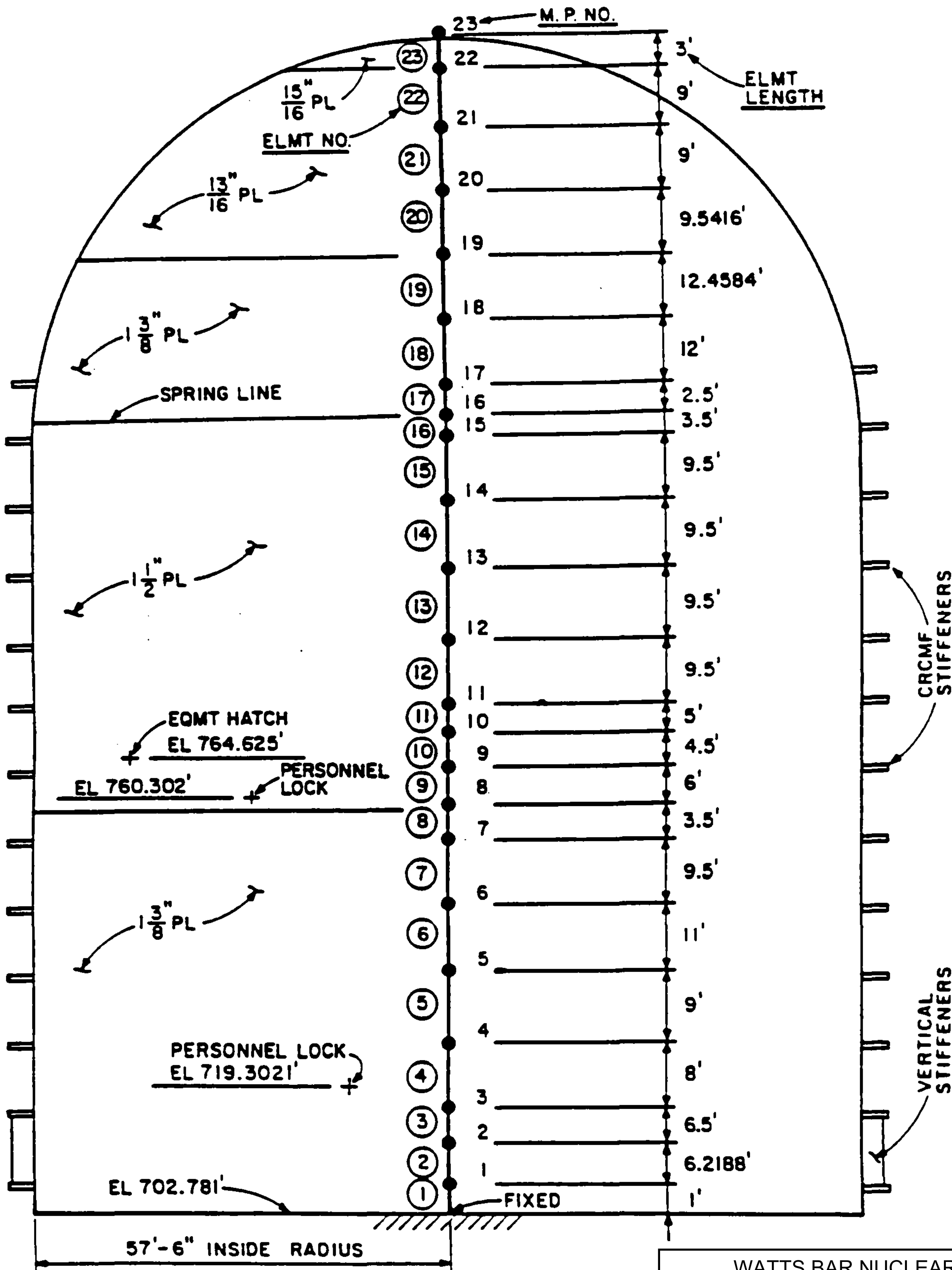


MASS NO.	I (In^4)	HORIZ A (In^2)	VERT A (In^2)	ELEV. (In)	WEIGHT (k)
1	$.1423 \times 10^{10}$	2981	5967	131.3	271.9
2	$.1423 \times 10^{10}$	2981	5967	262.6	509.5
3	$.1423 \times 10^{10}$	2981	5967	394.0	352.5
4	$.1423 \times 10^{10}$	2981	5967	525.3	425.7
5	$.1423 \times 10^{10}$	2981	5967	656.3	273.6
6	$.1553 \times 10^{10}$	3251	6510	793.4	501.2
7	$.1553 \times 10^{10}$	3251	6510	930.2	305.9
8	$.1553 \times 10^{10}$	3251	6510	1067.0	410.4
9	$.1553 \times 10^{10}$	3251	6510	1203.8	372.4
10	$.1423 \times 10^{10}$	2981	5967	1340.6	315.0
11	$.0578 \times 10^{10}$	1555	3111	1664.1	639.0
12	$.871 \times 10^{8}$	828	1656	1949.6	336.3
13	$.272 \times 10^{8}$	561	1123	1994.6	98.2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Containment Vessel Lumped Mass
Beam Model and Properties

FIGURE 3.7-7B

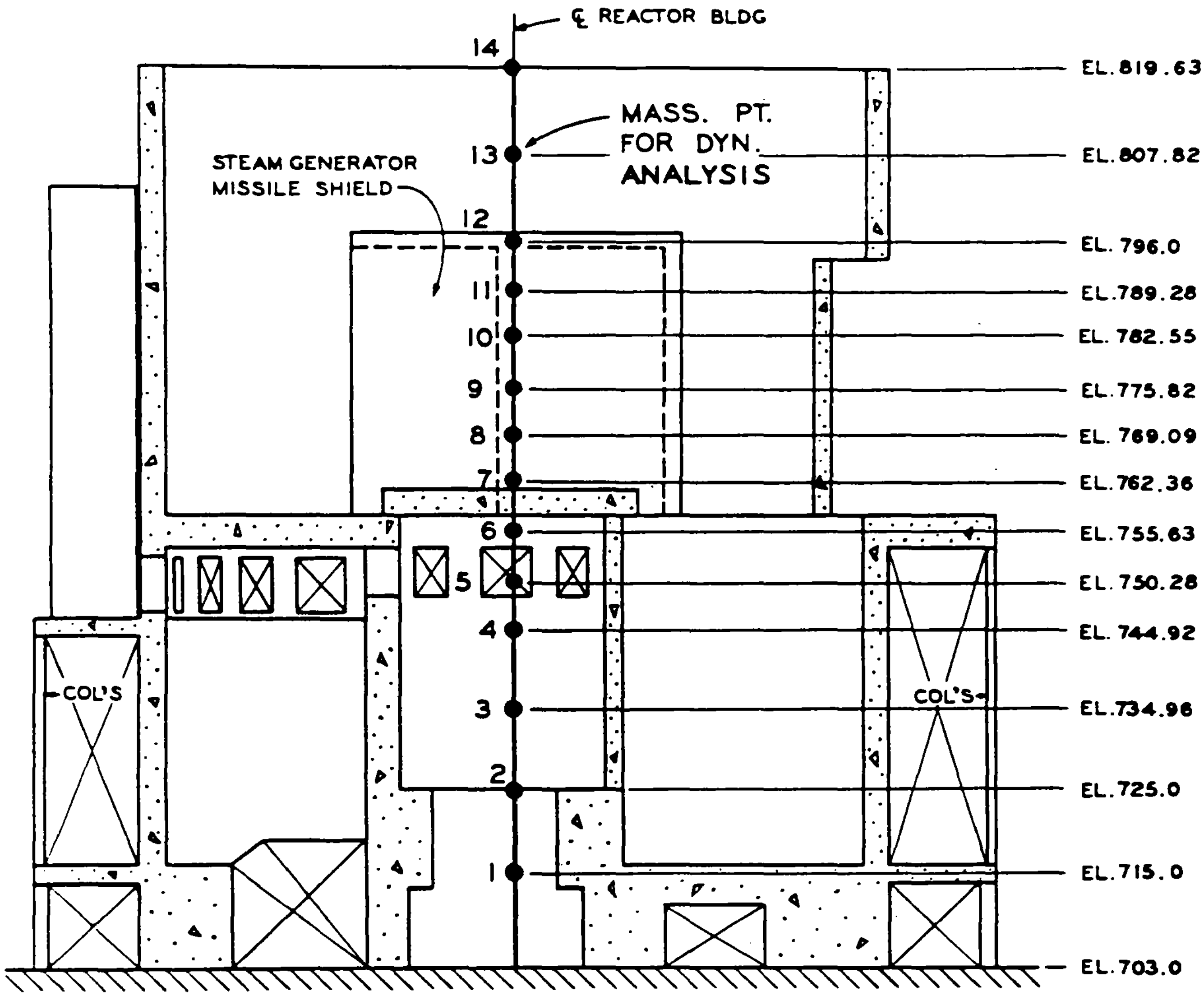


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of Steel
Containment Vessel
and Lumped Mass Model
for Seismic Analysis

FIGURE 3.7-7C

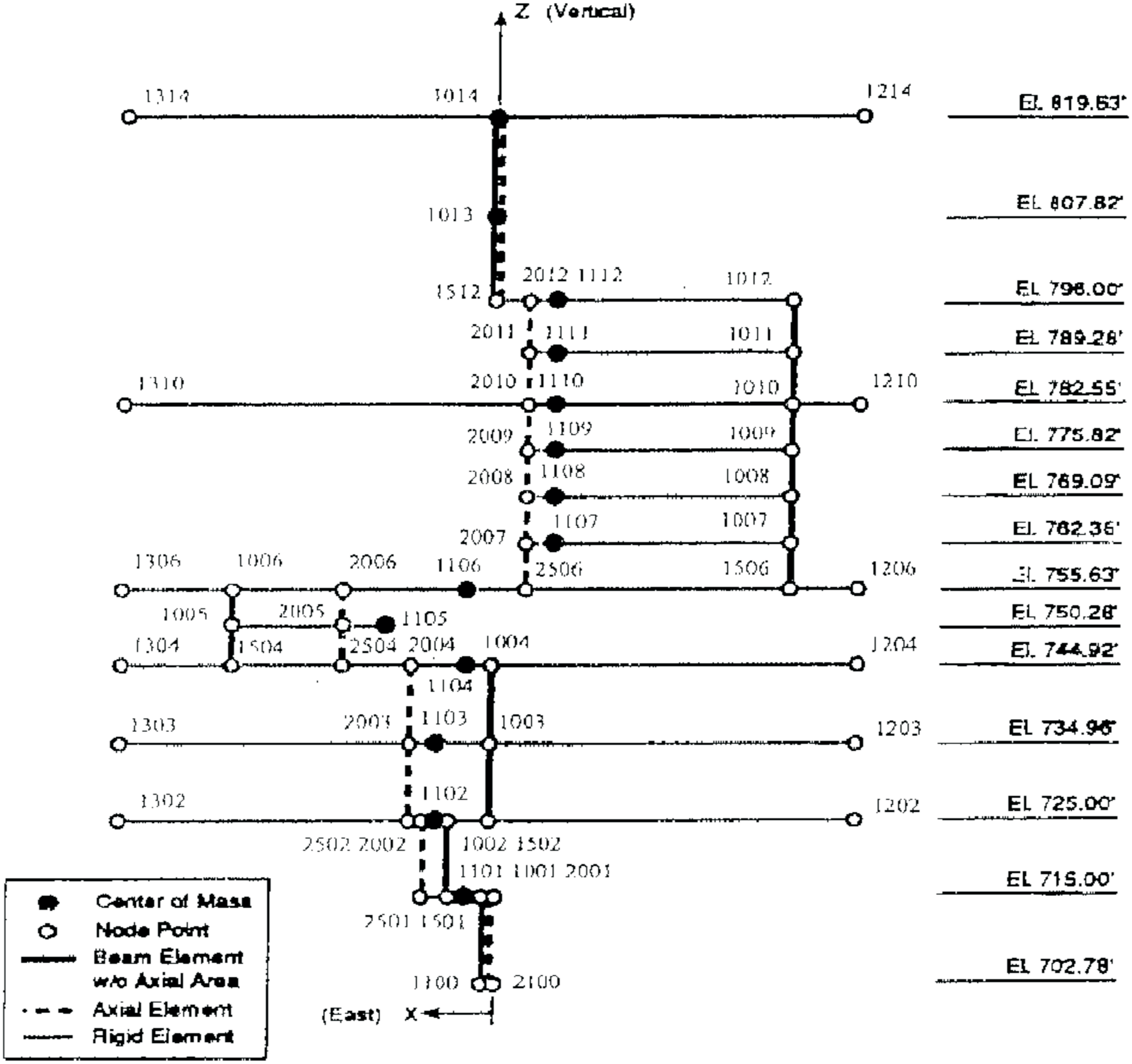
TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
REACTOR BUILDING - INTERIOR CONCRETE STRUCTURE



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation
Looking North
Lumped Mass Model
for Dynamic Analysis

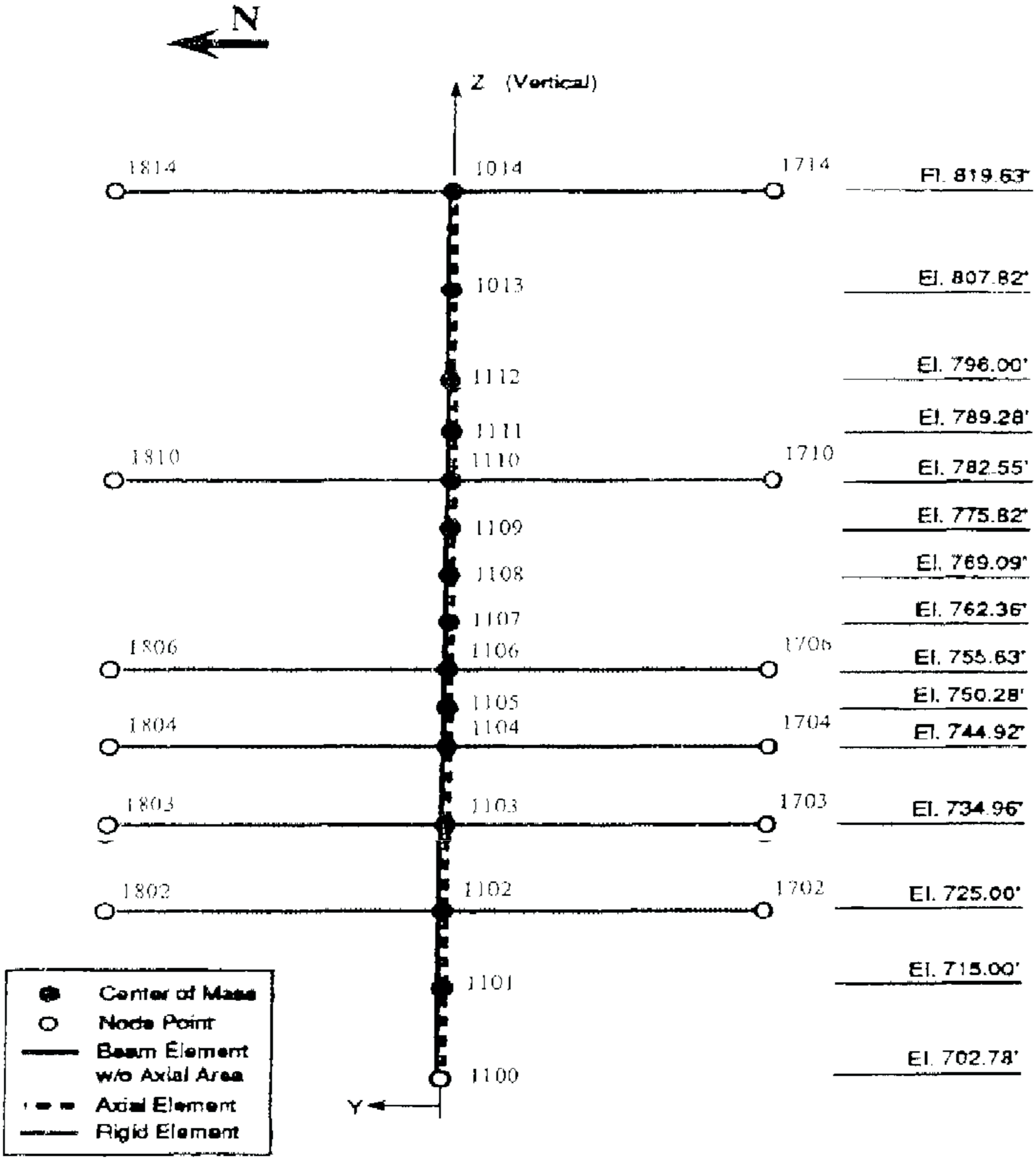
FIGURE 3.7-8



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Three-dimensional Multiple-Stick Seismic Model Used for Unit-1 Interior Concrete Structures (ICS) -XZ (EW-Vertical) Plane

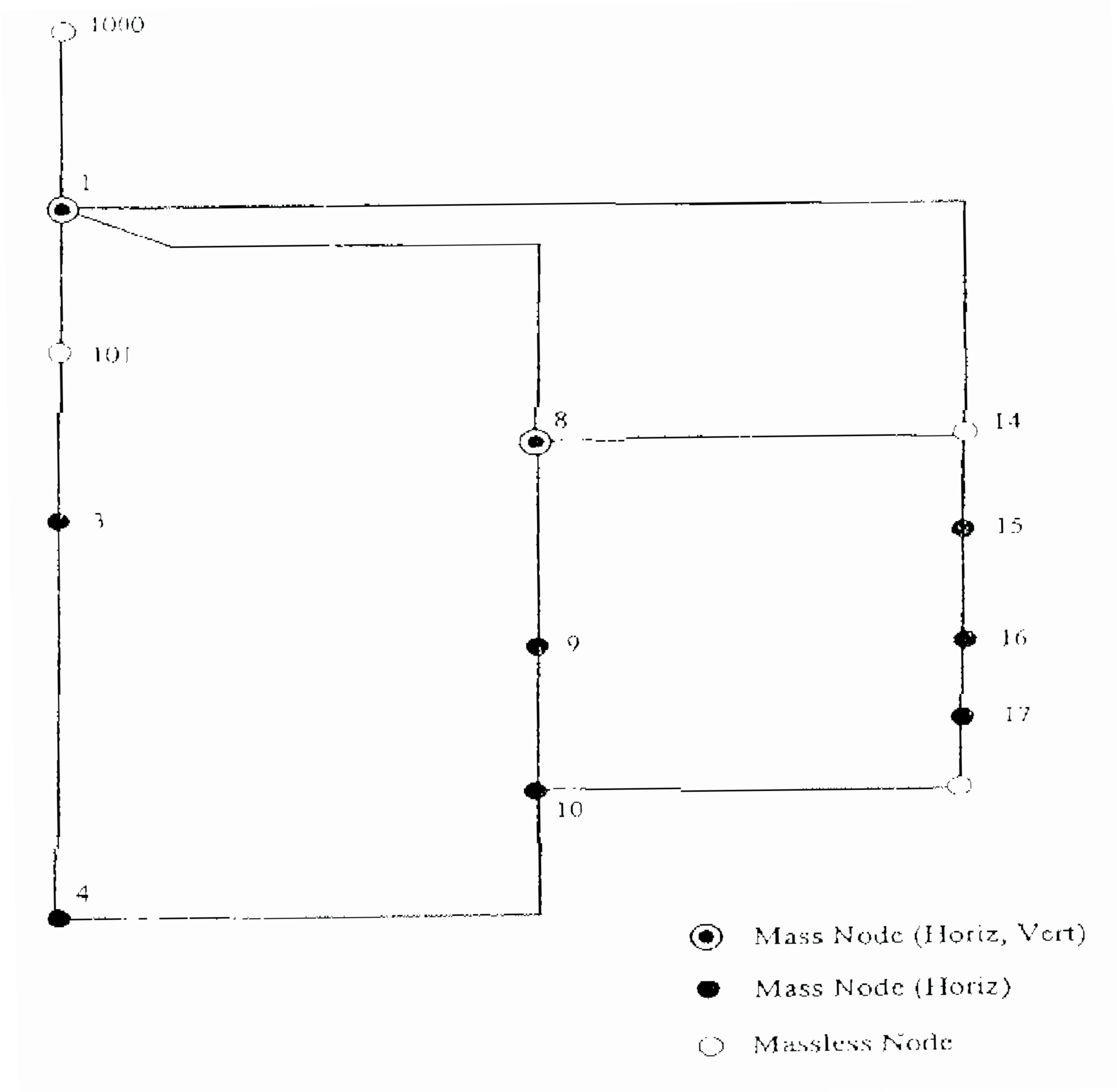
FIGURE 3.7-8a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Three-dimensional Multiple-Stick Seismic Model Used for Unit-1 Interior Concrete Structures (ICS) -YZ (NS-Vertical) Plane

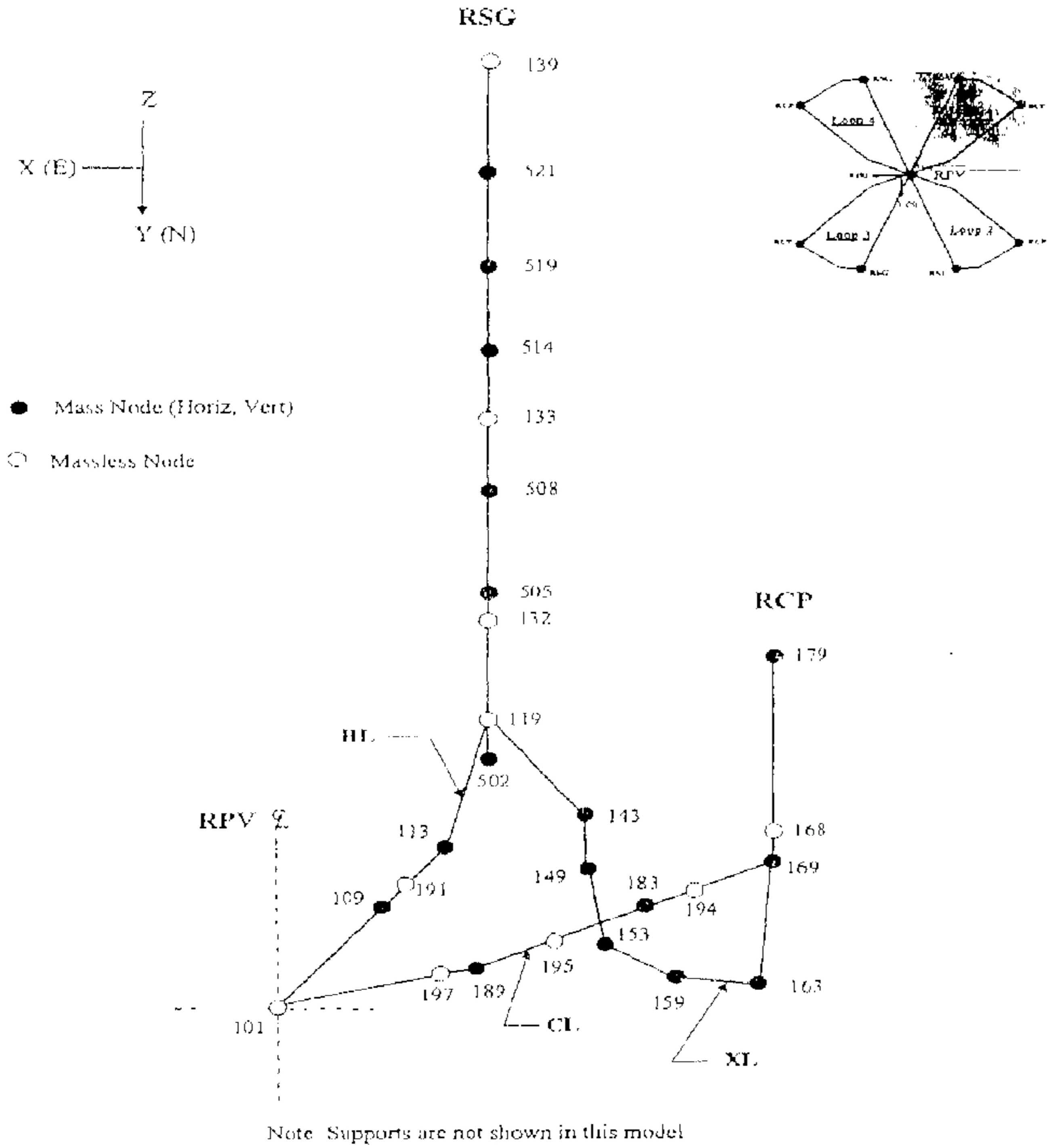
FIGURE 3.7-8b



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Configuration of the WBN Unit-1 Reactor Pressure Vessel (RPV) Seismic Model

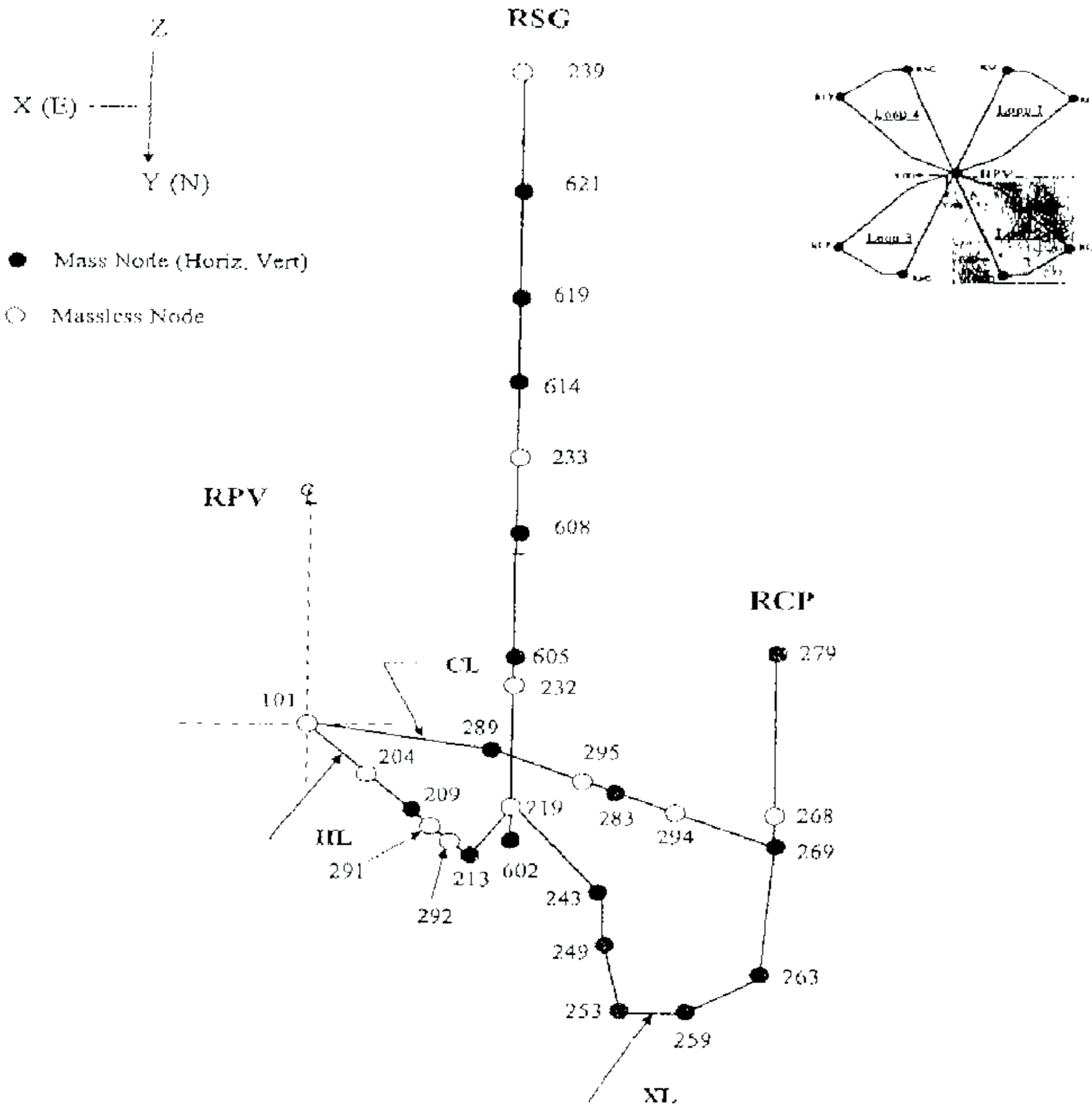
FIGURE 3.7-8c



WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 1

FIGURE 3.7-8d

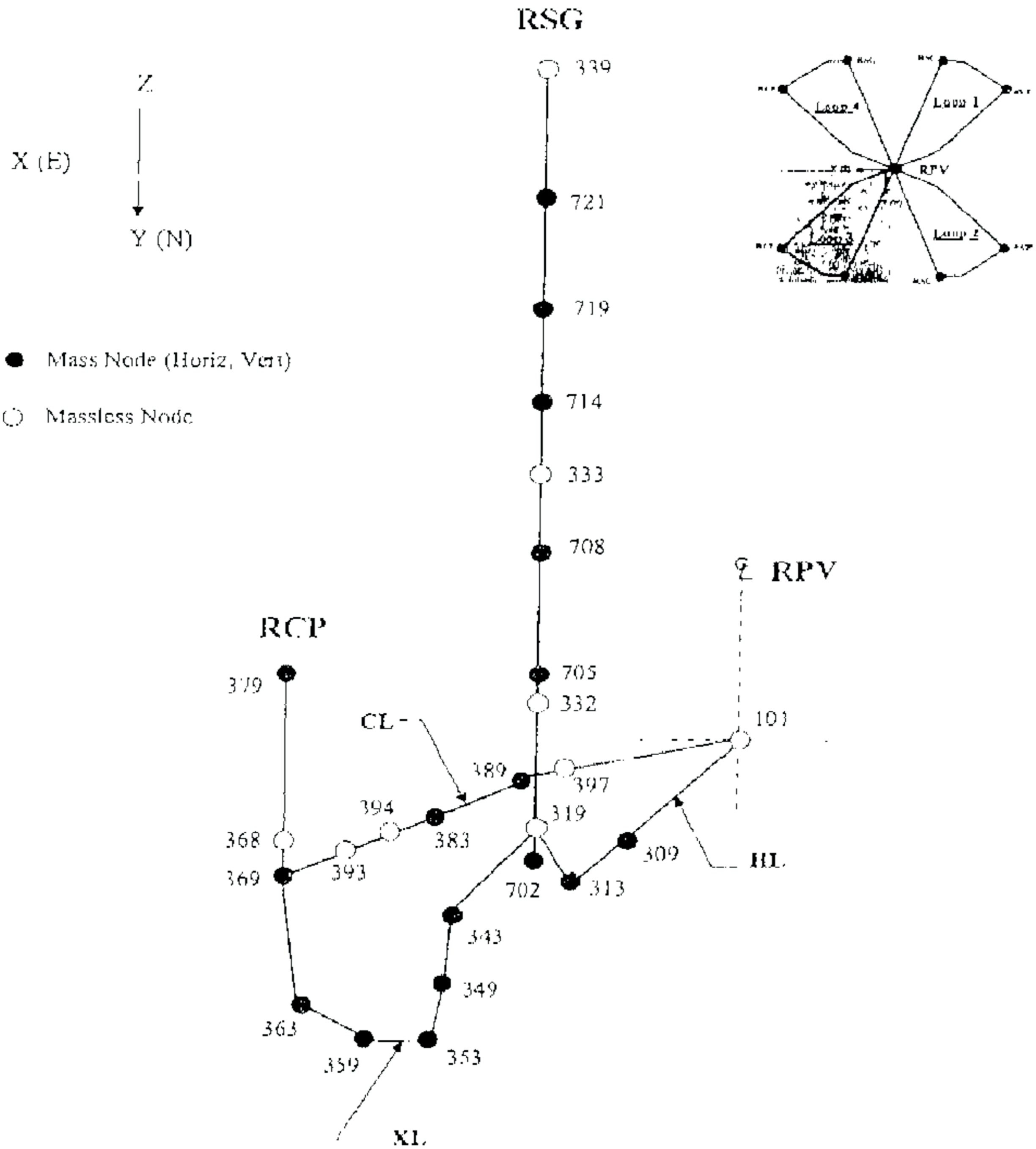


Note: Supports are not shown in this model.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 2

FIGURE 3.7-8e

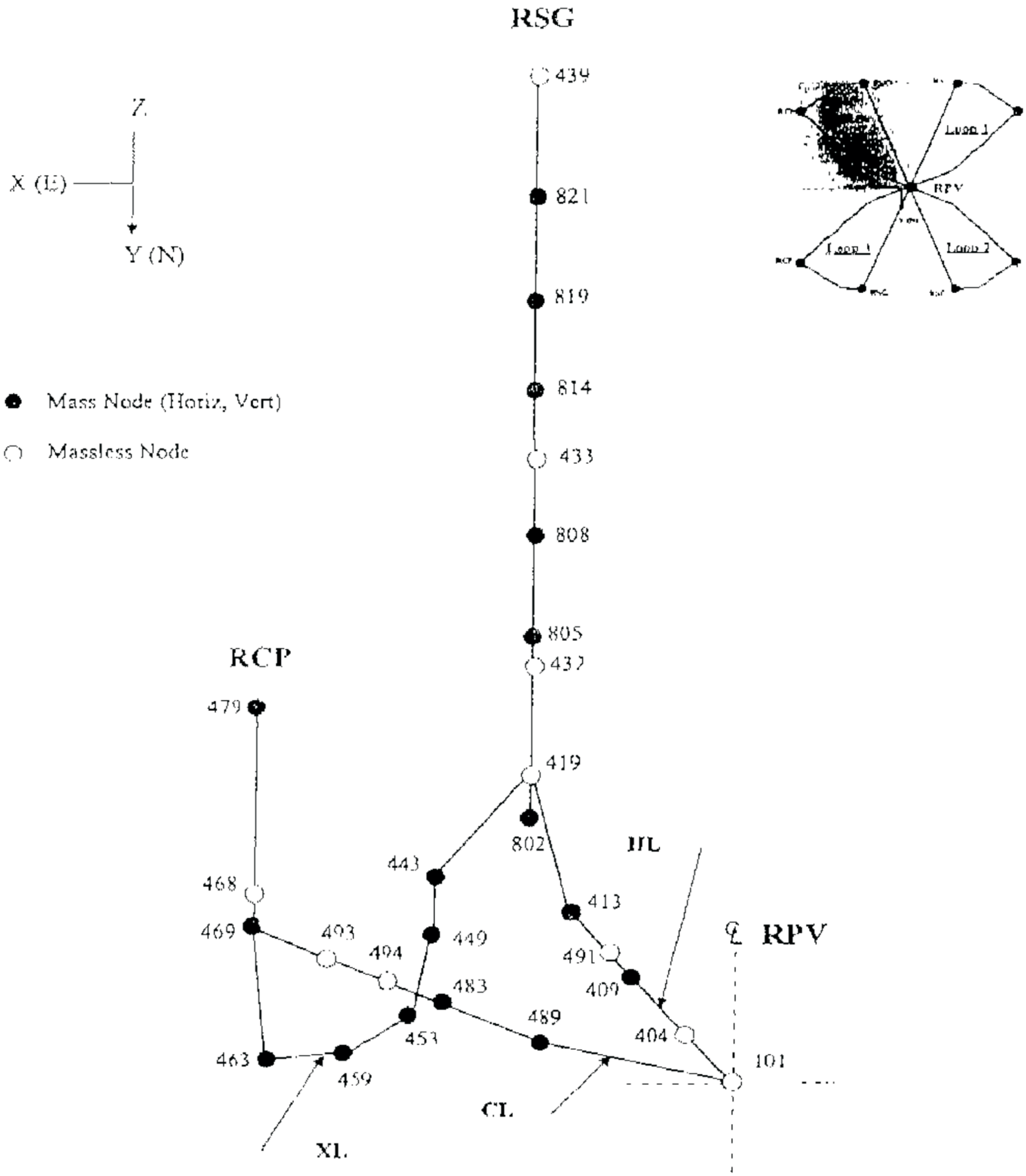


Note: Supports are not shown in this model

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 3

FIGURE 3.7-8f

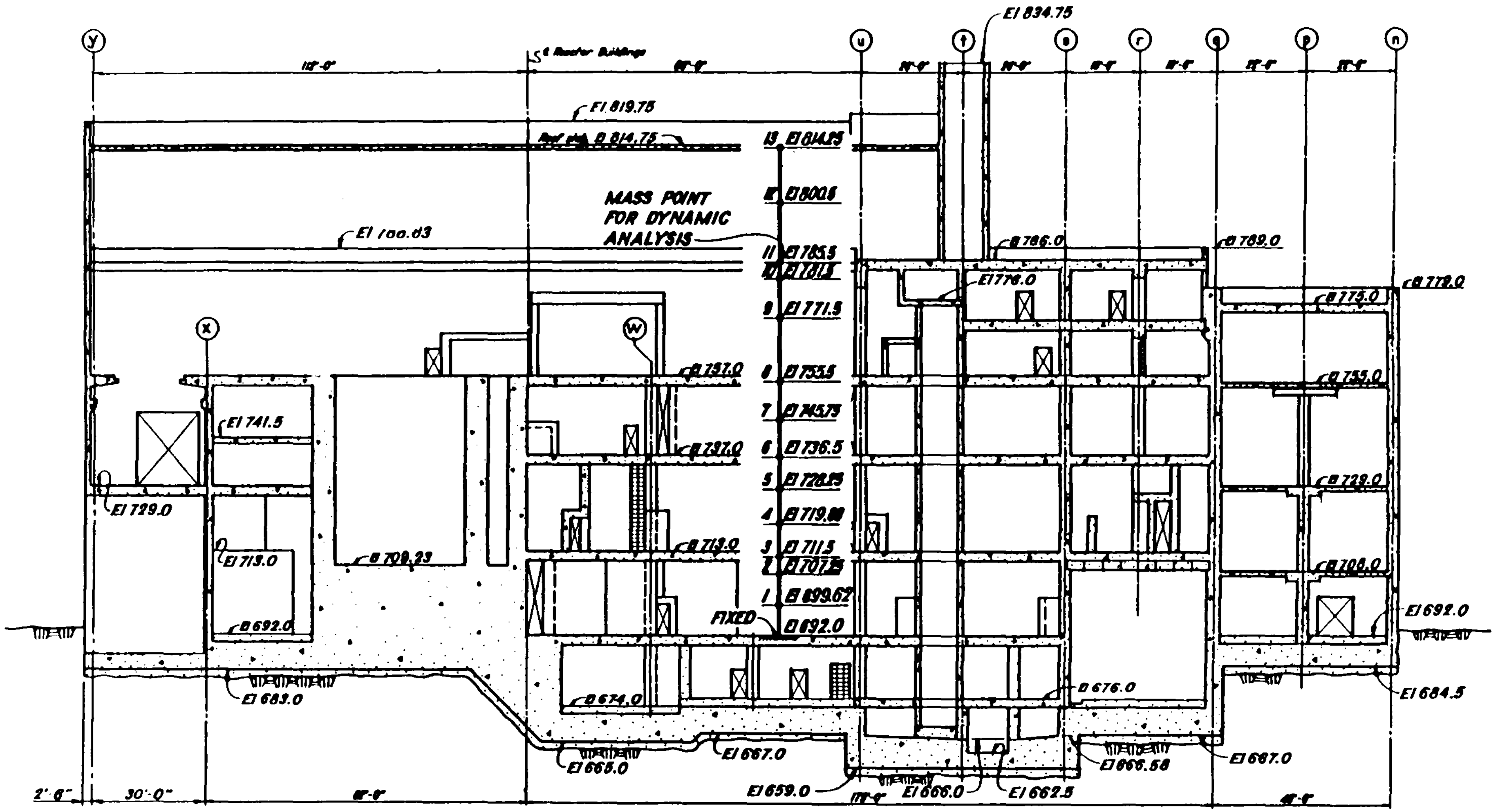


Note: Supports are not shown in this model.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

WBN Unit-1 Reactor Coolant Loop (RCL) Seismic Model Configuration - Loop 4

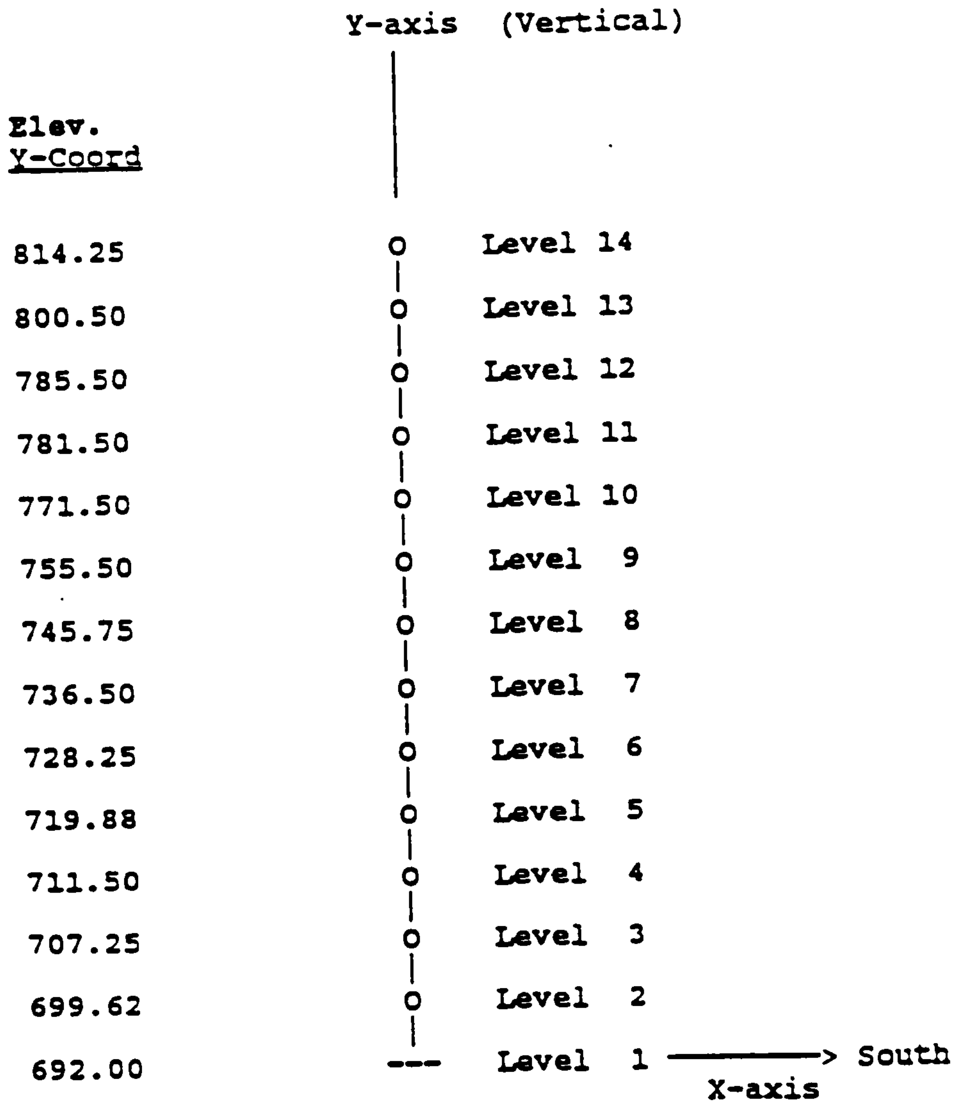
FIGURE 3.7-8g



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of Auxiliary-Control Building - Lumped Mass Model for Dynamic Analysis

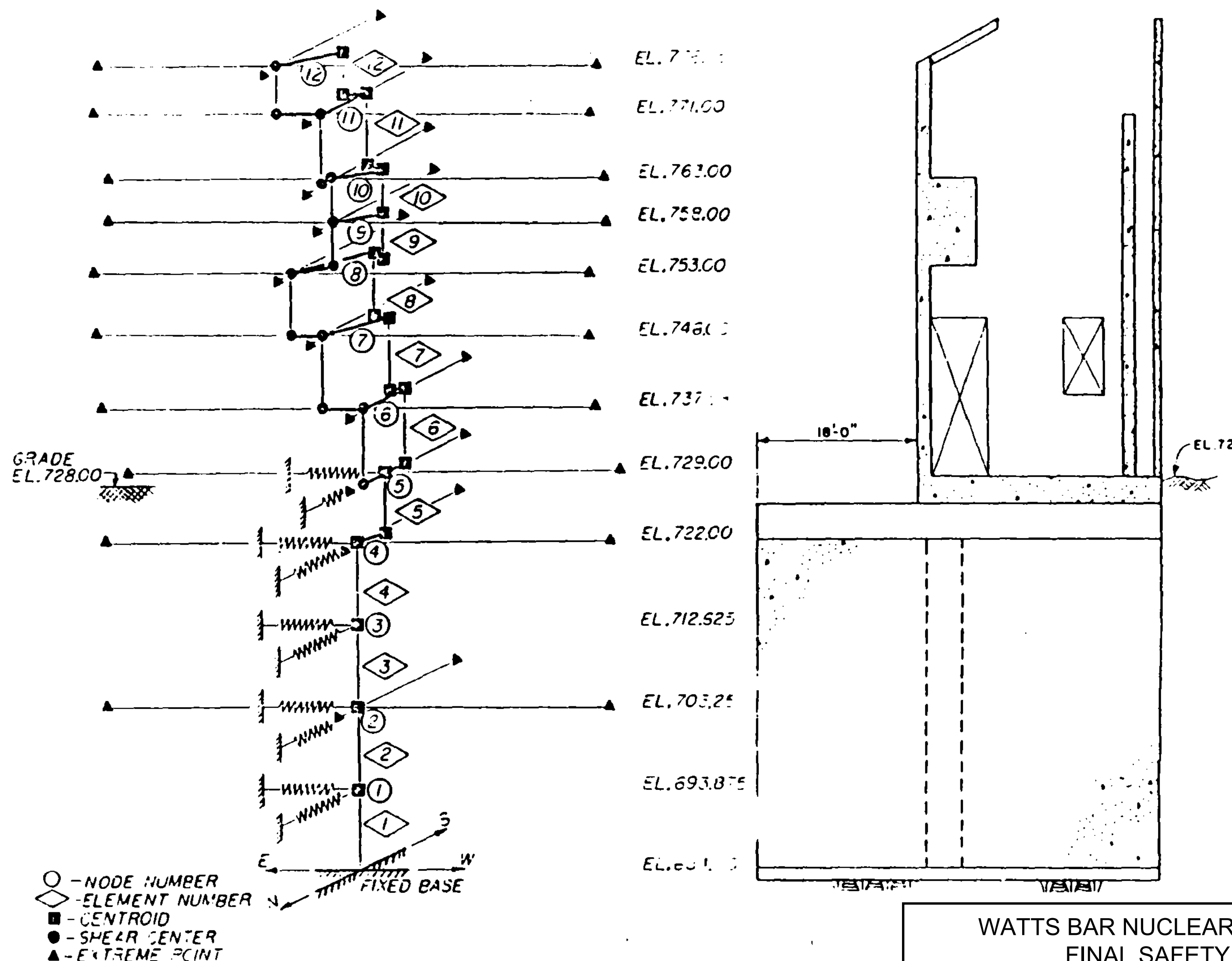
FIGURE 3.7-9



The coordinates of the Set B and Set C centers of mass and centers of rigidity are provided in Table 3.7-9A.

ACB Seismic Model
(Set B and Set C)

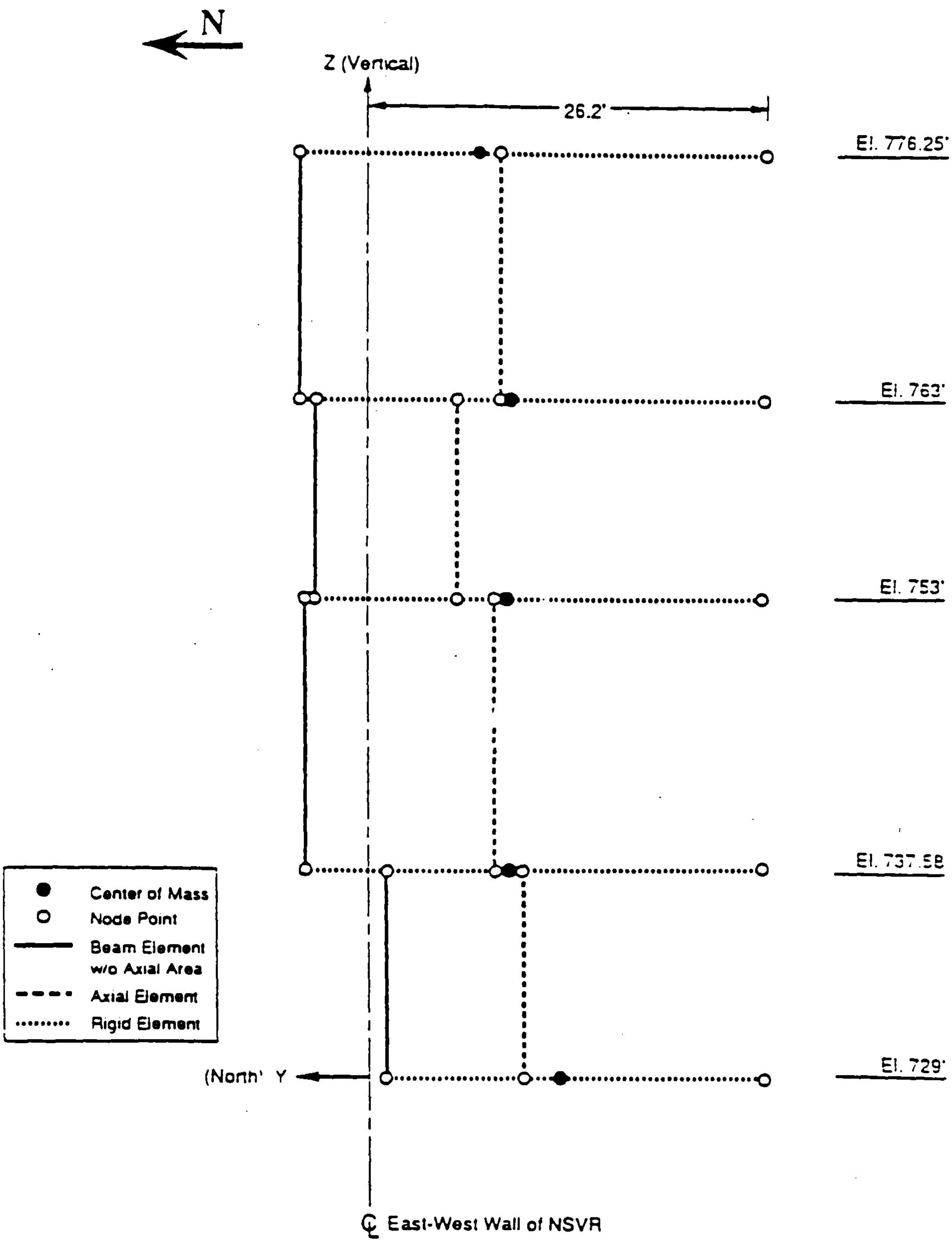
FIGURE 3.7-9A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of North Steam Valve Room and Lumped Mass Model for Seismic Analysis

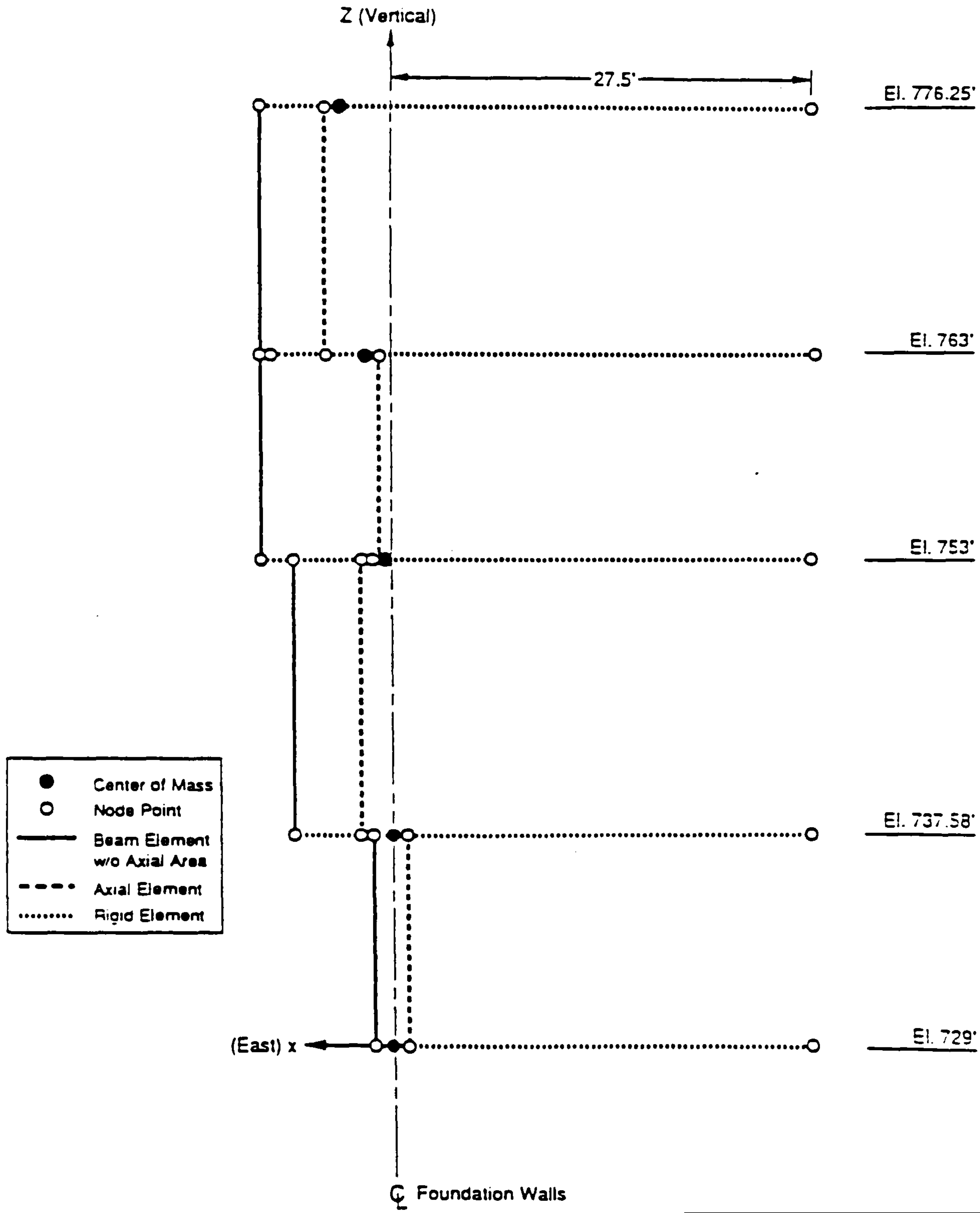
FIGURE 3.7-10



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lumped-Mass Stick Model for the NSVR Superstructure - XZ Plane

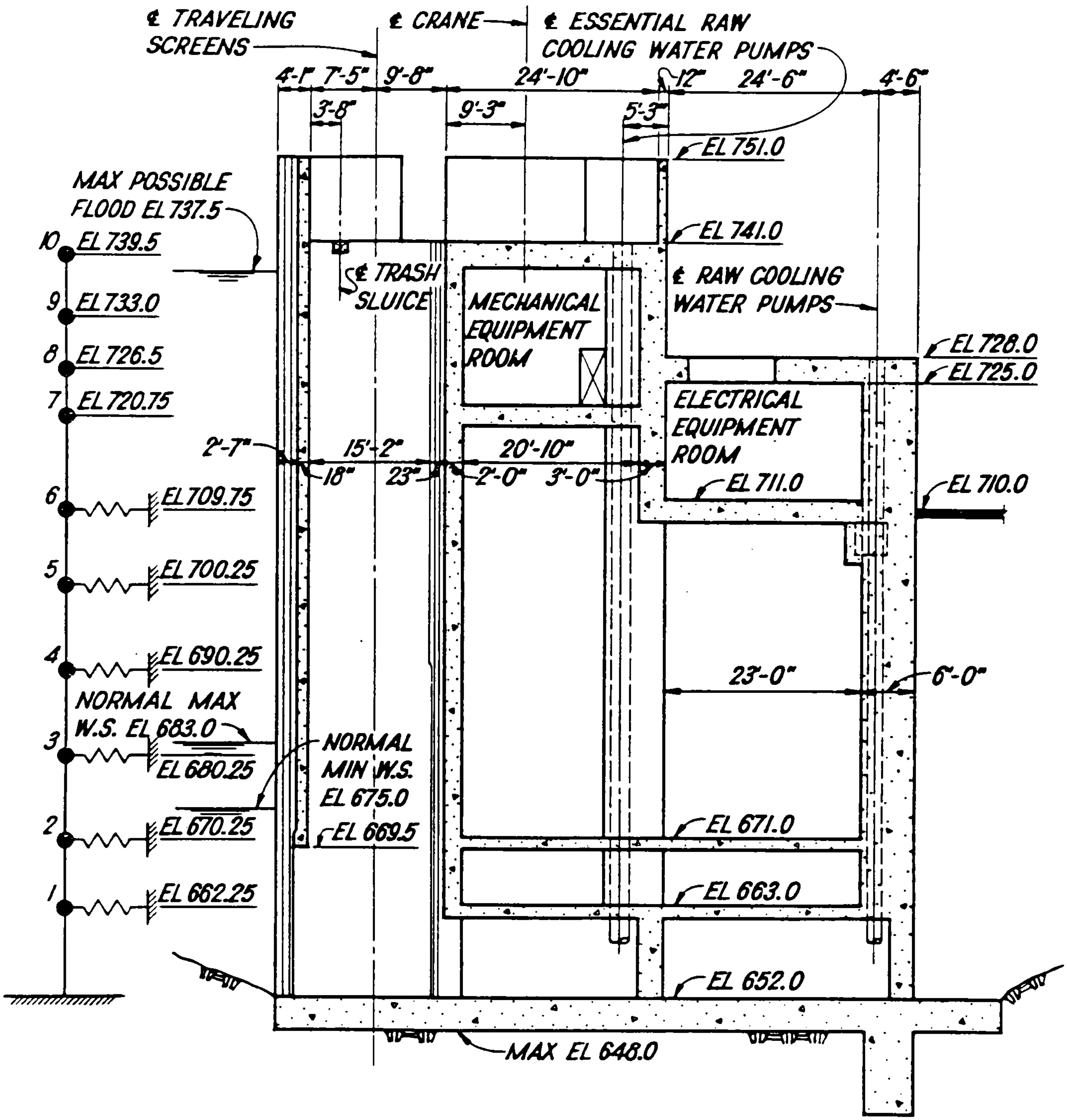
FIGURE 3.7-10A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lumped-Mass Stick Model for the NSVR Superstructure - XZ Plane

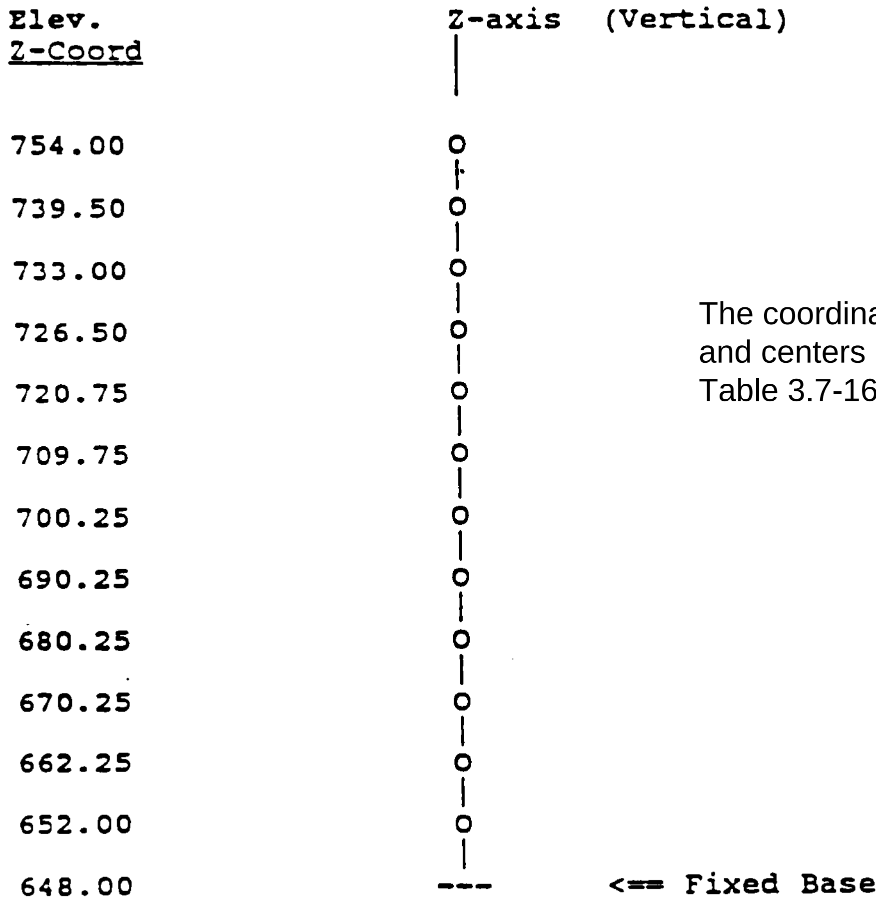
FIGURE 3.7-10B



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

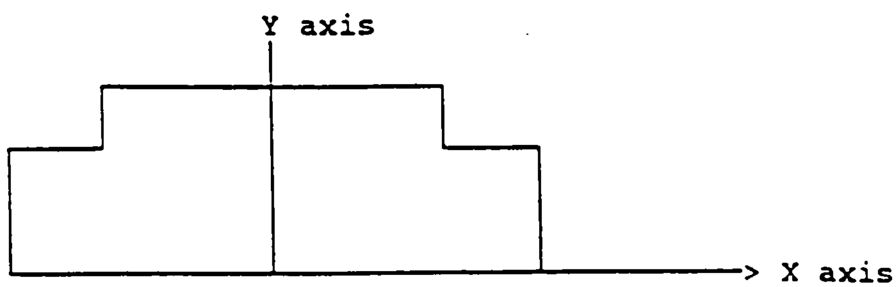
Sectional Elevation of
Intake Pumping Station
Lumped Mass Model for
Dynamic Analysis

FIGURE 3.7-11



The coordinates of centers of mass and centers of rigidity are presented in Table 3.7-16B.

Coordinate System:

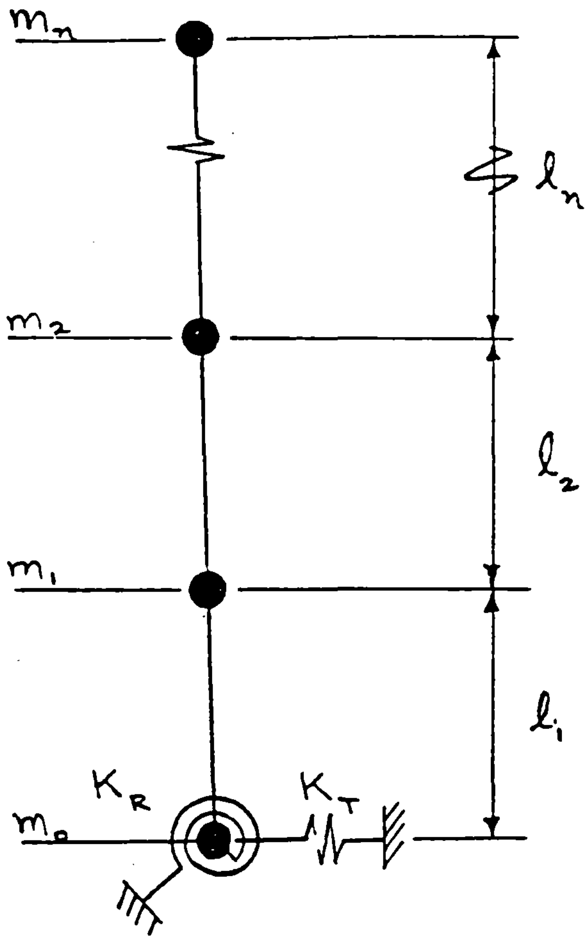


X Axis - Longitudinal direction
Y axis - Transverse direction
Z axis - Vertical direction

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

IPS Seismic Model

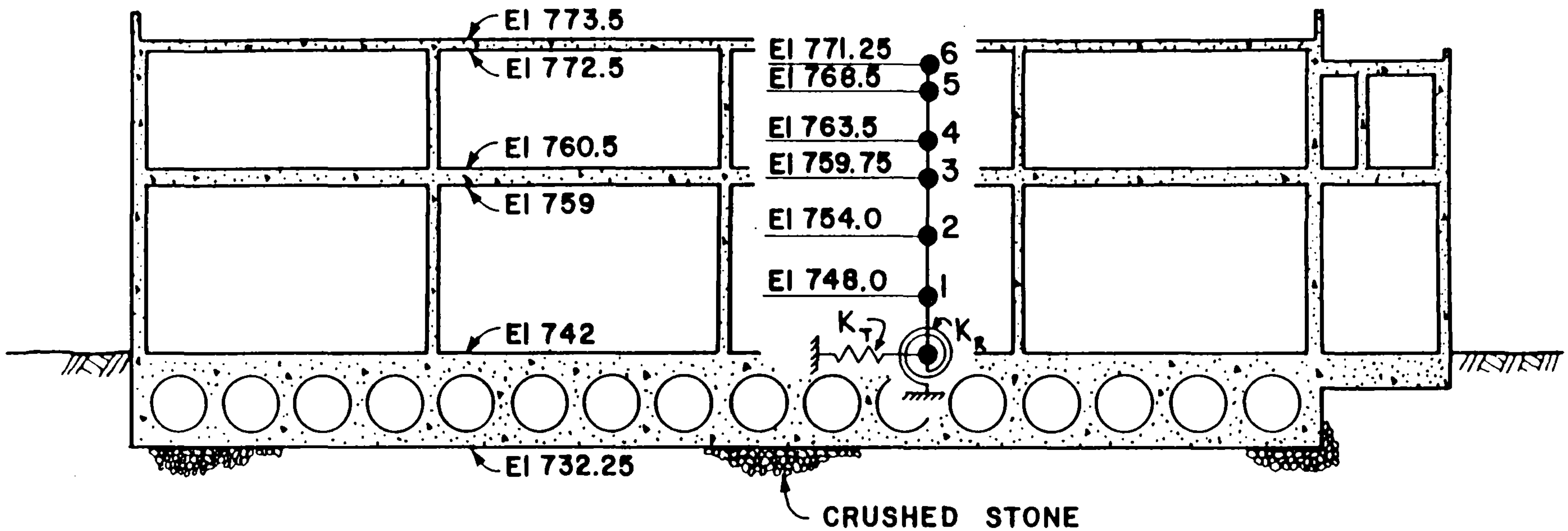
FIGURE 3.7-11A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Mathematical Model
for Soil Structure Interaction

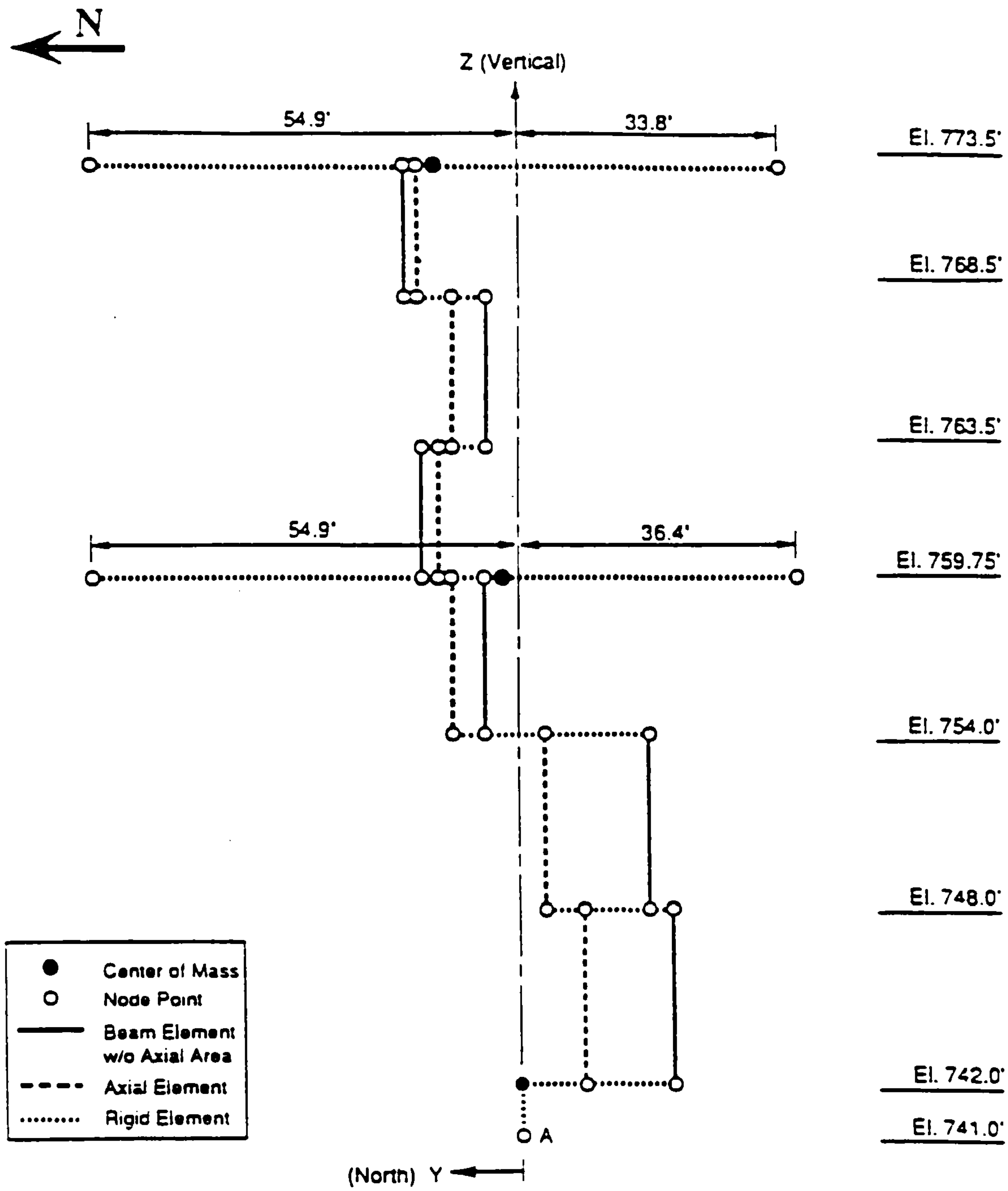
FIGURE 3.7-12



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sectional Elevation of
Diesel Generator Building
Lumped Mass Model for Dynamic Analysis

FIGURE 3.7-13

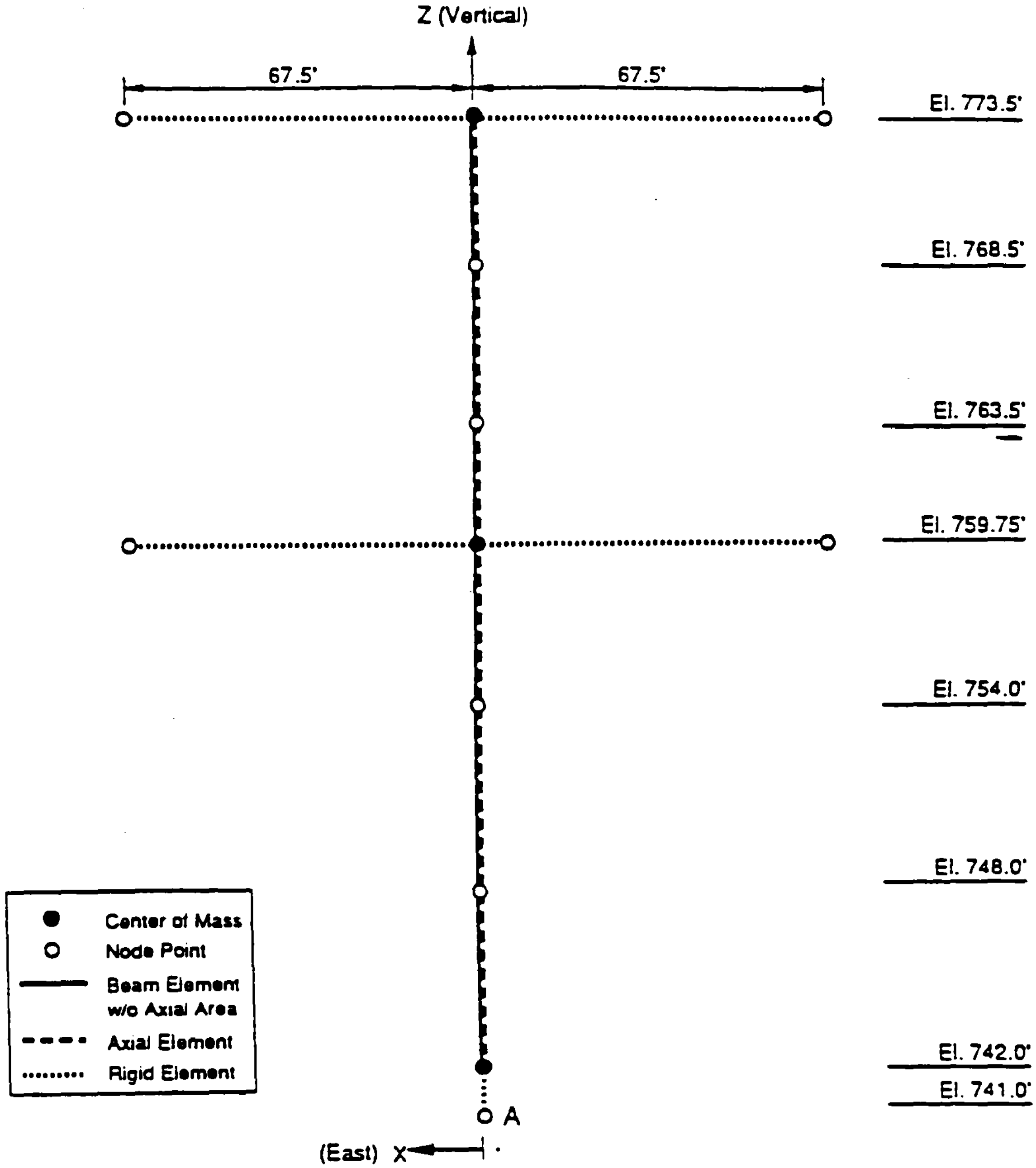


A: Geometric Center of Basemat

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Seismic Analysis Model for
Diesel Generator Building
XZ Plane

FIGURE 3.7-13A

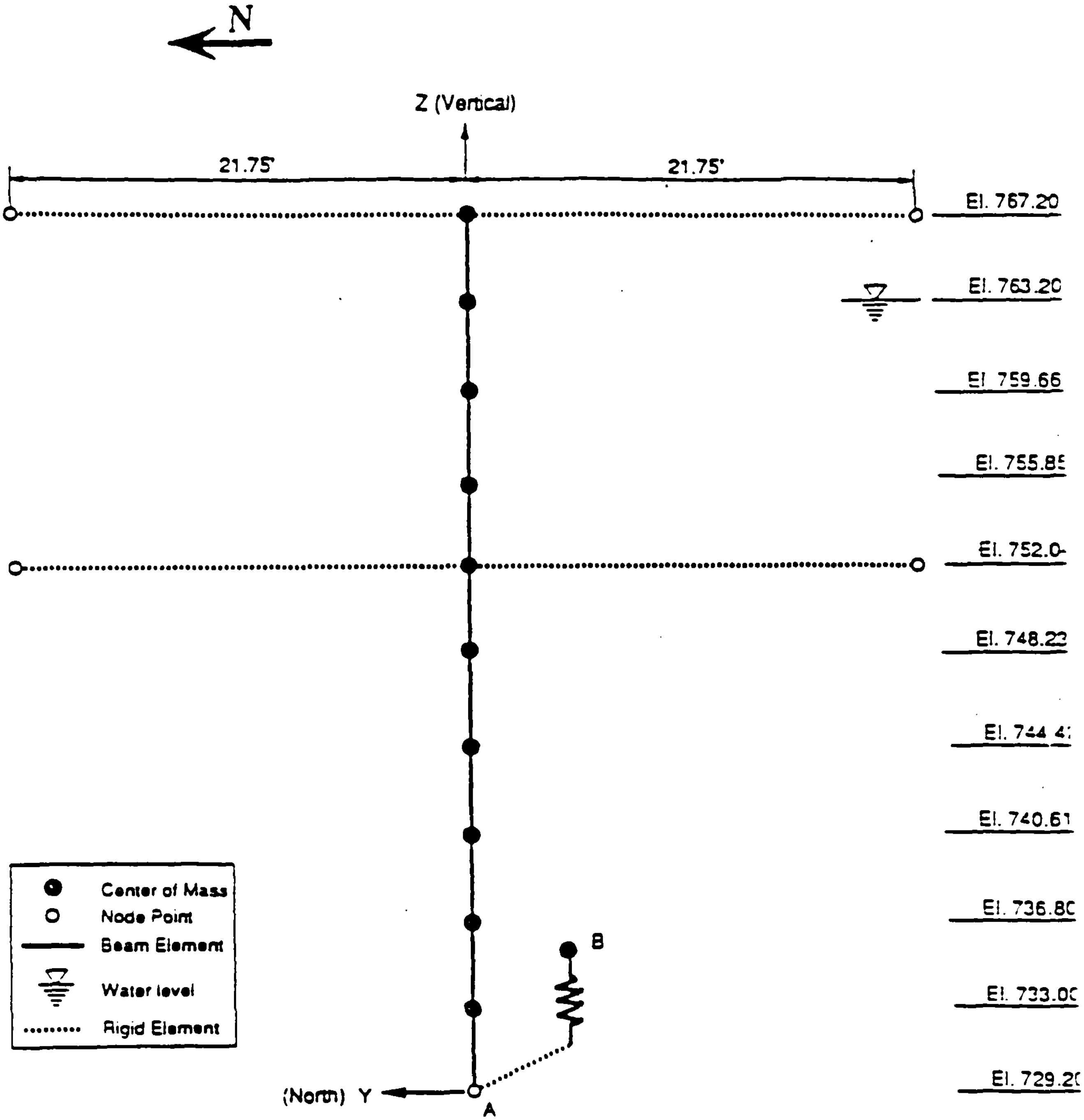


A: Geometric Center of Basemat

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

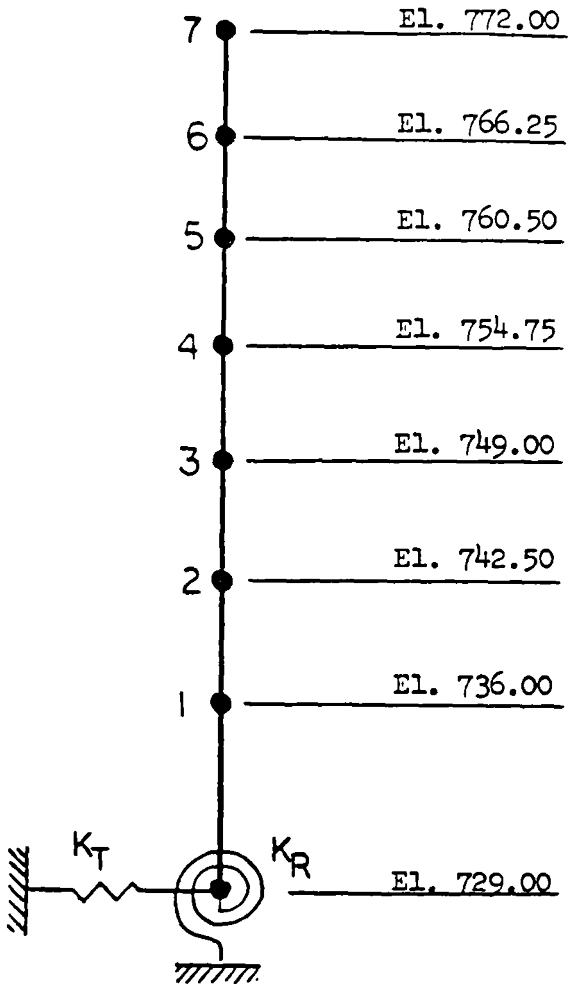
Seismic Analysis Model for
Diesel Generator Building
XZ Plane

FIGURE 3.7-13B



A: Geometric Center of Basemat
B: Hydrodynamic Vertical Oscillator

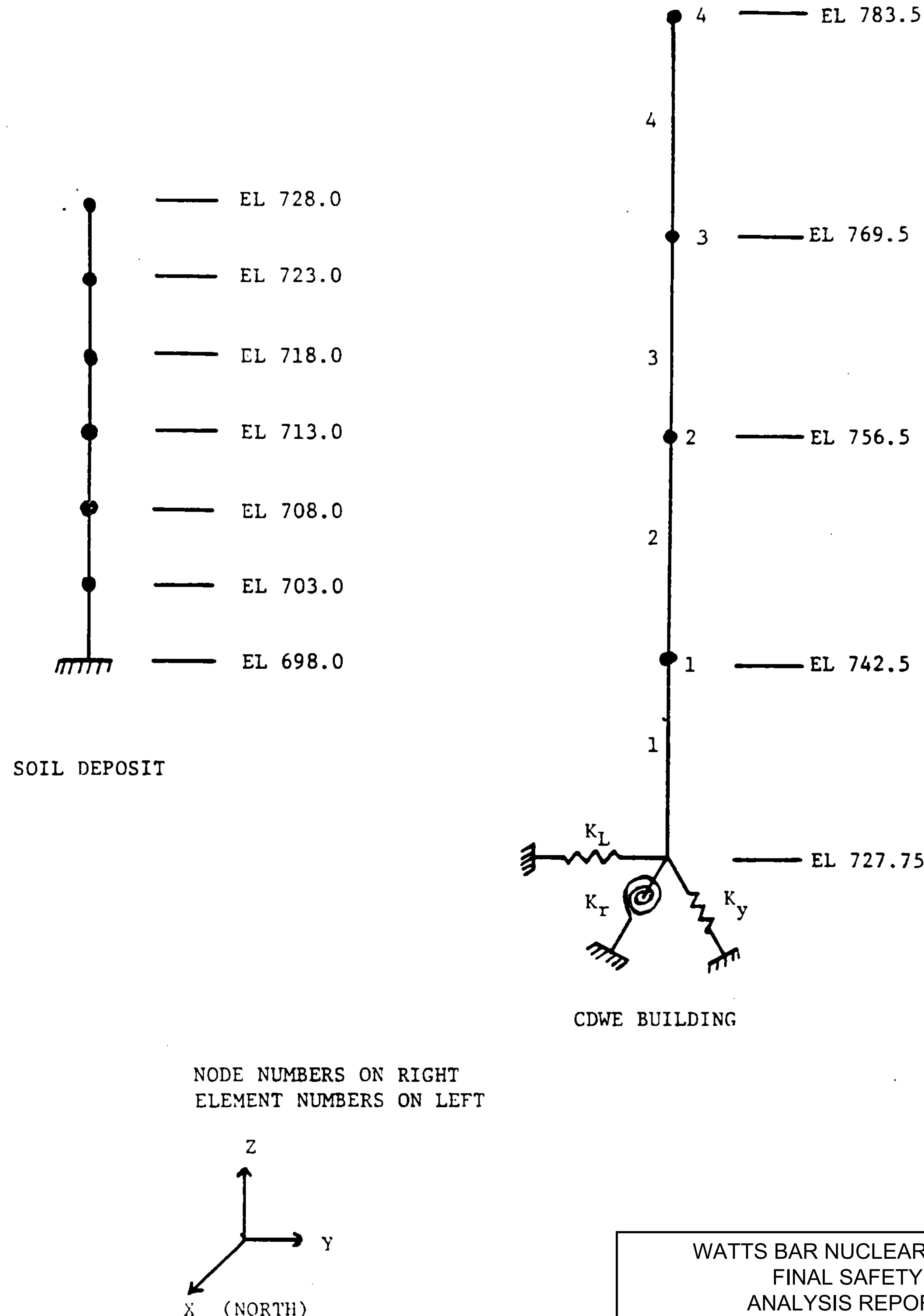
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Lumped-Mass Stick Model for Refueling Water Storage Tank FIGURE 3.7-13C



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Mathematical Model
for Dynamic Analysis of the
Waste Packaging Area

FIGURE 3.7-14



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Condensate Demineralizer Waste Evaporator Building - Lumped Models for Normal Mode Analysis

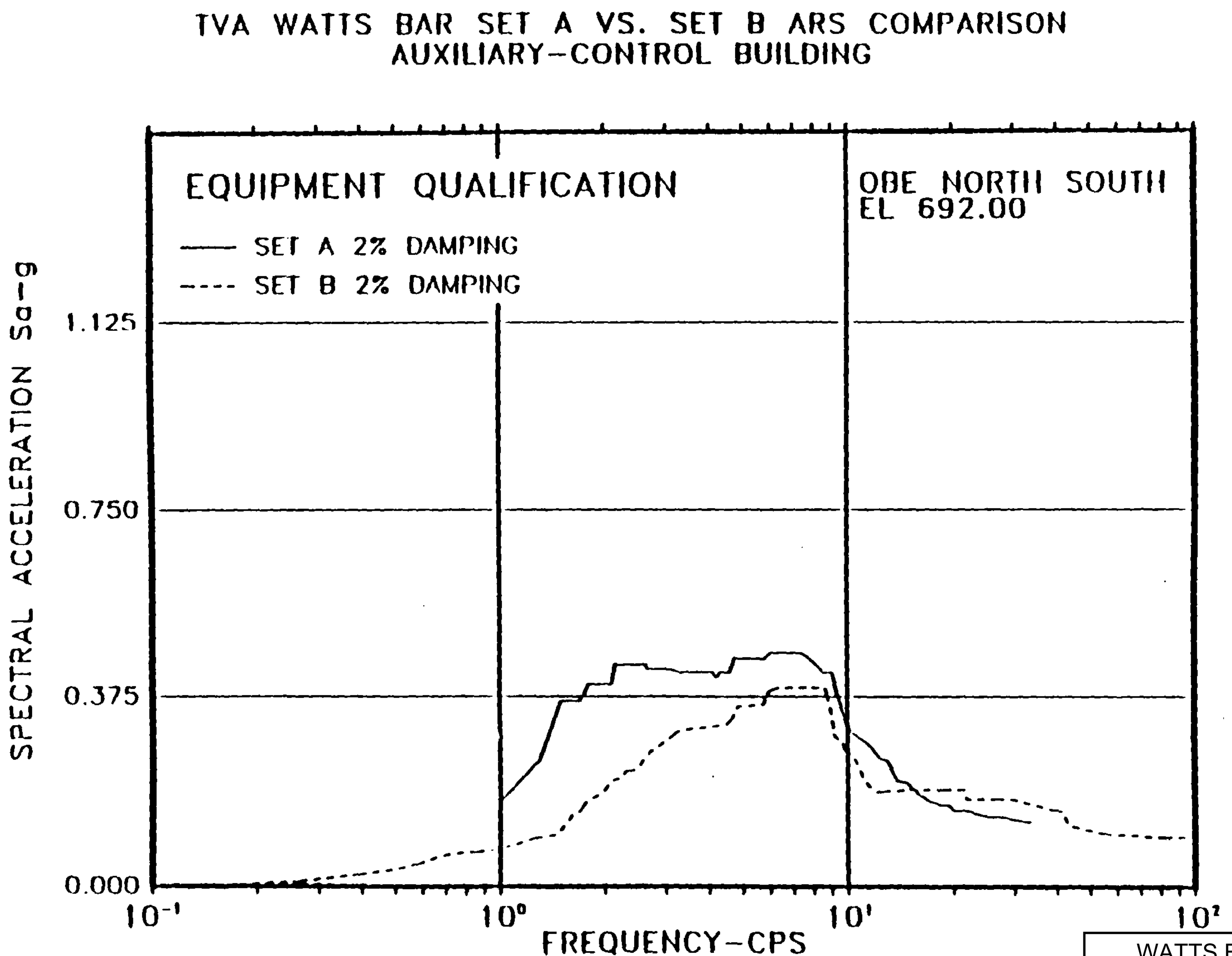
FIGURE 3.7-15A

Figure 3.7-15B

Deleted

Figure 3.7-15C

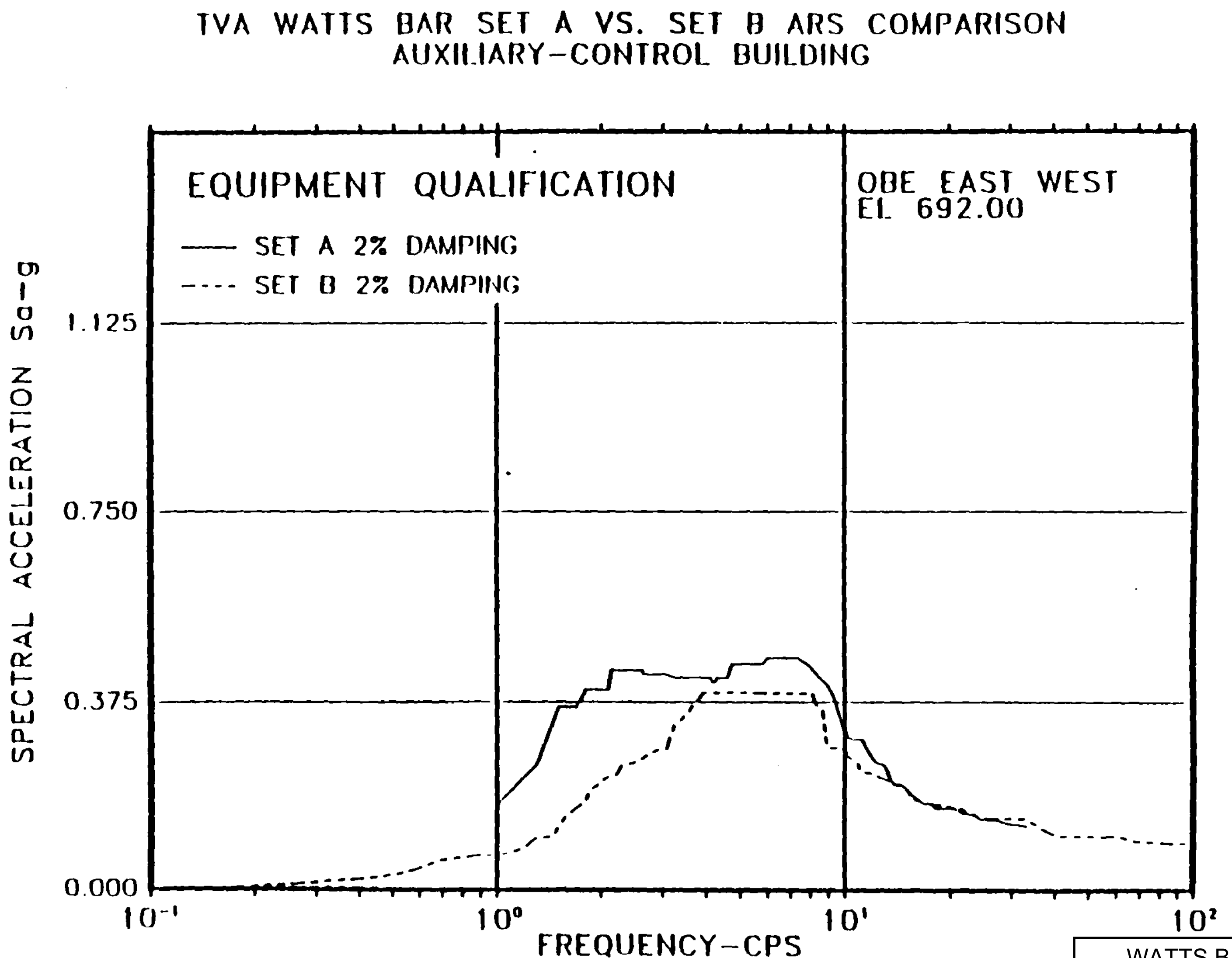
Deleted



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Set A VS. Set B

FIGURE 3.7-15D

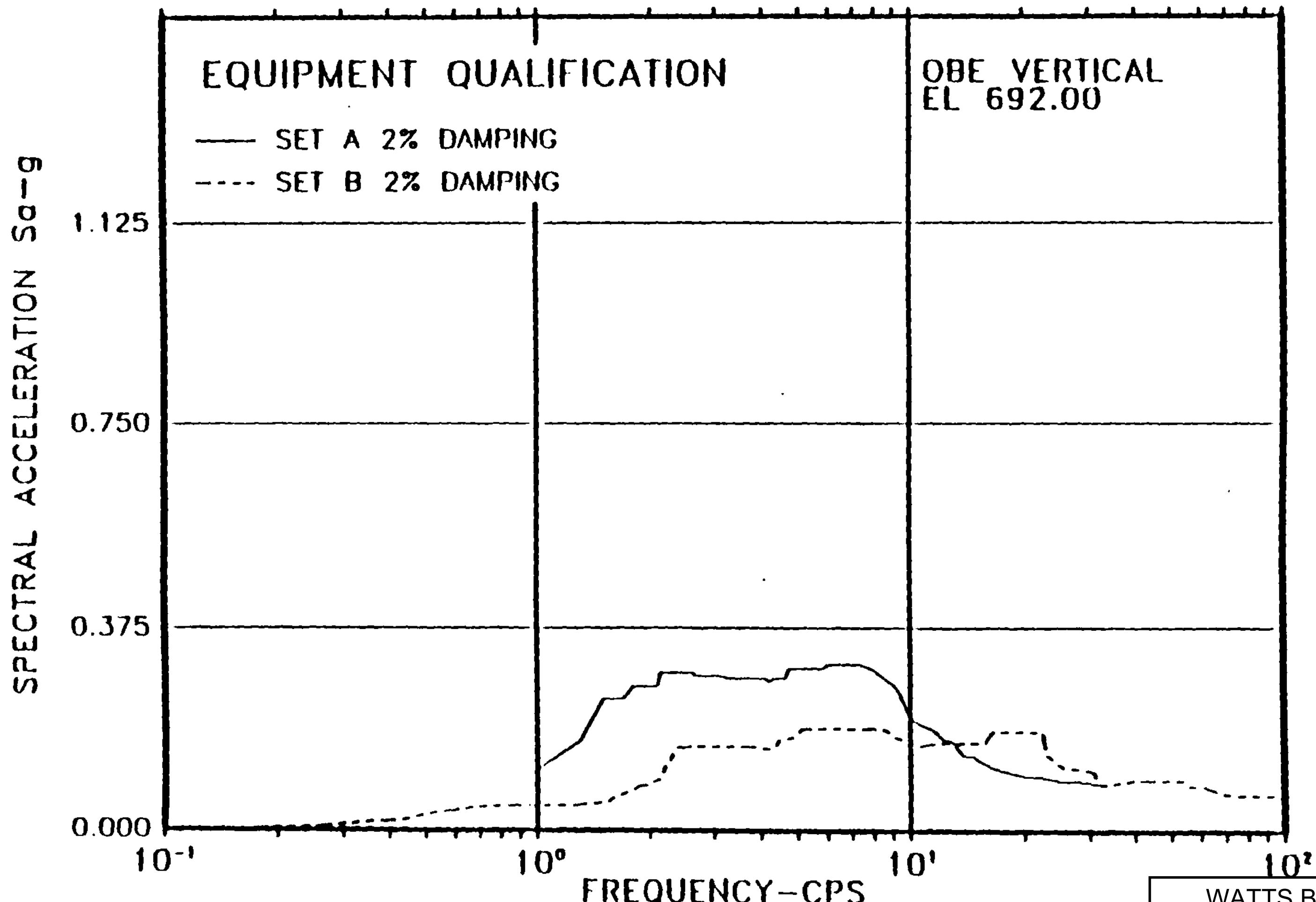


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

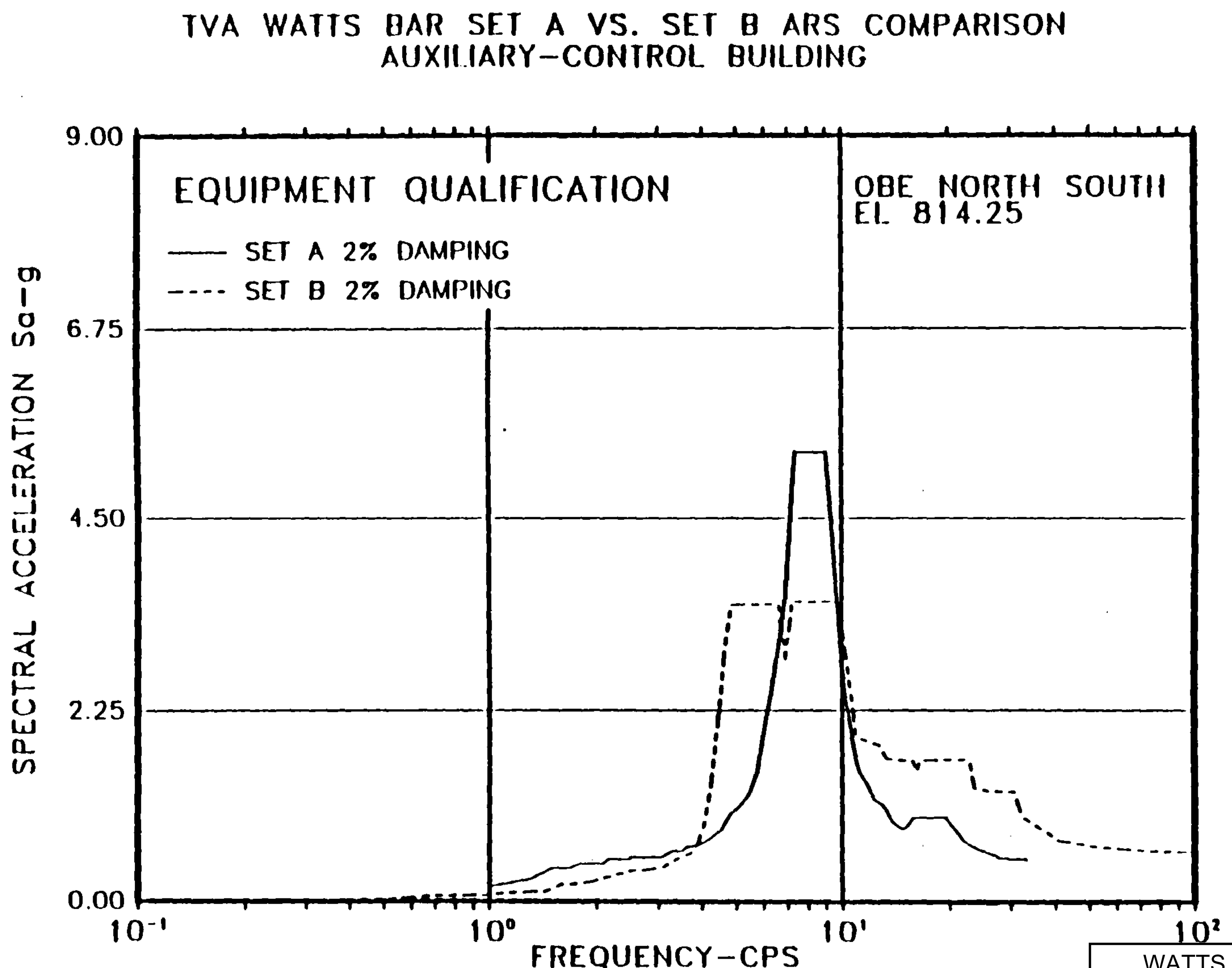
Set A VS. Set B

FIGURE 3.7-15E

TVA WATTS BAR SET A VS. SET B ARS COMPARISON
AUXILIARY-CONTROL BUILDING



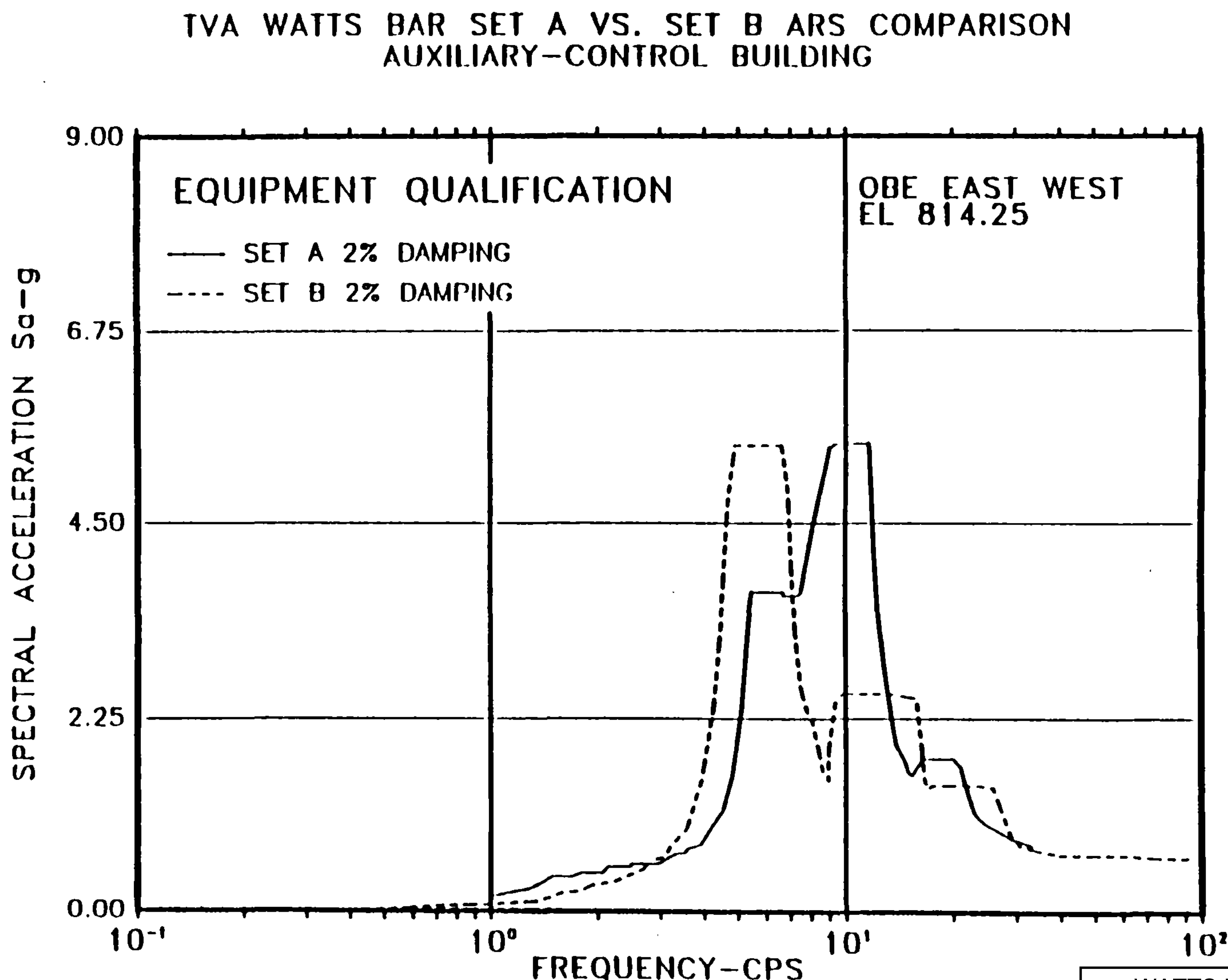
WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Set A VS. Set B FIGURE 3.7-15F



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Set A VS. Set B

FIGURE 3.7-15G

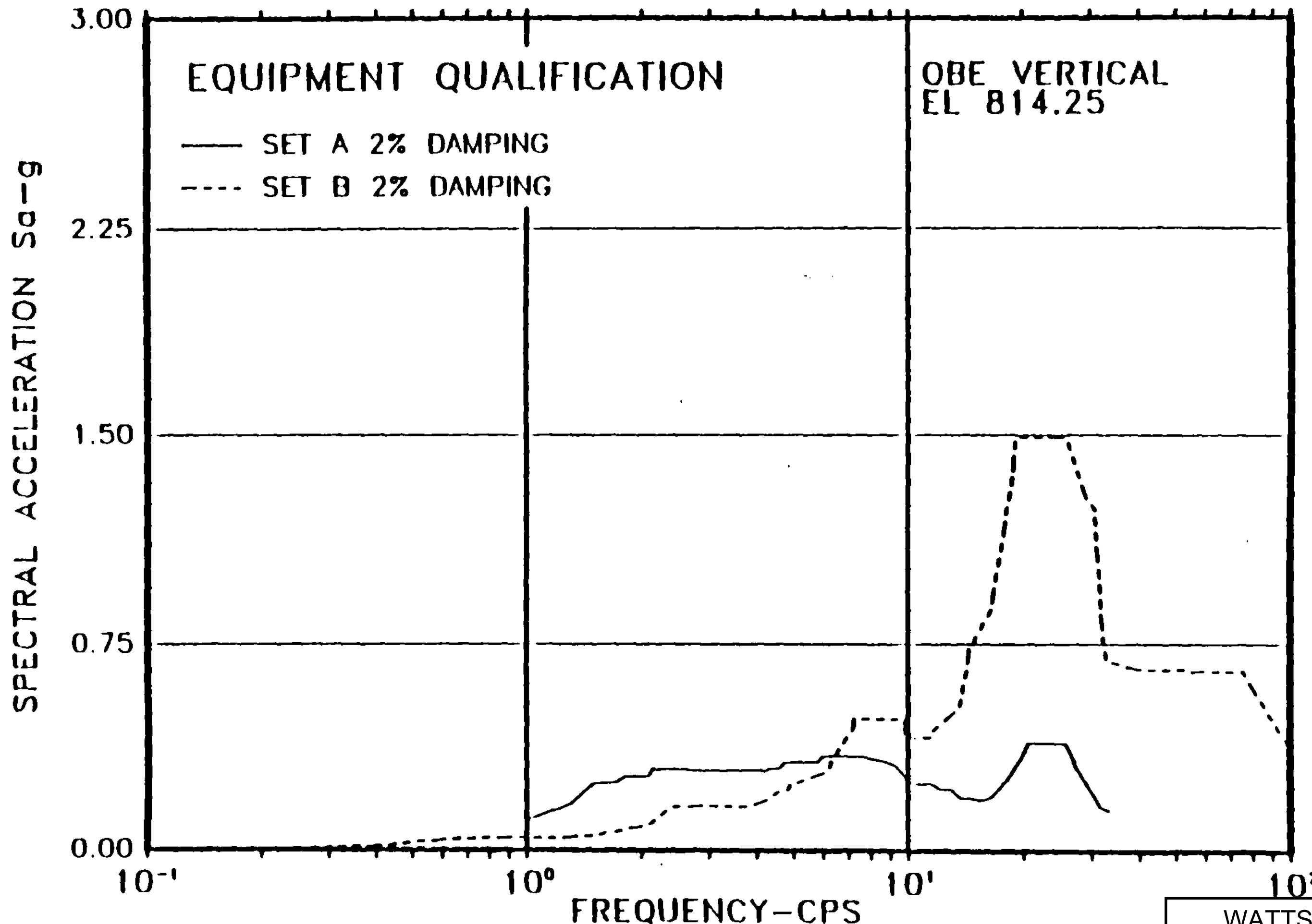


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Set A VS. Set B

FIGURE 3.7-15H

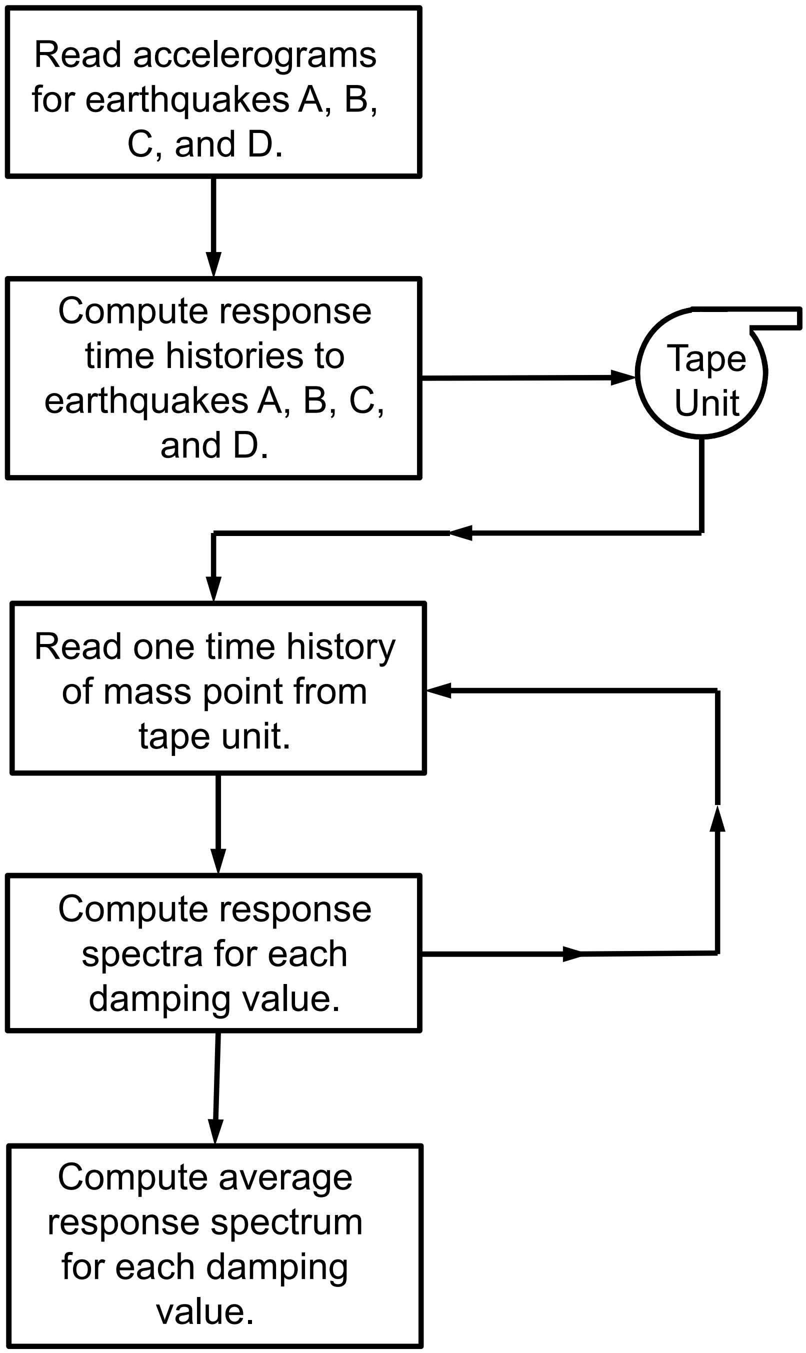
TVA WATTS BAR SET A VS. SET B ARS COMPARISON
AUXILIARY-CONTROL BUILDING



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Set A VS. Set B FIGURE 3.7-15I

FIGURE 3.7-16 THRU FIGURE 3.7-36

DELETED



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Flow Chart for
Development of Floor
Response Spectra

FIGURE 3.7-37

FIGURE 3.7-38

DELETED

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-39

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-40

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-41

CAD MAINTAINED DRAWING

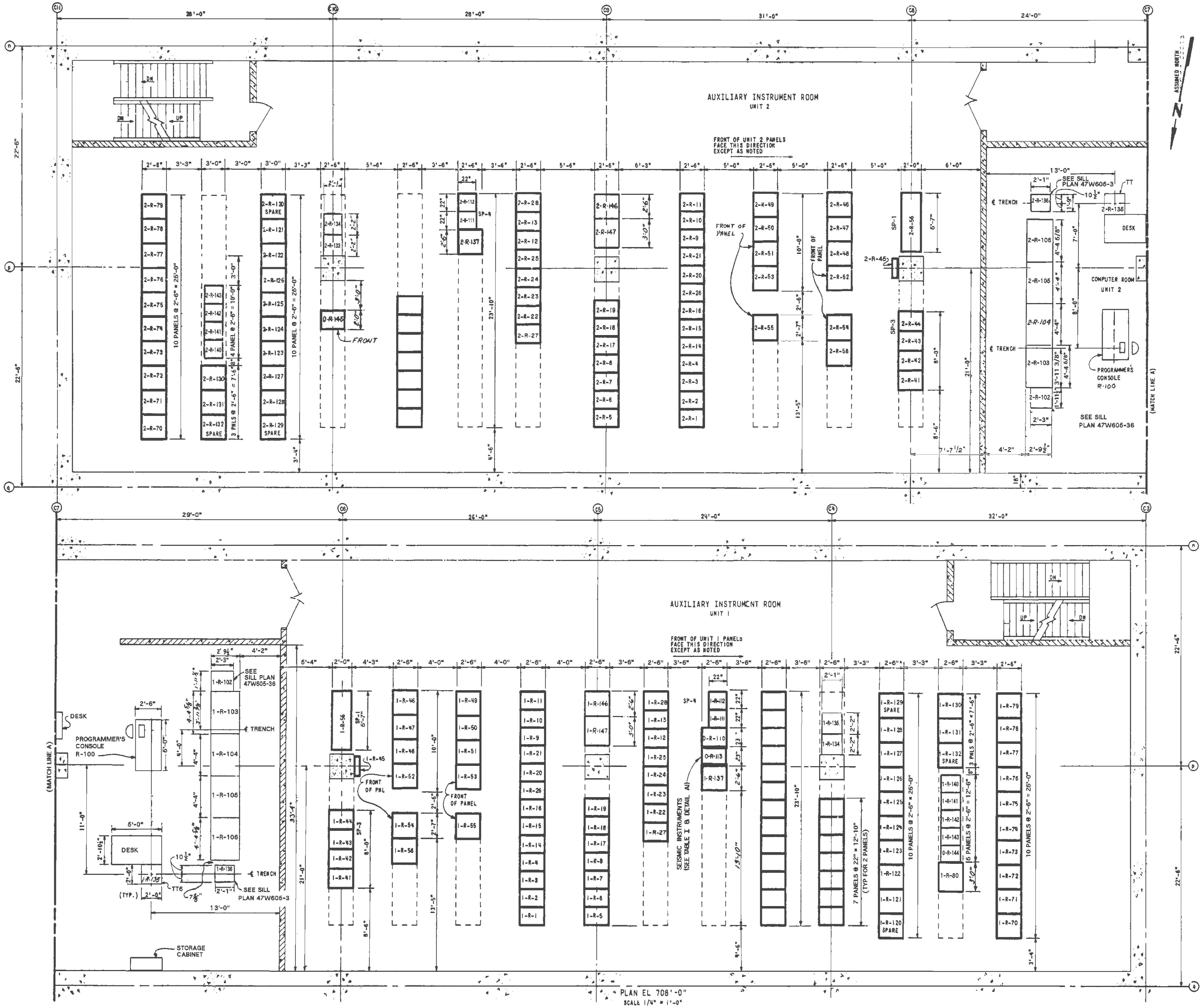
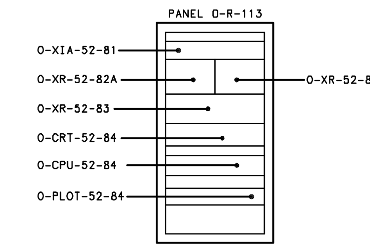


TABLE I
SEISMIC INSTRUMENTATION

INSTRUMENT NO.	FSAR SECTION NO.	FSAR ITEM NO.	TVA IDENTIFICATION NO.	INSTRUMENT TYPE
	3.7.4.2		O-XIA-52-81	ALARM PANEL
			O-XR-52-82A	RECORDER PANEL
			O-XR-52-82B	RECORDER PANEL
			O-XR-52-83	RECORDER PANEL
			O-CRT-52-84	LCD PANEL
			O-CPU-52-84	CENTRAL CONTROLLER
			O-PLOT-52-84	PRINTER PANEL



DET A1
SEISMIC INSTRUMENT LOCATION

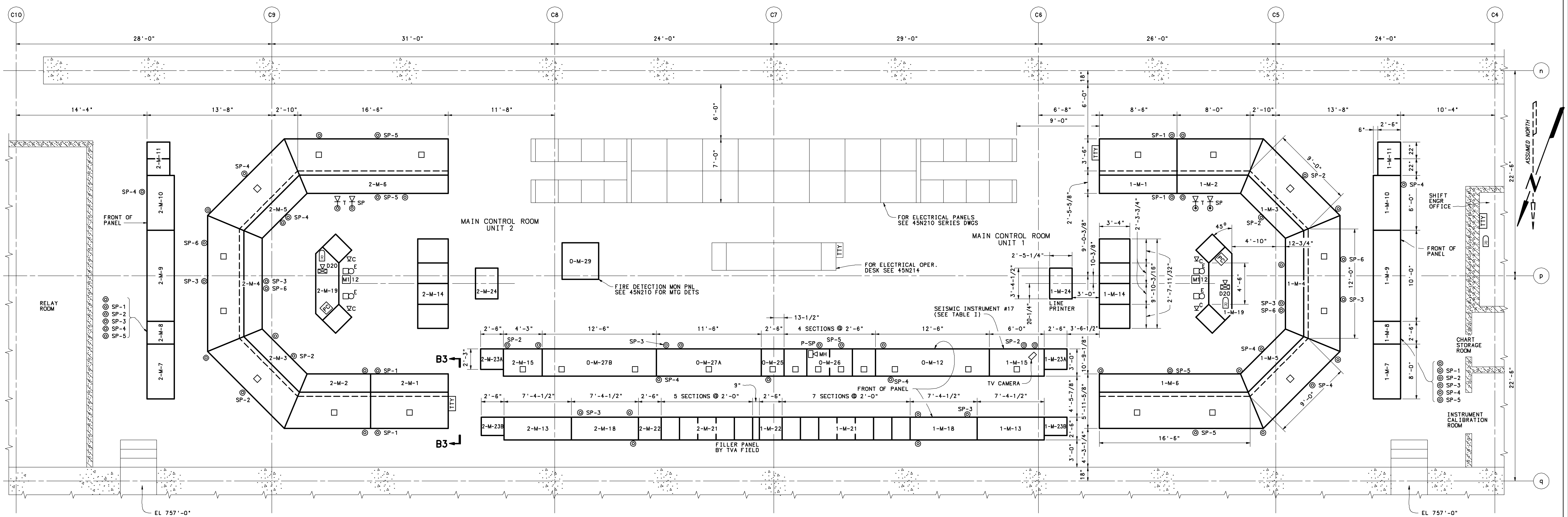
NOTES:
1. THIS DRAWING WAS ADAPTED FROM DRAWING NO. 47W605-2, REVISION 8.
2. REVISIONS OF DRAWING NO. 47W605-2 DO NOT REQUIRE UPDATING UNLESS EQUIPMENT IN TABLE I IS AFFECTED.

COMPANION DRAWINGS:
47W652-2,3,4
47N653-1,2,3
47B652

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

CONTROL BUILDING UNITS 1 & 2
SEISMIC INSTRUMENTATION
LOCATION OF SEISMIC INSTRUMENTS
AND PERIPHERAL EQUIPMENT
TVA DWG NO. 47W652-1 RD
FIGURE 3.7-42

CAD MAINTAINED DRAWING



PLAN EL 755'-0"

TABLE I
SEISMIC INSTRUMENTATION

INSTRUMENT NO.	FSAR SECTION NO.	FSAR ITEM NO.	TVA IDENTIFICATION NO.	INSTRUMENT TYPE	ORIENTATION
17	3.7.4.2	8	XA-55-15B	ANNUNCIATOR PANEL (SEE DRAWING 47W605-56)	

NOTES:

1. THIS DRAWING WAS ADAPTED FROM DRAWING NO. 47W605-1, REVISION 7.
2. REVISIONS OF DRAWING NO. 47W605-1 DO NOT REQUIRE UPDATING OF THIS DRAWING UNLESS EQUIPMENT IN TABLE I IS AFFECTED.

COMPANION DRAWINGS:
47W652-1,3,4
47N653-1,2,4
47B652

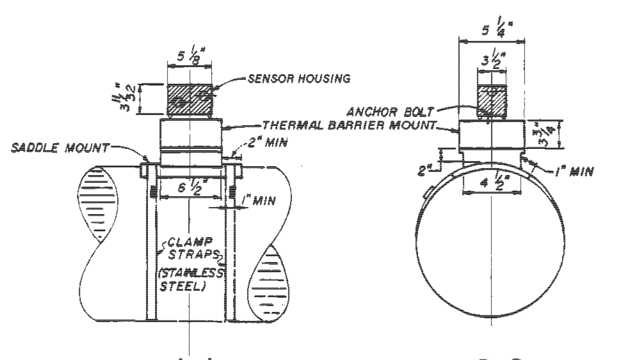
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

CONTROL BUILDING-UNITS 1 & 2
SEISMIC INSTRUMENTATION
LOCATION OF SEISMIC INSTRUMENTS
AND PERIPHERAL EQUIPMENT
TVA DWG NO. 47W652-2 RE
FIGURE 3.7-43

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 3.7-44

CAD MAINTAINED DRAWING



A-A
ALL TOLERANCES ± 1/16"
NTS

B-B
ALL TOLERANCES ± 1/16"
NTS

SADDLE MOUNT FOR PAR4CO-2 TRIAXIAL PEAK ACCELEROGRAPH

MOUNTING INSTRUCTIONS:

1. ALL PLATES OF THE SADDLE MOUNT ARE TO BE 1/2-INCH THICK STAINLESS STEEL.
2. LONG AXIS OF ACCELEROGRAPH UNIT IS TO BE ALIGNED WITH LONGITUDINAL AXIS OF PIPE.
3. SADDLE MOUNT AND STRAPS ARE TO BE ATTACHED DIRECTLY TO BARE PIPE.
4. THERMAL BARRIER MOUNT IS TO BE ENGDAHL ENTERPRISES MODEL NO. TBM400 AND WILL BE PROCURED BY THE FIELD.
5. SADDLE MOUNT IS TO HAVE FOUR 1/4" TAPPED THREAD HOLES IN A 4" x 4" SQUARE PATTERN. THERMAL BARRIER MOUNT IS TO BE ATTACHED TO SADDLE MOUNT WITH 1/4"-20UNC x 4" OR 4-1/4" STAINLESS STEEL BOLTS.

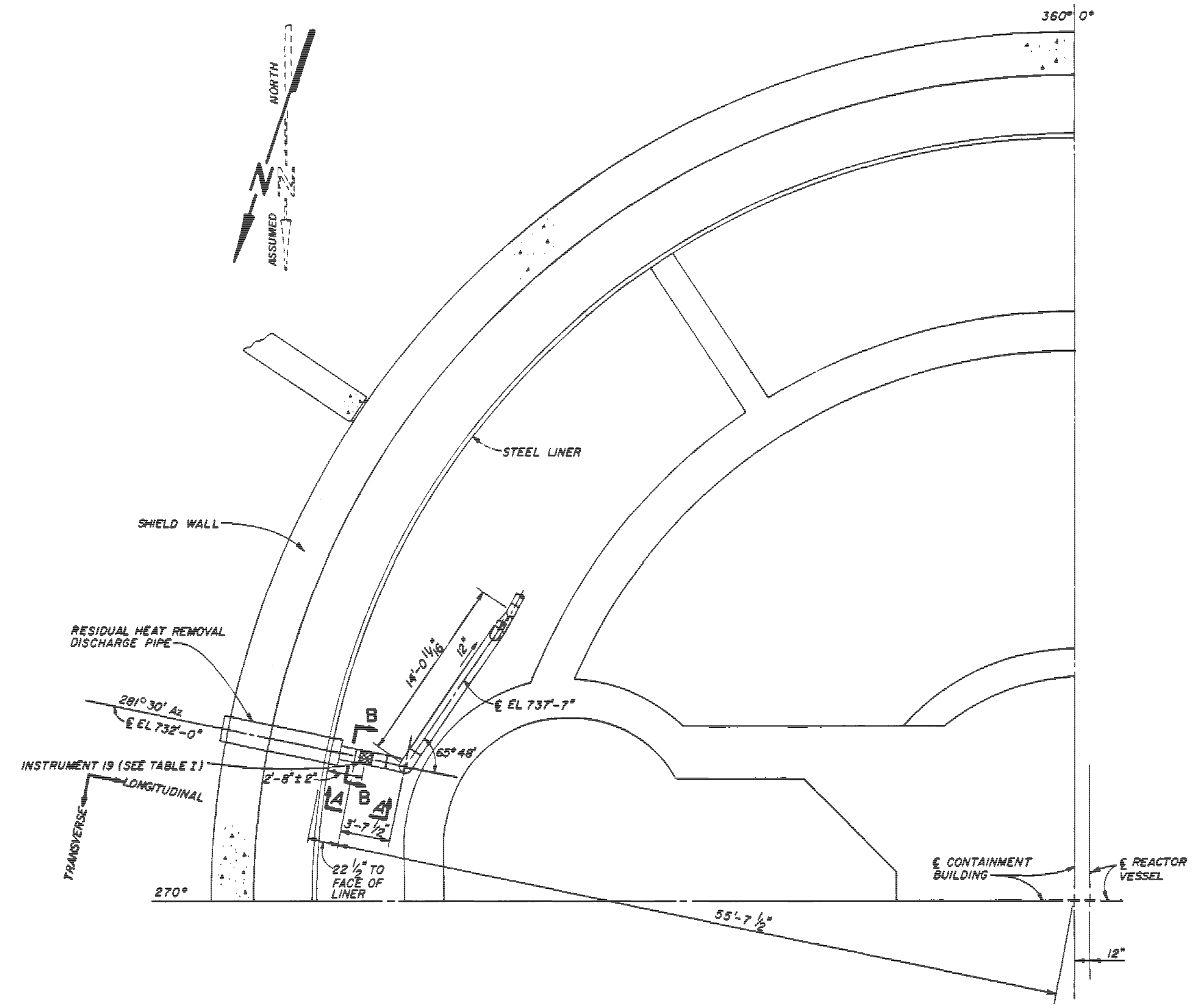


TABLE I
SEISMIC INSTRUMENTATION

INSTRUMENT NO.	FSAR SECTION NO.	FSAR ITEM NO.	TVA IDENTIFICATION NO.	INSTRUMENT TYPE
19			XR-52-76B ABANDONED IN PLACE	TRIAXIAL PEAK ACCELEROGRAPH PAR400-2 (SEE SECT. A-A & B-B FOR MOUNTING INSTRUCTIONS)

NOTES:

1. THIS DRAWING WAS ADAPTED FROM DRAWING NO. 47W432-3, REVISION 23.
2. REVISIONS OF DRAWING NO. 47W432-3 DO NOT REQUIRE UPDATING OF THIS DRAWING UNLESS EQUIPMENT IN TABLE I IS AFFECTED.
3. DESIGN INPUT DOCUMENTS; NRC REG GUIDE 1.12 AND ANSI/ANS-2.2, SECTION 4.1. THE LOCATION OF INSTRUMENT 19 DEVIATES FROM REG GUIDE 1.12 AS DOCUMENTED IN A MEMORANDUM DATED 3-5-79 (B45 880304 252).

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR UNIT 1
SEISMIC INSTRUMENTATION
LOCATION OF SEISMIC INSTRUMENTS
AND PERIPHEAL EQUIPMENT
TVA DWG NO. 47W652-4 RF
FIGURE 3.7-45