3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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3.0 - DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 CRITERION 1 - QUALITY STANDARDS AND RECORDS

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

CONFORMANCE

Discussion of the quality standards for those systems which are essential to the prevention of accidents which could affect the public health and safety or lead to a mitigation of their consequences are presented in appropriate sections of the Safety Analysis Report. (See chapters 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, 9.0, 10.0, 11.0, and 12.0.)

For example, components of the engineered safeguards systems are designed and fabricated in accordance with established codes and/or standards as required to assure that their quality is in keeping with the safety function of the component.

The mechanical components of the facility have been classified according to their importance in the prevention and mitigation of accidents which could cause undue risk to the health and safety of the public. These classifications are described in section 3.2, along with the codes to which each component was designed. The design criteria for mechanical components are listed in table 3.2-1. Those components listed as ANS Safety Class 1, 2a, 2b, or 3 are designed and manufactured to the quality standards required by Criterion 1.

The tests and inspections applied to the systems are included in the appropriate sections of the FSAR. The quality assurance program followed during plant design and construction was implemented in accordance with section 17.1. The quality assurance program used for plant operations is described in the SNC Quality Assurance Topical Report (QATR).

Drawings, records, correspondence, field books, forms, film, and temperature charts required to show compliance with the codes, tests, and quality control standards as outlined in section 17.1 are in the possession of or available to Southern Nuclear Operating Company throughout the life of the plant. This is discussed further in section 13.5.

3.1.2 CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety are 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 reflect:

- 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.

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

- The importance of the safety functions to be performed.

CONFORMANCE

The natural phenomena and their magnitude have been selected in accordance with the probability of their occurrence at the Farley plant site. The designs are based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in historical data. The natural phenomena postulated in the design are presented in sections 2.3, 2.4, and 2.5. The design criteria for the structures, systems, and components affected by each natural phenomenon are presented in sections 3.2, 3.3, 3.4, 3.5, 3.7, and 3.8. Those combinations of natural phenomena and plant originated accidents that are considered in the design are identified in sections 3.8, 3.9, and 3.10. The importance of the safety functions is identified with the classification system developed by the American Nuclear Society. Such identification is included in section 3.2.

3.1.3 CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials are 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 are provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

CONFORMANCE

The reactor facility is designed to minimize the probability and effect of fires and explosions. Noncombustible and heat resistant materials are used in the containment, in the control room, in rooms where safety-related equipment is located, and wherever required throughout the plant. Appropriate equipment and facilities for fire protection, including detection, alarm, and extinguishing of fire, are provided to protect both plant equipment and personnel from fire, explosion, and the resultant release of toxic vapors. Both wet and dry types of firefighting equipment are provided. Fire protection is provided by deluge sprays, sprinkler systems, piped carbon dioxide systems, hose stream, and portable extinguishers of the type appropriate for each area.

Firefighting systems are designed to assure that their rupture or inadvertent operation will not significantly impair safety-related systems. The fire protection system consists of a reliable, partially automatic unit engineered and installed in accordance with the requirements of the National Fire Protection Association (NFPA) or American National Standards Institute (ANSI), the Nuclear Energy Property Insurance Association (NEPIA), the Occupational Safety and Health Act (OSHA), Nuclear Regulatory Commission guidelines, and the Nuclear Electric Insurance Limited (NEIL) Property Loss and Prevention Standards, in addition to the applicable local codes and regulations.

The fire protection system is provided with test hose valves for periodic testing. All equipment is accessible for periodic inspection. The fire protection system is described further in subsection 9.5.1 and appendix 9B.

3.1.4 CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

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

CONFORMANCE

Structures, systems, and components important to safety are designed to accommodate the effects of any environmental conditions at the Farley site that are associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs, and are compatible with these conditions. Design criteria and implementation are presented in sections 3.5 and 3.6, and environmental factors are described in sections 3.11, 6.3, and 6.4. These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from any postulated failure of equipment or structural design or from environmental conditions and accidents outside the nuclear power unit. Postulated breaks in the reactor coolant loop, except for branch line connections, have been eliminated from the structural design basis for both Unit 1 and Unit 2, as allowed by the revised General Design Criterion 4. The elimination of these breaks is the result of the application of leak-before-break technology as described in section 3.6. Chapter 7.0 lists the motors, instrumentation, and associated cables of protection and safety features systems located inside the containment structure. It also gives the design

requirements in terms of the time that each must survive the extreme environmental conditions following a LOCA.
Details of the design, environmental testing, and construction of these systems, structures, and components are included in other sections of chapters 3.0, 5.0, 6.0, 7.0, and 9.0. Evaluation of the performance of safety features and analyses of possible accidents are contained in chapter 15.0.

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

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

CONFORMANCE

The Farley Plant is a two-unit plant with the following shared structures, systems, and components that provide features and functions important to safety. Such sharing has been evaluated to ensure that there are no adverse impacts on safety functions.

<u>Structures</u>

Control room
Diesel generator building
Auxiliary building HVAC penthouse
Ultimate heat sink storage pond and service water intake structure

<u>Systems</u>

Control room HVAC system
Diesel generator fuel storage/transfer system
Diesel generator HVAC systems for shared diesel generators and switchgear rooms
Service water intake structure HVAC
SWIS nitrogen backup system

<u>Components</u>

Three of five diesel generators and auxiliary support equipment
Diesel generator 600-V switchgear and MCCs
Service water intake structure 600-V switchgear, batteries, and battery chargers
Load centers K, L, R, and S
Service water discharge pipes to UHS pond
Carbon dioxide storage
Fire protection water tanksFire protection pumps
Spent-fuel cask crane
Pressurizer heater service transformer

Functional evaluation of structures, systems, and components shared by the two Farley units are addressed in corresponding sections of the FSAR.

3.1.6 CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control, and protection systems are 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.

CONFORMANCE

Each reactor core and associated coolant, control, and protection system is designed with adequate margins to:

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

B. Ensure return of the reactor to a safe state following a Condition III fault with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

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

Chapter 4.0 discusses the design bases and design evaluation of reactor components including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in chapter 7.0. This information supports the accident analyses in chapter 15.0 which show that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

3.1.7 CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems are 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.

CONFORMANCE

Prompt compensatory reactivity feedback effects are assured, when the reactor is critical, by the negative fuel temperature effect (Doppler effect) and by the operational limit on moderator temperature coefficient of reactivity at full power. The negative Doppler coefficient of reactivity is assured by the inherent design using low-enrichment fuel. The reload core is designed to have an overall moderator temperature coefficient of reactivity which is less than or equal to +7.0 pcm/°F for power levels up to 70% with a linear ramp to 0.0 pcm/°F at 100% power by the use of fixed burnable absorber (BA) rods, and/or integral fuel burnable absorbers (IFBAs) and/or control rods by limiting the reactivity held down by soluble boron.

The core inherent reactivity feedback characteristics are described in section 4.3, (Nuclear Design). Reactivity control by chemical injection is discussed in subsection 4.2.3, (Reactivity Control Systems), and subsection 9.3.4, (Chemical and Volume Control System). Plant Technical Specifications define allowable absorber concentrations.

3.1.8 CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems are 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.

CONFORMANCE

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

Oscillations due to xenon spatial effects in the radial, diametral, and azimuthal overtone modes are heavily damped because of the inherent design and 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.

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 (Nuclear Design). Details of the instrumentation design and logic are discussed in chapter 7.0.

3.1.9 CRITERION 13 - INSTRUMENTATION AND CONTROL

Instrumentation is 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.

CONFORMANCE

Instrumentation and control systems are provided to monitor and maintain plant variables including those variables that affect the fission process, integrity of the reactor core, the reactor coolant pressure boundary, and the containment over their expected range for normal operation, for anticipated operational occurrences, and under accident conditions. These systems comply with the intent of Criterion 13.

The following processes are controlled to maintain key variables within their normal ranges:

A. Reactor power level (manual or automatic by control of thermal load).

B. Reactor coolant temperature (manual or automatic by control of rod position).

C. Reactor coolant pressure (manual or automatic by control of heaters and spray in the pressurizer).

D. Reactor coolant water inventory, as indicated by pressurizer water level (manual or automatic control of charging flow).

E. Reactor coolant system boron concentration (manual or automatic control of makeup charging flow).

F. Steam generator inventory on secondary side (manual or automatic control of feedwater flow).

The reactor control system is designed to automatically maintain programmed average temperature in the reactor coolant during steady-state operation and to ensure that plant conditions do not reach reactor trip settings, as the result of design load change. Overall reactivity control is achieved by the combination of soluble boron and control rods. Long term regulation of the core reactivity is obtained by adjustment of the concentration of boron in the reactor coolant. Short-term reactivity control for power changes is performed by the reactor control system which automatically positions the rod cluster control assemblies. The measured inputs to this system are neutron flux, coolant temperature, and turbine load.

The instrumentation functional requirements are given in chapter 4.0 and the instrumentation and control systems are discussed in more detail in chapter 7.0. Operation of the reactor within prescribed limits is enforced by the reactor protection system. This system is discussed in section 7.2.

A wide spectrum of measurements is displayed for operator information and/or is processed to provide alarms. These measurements provide notification and allow correction of conditions having the potential of leading to abnormal or accident conditions. Typical indication (or alarm) measurements are rod position, rod deviation, insertion limit, rod bottom, rod control system failure, rod control system urgent failure, incore flux and temperature, protection system faults, and protection system test mode.

3.1.10 CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

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

CONFORMANCE

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation, including all anticipated transients, and maintain the stresses within applicable stress limits. The design criteria methods and procedures applied to components of the reactor coolant pressure boundary are discussed in subsection 5.2.1. Reactor coolant pressure boundary materials selection and fabrication techniques ensure a low probability of gross rupture or significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions such as pipe rupture and seismic, as discussed in sections 3.6 and 3.7, respectively. Fracture prevention measures are included to prevent brittle fracture. Refer to the discussion under Criterion 31 for additional information.

The system is protected from overpressure by means of the pressurizer high-pressure reactor trip (section 7.2) and by pressure relieving devices according to the provisions of subsection 5.2.2.

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

The pressure boundary has provisions for inspection, testing and surveillance of critical areas to assess the structural and leaktight integrity of the boundary. The reactor coolant pressure boundary leakage detection systems and inservice inspection program are discussed in subsections 5.2.7 and 5.2.8, respectively.

3.1.11 CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated auxiliary, control, and protection systems are 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.

CONFORMANCE

The reactor coolant system and associated auxiliary, control, and protection systems are designed to ensure the integrity of the reactor coolant pressure boundary with adequate margins during normal operation and during Condition I and Condition II transients. The system boundary accommodates loads due to the 1/2 safe shutdown earthquake during normal operation including normal operational transients (Condition II) within upset condition code stress limits. The system boundary accommodates loads due to the safe shutdown earthquake combined with loads due to piping failures such as circumferential pipe ruptures of reactor coolant pipes at junctures with equipment nozzles and connecting pipes at junctures to reactor coolant piping (Condition IV), without propagation of failure to remaining reactor coolant system loops, steam power conversion system, or other piping or equipment needed for emergency cooling. The components of the reactor coolant system and associated fluid systems are designed in accordance with appropriate ASME and ANSI codes. These codes are identified in

chapter 5.0. The protection system is designed in accordance with IEEE-Std 279. The protection system analyses are given in subsection 7.2.2. Overpressure protection is provided by automatic controls,
pressure relief valves, and code safety valves.

The selected design margins include operating transient changes due to thermal lag, coolant transport times, pressure drops, system relief valve characteristics, and instrumentation and control response characteristics.

3.1.12 CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems are 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.

CONFORMANCE

The containment, penetration rooms, and the engineered safeguards systems are designed to provide protection for the public from the consequences of an unlikely event of the LOCA, which is based on a postulated break of the reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe.

The containment, penetration rooms, and the engineered safeguards system are designed to safely sustain all internal and external loading conditions resulting from postulated transients and accidents as described in chapter 15.0. Functional capability of the containment, penetration rooms, and engineered safeguards will be maintained for as long as required to protect the public. Due consideration is given to site factors and local environments as they relate to public health and safety.

Refer to section 3.8 and chapters 6.0 and 15.0.

3.1.13 CRITERION 17 - ELECTRIC POWER SYSTEMS

An onsite electric power system and an offsite electric power system are 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) provides sufficient capacity and capability to assure that 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 the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, 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 is 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 is designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit. This assures that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits is 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 are included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

CONFORMANCE

An onsite electric power system and an offsite electric power system are provided to permit functioning of systems and components important to safety. As discussed in sections 8.2 and 8.3, sufficient capacity and capability have been provided in each system to assure that 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 the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric ac power supply is furnished by five diesel generators. The engineered safeguards loads are divided between the emergency buses of each unit in a balanced, redundant load grouping so that the failure of one diesel generator or one emergency bus will not prevent the safe shutdown of both units.

The onsite electric dc power supply for each unit consists of two redundant battery systems, either of which is adequate to supply the dc power required for the engineered safeguards. Failure of a single component in this system will not impair control of the minimum engineered safeguards required to maintain each unit in a safe condition.

Electric power from the transmission network to the onsite electric distribution system is provided for Units 1 and 2 by six 230- and 500-kV transmission lines. Four of these lines are connected to the 230-kV switchyard and two of these lines are connected to the 500-kV switchyard. The 230- and 500-Kv switchyards are interconnected at the plant by two 230/500-kV autotransformers. Power from the switchyard is provided for the engineered safeguards for each unit through two startup auxiliary transformers via two physically independent circuits which serve alternately as the redundant source for each emergency bus. A failure of a single active component in this power system will not prevent its required functioning.

In case of loss of all onsite power and of one offsite power circuit, power requirements for ensuring the specified fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded since they are met from the other redundant offsite power source.

In the event of a LOCA, all the emergency supplies are available to supply safety loads to assure that core cooling, containment integrity, and other vital safety functions are maintained.

The electric power supplies have been designed so as to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of offsite power, or the loss of onsite power.

3.1.14 CRITERION 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

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

CONFORMANCE

The inspection and testing of components of the electric power systems are divided into two types:

I. Inspection and testing performed during periods of unit outage.

II. Inspection and testing performed with the unit in service.

Type I covers checking for circuit integrity of wiring insulation and connections, of 4160-V switchgear, 600-V load centers, motor control centers, and the interphase circuitry between components falling under Type II. Type II covers the periodic testing of components of the electrical system such as diesel generators, relays, breakers, valves, and motors to assess their operability and functional performance. Testing of systems as a whole for the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system and the transfer of power from the nuclear power unit to the offsite power source, will be done on a periodic basis. The scope of testing involved for these types is covered in subsections 8.3.1 and 8.3.2.

3.1.15 CRITERION 19 - CONTROL ROOM

A control room is provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection is 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 is provided 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 with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

CONFORMANCE

Following proven power plant design philosophy, vital control stations, switches, controllers, and indicators necessary to start up, operate, and shut down each nuclear unit and maintain safe control of the facility are located in one control room. The control room and the associated post accident ventilation systems are designed in accordance with Category I requirements, as discussed in subsection 3.2.1.

The design of the control room will permit access and occupancy during a LOCA. Sufficient shielding and ventilation are provided to permit occupancy of the control room for a period of 30 days following the LOCA, without receiving more than 5 rem integrated whole body dose, or its equivalent, to any part of the body. The shielding is discussed in subsection 12.1.2, ventilation is discussed in subsection 9.4.1, and the control room habitability systems are discussed in section 6.4.

As discussed in subsection 7.4.1, it is possible to bring each unit to the hot shutdown condition, and maintain it in that condition, from outside the control room if access to the control room is prohibited during normal operation. Through the use of suitable procedures, the plant design also provides the potential capability of attaining cold shutdown from outside the control room.

3.1.16 CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system is designed 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 to sense accident conditions and initiate the operation of systems and components important to safety.

CONFORMANCE

The operational limits for the reactor protection system are defined by analyses of plant operating and fault conditions requiring rapid rod insertion to prevent or limit core damage. With respect to acceptable fuel design limits, the system design bases for anticipated operational occurrences are as follows:

A. Minimum departure from nucleate boiling ratio (DNBR) will not be less than the safety analysis limit.

B. Clad strain on the fuel element shall not exceed 1 percent.

C. No center melt shall occur in the fuel elements.

A region of permissible core operation is defined in terms of power, axial power distribution, and coolant flow and temperature. The protection system monitors these process variables (as well as other process variables and plant conditions). If the region limits are approached during operation, the protection system will automatically actuate alarms, prevent control rod withdrawal, or trip the reactor depending on the severity of the condition.

Operation within the permissible region and complete core protection is assured by the overtemperature ΔT and overpower ΔT reactor trips in the system pressure range defined by the pressurizer high pressure and pressurizer low pressure reactor trips, in the event of a transient that is slow with respect to piping delays from the core to the temperature sensors. In the event that a transient faster than the ΔT responses occurs, high nuclear flux and low coolant flow reactor trips provide core protection. Finally, the thermal transients are anticipated and avoided by reactor trips initiated by turbine trip.

The protection system operates by interrupting power to the rod control power supply. As a result, all full length control and shutdown rods insert by gravity. The Westinghouse protection system design meets the requirements of IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

The protection system measures a wide spectrum of process variables and plant conditions. All analog channels that actuate reactor trip, rod stop, and the permissive functions are indicated or recorded. In addition, visual and/or audible alarms are actuated for reactor trip; partial reactor trip, any input channel; and any control variable exceeding its setpoint, any input channel. These measurements and indications provide the bases for corrective action to prevent the development of accident conditions. In the event of accident conditions, however, the reactor protection system senses the condition, processes the signals used for engineered safety features actuation, and generates the actuation demand.

The reactor trip system is discussed in section 7.2 and the engineered safety features are discussed in section 7.3.

3.1.17 CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system is designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system are sufficient to assure that no single failure results in loss of the protection function and to assure that removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system is 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.

CONFORMANCE

The protection system is designed for the high functional reliability and inservice testability commensurate with the safety functions to be performed.

The system consists of a large number of input measurement channels, redundant logic trains, redundant reactor trip breakers, and redundant engineered safety features actuation devices. It performs indication and alarm functions in addition to its reactor trip and engineered safety features actuation functions. The design meets the requirements of IEEE Standard 279-1971, "Criteria for Nuclear Power Generating Station Protection Systems." The redundant logic trains, reactor trip breakers, and engineered safety features actuation relays are electrically isolated and physically separated. Further, physical separation of the channels is maintained within the separated trains to the point of logical processing.

Either of the two redundant logic trains performs the required protection function. All channels employed in power operation are sufficiently redundant so that individual testing and calibration, without degradation of the protection function or violation of IEEE Standard 279-1971, can be performed with the reactor at power. Such testing discloses failures or reduction in redundancy that may have occurred. Removal from service of any single channel or component employed during power operation does not result in loss of minimum required redundancy. For example, a two-of-three logic function is placed in the one-of-two mode when one channel is removed from service.

Semiautomatic testers are built into each of the two logic trains. These testers have the capability of testing the major part of the protection system rapidly with the reactor at power. Between tests, the testers continuously monitor a number of internal protection system points, including train power supply voltages and fuses. The outputs of these monitor circuits are logically processed to provide an alarm in the event of a single failure in either train, and automatic reactor trip in the event of one or more for failures in both trains. Self testing provision is designed into each tester.

The protection system is discussed in section 7.2.

3.1.18 CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system is 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 are demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, are used to the extent practical to prevent loss of the protection function.

CONFORMANCE

The protection system has been designed to perform its intended protective functions under the effects of accident conditions or postulated events. The design features that limit the effects of natural phenomena such as tornado, flood, earthquake, and fire, are physical separation and electrical isolation of redundant channels and subsystems, functional diversity of subsystems, and safe component and subsystem failure modes. The redundant logic trains, reactor trip breakers, and safety features actuation devices are physically separated and electrically isolated. Physically separate channel cable trays, conduit, and penetrations are provided to ensure independence of redundant elements of each train. Functional diversity and location diversity are designed into the system. For example, the loss of one feedwater pump would actuate one pressure trip, one high-level trip, one low-level trip, two temperature trips, and one flow trip. The system logic is designed so that, with exception of the safety features actuation devices, a zero input represents a trip demand. Hence, severed or shorted channel wiring, loss of power, and the majority of channel component failures are seen by the system as trip demands.

The protection system is tested and qualified under extreme environmental conditions. These tests ensure that the equipment will perform the required functions under accident conditions.

Loss of the protection function through improper testing or failure of the test equipment is guarded against by interlocks that enable the test of only one of the two trains at the same time, bypass trip breakers to maintain the protection function during test, annunciation of the test mode, unambiguous tester readout, and the indication, alarm, and status systems.

Functional and locative diversity designed into the system are defenses against loss of the protection function through postulated accident conditions. For the postulated loss of coolant accident, at least five diverse reactor trip demands and at least two diverse engineered safety features actuation demands would be generated. In addition, manual reactor trip and manual engineered safety features actuation means are provided.

The protection system has been quantitatively evaluated with respect to functional diversity and qualitatively evaluated with respect to common mode susceptibility. These studies indicate that the system is designed to have a very high probability of performing its function in any postulated occurrence.

The reactor protection system and the engineered safety features actuation system are discussed in sections 7.2 and 7.3, respectively.

3.1.19 CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system is 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.

CONFORMANCE

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of energy sources and the environment. Each reactor trip channel is designed on the 'deenergize to trip' principle so that a loss of power or disconnection or shorting of a channel causes that channel to go into its tripped mode. Likewise, loss of voltage to either of the two protection system output devices will trip the reactor. In a two-out-of-three logic circuit, the three channels are equipped with separate primary sensors and each channel is energized from independent electrical buses. A single failure can, in the worst case, cause a single channel to fail to deenergize. The trip signal furnished by the two remaining channels is unimpaired by this event. In addition, 15 internal points in each train are continuously monitored by the semiautomatic testers. Faults involving one logic train are annunciated; multiple faults involving both trains automatically trip the reactor, although such faults would not necessarily defeat the trip function. All full-length control and shutdown rods will insert by gravity if the rod power supply is lost.

The protection system components have been tested and qualified for the extremes of the normal environment to which they are subjected. In addition, components are tested and qualified according to individual requirements for the adverse environment specific to their location which might result from postulated accident conditions.

In the event of a loss of the preferred offsite power source, onsite diesel generators provide power to emergency loads. Station batteries are provided to power the vital instrumentation loads. The diesels are capable of supplying power required to operate engineered safeguards pumps and associated valves. A loss of power to one train of emergency core cooling equipment will not affect the ability of the other train to perform its function.

The rod control system, reactor trip system, and engineered safety features actuation systems are discussed in sections 4.2., 7.2, and 7.3, respectively.

3.1.20 CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system is 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 is limited so as to assure that safety is not significantly impaired.

CONFORMANCE

The failure of a single control system component or channel, or the failure or removal from service of any 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 is limited so as to assure that safety is not impaired.

Most functions performed by the reactor protection and the reactor control systems require the same process information. The design philosophy for these systems is to make maximum use of a wide spectrum of diverse and redundant process measurements. The protection system is separate and distinct from the control system. The control system is dependent on the protection system in that control input signals are derived from protection system measurements, where applicable. These control signals are transferred to the control system by isolation amplifiers which are classified as protection system components. No credible failure at the output of an isolation amplifier will prevent the corresponding protection channel from performing its protection function. Such failures include short circuits, open circuits, grounds, and the application of the maximum credible ac and dc voltages. The adequacy of system isolation has been verified by testing under these fault conditions. The design meets all requirements of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

The reactor protection system and the reactor control system are discussed in sections 7.2 and 7.7, respectively.

3.1.21 CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system is 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.

CONFORMANCE

The protection system design assures that acceptable fuel design limits are not exceeded in the event of single reactivity control malfunctions including accidental withdrawal of control rod groups. Analyses of postulated accidents are given in chapter 15.0.

Reactor shutdown with control rods is completely independent of the control functions. The trip breakers will interrupt power to the full-length rod drive mechanisms to trip the reactor regardless of the status of existing control signals.

The reactor control system provides visual displays of the control rod assembly positions and actuates an alarm in the event that deviation of rods occurs within their banks.

Additional information is given by the response to Criterion 10. The reactivity control systems are discussed in subsection 4.2.3; the protection system is discussed in section 7.2; and the electrical control systems are discussed in section 7.7.

3.1.22 CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles are provided. One of the systems uses control rods, preferably including a positive means for inserting the rods, and is 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 is 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 is capable of holding the reactor core subcritical under cold conditions.

CONFORMANCE

Two independent reactivity control systems of different design principles are provided. One of the systems uses control rods; the second system employs dissolved boron (chemical shim).

The rods are assembled in clusters and are manipulated as groups[a] of clusters. Two functional categories of rods are employed. These categories are full-length shutdown and full-length control. 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 shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to stick in/out of core position. The reactor protection system initiates reactor trip by interrupting power to the rod control power supply. This releases the magnetic latches, and the full-length control and shutdown rods are inserted. The design is inherently fail safe.

The boron system is capable of controlling the rate of reactivity change resulting from planned normal power changes, including xenon burnout, to assure fuel design limits are not exceeded. This system is capable of maintaining the reactor core subcritical under cold conditions with all rods withdrawn.

The reactivity control systems are discussed in subsection 4.2.3.

3.1.23 CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems are 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.

CONFORMANCE

The facility steam supply system is provided with the means of making and holding the core subcritical under any anticipated condition with appropriate margin for contingencies. Combined use of rod control and chemical shim control permit the necessary shutdown margin to be maintained during long term xenon decay and plant cooldown. The single highest worth control rod cluster is assumed to be stuck in its fully withdrawn position in postulated accident analyses. These means are discussed in detail in chapter 4 and subsection 9.3.4.

Under accident conditions when the safety injection system is actuated, concentrated boric acid is injected into the reactor coolant system. In case of a LOCA the accumulators will passively inject borated water. Reactivity effects of safety injection are discussed in section 6.3 and evaluated for accident conditions in chapter 15.

3.1.24 CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems are designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor 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.

CONFORMANCE

Core reactivity is controlled by a chemical poison dissolved in the coolant, rod cluster control assemblies, and burnable poison rods. The maximum reactivity insertion rates due to withdrawal of a bank of rod cluster control assemblies or by boron dilution are limited. These limits are set so that peak heat generation rate and the departure from nucleate boiling ratio (DNBR) do not exceed the specified limits at overpower conditions. The maximum worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values which prevent rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity. The reactor can be brought to the shutdown condition and the core will maintain acceptable heat transfer geometry following any Condition IV event, such as rod ejection or steam line break. Specifically, faulted condition stress limits are used to contain, within tolerable limits, the effects of extremely unlikely events.

3.1.25 CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

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

CONFORMANCE

The protection and reactivity control systems are designed to assure extremely high probability of performing their required safety functions in the event of anticipated operational occurrences. Redundancy, functional and locative diversity, testability, use of safe failure modes, and analyses are design measures which are employed to assure performance of the required safety functions. Detailed probabilistic analyses of the systems verify this high reliability. The protection system is further discussed under Criteria 20 through 25 and in section 7.2. The reactivity control systems are discussed in sections 4.2 and 7.7.

3.1.26 CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

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

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a. Two rod groups are considered a rod bank. The groups in a bank are electrically interlocked to move alternately and to remain within one step of each other.

The reactivity control systems are discussed in subsection 4.2.3 and section 4.3.

CONFORMANCE

Reactor coolant pressure boundary components are designed, fabricated, inspected, and tested in conformance with ASME Nuclear Power Plant Components Code, Section III. The design bases and evaluations of reactor coolant pressure boundary components are discussed in section 5.2.

Major components are classified as ANS N18.2 Safety Class 1 and are accorded the quality measures appropriate to this classification.

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 gaskets. Leakage inside the containment is drained to the containment sump.

Leakage is also detected by measuring the airborne activity and the rate of condensate drained from the containment air recirculation units. Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank, and coolant drain collection tanks makes available an accurate indication of integrated leakage.

The reactor coolant pressure boundary leakage detection system is discussed in subsection 5.2.7.

3.1.27 CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

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

CONFORMANCE

The Joseph M. Farley Units No. 1 and 2 are designed to conform to the intent of Criterion 31. The reactor coolant pressure boundary is designed so that, for all transients, normal, upset, and faulted, the reactor coolant pressure boundary behaves in a nonbrittle manner. The units were designed for 650° and 2500 psia. The normal service temperature and pressure are 550°F and 2250 psia. The reactor pressure vessels were designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, which considers cyclic loading, defect characterization, minimum material toughness, and maximum allowable stresses. Material selection and testing were in accordance with the Summer 1970 Addenda of the ASME Code.

Normal operating limits are calculated based on the toughness properties of the ferritic components of the pressure boundary in accordance with nonmandatory Appendix G, Section III of the ASME Boiler and Pressure Vessel Code. [For details, see heatup and cooldown limit curves for normal operation, which are provided in the Pressure Temperature Limits Reports (PTLR)].

Postulated accident conditions are analyzed using the concepts of fracture mechanics technology with which one quantitatively considers defect size and geometry, applied stresses (residual, thermal, and membrane), and the material properties.

The effects of irradiation are considered in the generation of the heatup and cooldown limit curves, analysis of post operational tests, and for analyses of all other transients and postulated accidents. A reactor vessel materials radiation surveillance program is performed in accordance with ASTM E-185. (For details see paragraph 5.2.4.4.)

3.1.28 CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary are designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity and to provide an appropriate material surveillance program for the reactor pressure vessel.

CONFORMANCE

The Joseph M. Farley Units' design conforms with the intent of Criterion 32. The Units' reactor coolant pressure boundary design meets the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, which requires access for all required inspections. The design also permits the conduct of a material surveillance program for the reactor pressure vessel. Additional details of these features can be found in subsections 5.2.8, Inservice Inspection Program, and 5.2.4, Fracture Toughness.

3.1.29 CRITERION 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary is provided. The system safety function is 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 is 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.

CONFORMANCE

The chemical and volume control system provides a means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below a preset level. High pressure centrifugal charging pumps are provided which are capable of supplying the required makeup and reactor coolant seal injection flow with power available from either onsite or offsite electric power systems. These pumps also serve as high head safety injection pumps. In the event of a loss of coolant larger than the capacity of the normal makeup path, these pumps discharge into the larger safety injection piping. A high degree of functional reliability is assured by provision of standby components and assuring safe response to probable modes of failure. Details of system design are included in subsection 9.3.4 and details of the electric power systems are given in chapter 8.0.

3.1.30 CRITERION 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat is provided. The system safety function is to transfer fission product decay heat and other residual heat from the reactor core at the 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 are 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.

CONFORMANCE

The residual heat removal system (RHRS), in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core within acceptable limits. The crossover from the steam power conversion system to the RHRS occurs at approximately 350°F.

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

Suitable redundancy at temperatures above approximately 350°F is provided by the steam generators and attendant piping. Details of the system design are given in subsection 5.5.7 and section 6.3.

3.1.31 CRITERION 35 - EMERGENCY CORE COOLING

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

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities are 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.

CONFORMANCE

By combining the use of passive accumulators with two safety injection pumps and two residual heat removal pumps, emergency core cooling is provided even if there should be a failure of any component in any system. The emergency core cooling system (ECCS) employs a passive system of accumulators, which do not require any external signals or source of power for their operation, to cope with the short-term cooling requirements of large reactor coolant pipe breaks. Two independent and redundant high-pressure and high-flow pumping systems, each capable of the required emergency cooling, are provided for small break protection and to keep the core in a coolable geometry after the accumulators have discharged, following a large break. Adequate design provisions assure the performance of the required safety functions even with the loss of a single component, assuming the electric power is available from either the offsite or onsite electric power sources. Borated water is injected into the reactor coolant system by accumulators, high-head safety injection pumps (charging pumps), and low-head safety injection pumps (RHR pumps). The design meets the intent of the "Interim Policy Statement Criteria for Emergency Core Cooling Systems for Light Water Power Reactors."

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, with its essential heat transfer geometry preserved. This protection is afforded for:

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

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

The basic criteria for LOCA evaluations are no clad melting, zircaloy water reactions will be limited to an insignificant amount, and the core geometry is to remain essentially in place and intact so that effective cooling of the core will not be impaired.

The zircaloy water reactions will be limited to an insignificant amount so that the accident:

A. Does not interfere with the emergency core cooling function to limit clad temperatures.

B. Does not produce H_2 in an amount that when burned would cause the containment pressure to exceed the design value.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the ECCS adds shutdown reactivity so that with a stuck rod, no offsite power, and minimum engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact. With no stuck rod, offsite power, and all equipment operating at design capacity, there is insignificant cladding damage. The ECCS is described in section 6.3.

Chapter 15.0 contains an analysis of LOCAs.

3.1.32 CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system is 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.

CONFORMANCE

Design provisions are made for inspection to the extent practical of all components of the emergency core cooling system. An inspection is performed periodically to demonstrate system readiness.

The pressure-containing systems are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

In addition, to the extent that is practical, the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and safety injection pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence, and by nondestructive inspection when such techniques are desirable and appropriate.

Details of the inspection program for the reactor vessel internals are included in section 5.4. Inspection of the emergency core cooling system is discussed in subsection 6.3.4.

3.1.33 CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system is designed to permit appropriate periodic pressure and functional testing to assure the structural and leaktight integrity of its components, the operability and performance of the active components of the system, and the operability of the system as a whole. Also tested, under conditions as close to design as practical, is 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.

CONFORMANCE

The components of the system located outside the containment will be accessible for leaktightness inspection during appropriate periodic tests. Each active component of the emergency core cooling system may be individually actuated on the normal power source or transferred to emergency power sources at any time during plant operation to demonstrate operability. The centrifugal charging/safety injection pumps are part of the charging system, and this system is in continuous operation during plant operation. The test of the safety injection pumps employs the minimum flow recirculation test line which connects back to the refueling water storage tank. Remotely operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers also may be checked during integrated system tests performed during a planned cooldown of the reactor coolant system.

Design provisions also include special instrumentation, testing, and sampling lines to perform the tests during plant shutdown to demonstrate proper automatic operation of the ECCS. A test signal is applied to initiate automatic action, and verification made that the safety injection pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.

These tests are described in subsection 6.3.4.

3.1.34 CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment is provided. The system safety function is 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 are provided to assure that for onsite electrical 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.

CONFORMANCE

Three systems are provided to reduce containment atmosphere temperature and pressure and/or to remove heat from the containment under post accident conditions. These are the low-head safety injection/residual heat removal system, the containment spray system, and the containment cooling system. The design, operation, and reliability of the low-head safety injection/residual heat removal system are discussed in section 6.3, Emergency Core Cooling System.

The containment spray system has been designed to spray water into the containment atmosphere, when appropriate, in the event of a MSLB or LOCA, to ensure the containment peak pressure is below its design value. This is accomplished by one of the two trains of containment spray. A detailed description of the containment spray system is provided in subsection 6.2.2.

The containment cooling system has been designed to remove heat which would be released to the containment atmosphere during any MSLB or LOCA up to and including the double-ended rupture of the largest system pipe. This is accomplished by one of the four containment air coolers. A detailed description of the containment cooling system is provided in subsection 6.2.2.

The containment heat removal systems are designed to ensure that the failure of any single active component, assuming the availability of either onsite or offsite power exclusively, does not prevent the systems from accomplishing their design safety functions.

3.1.35 CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

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

CONFORMANCE

The containment air coolers and associated service water system piping (subsection 6.2.2) inside containment can be inspected during shutdown. The service water system is outside containment and, with the exception of buried piping, can also be inspected periodically.

The containment spray system (subsection 6.2.2) essential equipment, except risers, distribution header piping, spray nozzles, and the containment sump are located outside containment. The containment sump and the spray pipe and nozzles can be inspected during shutdowns. Those portions of the containment spray suction piping from the containment sump and refueling water storage tank either embedded in concrete or buried are not accessible for inspections. Associated equipment outside the containment can be visually inspected.

3.1.36 CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system is designed to permit appropriate periodic pressure and functional testing to assure the structural and leaktight integrity of its components; the operability and performance of the active components of the system; 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.

CONFORMANCE

The normal operation of the four containment coolers will verify structural and leaktight integrity of the system components, as well as demonstrate the operation of the associated cooling system. System piping, valving, pumps, fans (at low speed), coolers, and other components of this system are arranged so that each component can be tested periodically for operability (see subsection 6.2.2). The delivery capability of the containment spray system can be tested periodically to the extent practical up to the last valve before the spray nozzles. These tests can also verify the structural and leaktight integrity of the spray system components (see subsection 6.2.2).

The containment air cooling and containment spray systems are designed to provide the capability for testing the full operational sequence as close to design as practical, including transfer to alternate power sources, during shutdown or refueling (see paragraphs 6.2.2.4 and 6.2.3.4).

3.1.37 CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

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

CONFORMANCE

The ECCS recirculation fluid pH is controlled to reduce airborne iodine activity levels inside containment and to retain the removed iodine in solution in the core sump. For a complete description of the design of the containment spray systems, see subsections 6.2.2 and 6.2.3.

A penetration room filtration system is provided for control of contaminants from ECCS recirculation leakage following a LOCA. This system is also used to mitigate the consequences of a fuel handling accident in the spent-fuel pool. For a complete description of the design of the penetration room filtration system, see subsection 6.2.3.

A hydrogen control system is provided to control hydrogen concentrations inside containment following an accident. This system consists of redundant hydrogen-oxygen recombiners and appropriate containment atmosphere mixing and sampling systems. Backup protection against excessive containment atmosphere hydrogen concentrations is provided by a post-accident containment purging system. For a complete description of the design of the containment combustible gas control system see subsection 6.2.5

Each of the containment atmosphere cleanup systems has suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available), its safety function can be accomplished assuming a single failure.

3.1.38 CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems are 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.

CONFORMANCE

Trisodium phosphate is added to the recirculation sump to help reduce airborne iodine activity levels inside containment and to retain the removed iodine in solution in the sump. Inspection of the remaining components of the containment spray system is discussed in the response to Criterion 39.

Components of the penetration room filtration system, including ducts, filters, fans, and dampers are located in the auxiliary building and are accessible for physical inspection (see paragraph 6.2.3.4).

The post-accident containment hydrogen-oxygen recombiners and atmosphere mixing system, and selective containment isolation valves associated with the post-accident containment sampling and purging systems are located inside containment and are available for inspection during shutdown or refueling. The trisodium phosphate baskets are also located inside containment and are available for inspection during shutdown and refueling. The remaining components of the containment atmosphere cleanup systems are located outside containment and are available for periodic inspection (see paragraphs 6.2.3.4 and 6.2.5.4).

3.1.39 CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems are designed to permit appropriate periodic pressure and functional testing to assure the structural and leaktight integrity of its components; the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves; 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.

CONFORMANCE

The capability for testing of the containment spray system has been discussed in the response to Criterion 40.

The penetration room filtration system is designed to provide the capability for testing the penetration room filtration system through the full operational sequence and as close to design conditions as practical, including transfer to alternate power sources, during shutdown or refueling. Such tests assure the operability and performance of the active system components and demonstrate the structural and leaktight integrity of the system components. A capability is also provided for periodic testing and surveillance of the penetration room filtration system filters to ensure that filter bypass paths have not developed and filter and adsorber materials have not deteriorated beyond acceptable limits.

The post-accident containment combustible gas control system is designed to allow appropriate periodic testing of the system through the full operational sequence and as close to design conditions as practical, including transfer to alternate power sources, to demonstrate system integrity as well as operability and performance of system active components. For further discussion of the testing capability of this system see subsection 6.2.5.

3.1.40 CRITERION 44 - COOLING WATER

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink is provided. The system safety function is 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 are provided to assure that for onsite electrical 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.

CONFORMANCE

Component cooling water, service water and spent-fuel pool cooling systems are provided to transfer heat from the plant to an ultimate heat sink. These systems have been designed to transfer their respective heat loads under all anticipated operating and accident conditions. Suitable redundancy, leak detection, and system interconnection and isolation capabilities have been incorporated into the design of these systems to assure that the systems can accomplish all required safety functions, assuming a single failure concurrent with either onsite or offsite power exclusively.

A more complete description of the spent-fuel pool cooling system design is presented in subsection 9.1.3. Descriptions of the design of the other cooling water systems are presented in section 9.2.

3.1.41 CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

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

CONFORMANCE

All safety-related components of the cooling water systems, with the exceptions of buried piping to the river and to pond intake structures and service water piping to the containment coolers that are located within the containment are accessible for periodic inspection. The service water piping inside the containment is accessible for inspection during reactor shutdown and refueling periods. The integrity of the buried pipe is demonstrated by pressure and functional tests.

Refer to section 9.2 for further discussion of these systems.

3.1.42 CRITERION 46 - TESTING OF COOLING WATER SYSTEM

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

CONFORMANCE

All cooling water systems operate either continuously or intermittently during normal plant operation. This functional operation serves to demonstrate the operability, performance, and structural and leaktight integrity of the system components. The cooling water systems are designed to include the capability for testing through the full operational sequence, as close to accident design conditions as practical, including transfer to alternate power sources (see section 9.2).

3.1.43 CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and the containment heat removal system are 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 reflects consideration of 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; the limited experience and experimental data available for defining accident phenomena and containment responses; and the conservatism of the calculational model and input parameters.

CONFORMANCE

The design of the containment is based on the LOCA, coupled with the partial loss of the redundant engineered safety features, which produces the post-LOCA temperature and pressure conditions described in subsection 6.2.1.

A minimum safety margin of 10 percent between the peak calculated post-LOCA containment pressure and the design pressure has been provided. The containment structural design is described in section 3.8.

3.1.44 CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary is designed with sufficient margin to assure that, under operating, maintenance, testing, and postulated accident conditions, its ferritic materials behave in a nonbrittle manner and to assure that the probability of rapidly propagating fracture is minimized. The design reflects 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 material properties; residual, steady-state, and transient stresses; and size of flaws.

CONFORMANCE

Principal load carrying components of ferritic materials used in the containment will not be exposed to the external environment. The ferritic material of the containment liner plate is designed to function as a leaktight membrane only.

In addition, nil ductility transition temperature (NDTT) requirements are not considered relevant for the design of the containment since this is a ligament type of structure wherein the brittle fracture of a ligament could not propagate to adjacent ligaments.

In all areas where the liner plate is the pressure resisting structural element without backup from the concrete, a minimum NDTT of 0°F has been specified, based on a minimum service temperature of 30°F. These areas are the containment access openings which are enclosed on the outside by heated buildings. The equipment hatch not protected by the auxiliary building may experience service temperatures as low as 0°F; this material has been specified to an NDTT of -30°F.

In all other areas, the liner plate has no structural function since it is backed up by concrete. Except for small locally thickened areas (up to 2 in.) in the floor and walls for crane brackets, anchorages for main steam pipe ruptures, frames, and similar items, the containment liner plate is 1/4 in. thick. No NDTT has been specified since the rules for NDTT from Section III of the ASME Code for Class B vessels apply to pressure vessels only. In addition, these rules are based on "Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of Steel Structures" by Pellini and Puzak and, as stated by the authors, are not applicable to plates less than 5/8 in. thick. See section 3.8 for details.

Although not classified as a part of the containment for the Farley Nuclear Plant, piping and penetrations through the containment pressure boundary out to the penetration isolation valves have been designed and fabricated with due consideration for brittle fracture prevention.

A selective review of the design and materials used for the Farley Nuclear Plant was performed by the NRC staff, which confirmed compliance with General Design Criterion 51. Documentation to support this conclusion was submitted to the NRC by Alabama Power Company letters dated August 15, 1980; September 22, 1980; and January 30, 1981.

3.1.45 CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

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

CONFORMANCE

The reactor containment and other equipment which may be subjected to containment test conditions are designed so that periodic integrated leakage rate tests can be conducted. For a complete description of the planned leak rate tests, including system design provisions to accommodate the attendant containment pressure refer to paragraphs 3.8.1.7, 6.2.1.4, and the plant Technical Specifications.

3.1.46 CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

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

CONFORMANCE

There are special provisions for conducting individual leakage rate tests on applicable penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals. Other inspections are conducted as required by Appendix J of 10 CFR 50. Inservice inspection of the metallic liner and the pressure retaining concrete structure of the containments of both units, meeting the requirements of Subsections IWE and IWL of the ASME Section XI Code and applicable edition and addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC, will be performed. Refer to paragraph 3.8.1.7, subsection 6.2.1, and the plant Technical Specifications, subsections 5.5.6 and 5.5.17.

3.1.47 CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment are provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems are 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.

CONFORMANCE

Piping systems penetrating the containment have been provided with appropriate isolation capabilities. The design of the isolation system incorporates the capability to test the operability of the isolation valves periodically and to determine that valve leakage is within specified limits.

Leak detection capability for the isolation valves is provided as described in paragraph 6.2.1.4. The containment isolation system is discussed in subsection 6.2.4.

3.1.48 CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment is 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, namely:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment.

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment.

(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.

(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 are located as close to the containment as practical (and upon loss of actuating power), automatic isolation valves are designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them are 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 includes consideration of the population density, use characteristics, and physical characteristics of the site environs.

CONFORMANCE

Each line that is part of the reactor coolant pressure boundary and that penetrates the reactor containment is provided with containment isolation valves, in accordance with this criterion, as discussed in subsection 6.2.4.

Isolation valves outside the containment are located as close to the containment as practical, and automatic isolation valves are 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 are 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, includes consideration of the population density, use characteristics, and physical characteristics of the site environs.

3.1.49 CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment is 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, namely:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment.

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment.

(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.

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

CONFORMANCE

Each line that connects directly to the containment atmosphere and penetrates the primary reactor containment is provided with containment isolation valves in accordance with this criterion, as discussed in subsection 6.2.4. Isolation valves outside the containment are located as close to the containment as practical. Upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

3.1.50 CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

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

CONFORMANCE

Each line that penetrates containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere is provided with an appropriate containment isolation valve arrangement. Refer to subsection 6.2.4 for a complete description of the design and operation of the containment isolation system.

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

The nuclear power unit design includes 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 is 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.

CONFORMANCE

Control of waste gas effluents is accomplished by holdup of waste gases in decay tanks until the activity of tank contents and existing environmental conditions permit discharges within Technical Specification requirements. Waste gas effluents are monitored at the point of discharge for radioactivity and rate of flow. Sufficient waste gas holdup capacity is provided, as discussed in section 11.3, to cope with all anticipated operational occurrences and site environmental conditions. A decay tank burst would not result in an activity release greater than 10 CFR 100 limits, based on 1-percent failed fuel.

Control of liquid waste effluents is accomplished by holdup of waste liquids in storage tanks, batch processing of all liquids, and sampling before controlled rate discharge. Liquid effluents are monitored for radioactivity and rate of flow. The liquid waste disposal system tankage and processing capacity, as described in section 11.2, is sufficient to cope with all anticipated operational occurrences and unfavorable site environmental conditions.

Station solid wastes are prepared in batches for offsite disposal by approved contractors in shielded and reinforced containers which meet Federal Regulation requirements. Sufficient handling capacity is provided, as discussed in section 11.5, to cope with all anticipated operational occurrences.

Chapters 11.0 and 15.0 provide additional information.

3.1.52 CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. These systems are designed with a capability to permit appropriate periodic inspection and testing of components important to safety; with suitable shielding for radiation protection; with appropriate containment, confinement, and filtering systems; 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 to prevent significant reduction in fuel storage coolant inventory under accident conditions.

CONFORMANCE

With the exception of spent fuel stored in the ISFSI in accordance with the general license provisions of 10 CFR 72, fuel storage and most waste handling facilities are contained in the auxiliary building; equipment is designed to prevent accidental releases of radioactive material directly to the environment. Components of these systems which are important to safety can be periodically inspected and tested.

The spent-fuel storage pool is designed for the underwater storage of spent-fuel assemblies and control rods after their removal from the reactor. Each unit is designed to accommodate a total of approximately 9 cores. The spent-fuel cask is accommodated in a separate pool adjoining the spent-fuel pool.

The spent-fuel storage racks are located to provide sufficient shielding water over stored fuel assemblies to limit radiation at the surface of the water to no more than 2.5 mrem/h during the storage period. The exposure time during refueling will be limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20.

The waste disposal system is designed to permit controlled handling and disposal of liquid, gaseous, and solid wastes generated during plant operation. The principal design criterion is to ensure that plant personnel are protected against exposure to radiation from wastes in accordance with limits defined in 10 CFR 20. During plant operations members of the public will be protected in accordance with limits defined in technical specifications.

The spent-fuel pool is located within the auxiliary building. The liquid waste processing equipment and the gaseous waste storage and disposal equipment are located within a separate area of the auxiliary building. Both of these areas provide confinement capability in the event of an accidental release of radioactive materials, and both are ventilated with discharges to the vent stack which is monitored. In the event of a fuel handling accident, the ventilation exhaust line will be automatically isolated and the fuel handling area ventilation fans will be automatically secured. The fuel handling area may then be remotely connected with the penetration room filtration system so that the air is processed by the particulate, absolute, and charcoal filters prior to being released through the vent stack.

Radioactive liquid discharged into the cooling tower blowdown is monitored prior to discharge. Any accidental leakage from liquid waste processing equipment is collected and transferred to other tanks to prevent uncontrolled releases to the environment.

The spent-fuel pool cooling system removes the residual heat from the spent-fuel pool. The system is required to handle the heat load from typical core discharges from the reactor as described in table 9.1-1, but it can safely accommodate the heat load from the spent-fuel pool while completely loaded with spent-fuel assemblies.

The spent-fuel pool cooling system design incorporates redundant heat exchangers and pumps, each with 100-percent capability. Normally, one pump draws water from the pool, circulates it through a heat exchanger, and returns it to the pool. Component cooling water cools these heat exchangers.

The spent-fuel storage pool is constructed of reinforced concrete and lined with stainless steel plate. The physical location of the suction and return lines to the storage pool is such that inadvertent loss of coolant from the pool is prevented.

For a more complete description of the design of the fuel storage and handling and radioactive waste systems, see section 9.1 and chapter 11.0.

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

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

CONFORMANCE

Criticality in new and spent-fuel storage areas is prevented by physical separation and administrative controls on placement of fuel assemblies. In addition, the presence of borated water controls the criticality in the spent-fuel storage pool. The separation and administrative controls on placement are provided so that criticality is precluded even if pure water replaces the air gap in the new fuel storage or the borated water in the spent-fuel storage area. Criticality prevention is discussed in paragraph 4.3.2.7.

3.1.54 CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

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

CONFORMANCE

Monitoring systems are provided to alarm on excessive temperature or low water level in the spent-fuel pool. In the event of a high temperature alarm, administrative procedures will provide for checking the cooling water flow to the fuel pool coolers, the operating status of the fuel pool cooling pumps, and the integrity of the fuel pool cooling and purification system. In the event of high airborne gaseous or particulate activity, automatic diversion of the normal fuel building or waste disposal building ventilation system exhaust to the supplementary leak collection and release system will occur.

A radiation monitor and an alarm are provided, as required, to warn personnel of an increase in the level of radiation or airborne activity. The radiation monitoring system is described in chapters 11.0 and 12.0.

3.1.55 CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means are 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.

CONFORMANCE

The containment atmosphere is continuously monitored during normal operation by the containment particulate and gas radiation monitor located outside the containment and by the area radiation monitors located within. In the event of accident conditions, samples of the containment atmosphere will be obtained from the containment post accident sample system for laboratory analysis of particulates, iodine, and noble gases that may be present. During normal operation ventilation systems within areas contiguous to the containment structure are monitored by particulate and/or gas radiation monitors. In addition, the service water outlet from each containment cooler is monitored to ensure that any leakage of radioactive fluids into the service water system will be detected. Radioactivity levels contained in the effluent discharge paths and in the environs are continuously monitored during normal and accident conditions by the plant radiation monitoring system and by the health physics program. Indications and alarms from the plant radiation monitoring system instruments are provided in the control room. The radiation monitoring system is described in chapters 11.0 and 12.0.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

3.2.1 SEISMIC CLASSIFICATION

A two-level system is used for the Seismic Classification of the structures, components, and systems of the facility.

1. Category I structures, components, and systems.

2. Category II structures, components, and systems.

3.2.1.1 Definitions

Structures, components, and systems required for safe shutdown, for immediate or long term core cooling, or for radioactive material confinement following a loss-of-coolant accident (LOCA) to ensure that the public is protected in accordance with 10 CFR 100 guidelines are designed Category I.

Category I structures, components, and systems are designed to withstand the effects of the safe shutdown earthquake (SSE) and 1/2 safe shutdown earthquake (1/2 SSE) as discussed in section 3.7.

When a system as a whole is referred to as Category I, portions not associated with loss of function of the system may be designated as Category II.

Category II structures, components, and systems are those whose failure would not result in the release of significant radioactive material and would not prevent reactor shutdown. All equipment not specifically listed as Category I is included as Category II.

The failure of Category II structures, components, and systems may interrupt power generation.

All Category II structures are designed to conform to Section 2.3.1.4 of the 1970 edition of the Uniform Building Code.

Seismic Classification of structures, systems, and components is in accordance with Regulatory Guide 1.29.

3.2.1.2 Category I Structures

1. Containment.

2. Auxiliary building, including all fuel handling equipment storage areas.

3. Diesel generator building.

4. River intake structure.[a]

5. Intake structure at storage pond.

6. Storage pond dam and dike.

7. Vent stack.[a]

8. Pond spillway structure.

9. Electrical cable tunnel structure.

10. Category I outdoor tanks.

11. Trisodium phosphate baskets in containment.

3.2.1.2.1 Category I Structures with Seismic Restraint Exclusionary Zones

Auxiliary building: Zones or areas, defined on drawings D506531 and D356597, do not have any safety-related systems or components. Within these zones push carts, hand tools, and other such devices may be placed without seismically restraining such devices. No safety-related systems or components may be mounted in or run through these areas.

3.2.1.3 Category I Mechanical Components and Systems

Refer to table 3.2-1 for Category I seismic mechanical components and systems.

3.2.1.4 Category I Electrical Equipment

1. 4160-V switchgear (engineered safeguard buses).

2. 4160-V to 600-V transformers (associated with engineered safeguard systems).

3. 600-V load centers (engineered safeguard buses).

4. 600-V and 208-V motor-control centers (associated with engineered safeguard systems).

a. Not required for safe shutdown of the plant. The original design (Category I) requirements for the river intake structure are no longer required.

5. Direct-current electrical distribution system (auxiliary building and service water building):

   a. 125-V dc station batteries.

   b. Inverters, 125-V dc to 120-V ac (vital ac instrumentation distribution panels).

   c. 125-V dc distribution panels.

   d. 125-v dc switchgear.

   e. 125-v dc battery chargers.

6. Vital ac instrumentation and regulated ac distribution panels.

7. Control panels and control boards:

   a. Auxiliary relay racks.

   b. Solid-state protection system cabinets.

   c. Nuclear instrumentation system cabinets.

   d. Process protection and control system cabinets.

   e. Emergency power board.

8. Cable tray and conduit supports (associated with engineered safeguard systems).

9. Containment penetration assemblies.

10. Direct-current emergency lighting.

11. Diesel generators.

12. Diesel generator control panels.

13. Diesel generator sequencers.

14. Boric acid heat-tracing equipment (functionally nonsafety related).

15. Turbine-driven auxiliary feedwater pump uninterruptable power supply.

3.2.1.5 Category I Instrumentation and Control Systems Equipment

1. Penetration room filtration system.
2. Radiation monitors for containment purge exhaust lines.
3. Radiation monitors for fuel handling area ventilation exhaust line.
4. Post-accident containment combustible gas control system.
5. Component cooling water system.
6. Service water system.
7. Auxiliary feedwater system.
8. Power supply inverters for balance of plant instrument panels.
9. Balance of plant instrument panels.
10. Portions of the sampling system which provide containment isolation and which interface with other Category I systems. (See drawings D-175009, sheet 1, D-175009, sheet 2, D-175009, sheet 3, D-205009, sheet 1, D-205009, sheet 2 and D-205009, sheet 3.)
11. Diesel generator control equipment.

3.2.1.6 Structures and Systems of Mixed Category

None of the structures in the Farley Nuclear Plant have classifications that are partially Category I and partially Category II. The boundaries of the nuclear classes of piping systems are shown on the Piping and Instrumentation Diagrams (P&ID) in sections as listed in table 3.2-3.

Nuclear Safety Classes 1, 2a, 2b, and 3 are designed as Seismic Category I systems. Systems that are Nuclear Safety Class NNS, and all piping systems not otherwise indicated as Category I, are non-seismic. The P&ID legend is shown on drawings D-175016, sheet 1, D-175016, sheet 2, D-175016, sheet 3 and figure 1.7-1.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

The design criteria are tabulated in table 3.2-1 for all mechanical system components. The design in general complies with the intent of Regulatory Guide 1.26. The actual design standards, however, conform to the standards of the American Nuclear Society, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," August 1970 draft. Regulatory Guide 1.26 was not available at the time of initial equipment design and purchase. Whenever practicable, equipment has been purchased to meet ASME Section III standards.

When equipment was purchased before ASME Section III became effective, other design codes, as indicated in table 3.2-1, were used.

The relationship between Safety Class and the ASME Section III Nuclear Class is indicated below.

ANS SAFETY CLASS	**ASME SECTION III NUCLEAR CLASS**
1	1
2a	2
2b	3
3	3
NNS (Non-nuclear safety)	-

The system quality group classifications are delineated on the piping and instrumentation diagrams in chapters 5.0, 6.0, 9.0, 10.0, and 11.0.

TABLE 3.2-1 (SHEET 1 OF 21)

SUMMARY OF CRITERIA - MECHANICAL SYSTEM COMPONENTS

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
REACTOR COOLANT SYSTEM							
Reactor vessel	W	1	III A	C	S	X	X
Full length CRDM housing	W	1	III A	C	S	X	X
Reactor coolant pump assembly (8)	W	1		C	S	X	X
Reactor coolant pump casing	W	1	P&V I	C	S	X	X
Reactor coolant pump internals	W	1	P&V I	C	S	X	X
Steam generator (tube side)	W	1	III A	C	S	X	X
(shell side including integral steam flow restrictor)	W	2a	III A	C	S	X	X
Pressurizer	W	1	III A	C	S	X	X
Reactor coolant piping to pressure boundary	W	1	III 1	C	S	X	X
RC system supports	W	-	-	C	N	X	X
Surge pipe and fittings	W	1	III 1	C	S	X	X
RC thermowells	W	1	III 1	C	S	X	X
Safety valves (16)	W	1	III A	C	S	X	X
Relief valves	W	1	III A	C	S	X	X

TABLE 3.2-1 (SHEET 2 OF 21)

Component		Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Valves to RC system boundary		W	1	P&V I	C	S	X	X
Pressurizer relief tank (11)		W	NSS	VIII	C	S	-	X
CRDM head adapter plugs		W	1	B31.7 I	C	S	X	X
CHEMICAL & VOLUME CONTROL SYSTEM								
Regenerative HX		W	2a	III C	C	S	X	X
Letdown HX	(tube side)	W	2a	III C	AB	S	X	X
	(shell side)		2b	VIII		P	X	X
Mixed bed demineralizer (11)		W	3	VIII	AB	S	X	X
Cation bed demineralizer (11)		W	3	VIII	AB	S	X	X
Reactor coolant filter		W	2a	III C	AB	S	X	X
Volume control tank		W	2a	III C	AB	S	X	X
Charging/high head safety injection pump (8)		W	2a	P&V II(30)	AB	S	X	X
Seal water injection filter		W	2a	III C	AB	S	X	X
Letdown orifices		W	2a	III 2	C	S	-	X
Excess letdown HX	(tube side)	W	2a	III C	C	S	X	X
	(shell side)		2b	VIII		P	X	X
Seal water return filter		W	2a	III C	AB	S	X	X

TABLE 3.2-1 (SHEET 3 OF 21)

Component		Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Seal water HX	(tube side)	W	2a	III C	AB	P	X	X
	(shell side)		2b	VIII			X	X
Boric acid tanks (19)		A	2b	API 650	AB	P	X	X
Boric acid filter (11)		W	2b	III C	AB	P	X	X
Boric acid transfer pump		W	2b	P&V III	AB	P	X	X
Boric acid blender		W	2b	III 3	AB	P	X	X
Resin fill tank (12) (13)		A	NNS	VIII	AB	N	-	X
Boric acid batching tank (13)		W	NNS	VIII	AB	N	-	X
Chemical mixing tank(12) (13)		W	NNS	VIII	AB	N	-	X
Chemical mixing tank orifice		A	NNS	-	AB	N	-	X
RCP No. 1 seal bypass orifice		W	1	III 1	C	S	-	X
Reactor makeup water storage tank		A	2b	III 3	O	-	X	X(23)
Reactor makeup water pump		A	2b	III 3	O	-	X	X(23)(26)
Demineralized water storage tank		A	NNS	-	O	-	-	-
EMERGENCY CORE COOLING SYSTEM								
Accumulators		W	2a	III C	C	P	X	X

TABLE 3.2-1 (SHEET 4 OF 21)

Component		Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
RESIDUAL HEAT REMOVAL SYSTEM								
Residual heat removal/low head safety injection pump (8)		W	2a	P&V II	AB	S	X	X
Residual heat exchanger	(tube side)	W	2a	III C	AB	S	X	X
	(shell side)		2b	VIII	AB	P	X	X
CONTAINMENT SPRAY SYSTEM								
Containment spray pump		W	2a	P&V II	AB	P	X	X
Eductor		W	2a	III 2	AB	P	X[(a)]	X
CONTAINMENT ISOLATION SYSTEM								
Valves		A	2a	P&V II	C,AB	S,P	X	X
CONTAINMENT COOLING SYSTEM								
Fans		A	2b	AMCA (14)	C	N	X	X
Heat exchanger		A	2b	VIII	C	N	X	X
COMPONENT COOLING SYSTEM								
Pumps		A	2b	P&V III	AB	P	X	X
Unit 1 Heat exchangers	(tube side)	A	2b	VIII	AB	N	X	X
	(shell side)	A	2b	VIII	AB	P	X	X

a. The components are included as part of the respective piping system model and seismic analysis.

TABLE 3.2-1 (SHEET 5 OF 21)

Component		Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Unit 2 Heat exchangers	(tube side)	A	2b	III	AB	N	X	X
	(shell side)	A	2b	III	AB	P	X	X
Surge tank (21)		A	2b	API 620	AB	P	X	X
SPENT FUEL POOL COOLING SYSTEM								
Spent fuel pool heat exchanger								
(tube side) (8)		W	2b	III C	AB	S	X	X
(shell side)		W	2b	VIII	AB	P	X	X
Spent fuel pool pump		W	2b	P&V III	AB	S	X	X
Spent fuel pool strainers		W	NNS		AB	S	-	X
Skimmer pump		W	NNS		AB	P	-	X
Spent fuel pool filter (10)(11)		W	NNS	III C	AB	S	-	X
Spent fuel pool demineralizer (10)(11)		W	NNS	III C	AB	S	-	X
BORON THERMAL REGENERATION SUBSYSTEM								
Moderating HX	(tube side)	W	3	VIII	AB	S	X	X
	(shell side)		3	VIII	AB	S	X	X
Letdown chiller HX	(tube side)	W	3	VIII	AB	S	X	X
	(shell side) (10)		NNS	VIII	AB	P	X	X
Letdown reheat HX	(tube side)	W	2a	III C	AB	S	X	X
	(shell side)		3	VIII	AB	S	X	X

TABLE 3.2-1 (SHEET 6 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Thermal regeneration demineralizer	W	3	III 3	AB	S	X	X
Chiller (8)	W	NNS	VIII	AB	N	-	X
Chiller surge tank (10)	W	NNS	VIII	AB	N	-	X
Chiller pumps	W	NNS	P&V III	AB	N	-	X
LIQUID RECYCLE AND WASTE SUBSYSTEM							
Recycle holdup tank (19)	A	NNS	API 650	AB	S	-	X
Recycle evap. feed pump	W	NNS	MS	AB	S	-	X
Recycle evap. feed demineralizer	W	NNS	VIII	AB	S	-	X
Recycle evap. feed filter	W	NNS	VIII	AB	S	-	X
Recycle evaporator	W	NNS	VIII	AB	S	-	X
Recycle evap. condensate demineralizer (10)	W	NNS	VIII	AB	P	-	X
Recycle evap. condensate filter (10)	W	NNS	VIII	AB	P	-	X
Recycle evap. concentrate filter (10)	W	NNS	VIII	AB	S	-	X
Recycle evap. reagent tank (13)	W	NNS	VIII	AB		-	X
R.C. drain tank	W	NNS	VIII	C	S	-	X

TABLE 3.2-1 (SHEET 7 OF 21)

Component		Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
R.C. drain tank pump		W	NNS	MS (N2G21P001 A-N) API-610 (N2G21P001 B-N)	C	S	-	X
R.C. drain tank HX	(tube side) (10)	W	NNS	VIII	C	S	-	X
	(shell side)		2b	III C	C	P	X	X
Waste holdup tank (12)(13)		W	NNS	VIII	AB	S	-	X
Waste evap. feed pump		W	NNS	MS	AB	S	-	X
Waste evap. reagent tank		W	NNS	VIII	AB		-	X
Waste evap. feed filter		W	NNS	VIII	AB	S	-	X
Waste evaporator		W	NNS	VIII	AB	S	-	X
Waste evap. condensate demin. (10)		W	NNS	VIII	AB	P	-	X
Waste evap. condensate filter (10)		W	NNS	VIII	AB	P	-	X
Waste evap. condensate tank (10)(12)		W	NNS	VIII	AB	P	-	X
Waste evap. condensate tank pump		W	NNS	MS	AB	P	-	X
Chemical drain tank (12)(13)		W	NNS	VIII	AB	S	-	X
Spent resin storage tank		W	NNS	VIII	AB	S	-	X

TABLE 3.2-1 (SHEET 8 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Spent resin sluice pump	W	NNS	MS	AB	S	-	X
Spent resin sluice filter	W	NNS	VIII	AB	S	-	X
Laundry and hot shower tank (10)(12)(13)	W	NNS	VIII	AB	P	-	X
Laundry and hot shower tank pump	W	NNS	MS	AB	P	-	X
Laundry and hot shower strainer (10)	W	NNS		AB	P	-	X
Laundry and hot shower filter (10)	W	NNS	VIII	AB	P	-	X
Floor drain tank (10)(12)(13)	W	NNS	VIII	AB	P	-	X
Floor drain tank pump	W	NNS	MS	AB	P	-	X
Waste monitor tank (10)(12)(13)	W	NNS	VIII	AB	P	-	X
Waste monitor tank pump	W	NNS	MS	AB	P	-	X
Waste monitor tank demineralizer (10)	W	NNS	VIII	AB	P	-	X
Waste monitor tank filter (10)	W	NNS	VIII	AB	P	-	X
Containment sump pump	A	NNS	MS	C	P	-	X
ES room sump pump	A	NNS	MS	AB	P	-	X
Drumming header strainer (10)	W	NNS		AB	S	-	X
Floor drain tank filter (10)	W	NNS	VIII	AB	P	-	X

TABLE 3.2-1 (SHEET 9 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Floor drain tank strainer (10)	W	NNS		AB	P	-	X
Disposable demineralizers	A	NNS	MS	AB	S	-	X
Disposable demineralizer pumps	A	NNS	MS	AB	S	-	X
GAS HANDLING SUBSYSTEM							
Gas compressor	W	NNS	VIII/MS	AB	S	-	X
Gas decay tanks	W	NNS	VIII	AB	S	X(D)	X
Hydrogen recombiner	W	NNS	VIII	AB	S	-	X
EMERGENCY DIESEL FUEL OIL SYSTEM							
Transfer pumps	A	2b	P&V III	DB	N	X	X
Fuel oil tanks	A	2b	API 620	B	N	X	X (29)
SERVICE WATER SYSTEM							
Pumps	A	2b	P&V III	S	N	X	X
Strainers	A	2b	VIII	S	N	X	X
Recirc pipe to wetpit	A	2b	VIII	O	N	X	(26)
RIVER WATER SYSTEM							
Pumps	A	NNS	P&V III	R	N	-	-

TABLE 3.2-1 (SHEET 10 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
FUEL HANDLING SYSTEM							
Fuel manipulator crane	W	3	-	AB	N	X	X
Fuel transfer tube (17)	W	2a	-	C/AB	N	X	X
Underwater fuel conveyor car and rail system (18)	W	3	-	AB	N	X	X
Fuel pool bridge crane	W	3	-	AB	N	X	X
Polar crane	A	NNS	(9)	C	N	X	X
Crane supports	A	NNS	-	C	N	X	X
SAMPLING SYSTEM							
Sampler cooler	A	NNS	(28)	AB	S	-	X
Sampler vessel	A	NNS	VIII	AB	S	-	X
Delay coil	A	2a	B31.7 II	C	S	X	X
REFUELING WATER SYSTEM							
Pump	A	NNS	-	AB	P	-	X
Storage tank (20)	A	2a	III 2	O	P	X	X(23)(29)
FIRE PROTECTION SYSTEM							
Fire pumps	A	NNS	(15)	O	N	-	-

TABLE 3.2-1 (SHEET 11 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
CONTAINMENT PURGE SYSTEM							
Fans	A	NNS	-	AB	N	-	X
Filters	A	NNS	-	AB	P	-	X
REACTOR VESSEL SUPPORT COOLING SYSTEM							
Fans	A	NNS	-	C	N	-	X
CONTROL ROD DRIVE MECHANISM COOLING SYSTEM							
Fans	A	NNS	-	C	N	-	X
AUXILIARY BUILDING VENTILATION SYSTEM							
Fan/coil units	A	NNS	-	AB	N	-	X
Filters	A	NNS	-	AB	P	-	X
Pump room air cooling units	A	2b	-	AB	P	X	X
Battery room exhaust fans	A	NNS	AMCA	AB	N	X	X
Battery charger room air cooling units	A	2b	III-3, AMCA	AB	N	X	X
Motor control center and 600 V load center air cooling units	A	2b	III-3, AMCA	AB	N	X	X
600-V load center cooling system fire damper	A	NNS	UL	AB	N	X	X

TABLE 3.2-1 (SHEET 12 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Battery room motor operated dampers	A	NNS	AMCA	AB	N	X	X
PENETRATION ROOM FILTRATION SYSTEM							
Fans	A	2b	AMCA(14)	AB	N	X	X
Filters (HEPA and charcoal)	A	2b	ORNL-NSIC	AB	P	X	X
Backdraft dampers	A	NNS	AMCA	AB	N	X	X
Spent Fuel Pool Area Duct	A	2b	AMCA	O	P	X	(26)
Suction dampers backup air supply piping	A	3	B31.1 (25)	AB	N	X	X
Suction dampers air supply accumulators Pressure relief valves	A	3	VIII (25)	AB	N	X	X
CONTROL ROOM VENTILATION SYSTEM							
Fans	A	2b	AMCA(14)	AB	N	X	X
Filters	A	2b	ORNL-NSIC	AB	P	X	X
Air conditioning unit	A	2b	AMCA, III-3	AB	N	X	X
Motor-operated dampers/valves	A	2b	AMCA, III-3	AB	N	X	X
Balancing dampers	A	NNS	AMCA	AB	N	X	X

TABLE 3.2-1 (SHEET 13 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
Fire dampers	A	NNS	UL	AB	N	X	X
DIESEL BUILDING VENTILATION SYSTEM							
Fans	A	2b	AMCA(14)	DB	N	X	X
Filters	A	2b	--	DB	N	X	X
MAIN STEAM SYSTEM							
Isolation valves	A	2a	P&V II	AB	N	X	X
FEEDWATER SYSTEM							
Isolation valves	A	2a	P&V II	AB	N	X	X
AUXILIARY FEEDWATER SYSTEM							
Auxiliary feedwater pumps Motor driven	A	2b	P&V III	AB	N	X	X
Steam turbine driven	A	2b	P&V III	AB	N	X	(26)
Condensate storage tank	A	2b	III 3	O	P	X	(24)(29)
STEAM DUMP SYSTEMS							
Turbine bypass	A	NNS	-	TB	N	-	-
Relief valves	A	2a	P&V II	AB	N	X	X(29)
Safety valves (16)	A	2a	P&V II	AB	N	X	X(29)

TABLE 3.2-1 (SHEET 14 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
STEAM GENERATOR BLOWDOWN TREATMENT SYSTEM							
Blowdown surge tank	W	NNS	VIII	AB	P	-	X
Blowdown inlet filters	W	NNS	VIII	AB	P	-	X
Blowdown outlet filter	W	NNS	VIII	AB	P	-	X
Blowdown discharge-recycle pumps	W	NNS	VIII	AB	P	-	X
Blowdown heat exchangers	W	NNS	VIII	AB	P	-	X
Blowdown cation demineralizers	W	NNS	VIII	AB	P	-	X
Blowdown mixed bed demineralizers	W	NNS	VIII	AB	P	-	X
Spent resin storage tank	W	NNS	VIII	AB	P	-	X
Spent resin sluice pump	W	NNS	VIII	AB	P	-	X
Spent resin sluice filter	W	NNS	VIII	AB	P	-	X
CONDENSER CIRCULATING WATER SYSTEM							
Circulating water pumps	A	NNS	-	O	N	-	-
COMPRESSED AIR SYSTEM(22)							
Compressors	A	NNS	-	TB	N	-	-

TABLE 3.2-1 (SHEET 15 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
After coolers	A	NNS	-	TB	N	-	-
Air tanks	A	NNS	VIII	TB	N	-	-
Air dryers	A	NNS	-	TB	N	-	-
Piping to safety grade component safety boundary	A	NNS	B31.1	TB/C/AB/O	N	-	-
Air filter	A	NNS	-	TB	N	-	-
Piping to within safety grade component safety boundary	W/A	(SAME AS COMPONENT - SEE APPROPRIATE COMPONENT DESCRIPTION)					
HYDROGEN SYSTEM							
Hydrogen vessels	A	NNS	VIII	O	N	-	-
NITROGEN SYSTEM							
Nitrogen vessels	A	NNS	VIII	O	N	-	-
POST-LOCA HYDROGEN CONTROL SYSTEM							
Post-LOCA hydrogen recombiners	W	2b	-	C	N	X	X
Containment post-LOCA hydrogen mixing system	W	2b	AMCA(14)	C	N	X	X
Post-LOCA, containment hydrogen monitoring equipment	W	2b	III-2	C	P	X	X

TABLE 3.2-1 (SHEET 16 OF 21)

Component	Design Responsibility (1)	ANS Safety Class (2)	Code (3)	Location (4)	Rad Source (5)	Rad Seismic (6)	Tornado (7)
MISCELLANEOUS COMPONENTS							
Diesel generators	A	3	-	DB	N	X	X(29)
Spent fuel pool	A	NNS	-	AB	S	X	X
Vent stack	A	NNS	-	O	N	X	X [(D)]
Spent fuel pool H & V system isolation dampers	A	NNS	AMCA	AB	P	X	X
Containment venting filter units	A	2b	III-3, ORNL	AB	P	X	X

TABLE 3.2-1 (SHEET 17 OF 21)

NOTES

(1)	A	Alabama Power Company
	W	Westinghouse
(2)	1	Safety Class 1 (ANS)
	2a	Safety Class 2a (ANS)
	2b	Safety Class 2b (ANS)
	3	Safety Class 3 (ANS)
	NNS	Non Nuclear Safety (ANS)
(3)	III A	ASME Boiler and Pressure Vessel Code - Section III, Class A
	III C	ASME Boiler and Pressure Vessel Code - Section III, Class C
	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
	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
	B31.1	ANSI B31.1 - Power Piping
	B31.7 I	ANSI B31.7 - Nuclear Power Piping, Class I
	B31.7 II	ANSI B31.7 - Nuclear Power Piping, Class II
	B31.7 III	ANSI B31.7 - Nuclear Power Piping, Class III
	D100	American Waterworks Association, Standard for Steel Tanks, Standpipes, Reservoirs, and Elevated Tanks for Water Storage, AWWA, D100

TABLE 3.2-1 (SHEET 18 OF 21)

	API 610	American Petroleum Institute, Centrifugal Pumps for General Refinery Services
	API 620	American Petroleum Institute Recommended Rules for Design and Construction of Large Welded Low Pressure Storage Tanks
	API 650	American Petroleum Institute, Welded Steel Tanks for Oil
	AMCA	Air Moving and Conditioning Association
	MS	Manufacturer's Standard
	ORNL-NSIC	Oak Ridge National Laboratory, Nuclear Information Center - Design, Construction, and Testing of High Efficiency Air Filtration Systems for Nuclear Application.
	UL	Underwriters Laboratory
(4)	C	Containment
	AB	Auxiliary Building
	TB	Turbine Building
	B	Buried in Ground
	DB	Diesel Generator Building
	R	River Water Intake Structure
	S	Service Water Intake Structure
	O	Outside
(5)	S	Source of radiation
	N	No source of radiation
	P	Possible source of radiation
(6)	X	Category I, (Methods used for seismic analysis of Category I systems and components are presented in table 3.7-4).

TABLE 3.2-1 (SHEET 19 OF 21)

	X(D)	Designed and constructed to the seismic requirements given in Regulatory Guide 1.143, Revision 1 with the exception of the seismic design criteria given in Regulatory Position C.5. The components, systems, and structures are designed to the seismic design criteria given in FSAR section 3.7.
	-	Category II
(7)	X	Protected by virtue of location in a structure designed for tornado wind.
	X(D)	Designed for tornado wind loads.
	-	No protection required
(8)		Portions of equipment containing component cooling water will be analyzed for seismic requirements.
(9)		Crane Manufacturers Association of America Specification No. 70 of 1971.
(10)		National Fire Underwriters and Underwriters Laboratory Certification.
(11)		Designed and fabricated to ASME III C, radiographed and so stamped; however, compliance with ASME VIII would be sufficient for this use.
(12)		Outside jurisdiction of ASME, but designed, fabricated, examined, and tested according to ASME Boiler and Pressure Vessel, Section VIII.
(13)		Built to code but not tested
(14)		Performance test required
(15)		National Fire Protection Association Standard No. 20.
(16)		Will meet pressure-relieving requirements of ASME Section III, Article 9.
(17)		That part which is part of the containment pressure boundary.
(18)		Protect against seismic overturning and possible impaling of fuel.
(19)		Quality control requirements include sidewall and nozzles to tank welds examined by magnetic-particle or liquid-penetrant methods; roof, roof-to-sidewall, and bottom welds visually examined; bottom and bottom-to-sidewall welds vacuum box tested
(20)		Quality control requirements include 100-percent radiograph of sidewall welds; roof, roof-to-sidewall, and bottom welds visually examined; bottom and bottom-

TABLE 3.2-1 (SHEET 20 OF 21)

to-sidewall welds vacuum box tested; bottom-to-sidewall and nozzles-to-tank welds examined by magnetic-particle or liquid penetrant methods

(21) Quality control requirements include sidewall welds 3/16 in. or under examined by magnetic-particle or liquid-penetrant methods; 100-percent radiograph of sidewall welds over 3/16 in.; roof, roof-to-sidewall, bottom, bottom-to-sidewall, and nozzles-to-tank welds examined by magnetic-particle or liquid-penetrant methods; roof and roof-to-sidewall welds soap tested; bottom and bottom-to-sidewall welds vacuum box tested.

(22) The compressed air system includes the instrument air system.

(23) This equipment is surrounded by a concrete wall to protect it from tornado missiles.

(24) In order to ensure the 164,000-gallon reserve required by Technical Specifications, the lower 12 ft of the tanks are designed to withstand ruptures caused by missiles generated by tornadoes. Certain connections to the Unit 1 and Unit 2 CSTs, within the lower 12 ft of the tank, are protected by structures. Reference paragraph 9.2.6.3 for a discussion of these connections.

(25) Deleted

(26) This component or portions of this system are not required to protect from tornado generated missiles per paragraph 3.5.1.2.2.1.

(27) Shall meet 10 CFR 50 Appendix B Requirements

(28) Sample coolers for the RCS sample stream were originally procured in accordance with ASME Code, Section III; however, compliance with ANSI B31.1 is sufficient for this use. The remaining sample coolers are designed in accordance with ANSI B31.1.

(29) This component or portions of this system have been analyzed for vulnerability to tornado generated missiles and found to have an acceptable probability of survival per paragraph 3.5.1.2.2.2.

(30) The 2A charging pump (Q2E21P002A), the 1B charging pump (Q1E21P002B), the 2B charging pump (Q2E21P002B), the 1C charging pump (Q1E21P002C), the 1A charging pump (Q1E21P002A), and the 2C charging pump (Q2E21P002C) casing and discharge head have been replaced. The replacement casing and discharge head were designed to meet the requirements of ASME III 1971 edition with summer 1972 addenda.

TABLE 3.2-1 (SHEET 21 OF 21)

GENERAL NOTES:

1. The safety-related systems outside the reactor coolant pressure boundary may have more than one quality class of piping and valves. Individual valves and sections of piping are assigned quality classes and codes appropriate to their locations and functions and consistent with the assignments of quality classes of system components in table 3.2-1.

2. All pressure-retaining cast parts of Safety Class 1a and 2a pumps and valves are radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic-particle or liquid-penetrant examination is substituted. Examination procedures and acceptance standards are at least equivalent to those specified in the applicable class in the code.

3. The reactor coolant system code requirements, including the applicable addenda, are presented in table 3.2-4.

TABLE 3.2-2

SUMMARY OF QUALITY CLASS REQUIREMENTS - MECHANICAL SYSTEM COMPONENTS

Equipment Category/	Analysis	Limits
1	SSE + NORMAL + LOCA	NLSF, permanent deformation permitted (faulted condition)
	SSE + NORMAL	
	1/2SSE + NORMAL	Applicable code stresses (upset condition)
2a and 2b	SSE + NORMAL	NLSF, permanent deformation permitted(faulted condition)
	1/2SSE + NORMAL	Applicable code stresses (upset condition)
3	SSE + NORMAL	NLSF, permanent deformation permitted(faulted condition)

SSE Safe-shutdown earthquake

1/2SSE 1/2 Safe-shutdown earthquake

NORMAL Those normal operation occurrences which are expected frequently and regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant.

LOCA Loss-of-coolant accident

NLSF No loss of safety function. Permanent deformation permitted to the extent that there is no loss of safety function.

TABLE 3.2-3 (SHEET 1 OF 6)

LISTING OF P&IDs

System	Drawing Number	
Reactor coolant system	D-175037	sheet 1
	D-175037	sheet 2
	D-175037	sheet 3
	D-205037	sheet 1
	D-205037	sheet 2
	D-205037	sheet 3
Residual heat removal system	D-175041	
	D-205041	
Containment cooling and purge system	D-175010	sheet 1
	D-175010	sheet 2
Penetration filtration system	D-175022	
Post-accident containment combustible gas control system	D-175019	
	D-205019	
Safety injection system	D-175038	sheet 1
	D-175038	sheet 2
	D-175038	sheet 3
	D-205038	sheet 1
	D-205038	sheet 2
	D-205038	sheet 3
Auxiliary feedwater system	D-175007	
Spent fuel pool cooling system	D-205043	
River water system	D-170119	sheet 6
	D-170119	sheet 7
Service water system	D-170119	sheet 1
	D-170119	sheet 2
	D-175003	sheet 1
	D-175003	sheet 2
	D-175003	sheet 3
	D-175003	sheet 4
	D-205003	sheet 1
	D-205003	sheet 2
	D-205003	sheet 3

TABLE 3.2-3 (SHEET 2 OF 6)

System	Drawing Number	
Component cooling water system	D-175002	sheet 1
	D-175002	sheet 2
	D-175002	sheet 3
	D-205002	sheet 1
	D-205002	sheet 2
	D-205002	sheet 3
Demineralized makeup water system	D-175047	sheet 1
	D-175047	sheet 2
	D-205047	
Potable and sanitary water system	D-170127	
Reactor makeup water system	D-175036	
	D-205036	
Plant water treatment system	figure 9.2-11	
Well water system	D-170110	sheet 1
Compressed air system	D-170131	sheet 1
	D-170131	sheet 2
	D-200019	sheet 1
	D-200019	sheet 2
Service air system	D-175035	sheet 1
	D-205035	
Instrument air system	D-175034	sheet 1
	D-175034	sheet 2
	D-175034	sheet 3
	D-205034	sheet 1
	D-205034	sheet 2
	D-205034	sheet 3
	D-205034	sheet 4
Sampling system	D-175009	sheet 1
	D-175009	sheet 2
	D-175009	sheet 3
	D-205009	sheet 1
	D-205009	sheet 2
	D-205009	sheet 3

TABLE 3.2-3 (SHEET 3 OF 6)

System	Drawing Number	
Nonradioactive drains and vents	D-175005	
Radioactive drains and vents	D-175004	sheet 1
	D-175004	sheet 2
Chemical and volume control system	D-175039	sheet 1
	D-175039	sheet 2
	D-175039	sheet 3
	D-175039	sheet 4
	D-175039	sheet 5
	D-175039	sheet 6
	D-175039	sheet 7
	D-205039	sheet 1
	D-205039	sheet 2
	D-205039	sheet 3
	D-205039	sheet 4
	D-205039	sheet 5
Boron thermal regeneration system	D-175040	sheet 1
	D-205040	
HVAC and filtration system (control room and computer room)	D-175012	
	D-205012	
Nonradioactive area heating, ventilation system	D-175014	sheet 1
	D-175014	sheet 2
	D-205014	sheet 1
	D-205014	sheet 2
Spent fuel pool ventilation	D-175045	
	D-205045	
Access control area heating, ventilating, and air conditioning	D-175001	
Engineered safety feature pump rooms ventilating and filtration system	D-175001	
Radwaste area heating, ventilating and filtration system	D-175011	sheet 1
	D-175011	sheet 2
	D-175011	sheet 3

TABLE 3.2-3 (SHEET 4 OF 6)

System	Drawing Number	
	D-205011	sheet 1
	D-205011	sheet 2
	D-205011	sheet 3
	D-205011	sheet 4
Turbine building chilled water system	D-175031	sheet 1
	D-175031	sheet 2
	D-175031	sheet 3
	D-205031	sheet 1
	D-205031	sheet 2
	D-205031	sheet 3
Turbine building heating, ventilating, air conditioning, and filtration system	D-175027	
Communication system	D-177331	
	D-177334	sheet 1
	D-177334	sheet 2
	D-177334	sheet 3
	D-175335	
	D-177336	
	D-177337	sheet 1
	D-177337	sheet 2
	D-177337	sheet 3
	D-177338	
	D-177339	
	D-207331	
	D-207334	sheet 1
	D-207334	sheet 2
	D-207336	
	D-207337	sheet 1
	D-207337	sheet 2
	D-207339	
Diesel generator fuel oil system	D-170060	
Diesel generator cooling water system	D-170119	sheet 3
	D-200013	sheet 3
Diesel generator starting air system	D-170806	sheet 1
Diesel generator starting air and control air systems	D-170807	sheet 1

TABLE 3.2-3 (SHEET 5 OF 6)

System	Drawing Number	
Extraction steam system	D-200009	
Main stream system	D-175033	sheet 1
	D-175033	sheet 2
	D-170114	sheet 1
	D-170114	sheet 2
	D-205033	sheet 1
	D-205033	sheet 2
	D-200007	
Chemical injection system	D-175000	sheet 1
	D-175000	sheet 2
Main condenser vacuum system	D-170064	
Circulation water system	D-170119	sheet 9
	D-170119	sheet 10
	D-200013	sheet 6
Condensate and feedwater system	D-170117	sheet 1
	D-170117	sheet 2
	D-170117	sheet 3
	D-170117	sheet 4
	D-175073	
	D-200011	sheet 1
	D-200011	sheet 2
	D-200011	sheet 3
	D-205073	
Steam generator blowdown processing system	D-175071	sheet 1
	D-175071	sheet 2
	D-175071	sheet 3
	D-205071	sheet 1
	D-205071	sheet 2
	D-205071	sheet 3
Waste processing system	D-175042	sheet 1
	D-175042	sheet 2
	D-175042	sheet 3
	D-175042	sheet 4
	D-175042	sheet 5
	D-175042	sheet 6

TABLE 3.2-3 (SHEET 6 OF 6)

System	Drawing Number	
	D-175042	sheet 7
	D-175042	sheet 11
	D-175042	sheet 12
	D-205042	sheet 1
	D-205042	sheet 2
	D-205042	sheet 3
	D-205042	sheet 4
	D-205042	sheet 5
	D-205042	sheet 6
	D-205042	sheet 7
	D-205042	sheet 9
	D-205042	sheet 10

TABLE 3.2-4

COMPONENT CODING

Component	Code	Edition	Applicable/Addenda
Reactor vessel	ASME III[a] Class A	1968	through Summer 1970
Full length CRD mechanisms	ASME III Class A	1968	through Winter 1969
Steam Generators			
Tube side	ASME III, Class A	1989	No Addenda
Shell side	ASME III, Class A		
Pressurizer	ASME III, Class A	1968	through Summer 1970
Reactor coolant pump casing	Nuclear Pump and Valve Code[b]	1968	through March 1970
Piping and fittings	ASME III, Class 1[c]	1971	through Summer 1971

a. ASME Boiler and Pressure Vessel Code Section III, Nuclear Vessels, including applicable mandatory Addenda.

Code Cases are not mandatory until included in a mandatory Addendum to the Code. The designer does not require that all Code Cases be applied. Where a specific Code Case is required by the designer it will be identified in the technical requirements. Where a supplier presents justification for applying a specific Code Case, the designer will review the justification and approve or disapprove the request.

Hardship exceptions to 10 CFR Part 50.55a are presented in Table 5.2-1.

b. ASME Code of Pumps and Valves for Nuclear Power. Reactor coolant pump casing in Unit 2 is to ASME III, 1971 ed.

c. Three 31-in., 90-degree vane elbows manufactured by Mitsubishi Steel Manufacturing Company, Nagasaki, Japan, meet requirements of ASME III, Class 1, through Summer 1971 addenda, except for "N" Stamp (Unit 1 only). At time of procurement of these fittings, Mitsubishi did not have "N" Stamp due to ASME not recognizing foreign manufacturers.

TABLE 3.2-5 (SHEET 1 OF 4)

ASME CODE CASES FOR CLASS 1 COMPONENTS

Code/Case	Title
1141	Foreign Produced Steel
1332	Requirements for Steel Forgings
1334	Requirements for Corrosion Resistant Steel Bars
1335	Requirements for Bolting Material
1337	Requirements for Special Type 403 Modified Forgings or Bars (Section III)
1344	Requirements for Nickel-Chromium Age-Hardenable Alloys
1345	Requirements for Nickel-Molybdenum-Chromium-Iron Alloys
1355	Electroslag Welding
1361	Socket Welds
1364	Ultrasonic Transducers SA-435 (Section II)
1384	Requirements for Precipitation Hardening Alloy Bars and Forgings
1388	Requirements for Stainless Steel-Precipitation Hardening
1390	Requirements for Nickel-Chromium, Age-Hardenable Alloys for Bolting
1395	SA-508, Class 2 Forgings-Modified Manganese Content
1401	Welding Repair, to Cladding
1407	Time of Examination
1423	Plate, Wrought Type 304 and 316 with Nitrogen Added
1433	Forgings, SA-387

TABLE 3.2-5 (SHEET 2 OF 4)

Code/Case	Title
1434	Class 8N Steel Casting (Postweld Heat Treatment for SA-487)
1448	Use of Case Interpretations of ANSI B31 Code for Pressure Piping
1456	Substitution of Ultrasonic Examination
1459	Welding Repairs to Base Metal
1461	Electron Beam Welding
1470	External Pressure Charts for Low Alloy Steel
1471	Vacuum Electron Beam Welding of Tube Sheet Joints
1474	Integrally Finned Tubes (Section III)
1477	B-31.7, ANSI 1970 Addenda
1484	SB-163 Nickel-Chromium-Iron Tubing at a Specified Minimum Yield Strength of 40,000 psi
1487	Evaluation of Nuclear Piping for Faulted Conditions
1492	Postweld Heat Treatment
1493	Postweld Heat Treatment
1494	Weld Procedure Qualification Test
1495	Stress Indices in Table NB-3681.2-1
1498	SA-508, Class 2, Minimum Tempering Temperature
1501	Use of SA-453 Bolts in Service Below 800 F without Stress Rupture Tests
1504	Electrical and Mechanical Penetration Assemblies
1508	Allowable Stresses, Design Stress Intensity and/or Yield Strength Values
1514	Fracture Toughness Requirements

TABLE 3.2-5 (SHEET 3 OF 4)

Code/Case	Title
1515	Ultrasonic Examination of Ring Forgings for Shell Section of Section III-Class 1 Vessels
1516	Welding of Non-Integral Seats in Valves for Section III Application
1517	Material Used in Pipe Fittings
1519	Use of A-105-71 in lieu of SA-105
1521	Use of H. Grades SA-240, SA-479, SA-336 and SA-358
1522	ASTM Material Specifications
1524	Piping 2-in. NPS and Smaller
1525	Pipe Descaled by Other Than Pickling
1526	Elimination of Surface Defects
1527	Integrally Finned Tubes
1528	High Strength SA-508 Class 2 and SA-541 Class 2 Forgings for Section III Construction of Class 1 Components
1529	Material for Instrument Line Fittings
1531	Electrical Penetrations, Special Alloys for Electrical Penetration Seals
1534	Overpressurization of Valves
1535	Hydrostatic Test of Class 1 Nuclear Valves
1539	Metal Bellows and Metal Diaphragm Steam
1542	Requirements for Type 403 Modified Forgings or Bars for Bolting Material
1544	Radiographic Acceptance Standards for Repair Welds
1545	Test Specimens from Separate Forgings for Class 1, 2, 3 and MC

TABLE 3.2-5 (SHEET 4 OF 4)

Code/Case	Title
1546	Fracture Toughness Test for Weld Metal Section
1547	Weld Procedure Qualification Tests; Impact Testing Requirements, Class 1
1552	Design by Analysis of Section III Class 1 Valves
1557	Plate Steel Refined by Electroslag Remelting
1567	Test Lots for Low Alloy Steel Electrodes
1568	Test Lots for Low Alloy Steel Electrodes
1571	Materials for Instrument Line Fittings, for SA-234 Carbon Steel Fittings
1573	Vacuum Relief Valves
1574	Hydrostatic Test Pressure for Safety Relief Valves
1621	Line Valve Internal and External Items Section III, Class 1, 2, and 3
1690	Stock Materials for Section III Construction

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

Wind loadings for Category I structures have been selected on the basis of ASCE Paper No. 3269, "Wind Forces on Structures"[1] or as provided in "TORMIS Missile Risk Analysis for Farley Nuclear Plant Units 1 and 2."[3]

3.3.1.1 Design Wind Velocity

Category I structures are designed to withstand a basic wind velocity of 115 mph. The recurrence interval of this wind velocity is estimated to be at least 100 years.[1] The variation of wind velocity with height is shown in table 3.3-1.

3.3.1.2 Basis for Wind Velocity Selection

The "fastest mile of wind" at the Farley Plant site is shown, according to Figure 1 (b) and the ASCE paper,[1] to be 90 mph. As a result of recent hurricane experiences on the Gulf Coast, a design velocity of 105 mph at ground level was selected. For additional conservatism, to account for uncertainties in historical data, this margin of safety has been increased and a ground level wind of 115 mph has been used as the basic design wind.

3.3.1.3 Vertical Velocity Distribution and Gust Factor

The wind pressures resulting from the wind velocities shown in table 3.3-1 incorporate the shape factors in both horizontal and vertical directions. A gust factor of 1.1 has been selected for the design and has been incorporated into the wind pressures shown in table 3.3-1.

The gust factor of 1.1 is selected on the basis of ASCE paper No. 3269, "Wind Forces on Structures."[1] This paper recommends that appropriate gust factors be used for structures that are small enough to be responsive to gusts involving less than 1 mile of passing wind, and that the gust factors bear some relation to the minimum size of gust necessary to envelop the structure and its accompanying pattern of flow. A gust factor of 1.1 will allow for gust of approximately 10-second duration which, in a 115-mph basic wind, would have a length downwind of about 1,700 ft; this factor is adequate for structures having a horizontal dimension, transverse to the wind, of 125 ft and larger.

3.3.1.4 Determination of Applied Forces

The design wind dynamic pressure is calculated by

$$q = 0.002558\ V^2$$

where q = pressure in psf

V = velocity in mph

A shape coefficient of 1.3 is applied for building wind loads. Of the total of 1.3 q , 0.9 q is applied as positive pressure to the windward walls, and 0.4 q is applied as negative pressure on the leeward walls, where applicable. A shape factor of 0.6 is applied for plant vent stack wind loads.

Wind loads are applied to the structures as uniform static loads on the surface area normal to the wind.

The applied force magnitude and distribution calculated for Category I structures are shown on figure 3.3-1.

3.3.2 TORNADO LOADINGS

All above ground Category I structures required to ensure the integrity of the reactor coolant pressure boundary, safe shutdown of the plant, long-term core cooling, or to prevent radioactive releases resulting in offsite exposures comparable to 10 CFR 100 guidelines are also designed to withstand tornado loadings and tornado generated missiles[(2)] or have been analyzed as discussed in paragraph 3.5.1.2.

3.3.2.1 Applicable Design Parameters

For Category I structures designed to withstand tornadoes and tornado generated missiles, the three following parameters are applied concurrently, in combinations producing the most critical conditions:

a) Dynamic Wind Pressure

The dynamic wind pressure is caused by a tornado funnel having a peripheral tangential velocity of 300 mph and a forward progression of 60 mph. The applicable portions of wind design methods described in ASCE Paper No. 3269 are used, particularly for shape factors. The provisions for gust factors and variation of wind velocity with height are not applied. The average tornado design dynamic wind pressure is q = 230 psf based on an average wind velocity of 300 mph.

b) Pressure Differential

The structure interior bursting pressure is taken as rising 1 psi/s for 3 seconds, followed by a 2-second calm, them decreasing at 1 psi/s for 3 seconds. This cycle accounts for reduced pressure in the eye of a passing tornado. All fully enclosed Category I structures are designed to withstand the full 3 psi pressure differential.

c) Missile Impingement

A tornado missile is defined as any object set in motion and propelled by a tornado. Three types of tornado missiles are considered; each type is assumed to act independently and only one type may be generated at any one time. It is also assumed that the missiles do not tumble while in flight, and are at any time oriented to have the maximum value:

$$\frac{C_d A}{W}$$

where

C_d = Drag coefficient

A = Projected area of missile exposed to wind

W = Weight of missile

The three types of missiles are as follows:

1. A 12-ft-long piece of wood 8 in. in diameter (114 lb) traveling end-on at a speed of 300 mph and striking the structure at any elevation.

2. A 10-ft-long steel pipe, schedule 40, 3 in. in diameter (75.8 lb), traveling end-on at a speed of 100 mph and striking the structure at any elevation.

3. A 4,000-lb automobile, traveling end-on at a speed of 50 mph and striking the structure on an impact area of 20 sq ft, with any portion of the impact area being not more than 25 ft above grade.

3.3.2.2 Determination of Forces on Structures

Tornado loads are applied to the Category I structures in the same manner as the wind loads described in subsection 3.3.1.4 with the exception that gust factor and variation of wind velocity with height do not apply. The load combinations involving tornadoes are given in subsections 3.8.1.3, 3.8.4.3, and 3.8.5.3.

The load factor selected for tornado loadings is 1.0, based on the short duration of the loading condition, the low probability of a tornado striking a specific geographic point, and the degree of conservatism in the selection of design tornado velocity. This subject is discussed in B-TOP-3.

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

Failure of Category II structures not designed for tornado loads will not affect the ability of Category I structures to perform their functions for the following reasons:

a. Tornado missiles that may be formed by the failure of Category II structures will not exceed the force of those postulated and described in subsection 3.3.2.1, against which Category I structures are designed.

b. The structural frame of the Category II turbine building in the vicinity of the auxiliary building has been designed against collapse when subjected to tornado loadings.

REFERENCES

1. "Wind Forces on Structures," Transactions of the ASCE, Paper No. 3269, 1961.

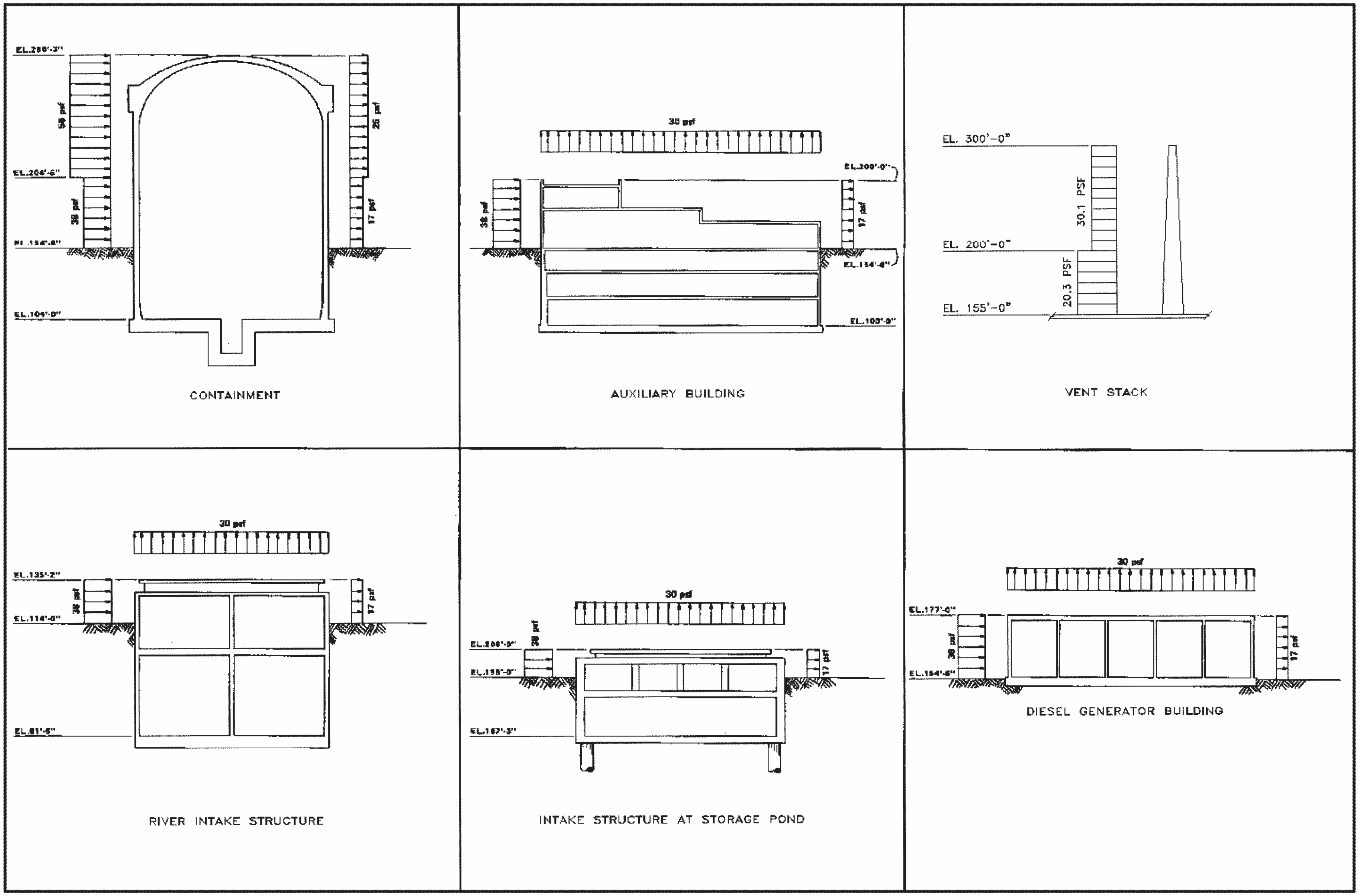
2. "Design Criteria for Nuclear Power Plants Against Tornadoes," Bechtel Topical Report, B-TOP-3, March 1970.

3. "TORMIS Missile Risk Analysis for Farley Nuclear Plant Units 1 and 2," ARA Report 4733, March 1999.

TABLE 3.3-1

WIND LOADS WITH GUST FACTOR

Height (ft)	Velocity (mph)	Dynamic Pressure q (psf)	Wall Load (psf) Pressure 0.9q	Wall Load (psf) Suction 0.4q	Roof Load (psf) Suction 0.7q
0-50	115	42	38	17	30
50-150	140	62	56	25	44
150-400	170	91	82	37	64



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WIND FORCES AND DISTRIBUTION ON CATEGORY I STRUCTURES

FIGURE 3.3-1

3.4 WATER LEVEL (FLOOD) DESIGN

All Category I structures are designed to protect the safety related systems, equipment, and components from the respective probable maximum flood and/or the highest groundwater levels.

Three probable maximum flood levels are used. The design bases for the PMF elevation of 127 ft are presented in subsections 2.4.2.1 and 2.4.2.2. The design bases for the PMF elevations of 144.2 ft and 192.2 ft are given in subsections 2.4.2.2 and 2.4.8.1, respectively.

3.4.1 FLOOD PROTECTION

The design maximum flood elevations for Category I structures are as follows:

1. Containment structure - elevation 144.2 ft

2. Auxiliary building - elevation 144.2 ft

3. Diesel generator building - elevation 144.2 ft

4. Electrical cable tunnel structure - elevation 144.2 ft

5. Category I outdoor tanks - elevation 144.2 ft

6. River intake structure - elevation 127 ft[a]

7. Intake structure at storage pond - elevation 192.2 ft

8. Pond spillway structure - elevation 192.2 ft

9. Storage pond dam and dike - elevation 192.2 ft

The safety-related systems, equipment, and components are protected against floods by virtue of their being located in flood protected structures. (See table 3.2-1.) These systems, equipment, and components, except those located in the river intake structure and the intake structure at the storage pond, are located on, above, or flood protected to the plant grade elevation of 154.5 ft. The systems, equipment, and components located in the river intake structure and intake structure at the storage pond are flood protected to elevation 127.0 ft and elevation 195.0 ft, respectively.

Descriptions of the Category I structures which house the safety-related systems, equipment, and components are given in subsections 3.8.1.1 and 3.8.4.1.

---

a. Original design (Category I) requirements are no longer required.

All exterior or access openings and penetrations are flood protected by watertight concrete walls to grade elevations which are 154.5 ft for the plant area and 195.0 ft for the storage pond area, respectively. Access to all Category I structures is possible only from above grade levels.

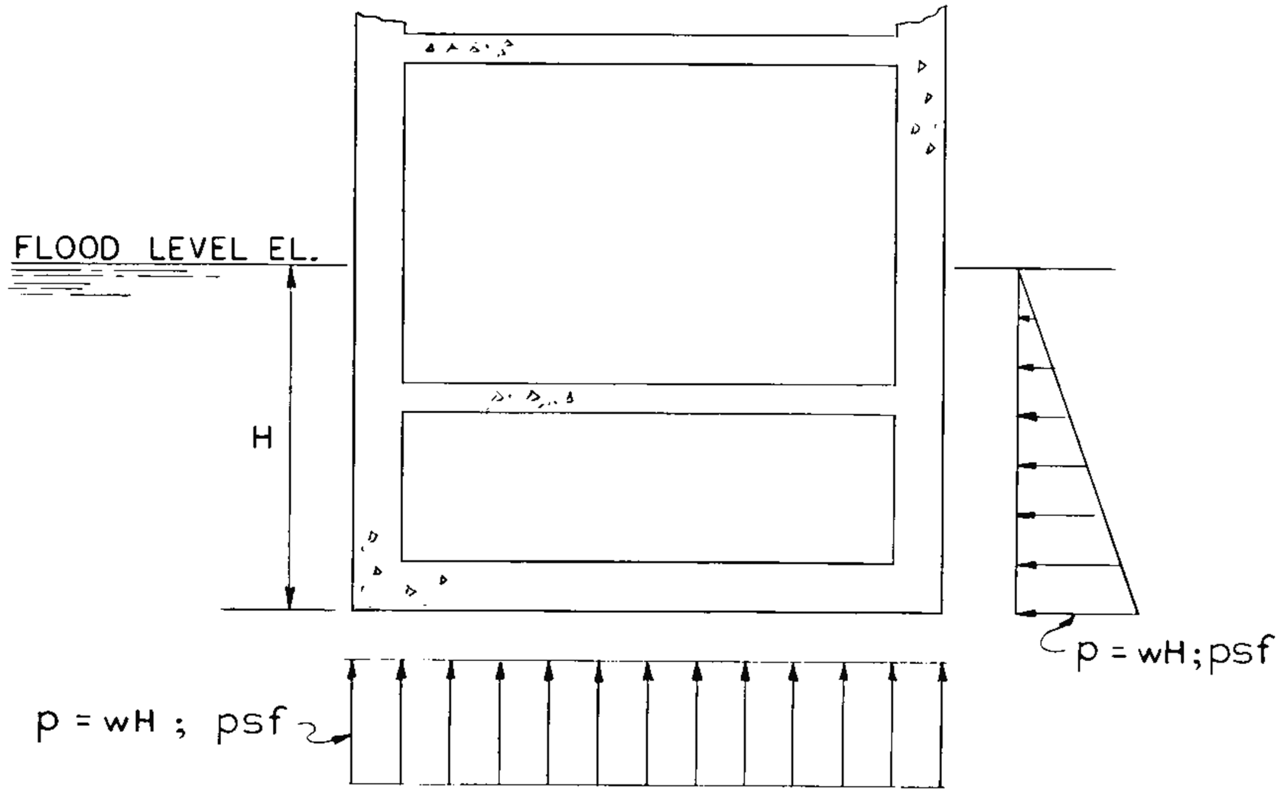
3.4.2 ANALYSIS PROCEDURES

The foundation slabs and exterior walls of the structures are designed to resist the upward and the lateral pressures caused by the maximum flood levels given in Section 3.4.1.

The hydrostatic pressure acting uniformly at the bottom of the structures is the product of the height to the design flood level and the weight of water which is taken as 63 lb/ft^3 (See figure 3.4-1.)

The horizontal pressure acting on the exterior walls varies with height, from the maximum at the bottom of the wall to zero at the design flood level. (See figure 3.4-1).

Dynamic water forces associated with phenomena such as flood currents, wind waves, hurricanes, and tsunamis are not considered in the design of the Category I structures. Justifications for their omission are given in subsections 2.4.3.6, 2.4.5 and 2.4.6.



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WATER PRESSURE ON STRUCTURES

FIGURE 3.4-1

3.5 MISSILE PROTECTION

Category I structures are designed to protect safety-related equipment and components from being damaged by internal and external missiles.

3.5.1 MISSILE BARRIERS AND LOADINGS

The missile barriers are designed to resist the missiles selected in subsection 3.5.2.

3.5.1.1 Accident/Incident Generated Missiles Inside Containment

A tabulation of barriers and the missiles they have been designed to contain is given in table 3.5-1. The postulated missile loadings are derived from the physical characteristics of the components involved and their respective kinetic energy levels. They are given in tables 3.5-2 through 3.5-5. The analytical method used to convert energies into forces and depths of penetration necessary to barrier design is described in subsection 3.5.4.

3.5.1.2 Environmental Load Generated Missiles

3.5.1.2.1 Missile Protection Methods

Those systems or components listed in table 3.2-1 that are required for safe shutdown, for immediate or long term core cooling, or to prevent a radioactive release resulting in offsite exposures comparable to 10 CFR 100 guidelines are provided with tornado missile protection by location within Category I structures, burial underground, or missile barriers/shielding or have been analyzed as discussed in paragraph 3.5.1.2.2.

Category I structures housing equipment and components vital to a safe shutdown have been designed against penetration by the tornado missiles described in paragraph 3.3.2.1 (c). These structures, having at least 2-ft-thick concrete exterior walls and roof slabs, constitute barriers against missile penetration. Calculations show that the deepest missile penetration of the concrete barriers would be 10 in. Therefore, the 2-ft-thick slabs provide ample protection. Where concrete spalling due to missile impact is considered, the inside surfaces of the following areas have been protected with corrugated sheet metal:

- Control room.
- HVAC equipment room for the control room.
- Component cooling water surge tank room.
- Spent-fuel pool area.

3.5.1.2.2 Components Not Requiring Unique Missile Protection

Certain Seismic Category I systems and components located outside of Seismic Category I structures are evaluated as not requiring unique tornado missile protection by burial or barriers. The following two approaches are used in the evaluation of these systems and components relative to a tornado event.

3.5.1.2.2.1 <u>Components Not Required for a Tornado Event</u>. The probability of occurrence of a tornado event coincident with another low probability design basis event is so small that no protection from tornado missiles is required for certain Seismic Category I structures, systems, and components which are not otherwise needed for safe shutdown; for immediate or long-term core cooling; to prevent a radioactive release resulting in offsite exposures comparable to 10 CFR 100 guidelines; or to support other systems or components which are required for one of those functions.

3.5.1.2.2.2 <u>Components with Acceptable Probability of Survival</u>. Safety-related systems and components required for safe shutdown, for immediate or long-term core cooling, or to prevent a radioactive release resulting in offsite exposures comparable to 10 CFR 100 guidelines required for a tornado event are generally protected. A limited amount of unprotected portions of these systems and components is analyzed using probabilistic missile damage analysis as permitted in Standard Review Plan 3.5.1.4, “Missiles Generated By Natural Phenomena.” This analysis is conducted to determine the probability per year of missiles generated by postulated tornadoes striking and damaging these systems and components beyond their failure point. For FNP, the specific acceptance criterion for tornado damage for the unprotected systems and components required for a tornado event is that the cumulative sum of the mean failure probabilities for these systems and components be less than 10^{-6} per year per unit. The allowable level of less than 10^{-6} per year per unit for the cumulative probability of failure of such systems and components is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

The analysis used for FNP is the computer program TORMIS[3][5][6], developed by the Electric Power Research Institute (EPRI)[4] and accepted by the NRC.

Systems and components whose analysis using the TORMIS methodology yield results that cause the less than 10^{-6} per year per unit acceptance criterion to be exceeded will be provided with unique barriers to reduce the total failure probability value to below the acceptance criterion.

3.5.1.2.3 TORMIS Methodology

TORMIS [3][4][5][6] is a methodology developed to predict the probability of damage to nuclear power plant structures and components from tornadoes. There are four fundamental models in the TORMIS analysis: wind hazard, site facility, load effects, and system models. Monte Carlo simulation is used to produce numerical estimates of hit and damage probabilities based on the site-specific models.
The wind hazard analysis for FNP Units 1 and 2 uses a site-specific analysis to generate a tornado hazard curve specifically for Farley.

The site facility model was conservatively developed based on a site area walkdown and the specific characteristics, materials, and failure points for Farley structures and components.

Load effects are determined based on the TORMIS model missiles, missile transport model, and component characteristics. The missiles utilized in the TORMIS model encompass the three design basis missiles described in paragraph 3.3.2.1.

TORMIS implements a methodology developed by EPRI. TORMIS determines the probability of striking walls and roofs of buildings on which penetrations or exposed portions of systems/ components are located. The probability is calculated by simulating a large number of tornado strike events at the site for each tornado wind speed intensity scale. After the probability of striking the walls or roof is calculated, the exposed surface area of the particular components is factored in to compute the probability of striking and consequently damaging a particular item.

The following provisions apply to the TORMIS analysis for FNP:

1. FSAR paragraph 2.3.1.3 estimates an occurrence rate of 3 tornadoes per year per 1-degree square (approximately 4000 square miles), or 7.5 E-04 tornadoes per square mile. This occurrence rate was based on conservative treatment of data from 1955 through 1967. As part of the FNP TORMIS analysis, the annual probability of a tornado was determined for the Fujita F-scale wind speeds using regional data available in TORMIS for NRC Region II. A site-specific analysis was performed to generate a tornado data set for the TORMIS analysis of Farley.

   The National Climatic Data Center (NCDC) files for the years 1950 through 1996 were used as the basic source of data for this investigation. These data were screened to eliminate coding errors in the record fields. In addition, corrections were introduced to account for reporting efficiency and time series, or other potential errors resulting from the indirect characteristics of the available data. The overall tornado occurrence rate computed from the updated regional data in TORMIS is 6.0 E-04 tornadoes per square mile per year. While this rate is slightly lower than the rate in FSAR paragraph 2.3.1.3, it is a more accurate figure based on conservative treatment of the best available data and will therefore be used in lieu of the data cited in FSAR paragraph 2.3.1.3 for those components or portions of systems analyzed in TORMIS.

2. The Fujita scale (F-scale) wind speeds will be used in lieu of the TORMIS wind speeds (F-scale) for the F0 through F5 intensities.

3. The tornado windfield parameters in the FNP TORMIS analysis were adjusted to increase the wind profile in the lowest 10 m over the original profile in TORMIS. This adjustment applied the ratio of V_0/V_{33} in a conservative manner in accordance with the NRC's October 26, 1983, TORMIS SER.

4. Detailed surveys of the plant site were performed to characterize and quantify potential missiles for use in the FNP TORMIS analysis. To ensure conservatism, these surveys were performed during a refueling outage when large amounts of material were temporarily stored in outside laydown areas around the site. Additionally, ground and aerial photographs were reviewed to estimate the number and type of missiles which

could originate from remote areas of the site. The total number of missiles used in the FNP TORMIS analysis was 51,864.

5. The FNP analysis will not deviate from the TORMIS program as described in reference 4 of FSAR section 3.5, except as noted in items 1 through 4 above.

3.5.1.3 Site Proximity Missiles

There are no guided missile installations in the vicinity of the Farley Nuclear Plant.

At the time of construction of Farley Nuclear Plant the only landing strip within a radius of 5 miles from the Farley site was a 3000-ft landing strip for the paper company at Cedar Springs, Georgia, approximately 3.5 miles south of the plant site.

Aircraft using the strip were light, twin-engined business planes comparable to a Cessna 401-A, which has a gross weight of 6300 lb. The orientation of this landing strip was N 30 degrees E; therefore, takeoffs and landing approaches were not in the direction of the Farley Nuclear Plant site. The landing strip is now abandoned.

A new 5400-ft landing strip, capable of handling jet engined aircraft, has been constructed by the paper company at Cedar Springs, Georgia. The new strip is located approximately 4 to 5 miles south of the old landing strip and 7 to 8 miles from the Farley site. The strip has approaches oriented NW and SE and is used by jet aircraft as well as conventional aircraft. The jet aircraft are six-to-eight-passenger business jets comparable to a Lear Jet Model 23, which has a gross weight of 12,500 lb. The paper company at Cedar Springs, Georgia, has indicated that pilots will be instructed to avoid the Farley Nuclear Plant site area during both takeoffs and landing operations.

For these reasons, aircraft generated missiles are not considered.

3.5.1.4 Accident/Incident Generated Missiles Inside Category I Structures Other than Containment

A tabulation of barriers and the missiles they have been designed to contain is given in table 3.5-6. The postulated missile loadings for the rod drive motor generator sets are derived from the physical characteristics of these components and their respective kinetic energy levels, as given in table 3.5-7.

3.5.2 MISSILE SELECTION

3.5.2.1 Missile Selection Within the Containment

The systems located inside the containment have been examined to identify and select potential missiles. The basic approach was to ensure design adequacy against generation of missiles, rather than allow missile formation and then contain their effects.

The following components have been considered to have a potential for missile generation:

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

b. Certain valves defined below.

c. Temperature and pressure element assemblies.

The limiting case considered for design is a drive shaft ejected from the reactor through the top of the rod travel housing. 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 ft 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 structure provided.

The valves considered for missile potential are those in the region where the pressurizer extends above the operating deck, such as 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 considered improbable, failure of the valve bonnet body bolts, nevertheless, has been considered and provisions made to ensure integrity of the containment liner from the resultant bonnet missile.

The only probable source of jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system is the type represented by the temperature and pressure element assemblies. The resistance temperature element assemblies can be of two types: "with well" and "without well". Two rupture locations have been assumed for each type of temperature element assembly: one around the weld between the boss and the pipe wall for each assembly, and another at the weld (or thread) between the temperature element assembly and the boss for the "without well" element or the weld (or thread) between the well and the boss for the "with well" element.

A temperature element 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 by a steel plate. In evaluating missile potential, it is assumed that this plate could break and the pipe plug on the external end of the hole could become a missile.

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

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

3.5.2.2 Missiles Selected Outside the Containment

The tornado generated missiles selected for the design of the Farley Plant structures are described in subsection 3.3.2.

3.5.2.3 Missile Selection Within Category I Structures Other Than Containment

The systems located inside Category I structures other than the containment have been examined to identify potential missiles. The following components are considered to have a potential for missile generation:

a. Flywheels of two rod drive power supply motor generator sets.

The electric motors of the rod drive power supply motor generator sets are designed to operate at 1800 rpm. In the unlikely event of an overspeed generated flywheel missile, the steel protective shield which closely encircles the flywheel would contain the missile and prevent it from impacting any safety-related components.

The steel protective shields are designed to contain a spectrum of probable flywheel fragment missiles generated at an overspeed of 150 percent of the operating speed as indicated in table 3.5-7. For conservation, the initial translational energy of the governing missile is increased by 10 percent for the design of the steel protective shield.

3.5.3 SELECTED MISSILES

The missiles selected inside the containment are given in tables 3.5-1 through 3.5-5.

The origin, weight, impact velocity, impact area, and all other parameters necessary to determine the missile penetration are listed in these tables. The calculated depth of penetration into a 2-ft-thick concrete slab is also given.

The missiles selected outside the containment are given in paragraph 3.3.2.1(c).

3.5.4 BARRIER DESIGN PROCEDURES

The internal and external missile barriers have been designed to resist missile penetration in order to protect systems and components so that the failure of one system or component cannot cause the failure of another system or component.

Missile barriers are constructed of concrete, steel or a combination of concrete and steel in order to provide protection from the effects of missiles.

Barriers are designed based on the pertinent characteristics of the potential targets, postulated missiles, and barrier materials including the materials ability to provide protection from penetration, perforation, and spalling. The methods and procedures used to evaluate missile impact on structures and barriers and the analytical methods used to convert energies into forces and depths of penetration necessary for barrier design are described in NAVDOCKS P-51[1] and Bechtel Topical Report BC-TOP-9A[2].

The analysis for the depth of missile penetration in reinforced concrete was carried out using the following modified Petry formula as presented in NAV DOCKS P-51.

$$D = KA_pV' \tag{3.5-1}$$

$$V' = \log_{10}[1 + V^2/215000] \tag{3.5-2}$$

$$D' = D \quad [1 + e^{-4} (a'-2)] \tag{3.5-3}$$

$$a' = T/D$$

where

D = depth of penetration of an infinitely thick slab (inches)

K = an experimentally obtained materials coefficient for penetration (k = 0.0022 for 5000 psi reinforced concrete)

A_p = sectional pressure, obtained by dividing the weight of the missile by the maximum cross sectional area (expressed as pounds per square foot)

V' = velocity factor

V = terminal or striking velocity in feet per second

D' = actual depth of penetration in a slab of finite thickness (inches)

T = thickness of resisting slab (inches)

The design basis for concrete barrier thickness within the reactor containment is planned to provide a barrier approximately three times thicker than the depth of missile penetration. As a result, 2 ft of concrete was chosen to satisfy the above criterion. Substituting the value of 2 ft for T in equation 3.5-3, the actual depth of penetration, D', was calculated as shown in tables 3.5-2 through 3.5-5.

For the external missiles, a minimum of 2 ft of concrete has also been used in the plant design, providing protection against penetration. A summary of Category I structures utilizing concrete designed against missile penetration and the thickness provided is given below:

	Thickness (in.)
Auxiliary building, exterior walls and roof slabs (see note 1)	24
Containment dome	39

Containment wall	45
Diesel generator building	24
River intake structure	24
Intake structure at storage pond	24
RWST & RMWST shield walls	24

Note 1: The walls of the main steam room venting structure are heavy welded steel grating which provides protection against penetration of tornado-generated missiles.

Equipment and piping located outside the containment which are required for safe shutdown, long-term core cooling, or to prevent a radioactive release resulting in offsite exposures comparable to 10 CFR 100 guidelines are provided with tornado missile protection either by location within Category I structures, burial under- ground, designed missile barriers/shielding, or have been analyzed as discussed as in paragraph 3.5.1.2.2.

3.5.5 MISSILE BARRIER FEATURES

Figure 3.8-2, drawing D-176151, figures 3.8-9, 3.8-10, 3.8-11, 3.8-13, 3.8-14, drawings D-205205, D-205206, D-205207, and figures 3.8-23, 3.8-24, 3.8-25, 3.8-26, 3.8-27, 3.8-28, and 3.8-29 show the layout and principal design features of the barriers and structures designed to resist missiles.

REFERENCES

1. NAVDOCKS P-51, “Design of Protective Structures,” Bureau of Yards and Docks, Dept. of the Navy, August 1950.

2. BC-TOP-9A, “Design of Structures for Missile Protection,” Revision 2, September 1974.

3. TORMIS Missile Risk Analysis for Farley Nuclear Plant Units 1 and 2, ARA Report 4733, March 1999.

4. EPRI NP-2005, Tornado Missile Simulation and Methodology, Volumes I and II, Final Report, August 1981.

5. REA 97-1409 response, SCS to SNC letter FP 99-0429, “Tornado Missile Broadness Review and PRA Analysis,” August 5, 1999.

6. REA FS040491501-01 response, SNC letter PS-04-2337, “Re-evaluate TORMIS Analysis of EDG Silencers,” December 3, 2004.

TABLE 3.5-1

MISSILE BARRIERS INSIDE CONTAINMENT

	Missiles	Barriers
1.	Control rod drive mechanism	Integral control rod drive missile shield. See figure 3.8-13.
2.	Deleted	
3.	Drive shaft	See figure 3.8-13.
4.	Drive shaft latched to drive mechanism	See figure 3.8-13.
5.	Valve bonnets in the area where pressurizer extends above the operating deck (el 155 ft) (Pressurizer safety valves, motor-operated isolation valves, air-operated relief valves, and air-operated spray valves.)	2-ft-thick concrete shield wall. See figures 3.8-13 and 3.8-14.
6.	Instrumentation assembly	Reactor primary shield walls. See figures 3.8-13 and 3.8-14.
7.	Pressurizer heater	2-ft concrete shield walls and pressurizer heater missile shield. See figures 3.8-13 and 3.8-14.

TABLE 3.5-2

CRDM - MISSILE CHARACTERISTICS

Missile Description	Weight (lb)	O.D. (in.)	Travel Outside Housing (ft)	Weight to Impact Area Ratio A (psf)	Velocity (ft/s)	Kinetic Energy (ft-lb)	Energy Ratio[1]
Drive shaft	120	1.75	3	7,200	130	32,000	0.38
Drive shaft latched to drive mechanism	1,500	3.75	3	19,565	(2)	(2)	(2)

1. The missile barrier is a 1 ½-in.-thick steel plate. The penetration evaluation is performed using the Stanford Research Institute formulae (equation 6.203 in U.S. Reactor Containment Technology, Vol. 1). The energy ratio is the ratio of missile energy to the energy required for the missile to completely penetrate the barrier.

2. The critical missile is the drive shaft alone. It is the limiting case and envelops the drive shaft latched to drive mechanism case.

TABLE 3.5-3

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, Ab (psf)	Velocity ft/s	Depth of Penetration in a 2-ft thick Barrier (in.)
Safety relief valve bonnet (3 x 6 in. or 6 x 6 in.)	350	2.86	80	24	2,100	110	1.31
3-in. motor-operated isolation valve bonnet (plus motor and stem)	400	5.5	113	28	2,057	135	1.92
2-in. air-operated relief valve bonnet (plus stem)	75	1.8	20	20	540	115	0.37
3-in. air-operated spray valve bonnet (plus stem)	120	5.5	50	50	345	190	0.61
4-in. air-operated spray valve	200	9.3	50	50	576	190	1.06

TABLE 3.5-4

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}
$A_p = \frac{\text{Missile weight}}{\text{Impact Area}}$	504 psf	697 psf
Velocity	20 ft/s	120 ft/s
Depth of penetration in a 2-ft thick barrier	0.012 in.	0.518 in.

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}
$A_p = \frac{\text{Missile weight}}{\text{Impact Area}}$	183 psf	279 psf
Velocity	75 ft/s	120 ft/s
Depth of penetration in a 2-ft thickness barrier	0.006 in.	0.205 in.

TABLE 3.5-5

CHARACTERISTICS OF OTHER MISSILES POSTULATED WITHIN CONTAINMENT

Characteristics	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}
$A_p = \frac{\text{Missile weight}}{\text{Impact Area}}$	72 psf	587 psf	900 psf
Velocity	260 ft/s	100 ft/s	55 ft/s
Depth of penetration in a 2-ft thick barrier	0.23 in.	0.30 in.	0.14 in.

TABLE 3.5-6

MISSILE BARRIERS AWAY FROM CONTAINMENT

Missiles	Barriers
Rod drive power supply motor-generator set flywheel	Steel protective shield over flywheel

TABLE 3.5-7

ROD DRIVE POWER SUPPLY MOTOR-GENERATOR SET FLYWHEEL MISSILE CHARACTERISTICS

	Missile 1	Missile 2	Missile 3	Missile 4	Missile 5	Missile 6	Missile 7
Flywheel fragment angle (degrees)	90	120	133	134[a]	135	150	180
Flywheel fragment weight (lb)	329	438	486	489	493	548	657
Rotational speed at failure (rpm)	2,700	2,700	2,700	2,700	2,700	2,700	2,700
Rotation speed at failure, percent of operating speed	150	150	150	150	150	150	150
Initial velocity (ft/s)	249	229	219	218	217	204	176
Initial translational energy (ft-lb x 10^6)	0.317	0.357	0.361	0.361	0.360	0.355	0.317

a. Governing the design of steel protective shield.

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

This section describes the design bases and protective measures which are used to ensure that the containment, vital equipment, and other vital structures are adequately protected from dynamic effects associated with the postulated rupture of piping, including the reactor coolant system.

Relative to interfaces for the piping systems described in this section, Bechtel designed and performed layouts for all auxiliary piping systems except the reactor coolant loop (RCL). The RCL is a generic layout except for the latitude of relocating, along the piping length, the various branch nozzles. This was jointly designed and approved by Westinghouse and Bechtel. Bechtel provided Westinghouse with the appropriate layout information to permit Westinghouse to analyze the RCL piping and other Class I branch piping for which Westinghouse is responsible.

Postulated breaks in the RCL, except for Accumulator and Residual Heat Removal branch line connections, have been eliminated from the structural design basis for both Unit 1 and Unit 2, as allowed by the revised GDC-4.[3] The elimination of these breaks is the result of the application of leak-before-break technology as presented in reference 4 and 5. Leak-before-break technology was evaluated in NRC miscellaneous letters dated 1/15/92 and 8/12/91 and satisfies the NRC acceptance criteria contained in NUREG 1061, Volume 3, dated November 1984, and GDC 4.

3.6.1 SYSTEMS IN WHICH DESIGN BASIS PIPING BREAKS ARE POSTULATED TO OCCUR

Design basis piping breaks and piping cracks are postulated to occur in the RCLs and in all lines outside the reactor coolant piping system that have a normal operating temperature above 200°F and a normal operating pressure above 275 psig.

PIPING SYSTEMS INSIDE CONTAINMENT

Piping systems inside containment in which piping breaks and cracks are postulated to occur are as follows:

A. RCLs (branch connections only, as discussed above).

B. ASME III Class 1 branch lines from the reactor coolant system.

C. Following ASME III Class 2 and 3 lines.

- Main steam lines (3).
- Main feedwater lines (3).
- Steam generator blowdown lines (3).
- Normal charging line (CVCS).

- Alternate charging lines (CVCS).
- Charging line to pressurizer spray lines (CVCS).
- Letdown line (CVCS).
- Reactor coolant pump seal water (CVCS).

PIPING SYSTEMS OUTSIDE CONTAINMENT

The piping systems outside containment in which piping breaks are postulated to occur are discussed and outlined in appendix 3K.

3.6.2 DESIGN BASIS METHODS AND PIPING BREAK CRITERIA

3.6.2.1 Criteria

The design basis for the postulated pipe rupture includes not only the break criteria, but also the criteria to protect other piping and vital systems from the effects of the postulated rupture.

A loss of reactor coolant accident is assumed to occur for a pipe break down to the restraint of the second normally open automatic isolation valve (Case II in figure 3.6-1) on outgoing lines,[a] and down to and including the second check valve (Case III in figure 3.6-1) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line are closed.

Accordingly, both of the automatic isolation valves are suitably protected and restraints positioned as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Case I and IV in figure 3.6-1) a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the reactor coolant system are defined as "large" for the purpose of these criteria as having an inside diameter greater than 4 in. up to the largest connecting line, generally the pressurizer surge line. Rupture of these lines results in a rapid blowdown from the reactor coolant system and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

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

Branch lines connected to the reactor coolant system are defined as "small" if they have an inside diameter equal to or less than 4 in. This size is such that emergency core cooling system analyses have shown acceptable peak clad temperature results for a break area of up to 12.5 in.^{2} corresponding to 4-in. inside diameter piping.

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a loss of reactor coolant or steam or feedwater line break incident, to ensure that the public is protected in accordance with 10 CFR 100 guidelines. These safety systems have been designed to provide protection for a reactor coolant system pipe rupture of a size up to and including a double ended severance of the reactor coolant system main loop.

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

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

B. The containment leaktightness is not decreased below the design value.

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

    1. A pipe break which is not a loss of reactor coolant will not cause a loss of reactor coolant or steam or feedwater line break.

    2. A reactor coolant system pipe break will not cause a steam feedwater system pipe break and vice versa.

In the unlikely event that one of the small pressurized lines should fail and result in a loss-of-coolant accident, the piping must be restrained or arranged to meet the following criteria in addition to A through C above:

A. Break propagation must be limited to the affected leg; i.e., propagation to the other leg of the affected loop and to other loops will be prevented.

B. Propagation of the break in the affected leg is permitted but must be limited to a total break area of 12.5 in.^{2} (4-in. inside diameter). The exception to this case is when the initiating small break is the high head safety injection line. Further propagation is not permitted for this case.

C. Damage to the high head safety injection lines connected to the other leg of the affected loop or to the other loops must be prevented.

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

3.6.2.2 Reactor Coolant Loops

Pipe break locations are postulated in the reactor coolant loop using the methods and criteria in WCAP 8082[2]. The applicability of WCAP 8082 to the Farley Plant has been verified by reactor coolant loop analysis. The results of the analysis indicate the following:

A. All locations of the reactor coolant loop piping are below 2.4 S_m in stress intensities and have fatigue usage factors that are less than 0.2 (the fatigue usage factors are, in fact, less than 0.1), except for the locations identified in WCAP 8082. The report thus applies to this plant. Consequently, no break locations other than those identified in WCAP 8082 need to be postulated.

B. The component displacements at support interfaces that are listed in WCAP 8082 are typical displacements and were included in the report to indicate the relative magnitude of displacements at the interface. The displacements at the interfaces for the Farley plant are of the same relative magnitude as the displacements provided in WCAP 8082.

   With respect to the component displacements at the design basis break locations, WCAP 8082 assumes, for purposes of analysis, double-ended area breaks for all points where circumferential breaks are postulated except at the reactor vessel nozzles. At the reactor vessel nozzles, circumferential breaks with limited break area are postulated since the concrete shield wall prevents the development of double-ended area breaks. The displacements at these two design basis break locations are 100 square inches. However, displacements at all other points are not required for review since the break areas employed at these points are absolute maximum. (See paragraph 6.2.1.3.10.B for further justification.)

Additional information for the reactor coolant loop analysis relative to methods and analytical procedures, jet impingement forcing functions, discharge flow areas, and break opening areas/displacements is contained in attachment F to appendix 3K and paragraph 6.2.1.3.10.

According to WCAP-8082 (Reference 2), eleven break locations were postulated in the RCS primary loop piping including breaks at the nozzle welds of three large branch lines (Accumulator, RHR and Surge lines). Nine of these break locations including the one at the surge line branch connection have subsequently been eliminated from the structural design basis through the application of leak-before-break (LBB) technology. Two postulated break locations at the nozzle welds of two branch lines (Accumulator and Residual Heat Removal) still exist. The detailed fracture mechanics techniques used in this evaluation are discussed in references 4 and 5.

3.6.2.3 Class 1 Branch Lines

Pipe break locations postulated for ASME III Class 1 Branch Lines meet the intent of Regulatory Guide 1.46. The specific criteria are as follows:

1. ASME Section III Code Class 1 piping[a] breaks should be postulated to occur at the following locations in each piping run[b] or branch run:

    a. The terminal ends.

---

a. Piping is pressure retaining components consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

b. A piping run interconnects components such as pressure vessels, pumps, and valves that act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection as a branch of the main pipe run.

b. At intermediate locations between terminal ends where the primary plus secondary stress intensities (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with specified seismic events[a] and operational plant conditions[b] exceed 2.4 S_m for austenitic steel.

c. At intermediate locations between terminal ends where the cumulative usage factor U[c] derived from the piping fatigue analysis under the loadings associated with specified seismic events and operational plant conditions exceeds 0.1.

Postulated breaks in the pressurizer surge line have been eliminated from the structural design basis through the application of leak-before-break (LBB) technology. The detailed fracture mechanics techniques used in this evaluation are discussed in reference 5. Application of LBB allows the elimination of the dynamic effects of pipe rupture for these break locations.

The requirement to postulate arbitrary intermediate breaks has been eliminated from the structural design basis (including resultant dynamic and environmental effects) as allowed by NRC Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements."

3.6.2.4 Class 2 and 3 Lines

Methods and criteria for ASME III Class 2 and 3 piping lines for piping systems inside and outside containment are outlined in appendix 3K.

Specific location criteria for break points are as follows:

ASME Section III Code Class 2 and 3 piping breaks will be postulated to occur at the following locations in each piping run or branch run:

---

a. Specified seismic events are earthquakes that produce at least 50 percent of the vibratory motion of the safe shutdown earthquake.

b. Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences), and testing conditions.

c. U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code.

a. The terminal ends.

b. At intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with specified seismic events and operational plant conditions exceed $0.8(S_h + S_A)$.[(a)]

The requirement to postulate arbitrary intermediate breaks has been eliminated from the structural design basis (including resultant dynamic and environmental effects) as allowed by NRC Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements."

3.6.2.5 Break Types

The following types of breaks will be postulated at the locations identified in subsections 3.6.2.3 and 3.6.2.4.

a. Longitudinal breaks will be considered only in piping runs and branch runs 4 in. nominal pipe size and larger. Circumferential breaks will be considered only in piping runs and branch runs exceeding 1 in. nominal pipe size.

b. The local stress field at the break location in the pipe will determine whether a circumferential or longitudinal break or both will be postulated.

c. Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference unless a preferential direction can be justified by analysis. The break area is equal to the sum of the effective cross sectional flow area upstream of the break location and downstream of the break location or is equal to a break area determined by test data which defines the break geometry. Dynamic forces resulting from such breaks will be normal to the pipe axis.

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a. S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

d. Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross sectional area of the ruptured pipe. Reduced cross sectional opening areas can be used if the pipe motion is physically restricted by pipe restraints or other restraining structures. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause whipping in any direction normal to the pipe axis.

3.6.3 DESIGN LOADING COMBINATIONS

In determining pipe break locations, a worst case combination of the following load conditions is considered--thermal expansion, deadweight, seismic, seismic anchor movement, internal pressure, and load due to steam hammer, relief thrust, etc., where applicable.

3.6.3.1 Reactor Coolant Piping

As described in section 5.2, the forces associated with rupture of reactor piping systems are considered in the design of supports and restraints in order to ensure continued integrity of vital components and engineered safety features.

3.6.3.2 Class 1 Branch Lines

For Class 1 piping, the design loading combinations and design stress limits are given in section 5.2.

3.6.3.3 Class 2 and 3 Lines

Stress analysis results used in determining pipe break locations for Class 2 and 3 lines outside containment are given in the applicable piping stress calculations.

Stress analysis results used in determining pipe break locations for Class 2 and 3 lines inside containment are given in the applicable piping stress calculations for the following systems: main steam, main feedwater, CVCS normal and alternate charging lines, CVCS letdown lines and steam generator blowdown lines. Thrust loads for Class 2 and 3 lines inside containment are given in Table 3.6-2.

The piping in the charging line to the pressurizer spray and the reactor coolant pump seal water lines are field run. The stress analyses of these lines are found in the applicable piping stress calculation for the subject piping.

3.6.4 DYNAMIC ANALYSIS

The dynamic analyses and orientations applicable to the main reactor coolant loop piping system are presented in WCAP 8082.

The dynamic analyses, postulated pipe break location, and orientations for lines outside the reactor coolant pressure boundary and for Class 2 and 3 lines inside containment are discussed in appendix 3K, High Energy Line Pipe Break (Outside Containment).

Class 1 system branch lines and Class 2 and 3 lines inside containment are analyzed using the methods outlined in appendix 3K. This analysis takes into account the movement of supports, forcing functionings, dynamics, and design criteria.

3.6.4.1 Postulated Break Locations

Reactor Coolant Loops

The breaks postulated in the reactor coolant loop for the Farley Nuclear Plant are identical to those postulated in WCAP 8082[2], except as explained in paragraph 3.6.2.2.

Class 1 Branch Lines

The analysis methods used by Westinghouse for Class 1 branch lines are described below:

1. Deadweight

   The deadweight loading is defined as consisting of the dry weight of the piping and the weight of the water contained in piping during normal operation.

2. Thermal Expansion

   The thermal movements of the terminal ends are considered in addition to the thermal expansion of the branch piping.

   The cold and hot moduli of elasticity, the coefficient of thermal expansion at the metal temperature, external movements transmitted to the piping as described above, and the temperature rise above the ambient temperature define the required input data to perform the flexibility analysis for thermal expansion.

3. Earthquake Loads

   The intensity and character of the earthquake motion that produces forced vibration of equipment mounted within the containment building are specified in terms of the floor response spectrum curves at various elevations within the containment building. The 1/2 SSE and SSE floor response spectrum curves for earthquake motions are given in reference 9, section 5.2. Code Case N-411 Damped Response Spectra may be used for piping as referenced in Section 3.7.

4. Pressure

   The design and steady state operating pressures are evaluated in accordance with the requirements of the ASME Section III code.

5. Transients

   In addition to the deadweight, thermal expansion, and seismic loads, the ASME code requires that the through-wall temperature distribution be evaluated for Class 1 piping. A heat transfer analysis is performed using the anticipated transients to determine this temperature distribution.

6. Analytical Methods

   The static and dynamic structural analyses assume linear elastic behavior and employ the displacement (stiffness) matrix method and the normal mode theory for lumped-parameter, multi-mass structural representation to formulate the solution. The complexity of the physical system to be analyzed requires the use of a computer for solution based on an idealized mathematical model.

7. Effect of Design Basis Accident (DBA)

   The motions induced at the reactor coolant loop branch piping nozzle interface as a result of a rupture in the primary coolant are applied as terminal displacements at the nozzle connections.

8. Static Load Solutions

   The static solutions for deadweight, thermal expansion conditions are obtained by using the WESTBYN computer program. The computer input consists of the piping model, stiffness matrices representing various supports for static behavior, and the appropriate load condition. Coordinate transformations for rotation from the local or support coordinate system to the global system are applied to the stiffness matrices prior to their input.

9. Normal Mode Response Spectral Seismic Load Solution

   The stiffness matrices representing various supports for dynamic behavior are incorporated into the model after transformations for rotation from local to the RCL global system. The response spectra for the 1/2 SSE or SSE load case are applied along the X, or Z, and Y axes simultaneously. From the input data, the overall stiffness matrix of the three-dimensional system is generated. The stiffness matrix is manipulated to obtain a reduced stiffness matrix, associated with the mass points only. The reduced matrix is inverted to give the flexibility matrix of the system. A product matrix (also known as the dynamic matrix) formed by the multiplication of the flexibility and mass matrices is used to solve for the natural frequencies and normal modes by the modified Jacobi method. The modal participation factor matrix is computed and combined with the appropriate seismic response spectra values to give the amplitude of the modal coordinate for each mode. Then the forces, moments, deflections, rotations, support structure reactions, and stresses are calculated for each significant mode. The total seismic response is computed by combining the contributions of the significant modes by the square-root-of-the-sum-of-the-squares method.

The method of analysis is presented in appendix 3L.

Postulated break locations for Class 1 branch lines are developed using the criteria outlined in subsection 3.6.2.3. The stress analysis methodology for Class 1 branch lines is presented in appendix 3L. Stress analysis results are found in the applicable stress calculations for the following systems: reactor coolant system drain, pressurizer auxiliary spray, and reactor coolant pump seal water injection. These stress results are utilized by the criteria of subsection 3.6.2.3 in determining pipe break locations.

Class 2 and 3 Lines

Postulated break locations for the following Class 2 and 3 piping systems are developed using the criteria outlined in subsection 3.6.2.4: main steam, main feedwater, CVCS normal and alternate charging, and CVCS letdown line.

Stress analysis results of the subject systems are utilized by the criteria of subsection 3.6.2.4 in determining pipe break locations.

3.6.5 PROTECTIVE MEASURES

The fluid discharged from the ruptured piping will produce reaction and thrust forces in the RCL system. These effects are considered in ensuring the continued integrity of the vital components and the engineered safety features.

To accomplish this in the design, a combination of component restraints, barriers, and layout are utilized to ensure that for a loss-of-coolant or steam feedwater line break, propagation of damage from the original event is limited, and the components as needed are protected and available.

Protective measures for high energy lines outside the reactor coolant pressure boundary are discussed in appendix 3K, High Energy Line Pipe Break (Outside Containment).

3.6.5.1 Pipe Whip Restraints

Reactor Coolant Loop

Large branch lines attached to the reactor coolant loop piping are restrained to meet the following criteria:

a. Propagation of the break to the unaffected loops must be prevented to ensure the delivery capacity of the accumulators and low head pumps.

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

c. Where restraints on the lines are necessary in order to prevent impact on and subsequent damage to the neighboring equipment or piping, restraint type and spacing will be chosen so that a plastic hinge on the pipe at the two support points closest to the break is not formed.

Additional discussion of pipe restraint design criteria is found in reference 2.

Class 1 Branch Lines

Pipe whip restraints for Class 1 branch lines are shown on applicable civil design drawings.

Class 2 and 3 Lines

Pipe whip restraint design criteria along with the description of a typical pipe whip restraint for Class 2 and 3 lines are given in appendix 3K.

Pipe whip restraint locations for Class 2 and 3 lines are shown on applicable civil design drawings.

3.6.5.2 Jet Impingement

In addition to pipe restraints, barriers and layout are used to provide protection from blowdown jet and reactive forces on cabling, instrumentation, and equipment necessary for safe shutdown of the reactor.

In addition, the refueling cavity walls, various structural beams, and the operating floor enclose each reactor coolant loop into a separate compartment, thereby preventing an accident, which may occur in any loop, from affecting another loop or the containment liner. The portion of the steam and feedwater lines within the containment have been routed behind barriers which separate these lines from all reactor coolant piping.

In reviewing the mechanical aspects of these lines, it has been demonstrated by Westinghouse Nuclear Energy System tests that lines hitting equal or larger size lines of same schedule will not cause failure of the equal or larger line; e.g., a 1-inch line, should it fail, will not cause subsequent failure of a 1 in. or larger size line. The reverse, however, is assumed to be probable; i.e., a 4-in. line, should it fail and whip as a result of the fluid discharged through the line, could break smaller size lines such as neighboring 3 in. or 2 in. lines.

Bending of a broken stainless steel pipe section such as that used in the reactor coolant system branch lines does not cause this section to become a missile. This design basis has been demonstrated by performing bending tests on large and small diameter, heavy and thin walled stainless steel pipes.

<u>RC Loop Jet Impingement</u>

Jet impingement loads on the primary equipment and supports due to the breaks postulated in RCL are based upon the dynamic piping displacement response determined from the loop LOCA analyses. The jet impingement loads on the adjacent structures are evaluated using the methods outlined in appendix 3K.

<u>Class 1, 2 and 3 lines Jet Impingement</u>

Jet impingement methods and analyses for Class 1, 2, and 3 lines for both inside and outside containment are outlined in appendix 3K.

3.6.5.3 Separation and Redundancy

The separation and redundancy of equipment and safety features that have been designed for protection against the effects of pipe break are outlined in appendix 3K.

3.6.6 STRUCTURAL ANALYSIS

3.6.6.1 Outside Containment

The structural analysis for high energy pipe breaks outside containment is discussed in attachment G, appendix 3K.

3.6.6.2 Inside Containment

The containment internal structures, which have the most critical loading conditions during a pipe break event, consist of:

1. The primary shield wall.
2. The secondary shield wall, which encloses the steam generator and pressurizer compartments.
3. The floor slab at elevation 129 ft 0 in.

The geometry of these structures is described in subsection 3.8.3.1.

The structural loads and loading combinations for each postulated break are in accordance with Sections B and C of "Structural Design Criteria for Evaluating the Effects of High Energy Pipe Breaks on Category I Structures Outside the Containment," Document (B) of the NRC.

The finite element method was employed for the analysis and Bechtel's SAP1.8 computer program was utilized for performing the finite element analysis. A description of the computer program is in attachment G, appendix 3K.

3.6.6.2.1 Finite Element Model

Three finite element models were developed for the analysis of the structures under investigation. Due to the symmetry or similarity of the geometry and the loads, only parts of each of these structures were modeled. The results and conclusions obtained from the analysis of these models will also apply to the other parts of the structures not included in the models.

Three-dimensional brick elements were used for the primary shield wall. Plate elements were used for the secondary shield walls (steam generator compartment walls), slab, and the pressurizer compartment walls. The boundary conditions include partial or complete fixity against displacement and rotation, depending upon the structure boundary restraint conditions under thermal and other loadings.

3.6.6.2.2 Results of Analysis

Table 3.6-7, summarizing the results of this linear, elastic, finite element analysis, indicates that the walls and slab are sufficiently strong to resist the various combinations of loads associated with a high energy pipe break inside the containment, with a margin of safety provided by the load increases and load factors used in the analysis.[a]

3.6.6.3 Pipe Whip Restraint Design

The analytical approach and design of the pipe whip restraints are described in attachment B, appendix 3K.

a. Reference section 6.2.

REFERENCES

1. Szyslowski, J. J. and Salvatori, R., "Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System," WCAP-7503, Rev 1, February 1972.

2. PWR Staff, "Westinghouse Technical Position on Discrete Break Locations and Types for the LOCA Analysis of the Primary Coolant Loop," WCAP-8082, May 1973.

3. Modification of GDC-4, Final Rule, 52 FR 41288, October 27, 1987.

4. "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants," (Proprietary Class 2) WCAP-12825, January 1991.

5. "Technical Justification for Eliminating Pressurizer Surge Line Rupture from the Structural Design Basis for Farley Units 1 and 2," (Proprietary Class 2) WCAP-12835, April 1991.

TABLE 3.6-2

THRUST LOADS DUE TO A FULL AREA PIPE RUPTURE (CLASS 2 AND 3 PIPING)

System	Line Size	Temperature (F)	Pressure (PSIG)	Thrust Force (lbs)
Main steam	32 in.	547	1005	285,000
Main feedwater	14 in.	442	1055	109,400
CVCS normal and alternate charging lines	3 in.	485	2350	12,000
CVCS letdown line from Class 1 interface to regenerative heat exchanger	3 in.	550	2350	12,000
CVCS letdown line from regenerative heat exchanger to containment penetration	2 in.	380	2250	4700
(before flow orifice)	3 in.	380	2250	12,000
CVCS letdown line from regenerative heat exchanger to containment penetration	2 in.	380	550	3150
(after flow orifice)	3 in.	380	550	6900
Steam generator blowdown line	2 in.	547	1055	4730

TABLE 3.6-7

ANALYSIS RESULTS

Structure	Thickness (in.)	Critical Postulated Pipe Break	Critical Load Combination	Maximum Pressure P	Allow Pressure P_{allow}	$\frac{P_{allow}}{P}$
Primary Shield Wall	108	Hot Leg Rupture	D + L + Ta + Ra + 1.5P	124 615	150 667	1.21 1.08
Steam Generator Compartment Wall	42	Cold Leg Rupture	D + L + Ta + Ra + 1.5P	57	60	1.05
Pressurizer Compartment Wall	24	Spray Line Rupture	D + L + Ta + Ra + 1.5P	20	22	1.10
Slab El 129'-0"	36 36	Cold Leg Rupture	D + L + Ta + Ra + 1.5P	57	58	1.02

Notes:

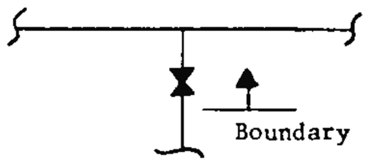
1. Loads and load combinations are in accordance with Sections B and C of "Structural Design Criteria for Evaluating the Effects of High-Energy Pipe Breaks on Category I Structures Outside the Containment," Document (B) of the NRC.

2. Two pressure values are given for the primary shield wall. The first one is the differential pressure uniformly applied inside the reactor cavity. The second one is the possible localized pressure within the inspection chamber at the reactor nozzles.

3. Maximum pressure values are the same as those used in the critical load combinations, and were obtained by multiplying the calculated peak pressure by a factor 1.4 x 1.2 = 1.68 to account for uncertainty and the dynamic factors, respectively.

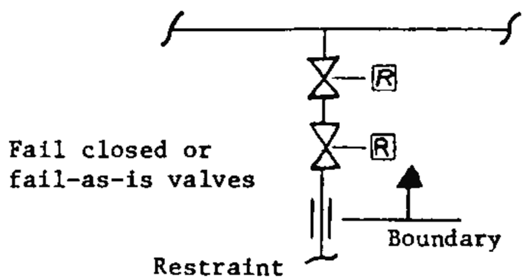
CASE I Outgoing lines with normally closed valve.

Reactor Coolant Piping

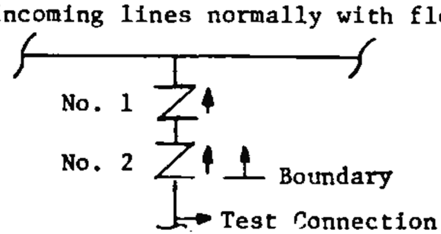


NOTE: Pressurizer safety valves are included under this case.

CASE II Outgoing lines with normally open valves.

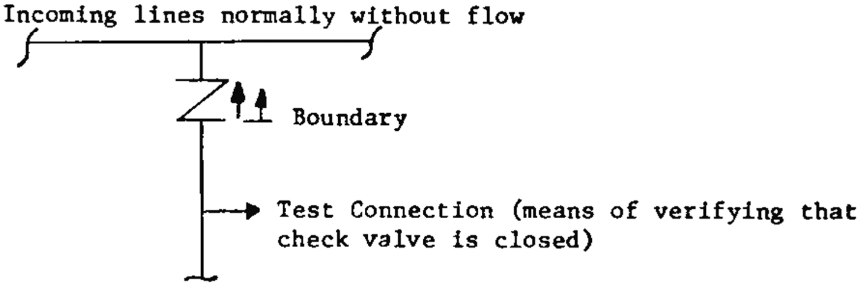


CASE III Incoming lines normally with flow



NOTE: The Reactor Coolant Pump No. 1 seal is assumed to be equivalent to first valve.

CASE IV Incoming lines normally without flow



CASE V All instrumentation tubing and instruments connected directly to the Reactor Coolant System are considered as a boundary. However, a break within this boundary results in a relatively small flow which can normally be made up with the charging system.

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SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS OF REACTOR COOLANT ACCIDENT BOUNDARY LIMITS
		FIGURE 3.6-1

3.7 SEISMIC DESIGN

The criteria for determining the adequacy of Seismic Category I mechanical and electrical equipment for the Farley Nuclear Plant are described in various areas of the FSAR. In some cases, the criteria are specified in general terms to require verification by tests or analyses. In other cases, more specific criteria are specified such as verification in accordance with IEEE Standard 344-1971.

Historically, it should be noted that the FNP Unit 2 seismic qualification program, i.e., IEEE 344-71 type qualification, was previously audited by the NRC's Seismic Qualification Review Team (SQRT). It was concluded in NUREG-0117 Supplement No. 5 (dated March, 1981) Safety Evaluation Report related to the operation of Unit 2 that "the licensee's seismic qualification program provides reasonable assurance that the seismic category I mechanical and electrical equipment is adequately qualified, meets the applicable requirements of General Design Criterion 2, and is, therefore, acceptable for full-power operation."

By letter dated February 19, 1987, the NRC issued Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46." On May 22, 1992, the NRC issued GL 87-02 Supplement 1. As documented in NUREG-1211, "Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants," GL 87-02 is applicable to Farley Nuclear Plant (FNP) Unit 1, since Unit 1 had not previously been audited by the SQRT. Southern Nuclear Operating Company (SNC) replied to GL 87-02 by letter dated September 10, 1992. The SNC letter included a commitment to use the Seismic Qualification Utility Group (SQUG) methodology as documented in the Generic Implementation Procedure (GIP) for resolution of seismic issues identified in GL 87-02 for FNP Unit 1. The SQUG methodology is based on application of earthquake experience data to verify the seismic adequacy of equipment. The seismic evaluation for FNP Unit 1 was completed, and the results were documented in a document entitled "Unresolved Safety Issue A-46 Summary Report." This document was submitted to the NRC by letter dated May 18, 1995 as a 10 CFR 50.54(f) response. SNC received an SER dated July 9, 1998, concerning FNP Unit 1 USI A-46 resolution and it stated that SNC's USI A-46 program implementation resulted in safety enhancements beyond the original licensing basis and SNC actions provide sufficient basis to close the USI A-46 review at the facility.

3.7.1 SEISMIC INPUT

Geologic and seismologic surveys of the site have been made to establish two "design earthquakes" with different intensities of ground motion. These are the 50 percent safe shutdown earthquakes (1/2 SSE) and safe shutdown earthquakes (SSE) with different intensities of ground motion. The 1/2 SSE, previously called operating basis earthquake (OBE) in the Preliminary Safety Analysis Report, is postulated to be the earthquake that could be expected to occur at the site during the operating life of the plant. The SSE represents the strongest earthquake that is hypothetically postulated to occur during an infinite period.

The plant site geologic and seismologic investigations and recommendations are discussed in section 2.5. As specified in the following paragraphs, the intensity postulated to occur at the site for both the 1/2 SSE and SSE is defined from the history of seismic activity in the area around the site.

3.7.1.1 Design Response Spectra

The safe shutdown earthquake and 50 percent safe shutdown earthquake are specified in terms of a set of idealized, smooth curves, called the design spectra because they specify a range of values for two of the important properties of an earthquake ground motion, i.e., the maximum ground acceleration and the frequency distribution.

The SSE is that earthquake which produces the vibratory ground motion for which Category I structures, systems and components are designed to remain functional. Category I structures, systems, and components will also be designed to withstand the effects of vibratory motion of at least 50 percent of the safe shutdown earthquakes in combination with other appropriate loads.

Figure 3.7-1 shows the 1/2 SSE spectra for 0, 0.5, 1.0, 2.0, 3.0, and 5.0 percent of critical damping, with a horizontal ground peak acceleration of 0.05 g and vertical ground acceleration of 0.033 g.

Figure 3.7-2 shows the SSE spectra for 0, 0.5, 1.0, 2.0, 3.0 and 5.0 percent of critical damping, with a horizontal ground peak acceleration of 0.10 g and vertical ground acceleration of 0.067 g.

The bases for the selection of 1/2 SSE and SSE ground accelerations are presented in section 2.5.

These design spectra are obtained by modifying Newmark's curves.[(1)] To obtain these curves, variations in site conditions, foundation properties, and amplification factors of previous distant and nearby earthquakes (where the location of the origin is known), were taken into account. One of the parameters defining the design spectra is the spectrum amplification ratio, which is the ratio of the peak spectrum acceleration to the ground acceleration for a particular magnitude of damping. For this site, a ratio of 3.5 is used for the period range of 0.15 to 0.50 second of the 2 percent critical damping design spectrum.

Section 2.5.1(6) of BC-TOP-4 (Rev. 1) discusses the derivation of the shape of the design spectra. These design spectra are based on the existing strong motion earthquake ground records of various durations, and are recorded at sites' having different geologic conditions, epicentral distances, and their associated spectral amplification factors.

A discussion of the effects of historical seismic events on the site is given in section 2.5. Because the modified design spectra are based on the properties of several strong motion records of the earthquakes recorded at sites of various geologic conditions and epicentral distances, the effects of duration, distance, and depth are automatically taken into account.

3.7.1.2 Design Response Spectra Derivation

The synthesized time history accelerogram, normalized to 0.10 g, is shown in figure 3.7-3 (SSE synthetic time history). The same synthesized time history accelerogram, normalized to 0.05 g, is used for 1/2 SSE analysis. In the vertical direction, the same accelerogram is normalized to 0.067 g for SSE and 0.033 g for 1/2 SSE.

The synthesized time history is obtained through modification of a time history selected from simulated motion, for a total duration of 24 seconds with a uniform time increment of 0.01 second.

The spectral values of the synthesized time history for the 1/2 SSE are equal to or greater than those on the 1/2 SSE ground response spectrum for 2 percent critical damping, as shown in figure 3.7-4. The spectral values of the synthesized time history for the SSE are equal to or greater than those on the SSE ground response spectrum for 5 percent critical damping as shown in figure 3.7-5. The damping values that are used for the generation of instructure response spectra are 2 and 5 percent for 1/2 SSE and SSE respectively for the prestressed concrete structures and reinforced concrete structures as presented in table 3.7-1. Because of this fact, 2 and 5 percent critical damping response spectra envelop the corresponding 1/2 SSE and SSE response spectra for the range of 115 frequencies tabulated in table 3.7-2. These 115 frequencies are sufficient to describe a response spectrum accurately for engineering purposes.

3.7.1.3 Critical Damping Values

The specific percentage of critical damping values used for Category I structures, systems, components, and soil are provided in table 3.7-1.

In lieu of damping values given in Table 3.7-1, ASME Code Case N-411 damping values may be used in piping analysis. Use of N-411 damping values will adhere to the conditions and limitations contained in the Code Case and Regulatory Guide 1.84.

Energy dissipation in structures is generally represented by equivalent viscous dampers. Evaluation of the damping coefficients is based on material, stress level, and the type of connections used in the structural system. The damping values used in the response spectrum design approach are those in table 3.7-1 which are based on a paper by N. M. Newmark and W. J. Hall, "Seismic Design Criteria for Nuclear Reactor Facilities," and another paper by N. M. Newmark, "Design Criteria for Nuclear Reactors Subjected to Earthquake Hazards." These values are used in conjunction with the modal representation of the structure and are expressed as a percentage of critical damping.

The allowable stress levels for 1/2 SSE and working load combinations have been established as normal code allowables. The allowable stress levels for SSE and yield load combinations have been established at 85 percent of the compressive strength for concrete and 90 percent of yield for steel which will maintain the materials of construction within the elastic range.

3.7.1.4 Bases for Site Dependent Analysis

Site dependent analysis is not used to develop the shape of the design response spectra.

3.7.1.5 Soil Supported Category I Structures

Outdoor tanks are the only major Category I structures founded on soil. The depth of soil over bedrock (Lisbon formation) is about 55 ft.

3.7.1.6 Soil Structure Interaction

Soil structure interaction is taken into account in the dynamic analysis of the containment and other Category I structures. For the lumped mass model, the soil stiffness of the foundation is represented by introducing equivalent springs for the foundation medium, whereas the base mat is assumed to be relatively rigid. Horizontal, vertical, and rocking spring constants are obtained from the theory of a rigid base resting on an elastic half-space.

A lumped mass model of a structure and the foundation is shown in figure 3.7-6. The constants k_x and k_ψ are the equivalent spring stiffnesses for horizontal translation and rocking, respectively, and k_z is the spring stiffness for vertical translation. The formulas for computing the equivalent spring stiffness for the cases of circular base mat and rectangular base mat are as follows:

a. Circular Base

Motion	Spring Constant
Horizontal	$k_X = \frac{32(1-\upsilon)GR}{7-8\upsilon}$
Rocking	$k_\psi = \frac{8GR^3}{3(1-\upsilon)}$
Vertical	$k_z = \frac{4GR}{1-\upsilon}$

in which

υ = Poisson's ratio of foundation medium

G = shear modulus of foundation medium

R = radius of the circular base mat

b. Rectangular Base

Motion	Spring Constant
Horizontal	$k_x = 2(1 + \upsilon) G\beta_x \sqrt{BL}$
Rocking	$k_\psi = \frac{G}{1 - \upsilon} \beta_\psi B^2 L$
Vertical	$k_z = \frac{G}{1 - \upsilon} \beta_z \sqrt{BL}$

in which υ and G are as defined previously, and

B = width of the base mat perpendicular to the direction of horizontal excitation

L = length of the base mat in the direction of horizontal excitation

β_x, β_ψ, β_z = constants that are functions of the dimensional ratio, L/B. See figure 3.7-7.

The shear modulus is obtained from the shear wave velocity and mass density of the soil using the following relationship:

$$G = \frac{\rho(V_s)^2}{144g}$$

G = shear modulus, lb/in.^{2}

ρ = density, lb/ft^3

Vs = shear wave velocity, ft/s

g = 32.174, ft/s^2

These springs are entered at the base of the model.

3.7.2 SEISMIC SYSTEM ANALYSIS

This subsection describes the seismic analysis performed for Category I structures. Category I structures were designed using a dynamic analysis.

3.7.2.1 Seismic Analysis Methods

The seismic analysis methods applied to all Category I structures, systems, and components are identified in tables 3.7-3 and 3.7-4.

Analysis of Category I structures, systems, and components is accomplished, where applicable, using the response spectra or time history approach, which utilizes the natural period, mode shapes, and appropriate damping factors of the particular system. Where analytical methods of analysis do not produce results of a significant confidence level or where analysis appears undesirable, dynamic testing of equipment is used to ensure functional integrity.

An important step in the seismic analysis of Category I systems or structures is the procedure used for modeling. The system is represented by lumped masses and a set of springs idealizing both the inertia and stiffness properties of the system.

A complete dynamic analysis including soil structure interaction is performed on the containment and all Category I structures to determine their behavior during an earthquake. The modal response spectrum technique is used in the seismic design of all Category I structures. The analysis is accomplished in the following 5 steps:

1. Reduce the structure into a mathematical model in terms of lumped masses and stiffness coefficients.

2. Obtain the natural frequencies and mode shapes of the model.

3. Evaluate and determine the proper damping values.

4. Determine the resulting internal forces on the structure, using the appropriate earthquake response spectra.

5. Determine the spectrum response curves to be used in the analysis of the equipment located at all levels. The spectrum response curves are generated at the modal mass points.

In building the mathematical model, the locations for lumped masses are chosen at floor levels and points considered of critical interest. Between mass points the structural properties are reduced to uniform segments of cross-sectional area, effective shear area and moments of inertia.

The analysis utilizes the values from the ground response spectra for this site. Acceleration values are selected for each mode, based on damping and natural frequency. The inertia forces, shears, moments, accelerations, and displacements of a sufficient number of the individual modes are combined by taking the square root of the summation of the squares (SRSS) of the individual modal values. Procedures for combining modal responses are presented in subsection 3.7.3.4.

A separate analysis is made on the model for the horizontal and vertical earthquake accelerations, the vertical being two-thirds of the horizontal spectral values. The results from both analyses are combined to obtain the critical response values. For structures, the model is analyzed separately for both horizontal directions, and the results of each are combined separately with those from the vertical analysis.

The mathematical model of the structure and results of seismic dynamic analysis for the Category I structures in the north to south, east to west, and vertical directions are shown in figures 3.7-8 through 3.7-21 and 3.7-23 through 3.7-56.

The following information is obtained from the preceding analyses:

a. Inertial forces.

b. Accelerations.

c. Structural displacements.

d. Horizontal shears.

e. Horizontal moments.

f. Vertical axial forces.

The mathematical model is analyzed for its frequencies and mode shapes; then the dynamic response at the mass points is obtained by application of the synthesized time history earthquake at the base of the structure. The input time history used is synthesized so that the computed spectral values are greater than or equal to the spectral values of the design spectrum for all periods. The output time history response is obtained for any mass point desired.

From the time history response of a particular mass point, a spectrum response curve is developed and enveloped into a design spectrum. These envelopes are widened by at least 10 percent by period to account for uncertainties in the structural model and input. For all rigid and flexible equipment, the maximum acceleration is obtained by the spectrum response curves developed at various elevations and other points of attachment. These curves are generated using a synthesized time history with horizontal components normalized to 0.05 g and 0.10 g ground accelerations for the 1/2 SSE and SSE, respectively. Both horizontal and vertical excitations are applied to the structure, and curves are generated for each direction.

The stability of structures from the combined horizontal and vertical earthquake excitation is considered by taking modal SRSS moments about the foundation level and comparing them against the resisting moment of the structural deadweight.

The general approach employed in the dynamic analysis of Category I equipment and component design is based on the response spectrum technique where applicable. The time history analysis of Category I structures, as previously explained, generates instructure response spectrum curves and time histories at various support elevations for use in analysis of systems and equipment.

At each level of the structure where vital items are located, horizontal response spectra for each of the two major axes of the structure and a vertical response spectrum are developed. The floor response spectrum is smoothed so that the response curve is an upper bound envelope of all the acceleration points. Whenever the response curve comes to a peak, the curve is made flat in a region ±10 percent of that peak frequency. When items are supported at two or more elevations, the response spectrum of each elevation is superimposed on each other and the resulting spectrum is the upper bound envelope of all the individual spectrum curves considered.

Simplified analytical models are used for analysis of systems and equipment; however, where one or two degree of freedom models do not provide a suitable representation of the systems or equipment under consideration, multi-mass models are used in accordance with the lumped parameter modeling techniques and normal mode theory. Piping analysis is handled in systems using lumped mass models outlined above. Special attention is given to the flexibility or rigidity characteristics of the piping networks using strategically placed restraints and snubbers to ensure predictability of structural integrity under the specified seismic conditions.

To determine the effect of an earthquake on Westinghouse equipment, a dynamic analysis based on a discrete mass mathematical model is performed. Although a mechanical component may be analyzed using a mathematical model with as much complexity as allowed by the capacity of the computer and the computer code, the analysis is meaningful only when this detailed model also represents the effective utilization of the theory on which the computer code is built. Specifically, there are at least three things that are considered when establishing the mathematical model. They are the limiting values for items such as the degrees of freedom, sections, members, anchors, joints, and bellows, etc; the maximum allowable ratio of member rigidity; and the basic theory limitations. A computer code such as WESTDYN can be used to obtain the natural frequencies, mode shapes, absolute and relative displacements, absolute accelerations, and the stresses. The equipment design is determined to be adequate from the stress margin and by displacements limited to the operating tolerance.

For certain Category I equipment and components where dynamic testing becomes a necessity to ensure functional integrity, test performance data and results reflect the following:

a. Performance data of equipment which, under the specified conditions, has been subjected to equal or greater dynamic loads than those to be experienced under the specified seismic conditions.

b. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to equal or greater dynamic loads than those specified.

c. Actual testing of equipment in accordance with one of the following methods:

    1. The equipment is subjected to a sinusoidal excitation, sweeping through the desired range of significant frequencies, using input acceleration amplitudes for the forcing function which simulates the specified seismic conditions.

    2. The equipment is subjected to a transient sinusoidal motion synthesized by pulse exciting a group of approximate octave filters so that the response of the shake table and the duration of load simulates the artificial response spectrum curve at the building floor elevation of interest.

A detailed description of dynamic analysis and testing requirements is given in sections 3.9 and 3.10. Table 3.7-4 identifies which qualification method is used.

The mathematical models used for the dynamic analysis of Category I structures, systems, and components, and the results of the analysis are shown in figures 3.7-8 through 3.7-21 and 3.7-23 through 3.7-56. Figure 3.7-6 shows a typical lumped mass model for a cantilevered system. Figures 3.7-64 and 3.7-65 show mathematical model and first mode of vibration of the reactor internals, respectively.

The allowable stresses for the 1/2 SSE and SSE loads in combination with other loads are in accordance with section 3.8.

For the 1/2 SSE, the resulting stresses and deflections are limited to those that do not interrupt normal operation of the plant; whereas for the SSE, the resulting stresses and deflections are limited to those that do not prevent a safe and orderly shutdown of the plant.

3.7.2.2 Natural Frequencies and Response Loads

A summary of natural frequencies is presented in table 3.7-5. Typical mode shapes for the containment and internal structures are shown in figures 3.7-57 and 3.7-58. Typical response loads (inertia forces) for each mode of the containment are shown in figures 3.7-59 and 3.7-60. The response spectrum at the reactor support elevation is shown in figure 3.7-61.

The natural frequencies of Westinghouse supplied components are considered in the system seismic analysis. The natural frequencies of the components themselves are above the seismic cutoff frequency.

The natural frequencies are listed in the component stress reports filed with the NRC.

3.7.2.3 Procedures Used to Lump Masses

The regular lumping techniques, which consist of lumping the continuous mass distribution at discrete joints referred to in section 3.7 as mass points are used in constructing some of the mathenatical models. The location of the lumped masses are chosen at floor levels and points considered of critical interest, such as equipment. The lumped masses are computed from tributary structure dead loads and fixed equipment loads. The model used to generate response spectra for the containment structure utilizes a consistent mass matrix. The term "consistent" describes both the inertial and deformation shapes of the structure that are consistent with stiffness formulation. In the matrix formulation for consistent masses, the floor masses and tributary fixed equipment masses are added to the diagonal mass matrix.

3.7.2.4 Rocking and Translational Response Summary

A fixed base mathematical model for the dynamic system analyses is not assumed. As described in subsection 3.7.1.6, a simplified lumped mass and soil spring approach has been used to characterize soil structure interaction. For more details, refer to BC-TOP-4 (Rev. 1), Section 3.3.

3.7.2.5 Methods Used to Couple Soil with Seismic System Structures

Methods used to couple soil with seismic system structures are discussed in subsection 3.7.1.6. A finite element analysis for the layered site has not been used to couple the soil and the Seismic System structures and components.

3.7.2.6 Development of Floor Response Spectra

A modal synthesis method is used to develop the response spectra as described in BC-TOP-4 (Rev. 1), Sections 4.2 and 5.2. The modal response spectra multi-mass method was not used to develop floor response spectra.

3.7.2.7 Differential Seismic Movement of Interconnected Components

Differential seismic movement of interconnected components has been considered. The stress and deformation criteria for structures are provided in section 3.8. The stress and deformation criteria for piping are described in BP-TOP-1.

The effect of differential seismic movement of interconnected components between floors is considered in the analysis when it is within Westinghouse scope of responsibility. The interconnected components subjected to differential movement will be within the applicable stress and deformation limits.

3.7.2.8 Effects of Variations on Floor Response Spectra

The instructure response spectra computed from the time history instructure acceleration response generally reflect two parameters, the amplification of the free field input produced by the soil and structural system, and the frequency content associated with these amplification regions.

The criteria selected for enveloping these computed curves with the smooth design spectra are:

a. The instructure design spectrum envelopes the computed spectra at all points.

b. The minimum frequency shift is either computed as above or ±10 percent, whichever is larger.

3.7.2.9 Use of Constant Vertical Load Factors

A vertical, seismic-system, multi-mass, dynamic analysis method has been used for seismic analysis of all Category I structures. The mathematical models are discussed in subsection 3.7.2.1.

Constant 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.

3.7.2.10 Methods Used to Account for Torsional Effects

The dynamic analysis of structures is covered in subsection 3.7.2.1. The method presented, in general, describes lumped mass modeling techniques used to analyze Category I structures. The effects of shear stress due to torsion are considered in the analysis.

A lumped mass mathematical model of the auxiliary building incorporating the eccentricity of the masses was generated. Calculated torsional frequencies are much higher than the translational frequencies for the auxiliary building and internal structures, as shown in table 3.7-6. Therefore, the torsional coupling has been neglected in the mathematical model of these structures. However, to assure the adequacy of the design, the effect of the torsional moment has been taken into account. The torsional moment is determined by the product of the shear force and eccentricity between the center of mass and the center of rigidity.

3.7.2.11 Comparison of Responses

Table 3.7-7 gives a comparison of results for the acceleration of the containment shell, based on the response spectrum and time history methods.

3.7.2.12 Methods for Seismic Analysis of Dams

The analytical methods and procedures that have been used for the seismic system analysis of the storage pond dam and dikes is described in Appendix 2B.

3.7.2.13 Methods to Determine Category I Structure Overturning Moment

The overturning moments of the Category I structures were calculated by the response spectrum method. The stability of the structures is checked by combining the overturning moment, dead load of the structure, and vertical acceleration. The soil reaction under the containment is obtained by considering the linear stress distribution under a rigid base mat subjected to the worst combined effects of overturning moment, dead load, and vertical acceleration.

3.7.2.14 Analysis Procedure for Dampings

In general, the models employed in the seismic analysis represent more than one material and the characteristic mode shapes have component deflections due to translation and rotation of the soil and structural deformation of steel and concrete. The mode shapes are broken down to component deflections due to the various material deformations. Damping values applied to each mode are computed as the summation of the absolute deflection multiplied by the associated damping for that material divided by the absolute summation of all component deflections.

As an example of the technique, consider a structure whose motion is primarily composed of flexural displacement and foundation rotation. The mode shape must be broken down into its rotational and flexural components, denoted as ϕ_R and ϕ_F, respectively. Since the rotation is due to the fact that the structure is supported on a flexible foundation, the foundation damping, denoted as

β_R, will influence the total damping. Denoting the structure's flexural or material damping by β_F, the composite damping is determined using the following equation:

$$C = \frac{\beta_R\phi_R + \beta_F\phi_F}{\phi_R + \phi_F}$$

The above formula may be regarded as an approximate technique to determine a composite damping value when the structural motion consists of both a rocking effect with the soil interaction and flexure of the building. If rocking is predominant, then soil damping alone is assigned to that mode. The converse holds true when flexure is predominant.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

3.7.3.1 Determination of Number of Earthquake Cycles

3.7.3.1.1 Category I Systems and Components Other Than NSSS

Procedures to determine the number of earthquake cycles for piping during one seismic event are discussed in BP-TOP-1, (Rev. 1) Section 6.0. For equipment designed on the basis of analytical results, the design criteria used assumed elastic behavior. Therefore, the number of loading cycles is of no concern. Neither is it of any concern for Category I structures, since the calculated stresses and strains are below yield.

3.7.3.1.2 NSS System

Where fatigue analyses of mechanical systems and components are required, Westinghouse specifies in the equipment specification that five occurrences of 1/2 SSE and SSE, each having ten cycles of maximum response for each occurrence, be analyzed. The fatigue analyses are performed as part of the stress report.

3.7.3.2 Basis for Selection of Forcing Frequencies

Forcing frequencies are not selected but are calculated in accordance with BC-TOP-4, Rev. 1, Section 5.3-2.

3.7.3.3 Root Mean Square Basis

The term used to describe the procedure for combination of modal responses is "square root of the sum of squares" (SRSS).

3.7.3.4 Procedure for Combining Modal Responses

The criteria for combining modal responses (shears, moments, stresses, deflections, and/or accelerations) for the response spectrum modal analysis are as follows:

a. The SRSS method of combining all modal responses is used.

b. All modes up to a frequency of 30 hz are used in the analysis.

c. When closely spaced frequencies of two or more modes occur, only those modes' responses are combined in an absolute manner; the resulting total is treated as that of a pseudo-mode and then combined with the rest of the modes in an SRSS manner.

d. The criterion used to determine whether mode frequencies are closely spaced is whether the frequencies differ from each other by less than 20% of the lower frequency. Also, multiples of lower mode frequencies are compared with higher mode frequencies, and the same 20 percent comparison is made.

e. For analyses for which Westinghouse has the responsibility, the total seismic response may be obtained by combining the individual modal seismic response and the individual modal responses, using 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 so that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10 percent of the lower frequency. The combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes, the product of the responses of the modes in each group of closely spaced modes and a coupling factor, ε. This can be represented mathematically as:

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

where

R_T	=	total response
R_i	=	absolute value of response of mode i
N	=	total number of modes considered
S	=	number of groups of closely spaced modes

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

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

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

$$\varepsilon_{K\ell} = \{1 + \left[\frac{\omega_K' - \omega_\ell'}{\beta_K'\omega_K + \beta_\ell'\omega_\ell}\right]^2\}^{-1}$$

and

$$\omega'_K = \omega_K \left[1 - (\beta'_K)^2\right]^{1/2}$$

$$\beta_K' = \beta_K + \frac{2}{\omega_k t_d}$$

ω_K = frequency of closely spaced mode K (rad/sec)

β_K = fraction of critical damping in closely spaced mode K

t_d = duration of the earthquake (sec.)

In addition to the above methods, any of the methods described in USNRC Regulatory Guide 1.92, Revision 1 may be used for modal combination in the analysis of replacement components.

f. Modal response combination for piping analysis is described in appendix 3L.

3.7.3.5 Significant Dynamic Response Modes

BC-TOP-4 (Rev. 1), Appendices F and H, describe the analysis techniques used when the peak of the spectra method is employed by the Category I equipment suppliers.

For piping, this is covered in BP-TOP-1, Rev. 1, Section 2.0 and Appendix D.

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components 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 Design Criteria and Analytical Procedures For Piping

The relative seismic movements between buildings, between floors in buildings, and between major components and buildings are applied to the pipe anchors and restraints in a rational or conservative manner. Movements between buildings and between buildings and components are always considered to be out of phase in such a way that their relative movements are maximum. The resulting stresses are classed as secondary and are combined with thermal expansion stresses. These stresses are held below the appropriate code allowable limits.

3.7.3.7 Basis for Computing Combined Response

The bases for the methods used to determine the combined horizontal and vertical amplified response loadings for the seismic design of the piping and equipment are discussed in BP-TOP-1, Revision 1, and BC-TOP-4, Revision 1, except that in all cases the maximum horizontal response in one direction is combined with the vertical response by the "SRSS" approach. The combined response is then used in the stress analyses.

3.7.3.8 Amplified Seismic Responses

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

3.7.3.9 Use of Simplified Dynamic Analysis

The simplified seismic analysis methods and procedures are used only for the design of 2 in. and under piping that is field routed. The design requires the piping system to be supported by means of hangers, restraints, and anchors in a continuous run of simple shapes, for which the natural frequencies have been established by previous analyses and found to fall within the rigid frequency range or acceptable stress limits.

A summary of typical results is given in table 3.7-8.

3.7.3.10 Modal Period Variation

The procedures used to account for modal period variation in the mathematical models for Category I structures due to variation in material properties are discussed in subsection 3.7.2.8.

The materials employed in safety-related systems under Westinghouse scope of supply are standard. The material properties that can affect a variation in modal period are well known, and the known variation in these properties does not account for any measurable or significant shift in period or increase in seismic loads.

3.7.3.11 Torsional Effects of Eccentric Masses

The seismic mass model accounts for the effect of masses that are offset from the pipe centerline. Components with eccentric masses are modeled by placing the component's mass at its calculated center of gravity and connecting this mass to the pipe centerline with a rigid connection. The inertia forces calculated from the response spectra curves are applied at this lumped mass point. Therefore, any forces or moments, including torsion, resulting from eccentric masses are accounted for in the seismic analysis.

3.7.3.12 Piping Outside Containment

The differential movement of all Category I piping located outside containment is included in the stress analysis.

Movements are always considered to be out of phase in such a manner that the relative movements are maximum. These movements are then imposed on all anchors and restraints. The resulting stresses are classed as secondary and combined with thermal expansion stresses. These stresses are held below the appropriate code allowable limits.

BC-TOP-4 (Rev. 1), Section 6, discusses the techniques used to predict structural stresses in buried Category I piping for seismic loadings. The criteria require piping to remain functional when exposed to loadings predicted by use of the site design spectra. This is assured by limiting the strains to 40 percent of the ultimate strain of the pipe material.

3.7.3.13 Interaction of Other Piping With Category I Piping

The interface between Category I piping and non-Category I piping is always an anchor. The anchor prevents seismic motion on the non-Category I side from affecting the Category I side. The anchor is designed so that under the most conservative combination of thermal, weight, and seismic loads from both sides of the anchor, the anchor can maintain separation of motions. Seismic loads from the non-Category I side are estimated using a simplified dynamic analysis.

3.7.3.14 Field Location of Supports and Restraints

Seismic supports and restraints for seismic Category I piping are located so that the stresses, as determined by the dynamic analysis, are less than the appropriate code allowable limits. When rigid seismic supports result in excessive thermal loads on piping or equipment, snubbers or dampers are used.

The pipe support contractors' pipe restraint locations and detailed support drawings are reviewed by pipe stress engineers to ensure that they conform to requirements. In addition, a field inspection of the pipe supports is made by stress engineers to ensure that supports have been installed properly and meet design requirements.

For 2 in. and under Category I piping, a Bechtel field installation manual is provided so that field engineers can properly design and locate pipe supports and restraints. When the field engineers have completed their designs, they are reviewed by pipe stress engineers.

3.7.3.15 Seismic Analyses for Fuel Elements, Control Assemblies, and Control Rod Drives

Fuel assembly responses resulting from a safe shutdown earthquake were analyzed using time history integration techniques. The time history motions of the core plates and the core barrel used as the seismic input were obtained from the reactor vessel and internals system model. The acceleration spectra of time histories at the reactor vessel support elevation encompass the corresponding design spectra for the plant site.

The seismic response of the fuel assemblies is analyzed to determine structural design adequacy. Component stresses were obtained through the use of finite element computer modeling. Detailed discussions of the analysis methodology for evaluating the faulted condition loads on the fuel assembly design are contained in references 2, 4, and 5. The resulting combined seismic and LOCA loads are given in paragraph 4.2.1.1.2.

The control rod drive mechanisms (CRDMs) 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.

3.7.4 SEISMIC INSTRUMENTATION PROGRAM

3.7.4.1 Comparison with NRC Regulatory Guide 1.12

The original seismic instrumentation for the FNP was installed to meet the guide lines of NRC Regulatory Guide 1.12. The seismic instrumentation provides data to determine if the plant can continue to operate safely after an earthquake. New advances in seismic instrumentation technology have made available instruments that can analyze seismic data on site, faster and more accurately than the original instrumentation installed at FNP. The NRC has accepted the following EPRI reports as an acceptable approach to redefine the seismic monitoring requirements and determine plant action following an earthquake:

Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guideline for Implementation, EPRI TR-104239, July 1994

A Criterion for Determining Exceedance of the Operating Basis Earthquake, EPRI NP-5930, July 1988

Guidelines for Nuclear Plant Response to an Earthquake, EPRI NP-6695, December 1988

Standardization of the Cumulative Absolute Velocity (CAV), EPRI TR-100082, December 1991

The seismic monitoring system for FNP consists of the following instruments which meet the seismic requirement of EPRI TR-104239:

A. Three triaxial strong-motion acceleration sensors (time history) connected to an on-line computer for automatic data retrieval and analysis. System computes OBE exceedance and provides operator alarm.

B. Two self contained strong-motion accelerographs (time history).

3.7.4.2 <u>Location and Description of Instrumentation</u>

For location and summary of instrumentation see figure 3.7-62.

1. <u>Acceleration Sensors Connected to Seismic Monitoring Panel in Control Room</u>

   Three triaxial strong-motion acceleration sensors are located in the following plant areas:

   a. Two sensors are located on the exterior surface of the containment. One sensor is rigidly mounted on the containment base slab (elevation 104 ft) and the other is rigidly fastened to the containment wall directly above at elevation 212 ft 7 in.

   b. One sensor is located in the free field south of the containment, to be used as a free field instrument. It is far enough from the containment to avoid any effect of the containment on the sensor.

   The three axes of each triaxial strong-motion acceleration sensor have an orientation common with the others to permit accurate phase correlation of all channels. The sensors are rigidly mounted to the structure so that seismic records can be directly related to any structure movement.

The function of each acceleration sensor is to provide data on the frequency, amplitude, and phase relationship of the seismic response of the containment and other seismic Category I structures. These measurements will be used to evaluate the effect of the seismic disturbance on the structures and to assess their post-disturbance integrity. The seismic monitoring instrumentation is not connected to the plant safety systems.

The monitoring system is an automatic data retrieval and analysis system based on a high speed computer. The system remains on at all times continuously monitoring the signals from the acceleration sensors. The digital triggering system continuously monitors the signals and when the motion exceeds the adjustable, preset threshold, the system retrieves the time history data from storage and automatically performs the preprogrammed calculations for operator review. The system printer provides print and plot copies of the results for record and review. The system provides alarms for event, OBE, and loss of power. The system is powered by internal rechargeable batteries, which provide sufficient reserve power in the event of ac power failure.

2. <u>Self-Contained Triaxial Accelerographs (Time History)</u>

Two self-contained strong-motion triaxial accelerographs are installed at elevation 155 ft 0 in. in the diesel generator building and at elevation 167 ft 3 in. in the service water intake structure. These instruments will not be connected to the control room. These instruments are actuated by integral, digital triggers which have an adjustable, preset threshold for each channel. When the motion exceeds the threshold setting, the integral digital solid state recorder retrieves the time history data from storage and continues to record data until the unit de-triggers. The recorded data can be retrieved and analyzed to determine plant impact. The unit is powered by internal rechargeable batteries, which provide power in the event of ac power failure. The unit has external indicators for event and loss of ac power.

3.7.4.3 Control Room Operator Notification

The seismic monitoring panel will annunciate in the control room to alert the operator for an event, OBE exceedance and loss of power.

3.7.4.4 Comparison of Measured and Predicted Responses

If the seismic monitoring panel in the control room is triggered by an event from one or more of the strong-motion acceleration sensors connected to the panel, the system will automatically retrieve and analyze the data. The data analysis is rapid and automatic so operators can evaluate the event. An outline of the order of actions to be taken is given in figure 3.7-63.

3.7.5 SEISMIC DESIGN CONTROL

3.7.5.1 Seismic Design Control - Construction Phase

This section describes the design control measures which were used during the construction phase to ensure that adequate seismic input data (including necessary feedback from structural and system dynamic analysis) were specified to vendors of purchased Category I components and equipment. Three organizations are involved in procuring Category I components. These include Southern Company Services, Inc., Bechtel Power Corporation, and the nuclear steam supplier, Westinghouse.

The primary design organizations involved in the seismic design of the various structures, systems and components for the FNP are Westinghouse Electric Corporation, Bechtel Power Corporation, and Southern Company Services, Inc.

Components designed by others which fall under one of the three primary areas of responsibility are designed to the overall seismic requirements and checked by one of these organizations.

Responsibilities are as follows:

a. Westinghouse Electric Corporation is responsible for design of the nuclear steam supply system (NSSS). This includes the reactor vessel, steam generator, pressurizer, NSSS supports, primary coolant piping, and the emergency core cooling systems.

b. Bechtel Power Corporation is responsible for the design of the containment, auxiliary building, and all of the safety-related systems in these two buildings not furnished by Westinghouse. In addition, Bechtel has responsibility for reviewing seismic designs originated by Southern Company Services.

c. Southern Company Services, Inc., has responsibility for designing those structures and systems not contained in either the containment or the auxiliary building. These seismic designs are reviewed by Bechtel Power Corporation.

Westinghouse Supplied Equipment and Components

1. To ensure that Westinghouse supplied NSSS Category I mechanical components meet the seismic design criteria, the following procedures are implemented:

    a. Equivalent static acceleration factors are determined for each Category I component based upon the amplified ground acceleration response spectrum curves and the location of the component within the structure. This acceleration factor is included in the equipment specification, and the vendor must certify the adequacy of the component to meet this seismic requirement. Equipment specifications to vendors require that Westinghouse supplied Category I auxiliary pumps are designed by the vendor to operate during horizontal and vertical accelerations of 1.0 g and 0.6 g respectively and simultaneously. The sum of the primary stresses does not exceed Section III of the ASME Code for pressure-containing members. If qualification is by test results or by response analysis, the input frequencies for referenced "g" loadings is 5 to 15 hz.

Category I tanks are designed by Westinghouse to withstand the simultaneous horizontal and vertical forces resulting from the SSE. The vendor is also required to perform a static analysis and to comply with ASME Section III.

Category I valves are designed by the vendor to withstand seismic loadings equivalent to 3.0 g in the horizontal direction and to 2.0 g in the vertical direction and perform all functions within the specification.

b. The vendor's drawings and calculations are reviewed to determine whether the component meets all specification requirements.

c. Based upon engineering judgment and detailed analyses on similar equipment, the cognizant engineer will:

   i. Accept the component.

   ii. Reject the component as inadequate or recommend modifications.

   iii. Require that the engineering analysis section review the drawing details and perform a detailed analysis, if deemed necessary, using one of the methods described in the following paragraph.

d. To conform to the above criteria, seismic analysis of selected NSSS Category 1 components, including heat exchangers, pumps, tanks, and valves, is performed using one of three methods depending on the relative rigidity of the equipment being analyzed:

   i. Equipment that is rigid and rigidly attached to the supporting structure is analyzed for a g-loading equal to the acceleration of the supporting structure at the appropriate elevation

   ii. Equipment that is not rigid, and therefore a potential for response to the support motion exists, is analyzed for the peak of the floor response curve with appropriate damping values

   iii. In some instances, nonrigid equipment is analyzed using a multiple degree of freedom modal analysis, including the effect of modal participation factors and mode shapes, together with the spectral motions of the floor response spectrum defined at the support of the equipment. The inertial forces, moments, and stresses are determined in each mode. They are then summed using the square-root-of-the-sum-of-the-squares method.

The analyses described above are performed on mechanical equipment selected on a generic and size basis to verify that the equipment meets the seismic criteria listed in the equipment specification. Westinghouse has established the following criteria for protection and engineered safety system equipment:

For the SSE, the equipment is analyzed to ensure that it does not lose capability to perform its function; i.e., shut the plant down and maintain it in a safe shutdown condition.

To ensure that the equipment will perform its intended function during a safe shutdown earthquake, the deflections and stresses obtained by the seismic analysis are added to those associated with the operational mode of the equipment to verify that clearances are not exceeded and stresses are within allowable limits.

2. For protection grade instrumentation and control equipment:

For either earthquake (1/2 SSE or SSE), the equipment is designed to ensure that it does not lose its capability to perform its function; i.e., shut the plant down and maintain it in a safe shutdown condition.

For the SSE there may be permanent deformation of the equipment provided that the capability to perform its function is maintained.

Typical protection system equipment is subjected to type tests under simulated seismic accelerations to demonstrate its ability to perform its functions.

Type testing is being done on equipment by Westinghouse using conservatively large accelerations and applicable frequencies. Analyses such as those done for structures are not done for the reactor protection system equipment. However, the peak accelerations and frequencies used are checked against those derived by structural analyses of 1/2 SSE and SSE loadings.

A Westinghouse topical report, WCAP-7397-L, (and supplement), provides the seismic evaluation of safety related equipment. The type tests covered by this report are applicable to the Farley Nuclear Plant.

The control board is not considered to be protection equipment. Typical switches and indicators for safeguards components have been tested to determine their ability to withstand seismic forces without malfunction which would defeat automatic operation of the required component.

The control boards are stiff and past experience indicates that the amplification due to the board structure is sufficiently low so that the acceleration seen by the device is considerably less than the acceleration that the device was shown to withstand in testing.

Bechtel and Southern Company Services Specified Equipment and Components

Bechtel and Southern Company Services Specifications for Category I equipment incorporate a section on seismic design criteria. Category I valves and dampers are designed by the vendor to withstand seismic loadings vertical direction and to perform all functions within the specification. For other Category I equipment, the vendor was provided, as a part of the design specifications, the seismic response spectra, generated by a time history, which have been developed for the particular equipment location, and a list of damping factors. The specification requires the vendor to do one of the following:

1. Perform a seismic analysis based on the appropriate damping factor and response spectrum as well as the natural frequency of his equipment.

2. If it is not practical to calculate the natural frequency of the equipment, use the maximum acceleration of the spectrum curve for the seismic analysis.

3. Subject prototype equipment to a test demonstrating its ability to perform its intended function during and after seismic disturbance.

The current applicable seismic design data are provided to each vendor and certification is required from each vendor that his equipment will function during the SSE. This certification may consist of calculations checked by an engineer knowledgeable in the design of such equipment or of a written certification that the equipment has successfully passed tests of forces equal to or higher than those stated in the seismic requirement and has been exposed to these severe vibration requirements. The method of analysis, calculation, or testing is reviewed and approved by the responsible engineer.

3.7.5.2 Seismic Design Control - Operational Phase

During the operational phase, FNP will exercise the same controls as described in paragraph 3.7.5.1, except the responsibilities shall be as directed by Southern Nuclear Operating Company.

REFERENCES

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2. Gesinski, T. L., "Fuel Assembly Safety Analysis For Combined Seismic and Loss of Coolant Accident," WCAP-7950, July 1972.

3. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment," WCAP-7817 and Supplement I, December 1971.

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TABLE 3.7-1

PERCENTAGE OF CRITICAL DAMPING FACTORS

Type of Structure	1/2 Safe Shutdown Earthquake 0.05 g Ground Acceleration	Safe-Shutdown Earthquake (E') 0.10 g Ground Acceleration
Vital piping [(a)]	0.50	1.00
Welded steel plate assemblies	1.00	2.00
Welded steel frame structures	2.00	5.00
Bolted and riveted steel [(b)]	3.00	5.00
Reinforced concrete structures and equipment supports	2.00	5.00
Prestressed concrete structures	2.00	5.00
Soil damping	4.00	7.00

a. ASME Code Case N-411 damping values may be used as stated in paragraph 3.7.1.3.

b. Regulatory Guide 1.61 damping values are used in the analysis of the reactor vessel head assembly structure.

TABLE 3.7-2

SYSTEM PERIOD INTERVAL

No.	Frequency (Hz)	No.	Frequency (Hz)	No.	Frequency (Hz)	No.	Frequency (Hz)
1	0.10	31	0.43	61	1.87	91	8.07
2	0.105	32	0.45	62	1.96	92	8.48
3	0.11	33	0.48	63	2.06	93	8.90
4	0.115	34	0.50	64	2.16	94	9.35
5	0.12	35	0.53	65	2.27	95	9.81
6	0.125	36	0.55	66	2.38	96	10.30
7	0.13	37	0.58	67	2.50	97	10.82
8	0.14	38	0.61	68	2.63	98	11.36
9	0.15	39	0.64	69	2.76	99	11.93
10	0.155	40	0.67	70	2.90	100	12.52
11	0.16	41	0.70	71	3.04	101	13.15
12	0.17	42	0.74	72	3.19	102	13.81
13	0.18	43	0.78	73	3.35	103	14.50
14	0.19	44	0.81	74	3.52	104	15.22
15	0.20	45	0.86	75	3.70	105	15.98
16	0.21	46	0.90	76	3.88	106	16.78
17	0.22	47	0.94	77	4.08	107	17.62
18	0.23	48	0.99	78	4.28	108	18.50
19	0.24	49	1.04	79	4.50	109	19.43
20	0.25	50	1.09	80	4.72	110	20.40
21	0.27	51	1.15	81	4.96	111	21.42
22	0.28	52	1.20	82	5.20	112	22.49
23	0.29	53	1.26	83	5.46	113	23.62
24	0.31	54	1.33	84	5.74	114	24.80
25	0.32	55	1.39	85	6.02	115	26.04
26	0.34	56	1.46	86	6.33		
27	0.36	57	1.54	87	6.64		
28	0.37	58	1.61	88	6.97		
29	0.39	59	1.69	89	7.32		
30	0.41	60	1.78	90	7.69		

TABLE 3.7-3

METHODS USED FOR SEISMIC ANALYSES OF CATEGORY I STRUCTURES

	Method of Analysis			
Category I Structures	Response Spectra Analysis	Time-History Analysis	Applicable Stress or Deformations Criteria	Remarks
Containment	X	X	Refer to Section 3.8.1.5	
Auxiliary building	X	X	Refer to Section 3.8.4.5	
Diesel generator building	X	X	"	
River intake structure [a]	X	X	"	
Intake structure at storage pond	X	X	"	
Storage pond dam and dike (earth fill)	-	-	-	See Section 2.5
Vent stack	X	-	Refer to Section 3.8.4.5	
Pond spillway structure	X	-	"	
Electrical cable tunnel structure	X	-	"	
Category I outdoor tanks	X	-	"	

a. Original design (Category I) requirements are no longer required.

TABLE 3.7-4 (SHEET 1 OF 10)

METHODS USED FOR SEISMIC ANALYSES OF CATEGORY I SYSTEMS AND COMPONENTS

Category I Systems and Components	Method of Analysis: 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 loop piping [(1)] and piping to pressure boundary [(2)]		X[(2)]	X[(1)]		“	
RC system supports		X			“	
Surge pipe and fittings		X			“	
RC Thermowells				X	“	
Safety valves	X				“	
Relief valves	X				“	
Valves to RC system boundary	X				“	
CRDM head adapter plugs		X			“	
CHEMICAL AND VOLUME CONTROL SYSTEM						
Generative HX		X			See section 3.9	
Letdown HX		X			"	
Mixed-bed demineralizer		X			"	
Cation bed demineralizer		X			"	

TABLE 3.7-4 (SHEET 2 OF 10)

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 filter	X				"	
Volume control tank		X			"	
Charging/high head safety injection pump	X			X	"	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				Per API 650
Boric acid filter	X				"	
Boric acid transfer pump	X					
Boric acid blender		X			"	
Reactor makeup water storage tank		X			Wt/% 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	

TABLE 3.7-4 (SHEET 3 OF 10)

Category I Systems and Components	Method of Analysis: Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
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				"	
CONTAINMENT COOLING SYSTEM						
Fans		X			"	
Heat exchanger		X			"	
COMPONENT COOLING SYSTEM						
Pumps	X				"	
Heat exchangers		X			"	
Surge tank		X			"	Per API 650
SPENT FUEL POOL COOLING SYSTEM						
Spent fuel pool heat exchanger		X				
Spent fuel pool pump	X					

TABLE 3.7-4 (SHEET 4 OF 10)

Category I Systems and Components	Method of Analysis Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
BORON THERMAL REGENERATION SUBSYSTEM						
Moderating HX		X			"	
Letdown chiller HX		X			"	
Letdown reheat HX		X			"	
Thermal regeneration demineralizer		X			"	
LIQUID RECYCLE AND WASTE SUBSYSTEM						
Recycle holdup tank		X				Per API 650
Recycle evaporator feed pump	X				"	
Recycle evaporator feed demineralizer		X			"	
Recycle evaporator feed filter	X				"	
Recycle evaporator	X				"	
LIQUID RECYCLE AND WASTE SUBSYSTEM						
R.C. drain tank HX		X			"	
Waste holdup tank		X			"	
Waste evaporator feed pump	X				"	
Waste evaporator feed filter	X				"	
Waste evaporator	X				"	
Spent resin storage tank		X			"	
Spent resin sluice pump	X				"	

TABLE 3.7-4 (SHEET 5 OF 10)

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
	Method of Analysis					
Spent resin sluice filter	X				"	
Floor drain tank		X			"	
ES room sump pump [(a)]	X				"	
GAS HANDLING SUBSYSTEM						
Gas compressor	X			X	"	Vibration tests were conducted To determine seismic capability
Gas decay tanks		X			"	
Hydrogen recombiner	X				"	
EMERGENCY DIESEL FUEL OIL SYSTEM						
Transfer pumps		X			"	
Fuel oil tanks	X				"	
SERVICE WATER SYSTEM						
Pumps		X			"	
Strainers		X			"	
RIVER WATER SYSTEM						
Pumps[(a)]		X			"	
FUEL HANDLING SYSTEM						
Fuel manipulator crane		X			"	
Fuel transfer tube	X				"	
Underwater fuel conveyor car and rail system	X				"	
Fuel pool bridge crane		X			"	
Polar crane	X				"	

a. Pumps originally seismically analyzed as Seismic Category I, but have been downgraded to Seismic Category II.

TABLE 3.7-4 (SHEET 6 OF 10)

Category I Systems and Components	Method of Analysis: Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Crane supports	X				"	
REFUELING WATER SYSTEM						
Storage tank		X			Wt/% exceeding 90% of yield stresses and/or loss of function	
AUXILIARY BUILDING VENTILATION SYSTEM						
ES AIR COOLING UNITS						
Heat exchanger		X			"	
Fan		X			"	
PENETRATION ROOM FILTRATION SYSTEM						
Fans		X			"	
Filters (HEPA and charcoal)		X			"	
CONTROL ROOM VENTILATION SYSTEM						
Fans		X			"	
Filters		X			"	
Air handling unit		X			"	
Condensing unit				X	"	
DIESEL BUILDING VENTILATION SYSTEM						
Fans				X	"	
Filters				X	"	
MAIN STEAM SYSTEM						
Isolation valves	X				"	

TABLE 3.7-4 (SHEET 7 OF 10)

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
FEEDWATER 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				"	
ELECTRICAL COMPONENTS AND SYSTEMS						
4160-v switchgear (engineered safe-guard buses)				X	"	
4160-v to 600-v trans-formers (associated with engineered safe-guard systems)		X			"	
600-v load centers (engineered safe-guard buses)				X	"	Test on prototype
600-v and 208-v motor-control centers (associated with engineered safeguard systems)				X	"	Test on prototype
125-v dc station batteries				X	"	Test on three cells

TABLE 3.7-4 (SHEET 8 OF 10)

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
	Method of Analysis					
Inverters, 125-v dc to 120-v ac (vital ac instrumentation distribution panels)				X	See section 7.1	
ELECTRICAL COMPONENTS AND SYSTEMS						
125-v dc distribution panels				X	"	Tests on two panels selected at random
120-v vital ac instrumentation and regulated ac distribution panels				X	"	
125-v dc switchgear				X	"	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	Wt/% 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 1000-V dc power and 600-V control and instrument cables
Turbine driven auxiliary feedwater pump uninterruptable power supply				X	See section 7.1	

TABLE 3.7-4 (SHEET 9 OF 10)

Method of Analysis

Category I Systems and Components	Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
Emergency power board		X		X	"	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	Wt/% loss of function	
Equipment contained within balance of plant instrument cabinets		X		X	"	
Containment purge radiation monitors		X		X	"	
Fuel handling area radiation monitors		X		X	"	
SAMPLING SYSTEM						
1. Cabinet		X			Wt/% exceeding 90% of yield stresses and w/o loss of function	
2. Tubing, valves, coolers, sample vessels	X				Wt/% loss of function	

TABLE 3.7-4 (SHEET 10 OF 10)

Category I Systems and Components	Method of Analysis: Equivalent Static Load	Response Spectra Analysis	Time-History Analysis	Tests	Applicable Stress or Deformation Criteria	Remarks
ELECTRICAL COMPONENTS AND SYSTEMS						
Balance of plant field mounted instruments	X			X	"	
Instrument valves for field mounted instruments	X				"	
Instrument lines for field mounted instruments	X				Wt/% exceeding code allowable stresses	
Isolation devices for output of AMSAC				X		

TABLE 3.7-5

NATURAL FREQUENCIES FOR CATEGORY I STRUCTURES

Structure	Direction: North-South Mode Frequency (Hz)					East-West Mode Frequency (Hz)					Vertical Mode Frequency (Hz)				
	1st	2nd	3rd	4th	5th	1st	2nd	3rd	4th	5th	1st	2nd	3rd	4th	5th
Containment & internal structure	4.20	12.62	16.87	25.00	33.41	4.20	13.19	17.54	27.01	33.76	10.77	21.17	43.47	-	-
Auxiliary building	8.89	25.79	35.79	41.76	-	8.20	23.10	29.66	38.82	-	9.85	49.0	-	-	-
Diesel generator building	1.44	31.54	54.69	-	-	1.44	34.44	53.35	-	-	14.01	74.09	-	-	-
River intake [b] structure	.23	6.60	16.33	28.69	41.86	.26	7.66	21.68	44.28	-	10.22	41.65	-	-	-
Intake structure storage pond	.15	.27	1.19	11.28	44.29[a] 12.47	.24	1.14	9.71	12.51	36.51	5.14	42.72	-	-	-

---

a. 6th mode

b. The river intake structure was originally designed as a Category I structure, but has since been downgraded to non-seismic.

TABLE 3.7-6

COMPARISON OF TRANSLATIONAL AND TORSIONAL FREQUENCIES

Structure	Mode NR	Horizontal Translational Frequency (Hz)	Torsional Frequency (Hz)[a]
Internal structures	1	16.21	26.72
	2	41.51	63.31
Auxiliary building	1	8.89	34.92
	2	25.77	120.35
	3	35.97	-
	4	41.76	-

a. Frequencies greater than 25 Hz.

TABLE 3.7-7

CONTAINMENT SHELL
COMPARISON OF RESPONSE SPECTRUM AND TIME HISTORY ANALYSIS,
SAFE SHUTDOWN EARTHQUAKE
(EAST - WEST DIRECTION)

	Absolute Acceleration, g	
Elevation, ft	Response Spectrum Analysis	Time History Analysis
99.5	.0630[(a)]	.1206
116.50	.0782[(a)]	.1293
129.00	.0933[(a)]	.1341
142.00	.1077	.1384
155.00	.1215	.1564
174.00	.1451	.1742
193.00	.1705	.2008
212.58	.2011	.2380
228.12	.2255	.2706
243.67	.2519	.3049
251.42	.2659	.3216
269.17	.2927	.3514
282.17	.3101	.3705

a. A minimum of 0.10 g has been used.

TABLE 3.7-8

MAXIMUM ALLOWABLE SPAN BETWEEN SEISMIC RESTRAINTS FOR PIPES 2 IN. AND UNDER

Pipe Diam. (in.)	Max. Allowable Span of Pipe + Water + Insulation (ft)	Natural Frequency fn[(a)] (Hz)
3/8	3.5	26.6
1/2	4.0	27.1
3/4	5.0	22.6
1	6.0	20.2
1-1/2	7.0	22.1
2	8.0	21.4

a. Figures based on a limiting natural frequency of 20 Hz, derived from the following equation:

$$fn = \frac{\alpha}{L^2}\sqrt{\frac{EI}{W}}$$ where:

fn = natural frequency - Hz

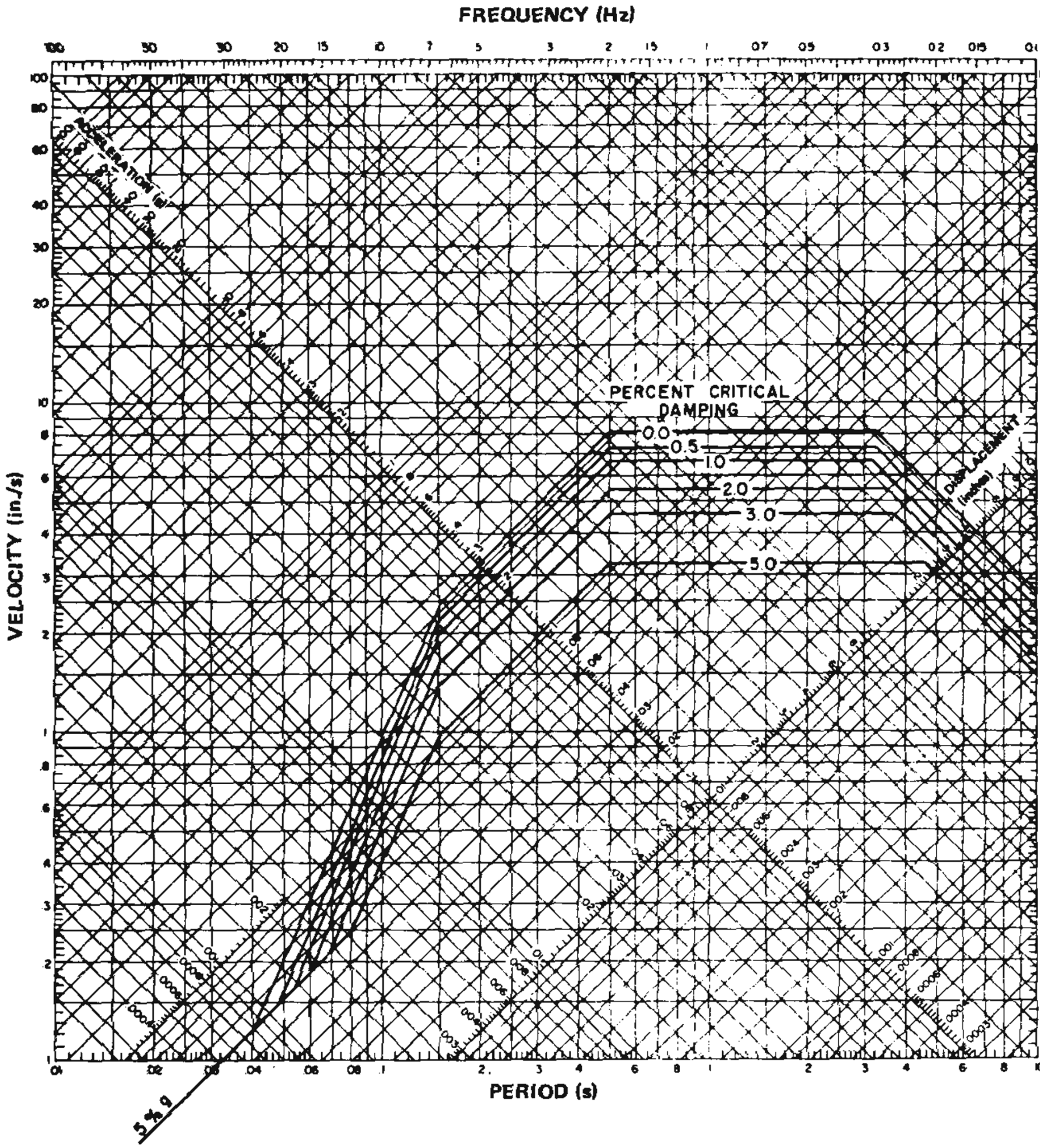
α = 0.743

L = span ft.

E = Young's modulus of elasticity - psi

I = Moment of inertia (in.^{4})

W = Pipe weight/ft



NOTES:

1 For horizontal ½ safe shutdown earthquake spectra, multiply by 1.00 for all values of acceleration.

2 For vertical ½ safe shutdown earthquake spectra, multiply by 0.67 for all values of acceleration.

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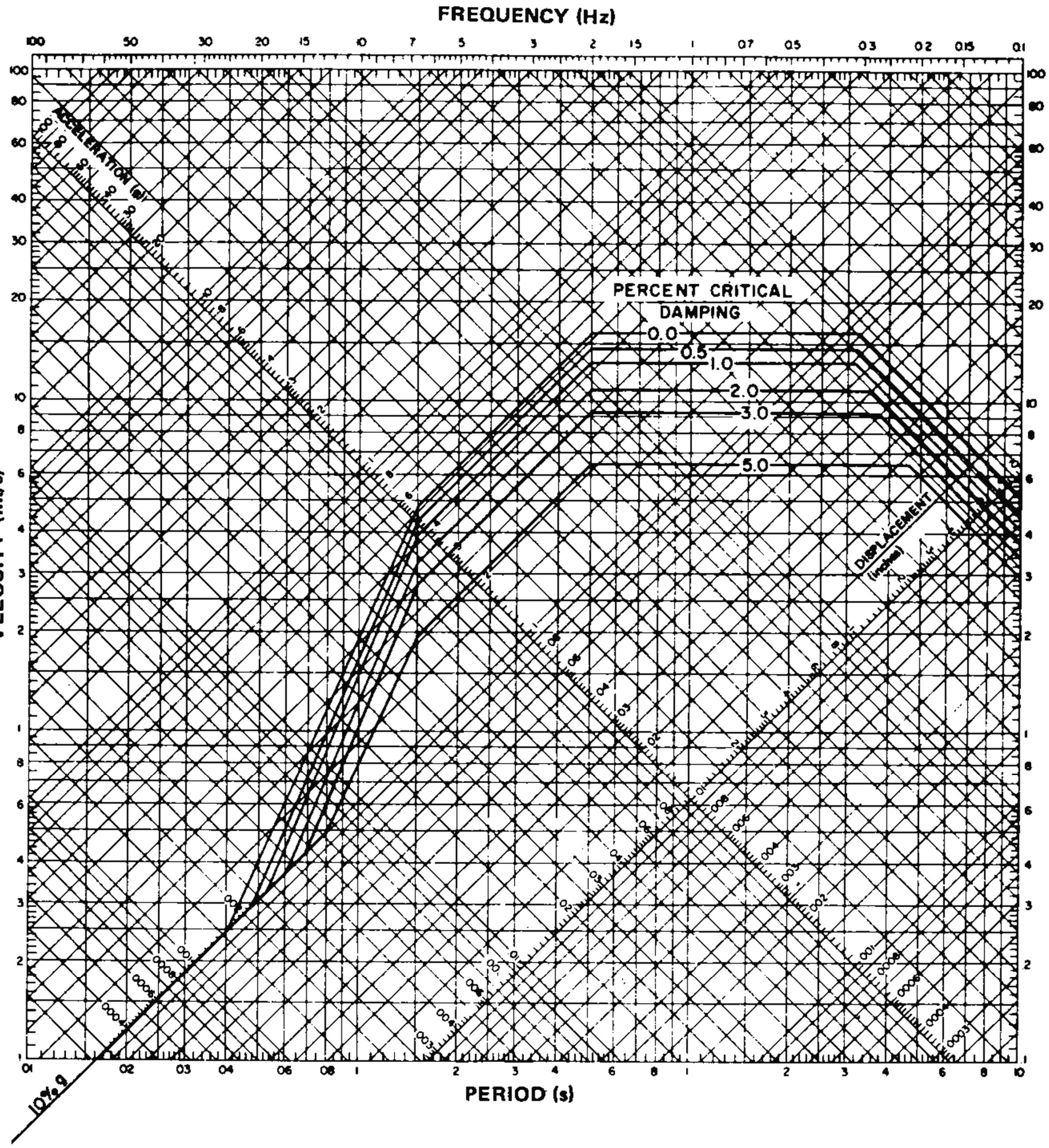
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	½ SAFE SHUTDOWN EARTHQUAKE GROUND SPECTRA 0.05 g (HORIZONTAL & VERTICAL) FIGURE 3.7-1



NOTES:

1 For horizontal safe shutdown earthquake spectra, multiply by 1.00 for all values of acceleration.

2 For vertical safe shutdown earthquake spectra, multiply by 0.67 for all values of acceleration.

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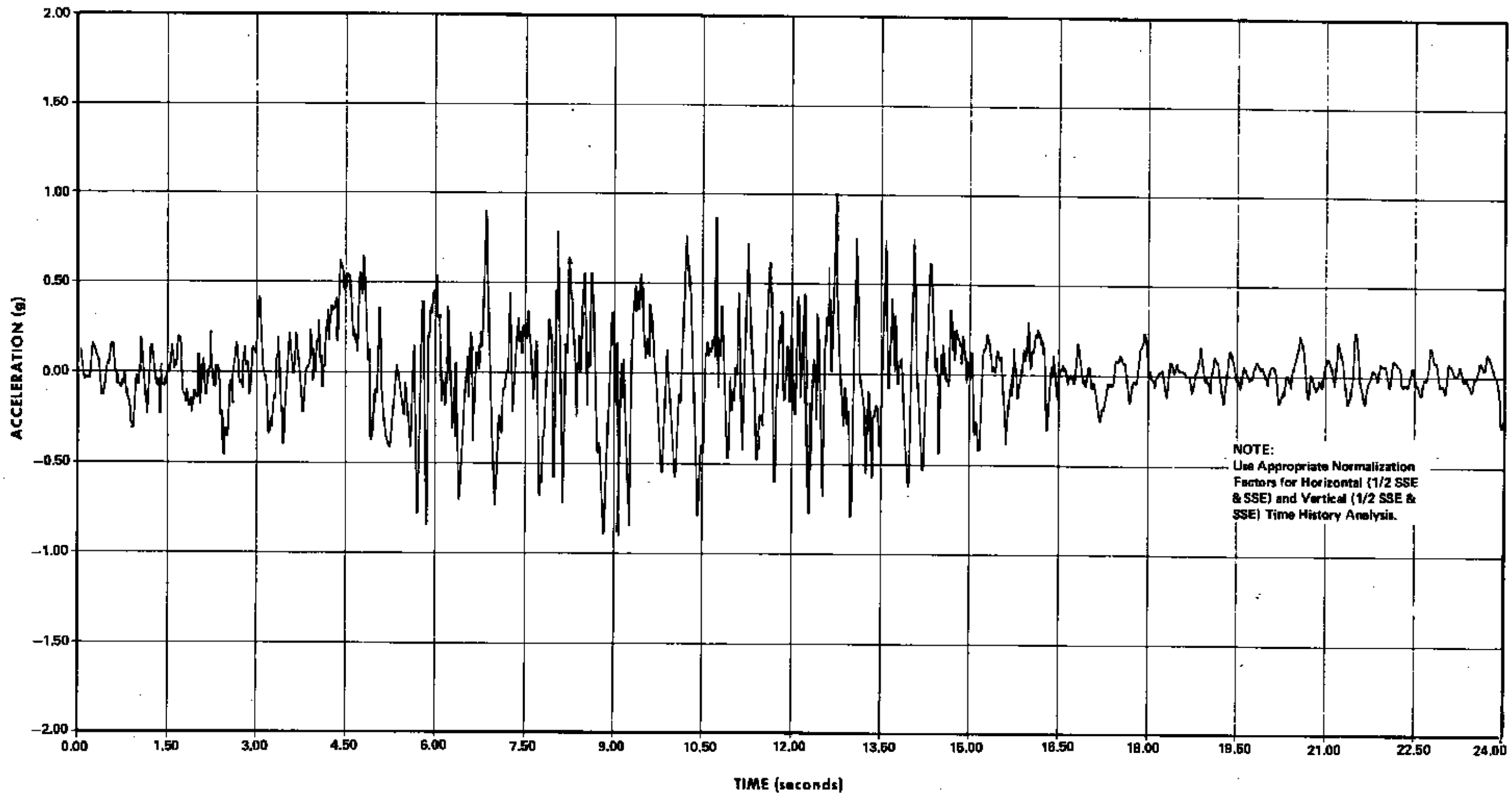
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	SAFE SHUTDOWN EARTHQUAKE GROUND SPECTRA 0.10 g (HORIZONTAL & VERTICAL)
		FIGURE 3.7-2



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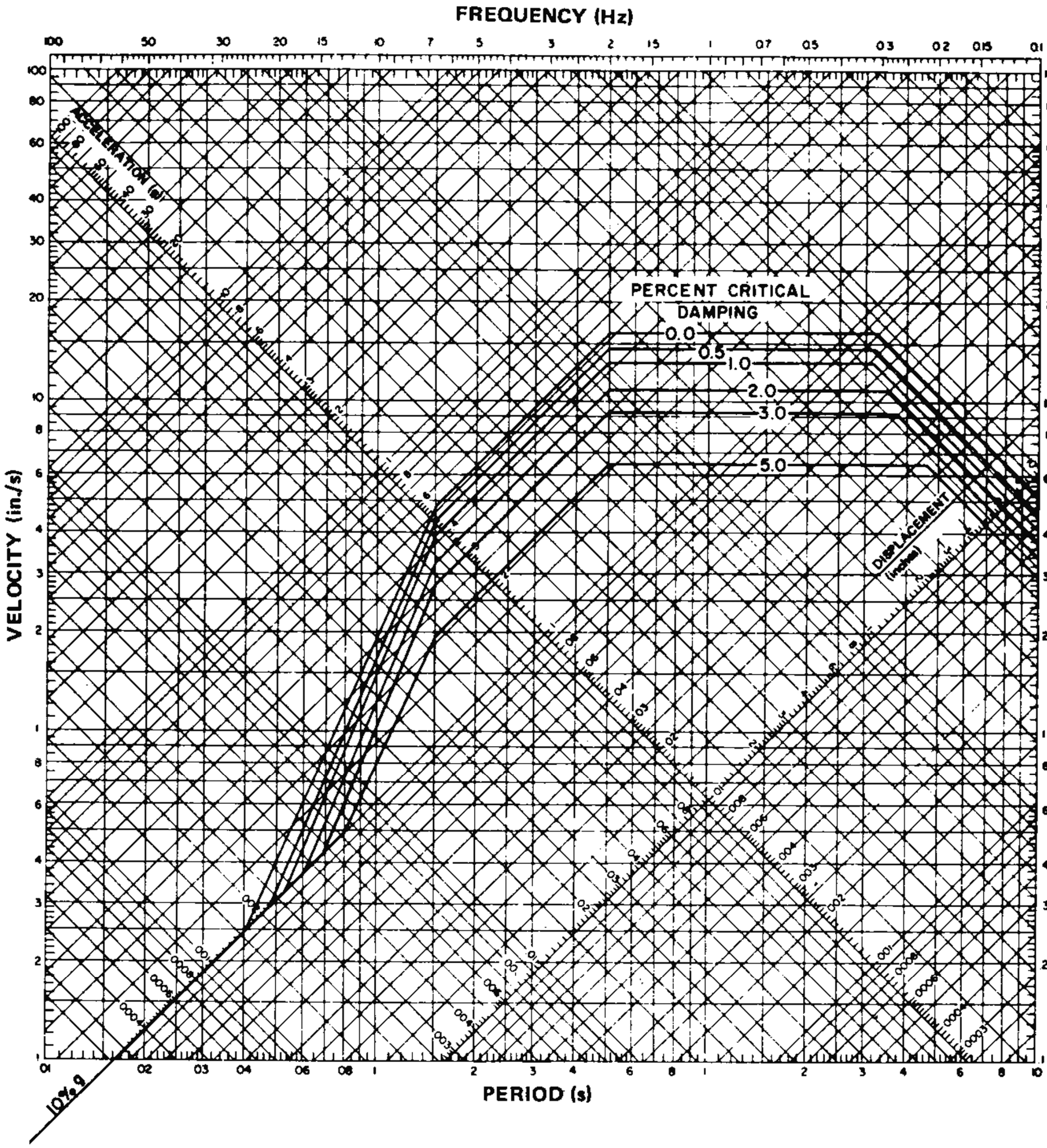
SYNTHESIZED TIME HISTORY (1/2 SSE & SSE)

FIGURE 3.7-3



NOTES:

1 For horizontal safe shutdown earthquake spectra, multiply by 1.00 for all values of acceleration.

2 For vertical safe shutdown earthquake spectra, multiply by 0.67 for all values of acceleration.

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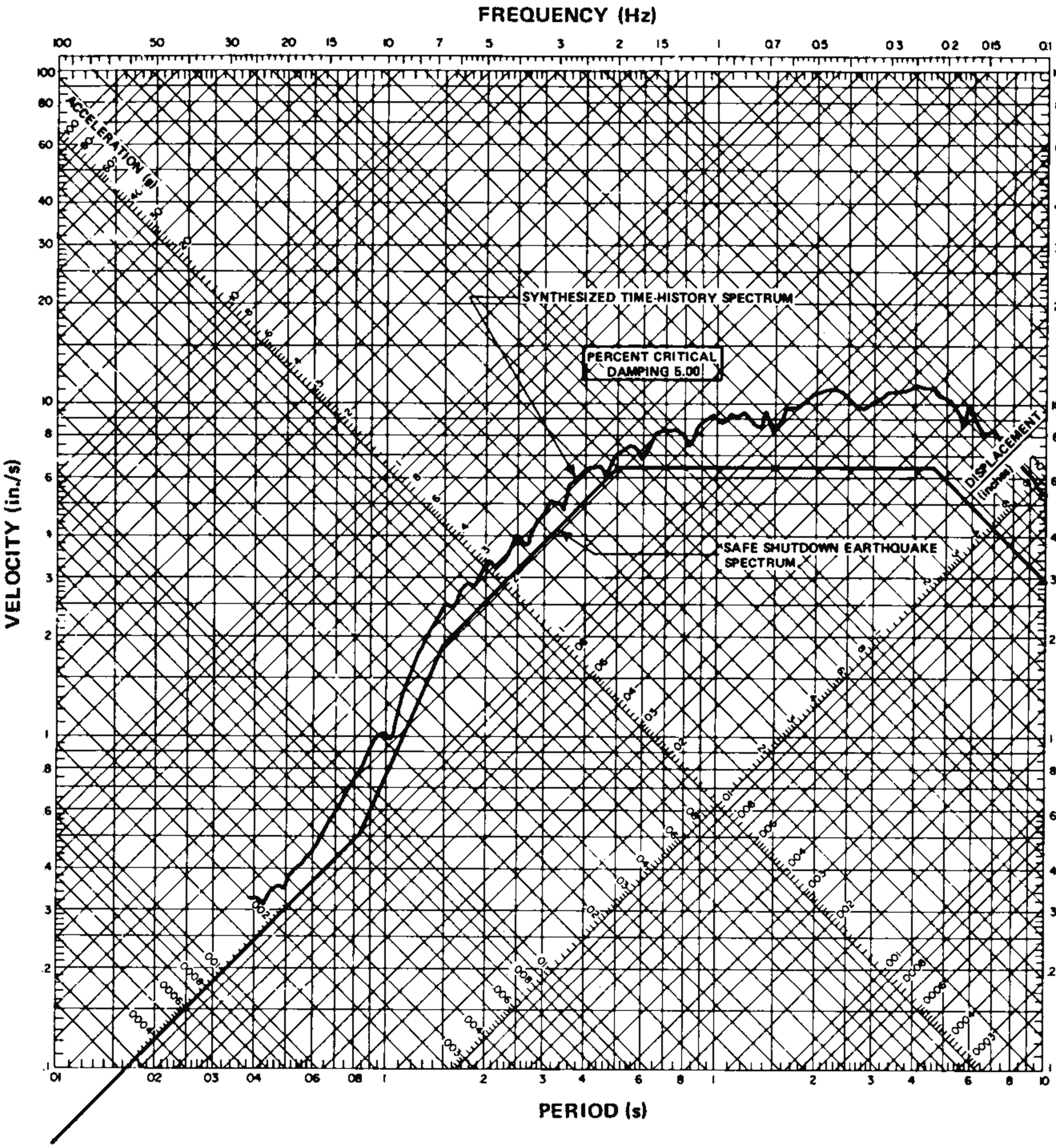
TIME HISTORY SPECTRUM ENVELOPE ON RESPONSE SPECTRUM (1/2 SSE)

FIGURE 3.7-4



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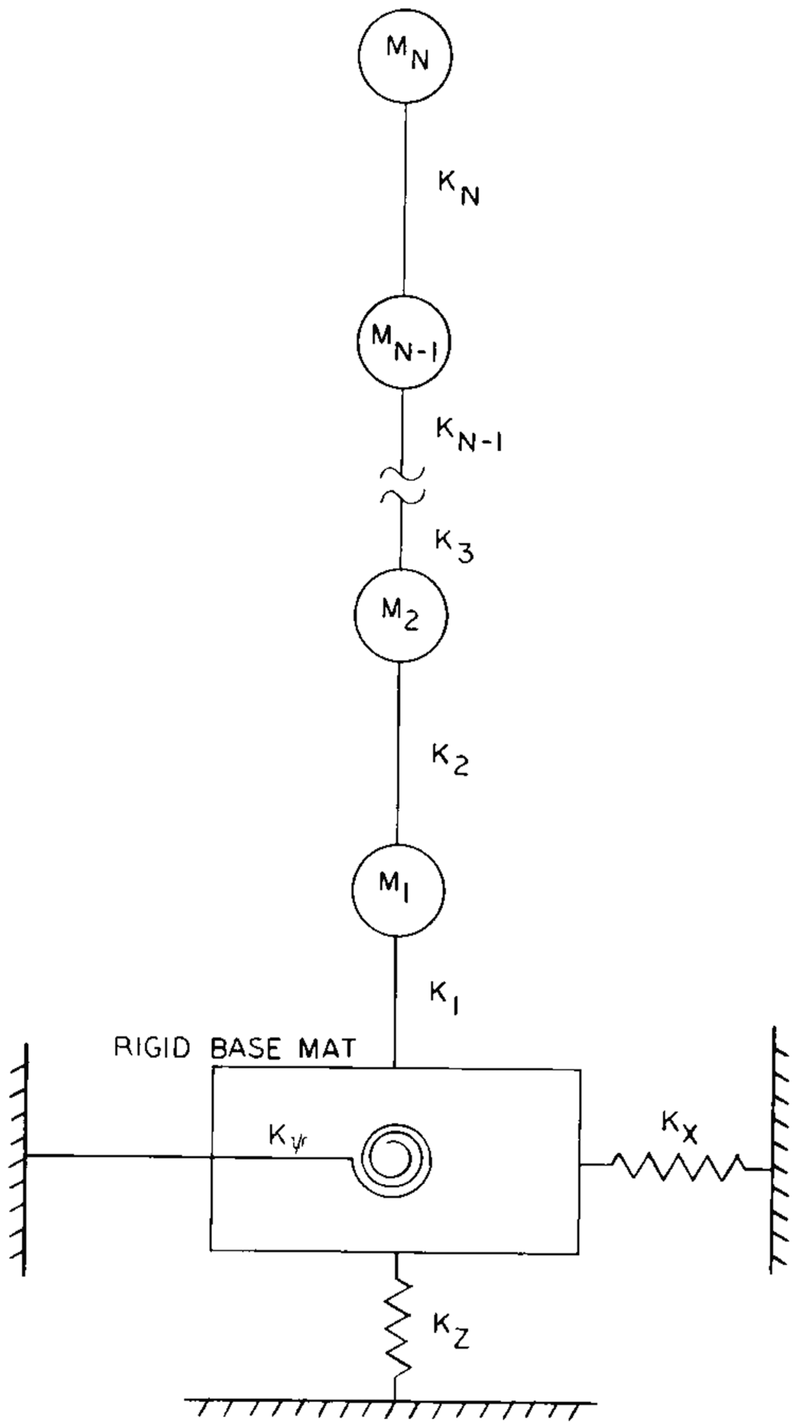
TIME HISTORY SPECTRUM ENVELOPE ON RESPONSE SPECTRUM (SSE)

FIGURE 3.7-5



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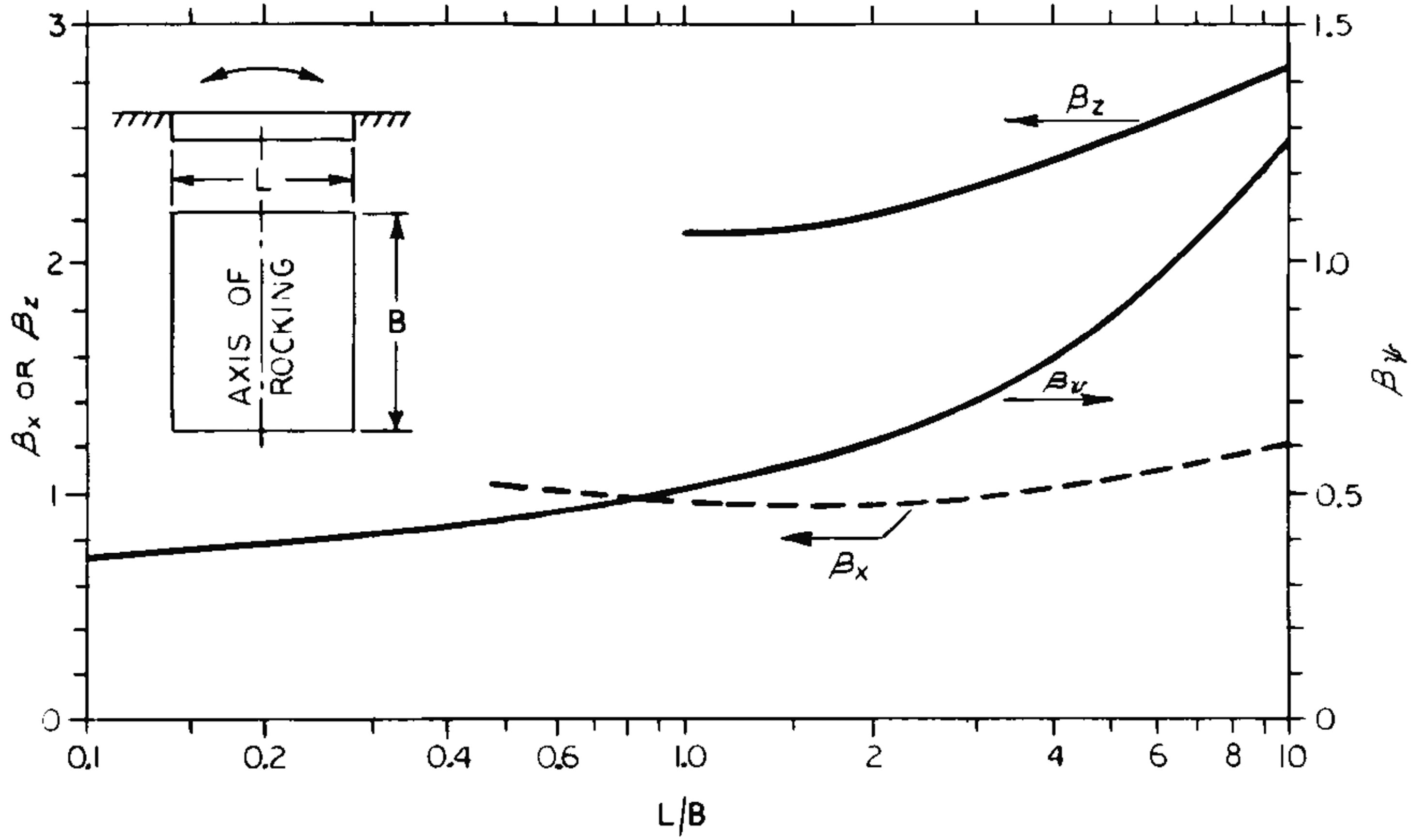
A LUMPED-MASS MODEL OF
STRUCTURE FOUNDATION SYSTEM

FIGURE 3.7-6



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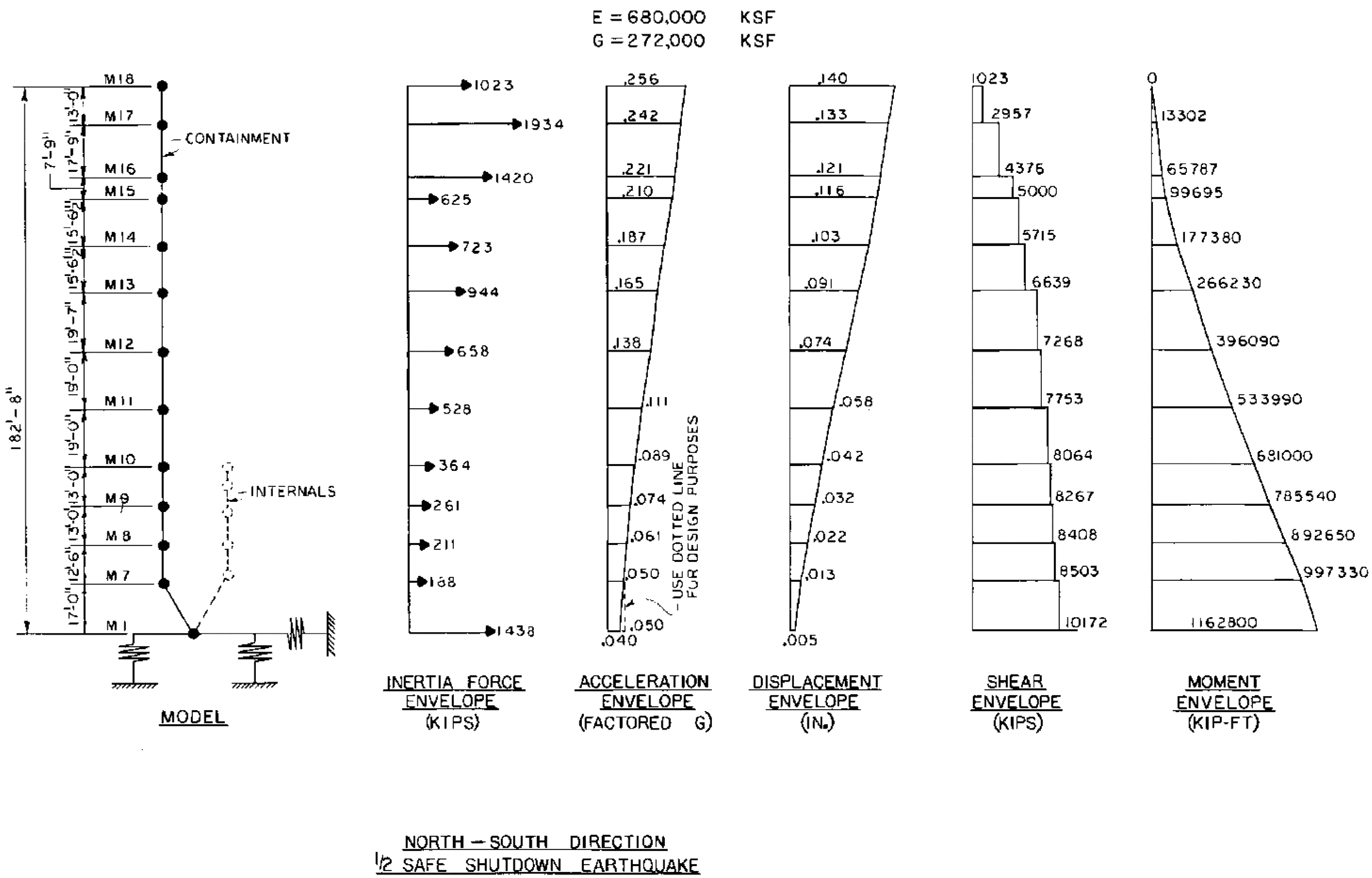
CONSTANTS β_x, β_ψ AND β_z FOR RECTANGULAR BASES

FIGURE 3.7-7



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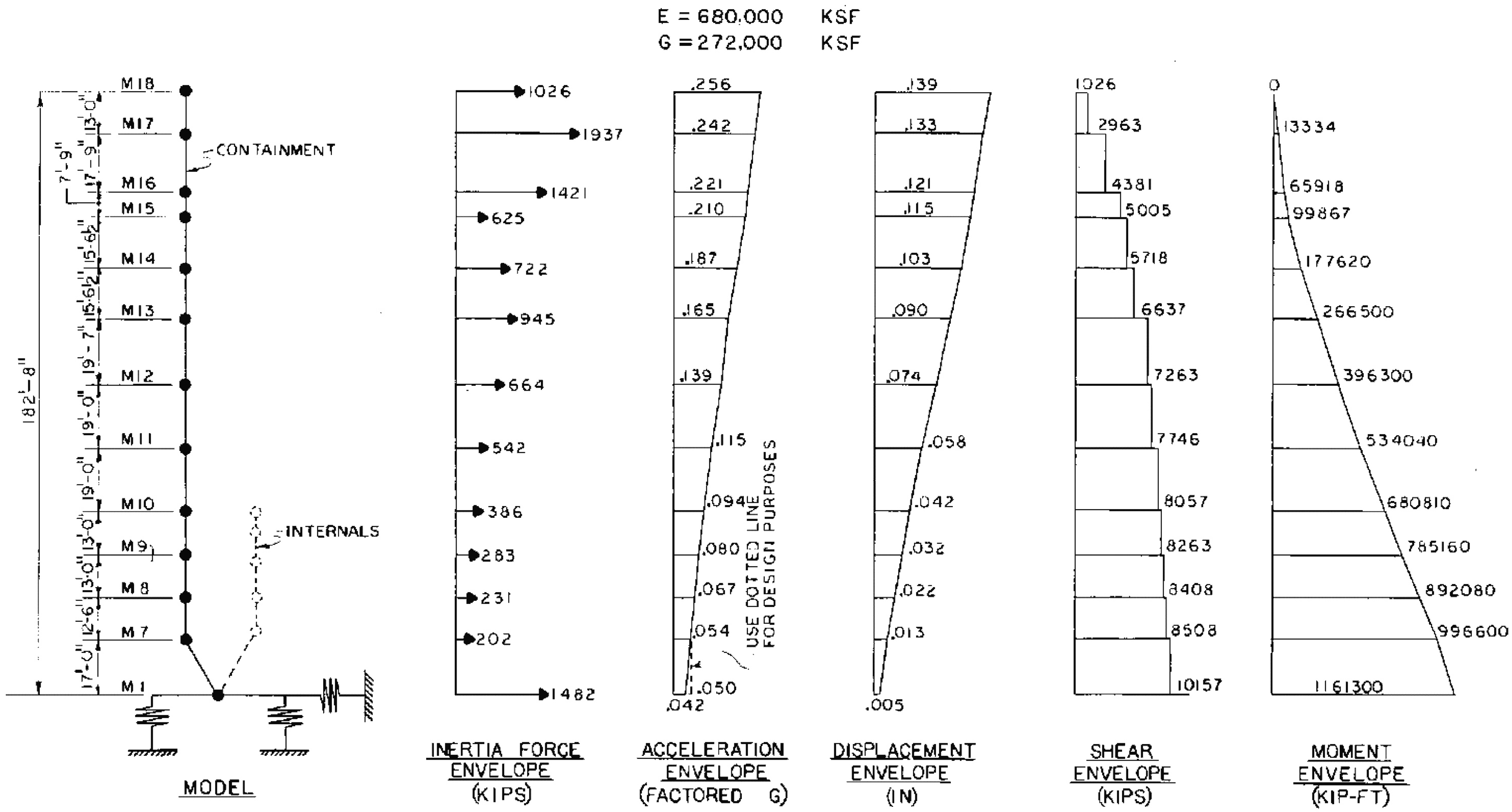
CONTAINMENT – SEISMIC RESULTS

FIGURE 3.7-8



EAST — WEST DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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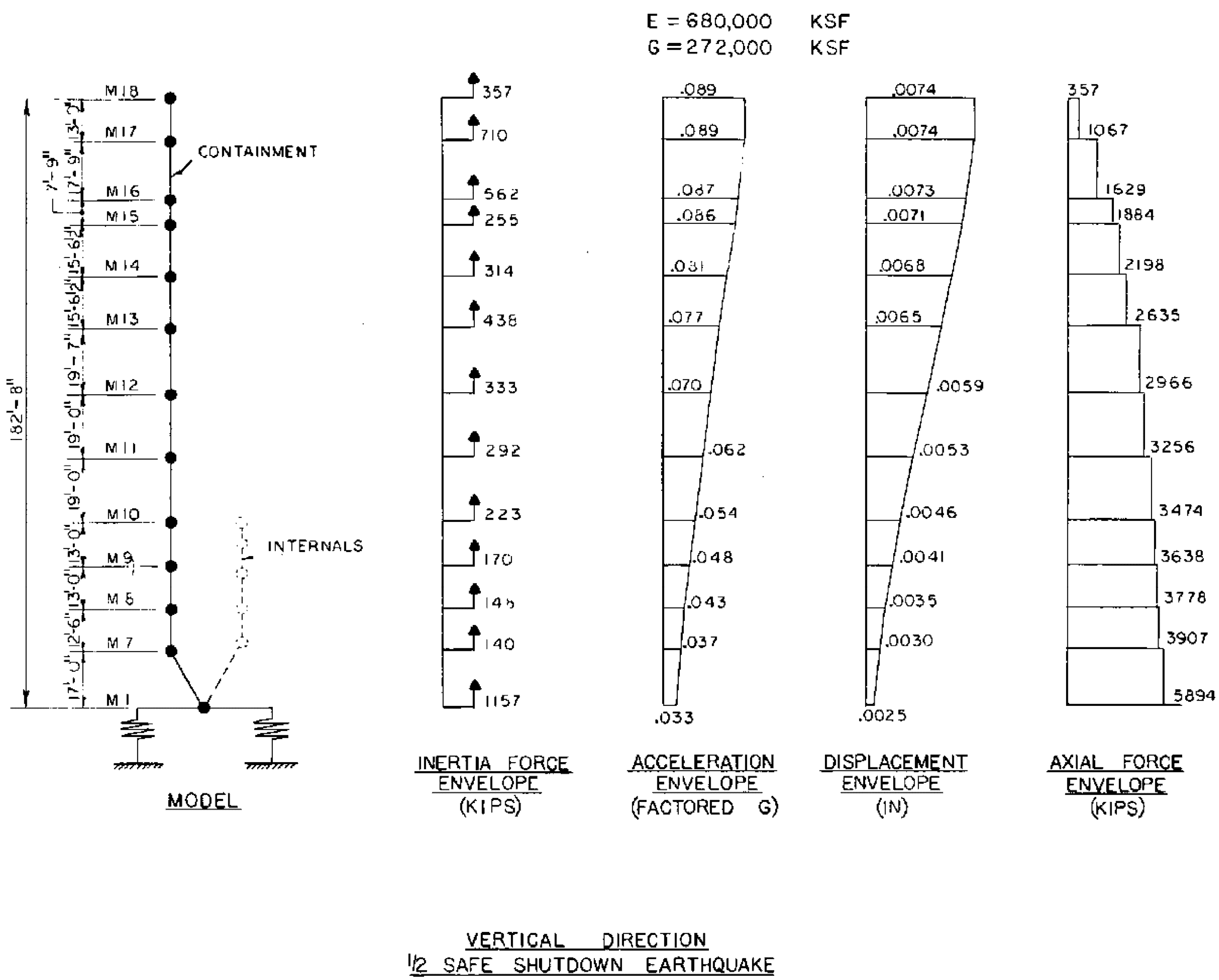
CONTAINMENT – SEISMIC RESULTS

FIGURE 3.7-9



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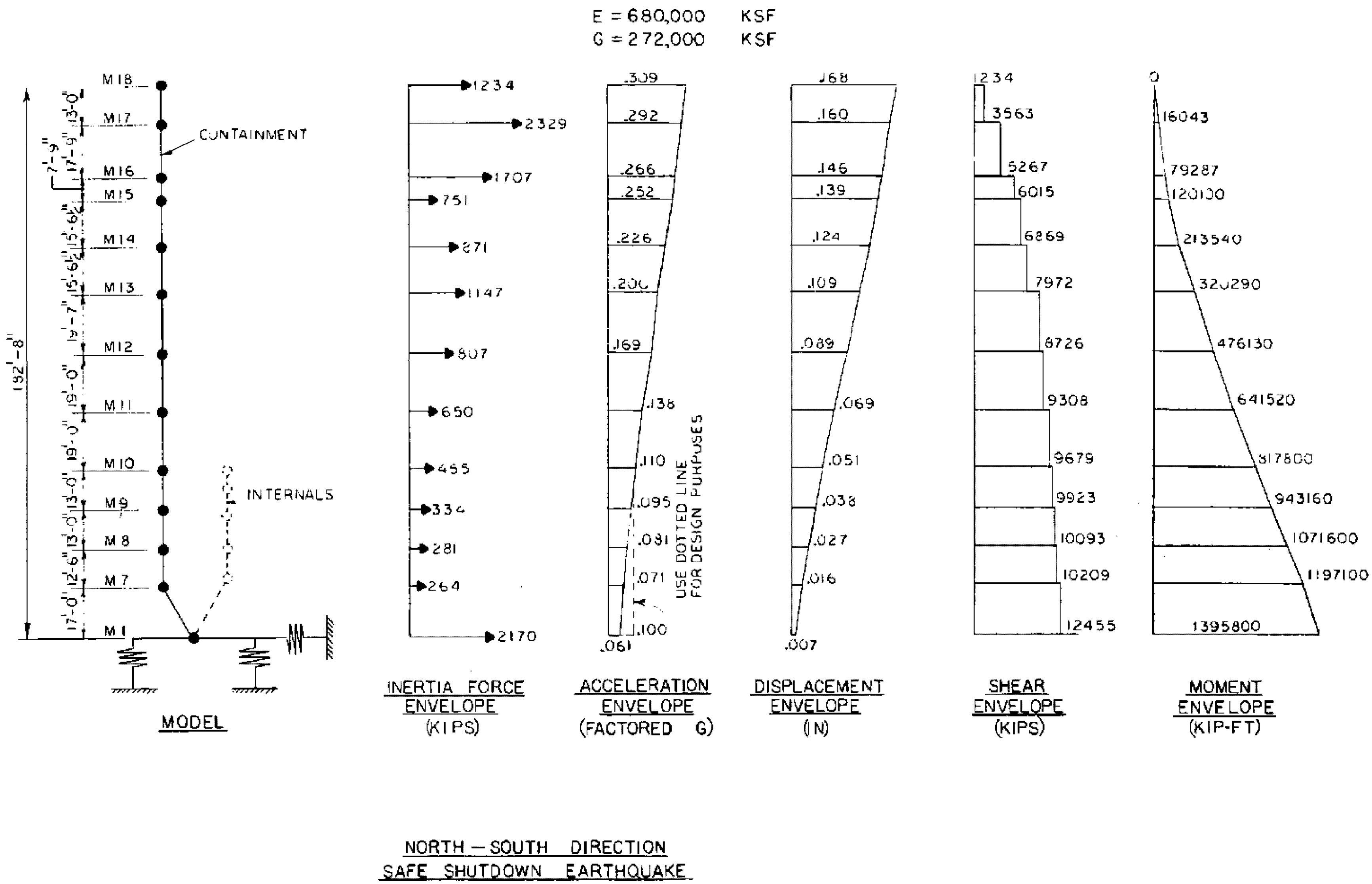
CONTAINMENT – SEISMIC RESULTS

FIGURE 3.7-10



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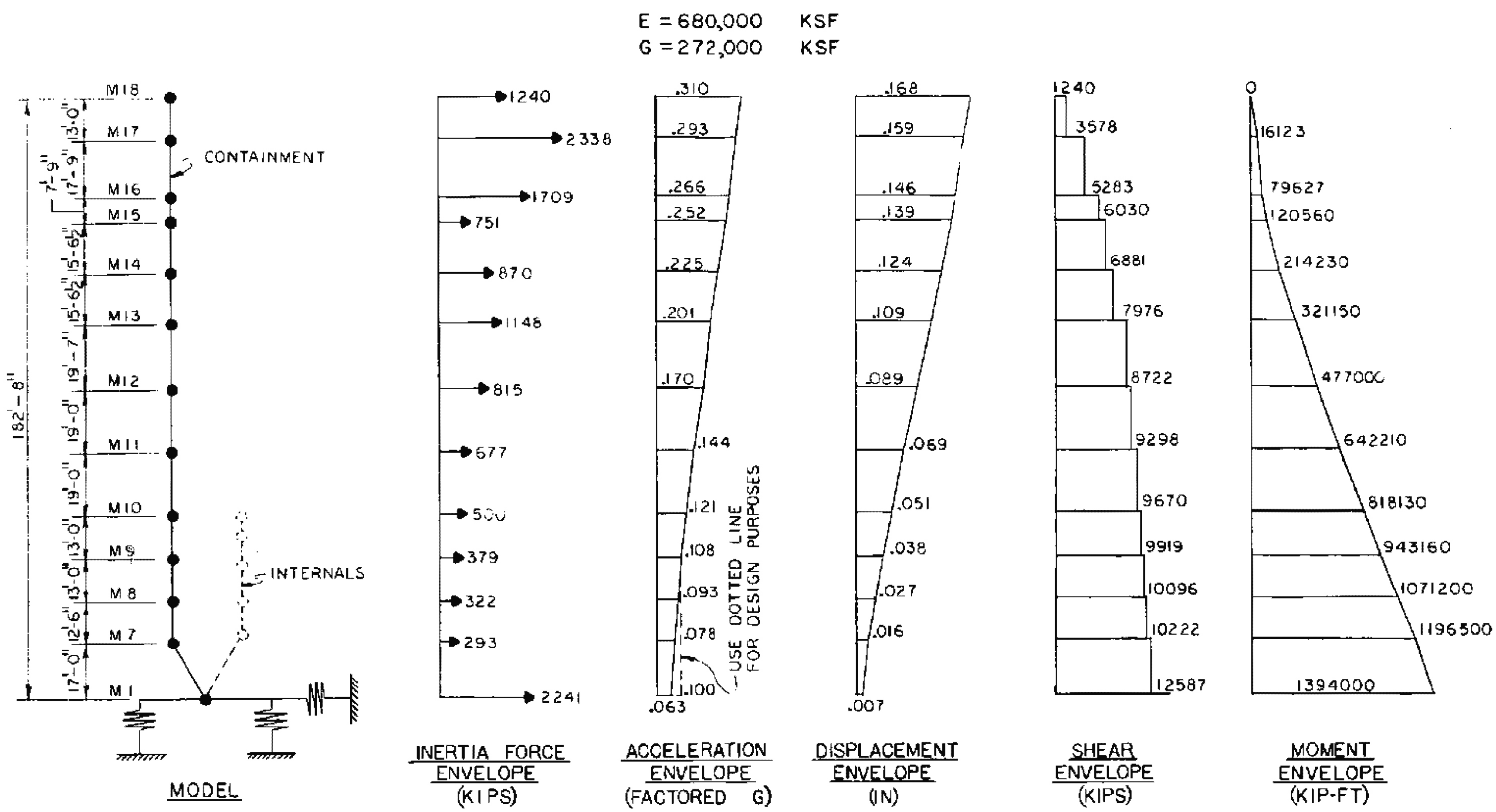
CONTAINMENT – SEISMIC RESULTS

FIGURE 3.7-11



EAST – WEST DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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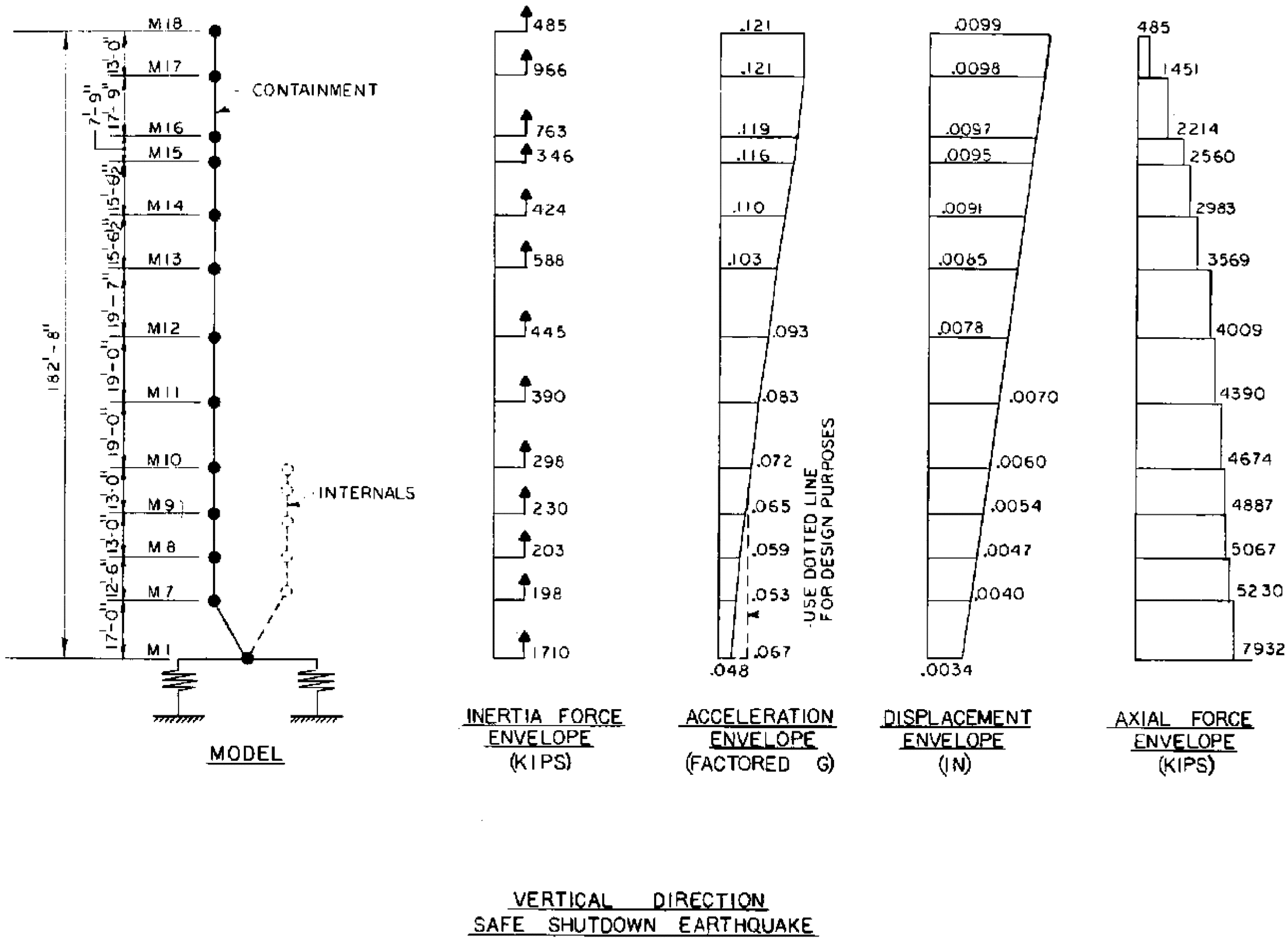
CONTAINMENT – SEISMIC RESULTS

FIGURE 3.7-12



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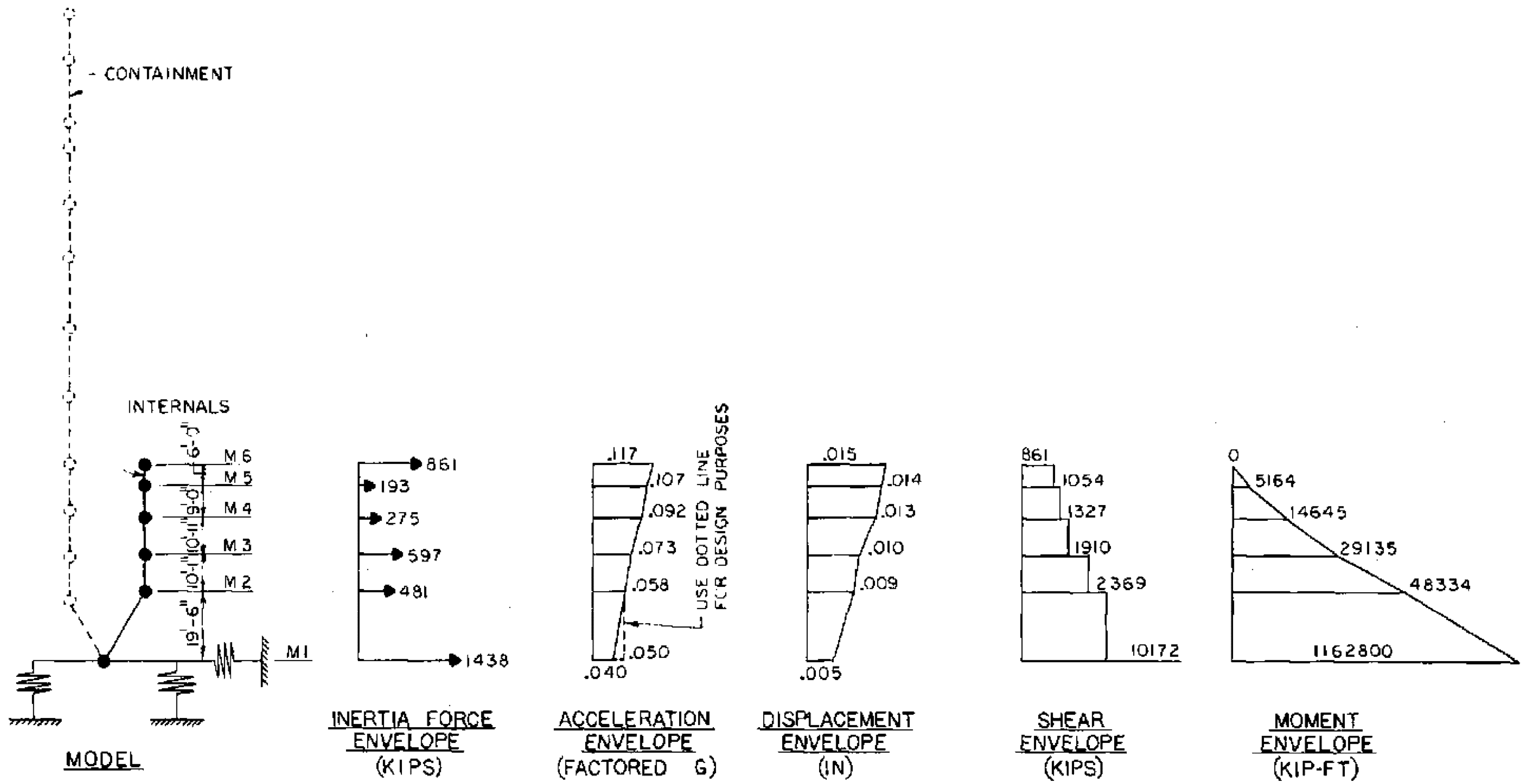
CONTAINMENT – SEISMIC RESULTS

FIGURE 3.7-13

E = 680,000 KSF
G = 272,000 KSF



NORTH – SOUTH DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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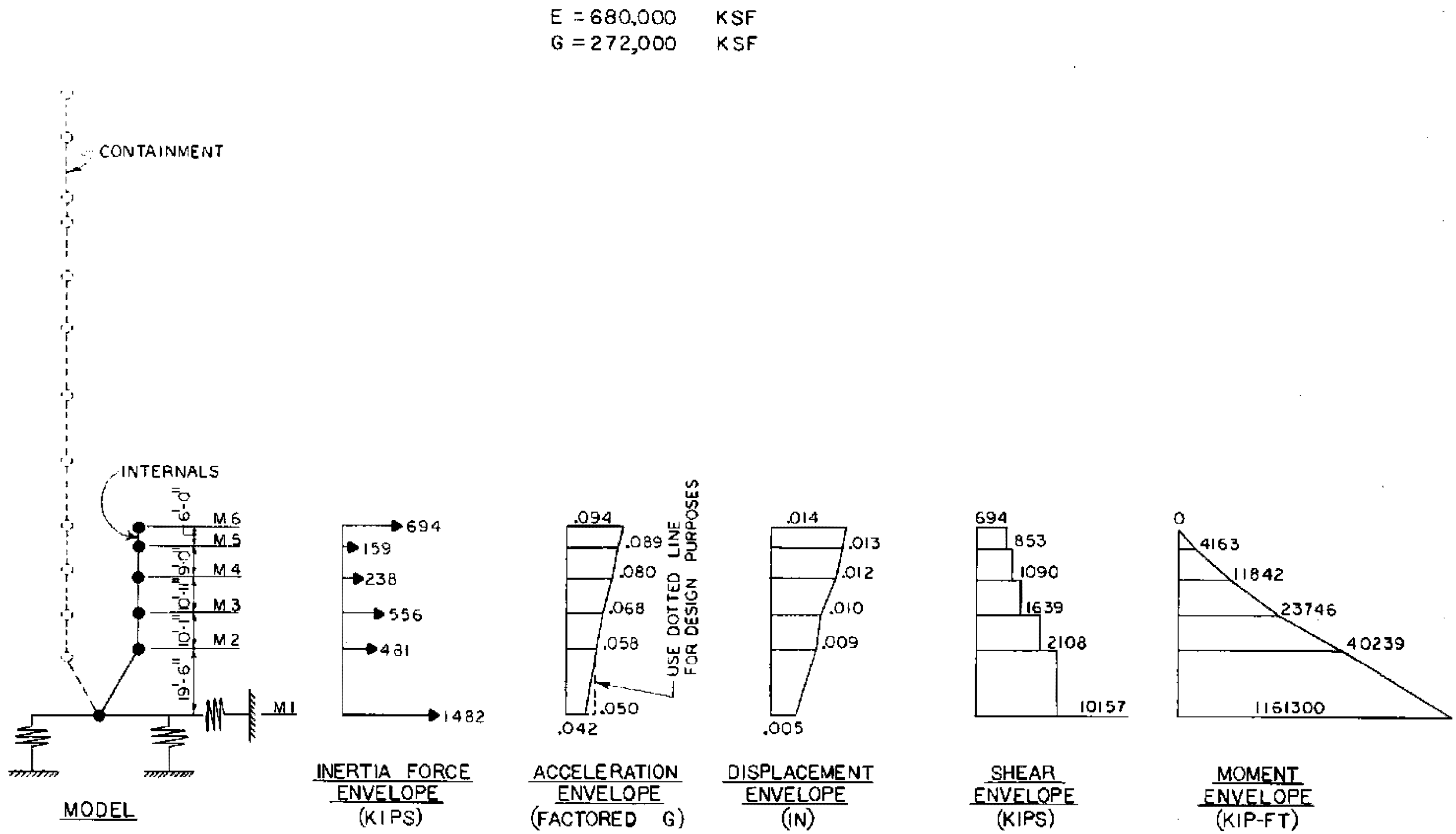
INTERNAL STRUCTURE – SEISMIC RESULTS

FIGURE 3.7-14



EAST – WEST DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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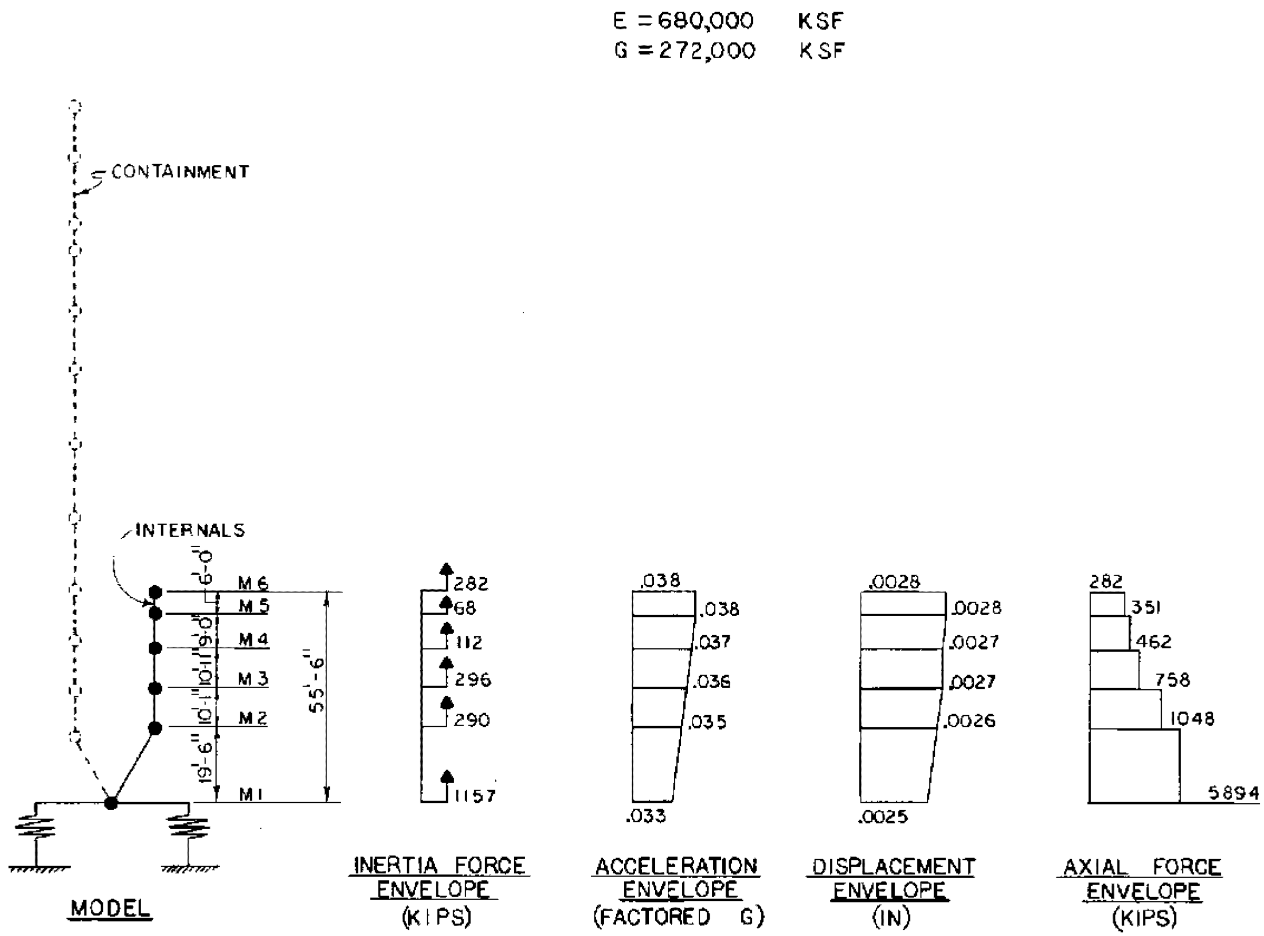
INTERNAL STRUCTURE – SEISMIC RESULTS

FIGURE 3.7-15



VERTICAL DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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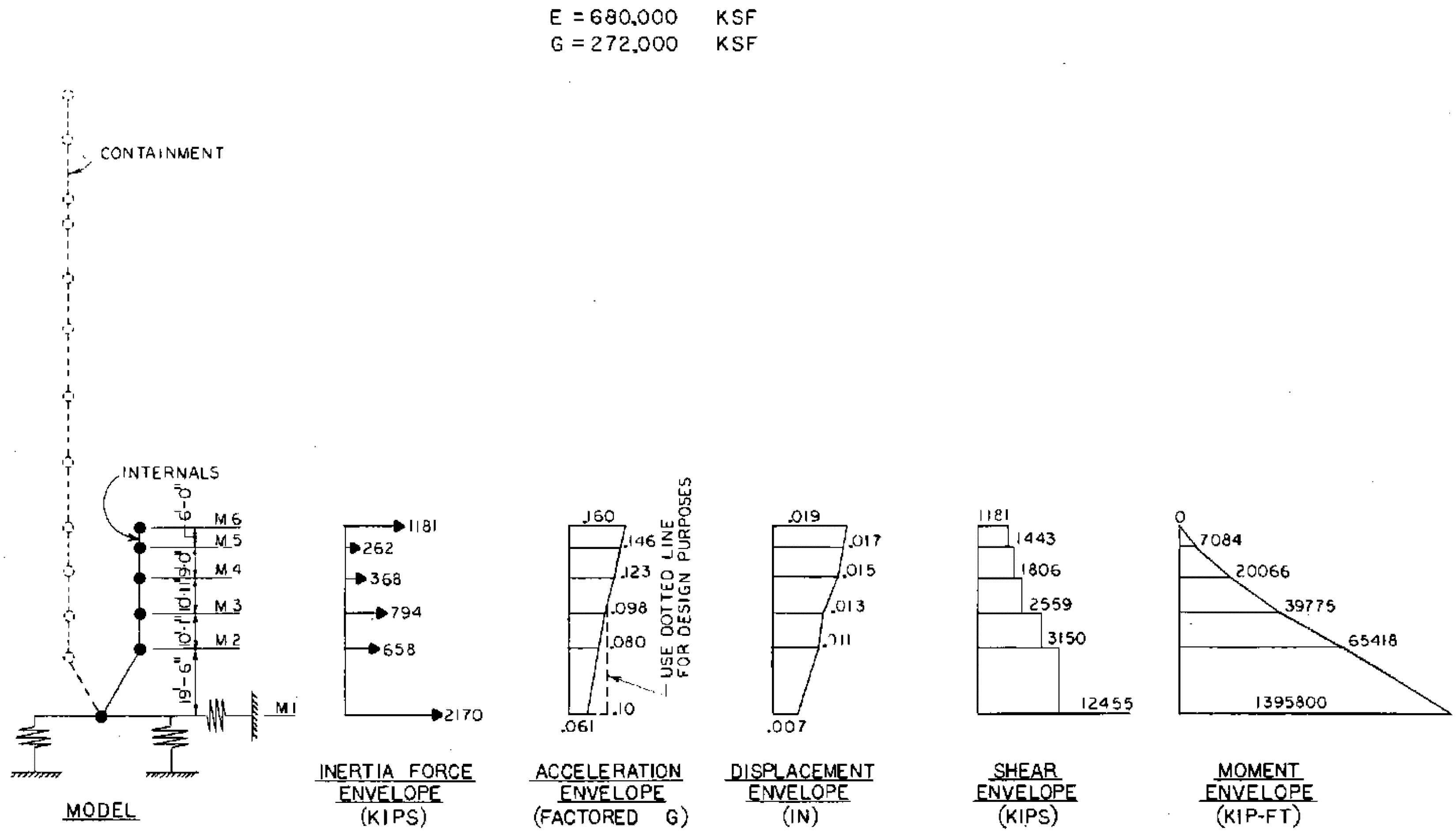
INTERNAL STRUCTURE – SEISMIC RESULTS

FIGURE 3.7-16



NORTH — SOUTH DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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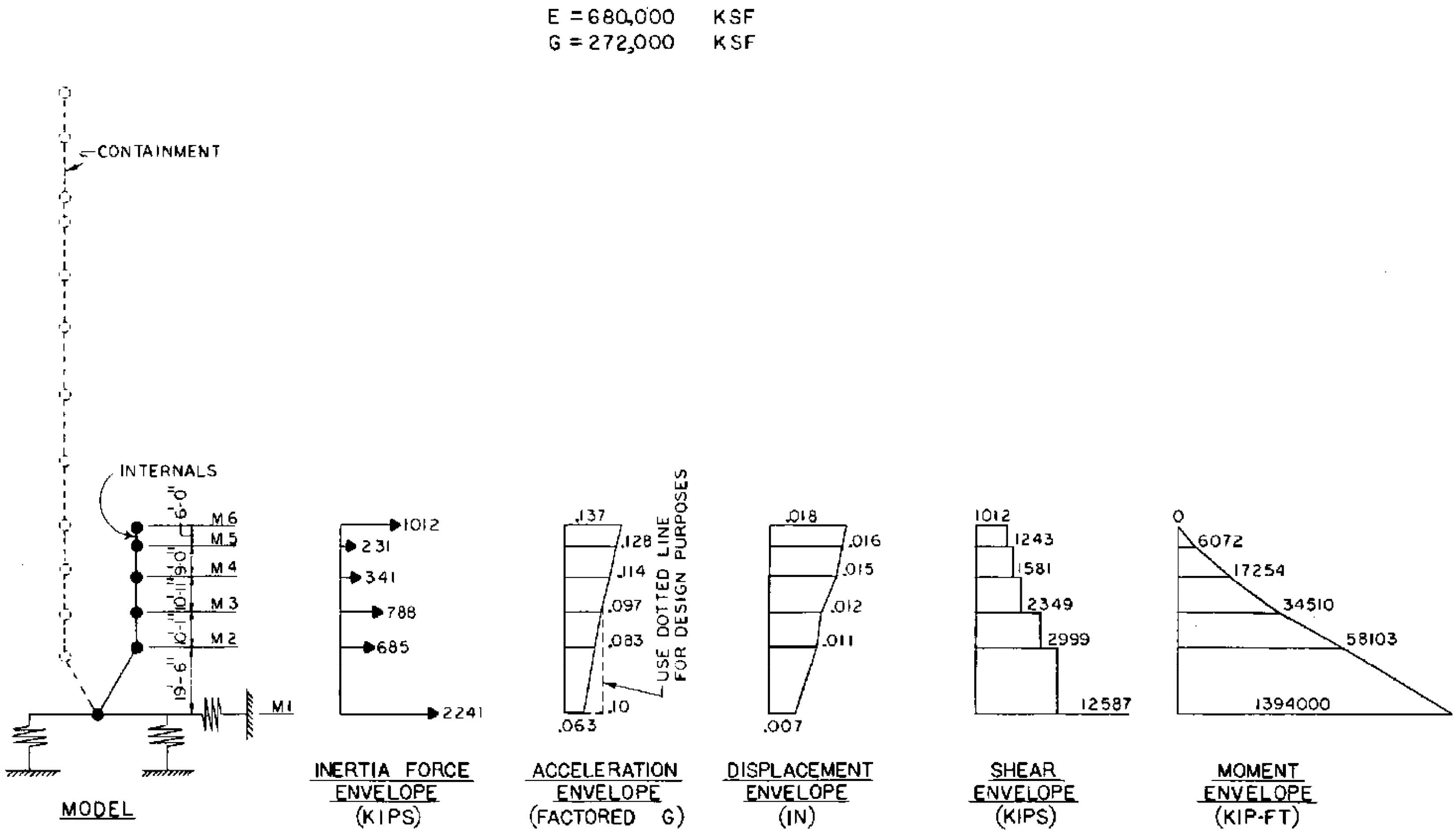
INTERNAL STRUCTURE – SEISMIC RESULTS

FIGURE 3.7-17



EAST – WEST DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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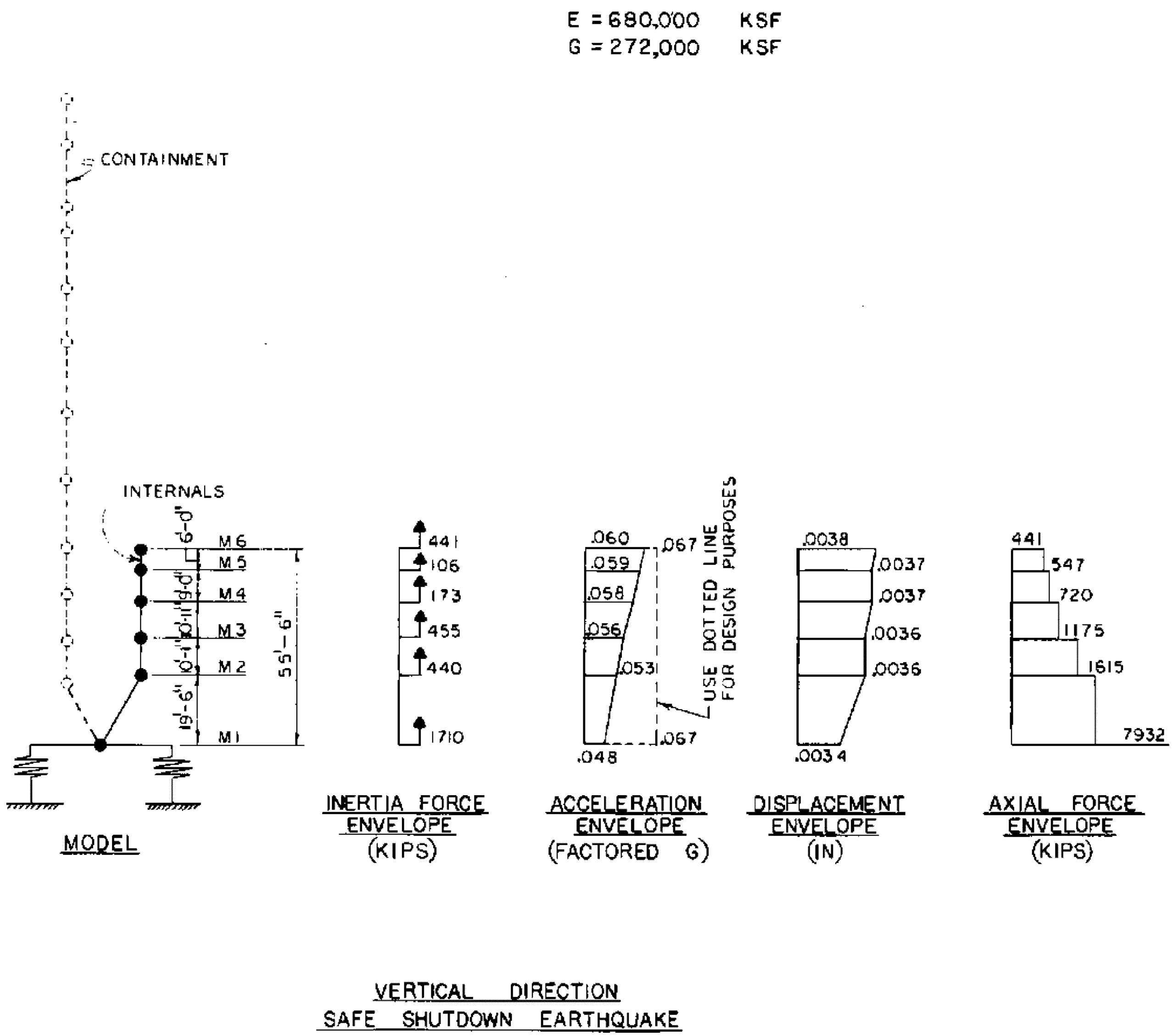
UNIT 1 AND UNIT 2**

INTERNAL STRUCTURE – SEISMIC RESULTS

FIGURE 3.7-18



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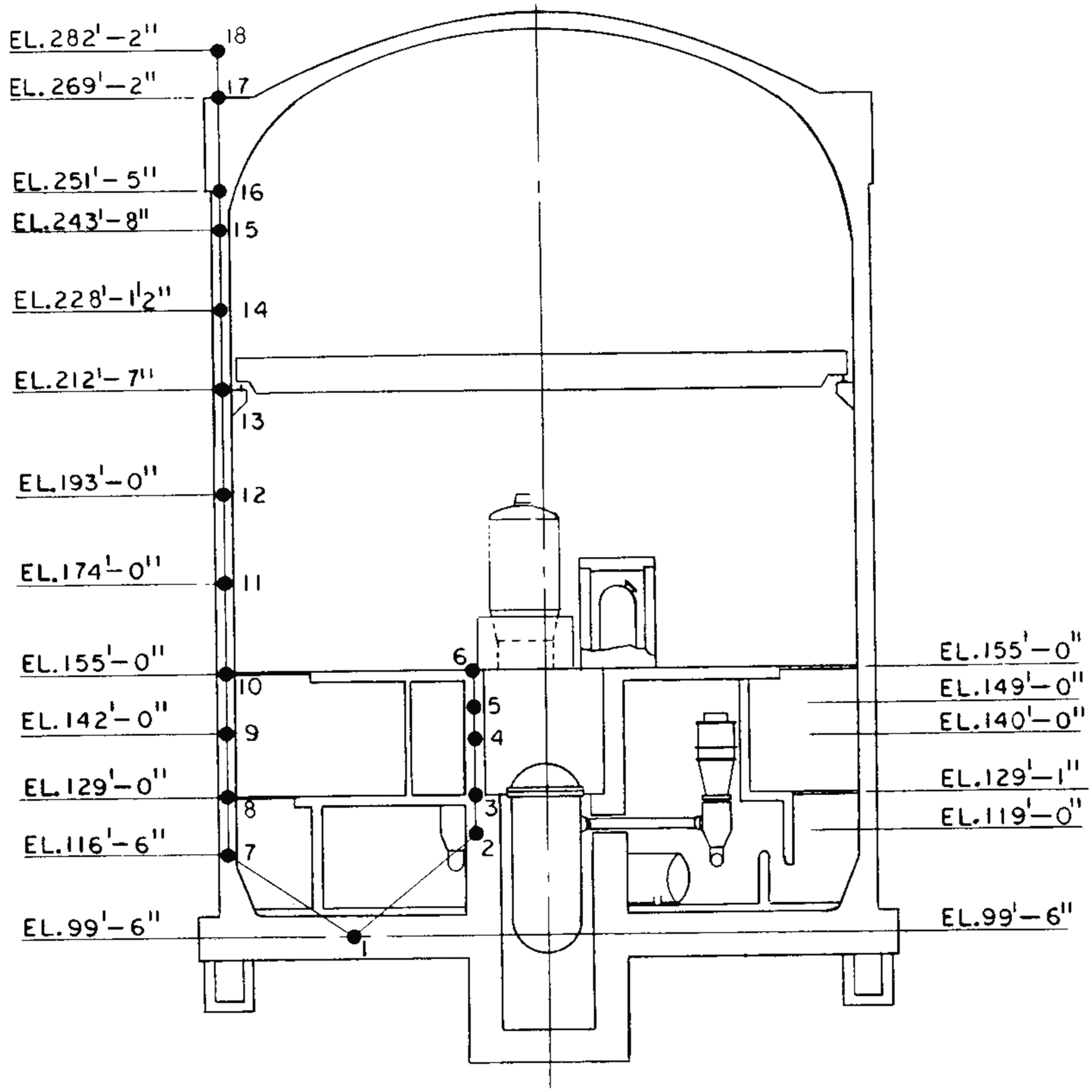
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTERNAL STRUCTURE – SEISMIC RESULTS

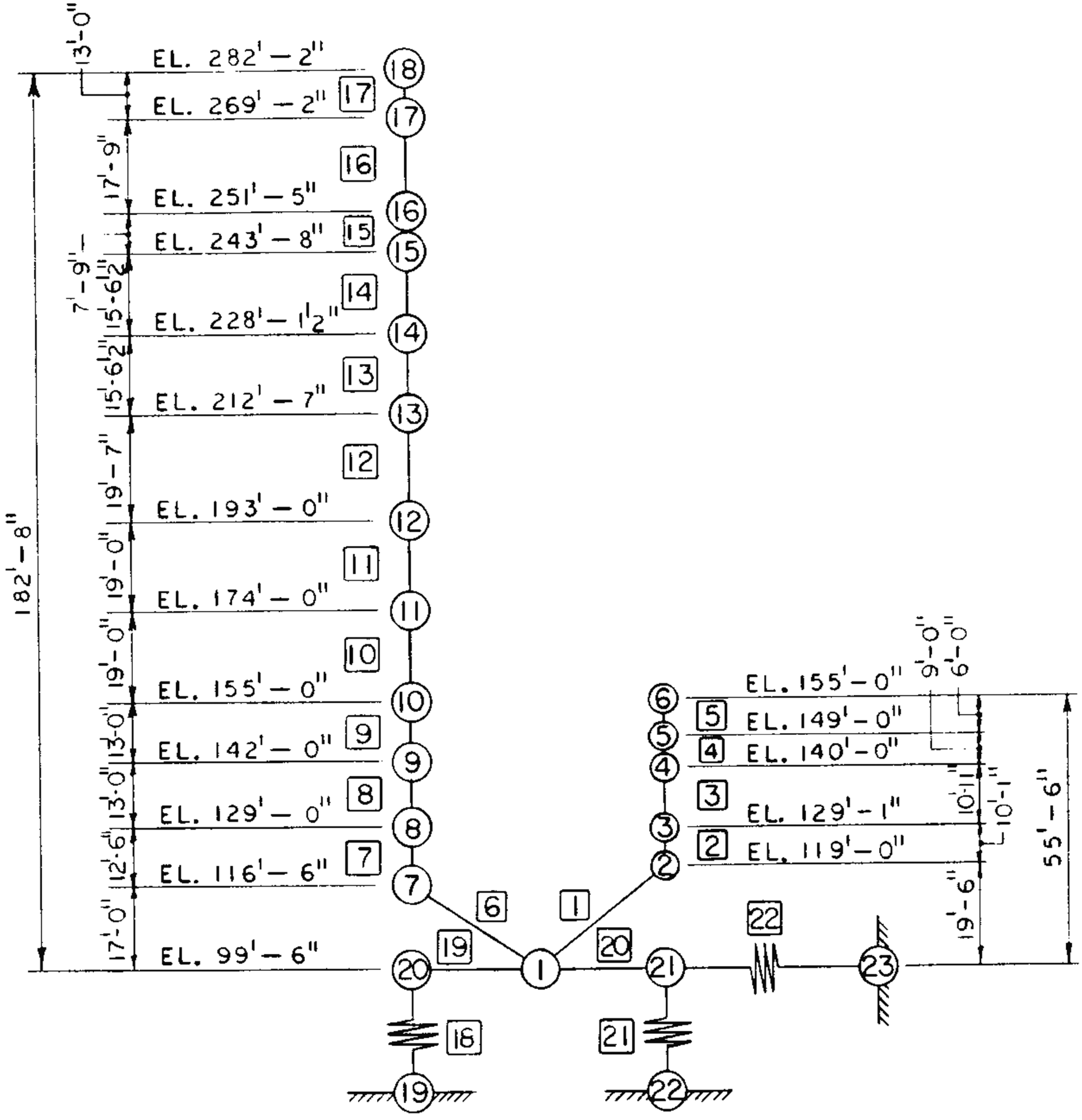
FIGURE 3.7-19



NORTH – SOUTH ℄ EAST – WEST

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	CONTAINMENT AND INTERNAL STRUCTURE MATHEMATICAL MODEL FIGURE 3.7-20



NORTH-SOUTH & EAST-WEST

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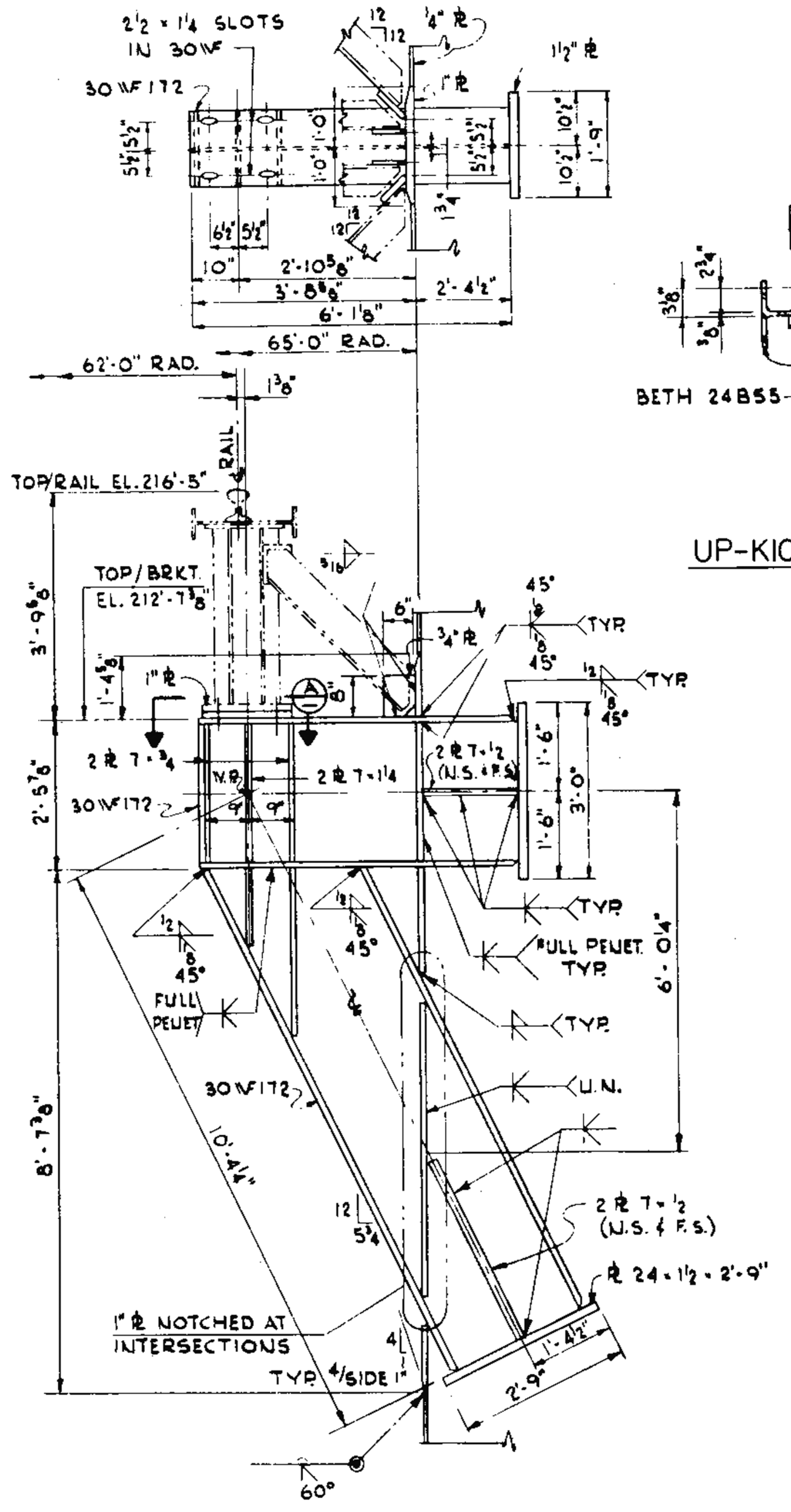
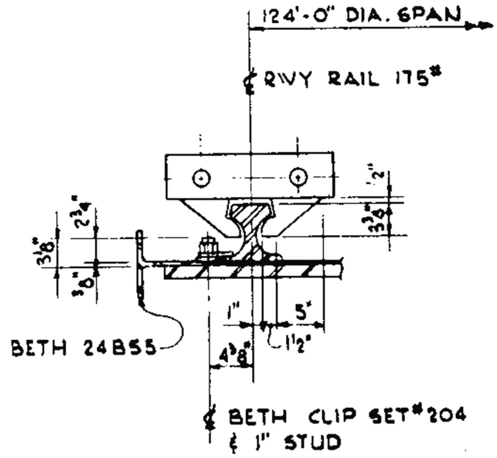
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AND INTERNAL STRUCTURE
MATHEMATICAL MODEL

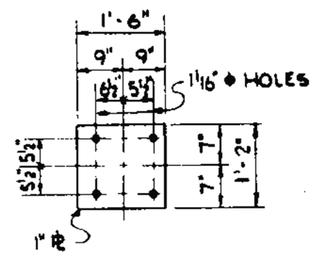
FIGURE 3.7-21



POLAR CRANE BRACKET



UP-KICK LUG DETAIL



SECTION A

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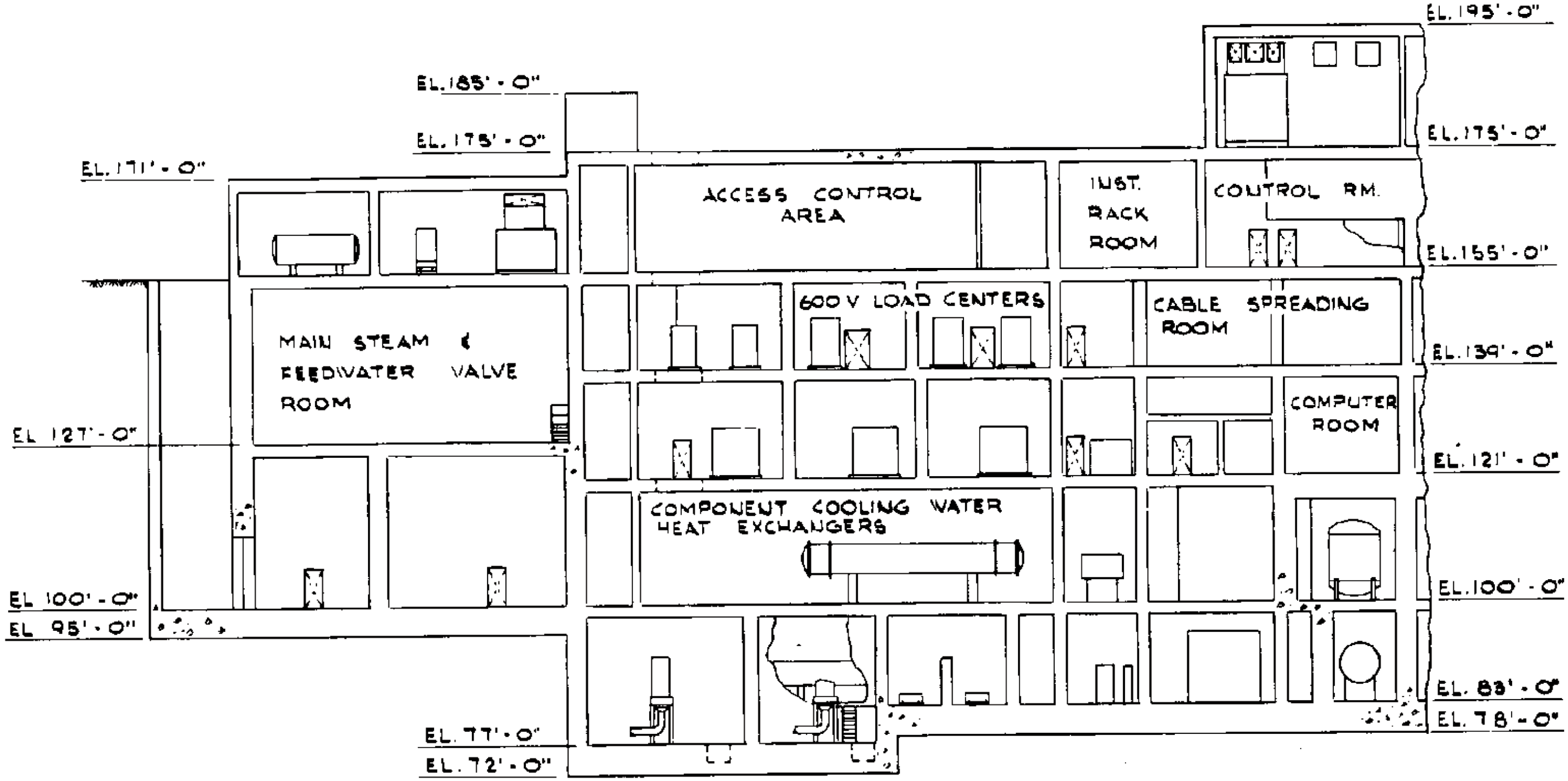
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POLAR CRANE BRACKET AND SEISMIC RETAINER

FIGURE 3.7-22



SECTION — LOOKING WEST

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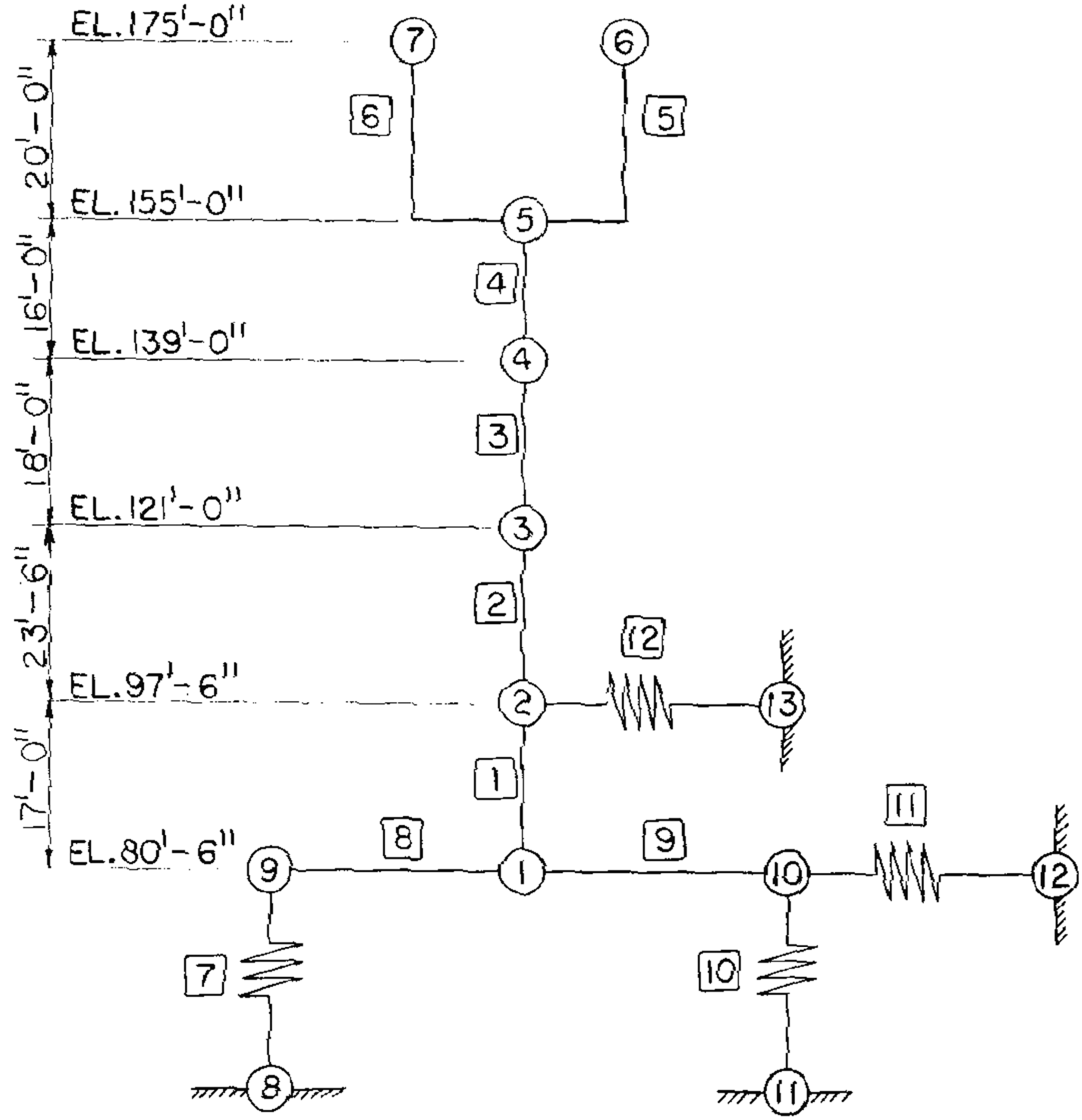
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING MATHEMATICAL MODEL

FIGURE 3.7-23



NORTH-SOUTH, EAST-WEST AND VERTICAL DIRECTION

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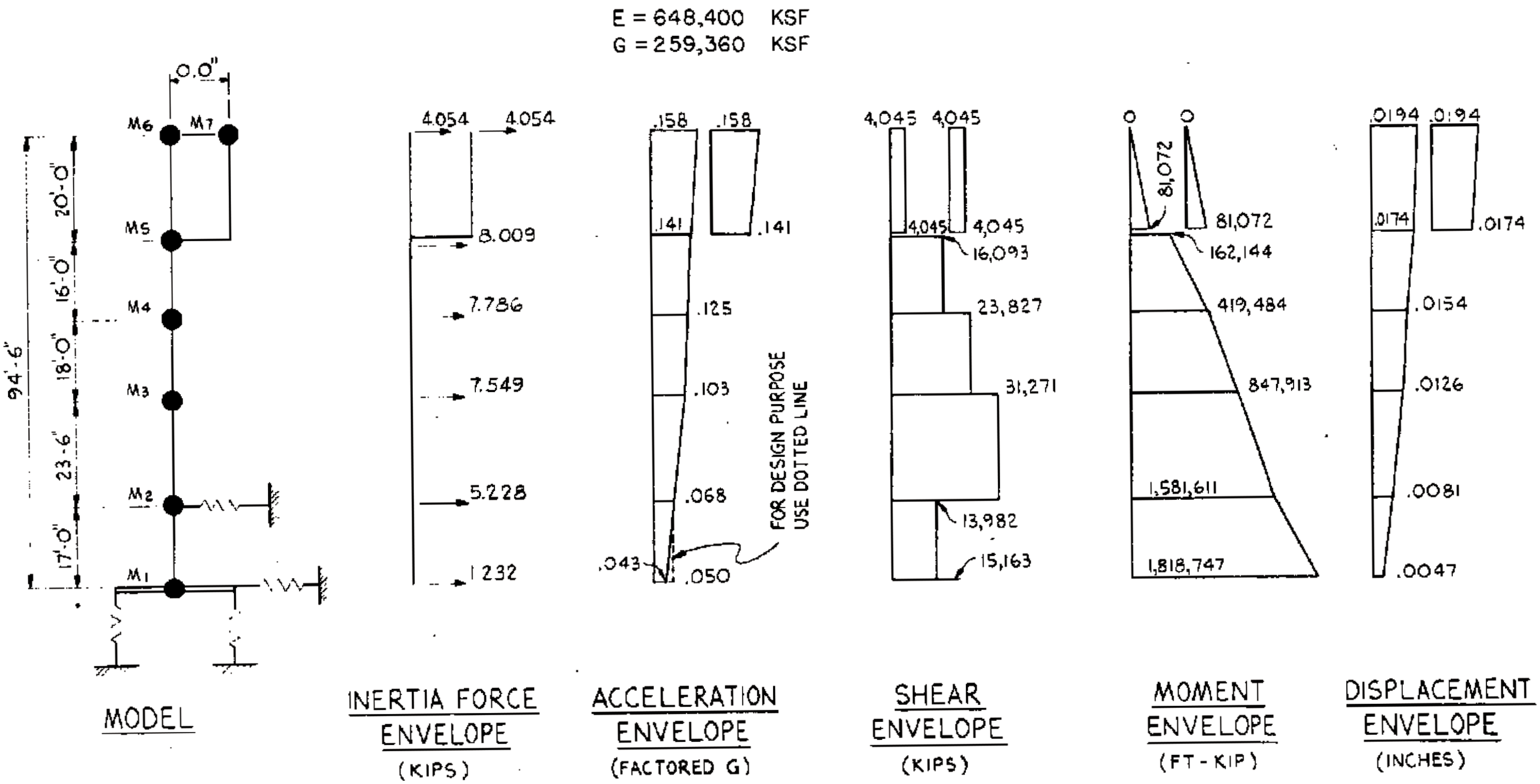
AUXILIARY BUILDING MATHEMATICAL MODEL

FIGURE 3.7-24



NORTH - SOUTH DIRECTION

1/2 SAFE SHUTDOWN EARTHQUAKE 0.05 G

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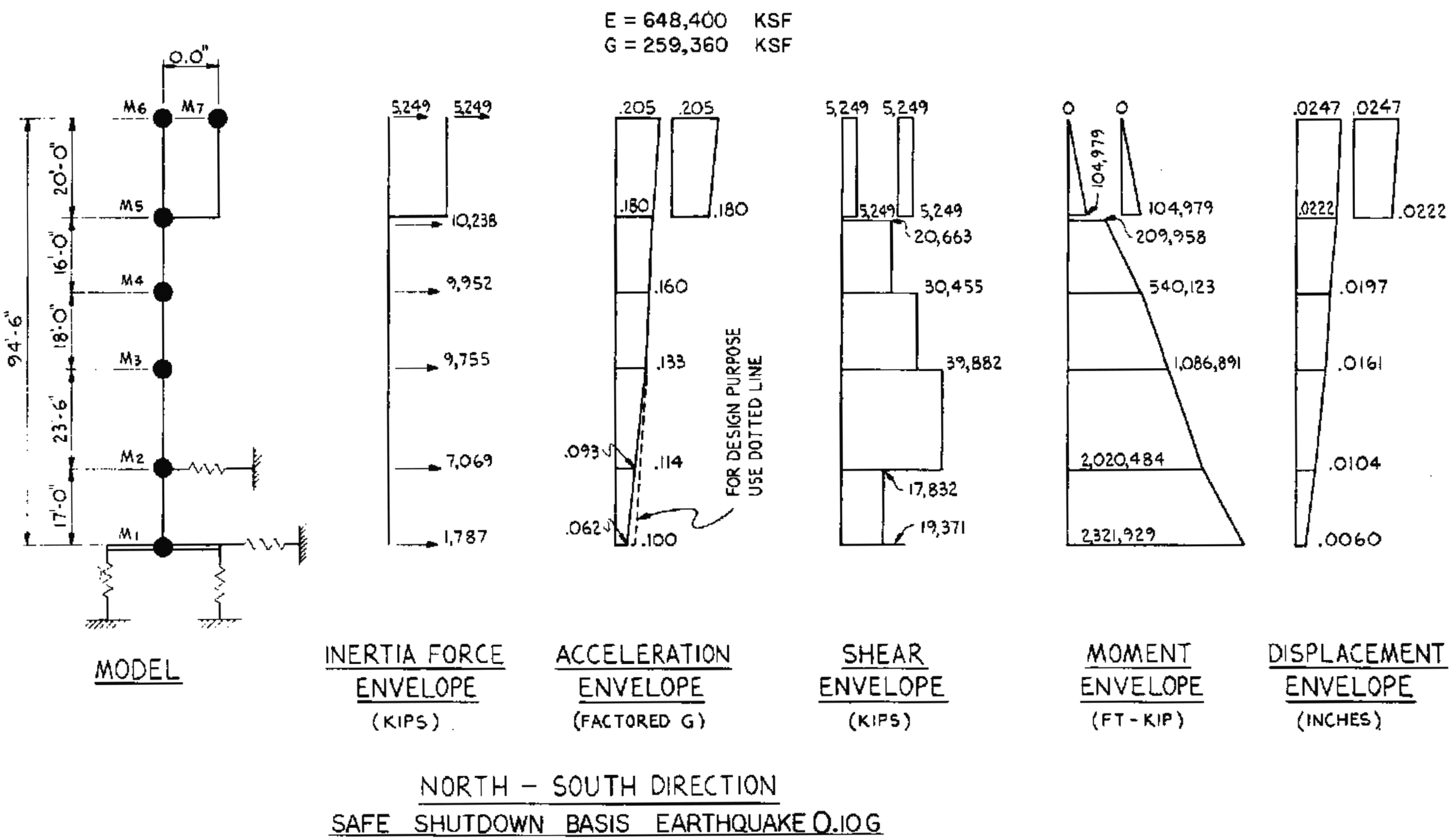
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING SEISMIC RESULTS

FIGURE 3.7-25



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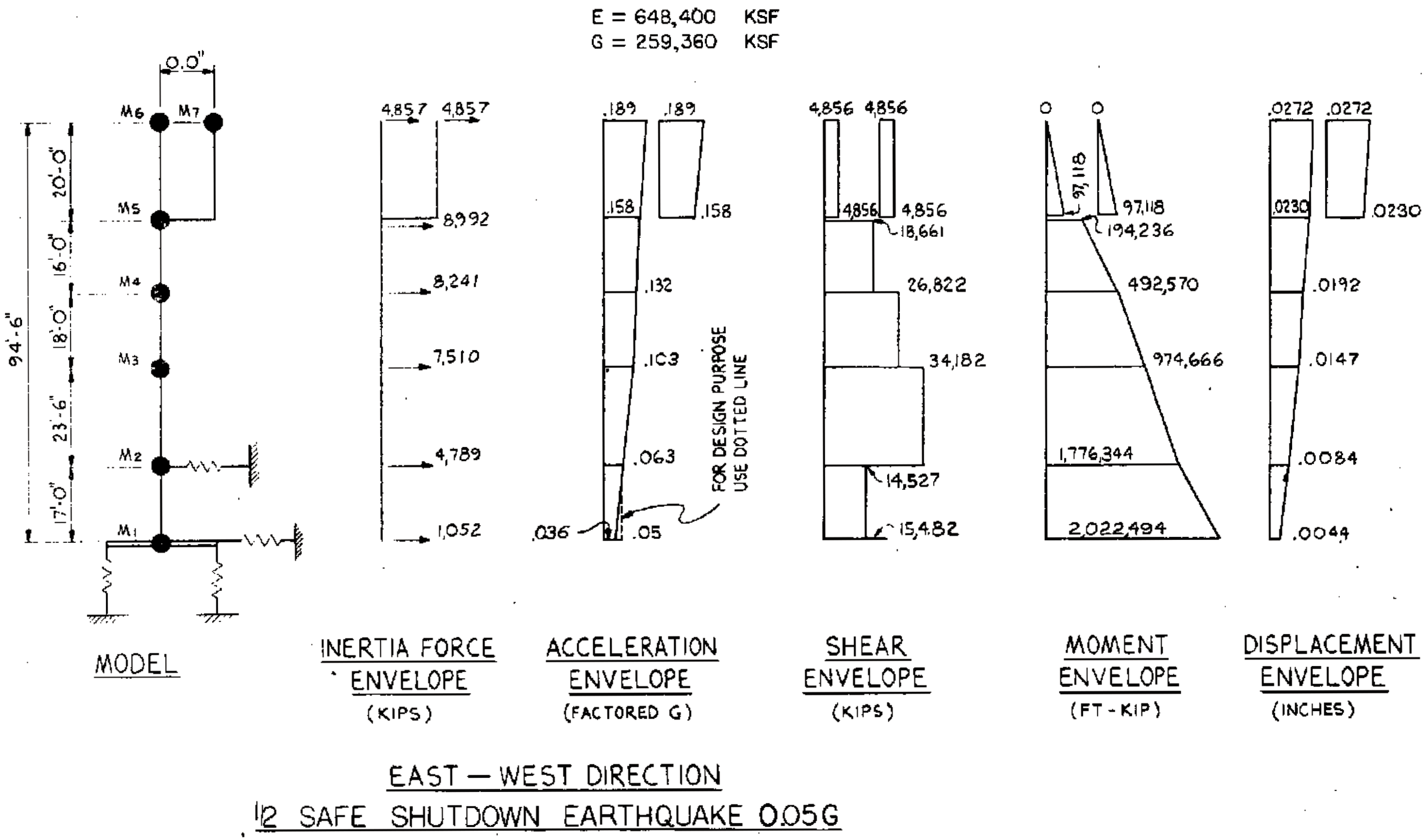
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING SEISMIC RESULTS

FIGURE 3.7-26



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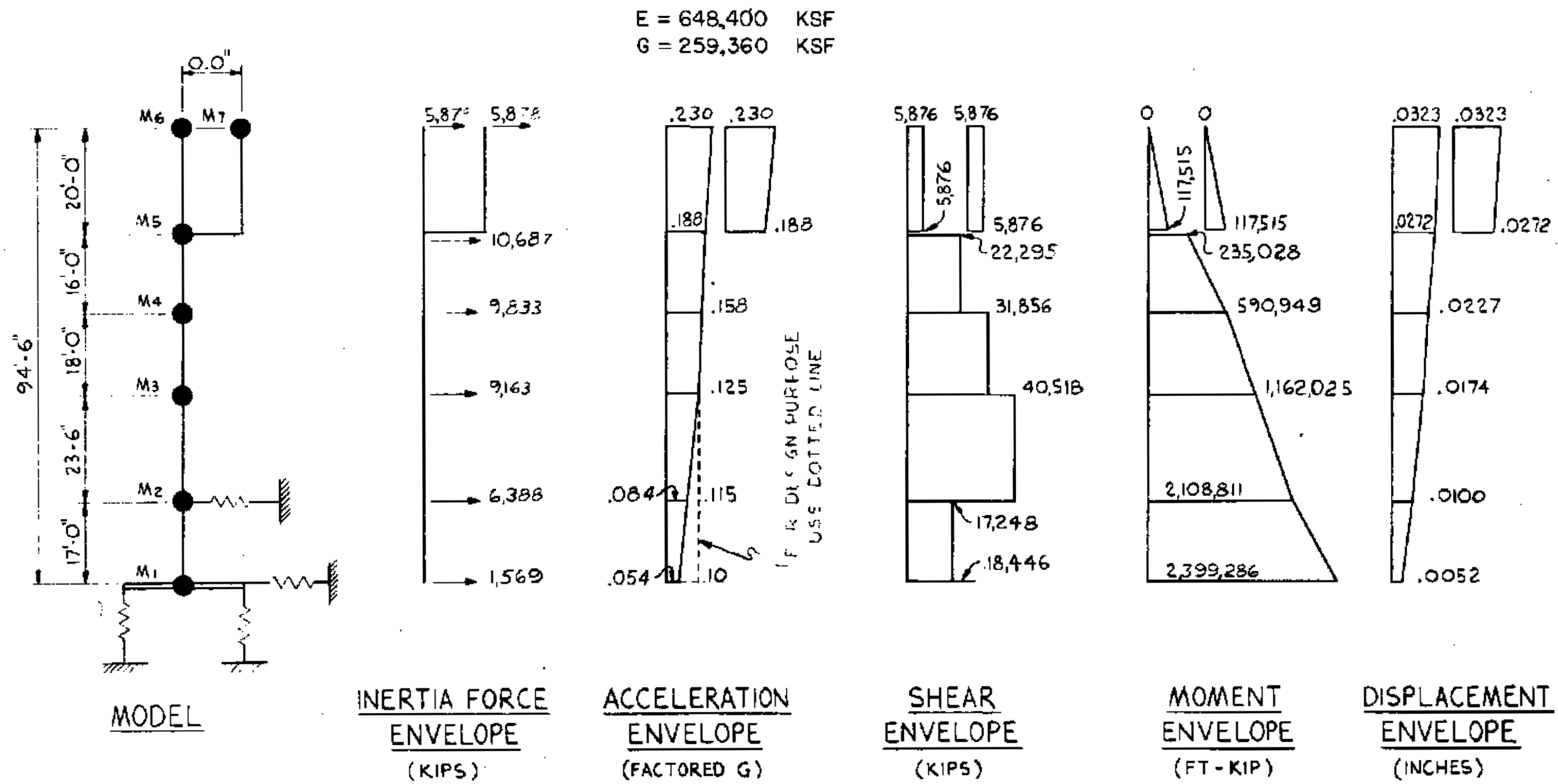
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING SEISMIC RESULTS

FIGURE 3.7-27



EAST — WEST DIRECTION

SAFE SHUTDOWN BASIS EARTHQUAKE 0.10G

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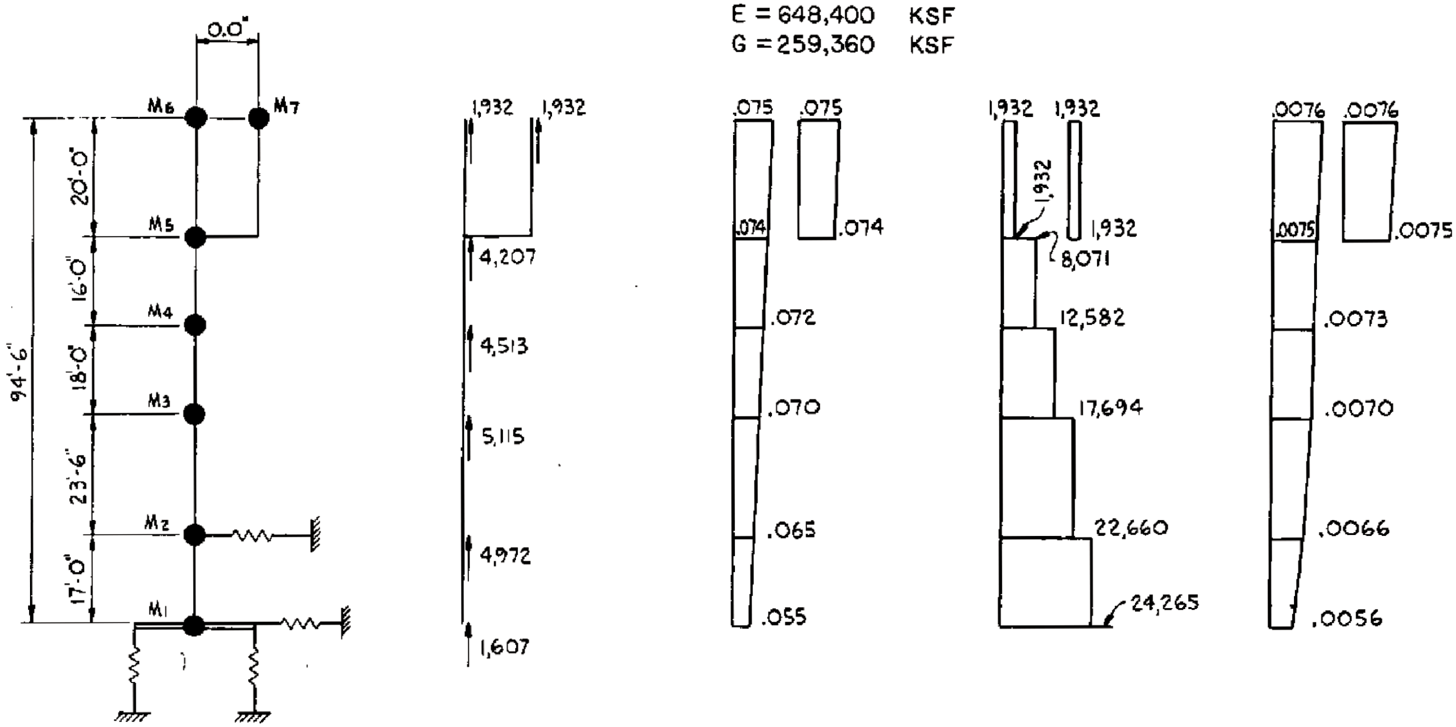
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING SEISMIC RESULTS

FIGURE 3.7-28

E = 648,400 KSF
G = 259,360 KSF



MODEL

INERTIA FORCE ENVELOPE (KIPS)

ACCELERATION ENVELOPE (FACTORED G)

AXIAL FORCE ENVELOPE (KIPS)

DISPLACEMENT ENVELOPE (INCHES)

VERTICAL DIRECTION

1/2 SAFE SHUTDOWN EARTHQUAKE 0.033 G

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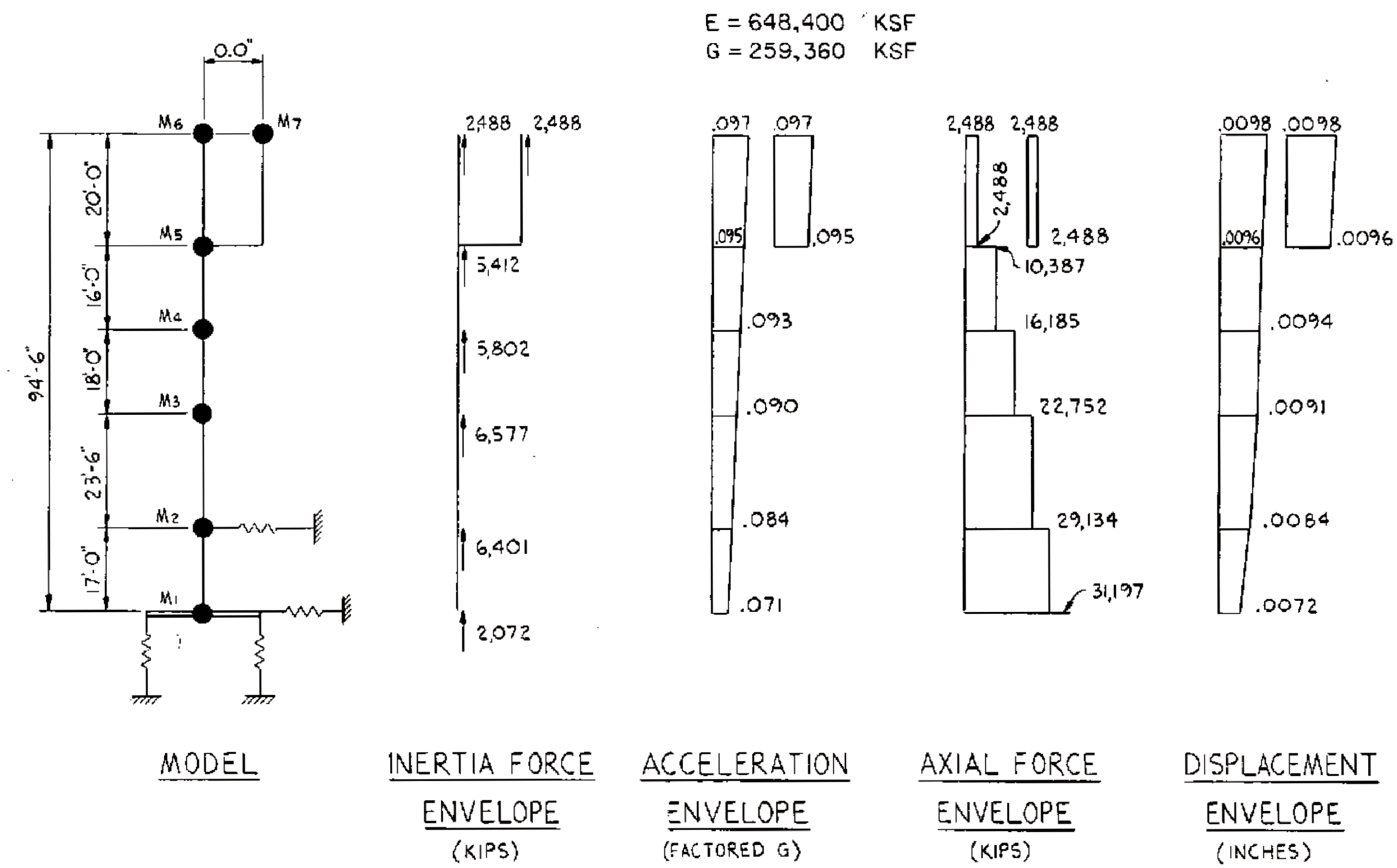
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

AUXILIARY BUILDING SEISMIC RESULTS

FIGURE 3.7-29



VERTICAL DIRECTION

SAFE SHUTDOWN BASIS EARTHQUAKE 0.067 G

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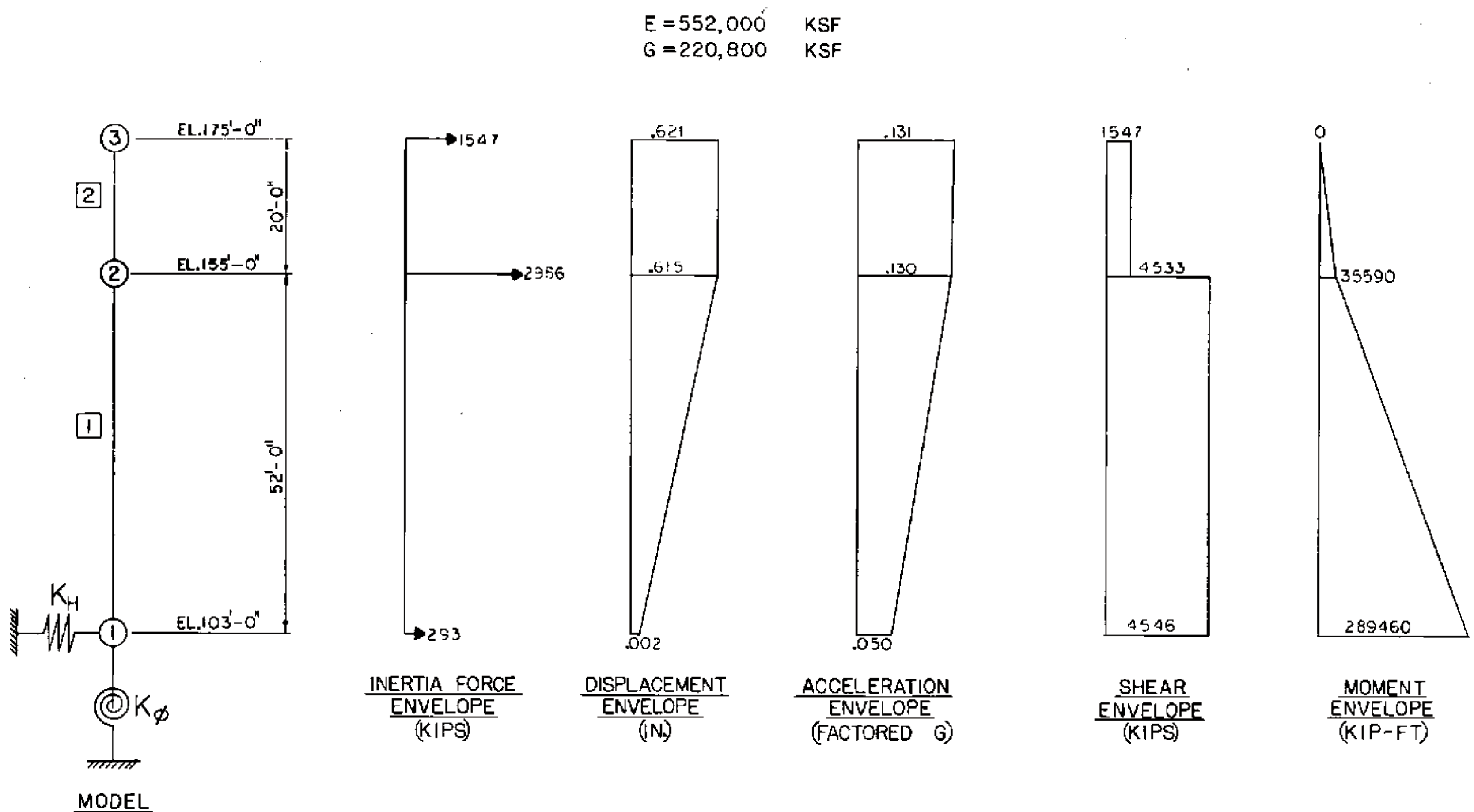
**JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2**

AUXILIARY BUILDING SEISMIC RESULTS

FIGURE 3.7-30



NORTH - SOUTH DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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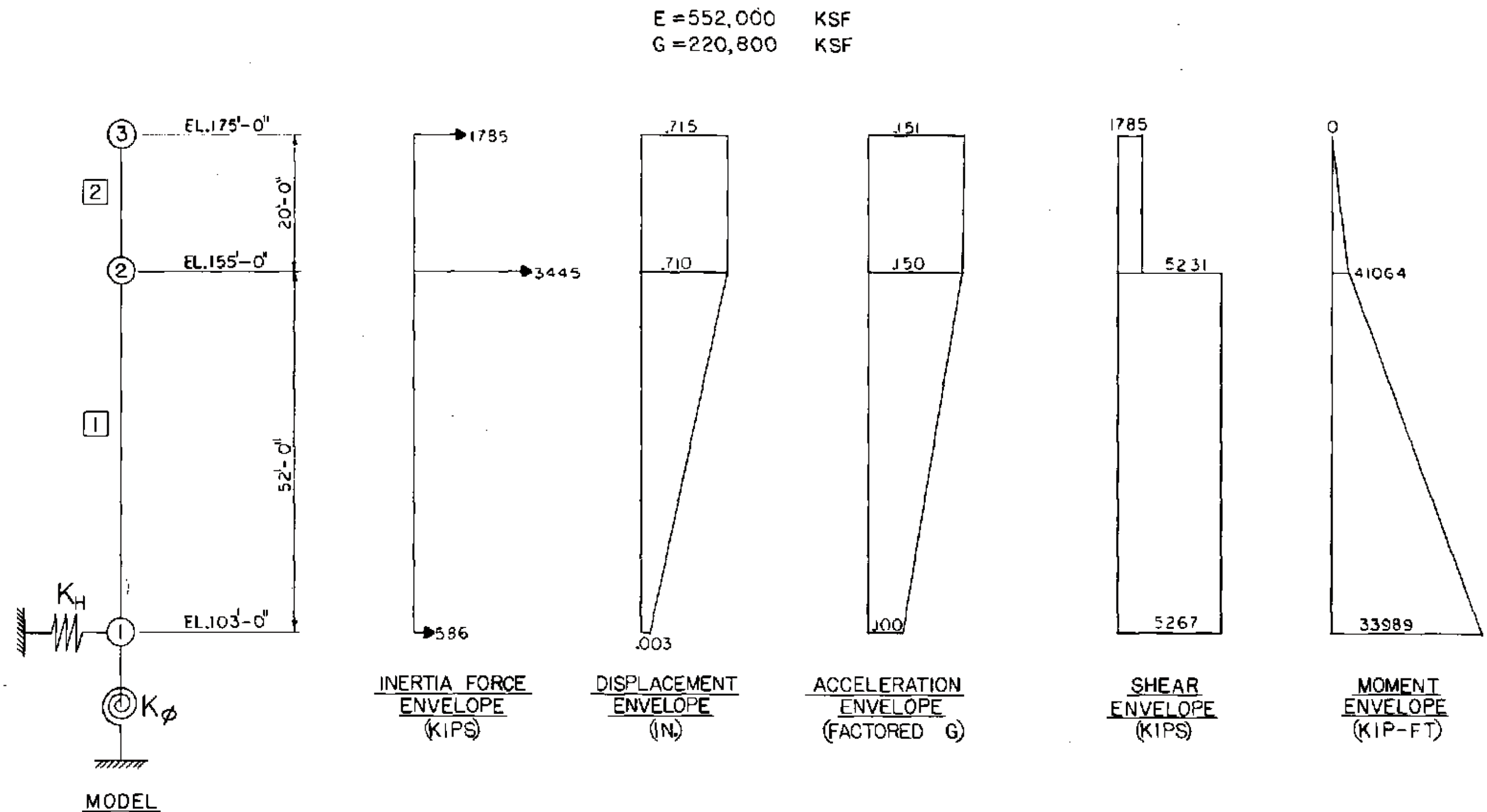
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DIESEL GENERATOR BUILDING SEISMIC RESULTS

FIGURE 3.7-31



NORTH-SOUTH DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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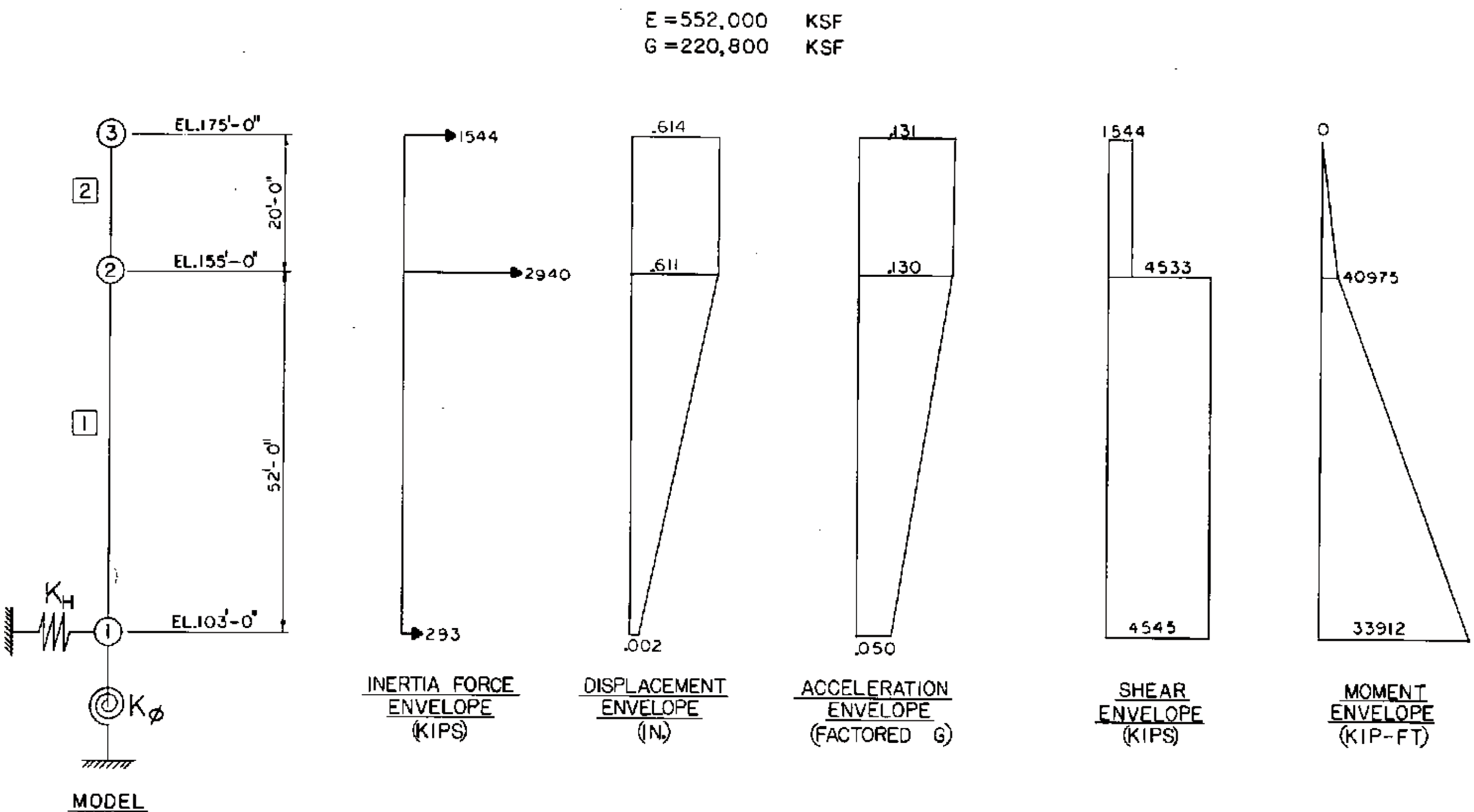
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DIESEL GENERATOR BUILDING SEISMIC RESULTS

FIGURE 3.7-32

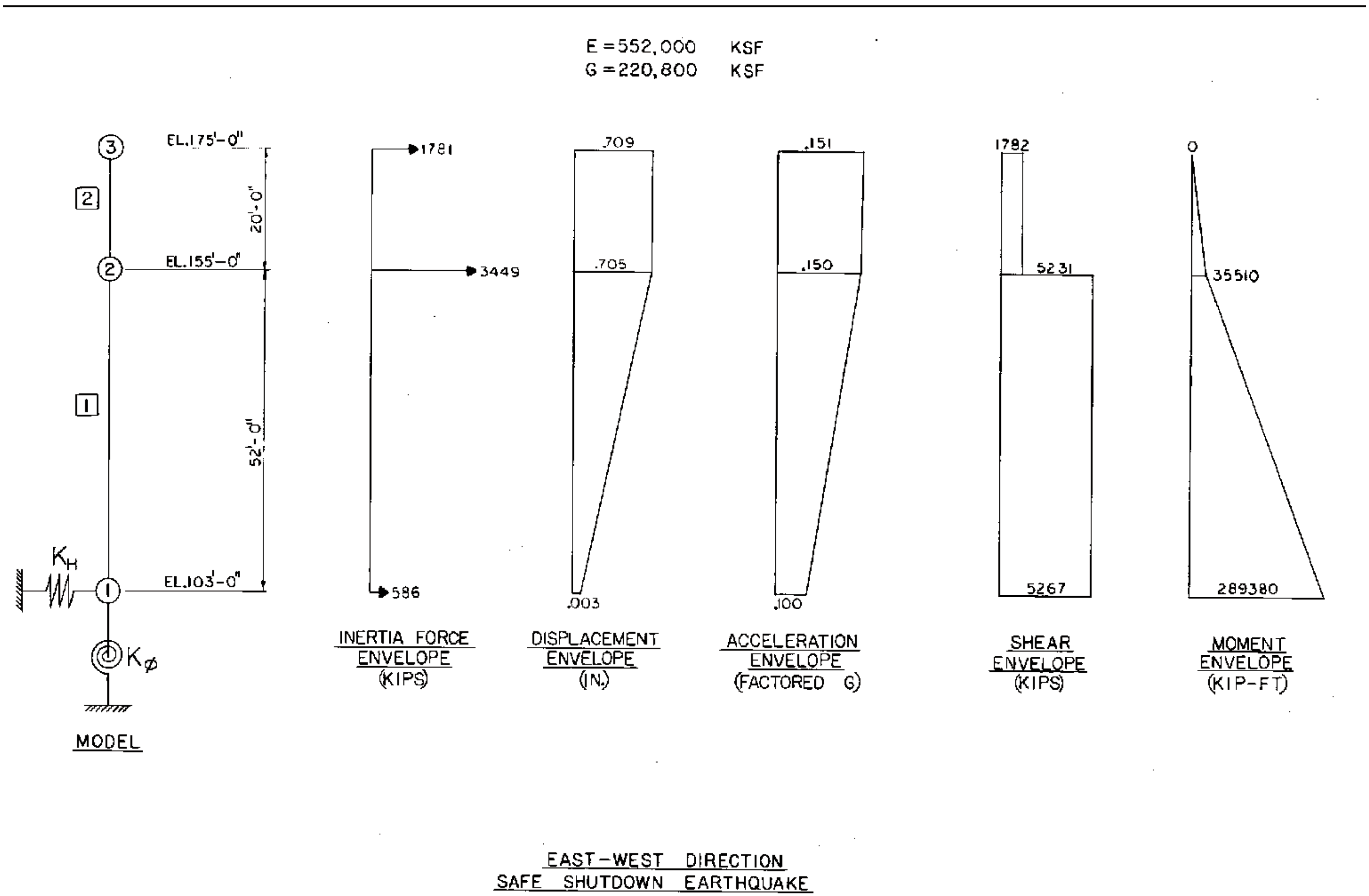


EAST-WEST DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

REV 21 5/08

DIESEL GENERATOR BUILDING SEISMIC RESULTS

FIGURE 3.7-33



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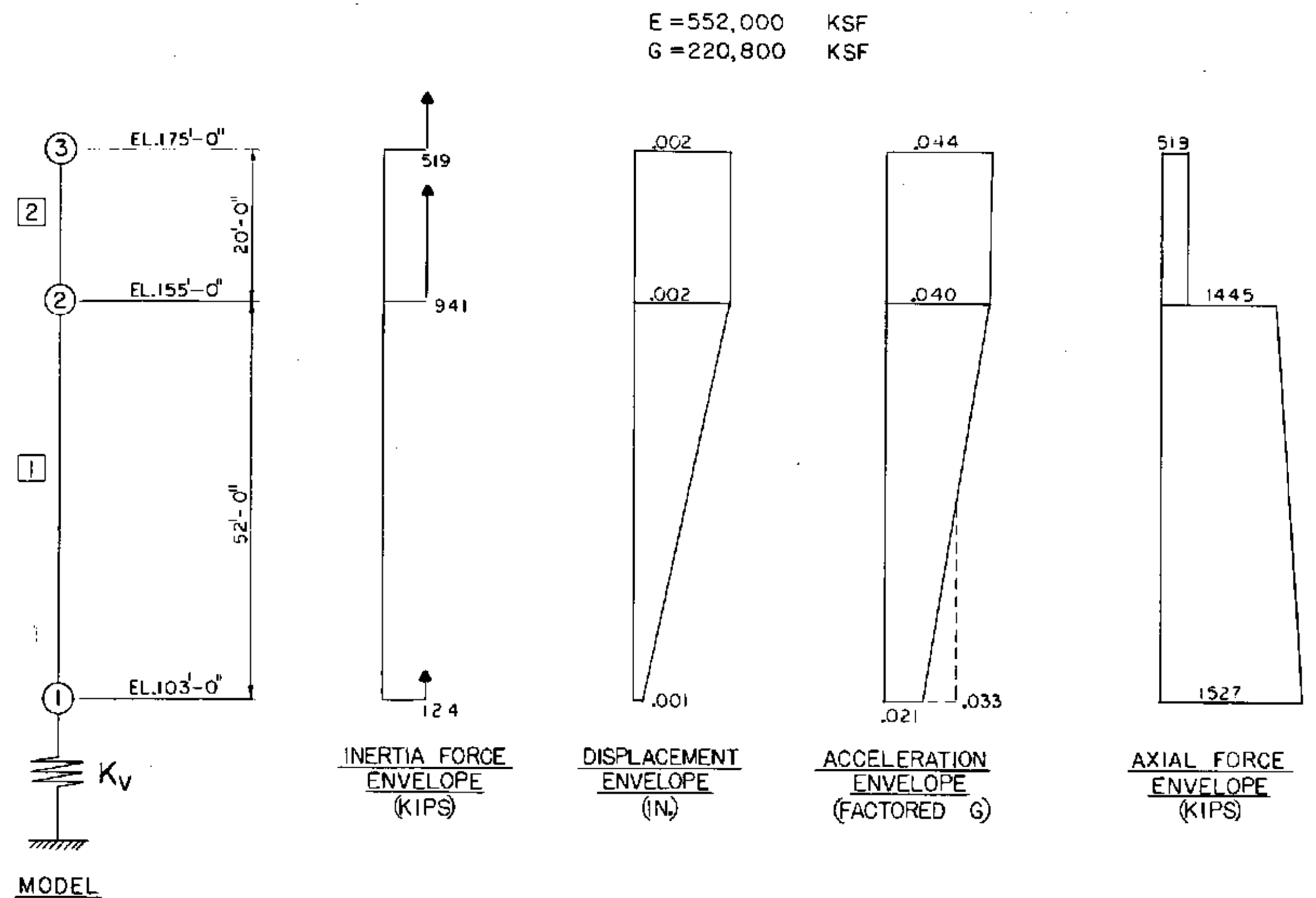
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DIESEL GENERATOR BUILDING SEISMIC RESULTS

FIGURE 3.7-34



VERTICAL DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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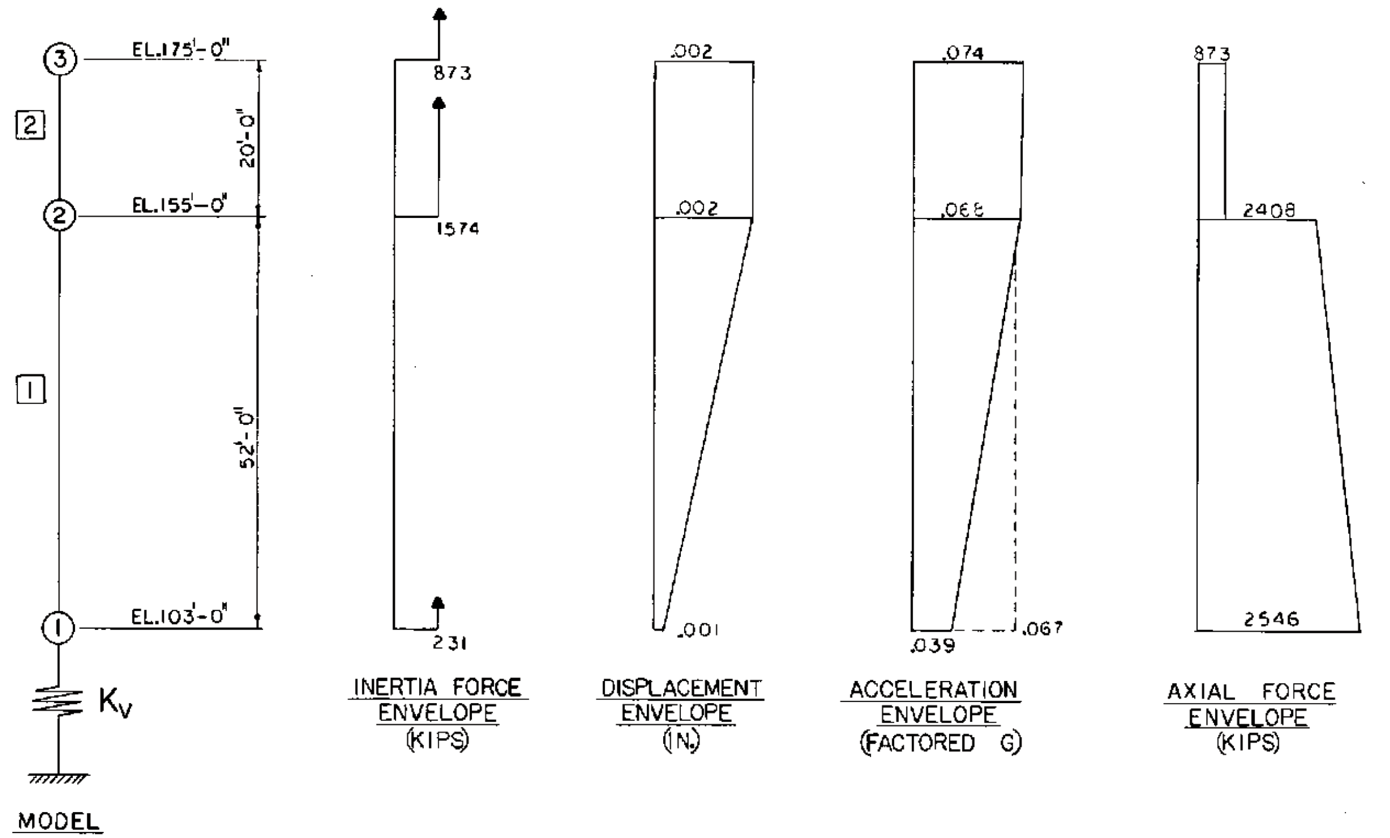
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

DIESEL GENERATOR BUILDING SEISMIC RESULTS

FIGURE 3.7-35

E =552,000 KSF
G =220,800 KSF



VERTICAL DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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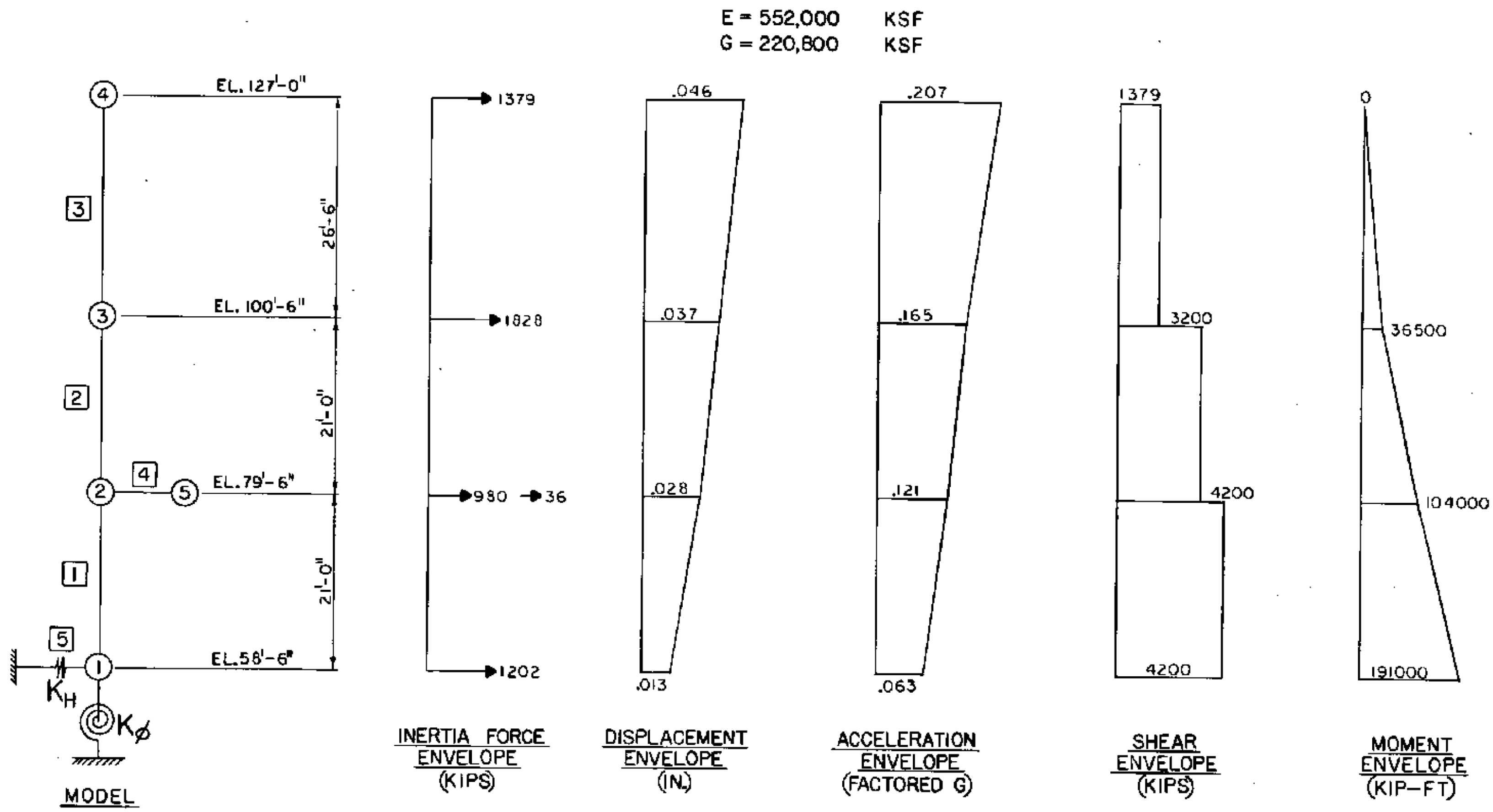
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

DIESEL GENERATOR BUILDING SEISMIC RESULTS

FIGURE 3.7-36



NORTH-SOUTH DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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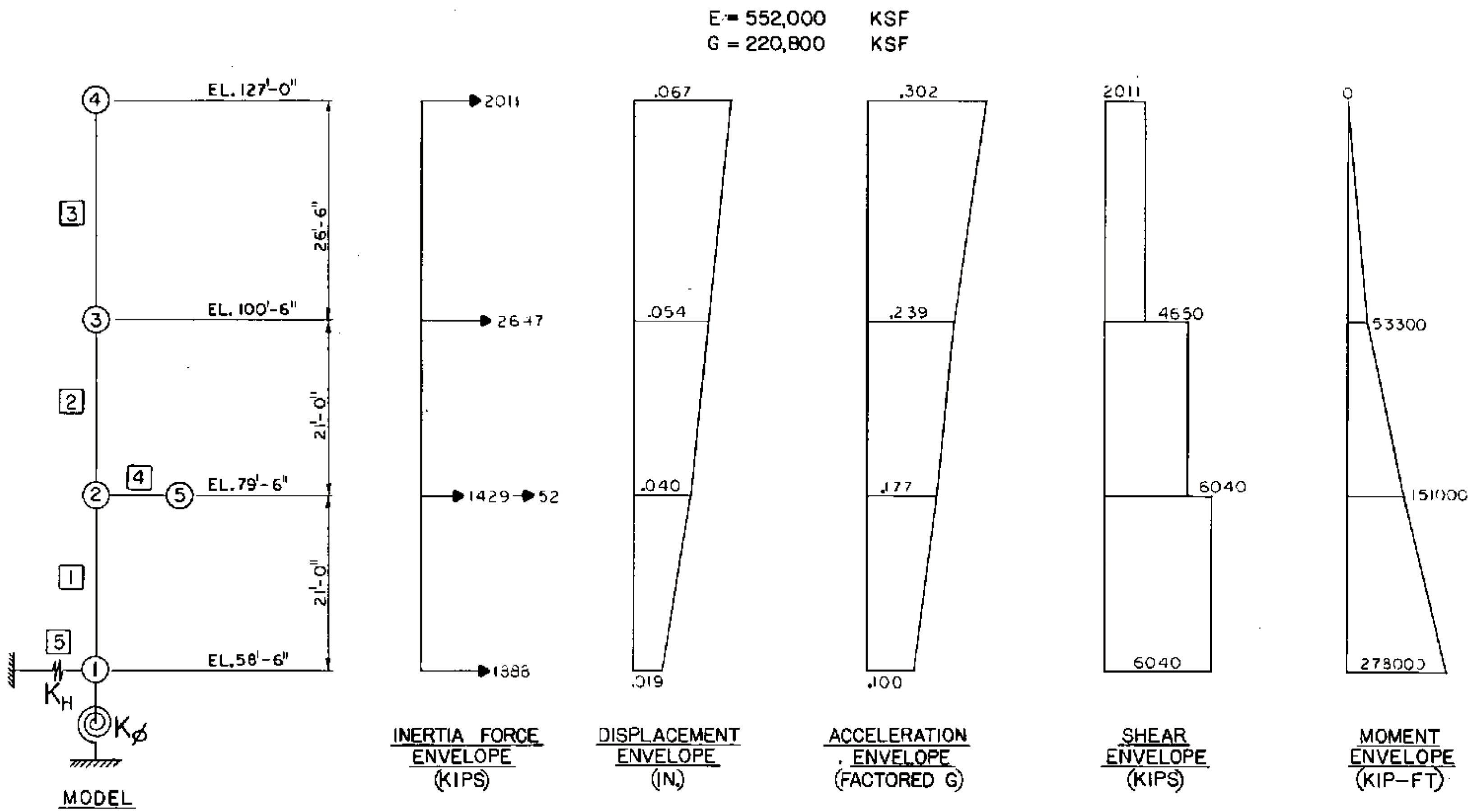
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RIVER INTAKE STRUCTURE SEISMIC RESULTS

FIGURE 3.7-37



NORTH-SOUTH DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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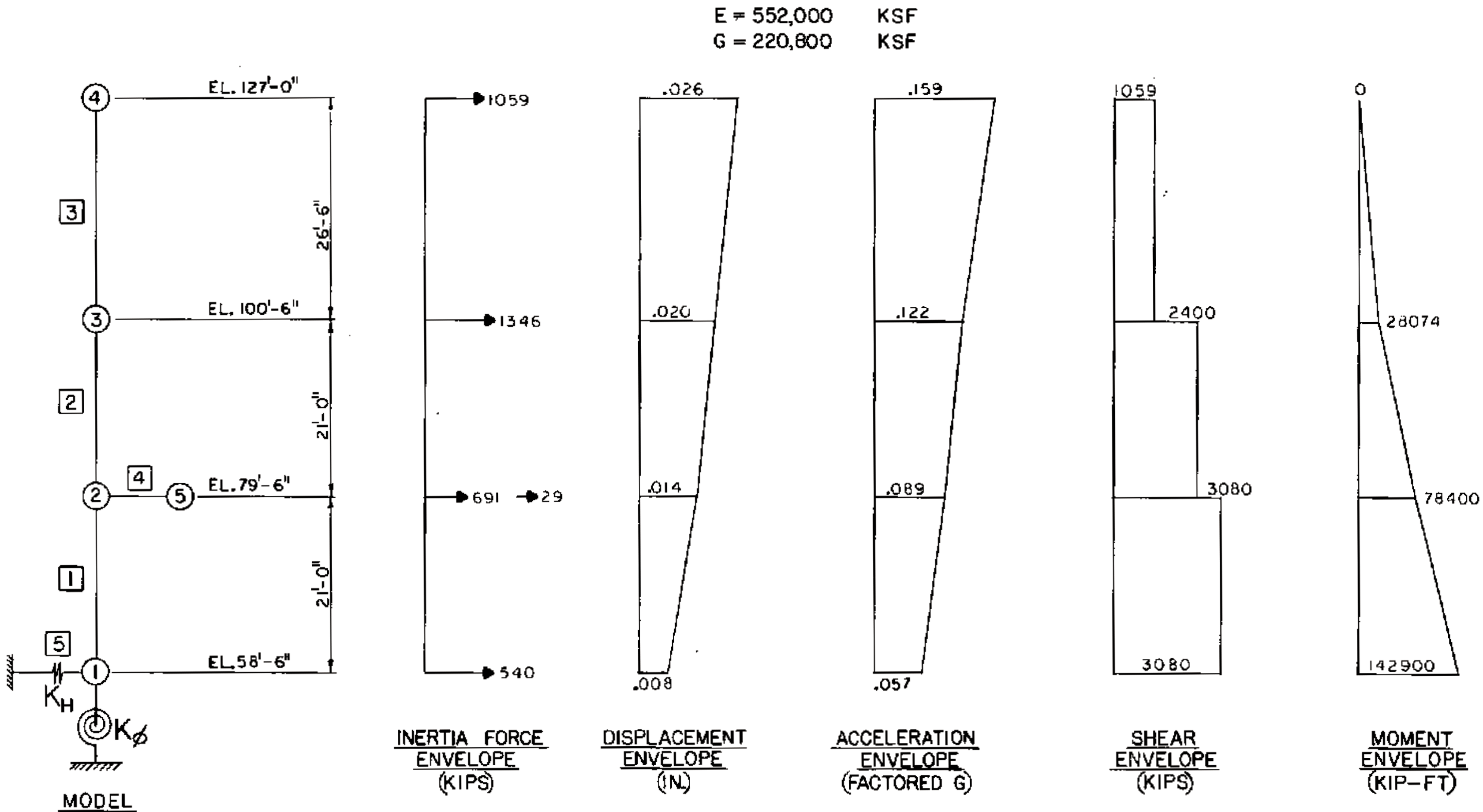
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

RIVER INTAKE STRUCTURE SEISMIC RESULTS

FIGURE 3.7-38



EAST-WEST DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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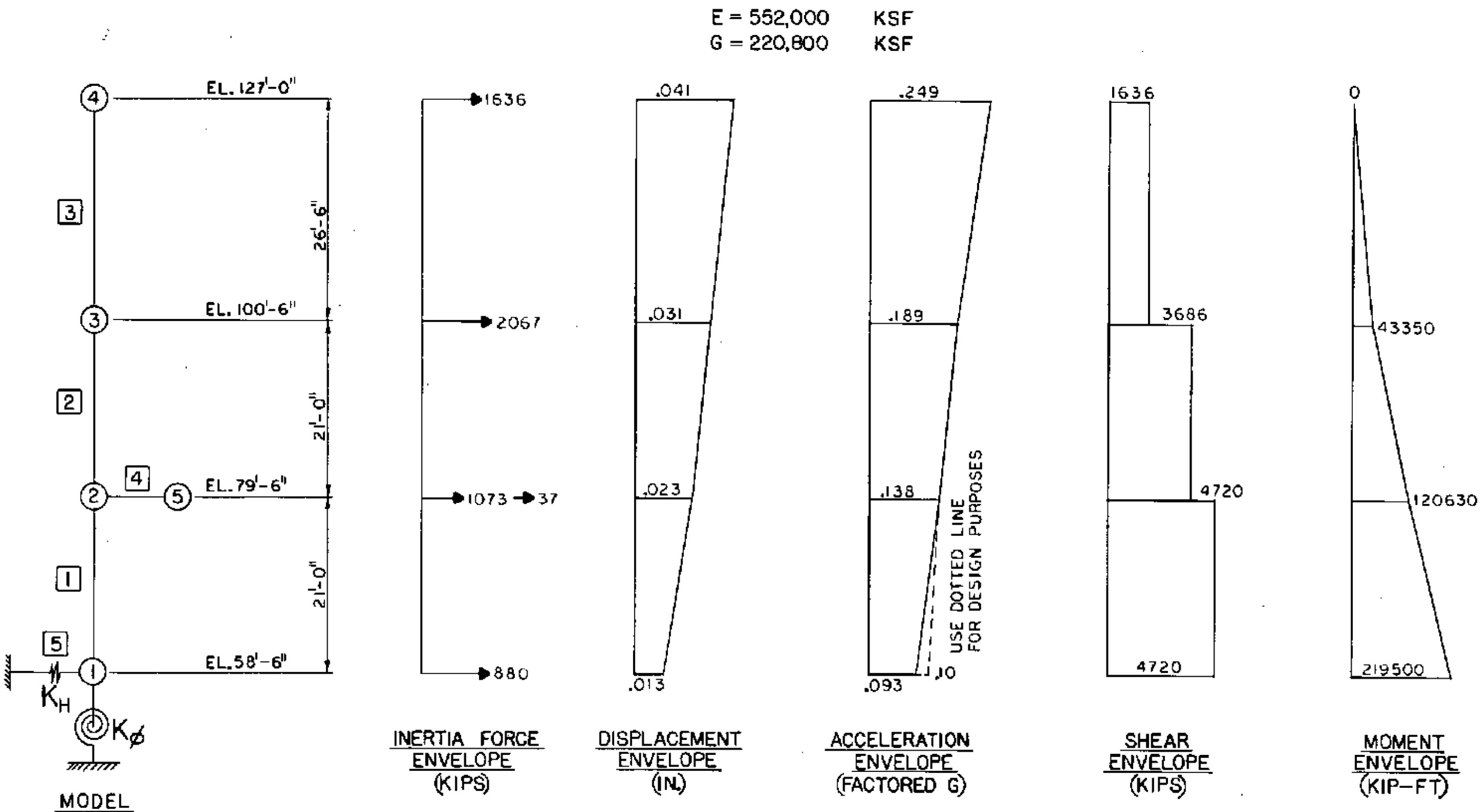
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RIVER INTAKE STRUCTURE SEISMIC RESULTS

FIGURE 3.7-39



EAST—WEST DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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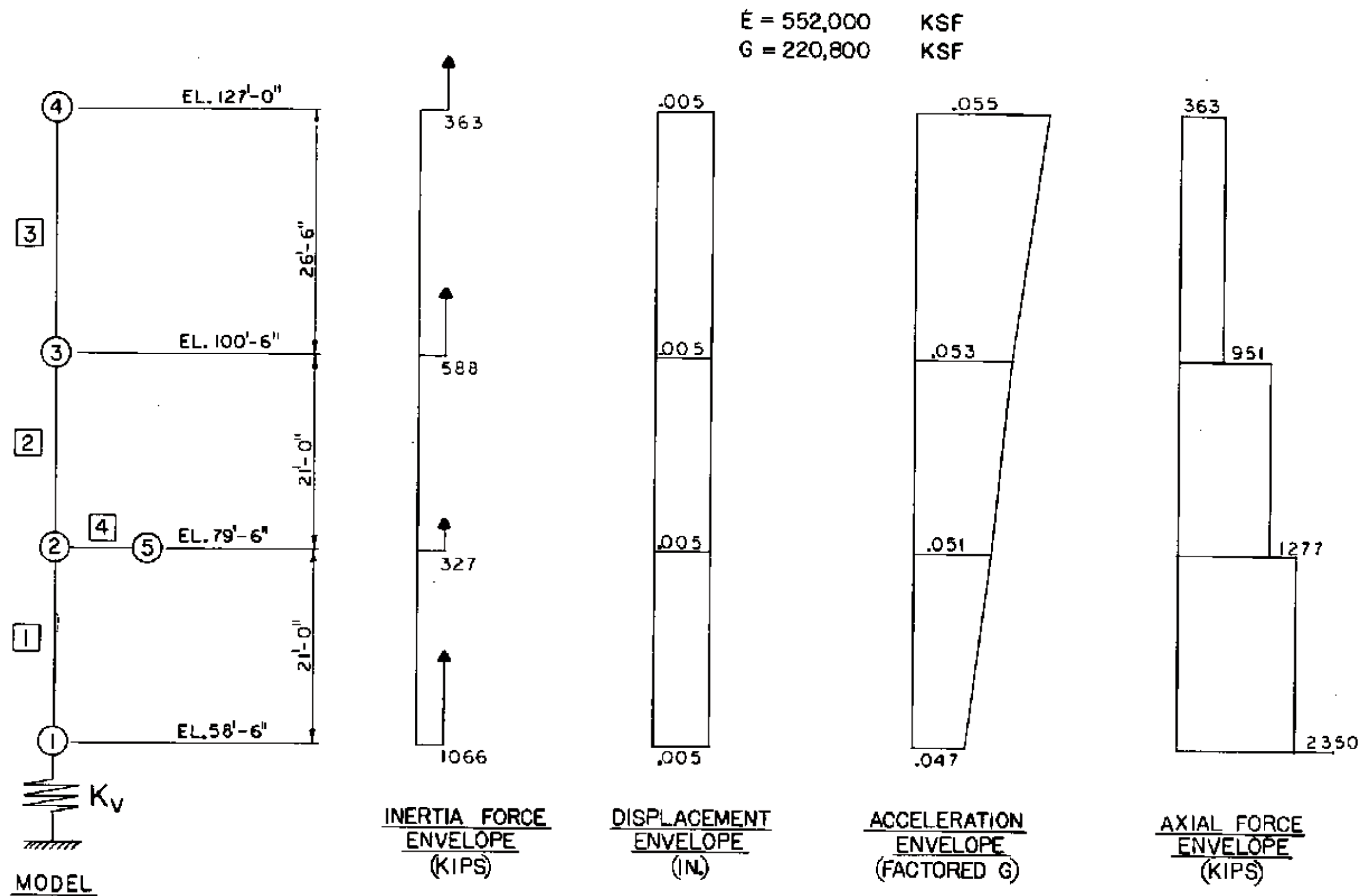
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RIVER INTAKE STRUCTURE SEISMIC RESULTS

FIGURE 3.7-40



VERTICAL DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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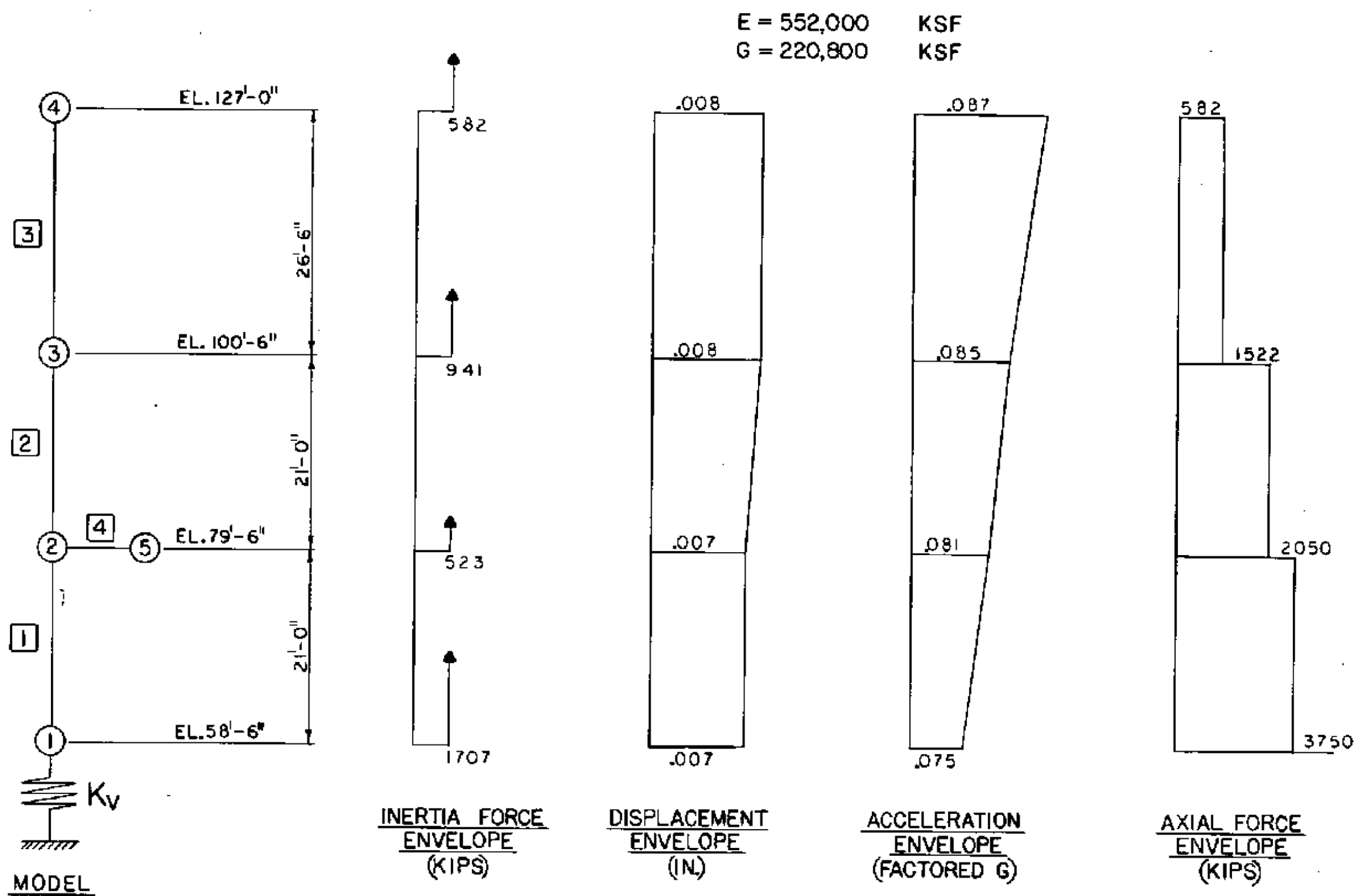
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RIVER INTAKE STRUCTURE SEISMIC RESULTS

FIGURE 3.7-41



VERTICAL DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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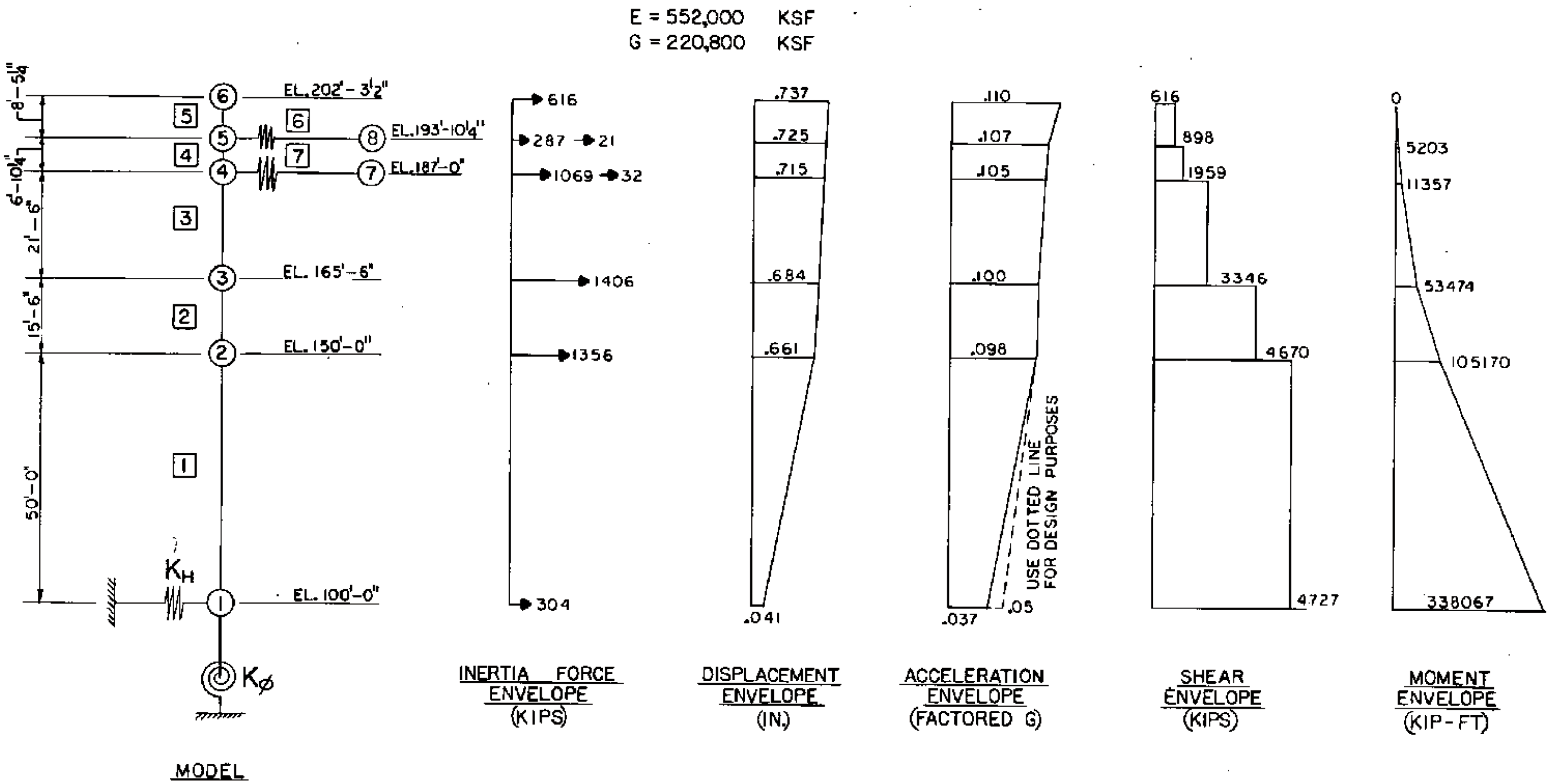
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RIVER INTAKE STRUCTURE SEISMIC RESULTS

FIGURE 3.7-42



NORTH – SOUTH DIRECTION
½ SAFE SHUTDOWN EARTHQUAKE

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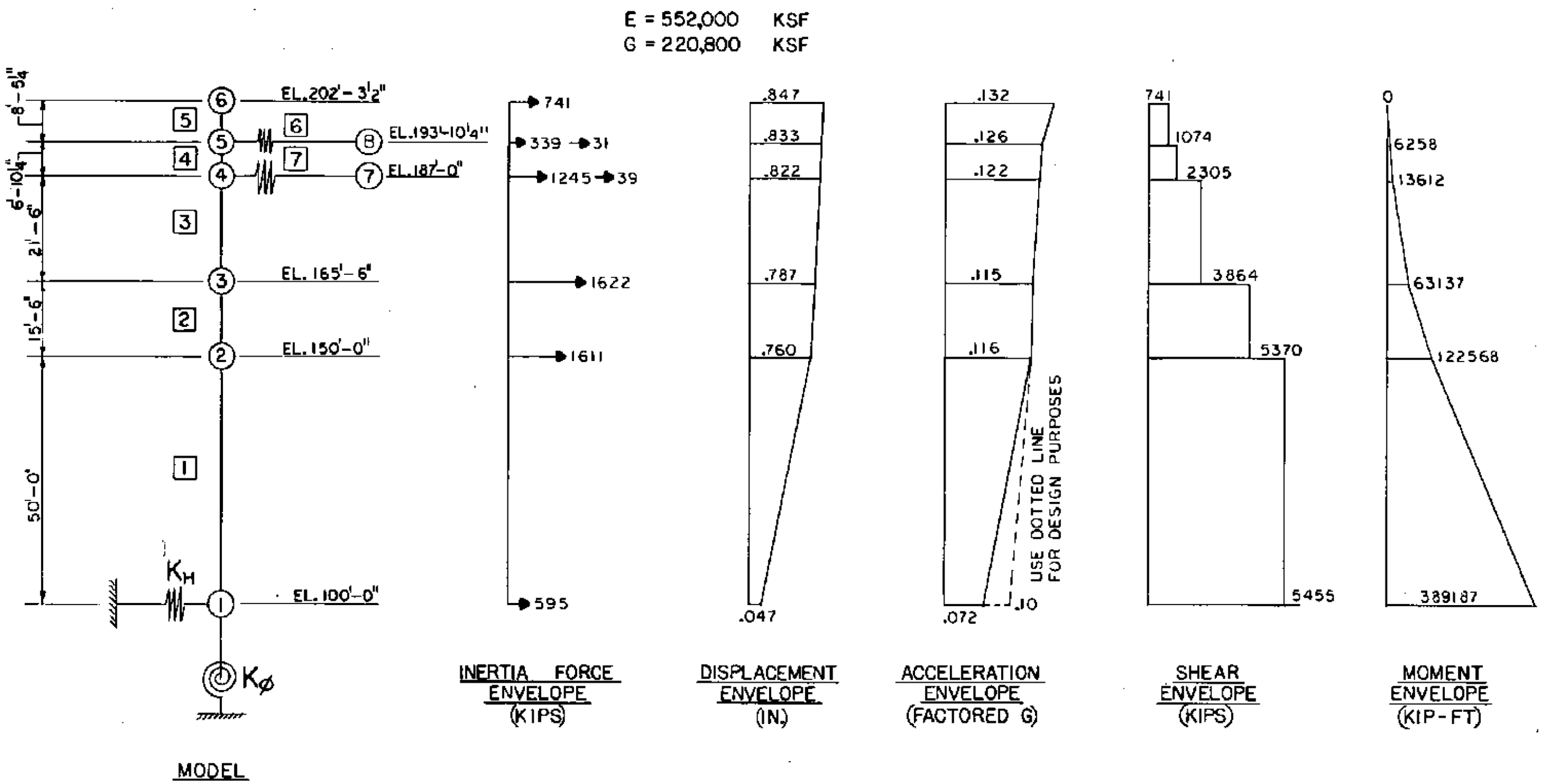
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTAKE STRUCTURE AT STORAGE POND
SEISMIC RESULTS

FIGURE 3.7-43



NORTH – SOUTH DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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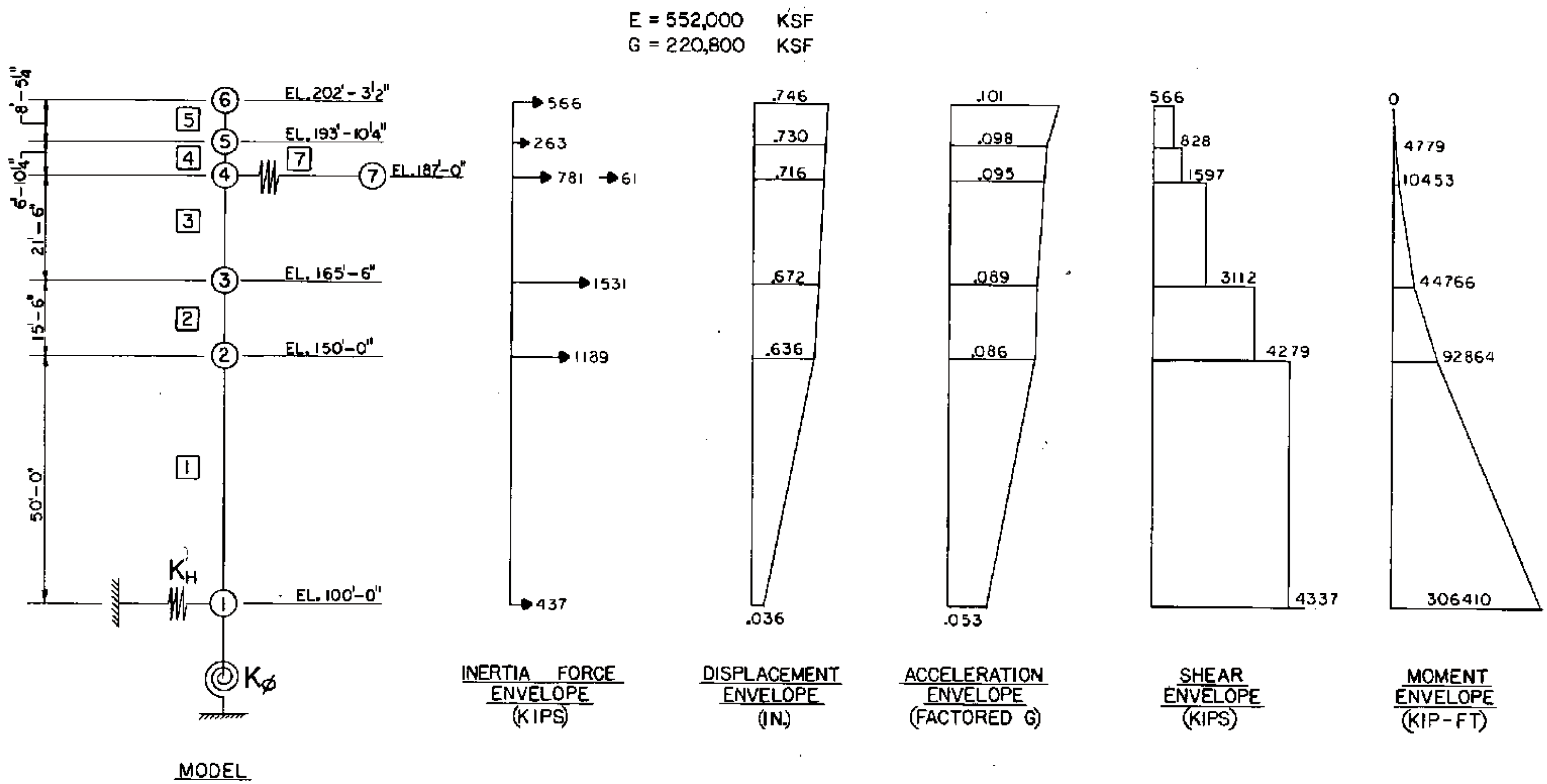
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTAKE STRUCTURE AT STORAGE POND
SEISMIC RESULTS

FIGURE 3.7-44



EAST - WEST DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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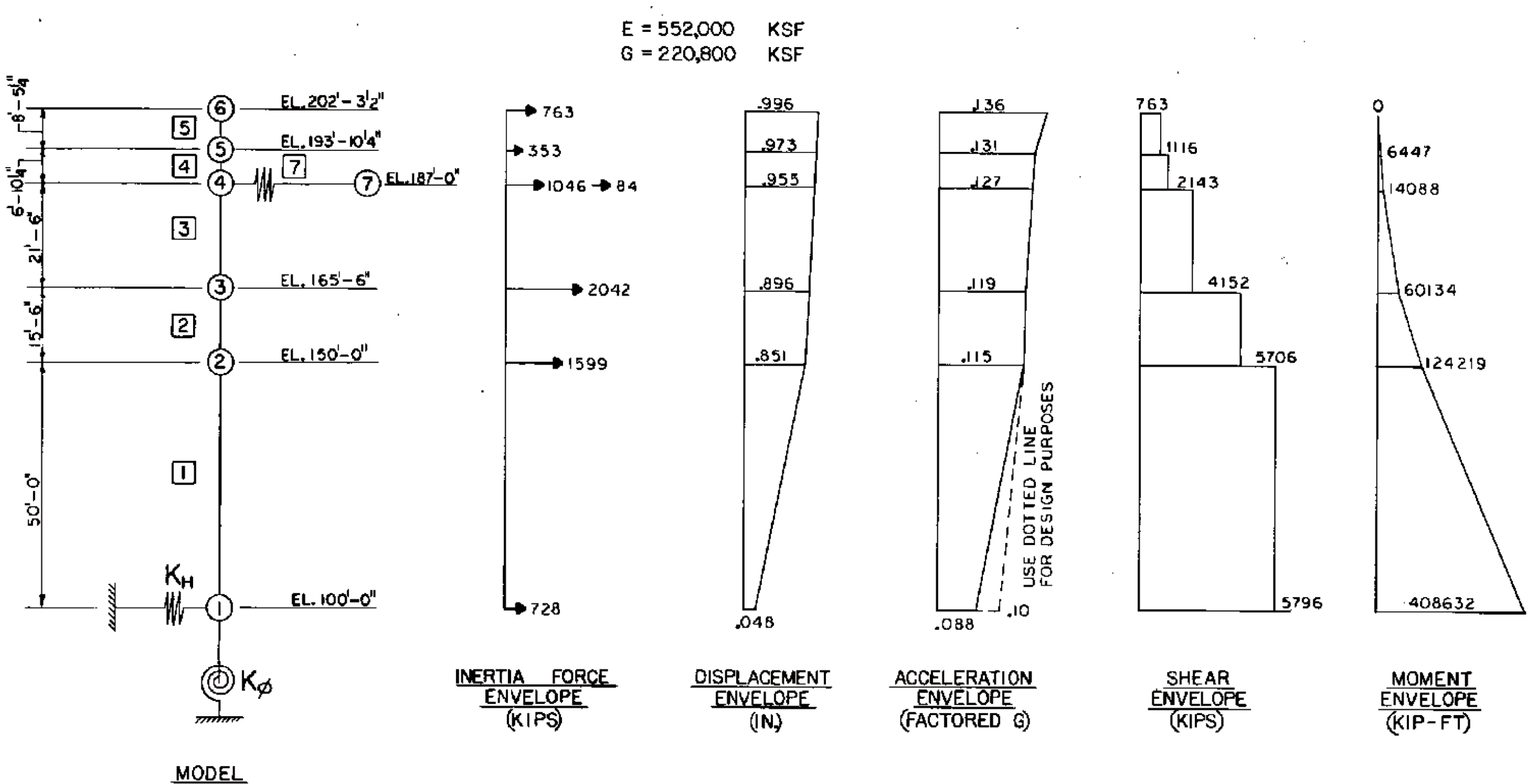
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTAKE STRUCTURE AT STORAGE POND
SEISMIC RESULTS

FIGURE 3.7-45



EAST-WEST DIRECTION
SAFE SHUTDOWN EARTHQUAKE

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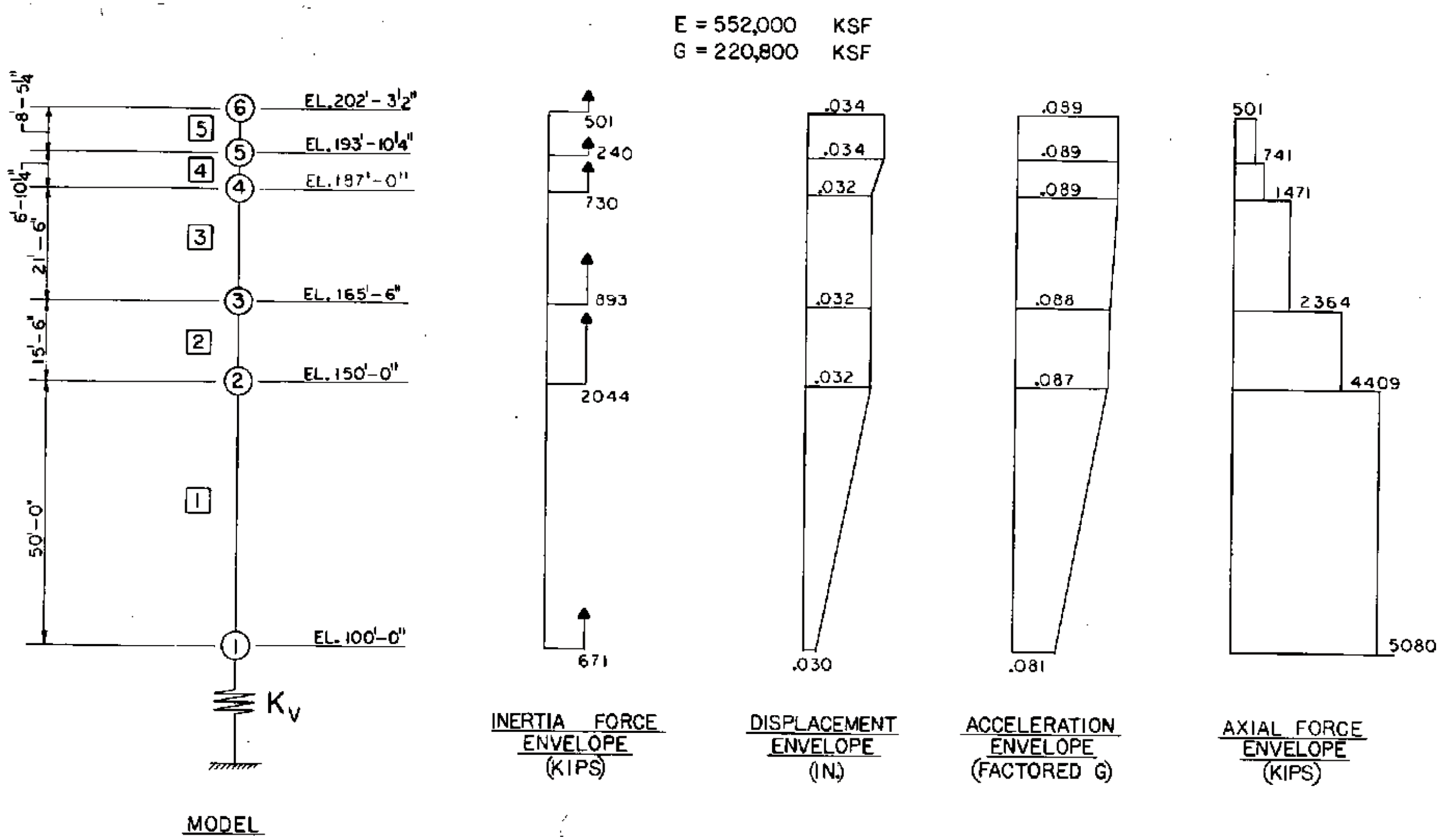
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTAKE STRUCTURE AT STORAGE POND
SEISMIC RESULTS

FIGURE 3.7-46



VERTICAL DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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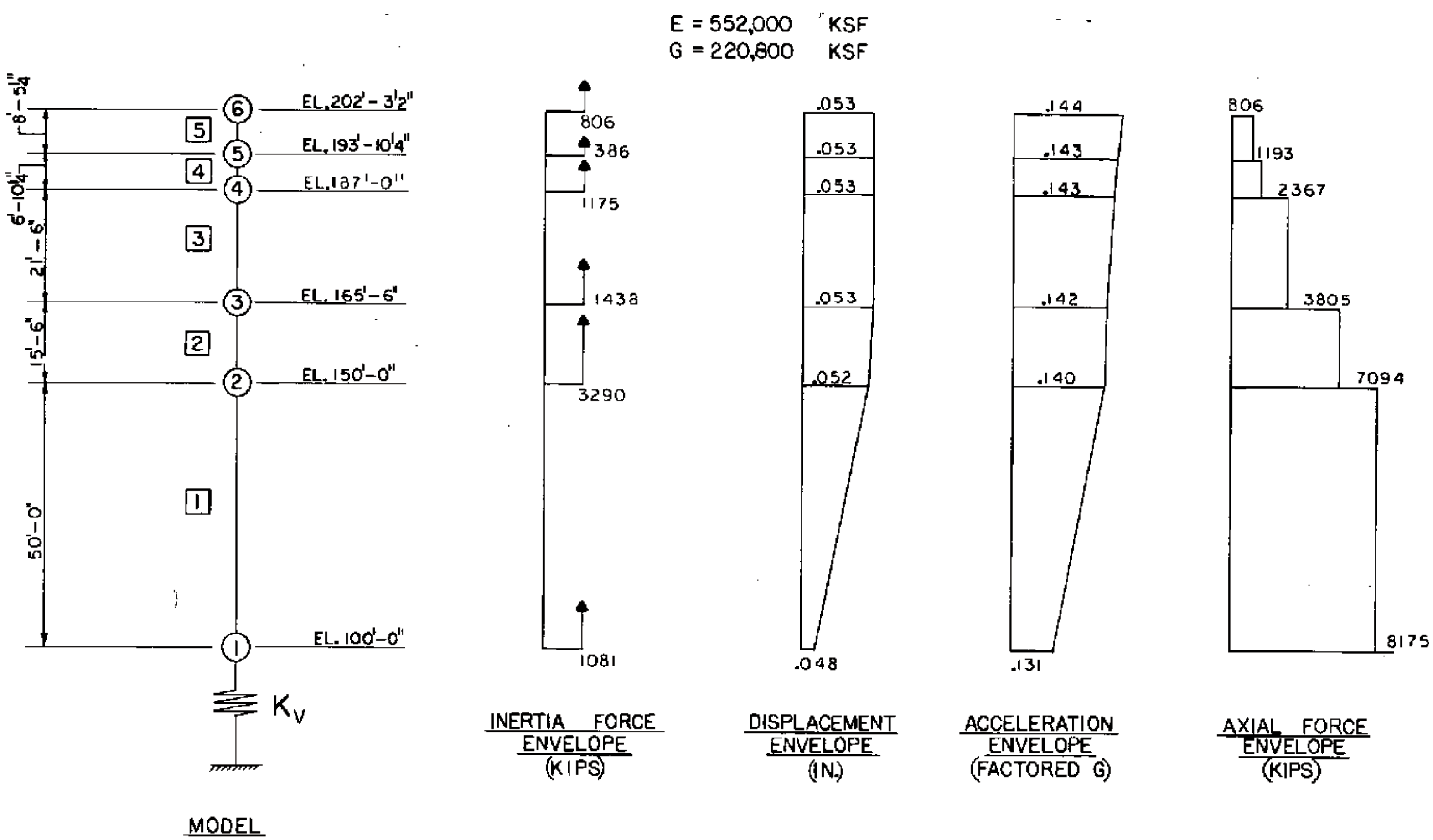
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

INTAKE STRUCTURE AT STORAGE POND SEISMIC RESULTS

FIGURE 3.7-47



VERTICAL DIRECTION
SAFE SHUTDOWN EARTHQUAKE

REV 21 5/08

SOUTHERN COMPANY
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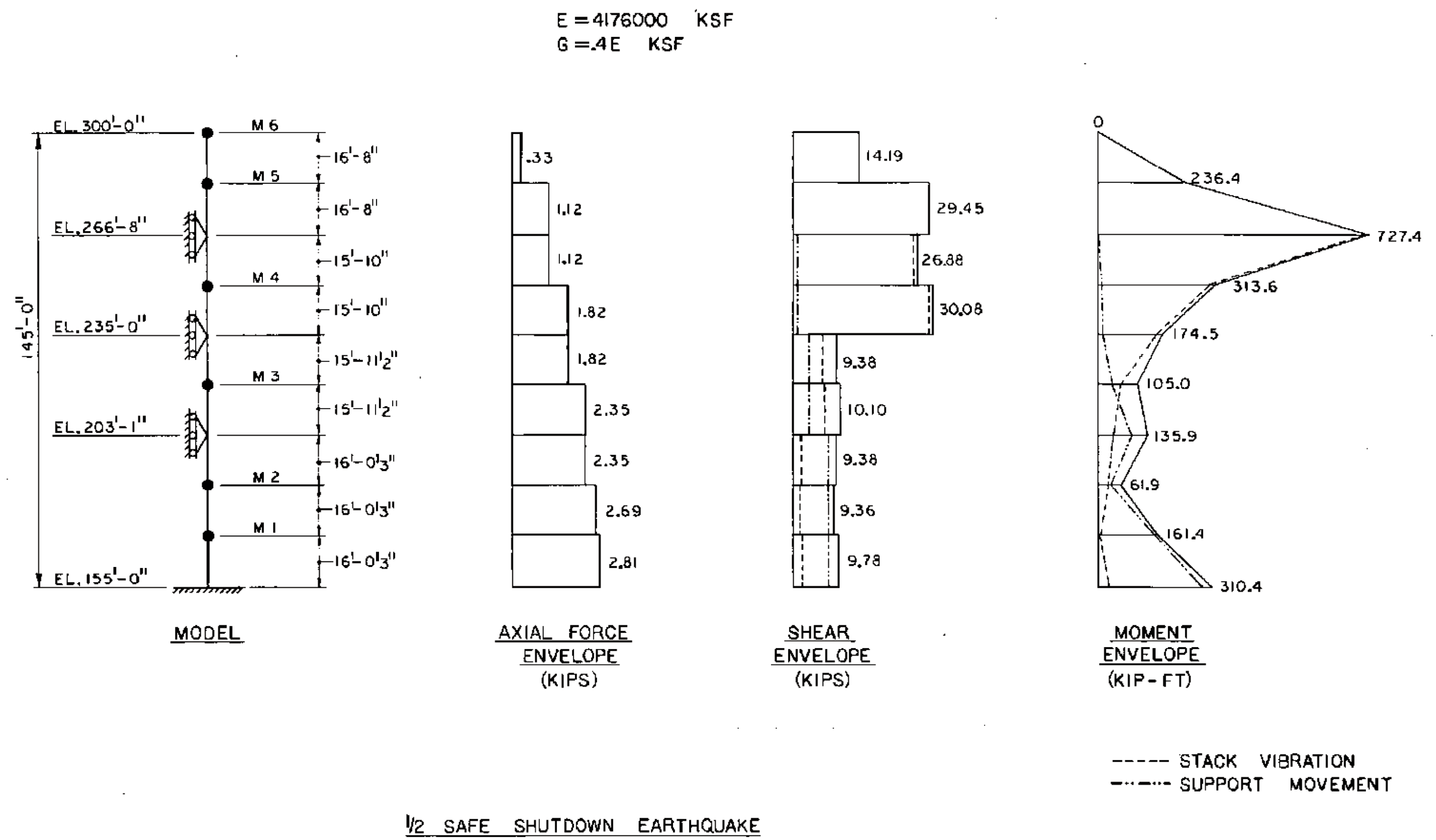
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

INTAKE STRUCTURE AT STORAGE POND SEISMIC RESULTS

FIGURE 3.7-48



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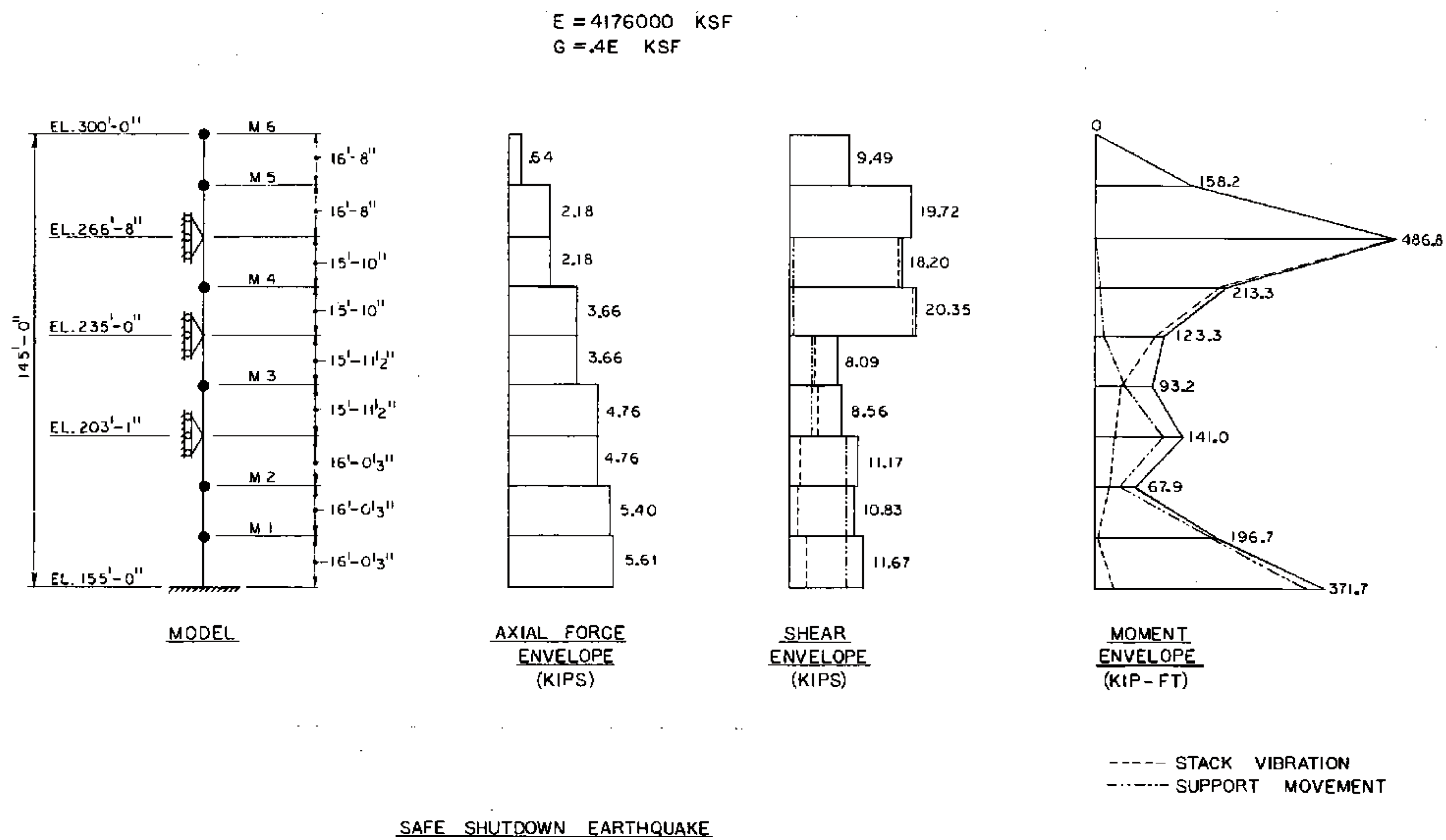
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

VENT STACK SEISMIC RESULTS

FIGURE 3.7-49



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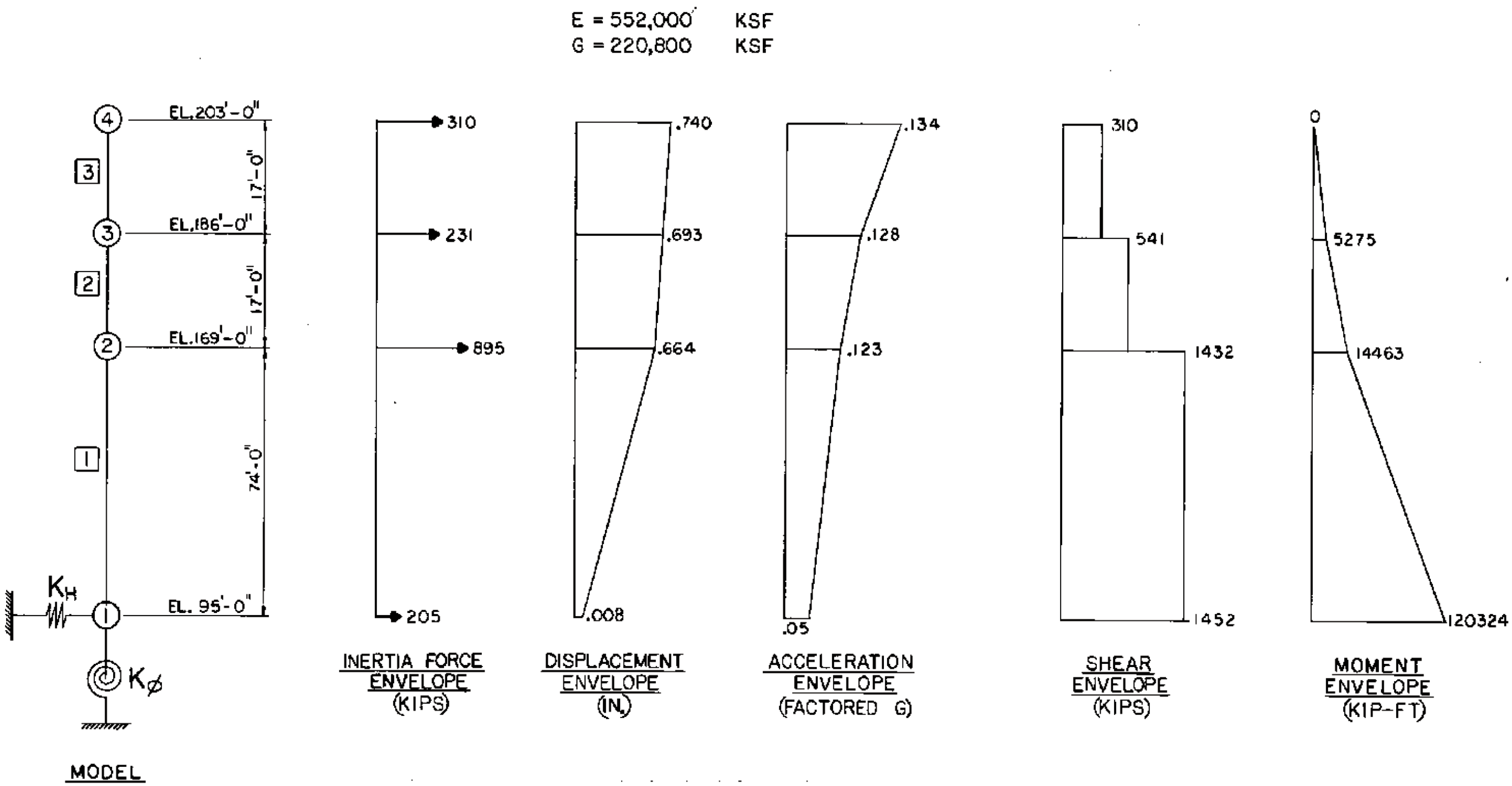
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

VENT STACK SEISMIC RESULTS

FIGURE 3.7-50



NORTH – SOUTH DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

REV 21 5/08

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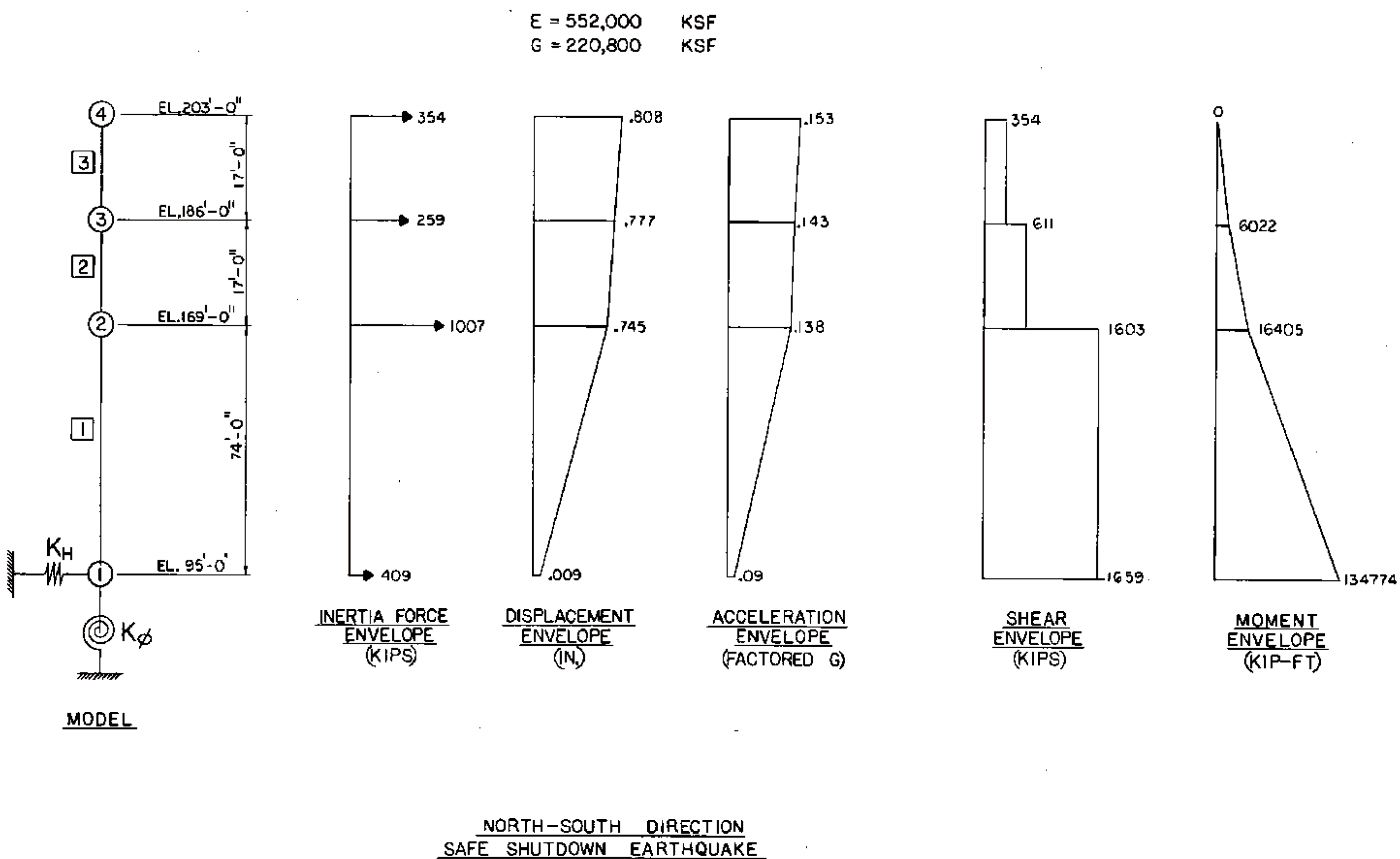
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE SEISMIC RESULTS

FIGURE 3.7-51



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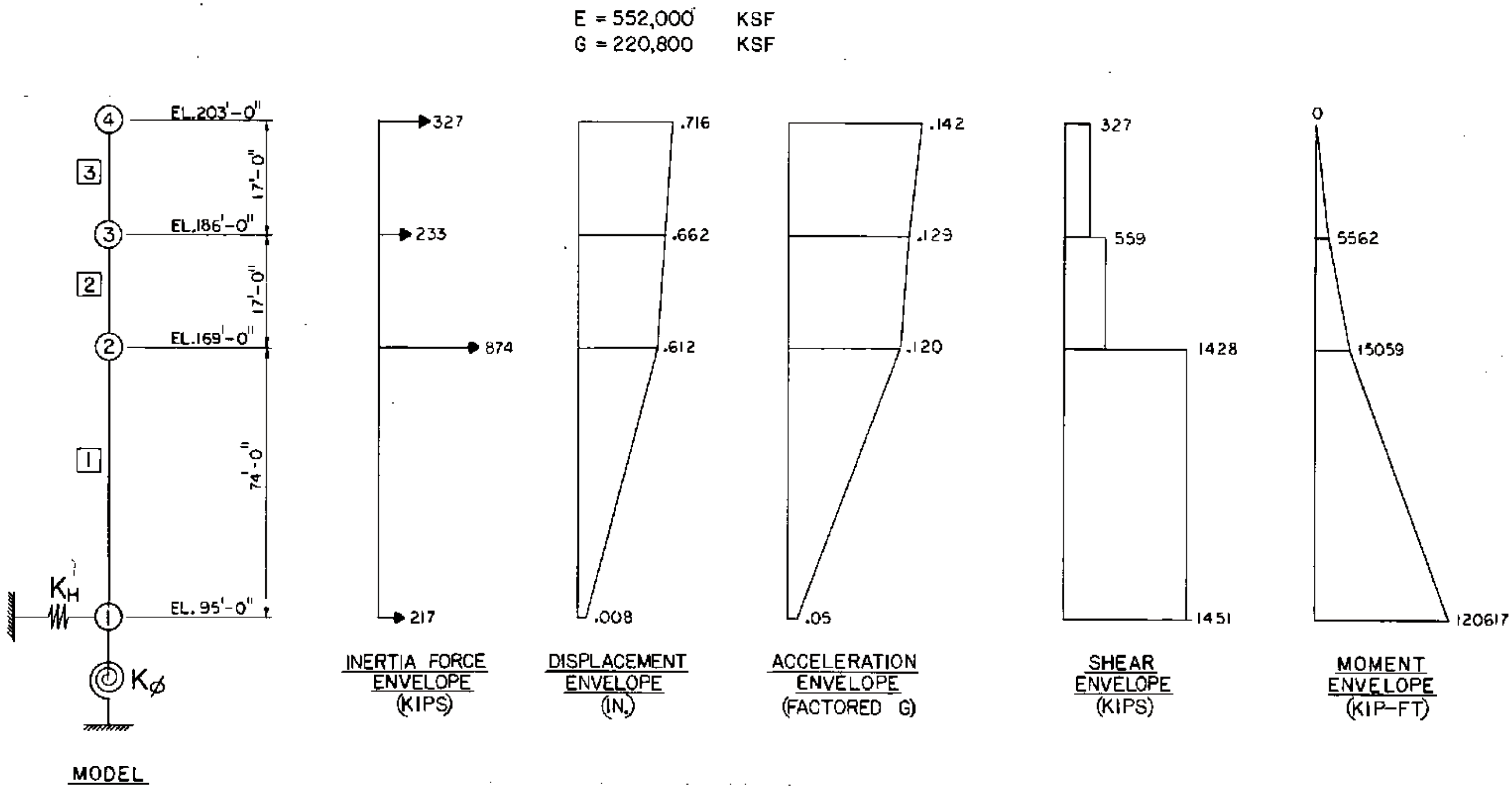
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE SEISMIC RESULTS

FIGURE 3.7-52



EAST–WEST DIRECTION

1/2 SAFE SHUTDOWN EARTHQUAKE

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SOUTHERN COMPANY

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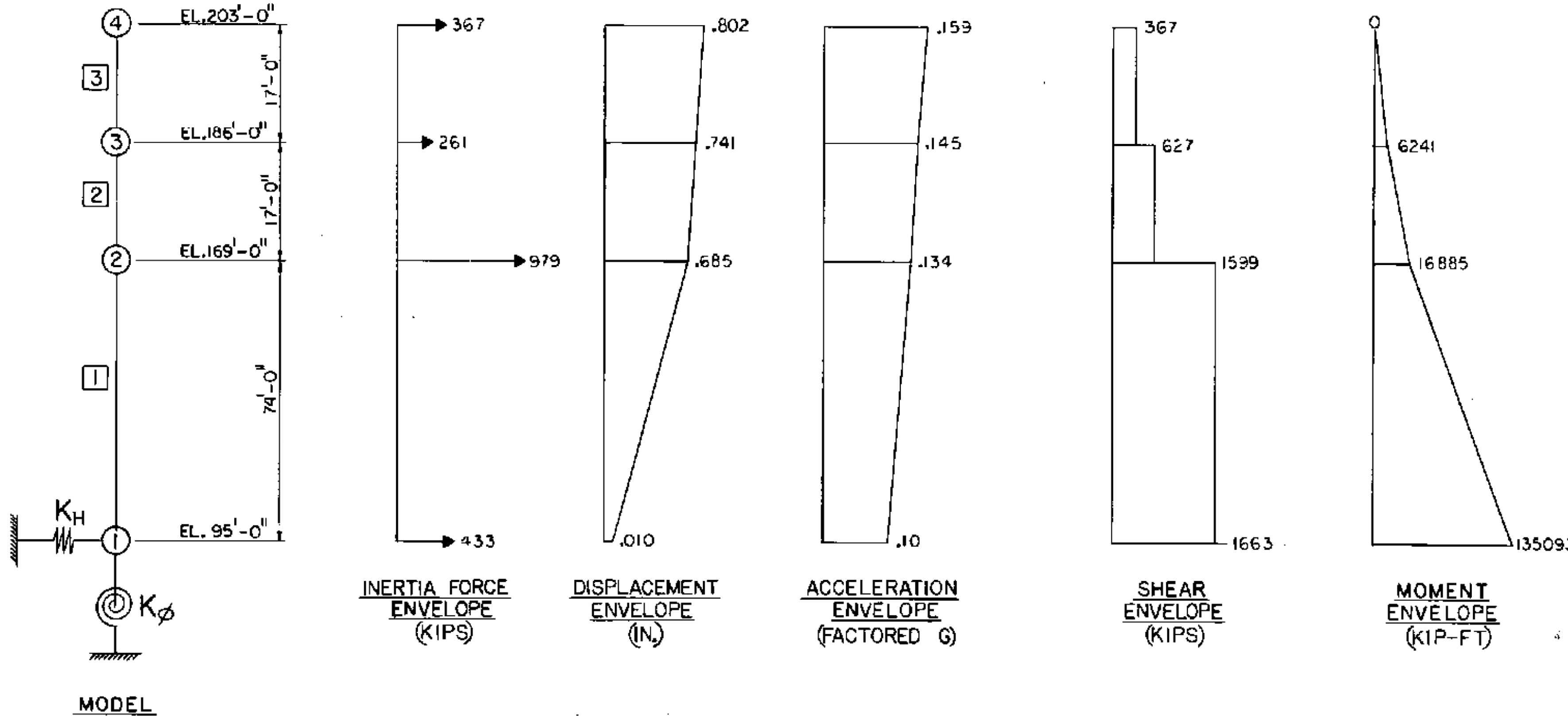
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE SEISMIC RESULTS

FIGURE 3.7-53

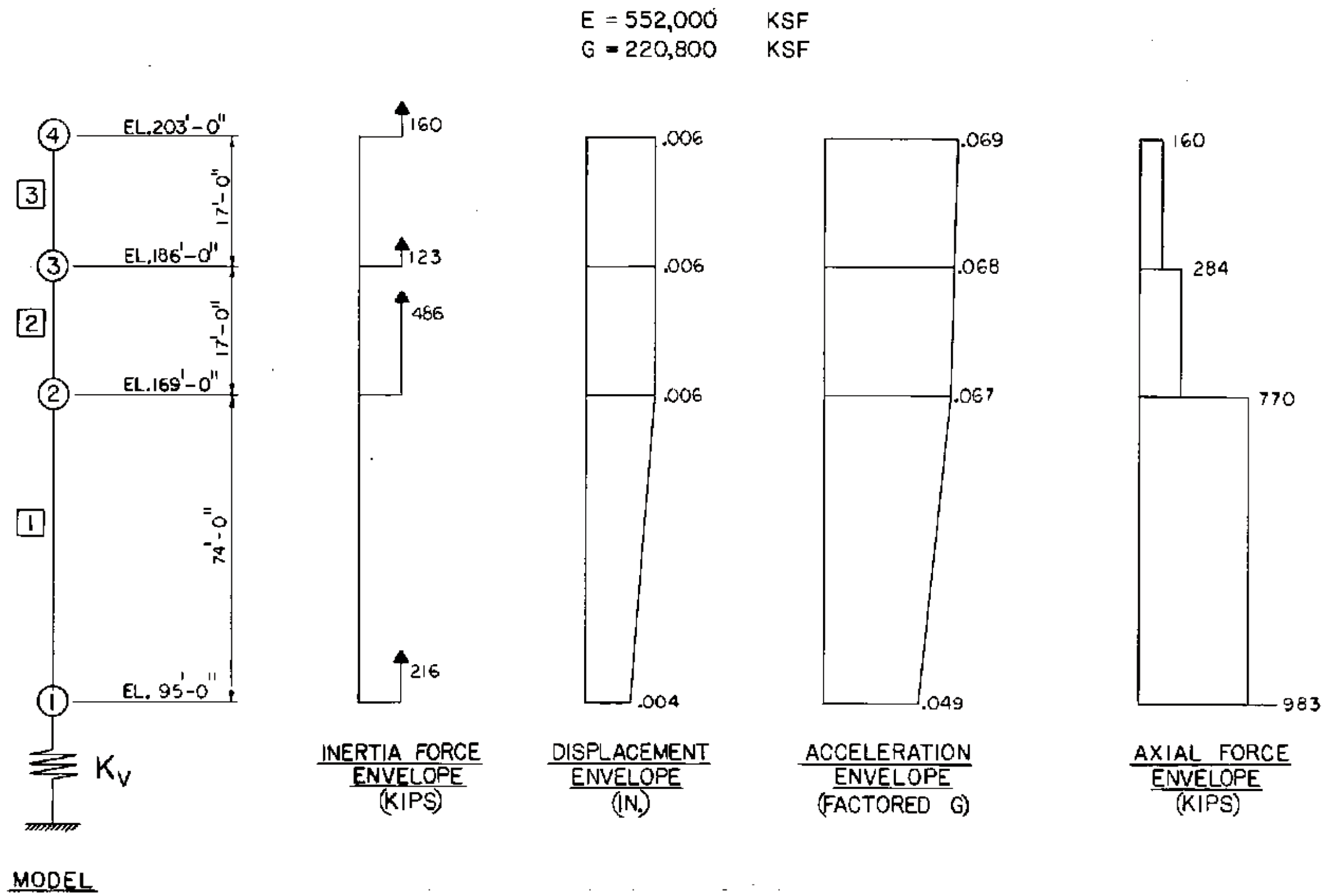
E = 552,000 KSF
G = 220,800 KSF



EAST-WEST DIRECTION
SAFE SHUTDOWN EARTHQUAKE

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	POND SPILLWAY STRUCTURE SEISMIC RESULTS
		FIGURE 3.7-54



VERTICAL DIRECTION
1/2 SAFE SHUTDOWN EARTHQUAKE

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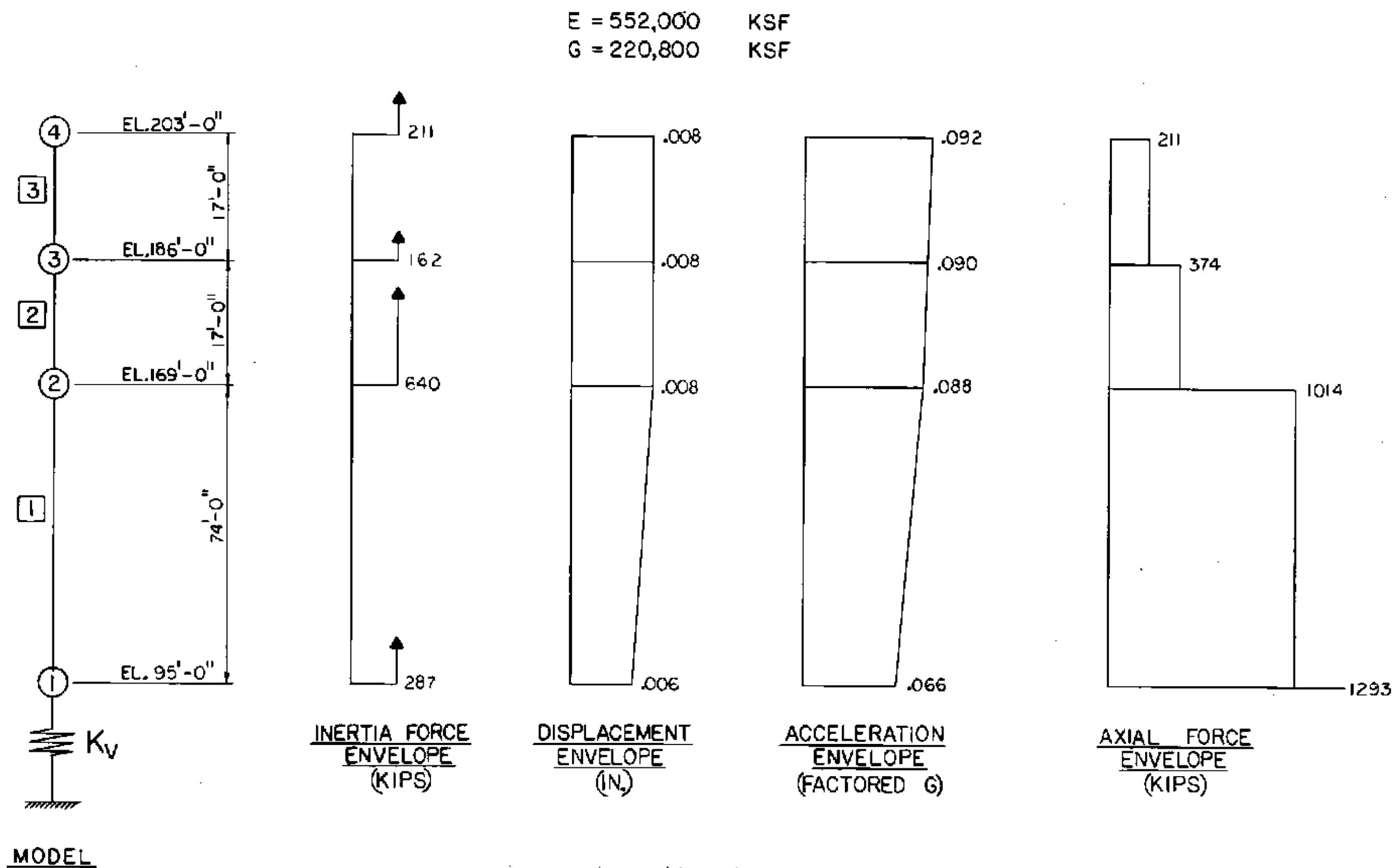
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE SEISMIC RESULTS

FIGURE 3.7-55



VERTICAL DIRECTION
SAFE SHUTDOWN EARTHQUAKE

REV 21 5/08

SOUTHERN COMPANY
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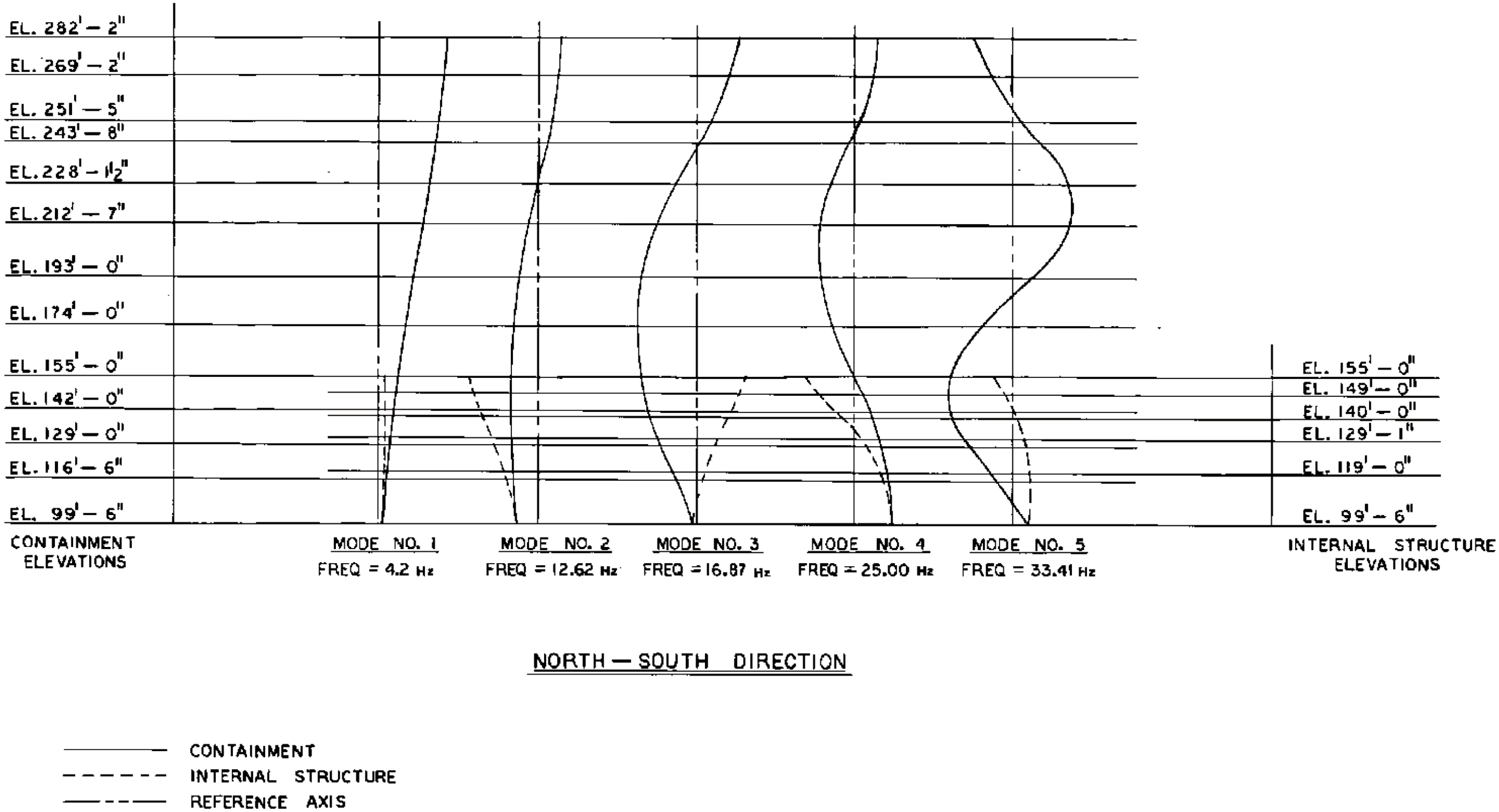
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE SEISMIC RESULTS

FIGURE 3.7-56



REV 21 5/08

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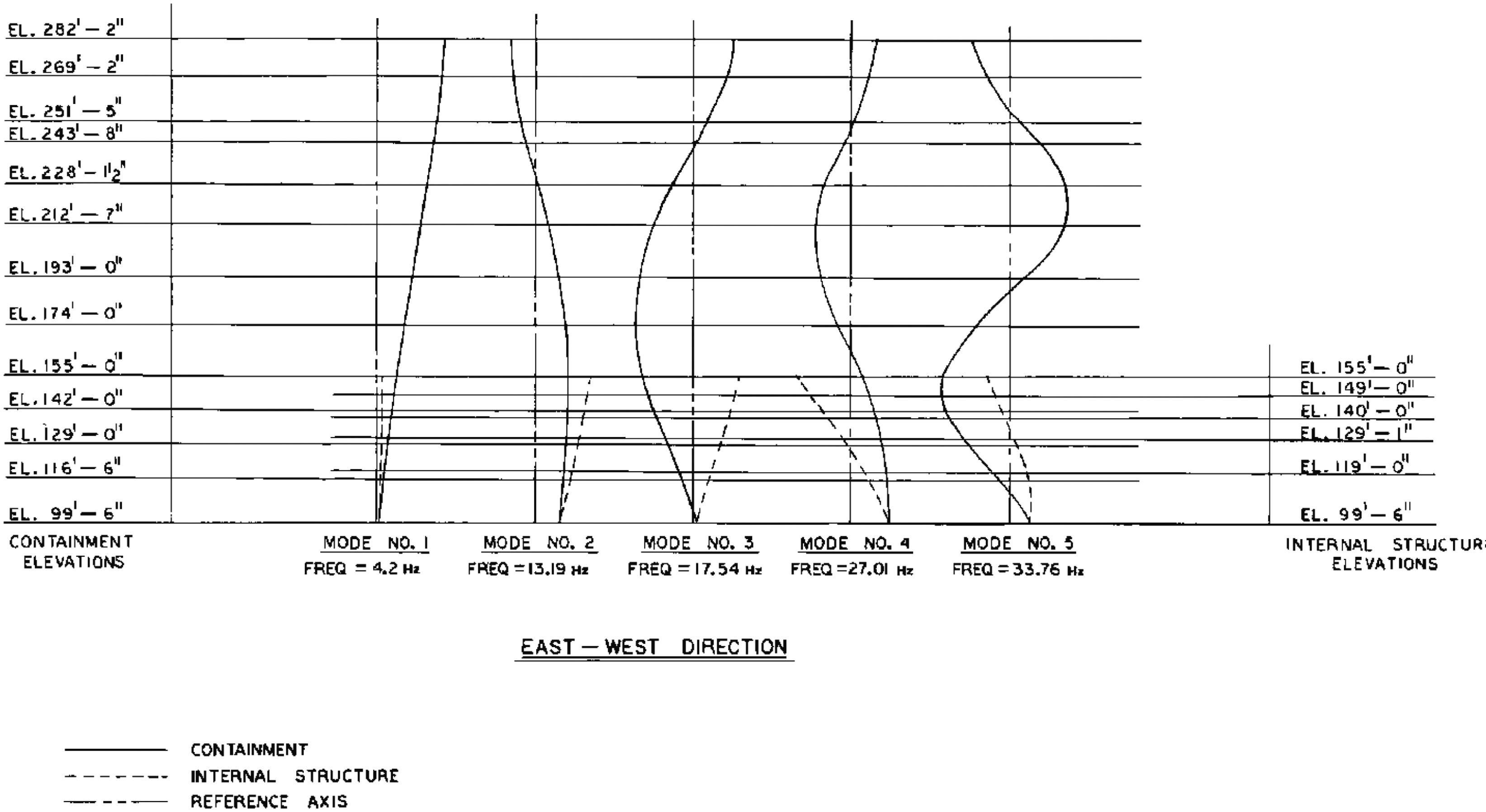
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AND INTERNAL STRUCTURE
FREQUENCIES AND MODE SHAPES

FIGURE 3.7-57



REV 21 5/08

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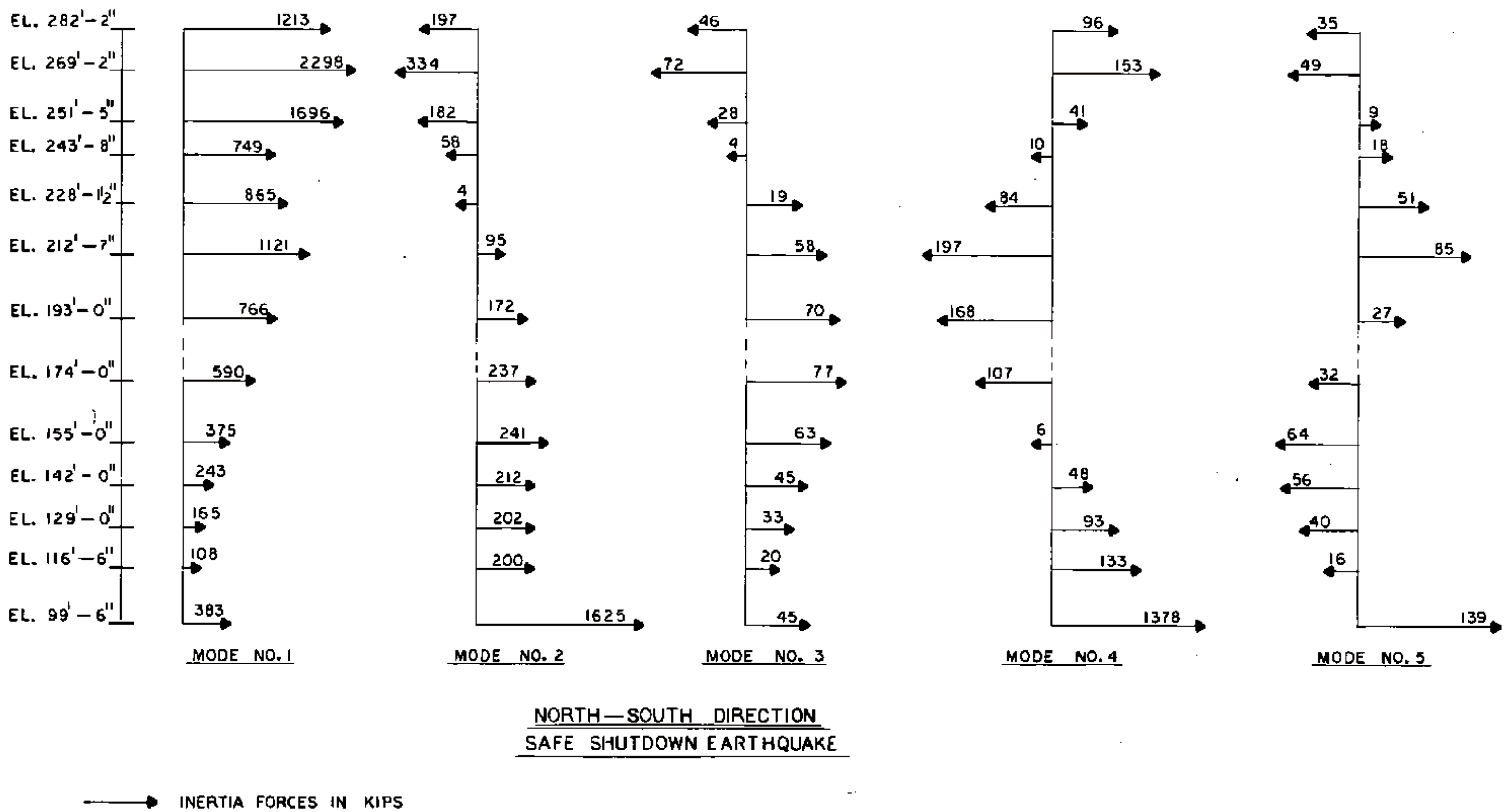
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AND INTERNAL STRUCTURE
FREQUENCIES AND MODE SHAPES

FIGURE 3.7-58



REV 21 5/08

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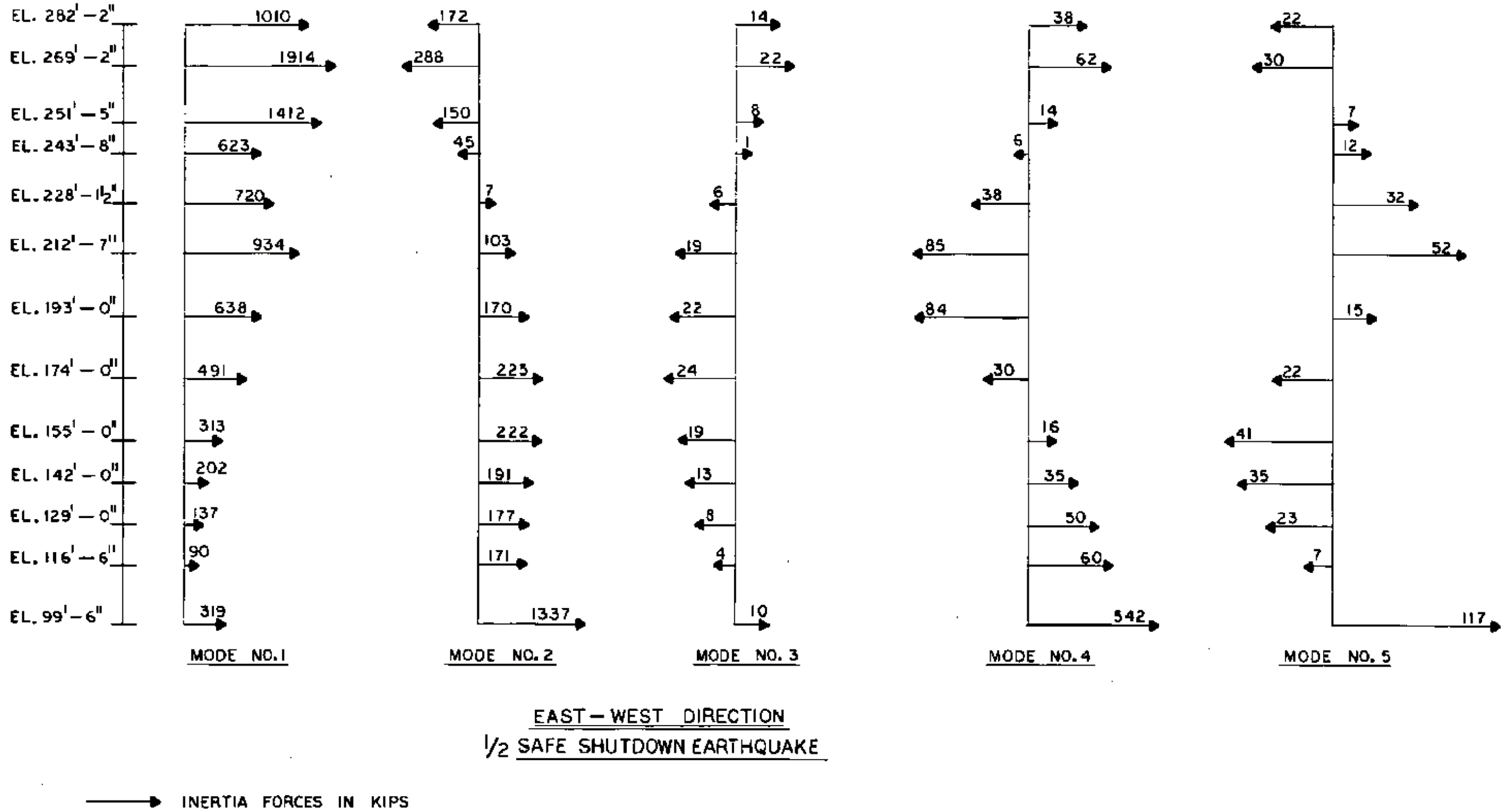
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MODAL INERTIA FORCES FOR CONTAINMENT

FIGURE 3.7-59



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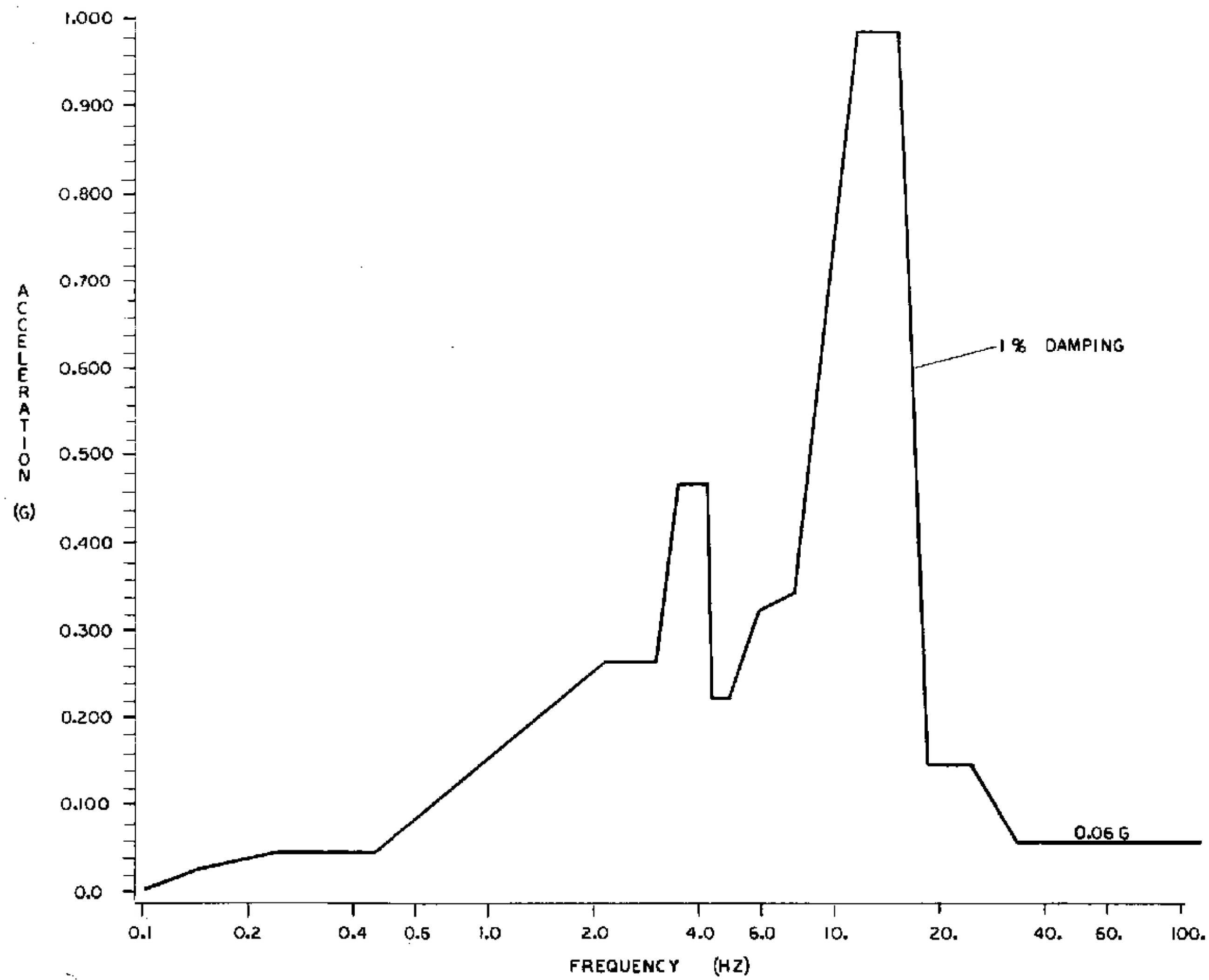
MODAL INERTIA FORCES FOR CONTAINMENT

FIGURE 3.7-60



NORTH - SOUTH DIRECTION ½ SAFE SHUTDOWN EARTHQUAKE

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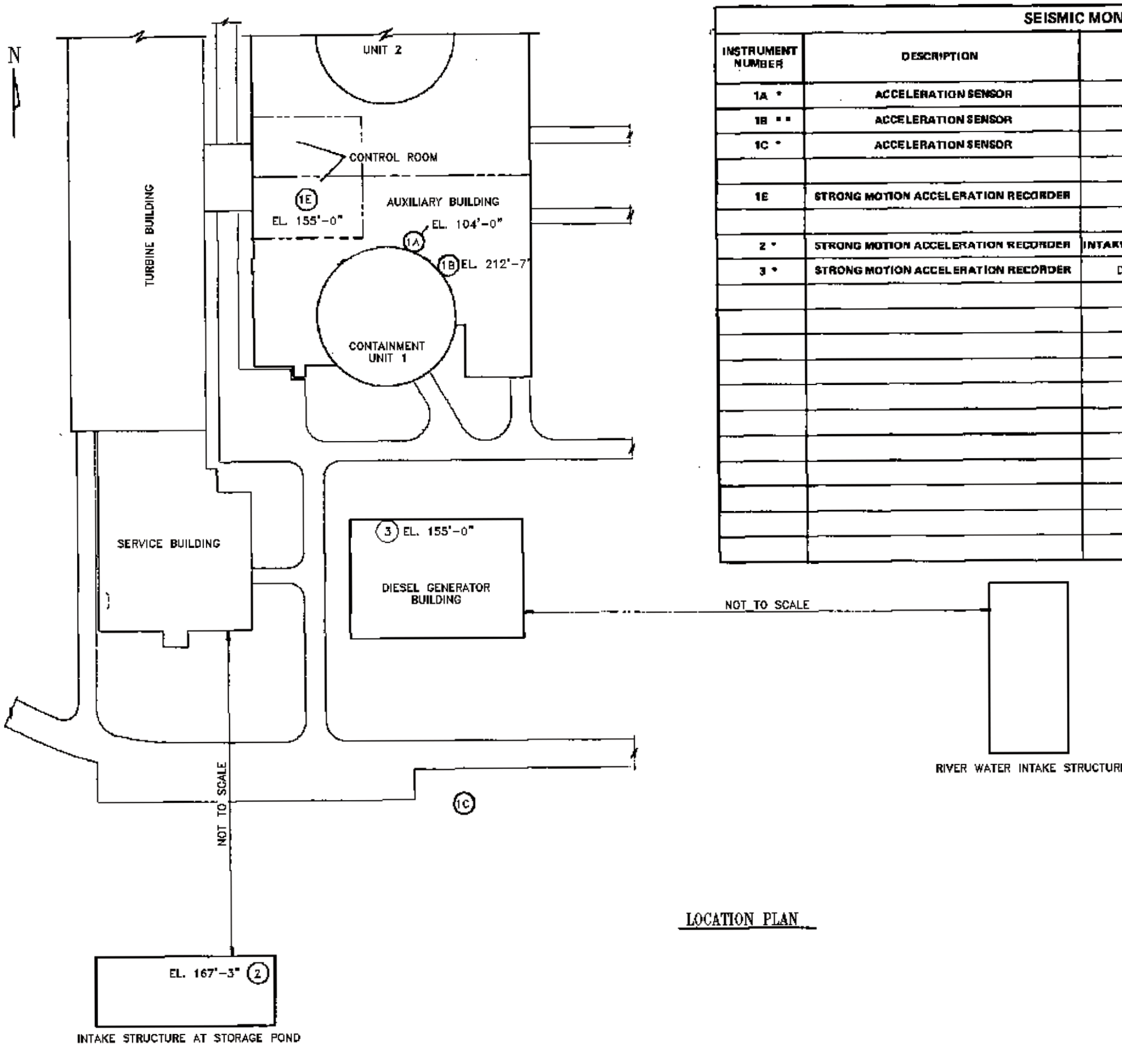
RESPONSE SPECTRUM AT
REACTOR SUPPORT ELEVATION

FIGURE 3.7-61



LOCATION PLAN

SEISMIC MONITORING DEVICES

INSTRUMENT NUMBER	DESCRIPTION	LOCATION	REMARKS
1A *	ACCELERATION SENSOR	CONTAINMENT	TRIAXIAL
1B **	ACCELERATION SENSOR	CONTAINMENT	TRIAXIAL
1C *	ACCELERATION SENSOR	FREE FIELD	TRIAXIAL
1E	STRONG MOTION ACCELERATION RECORDER	CONTROL ROOM	TIME HISTORY RECORD & PLAYBACK
2 *	STRONG MOTION ACCELERATION RECORDER	INTAKE STRUCTURE AT STORAGE POND	TIME HISTORY RECORD
3 *	STRONG MOTION ACCELERATION RECORDER	DIESEL GENERATOR BUILDING	TIME HISTORY RECORD

** 1.00g FULL SCALE

* 0.50g FULL SCALE, THIS RANGE WILL GIVE INCREASED ACCURACY AND SENSITIVITY.

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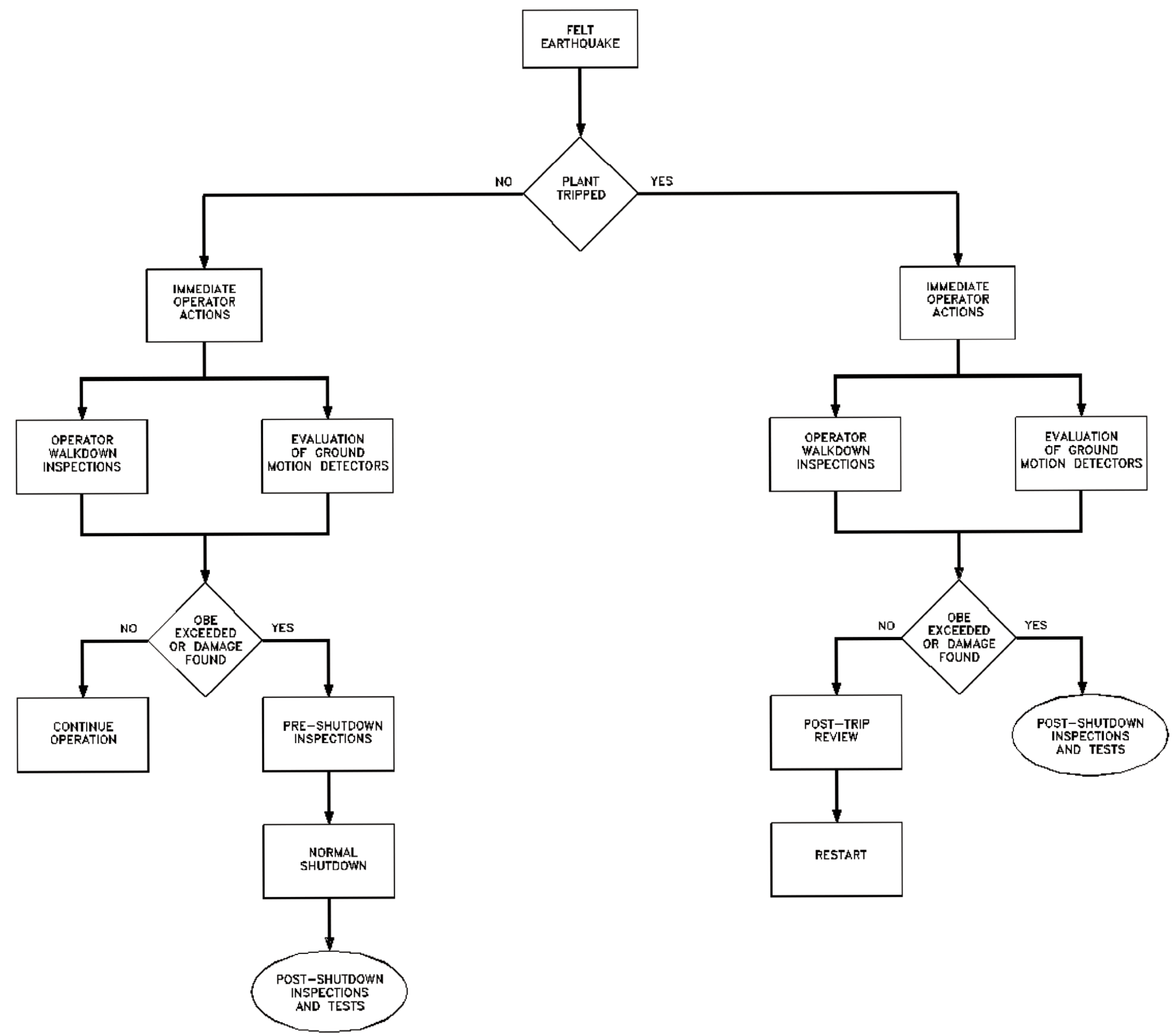
SEISMIC INSTRUMENTATION

FIGURE 3.7-62



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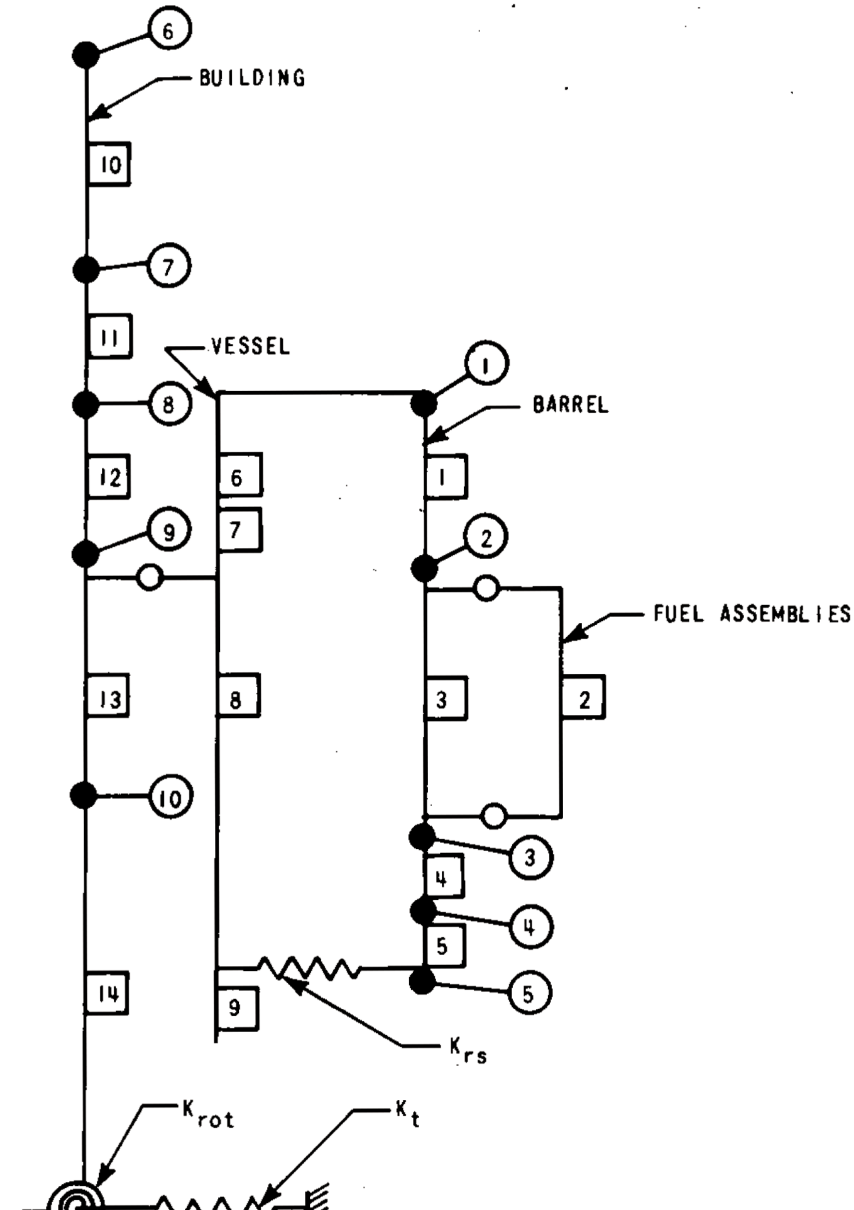
EARTHQUAKE EVALUATION PROCEDURE FOR CATEGORY 1 STRUCTURES

FIGURE 3.7-63

K_{rs} = RADIAL SUPPORT SPRING CONSTANT

K_{rot} = ROTATIONAL GROUND SPRING CONSTANT

K_t = TRANSLATIONAL GROUND SPRING CONSTANT



NOT TO SCALE

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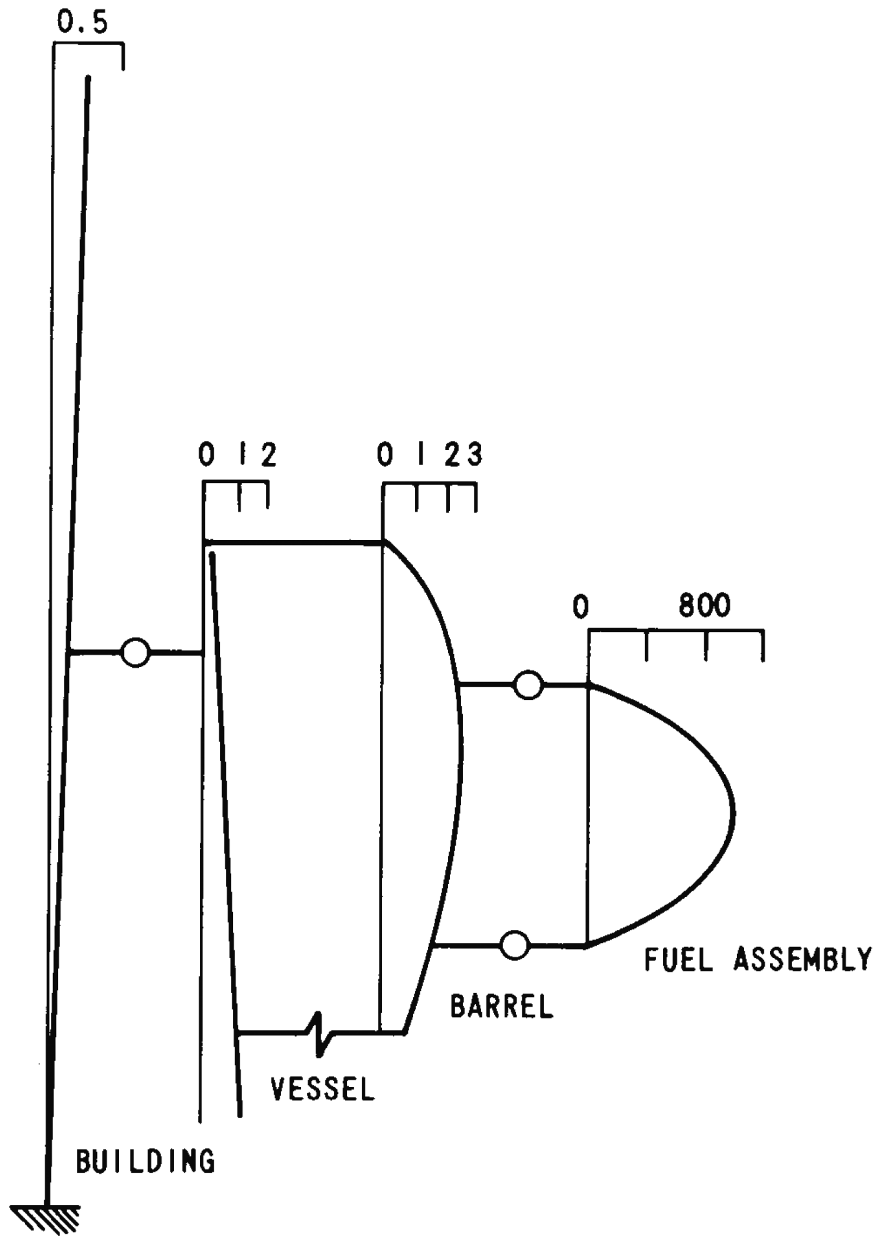
UNIT 1 AND UNIT 2

MATHEMATICAL MODEL OF REACTOR INTERNALS

FIGURE 3.7-64



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FIRST MODE VIBRATION OF REACTOR INTERNALS

FIGURE 3.7-65

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

The containment completely encloses the reactor, the reactor coolant systems, the steam generators, and portions of the auxiliary and engineered safeguards systems. It ensures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the reactor coolant system occurs. The structure provides biological shielding during normal operation and following a loss-of-coolant accident (LOCA). It also provides a vapor containment following an accident inside the containment. Further information relative to the containment is covered in topical BC-TOP-5, which provides the bases for design, construction, testing, and surveillance for the prestressed concrete containment.

3.8.1.1 Description of the Containment

a. Physical Description

The containment is a prestressed, reinforced concrete cylindrical structure with a shallow domed roof and a reinforced concrete foundation slab with provision for a reactor cavity at the center. The cylindrical portion of the containment is prestressed by a post-tensioning system composed of horizontal and vertical tendons. The horizontal tendons are placed in three 240-degree segments using three buttresses spaced 120 degrees apart as supports for the anchorages. The dome has a three-way tendon pattern in which groups of tendons intersect at 120 degrees. The concrete foundation is a conventionally reinforced mat. A continuous access gallery is provided beneath the base slab for installation and inspection of the vertical tendons.

A 1/4-in.-thick welded steel liner is attached to the inside face of the concrete. The floor liner is installed on top of the foundation slab and is then covered with concrete.

Principal nominal dimensions of the containment are as follows:

Interior diameter (ft)	130
Interior height (ft)	183
Cylindrical wall thickness (ft)	3 3/4
Dome thickness (ft)	3 1/4
Foundation slab thickness (ft)	9
Liner plate thickness (in.)	1/4
Internal free volume (ft^3)	2.0×10^6

The geometry and typical details of the containment and liner plate are shown in figures 3.8-1, 3.8-2, and drawing D-176145.

b. Description of Post-Tensioning System

The prestressed, post-tensioning system is a low relaxation Inland-Ryerson BBRV buttonhead system using 170 wires of 1/4-in. diameter per tendon. The pertinent features of the post-tensioning system are given in table 3.8-1. The tendons are installed in metal sheaths which form ducts through the concrete between anchorage points. Trumpets, which are enlarged ducts attached to the bearing plate, allow the wires to spread out at the anchorage to suit washer hole spacing and facilitate field buttonheading of wires. Figure 3.8-5 shows details of trumpet and sheathing. Sheaths are provided with a valved vent at the highest points of curvature to permit release of pockets of entrapped air during greasing operations. Drains are provided at the lowest points of curvature to remove accumulated water prior to installing tendons. In the process of greasing operations, the vents and drains are closed and sealed.

The prestressing wire is protected against atmospheric corrosion during its shipment and installation, and during the life of the containment. Prior to shipment, the wire is coated with a thin film of petrolatum containing rust inhibitors. The interior surface of the sheathing is coated with a suitable material during manufacture to minimize removal of the petrolatum from the tendon wires during pulling through the sheathing. The sheathing filler material used for permanent corrosion protection is a modified, refined petroleum oil base product. The material is pumped into the sheathing after stressing.

The vertical tendons are anchored at the top of the ring girder and at the bottom of the foundation slab. The hoop tendons are anchored at buttresses 240 degrees apart, bypassing an intermediate buttress. The anchorages of each successive hoop tendon are progressively offset 120 degrees from the one beneath it. The three way dome tendons are anchored at the side of the ring girder.

For the arrangement of the prestressing tendons, especially at penetrations, anchorage zones, connections and joints, see figures 3.8-1, 3.8-2, and drawing D-176145.

c. Description of the Equipment Hatch, Personnel Locks, and Electrical Penetration.

The geometry of the equipment hatch is shown on drawing D-176145. The personnel access lock and the auxiliary access lock are both shown on figure 3.8-40. The electrical penetrations are shown on drawing D-176151.

3.8.1.2 Applicable Codes, Standards and Specifications

The following codes, standards, specifications, design criteria, NRC Regulatory Guides, and industry standard practices constitute the basis for the design and construction of the containment. Modifications to these codes, standards, etc. are made when necessary, to meet the specific requirements of the structure. These modifications are indicated in the sections where references to the codes, standards, etc. are made.

Codes

1.	ACI 214-65	"Recommended Practices for Evaluation of Compression Test Results of Field Concrete".
2.	ACI 301-66	"Specifications for Structural Concrete for Buildings".
3.	ACI 306-66	"Recommended Practice for Cold Weather Concreting".
4.	ACI 307-69	"Specification for the Design and Construction of Reinforced Concrete Chimneys".
5.	ACI 311-64	"Recommended Practice for Concrete Inspection".
6.	ACI 315-65	<u>Manual of Standard Practice for Detailing Reinforced Concrete Structures.</u>
7.	ACI 318-63	“Building Code Requirements for Reinforced Concrete".
8.	ACI 605-59	"Recommended Practice for Hot Weather Concreting".
9.	ACI 613-54	"Recommended Practice for Selecting Proportions for Concrete".
10.	ACI 614-59	"Recommended Practices for Measuring, Mixing, and Placing Concrete".
11.	AISC	<u>Manual of Steel Construction,</u> 1969 Edition.
12.	AWS D2.0-69	"Specifications for Welded Highway and Railway Bridges".
13.	ASME	"Boiler and Pressure Vessel Code", Sections III, VIII, and IX - 1968 Edition.
14.	ICBO	"Uniform Building Code" - 1970 Edition.
15.	SBCC	"Southern Standard Building Code" - 1969 Edition.

16. CFR — Federal Register, Title 29, Part 1910, Department of Labor, Occupational Safety, and Health Standards.

17. CFR — Code of Federal Regulations, Title 10, Part 100, Appendix A, Part 50.

18. AWS D1.1-86 — "Structural Welding Code - Steel".

19. NCIG-01, Rev. 2 — "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants" - EPRI NP-5380.

Standards

1. ACI — Manual of Concrete Inspection - sp 2

Specifications

1. CMAA — "Specifications for Electric Overhead Traveling Cranes" - No. 70 - 1970 Edition.

2. ASTM — The specifications utilized are identified in the applicable sections.

Design Criteria

1. ASCE — "Wind Forces on Structures", Paper No. 3269.

2. NRC — "Nuclear Reactor and Earthquake", Publication TID 7024.

Industry Standard Practices

Naval Documents — "Design of Protective Structures - A New Concept of Structural Behavior" - Naval Document P-51.

NRC Regulatory Guides

Regulatory Guide No. 1.10 — "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures."

Regulatory Guide No. 1.12 — "Instrumentation for Earthquakes."

Regulatory Guide No. 1.13 — "Fuel Storage Facility Design Basis."

Regulatory Guide No. 1.15 — "Testing of Reinforcing Bars for Category I Concrete Structures."

Regulatory Guide No. 1.18 — "Structural Acceptance Test For Concrete Primary Reactor Containments."

Regulatory Guide No. 1.19	"Nondestructive Examination of Primary Containment Liner Welds."
Regulatory Guide No. 1.28	"Quality Assurance Program Requirements - Design and Construction."
Regulatory Guide No. 1.29	"Seismic Design Classification."
Regulatory Guide No. 1.31	"Control of Stainless Steel Welding."
Regulatory Guide No. 1.35	Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures."
Regulatory Guide No. 1.38	"Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water Cooled Nuclear Power Plants,"
Regulatory Guide No. 1.46	"Protection Against Pipe Whip Inside Containment."
Regulatory Guide No. 1.54	"Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants."
Regulatory Guide No. 1.55	"Concrete Placement in Category I Structures."
Regulatory Guide No. 1.59	"Design Basis Floods for Nuclear Power Plants."
Regulatory Guide No. 1.64	"Quality Assurance Program Requirements for the Design of Nuclear Power Plants."

Discussions on the compliance with and interpretations of Regulatory Guides are presented in appendix 3A.

BECHTEL POWER CORPORATION TOPICAL REPORTS

B-TOP-3 "Design Criteria for Nuclear Power Plants Against Tornadoes," March 1970.

BP-TOP-1 "Seismic Analysis of Piping Systems," February 1974.

BC-TOP-4 "Seismic Analysis of Structures and Equipment for Nuclear Power Plants," September 1972.

BC-TOP-5 "Prestressed Concrete Nuclear Reactor Containment Structures," December 1972.

BC-TOP-7 "Full Scale Buttress Test for Prestressed Nuclear Containment Structures," Revision 0, September 1972.

BC-TOP-8 "Tendon End Anchor Reinforcement Test," Revision 0, September 1972.

BC-TOP-1 "Containment Building, Liner Plate Design Report," Revision 1, December 1972.

BN-TOP-1 "Testing Criteria for Integrated Leak - Rate Testing of Primary Containment Structures for Nuclear Power Plants," Revision 1, November 1972.

BC-TOP-9A "Design of Structures for Missile Protection," Revision 2, September 1974.

3.8.1.2.1 Discussions of Codes and Standard Specifications

a. American Concrete Institute, Building Code Requirements for Reinforced Concrete (ACI-318-71)

The ACI-318-71 code has not been used in the design of the containment.

b. American Concrete Institute, Concrete Shell Structures (ACI-334)

In order to ensure consistency of the analysis results, equilibrium checks of internal stresses and external loads were made in a similar manner to that described in Sections 202 (d) and 202 (e) of the ACI Committee 334 Report, "Concrete Shell Structures Practice and Commentary," Journal of the ACI, Title No. 61, Proceedings V 61, No. 9, pp. 1091-1108, 1964.

c. American Concrete Institute, Specifications for the Design and Construction of Reinforced Concrete Chimneys (ACI-307-69)

The Specifications for the Design and Construction of Reinforced Concrete Chimneys (ACI-307-69) is not used for consideration of self-relieving effects of thermal stresses. Instead, reference is made to the considerations outlined in Section 2.5.6.3.3 of ACI-349 Committee Report.

d. American Concrete Institute and American Society of Mechanical Engineers, "Proposed Standard Code for Concrete Reactor Vessels and Containments." (ACI-359)

The joint-ACI-ASME (ACI-359) document, Proposed Standard Code for Concrete Reactor Vessels and Containments, has not been used in the design of the containment.

3.8.1.2.2 Structural Specifications

Structural specifications are prepared to cover the areas related to design and construction of the containment. These specifications emphasize important points of the industry standards for the design and construction of the containment, and reduce options that otherwise would be permitted by the industry standards. Unless specifically noted otherwise, these specifications do not deviate from the applicable industry standards. They cover the following areas:

a. Concrete material properties.

b. Placing and curing of concrete.

c. Reinforcing steel and splices.

d. Post-tensioning system.

e. Liner plate and penetration assemblies.

3.8.1.3 Loads and Loading Combinations

The containment is designed for all credible conditions of loadings, including normal loads, loads resulting from a loss-of-coolant accident, test loads, and loads due to adverse environmental conditions.

Critical loading combinations are those caused by a postulated loss of reactor coolant, by a postulated earthquake, or by a pipe rupture in the containment.

Wind and tornado loads, flood design bases, and seismic loads are given in sections 3.3, 3.4 and 3.7, respectively. Missile effects and the postulated pipe rupture effects are discussed in sections 3.5 and 3.6. Chapter 15.0, "Accident Analyses", provides information on the design pressure load.

1. Loads

The following loads are considered: dead loads; live loads; prestressing loads; earthquake loads; pipe rupture loads; loss-of-coolant accident loads; operating thermal loads; wind and tornado loads; external pressure loads; hydrostatic loads; and test loads.

a. Dead Loads:

Structural dead loads consist of the weight of the containment wall, dome, base slab, interior framing and slabs, all internal structures, equipment, and major piping and electrical conductors.

b. Live Loads:

Live loads consist of design floor loads, equipment live loads, and all live loads transmitted by the internal structures.

The operating floor slab is designed for either of these live loads:

Floor gratings	450 psf
Concrete slabs	1000 psf

Equipment live loads are those specified on drawings supplied by the manufacturer of the equipment.

c. Prestressing Loads:

The compressive forces due to the prestressing tendons are taken into consideration.

d. Earthquake Loads:

Earthquake loads are predicated on a basis of the 1/2 safe shutdown earthquake (1/2 SSE), having a horizontal ground acceleration of 0.05 g and a vertical ground surface acceleration of 0.033 g.

In addition, a safe shutdown earthquake (SSE), having a horizontal ground acceleration of 0.10 g and a vertical ground surface acceleration of 0.067 g, is used to check the design to ensure that loss of structural functions will not occur.

Seismic response spectrum curves are given in section 3.7 for both horizontal and vertical ground motions. A dynamic analysis is used to compute the seismic loads for the design of structural elements.

e. Pipe Rupture Loads:

Pipe rupture loads represent the forces or pressure on the structure due to the rupture of any one pipe.

f. Loss-of-Coolant Accident Loads:

The design pressure and temperature of the containment are greater than the peak pressure and temperature that would result from a postulated complete blowdown of the reactor coolant. This might occur through the rupture of the reactor coolant system, up to and including the hypothetical double ended severance of the largest reactor coolant pipe.

Pressure transients resulting from the LOCA (see section 6.2) serve as the basis for a containment design pressure of 54 psig.

The design pressure will not be exceeded during any subsequent long term pressure transients caused by the combined effects of heat sources. These effects will be overcome by the combination of safety features and heat sinks.

The temperature gradient through the containment wall during operating conditions and during LOCA is shown in figure 3.8-15. The variation of temperature with time and the expansion of the liner plate with temperature are considered in determining the thermal stresses associated with the LOCA.

g. Operating Thermal Loads:

The temperature gradient through the containment wall during normal operating condition is shown in figure 3.8-15. For this condition, a low mean winter temperature of 20°F is assumed at the exterior surface of the concrete, and an operating temperature of 120°F on the interior surface of the concrete. For normal operation or any other long term period, the concrete temperature shall not exceed 150°F, except for local areas, which are allowed to have increased temperatures not to exceed 200°F, such as heat affected zones around penetrations. Thermal loads caused by transient wall temperatures during a prolonged shutdown are also considered in the design.

The thermal loads caused by the expansion and contraction of piping and equipment are considered in the design whenever applicable.

h. Wind and Tornado Loads:

The wind loadings and tornado loadings are discussed, respectively, in subsections 3.3.1 and 3.3.2.

The containment is designed to withstand the effects of these wind and tornado loadings, and to provide protection against tornado missiles.

The structure is analyzed for tornado loadings not coincident with the SSE.

i. External Pressure Loads:

A pressure of 3 psi from the exterior to the interior of the containment is assumed and applied as an external pressure on the containment.

j. Hydrostatic Loads:

Lateral hydrostatic pressure resulting from ground or flood water, as well as buoyant forces resulting from the displacement of ground or flood water by the structure, are accounted for in the design.

The water levels considered are:

Ground water, elevation 125 ft.
Flood water, elevation 144.2 ft.

k. Test Loads:

Upon completion of construction, the containment and its penetrations are tested at 115 percent of the design pressure. This pressure is considered in the design.

2. Loading Combinations

In general, two types of loading cases are considered in the design of the containment.

a. The design loading case for which the working stress method is used.

b. The factored loading case for which the ultimate stress method is used.

The following terms are used in the loading combination equations:

C = Required capacity of the containment to resist factored loads.

ϕ = Capacity reduction factor (defined in subsection 3.8.1.3.1)

D = Dead loads of containment, interior structures, and equipment, plus any other permanent contributing loads.

E = 1/2 Safe-shutdown earthquake load.

E'= Safe Shutdown Earthquake load.

F = Prestress load.

H = Force on structure due to thermal expansion of pipes under operating conditions.

L = Appropriate live load.

P = Design accident pressure load.

R = Force or pressure on structure due to rupture of any one pipe.

T_a= Thermal loads due to the accident temperature gradient through the wall, based on a temperature corresponding to the unfactored design accident pressure.

T_o= Thermal loads due to the normal operating temperature gradient through the walls.

T_s= Thermal loads due to transient wall temperatures over a prolonged shutdown. (20°F at outside face, 70°F at center, 212°F at inside face.)

W = Wind load.

W_t= Tornado load.

a. Design Loading Case:

In the basic working stress design, the containment is designed for the following loading combinations:

1) $D + F + L$ (construction case)

2) $D + F + L + T_o + E$ (or W) (operating case)

3) $D + F + L + P + T_a$ (design accident case)

4) $D + F + L + T_S + E$ (or W) (prolonged shutdown case)

5) $D + F + L + 1.15P$ (test case)

b. Factored Loading Case:

This loading case utilizes the capacity of the structure to verify its ability to withstand loading combinations in excess of the maximum that could be expected under the LOCA conditions. The design of the containment satisfies the following loading combinations:

1) $C = 1/\phi\ (1.0D + 1.5P + 1.0T_a + 1.0F)$

2) $C = 1/\phi\ (1.0D + 1.25P + 1.0T_a + 1.25H + 1.25\ E\ (\text{or } 1.25W) + 1.0F)$

3) $C = 1/\phi\ (1.0D + 1.25H + 1.0R + 1.0F + 1.25E\ (\text{or } 1.25W) + 1.0T_o)$

4) $C = 1/\phi\ (1.0D + 1.25H + 1.0F + 1.0W_t + 1.0T_o)$

5) $C = 1/\phi\ (1.0D + 1.0P + 1.0T_a + 1.0H + 1.0E' + 1.0F)$

6) $C = 1/\phi\ (1.0D + 1.0H + 1.0R + 1.0E'$
$1.0F + 1.0T_o)$

Equation 1 ensures that the containment will have the capacity to withstand pressure loadings at least 50 percent greater than those calculated for the postulated loss-of-coolant accident alone.

Equation 2 ensures that the containment will have the capacity to withstand loadings at least 25 percent greater than those calculated for the postulated loss-of-coolant accident with a coincident 1/2 SSE.

Equation 3 ensures that the containment will have the capacity to withstand earthquake loadings 25 percent greater than those calculated for the 1/2 SSE coincident with the associated rupture of any attached piping.

Equation 4 ensures that the containment will have the capacity to withstand a tornado loading.

Equations 5 and 6 ensure that the containment will have the capacity to withstand either the postulated loss-of-coolant accident or the rupture of any attached piping coincident with the SSE.

3.8.1.3.1 Capacity Reduction Factors

The capacities of all load carrying structural elements are reduced by capacity reduction factors (ϕ) as given below. These factors provide for the possibility that small adverse variations in material strengths, workmanship, dimensions, control, and degree of supervision, while individually within the required tolerances and the limits of good practice, occasionally may combine to result in undercapacity.

Capacity reduction factors:

ϕ = 0.90 for concrete in flexure.

ϕ = 0.85 for tension, shear, bond, and anchorage in concrete.

ϕ = 0.75 for spirally reinforced concrete compression members.

ϕ = 0.70 for tied compression members.

ϕ = 0.90 for fabricated structural steel.

ϕ = 0.90 for mild reinforcing steel (not prestressed in direct tension excluding splices).

ϕ = 0.90 for mild reinforcing steel with welded or mechanical splices. (For lap splices, ϕ = 0.85 as above for bond and anchorage.)

ϕ = 0.95 for prestressed tendons in direct tension.

3.8.1.3.2 Prestress Losses

In accordance with ACI 318-63, the design provides for the following prestress losses:

a. Slip at anchorage.

b. Elastic shortening of concrete.

c. Creep of concrete.

d. Shrinkage of concrete.

e. Relaxation of steel stress.

f. Frictional loss due to intended or unintended curvature in the tendons.

The following relationships and assumed values have been used in conjunction with these categories of prestress losses:

a. Slip at Anchorage

There is no loss for hoop, dome, or vertical tendons due to slippage at anchorage since the buttonheaded BBRV system has physical characteristics that eliminate this loss. Lift off readings will be made to confirm that seating losses are negligible.

b. Elastic Shortening of Concrete

$$\Delta\sigma_s = \frac{\sigma_c x E_s}{2E_c}$$

where $\Delta \sigma_s$ is the change in tendon stress due to elastic shortening, σ_c is the maximum concrete stress for that general area, E_S is the modulus of elasticity of the steel, and E_c is the modulus of elasticity of the concrete at the time of stressing.

c. Creep of Concrete

$$\text{Creep loss} = 317 \times 10^{-6} \frac{in}{in} x \frac{\sigma_c}{\sigma_c'} x E_s$$

where σ_c is the maximum concrete stress for that general area, σ_c' is the concrete stress level at which the value of 317 x 10^{-6} in/in was determined, and ES is the modulus of elasticity of the steel.

d. Shrinkage of Concrete

Shrinkage loss = 170 x 10^{-6} in/in x E_S where E_S is the modulus of elasticity of the steel.

e. Relaxation of Steel Stress

The relaxation loss was assumed to be 8 1/2 percent of the seating stress of the tendon.

f. Frictional Losses

The curvature friction coefficient, μ, and the wobble friction coefficient, K, which were used are 0.08 and 0.0003, respectively.

The following tabulation shows the magnitude of losses and the final effective prestress at the end of 40 years for a typical dome, hoop, and vertical tendon: [(a)]

	DOME (ksi)	HOOP (ksi)	VERTICAL (ksi)
Average overstress	179.8	192.2	176.7
Average friction loss	11.3	36.0	6.4
Seating loss	0.0	0.0	0.0
Average seating stress	172.4	173.4	172.3
Elastic shortening loss	5.0	5.0	2.7
Creep loss	7.6	7.6	4.1
Shrinkage loss	4.9	4.9	4.9
Relaxation loss	14.7	14.7	14.6
40-year minimum effective stress	136.3	124.0	144.0

a. The operating licenses for both FNP units have been renewed and the original licensed operating terms have been extended by 20 years. Containment tendon prestress was evaluated as a time-limited aging analysis in accordance with 10 CFR 54.21 during the license renewal process. Continued trending of prestress losses for tendons will ensure that minimum effective prestress for typical dome, hoop, and vertical tendons will remain above the required values for the period of extended operation (see chapter 18, subsection 18.4.3).

3.8.1.4 Design and Analysis Procedures

The containment is analyzed for various loading combinations, considering the values of individual loads that generate the most significant stress condition for each component and member of the structure.

The critical areas for analysis are as follows:

a. The intersection between cylinder wall and base slab.

b. Ring girder.

c. Behavior of the base slab relative to an elastic foundation.

d. Transient temperature gradients in the steel liner plate and concrete.

e. Tendon anchorage zones.

Classical theory, empirical equations, and numerical methods are applied as necessary for the analysis of structural elements. They are described in Section 7 of BC-TOP-5.

The design methods incorporate several phases as described in Section 6 of BC-TOP-5. Improved assumptions as to material properties including the effects of creep, shrinkage and cracking on concrete are used in design. Analysis and design of tendon anchorage zones and reinforcement in buttresses are discussed in BC-TOP-5, BC-TOP-7, and BC-TOP-8. The method of analyzing the effects of penetrations, the thickening of walls, reinforcements and embedments, etc., are discussed in Section 7 of BC-TOP-5. The design of the liner and its anchorage system is covered in BC-TOP-1 and BC-TOP-5. Information on analyses for computation of seismic loads is provided in section 3.7.

3.8.1.4.1 Analytical Techniques

The analysis of the containment consists of two parts: the axisymmetric analysis and the non-axisymmetric analysis. The axisymmetric analysis is performed by utilizing a finite element computer program for combinations of the individual loading cases of dead, live, temperature, pressures, and prestress loads. The axisymmetric finite element representation of the containment assumes that the structure is axisymmetric. This does not account for the buttresses, penetration, brackets, and anchors. These items, together with the lateral loads due to earthquakes, winds, tornados, and various concentrated loads, are considered in the non-axisymmetric analysis.

1. Axisymmetric Analysis

   The containment is considered an axisymmetric structure for the overall analysis. Although there are deviations from this ideal shape, the deviations are usually localized and can be handled by special analyses; hence, axisymmetric analyses are considered acceptable.

The axisymmetric analysis of the containment is performed by Bechtel's "FINEL" computer program (CE 316-4) based on the finite element method. Because of program limitations, the upper and lower portions of the containment are analyzed separately, to permit the use of a greater number of elements for those areas of the structure which are of major concern, i.e., the ring girder and the haunch connecting the cylindrical shell to the base slab.

The entire concrete structure is modeled by continuously interconnected elements. The geometry of the mesh allows for the locations and shapes of narrow elements representing reinforcing steel superimposed on the corresponding concrete elements.

The liner plate is simulated by a layer of elements attached to the interior surfaces of the concrete structure.

The finite element mesh of the structure is extended into the foundation to account for the elastic nature of the foundation materials and its effect on the behavior of the base slab. The tendon access gallery is designed as a separate structure.

The use of the finite element analysis permits accurate determination of the stress pattern at any location on the structure.

The finite element mesh for axisymmetric loads is shown on figures 3.8-16 and 3.8-17.

2. Non-Axisymmetric Analysis

The non-axisymmetric analyses of the containment include the following:

a. Seismic loadings.

b. Wind and tornado loadings.

c. Buttress and tendon anchorage zones.

d. Large penetration openings.

e. Small penetration openings.

f. Non-axisymmetric internal structure and equipment.

a. Seismic Loadings

The analysis of the non-axisymmetric seismic loadings is performed by Bechtel's "Axisymmetric Shell and Solid Computer Program" (ASHSD), described in appendix 3F.

Details of the analysis are described in section 3.7.

b. Wind and Tornado Loadings

Wind and tornado loadings are discussed in section 3.3.

c. Buttress and Tendon Anchorage Zones

The containment has three buttresses. At each buttress, two out of any group of three-hoop tendons are spliced by anchoring on the opposite faces of the buttress, with the third tendon continuous through the buttress.

Between the opposite anchorages in the buttress, the compressive forces exerted by the spliced tendons are twice as large as elsewhere on the buttress. This value, combined with the effect of the tendon which is continuous throughout the buttress, is 1.5 times the prestressing forces acting outside the buttress. The thickness of the buttress is approximately 1.5 times the thickness of the wall. Hence, the hoop stresses and strains, as well as the radial displacements, may be considered as being nearly constant all around the structure.

The vertical stresses and strains, caused by the vertical post-tensioning become constant a short distance away from the anchorages because of the stiffness of the cylindrical walls. The effects of the buttresses on the overall behavior of the containment are negligible under dead and prestressing loads. The stresses and strains remain nearly axisymmetric despite the presence of the buttresses.

The design of the tendon anchorage zones is based on two test programs conducted by Bechtel to demonstrate the adequacy of several reinforcing patterns for use in anchorage zone concrete in the base slab, buttresses and ring girder. These tests have been undertaken to develop a more efficient design to reduce reinforcement congestion and thereby facilitate the placement of high quality concrete around the tendon anchorages. The test programs are as follows:

1. A full scale model of a simulated containment buttress containing several patterns of reinforcememt and types of tendon anchorages was constructed and tested. A detailed description of the test is presented in Bechtel Topical Report BC-TOP-7.

2. Two large concrete test blocks containing two patterns of reinforcement with different proportions of reinforcing bars were constructed and tested. A detailed description of the test is presented in Bechtel Topical Report BC-TOP-8.

The test results demonstrated satisfactory performance of the test anchorages. The design of the tendon anchorage zones is based on the results and recommendations of these tests.

In addition, the local stress distribution in the immediate vicinities of the bearing plates has been investigated using the following methods.

1. The Guyon Equivalent Prism Method: This method is based on the experimental photoelastic results and the equilibrium considerations of homogeneous and continuous media. It also considers the relative bearing plate dimensions of the anchorages.

2. The experimental test data presented by S. J. Taylor at the March 1967 London Conference of the Institute of Civil Engineers: These data are used to evaluate the effect of the biaxial stresses at the anchorages, including the effects of the trumpets welded to the bearing plates.

3. Leonhardt's Formula for determining the bursting forces in the anchorage zone of a prestressed concrete member.

d. Large Penetration Openings

Large penetrations are defined as those having an inside diameter equal to or greater than 2.5 times the containment wall thickness. The equipment hatch falls into this category.

The stresses at the opening are predicted by Bechtel's Computer program "SAP" (CE 779), which is capable of performing a static analysis of linear elastic three dimensional structures utilizing the finite element method.

The points delineating the outermost boundaries of the analytical model are located at two penetration diameters beyond the edges of the opening, so that the behavior of the model along the boundaries is compatible with that of the undisturbed cylindrical wall.

Typical details of the equipment hatch are shown on drawing D-176145. Figure 3.8-30 shows the equipment hatch boundary lines. Figure 3.8-31 shows the finite element mesh used for the analyses of the equipment hatch. Figures 3.8-32 through 3.8-35 show typical iso stress plots for the load combination D + F + 1.5P + T_a.

Typical results of the analyses of the equipment hatch are given in table 3.8-4.

e. Small Penetration Openings

Small penetration openings are defined as those having an inside diameter less than 2.5 times the containment wall thickness.

The determination of the stresses at the openings due to applied moments and forces are based on a paper by Eringen et al.[1]

Results of these analyses show the stresses to be well within the allowable limits. Typical details of small penetrations are shown on drawings D-176145 and D-176151.

f. Non-Axisymmetric Internal Structure and Equipment

The stresses due to the non-axisymmetric portion of the internal structure and equipment are combined with the stresses resulting from the other loading combinations described in subsection 3.8.1.3.

3.8.1.4.2 Steel Liner Plate and Penetrations

The steel liner plate and penetrations are designed to serve as the leakage barrier for the containment. Typical details for the liner plate and penetrations are shown in figures 3.8-1, 3.8-2, and drawings D-176145 and D-176151.

The design of the liner plate considers the composite action of the liner and the concrete structure and includes the transient effects on the liner due to temperature changes during construction, normal operation, and the loss-of-coolant accident. The changes in strains to be experienced by the liner due to these effects, and those at the pressure testing of the containment, are considered.

The stability of the liner is achieved by anchoring it to the concrete structure. At all penetrations, the liner is thickened to reduce stress concentration. The thickened plate is also anchored to the concrete.

For a detailed description of the liner plate stability, see appendix 3I.

Insert plates are provided in the liner to transfer concentrated loads to the wall, slab, and dome of the containment. Examples of these concentrated loads are polar crane brackets and floor beam brackets. A typical bracket detail is shown in figure 3.8-1.

The topical Report BC-TOP-1, "Consumers Power Company Palisades Nuclear Power Plant, Containment Building Liner Plate Design Report," October 1969, prepared by Bechtel Power Corporation, and submitted to the Nuclear Regulatory Commission, constitutes the basic approach used in the design of the liner plate.

There are minor differences in the design of the Farley Nuclear Plant from that presented in the topical report. They are listed below.

a. The 1/4-in. liner plate material is ASTM A-285, Grade A, with a specified yield stress of 24,000 psi, instead of ASTM A-442, which has a specified yield stress of 30,000 psi. The lower yield stress would only tend to decrease the loads on the anchors, as indicated in Section 3.4 of the report.

b. The welding of the stiffeners is 3/16 - 6 x 12 rather than 3/16 - 4 x 12. This does not invalidate the analysis, since the spring constants used are similar.

c. The stiffeners on the thickened plates are not welded with a double fillet weld. Instead, the 3/16 - 6 x 12 welding is used for all stiffeners.

d. A self-supporting dome is used rather than the truss supported dome. Details of the dome are shown in figure 3.8-2.

3.8.1.4.3 Description of Computer Programs

Computer programs used in the design and analysis of the containment are described in appendix 3F.

3.8.1.5 <u>Structural Acceptance Criteria</u>

The fundamental acceptance criterion for the containment is the successful completion of the structural integrity test, with measured responses within the limits predicted by analyses. The limits are predicted based on test load combinations and code allowable values for stress, strain, or gross deformation for the range of material properties and construction tolerances specified as described in Topical Report BC-TOP-5. In this way the margins of safety associated with the design and construction of the containment are, as a minimum, the accepted margins associated with nationally recognized codes of practice.

The structural integrity test is planned to yield information on both the overall response of the containment and the response of localized areas, such as major penetrations and buttresses, which are important to its design functions.

The design and analysis methods, as well as the type of construction and construction materials, are chosen to allow assessment of the structure's capability throughout its service life. Additionally, surveillance testing provides further assurances of the structure's continuing ability to meet its design functions.

Tables 3.8-2 and 3.8-3 show the calculated stresses and strains, respectively, as well as the allowables, taken from critical sections of the containment structure. The ratios of the allowable stresses and strains to the calculated stresses and the strains yield the margins of safety at selected critical sections. Deviations in allowable stresses for the design loading conditions in the working stress method are permitted if the factored load capacity requirements are fully satisfied.

For the margins of safety related to major local areas of the containment, such as the equipment hatch, refer to table 3.8-4. The ratio of the allowable stress to the calculated stress, yields the margins of safety. This information, together with the test information documented in BC-TOP-7 and BC-TOP-8, permits the assessment of the margins of safety for anchorage zones.

The effect of three dimensional stress/strain fields on the behavior of the structure has been considered in the "FINEL" computer program.

3.8.1.6 <u>Materials, Quality Control, and Special Construction Techniques</u>

The following basic materials are used in the construction of the containment structure:

a. Concrete

For: Tendon access gallery		f'_c (psi) = 5,000
Base slab		f'_c (psi) = 5,000
Cylindrical wall and dome		f'_c (psi) = 5,500

b. Reinforcing steel

Deformed bars	ASTM A-615 Grade 60	f_y (psi) = 60,000
Spiral bars	ASTM A-82	f_y (psi) = 70,000

c. Structural and miscellaneous steel

Rolled shapes, bars and plates	ASTM A-36	f_y (psi) = 36,000
High strength bolts	ASTM A-325 or A-490	
Stainless steel	ASTM A-240 Type 304	

d. Containment steel liner plate and penetration sleeves

1/4-in. liner plates	ASTM A-285 Grade A	f_y (psi) = 24,000
Insert plates	ASTM A-516 Grade 70	f_y (psi) = 38,000
Penetration sleeves pipes	ASME SA-333 Grade 6	f_y (psi) = 32,000
Plates	ASME SA-516 Grade 70	f_y (psi) = 38,000

e. Post-tensioning system

Prestressing wires	ASTM A-421-65 Type BA	f'_s (psi) =240,000
Bearing plates	ASTM A-36-70a	f_y (psi) = 36,000
Anchor heads	HR 4142 or 4140 Alloy steel	
Bushing	HFSM 4142 Tubing	
Shims	ASTM A 36-70a	f_y (psi) - 36,000 0.40/0.50 carbon sheet steel
Sheathing		22-gauge galvanized corrugated tubing

Materials and their quality control requirements are described in the following subsections.

3.8.1.6.1 Reinforced Concrete

a. Concrete

All concrete work is done in accordance with ACI 318-63, "Building Code Requirements for Reinforced Concrete", and ACI 301-66, "Specifications for Structural Concrete for Buildings", except as otherwise stated herein, or in the appropriate job specifications or design drawings.

The concrete is a dense, durable mixture of sound coarse aggregates, fine aggregates, cement, and water. In some areas, fly ash is substituted for portions of cement used in the concrete. Admixtures are added to improve the quality and workability of the plastic concrete during placement and to retard the set of concrete. The sizes of aggregates, water reducing additives, and slumps are selected to maintain low limits on shrinkage and creep.

b. Aggregates

Aggregates comply with ASTM C-33-69, "Specifications for Concrete Aggregates". Acceptability of the aggregates is based on the initial tests listed in table 3.8-5.

Certain user tests, as indicated in table 3.8-5, are performed on the aggregates used in every 5,000 cubic yards of concrete produced.

In addition, a daily inspection control program is carried out during construction to ascertain the consistency in the potentially variable characteristics such as gradation and organic content.

c. Cement

Cement is Type II, low-alkali cement in accordance with ASTM C-150-68, "Specification for Portland Cement", and is tested to comply with the requirements of ASTM C-114-67, "Chemical Analysis of Hydraulic Cement". The inspection and testing of cement, in addition to the initial tests performed by the cement manufacturer, are indicated in table 3.8-6.

User tests are performed on the cement used in every 5,000 cubic yards of concrete produced. In addition, ASTM Tests C-191-65 and C-109-64 are performed periodically during construction to check the environmental effects of storage on cement characteristics.

d. Fly Ash

Fly ash conforms to ASTM C-618-68T Class F, "Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete," and is tested to comply with the requirements of ASTM C-311-68, "Sampling and Testing Fly Ash for Use as an Admixture in Portland Cement Concrete."

The producer is required initially to test and then submit data on each lot of fly ash furnished. User tests, as indicated in table 3.8.7, are performed for each 2,500 cubic yards of concrete produced. In addition, periodic tests in accordance with ASTM C-109-64 are performed during construction to check the environmental affects of storage on fly ash.

e. Water

Water used in mixing concrete is free from injurious amounts of acid, alkali, organic matters, and other deleterious substances as determined by AASHO-T-26. In addition, user tests are performed quarterly on the mixing water after the initial tests.

The acceptance criteria for the mixing water are as follows:

Criteria	Percentage (Max.)
Alkalinity in terms of Calcium carbonate	0.025
Total organic solids	0.025
Total inorganic solids	0.050
Total chlorides	0.025
Total sulfates as SO_4	0.025

f. Admixtures

The selected water reducing agent MBHC, manufactured by the Masters Builders Company, possesses a shrinkage reduction effect similar to the types prescribed by ASTM C-494-68, "Specifications for Chemical Admixtures for Concrete."

An air entraining agent, Vinsol Resin, also manufactured by the Masters Builders Company, is added to the concrete mix to increase workability.

Admixtures containing chlorides are not used.

g. Concrete Mix Design

Concrete mixes are designed in accordance with ACI 613-54, "Recommended Practice for Selecting Proportions for Concrete," using materials qualified and accepted for this work. Only concrete mixes meeting the design requirements specified for the structures are used.

Trial mixes are tested in accordance with the applicable ASTM specifications as indicated below.

ASTM	**Test**
C-39-66	Compressive strength of molded concrete cylinders.
C-143-69	Slump of Portland Cement concrete.
C-192-69	Making and curing concrete test specimens, in the laboratory.
C-231-68	Air content of freshly mixed concrete by the pressure method.
C-232-58	Bleeding of concrete.

Concrete test cylinders are cast from the mix proportions selected for construction, to determine the following properties:

1. Compressive strength (ASTM C-39-66).

2. Thermal diffusivity (ASTM C-342-67).

3. Autogenous shrinkage (ASTM C-342-67).

4. Thermal coefficient of expansion (ASTM-342-67).

5. Modulus of elasticity and Poisson's ratio (ASTM-469-65).

6. Uniaxial creep (ASTM C-512-69).

h. Concrete Testing

During construction, concrete is sampled and tested to ascertain conformance to the specifications. Concrete samples are taken from the mix in accordance with ASTM C-172-68, "Sampling Fresh Concrete," and six cylinders, three sets of two cylinders each, are prepared from each sampling and cured in accordance with ASTM C-31-69, "Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field."

The tests consist of the following:

1. Determination of air content in accordance with ASTM C-231-68.

2. Determination of unit weight in accordance with ASTM C-138-63.

3. Slump test in accordance with ASTM C-143-69.

4. Compressive strength test in accordance with C-39-66.

5. Determination of temperature.

The frequency and extent of these tests are as follows:

One partial test for every 35 cubic yards mixed at the batch plant, consisting of a determination of slump and air content only.

One complete test for every 100 cubic yards mixed at the batch plant.

One complete test for every 210 cubic yards discharged from the truck.

One partial test for every 210 cubic yards discharged into the forms, consisting of a determination of temperature and slump only. This partial test is performed at the forms, and is required only when concrete is discharged into the forms using pumps or conveyors.

The tests conducted at the truck discharge are performed on the concrete previously tested at the batch plant. The tests conducted at the forms are performed on concrete previously tested at the truck discharge. Hence, the concrete tested at the forms is previously tested at the truck discharge and also at the batch plant.

In addition, all concrete discharged from the truck is visually examined by an experienced inspector during the course of discharge from the truck and samples are obtained and tested whenever the concrete appears to have excessive slump.

The locations at which the sampled concrete is placed are marked.

i. Concrete Placement

All concrete for the containment base slab, cylindrical wall, dome and all walls exceeding 2 1/2 ft in thickness has a placing temperature of not less than 40°F nor more than 70°F. All other concrete including walls and elevated slabs has a placing temperature of not less than 40°F nor more than 85°F.

If it is necessary to keep the temperature of the concrete from exceeding the above maximums, approved measures for reducing the temperature of the concrete are employed, such as:

1. Cooling the mixing water.

2. Cooling the aggregates.

3. Shading the materials and facilities from direct rays of the sun.

4. Insulating water supply lines.

5. Introducing flaked ice into the mix.

In general, all procedures for hot weather concreting are in accordance with ACI-605-59.

During cold weather, frozen materials or materials containing ice are not used. To protect the new poured concrete from freezing, enclosures or coverings are placed over the concrete. Whenever the outdoor temperature is below 40°F, steam or heat is introduced into the enclosures or coverings to maintain a temperature of not less than 50°F for at least 72 hours, or for as much longer as is necessary to insure proper rate of curing the concrete. The enclosures or coverings used in connection with curing remain in place and intact for at least 24 hours after the artificial heating is discontinued.

In general, all procedures for cold weather concreting are in accordance with ACI-306-66.

j. Bonding of Concrete Between Lifts

Horizontal construction joints are prepared for receiving the next lift by either wet sandblasting, by cutting with an air water jet, or by bush hammering.

When wet sandblasting is employed, it is continued until all laitance, coatings, stains, and other foreign materials are removed. The surface of the concrete is washed thoroughly to remove all loose materials.

When air water jet cutting is used, it is performed after initial set has taken place but before the concrete has taken its final set. The surface is cut with a high pressure air water jet to remove all laitance and to expose clean, sound aggregates, but not so as to undercut the edges of the larger particles of the aggregates. After cutting, the surface is washed and rinsed as long as there is any trace of cloudiness of the wash water. When it is necessary to remove accumulated laitance, coatings, stains, and other foreign materials, wet sandblasting is used before placing the next lift, to supplement air water jet cutting.

The horizontal surface is wet immediately before the concrete is placed.

Surface set retardant compounds are not used.

3.8.1.6.2 Reinforcing Steel

All reinforcing steel conforms to ASTM A-615-68, "Deformed Billet-Steel Bars for Concrete Reinforcement," Grade 60. Spiral reinforcing steel conforms to ASTM A-82-66, "Cold Drawn Steel Wire for Concrete Reinforcement,"

Mill test reports are obtained from the reinforcing steel supplier for each heat of steel, to ensure that the physical and chemical properties of the steel are in compliance with the ASTM specifications. In addition, user tests consisting of tension and bend tests, in accordance with ASTM A-615-68, are performed to supplement the standard mill tests. One tension test and one bend test are required for each 50 tons of each bar size from each heat of steel, with the exception that bend tests are not performed on No. 14 and No. 18 bars.

All user tests are performed on full size bars.

Bars No. 11 and smaller are generally lap spliced in accordance with ACI 318-63. Bars No. 14 and No. 18 are Cadweld spliced exclusively.

Splicing reinforcing bars by welding is not done.

Procedures for splicing reinforcing bars using the Cadweld process are defined in appendix 3C.

3.8.1.6.3 Structural and Miscellaneous Steel

All structural and miscellaneous steel conforms to the following ASTM specifications:

Rolled shapes, bars, and plates	A-36-69
High strength bolts	A-325 or A-490
Stainless steel	A-240, Type 304

Mill test reports are obtained for all materials used with the exceptions of hand rails, toe plates, kick plates, stairs, and ladders.

Detailing, fabrication, and erection of the structural and miscellaneous steel are in accordance with the Manual of Steel Construction, 1969 edition.

Welding is done in accordance with AWS D 2.0, "Specification for Welded Highway and Railway Bridges."

3.8.1.6.4 Containment Steel Liner Plate and Penetration Sleeves

Since the liner plate is not a pressure vessel and its function is to serve as a leaktight membrane, the design, construction, inspection, and testing of the liner plate are not covered by any recognized codes or standards. However, for Unit 1, components of the liner plate that must resist the full containment design pressure, such as the penetrations, personnel locks and equipment hatch, are designed, fabricated, constructed, and tested to meet the requirements of Subsection B of Section III, Nuclear Vessels of the 1968 ASME Code. For Unit 2, components that must resist the full containment design pressure are designed, fabricated, constructed, and tested to meet requirements of Subsection NE of Section III of the 1971 ASME Code.

Since the principal stresses of the liner due to thermal expansion are in compression, and no significant tensile stresses are expected from the internal pressure loadings, special nil ductility transition temperature requirements are not applied to the liner plate materials. However, for Unit 1, all materials for the liner components which must resist tensile stresses resulting from internally applied pressure, such as the penetration sleeves, are impact tested in accordance with the requirements of Article 12 of Section III, Nuclear Vessels, of the 1968 ASME Code. For Unit 2, impact tests are performed to paragraph NE 2300 of Section III of the 1971 ASME Code.

All welding procedures, welders, and welding operators used in the fabrication and erection of the steel liner plate and penetration sleeves are qualified by tests in accordance with Section IX of the 1968 ASME Code, for Unit 1, and to the 1971 ASME Code for Unit 2.
The quality control procedures for welding and for non-destructive inspection of welds are defined in Appendix 3-G.

3.8.1.6.5 Post-Tensioning System

The prestressed, post-tensioning system selected for the containment is an Inland-Ryerson BBRV buttonhead system.

1. Tendons

   The tendons are composed of stabilized, low relaxation wires of 1/4-in. diameter with a minimum tensile strength of 240,000 psi in accordance with ASTM A-421-65, Type BA. The pertinent features of the tendons are as follows:

Number of wires	170
Ultimate tensile capacity (kips)	2000
End anchorage	Buttonheads

   Sampling and testing of the tendon material conform to ASTM A-421-65.

2. Anchorages

   The basic performance requirements for the end anchors of the tendons are stated qualitatively by the Seismic Committee of the Prestressed Concrete Institute and published in their Journal of June, 1966, as follows:

> "All anchors of unbonded tendons should develop at least 100 percent of the guaranteed ultimate strength of the tendons. The anchorage gripping should function in such a way that no harmful notching would occur on the tendons. Any such anchorage system used in earthquake areas must be capable of maintaining the prestressing force under sustained and fluctuating loads and the effect of shock. Anchors should also possess adequate reserve strength to withstand any overstress to which they may be subjected during the most severe probable earthquake. Particular care should be directed to accurate positioning and alignment of end anchors."

The end anchors used are capable of developing 100 percent of the minimum tensile strength of the tendons. Furthermore, the end anchors are capable of maintaining integrity for 500 cycles of loads corresponding to an average axial stress variation between 0.7 and 0.75 f'_s, at a repetition rate of one cycle in 0.1 second. This requirement sets the minimum acceptable limits on fatigue effects due to notching by the end anchor and tendon performance in response to earthquake loads.

The number of cycles caused by the earthquake loads is predicted as only 30 of a total of 100, resulting from using all the strong ground motions which exceed one half of the peak ground motion of the earthquake. However, it is conservatively set at 500.

The stress variations due to the earthquake motion alone are predicted as being 10 percent of the total of the estimated stress variations of 0.04 f'_s. The estimated 0.04 f'_s stress variations, in turn, result from the combinations of earthquake, wind, and incident loadings. Analyses made during the investigation include consideration of tendon excitation, both parallel and perpendicular to the tendon axis.

The anchorage assemblies, including the bearing plates, are capable of transmitting the ultimate loads of the tendons into the structure without brittle fracture at an anticipated lowest service temperature of 20°F.

3. Sheathing

Sheaths for the tendons are classified as concrete forms and are not subjected to any Standard codes. They provide a void in the concrete in which the tendons are installed, stressed and greased after the concrete is placed.

The sheaths are fabricated from 22 gauge, galvanized and corrugated ferrous metal tubing, which has an internal diameter of 4-3/4 in. clear of corrugations. Couplers are provided at all field splices and sealed by tapes.

After sheathing installation, and prior to concrete placement, the sheathing is surveyed to assure accurate alignment. An inspection is also performed to ascertain that all sheaths are continuous and unblocked by obstructions.

Before installation of the tendons, the sheathing is cleaned to remove all water and debris.

Vent tubing and temporary valves are provided to permit drainage at all low points. Splash caps at the ends of all sheaths, to prevent concrete and laitance from entering into the sheaths during construction, are provided.

4. Corrosion Protection

Suitable atmospheric corrosion protection is maintained for the tendons from the point of manufacture to the installed locations. The atmospheric corrosion protection provides assurance that the tendon integrity is not impaired due to exposure to the environment.

Prior to shipment, a thin film of petroleum oil based rust inhibitor, Visconorust 1601 and 1702 Amber, manufactured by the Viscosity Oil Company, is applied to the tendons in accordance with the manufacturer's instructions. After the tendons are installed and stressed, the interior of the sheathing is pumped full of a modified, thixotropic, refined petroleum oil based product to provide corrosion protection. The tendon and anchors are also encapsulated by gasketed end caps which are filled with the corrosion protection material and sealed against the bearing plates.

Testing of the permanent corrosion protection material indicates that there are no significant amounts of chlorides, sulfides, or nitrates present. However, to further verify the chemical composition of the filler material, test samples are obtained from each shipment and analyzed as follows:

a. Water soluble chlorides (Cl) are determined in accordance with ASTM D512-67 with a limit of accuracy of 0.5 ppm.

b. Water soluble nitrates (NO3) are determined by the Water and Sewage Analysis Procedure of the Hach Chemical Company, Ames, Iowa, or by ASTM D-992 Brucine Method.

c. Water soluble sulfides (S) are determined in accordance with American Public Health Association (APHA) standards with a limit of accuracy of 1 ppm. The APHA Standard methods (Methylene Blue procedure) or the Hach Chemical Company method are used.

Acceptance criteria of the corrosion protection materials are as follows:

1. As shipped from the point of manufacture:

a)	Chlorides	-	2 ppm max.
b)	Nitrates	-	4 ppm max.
c)	Sulfides	-	2 ppm max.

2. Onsite user test

   a) Chlorides - 5 ppm max.
   b) Nitrates - 5 ppm max.
   c) Sulfides - 5 ppm max.

5. Prestressing Sequences

   The criteria of the prestressing sequence are based on the design requirements to limit the membrane tension in concrete to 1.0 $\sqrt{f'_c}$ to minimize unbalanced loads or differential stresses in the structure.

   Prestressing begins after the concrete in the wall and the dome has reached the specified f'_c (5500 psi). The construction opening will be closed prior to prestressing. All tendons are tensioned from both ends. Each hoop tendon wraps approximately 240 degrees around the containment. Three hoop tendons forming one complete band are tensioned simultaneously and every other band is stressed starting from the top of the base slab to approximately 30 ft below the bottom of the ring girder. Vertical tendons are placed in two layers spaced equally on either side of the containment wall center line. Three vertical tendons, approximately 120 degrees apart, are tensioned simultaneously; then every fourth tendon is stressed, taking three at a time. Three groups of dome tendons are oriented at 60 degrees to each other and are anchored at three different levels in the ring girder. Three dome tendons, one in each group, are tensioned simultaneously. The procedure for prestressing is carefully worked out with the vendor so that all the tendons proceed in such a manner that the containment structure will not be eccentrically loaded at any phase.

   Table 3.8-8 summarizes the prestressing sequences.

3.8.1.6.6 Containment Interior Coating System

Coating materials specified have been tested by their manufacturers under simulated operating and incident conditions and shipments of the coating materials are accompanied by vendor certification of compliance. Coatings will be selected from inorganic zinc and epoxy compounds that are resistant to radiation and have successfully withstood tests in an environment simulating post design basis accident conditions, including exposure to chemical sprays. Such tests are designed to ensure that the selected coatings will suffer no significant loss of adhesion or deterioration which could contribute particles capable of interfering with free flow of the emergency core cooling system. Current references for the selection of coatings include B. J. Newby, "Applicability of Conventional Protective Coatings to Reactor Containment Building," IN-1169, June 1968.

Table 6.2-37 summarizes the coating systems used and surface areas covered inside the containment.

1. Containment Steel Liner Plate Coatings

   Surface preparation of the interior (exposed) surfaces of the steel liner plate is done in the shop by abrasive blast cleaning of each plate from edge to edge in accordance with the Steel Structures Painting Council (SSPC) Specification SSPC-SP6-63, "Commercial Blast Cleaning" for inorganic zinc primer or with SSPC-SP10 "Near White Metal Blast Cleaning" for epoxy primer. The plates are then primed with one coat of Ameron Dimetcote No. 6 inorganic zinc coating or Ameron Amercoat 90 to an average dry film thickness of 3.0 mils or 5.0 mils, respectively.

   Areas damaged by welding, such as strikes, or those damaged by the removal of temporary attachments for erection, are repaired and recoated so that they are equivalent to the original conditions.

2. Containment Interior Coatings

   a. Steel Surfaces

      Carbon steel surfaces, including structural and miscellaneous steel, uninsulated piping, and equipment, which are located in areas subject to hard usage or radioactive contamination are blast cleaned in accordance with Specification SSPC-SP6-63, "Commercial Blast Cleaning" for inorganic zinc primer or with SSPC-SP10 "Near White Metal Blast Cleaning" for epoxy primer. Within 8 hours after blast cleaning, the surfaces are primed with one coat of Ameron Dimetcote No. 6 (inorganic zinc primer), or Ameron Amercoat 90 to an average dry film thickness of 3.0 mils or 5.0 mils, respectively.

   b. Concrete and Masonry Surfaces

      All concrete and masonry surfaces, including floors, wainscot, walls, columns, pilasters, and ceilings, are chemically cleaned by either caustic wash or acid etching, or by blast cleaning. A surfacer or epoxy coating is then applied over the entire area.

3. Curing

   Curing of the newly painted surface areas is as recommended by the coating manufacturer. Special precautions are taken during the curing period so that rain, snow, fog, moisture, condensation, or any foreign contaminant does not come into contact with or adhere to the freshly painted surface areas.

4. Recoating (One Additional Finish)

   The recoating of the existing coating systems (except Amercoat 90) in items 1 and 2 above will be performed with Amercoat 90 or 66 finish as per NMP-MA-011, Nuclear Coatings Program. The maximum dry film thickness of existing finish (Amercoat 66) and recoat finish (Amercoat 90 or 66) will be in accordance with FSAR tables 6.2-2 and 6B-1.

3.8.1.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance requirements for the containment include the following:

a. Pre-operational structural acceptance test.

b. Pre-operational integrated leak rate test.

c. Post-tensioning system tendon surveillance.

d. Post-tensioning system tendon end anchorage concrete surveillance.

e. Containment surfaces and steel liner plate surveillance.

f. Inservice Inspection of pressure retaining concrete surfaces and metallic liner and attachments.

3.8.1.7.1 Pre-Operational Structural Acceptance Test

In accordance with NRC acceptance criteria (NRC SER NUREG 75/034 dated May 2,1975), prior to initial fuel loading, the containment is subjected to a pressure proof test equivalent to 115 percent of the containment design pressure as required by NRC Regulatory Guide 1.18. This test demonstrates that the containment is capable of resisting the postulated accident pressure. In addition, by measuring the structural response and comparing the results with analytical predictions, the test also serves to verify the anticipated structural behavior.

Measuring systems, pressurization procedure, deflection and temperature measurements, crack pattern mapping, and data acquisition schedules for the pre-operational structural acceptance test are described in appendix 3H, "Containment Structural Acceptance Test".

3.8.1.7.2 Pre-Operational Integrated Leakage Rate Testing

Pre-operational integrated leakage rate testing of the containment is conducted in accordance with the procedures described in the Bechtel Power Corporation Topical Report BN-TOP-1, "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants", Revision 1, November 1, 1972.

3.8.1.7.3 Post-Tensioning System Tendon Surveillance

The objective of the containment tendon surveillance program during the lifetime of the plant is to provide a systematic means of assessing the continued quality of the post tensioning system. The program is intended to furnish sufficient inservice historical evidence to provide a measure of confidence in the condition and the functional capability of the system, as well as an opportunity for timely corrective measures should adverse conditions, such as excessive corrosion, be detected.

[HISTORICAL]

[Conformance to NRC Regulatory Guide 1.35, "Inservice Surveillance of the Ungrouted Tendons in Prestressed Concrete Containment Structures," is discussed in appendix 3A.

In general, the containment tendon surveillance program for the Farley Nuclear Plant is modeled after the programs now in progress and is similar to those established by the Point Beach Nuclear Power Plant Units 1 and 2 (Docket Nos. 50-266 and 50-310, respectively), Turkey Point Unit 3 (Docket No. 50-250), and the Palisades Plant Unit 1 (Docket No. 50-255), with the exception of the number of surveillance tendons involved.

To implement the surveillance program for each unit, 21 surveillance tendons are provided. Nine of these 21 tendons are redundant and are not required for the design prestress level. All tendons in the containment structure are uniformly spaced. Those to be used for surveillance will be selected after installation, tensioning, and greasing, from groups of tendons similar with respect to environmental exposure, structural function, and prestress losses, as follows:

a. Six dome tendons, randomly but representatively distributed.

b. Five vertical tendons, randomly but representatively distributed.

c. Ten hoop tendons, randomly but representatively distributed.

The program also provides for additional testing and inspection should nonconformance or unexpected conditions be detected.

The frequency of the tendon surveillance, after the structural integrity test, will be as follows:

a. One year after test.

b. Three years after test.

c. Five years after test.

d. 5 years thereafter for the life of the plant.

The containment tendon surveillance program will include the following:

a. Liftoff

Liftoff will be performed by properly calibrated jacks at the stressing ends of all 21 surveillance tendons, with simultaneous measurements of tendon elongations and jacking forces. Allowable elongations and jacking forces, temperature effects, and allowable tolerances will be established prior to the tests. The liftoff will include an unloading cycle going down to essentially complete detensioning of the tendon to identify broken or damaged wires.

b. Tendon Inspection

One dome tendon, one vertical tendon, and one hoop tendon will be completely detensioned. One wire from each of these tendons will be removed as a sample, visually inspected for corrosion and pitting, and tested for physical properties. In addition, both ends of all surveillance tendons will be inspected for any deformed buttonheads and broken wires. At each successive inspection the samples will be selected from different tendons.

c. Wire Tests

Tensile and elongation tests will be performed on at least three wire samples taken from the removed wire, one at each end and one at the middle of the wire. The test results for each wire will include, as a minimum, the yield and ultimate strength and percent of elongation under load at failure.

d. Anchorage Assembly Inspection

The tendon anchorage assemblies will be visually inspected for any deleterious conditions such as corrosion or cracks. The shim thickness will be measured. The surrounding concrete will be visually inspected for indications of abnormal material behavior.

e. Sheath Filler Inspection

A sample of sheath filler will be taken from each surveillance tendon for visual and laboratory examinations.

f. Re-Tensioning

After wire removal, the tendons will be re-tensioned to the stress level measured at the liftoff reading, and then checked by a final liftoff reading. Any change in total shim thickness will be recorded.

The acceptance criteria for containment tendon surveillance are as follows:

a. The prestress force for each tendon shall be not less than the allowable lower bound nor greater than the allowable upper bound forces at the time of the test.

b. An acceptance limit of not more than one defective tendon out of the total sample population shall be set. If one sample tendon is defective, an adjacent tendon on each side of the defective tendon shall also be tested. If both these tendons are acceptable, as defined in "a", the surveillance program shall then proceed considering the single deficiency as unique and acceptable. However, if either adjacent tendon is defective, or if more than one tendon out of the original sample population is defective, abnormal degradation of the containment structure is indicated. The commission will be notified in accordance with Regulatory Guide 1.16, "Reporting of Operating Information," except that the initial report may be made within 30 days of the completion of the tests, and the detailed report may follow within 90 days of the completion of the tests.

c. *Failure below the guaranteed ultimate strength of any one of the three tendon material sample tests will be considered an indication of abnormal degradation of the containment, and the commission will be notified as indicated above.*

Unless there is evidence of abnormal degradation of the containment, after the third surveillance the tendon sample population will be reduced from a total of 21 tendons to nine tendons. These will include three dome tendons, three vertical tendons, and three horizontal tendons, selected randomly but representatively.

Since the FNP Unit 2 containment is identical to that of Unit 1, and since these two units are located on the same site and are built by the same constructor in the same manner at the same time, the surveillance program for Unit 2 consists of visual examination of the same numbers and types of tendons as those of Unit 1, to the extent practical, without dismantling the load-bearing components of the anchorage.

3.8.1.7.4 Tendon End Anchorage Concrete Surveillance

Surveillance will be performed on the concrete surfaces surrounding some tendon anchorages. The locations of these concrete surfaces will be designated prior to the pre-operational structural acceptance test, on the following basis:

a. *Hoop Tendon Anchorage*

Four locations on one buttress and one location between the faces of the buttress.

b. *Vertical Tendon Anchorages*

Two locations on the top of the ring girder.

c. *Dome Tendon Anchorages*

Two locations on the side of the ring girder.

The frequency of the tendon anchorage concrete surveillance will be as follows:

a. *Immediately prior to and after the pre-operational structural integrity test.*

b. *During the first type A containment leakage rate test only.*

The surveillance program will include the following:

a. *Visual inspection of the end anchorage concrete exterior surfaces.*

b. *The mapping of the predominantly visible concrete crack patterns.*

c. *The measurement of crack widths, by the use of optical comparators or wire feeler gauges, and the notation of the length, orientation and location of the cracks.*

The acceptance criteria for the tendon anchorage concrete surveillance are as follows:

> *The measurements and observations of the cracks and crack patterns of the concrete surfaces will be compared with those taken from other post-tensioned concrete containment structures, and with the previous measurements and observations at the same location on the containment. The cracks and crack patterns will be acceptable if the comparison is in agreement with the predictions.]*

3.8.1.7.5 Containment Surfaces and Steel Liner Plate Surveillance

3.8.1.7.5.1 Containment Surfaces Visual Inspection

A periodic visual inspection of exposed accessible interior and exterior surfaces of containment is conducted in accordance with the FNP Containment Leakage Rate Testing Program and containment inspection plan.

3.8.1.7.5.2 Liner Plate Surveillance

[HISTORICAL]*[The steel liner plate will be examined and measured at four easily accessible locations to monitor its performance. The areas used for surveillance will be chosen as follows:*

a. Two areas where the liner plate has a measurable initial inward curvature.

b. Two areas where the local liner plate has a measurable initial outward curvature.

The frequency of the liner plate surveillance will be as follows:

a. Prior to and shortly after the pre-operational structural integrity test.

b. During the shutdown for the first type A containment leakage test only.

The surveillance program will include the following:

a. Measurement of the inward/outward curvature relative to a fixed chord at each location.

b. Measurement of temperature at the liner plate and exterior concrete at these locations.

c. In addition to the four areas profiled, four more areas will be surveyed for any indication of strain concentrations.

The acceptance criteria for the steel liner plate surveillance are as follows:

> *Measurements and observations will be compared with those made previously at the same locations. The performance of the liner plate will be considered satisfactory if there are no significant differences in the measurements and observations.]*

3.8.1.7.6 Inservice Inspection of Pressure Retaining Concrete Surfaces and Metallic Liner and Attachments.

[HISTORICAL]*[On September 9, 1996, the NRC amended its regulations to incorporate by reference the 1992 edition with 1992 addenda of Subsections IWE and IWL of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code with modifications, in 10 CFR 50.55a. The new rules require certain containment liner and concrete inspections/examinations to be performed prior to September 9, 2001, and to be repeated on a regular basis thereafter. Containment repair and replacement requirements of the new rules including preservice examinations after repair or replacement were effective on September 9, 1996. Relief from this effective date until March 15, 1997, was requested by FNP in order to revise the plant repair/replacement program. The periodic inservice inspections will be incorporated into the individual Inservice Inspection Programs and Plans prior to the required implementation date of September 9, 2001.]*

Some information in paragraph 3.8.1.7.3 through 3.8.1.7.5 above was applicable until 10 CFR 50.55a was amended in 1996 (ref. 61 FR 41303, August 8, 1996). 10 CFR 50.55a was amended in 1996 to incorporate Subsections IWE and IWL of the 1992 edition and 1992 Addenda of the ASME Section XI Code. This amendment also required an expedited implementation schedule.

The tendon surveillance program will be implemented per ASME Section XI, Subsection IWL. The main features of this program are mentioned below, as described in the Containment Inspection Plan.

a. Subsection IWL specifies the required number of tendons to be examined during an inspection. Both FNP units require a minimum and maximum number of each type of tendon as long as other applicable criteria are met.

b. To develop a history and to correlate the observed data, one tendon from each group should be kept unchanged after the initial selection, and these unchanged tendons should be identified as control tendons.

c. Tendon forces are acceptable if the average of all measured tendon forces for each type of tendon is equal to or greater than the minimum required prestress specified at the anchorage for that type of tendon.

The ASME Section XI Inservice Inspection Program's IWE and IWL inspections are credited as license renewal aging management activities (see chapter 18, subsection 18.2.1).

3.8.2 STEEL CONTAINMENT SYSTEM (ASME CLASS MC COMPONENTS)

As described in subsection 3.8.1, the containment is a prestressed, reinforced concrete structure; therefore this section does not apply to the design of the Farley Nuclear Plant.

3.8.3 INTERNAL STRUCTURES

The containment internal structures are all Category I, consisting of:

A. The reactor cavity and primary shield wall.

B. The secondary shield wall, which encloses the steam generator and pressurizer compartments.

C. The refueling canal.

D. The floor slabs.

3.8.3.1 Description of the Internal Structures

A. Reactor Cavity and Primary Shield Wall

The reactor cavity is a heavily reinforced concrete structure which houses the reactor and provides the primary shielding barrier.

The reactor is supported on six special pads and shoes; these vessel supports are mounted on structural steel supports embedded in the reactor cavity concrete. The wall of the cavity structure provides missile protection for the containment structure and liner plate in the event of a hypothetical loss-of-coolant accident. The reactor cavity is basically designed to contain the internal pressures resulting from the loss-of-coolant accident. During normal operating and maintenance inspection, the cavity wall provides biological shielding. Finally, the cavity wall structure acts as the support structure for the reactor and transmits loads to the foundation mat. Refer to figure 3.8-11 for typical dimensions, location, and orientation of the cavity.

B. Secondary Shield Walls

The secondary shield walls are thick reinforced concrete walls anchored into the base slab to ensure stability and prevent uplift. See figure 3.8-7 for base anchorage detail. Figures 3.8-9 and 3.8-10 show the dimensions and extent of the secondary shield wall.

The steam generator compartment is a reinforced concrete structure housing the steam generator, reactor coolant pumps, and the reactor coolant loops. There are three steam generator compartments with one reactor coolant loop in each. Refer to figures 3.8-13 and 3.8-14 for the layout and typical dimensions of the compartments. The compartments are formed by the secondary shielding walls on the exterior and by the reactor and the refueling canal walls on the interior. The steam generator compartment, which extends 11 ft 6 in. above the operating floor slab, and the foundation base mat form the bottom. The pressurizer compartment is built integrally with the secondary shielding wall.

C. Refueling Canal

The refueling canal is used during refueling operations to transfer the new and spent fuel elements between the reactor and the fuel handling building. It is also a laydown area for the reactor upper and lower internals. The canal is lined with stainless steel plate and is filled with borated water to a depth of 34 ft during refueling. The canal is open at the top. A missile protection slab above the reactor head serves also as biological shielding in the vertical direction. Refer to figures 3.8-13 and 3.8-14 for the structural arrangement showing the refueling canal shield.

D. Floor Slabs

The operating floor surrounds the refueling cavity wall, the secondary shield walls and the containment wall. The floor slab is supported by the refueling cavity walls and the secondary shielding walls.

The floor gratings and the floor slab supporting the containment coolers in turn are supported by structural steel beams spanning between the operating floor slab and the containment wall.

3.8.3.2 Applicable Codes, Standards and Specifications

The applicable codes, standards, specifications, regulatory guides, and other documents used in the structural design of the internal structures are covered in subsection 3.8.1.2.

3.8.3.3 Loads and Loading Combinations

The internal structures are designed for all credible conditions of loadings, including normal loads, loads resulting from a loss-of-coolant accident, test loads, and missile generated loads.

Critical loading combinations are those caused by a postulated earthquake, or by a pipe rupture within the containment.

1. Loads

The following loads are considered: dead loads, live loads, earthquake loads, pipe rupture loads, operating thermal loads, and test loads.

a. Dead Loads (D)

Structural dead loads consist of the weight of walls, framing and slabs, partitions, platforms, all permanent equipment, and major piping and electrical conductors.

b. Live Loads (L)

Live loads consist of design floor loads, laydown loads, equipment live loads, fuel handling equipment and material loads.

The operating floor slab is designed for either of these live loads:

Floor gratings	450 psf
Concrete slabs	1000 psf

Equipment live loads are those specified on drawings supplied by the manufacturer of the equipment.

c. Earthquake Loads (E, E')

Earthquake loads are predicated on a basis of (E) the 1/2 safe shutdown earthquake (1/2 SSE), having a horizontal ground surface acceleration of 0.05 g and a vertical ground surface acceleration of 0.033 g.

In addition, a safe shutdown earthquake (SSE), (E') having a horizontal ground surface acceleration of 0.10 g and a vertical ground surface acceleration of 0.067 g, is used to check the design to ensure that loss of structural function will not occur.

Seismic response spectrum curves are given in section 3.7 for both horizontal and vertical ground motions. A dynamic analysis is used to compute the seismic loads for the design of structural elements.

d. Pipe Rupture Loads (R)

Pipe rupture loads are the local jet forces or pressures resulting from a postulated rupture of any one pipe.

e. Thermal loads (T_o, T_a)

The thermal effects during normal operating condition (T_o), as well as the thermal effect due to the accident temperature gradient (T_a), are included.

2. Loading Combinations

In general, two types of loading cases are considered in the design of the internal structures.

a. The design loading case for which the working stress method is used.

b. The factored loading case for which the ultimate stress method is used.

The following terms are used in the loading combination equations:

C = Required capacity of the containment to resist factored loads.

ϕ = Capacity reduction factor.

D = Dead loads of containment, interior structures, and equipment, plus any other permanent contributing loads.

E = 1/2 safe-shutdown earthquake load.

E' = safe shutdown earthquake load.

H = Force on structure due to thermal expansion of pipes under operating conditions.

L = Appropriate live load.

P = Design accident pressure load.

R = Force or pressure on structure due to rupture of any one pipe.

T_a = Thermal loads due to the accident temperature gradient through the wall, based on a temperature corresponding to the unfactored design accident pressure.

T_o = Thermal loads due to the normal operating temperature gradient through the walls.

The load combinations, equations and load factors are as follows:

a. The design loading case for which the working stress method is used:

D + L	(construction case)
$D + L + H + T_o + E$	(operating case)
$D + L + P + T_a$	(design accident case)

b. The factored loading case for which the ultimate stress method is used:

$C = 1/\phi\ (1.0\ D + 1.0\ R + 1.25\ E)$
$C = 1/\phi\ (1.0\ D + 1.25\ H + 1.25\ E)$
$C = 1/\phi\ (1.0\ D + 1.0\ R + 1.0\ E')$
$C = 1/\phi\ (1.0\ D + 1.25\ H + 1.0\ E' + 1.0\ R)$
$C = 1/\phi\ (1.0\ D + 1.25\ H)$

The capacities of all load carrying structural elements are reduced by capacity reduction factors (ϕ) as given below. These factors provide for the possibility that small adverse variations in material strengths, workmanship, dimensions, control, and degree of supervision, which individually are within the required tolerances and the limits of good practice, occasionally may combine to result in undercapacity.

Capacity reduction factors:

ϕ = 0.90 for concrete in flexure.

ϕ = 0.85 for tension, shear, bond and anchorage in concrete.

ϕ = 0.75 for spirally reinforced concrete compression members.

ϕ = 0.70 for tied compression members.

ϕ = 0.90 for fabricated structural steel.

The deflections and deformations of the internal structures and supports are checked to ensure that the functions of the equipment and engineered safeguards are not impaired.

3.8.3.4 Design and Analysis Procedures

The basic techniques of analyzing the internal structures can be classified into two groups: conventional methods involving simplifying assumptions such as those found in beam theory, and those based on plate and shell theories of different degrees of approximation. The strength methods given in the ACI-318-63 code are used. The internal structures are provided with connections capable of transmitting axial and lateral loads to the containment base slab.

The containment interior structure is designed to provide structural supporting elements for the entire nuclear steam supply system (NSSS) as well as required shielding. Basic supporting components are designed using both reinforced concrete and structural steel as appropriate. All design aspects are integrated with the design criteria of the nuclear steam system supplier and include particular attention to the combined thermal and dynamic effects particularly evident during earthquake conditions. Thrusts are taken by rigid members and by shock suppressors. Design loads for the interior structure are listed and described in subsection 3.8.3.3.

Design of the interior structures evolves around four basic systems: the reactor coolant system; the main steam system; the engineered safeguards system; and the fuel handling system. The structures which house or support the basic systems are designed to sustain the factored loads described in subsection 3.8.3.3.

The design bases to be applied are given as follows:

a. All operating loads, thermal loads, seismic loads, and thermal deformations at the levels indicated in subsection 3.8.3.3.

b. Loads and deformations resulting from a LOCA and its associated effects.

c. Pressure buildup in locally confined areas such as the primary shielding cavity or the secondary shielding room.

d. Jet forces resulting from the impingement of the escaping fluid upon adjacent structures.

e. Pipe whipping following a break in the reactor coolant system pipe.

f. Rapid rise in temperature and accompanying rise in pressure.

g. Missiles as described in section 3.5.

The magnitude of thrust forces and pressure buildup resulting from a pipe break is determined from appropriate blowdown values.

The interior areas where local pressure buildup is significant are the reactor cavity and the three secondary shield areas. Ultimate strength design is used for these interior areas in accordance with ACI 318-63. A strain limit of 0.003 in./in. is used for concrete and steel. Sketches showing structural details of the interior structures are shown in figures 3 8-9 to 3 8-14.

Seismic analyses for the interior structures conform to the appropriate procedures outlined in section 3.7.

The mathematical model includes equipment of significant mass values as part of the lumped masses at the appropriate elevation. The seismic loads are determined using the procedures of the design response spectrum technique of analysis. Bending moments and shears resulting from appropriate earthquake loads are combined according to the load combinations described in subsection 3.8.3.3. The equipment seismic shear is resisted by the anchorage system, anchor bolts, and by additional shear studs. See figure 3.8-6 for details.

Various structural components of the interior structures are analyzed and designed individually for governing loading conditions.

For the operating condition analysis, the concrete is assumed to be uncracked and the stresses are limited to those specified in ACI 318-63. For the accident condition analysis, ultimate strength approach is employed. In this analysis the results of the unfactored loading cases are multiplied by appropriate load factors as described in subsection 3 8 3 3. The resulting concrete stresses are limited to 0.85 $f_c^{'}$ and the reinforcing steel stresses are limited to 90 percent of the guaranteed minimum yield given in the appropriate ASTM specification.

The main considerations in establishing the structural design criteria for the internal structures are to provide a structure that will withstand the differential pressure within the cavity in the event of an accident, and to minimize the effects of the pipe rupture force utilizing supports and restraints. Loads and deformations resulting from a LOCA and its associated effects on any one of the basic systems are restricted so that propagation of the failure to any other system is prevented. In addition, a failure in one loop of the nuclear steam supply system is restricted so that propagation of the failure to the other loop is prevented. Localized concrete yielding is permitted when it is demonstrated that the yield capacity of the component is not affected, and that this small localized yielding does not generate missiles that could damage the structure. Full recognition is given to the time increments associated with these postulated failure conditions, and yield capacities are appropriately increased when a transient analysis demonstrates that the rapid strain rate justifies this approach. The walls are also designed to provide adequate protection for potential missile generation that could damage the containment liner.

The effect of radiation generated heat on the internal structures has been considered in the design of the primary and secondary shield walls. The shield wall thicknesses were determined on the basis of the radiation shielding requirements, much higher than those required for structural purposes. This additional thickness provides a reserve strength greater than required to offset minor damages to the structures due to a LOCA. Since high temperatures are damaging to concrete, provisions are made to maintain a constant temperature through ventilation. The ventilation within the containment has been designed to cool the area surrounding the shield walls in order to prevent any appreciable loss of structural strength due to gamma and neutron heating.

The final designs of the interior structure and equipment supports are reviewed to assure that they can withstand applicable pressure loads, jet forces, pipe reactions, and earthquake loads without loss of function. The deflections or deformations of the structures and supports are checked to ensure that the functions of the containment and safety feature systems are not impaired.

3.8.3.4.1 Reactor Cavity and Primary Shield

For the normal operating condition, the reactor cavity is designed to withstand the stresses due to dead loads and thermal loads. Under this condition, the stresses in the concrete and the reinforcing steel are kept below those permissible for the working stress design as stipulated in ACI 318-63 Code. In the stress analysis, flexure tensile cracking is permitted but is controlled by the bonded reinforcing steel.

For the hypothetical LOCA condition, the cavity wall is designed to withstand jet forces and internal pressurization without gross damage to the cavity structure. Local damage to the cavity in the immediate vicinity of the NSSS component failure is inevitable. However, vital parts of the containment are protected from this failure to ensure a post accident leaktight containment structure.

The reactor cavity is designed to withstand a static equivalent internal pressure of 225 psi due to the LOCA. This pressure is assumed to be acting on the entire cavity for a duration of one second. For this loading case the concrete is assumed cracked across the entire section. The reinforcing steel resists all the stresses. The maximum stress level in the rebar under this loading condition is limited to the ultimate capacity of the rebar as modified by appropriate capacity reduction factor.

3.8.3.4.2 Refueling Canal

For the refueling condition, the walls are designed for the hydrostatic head due to 35 ft of water and are checked for the effect of hydrodynamic loads due to 1/2 SSE and SSE motions. The pressure loads on the side of the steam generator compartment and hydrostatic loads do not occur simultaneously.

3.8.3.4.3 Steam Generator Compartments

The compartments are designed for an internal pressure of 30 psi due to a loss-of-coolant accident resulting from a hypothetical double ended rupture of a reactor coolant pipe.

3.8.3.5 Structural Acceptance Criteria

The limiting values of stress, strain, and gross deformations are established by the following criteria:

a. To maintain the structural integrity when subjected to the worst load combinations.

b. To prevent structural deformations from displacing the equipment to the extent that the equipment suffers a loss of safety-related function.

The allowable stresses are those specified in the applicable codes. The stress contributions due to earthquakes are included in the load combinations described in subsection 3.8.3.3.

Structural deformations were found not to be a controlling criterion in the design of the internal structures.

Table 3.8-9 lists the load combinations, the calculated and allowable stresses, as well as the method of analysis used in the design of the main structural components of the internal structures. The ratio of the allowable to the calculated stresses yields the safety margins.

3.8.3.6 Materials, Quality Control and Special Construction Techniques

The following basic materials are used in the construction of the internal structures:

a.	Concrete		f'_c (psi)= 5,500
b.	Reinforcing steel		
	Deformed bars	ASTM A-615 Grade 60	f_y (psi) = 60,000
	Spiral bars	ASTM A-82	f_y (psi) = 70,000
c.	Structural and miscellaneous steel		
	Rolled shapes, bars and plates	ASTM A-36	f_y (psi) = 36,000
	Crane rails	ASCE	
	High strength bolts	ASTM A-325 or A-490	
	Stainless steel	ASTM A-240 Type 304	

d. Reactor cavity steel liner plate and penetration sleeves

1/4-in. liner plates	ASTM A-285 Grade A	f_y (psi) = 24,000
Insert plates	ASTM A-516 Grade 70	f_y (psi) = 38,000
Penetration sleeves		
pipes	ASME SA-333 Grade 6	f_y (psi) = 32,000
plates	ASME SA-516 Grade 70	f_y (psi) = 38,000

e. Interior coating system

Steel liner plate (reactor cavity)
Primer
Finish coat (wainscot)

Carbon steel surface
Primer
Finish coat (wainscot)
Finish coat (above wainscot)

Concrete and masonry surfaces
Surfacer
Primer

The materials and the quality control procedures have been described in subsection 3.8.1.6. Part of the "C" Secondary Shield Wall for Unit 1 and the "A" Secondary Shield Wall for Unit 2 was removed and restored as a part of the steam generator replacement activities. The Cadwelding process used for wall restoration differed in some respects from the one used during initial construction and described in Appendix 3C. The differences are:

1. The original Cadwelding process used for construction of the shield walls, as described in Appendix 3C, specified preheating the reinforcing steel to 300°F to ensure that Cadweld splices were free of moisture. The Cadwelding process used for the restoration of the shield walls did not specify a specific preheat temperature. Vendor practices at the time of shield wall restoration provided detailed instructions for preheating, but did not specify a temperature.

2. The original Cadwelding process used both sister and production splice testing as specified in Appendix 3C. The Cadwelding process used for the restoration used sister splice methodology only. The sister splice testing is in conformance with the statement of conformance for Regulatory Guide 1.10 in Appendix 3A.

Additionally, the concrete mix used for restoration of the wall meets the later editions of the applicable codes and standards described in Section 3.8.1.6.1. The restored wall meets the design strength requirements and is adequate to satisfy the other functions of the wall described in the FSAR.

The internal structures are built of reinforced concrete and structural steel, using proven methods common to heavy industrial construction. No special construction techniques have been employed in the construction of the internal structures.

The effect of various amounts of radiation on the internal structures has been considered in the calculations for the primary and secondary shield walls.

The shield wall thicknesses were determined on the basis of the radiation shielding requirements, much higher than those required for structural purposes. This additional thickness provides a reserve strength greater than required to offset minor damages to the structures due to a LOCA. Since high temperatures are damaging to concrete, provisions are made to maintain a constant temperature through ventilation. The ventilation within the containment has been designed to cool the area surrounding the shield walls in order to prevent any appreciable loss of structural strength due to gamma and neutron heating.

3.8.3.7 Testing and Inservice Surveillance Requirements

A formal program of testing and inservice surveillance is not considered for the internal structures. The internal structures are not directly related to the functioning of the containment concept. Therefore, no testing or surveillance is required.

For the period of extended operation, periodic inspections of the Category I containment internal structures by the Structural Monitoring Program are required license renewal aging management program activities (see chapter 18, subsection 18.2.10).

3.8.4 OTHER CATEGORY I STRUCTURES

Category I structures other than the containment and the internal structures are listed below:

a. Auxiliary building.

b. Diesel generation building.

c. River intake structure.[a]

d. Intake structure at storage pond.

e. Storage pond, dam, and dike.

f. Pond spillway structure.

g. Electrical cable tunnel.

h. Outdoor Category I tanks.

i. Plant vent stack.[a]

3.8.4.1 Description of the Structures

A. Auxiliary Building

The auxiliary building consists of four floors below grade (el 155 ft) and two floors above. The containment and the auxiliary building are separated by a 3-in.-wide expansion joint.

According to analysis, this gap is adequate to prevent the two structures from coming in contact with one another during an earthquake or design basis accident. Flood protection design of the auxiliary building is described in section 3.4.

The auxiliary building houses the following major plant facilities related to safety:

1. New fuel and spent fuel handling, storage, and shipment facilities.

2. Control room and related facilities (shared by both units).

3. Radwaste disposal system facilities.

4. Chemical and volume control facilities.

5. Engineered safety features system (ESF).

6. Penetration room.

7. Access control area (area is located in unit 1 auxiliary building, but is shared by both units).

The auxiliary building is constructed of reinforced concrete below and above grade. All columns, slabs, and structural walls are of reinforced concrete. The roof is a reinforced concrete slab, with a minimum thickness of 2 ft, designed to prevent penetration of missiles. Drawings D-176002 through D-176007 for Unit 1 and D-206002 through D-206007 for Unit 2, and figures 3.8-23, 3.8-24, and 3.8-25 show various plans and sections of the auxiliary building. The principal features of the new and spent fuel handling, storage, and shipment facilities are shown in figures 3.8-23 and 3.8-25.

The fuel handling facilities are served by a 125-ton overhead crane capable of handling heavy loads, such as the spent fuel cask, and a spent fuel handling bridge crane which runs on rails mounted on the operating floor. The overhead spent fuel cask crane is prevented from traveling over the spent fuel pool by means of mechanical stops and administrative controls.

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a. Not required for safe shutdown of the plant and the river intake structure is no longer required to be maintained as Category I.

Mechanical anti-derailing devices mounted on the wheel assemblies of the overhead crane bridge and trolley prevent the crane from being dislodged from its rails due to horizontal motion during an earthquake. The vertical acceleration due to an earthquake is not large enough to overcome the crane's downward load due to gravity. The same devices prevent overturning of the crane gantry in tornado winds. The gantry also has manual locking devices to prevent horizontal movement during design wind conditions and tornado winds.

The spent fuel cask storage area is separated from the fuel transfer canal by a 3 ft 9-in.-thick, reinforced concrete wall and a gate at the fuel transfer slot. The gate will be in a closed position during cask handling, completely isolating the cask storage area from the fuel transfer canal. The base slab of the cask storage area is composed of 5 ft of reinforced concrete placed over a 9 ft 5 in. concrete fill, which in turn rests on a 5 ft-thick reinforced concrete mat bearing directly on the Lisbon formation.

The spent fuel cask wash area is separated from the cask storage area and the fuel transfer canal by 3 ft-thick and 3 ft 9 in.- thick, respectively, reinforced concrete isolation walls. The base slab of the cask wash area is composed of 5 ft reinforced concrete placed over a 30 ft compacted backfill, which in turn rests on a 5 ft-thick reinforced concrete mat bearing directly on the Lisbon formation.

Figures 3.8-41 and 3.8-42 show the details of the spent fuel cask storage and the cask wash areas.

The fuel transfer from and into the containment is accomplished through the fuel transfer tube. Expansion joint bellows at the fuel transfer tube provide for the relative movement between containment, containment internals, and the spent fuel pool. The bellows allow for all loading conditions including the safe shutdown earthquake and maximum hydraulic pressure. The design of the expansion bellows considers the maximum computed relative axial and lateral displacement of the fuel transfer tube occurring simultaneously. Access for inspection and maintenance of the bellows is provided. See figure 3.8-2 for details of the refueling transfer tube. The spent fuel bundles are stored in stainless steel racks in the spent fuel pool.

The spent fuel pool walls and base slab are constructed of thick (6 ft to 7 ft walls and 5-ft slab) reinforced concrete. The inside face of the walls and base slab are lined with 1/4-in.-thick stainless steel liner plate to provide leaktightness. The reinforced concrete superstructure of the fuel handling area protects the spent fuel pool from the environment.

B. Diesel Generator Building

This reinforced concrete building, housing the diesel generators essential to safe plant shutdown, is a one-story, box-type structure. Reinforced concrete interior walls are provided to physically separate the diesel generators from each other. Figure 3.8-26 shows the general configuration of the building.

The foundation, which consists of a reinforced concrete mat slab supported by concrete caissons, is anchored to the Lisbon formation, and is structurally separated from the electrical cable tunnels by means of free joints. The 2 ft 6 in. exterior reinforced concrete walls and the 2 ft roof slab provide protection against missiles, as described in section 3.5.

C. River Intake Structure

The river intake structure was originally designed as a Category I structure; however, it is not required to be maintained as Category I. It is a reinforced concrete, box-type structure which houses the river water pumps. It consists basically of two levels. The upper level is a concrete enclosure which houses the river water pumps. The lower level consists of passages to supply the pumps with water. Trash racks and traveling screens keep debris from entering the pump suction lines via the water passages. Figure 3.8-27 shows the configuration of the structure. It is placed on a base mat which bears directly on the Lisbon formation.

D. Intake Structure at Storage Pond

The intake structure at the storage pond is a redundant Category I, reinforced concrete, box-type structure which houses the pumps, wetwell, and traveling screens. Figure 3.8-28 shows the configuration of the structure. The structure has a reinforced concrete caisson foundation which is anchored into the Lisbon formation.

E. Storage Pond, Dam, and Dike

See appendix 2B to chapter 2.0 for the description of this structure.

F. Pond Spillway Structure

A description of this structure is given in section 2.4.8.1. In addition, figure 3.8-29 shows its configuration.

G. Electrical Cable Tunnel

The electrical cable tunnel is a Category I, reinforced concrete, tubular type structure which encloses the emergency electrical cable between the diesel generator and auxiliary buildings. The emergency electrical cable is required for a safe shutdown of the plant. The reinforced concrete tunnel has a mat foundation which bears for the most part on the Moody's Branch formation.

H. Outdoor Category I Tanks

The following tanks are Category I:

1. Refueling water storage tank.
2. Reactor makeup water storage tank.
3. Condensate storage tank.

These tanks are cylindrical in shape and are supported by concrete mats resting on compacted backfill. The tanks are of steel plate construction and are designed to withstand seismic generated loads. Shield and retaining walls are provided for the RWST and RMWST only to safeguard the quantities of water in the tanks required for a safe shutdown of the plant. The CST contains no shield or retaining walls, but has its bottom 12 feet reinforced to withstand ruptures caused by tornado-generated missiles. A tabulation of the tank capacities and dimensions is given below:

Tank	**Capacity (gal)**	**I.D. (ft)**	**Height (ft)**
Refueling water storage tank	500,000	46	41
Reactor makeup water storage tank	200,000	34	32
Condensate storage tank	500,000	46	41

3.8.4.2 Applicable Codes, Standards and Specifications

The following codes, standards, specifications, design criteria, NRC Regulatory Guides and industry standard practices constitute the basis for the design and construction of all Category I structures other than the containment. Modifications to these codes, standards, etc. are made where necessary, to meet the specific requirements of the structures. These modifications are indicated in the sections where references to the codes, standards, etc. are made.

Codes

ACI 214-65 "Recommended Practices for Evaluation of Compression Test Results of Field Concrete."

ACI 301-66 "Specifications for Structural Concrete for Buildings."

ACI 306-66 "Recommended Practice for Cold Weather Concreting."

ACI 311-64 "Recommended Practices for Concrete Inspection."

ACI 315-65 "Manual of Standard Practice for Detailing Reinforced Concrete Structures."

ACI 318-63 "Building Code Requirements for Reinforced Concrete."

ACI 605-59 "Recommended Practice for Hot Weather Concreting."

ACI 347-63 "Recommended Practice for Concrete Formwork."

ACI 613-54 "Recommended Practice for Selecting Proportions for Concrete."

ACI 614-59 "Recommended Practice for Measuring, Mixing, and Placing Concrete."

AISC Manual of Steel Construction, 1963 and 1969 Editions.

AWS D1.1-86 Structural Welding Code - Steel.

AWS D2.0-69 Specifications for Welded Highway and Railway Bridges.

NCIG-01, “Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants” - Rev. 2 EPRI NP-5380.

ICBO Uniform Building Code, 1970 Edition.

SBCC Southern Standard Building Code, 1969 Edition.

CFR Code of Federal Regulations, Title 29, Chapter XVII, "Occupational Safety and Health Standards."

Specifications

CMAA Specifications for Electric Overhead Traveling Crane - No. 70, 1970 Edition.

ASTM The specifications used are identified in the applicable subsections.

Design Criteria

ASCE "Wind Forces on Structures," Paper No. 3269.

AEC "Nuclear Reactor and Earthquake" - Publication TID 7024.

NRC Regulatory Guides

Regulatory Guide No. 1.10 "Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments."

Regulatory Guide No. 1.13 "Fuel Storage Facility Design Basis.”

Regulatory Guide No. 1.15 "Testing of Reinforcing Bars for Concrete Structures."

Regulatory Guide No. 1.28 "Quality Assurance Program Requirements - Design and Construction."

Regulatory Guide No. 1.29 "Seismic Design Classification."

Regulatory Guide No. 1.31 "Control of Stainless Steel Welding."

Regulatory Guide No. 1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants."

Regulatory Guide No. 1.55 "Concrete Placement in Category I Structures."

Regulatory Guide No. 1.64 "Quality Assurance Program Requirements for the Design of Nuclear Power Plants."

<u>Bechtel Corporation Topical Reports</u>

BC-TOP-4 (Rev. 1) "Seismic Analysis of Structures and Equipment for Nuclear Power Plants," Sept 1, 1972

3.8.4.3 Loads and Loading Combinations

All Category I structures are designed for all credible conditions of loadings, including normal loads, loads resulting from a pipe rupture where applicable, and loads due to adverse environmental conditions.

3.8.4.3.1 Loads

The following loads are considered in the design:

a. Dead loads.

b. Live loads.

c. Earthquake loads.

d. Pipe rupture loads.

e. Thermal loads.

f. Wind and tornado loads.

g. Hydrostatic loads.

h. Cask drop loads.

A. Dead loads.

Structural dead loads consist of the weight of framing, roof, floors, walls, partitions, platforms, hangers, cable trays, pipes with fluid, and equipment dead loads, as specified on the drawings supplied by the manufacturers of the equipment installed within the structure.

B. Live Loads

Live loads consist of design floor loads, pool and tank liquid weights, and equipment live loads as specified on the drawings supplied by the manufacturers of the equipment installed within the structure.

C. Earthquake Loads

Earthquake loads are predicated on a basis of 1/2 of the safe shutdown earthquake (SSE) and having a horizontal ground acceleration of 0.05g, and a vertical ground acceleration of 0.033g.

In addition, a safe shutdown earthquake (SSE) having a horizontal ground acceleration of 0.10g, and a vertical ground acceleration of 0.067g, is used to check the design to ensure that loss of structural functions would not occur.

Seismic response spectrum curves are given in section 3.7 for both horizontal and vertical ground motions. A dynamic analysis is used to compute the seismic loads for the design of structural elements.

D. Pipe Rupture Loads

Pipe rupture loads include the jet-impingement forces from postulated pipe breaks, differential pressures that might build up across compartments, and loads due to pipe whipping or pipe restraint.

E. Thermal Loads

Thermal loads are those induced in the spent fuel pool floor and walls due to the thermal gradients across these elements. Thermal gradients may be caused by an increase in water temperature during operating conditions, or by an accident. The interior temperatures of the pool are assumed to be 180°F for both normal and abnormal conditions. The ambient temperature is assumed to be 50°F for exterior walls and 70°F for the internal walls.

F. Wind and Tornado Loads

The wind loadings and tornado loadings are discussed in section 3.3.

All Category I structures are designed to withstand the effects of the wind and tornado loadings, and to provide protection against tornado missiles for all Category I systems and components within the structures.

The structures are analyzed for tornado loadings not coincident with the safe shutdown earthquake.

G. Hydrostatic Loads

Lateral hydrostatic pressure loads and buoyant forces resulting from the displacement of ground and flood waters are applied to the structures, as discussed in section 3.4.

H. Cask Drop Loads

Special lifting devices, as discussed in section 9.1.4.2.2.5, are provided for cask crane operation to prevent the dropping of the cask. However, a static equivalent load of 1000 Kips (over a 38.5 ft^2 area) has been considered in the design of the spent fuel pool slab.

3.8.4.3.2 Loading Combinations

The load combinations and load factors for Category I structures listed in Section 3.8.4.1 are as follows:

$C = 1/\phi\ (1.0D + 1.0R + 1.25E\ (\text{or } 1.25W))$

$C = 1/\phi\ (1.0D + 1.25H + 1.25E\ (\text{or } 1.25W))$

$C = 1/\phi\ (1.0D + 1.0R + 1.0E')$

$C = 1/\phi\ (1.0D + 1.25H + 1.0E' + 1.0R)$

$C = 1/\phi\ (1.0D + 1.0W_t + 1.25H)$

Where:

C = required capacity of the structures.

D = dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads. In addition, a portion of "live load" will be added when such load is expected to be present during plant operation. An allowance will also be made for future permanent loads.

R = force of pressure on structure due to rupture of any one pipe.

H = force on structure due to thermal expansion of pipes under operation conditions.

E = 1/2 safe shutdown earthquake resulting from horizontal ground surface acceleration of 0.05g; vertical acceleration is 2/3 horizontal acceleration.

E'= safe shutdown earthquake resulting from horizontal ground surface acceleration of 0.10g; vertical acceleration is 2/3 horizontal acceleration.

W = wind load.

W_t= tornado wind load including differential pressure.

The capacity reduction factors that will be used in the design are:

ϕ = 0.90 for reinforced concrete in flexure.

ϕ = 0.85 for tension, shear, bond, and anchorage in reinforced concrete.

ϕ = 0.75 for spirally reinforced concrete compression members.

ϕ = 0.70 for tied compression members.

ϕ = 0.90 for fabricated structural steel.

ϕ = 0.90 for reinforcing steel in direction tension.

3.8.4.4 Design and Analysis Procedures

The analysis procedures for the Category I structures listed in subsection 3.8.4.1 are based on conventional methods, as found in standard textbooks and handbooks used in universities and engineering practice.

The design procedures of these Category I structures are in accordance with design methods of accepted standards and codes where applicable for normal operating loads.

The seismic analysis of these structures is covered in section 3.7. The structures are proportioned to maintain elastic behavior when subjected to various combinations of dead loads, live loads, wind loads, tornado loads, seismic loads, and LOCA loads. The upper limit of elastic behavior is the yield strength of the effective load carrying structural materials.

3.8.4.5 Structural Acceptance Criteria

The limiting values of stress, strain, and gross deformations are established by the following criteria:

a. To maintain the structural integrity when subjected to the worst load combinations.

b. To prevent structural deformations from displacing the Category I equipment to the extent that it suffers a loss of function.

The allowable stresses are those specified in the applicable codes. The stress contributions due to earthquakes are included in the load combinations described in subsection 3.8.4.3.

Structural deformations were found not to be a controlling criterion in the design of the Category I structures, listed in subsection 3.8.4.1.

Tables 3.8-10 through 3.8-14 give the load combinations, the calculated and allowable stresses, as well as the method of analysis used in the design of the main structural components of the structures. The ratio of the allowable to the calculated stresses yields the safety margins.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

The following basic materials are used in the construction of the Category I structures listed in subsection 3.8.4.1.

a. Concrete

Auxiliary building		f'_c (psi) = 5,000
Diesel generator building		f'_c (psi) = 3,000 f'_c (psi) = 4,000
All other structures listed in subsection 3.8.4.1		f'_y (psi) = 4,000

b. Reinforcing steel

Deformed bars	ASTM A-615 Grade 60	f_y (psi) = 60,000
Spiral bars	ASTM A-82	f_y (psi) = 70,000

c. Structural and miscellaneous steel

Rolled shapes, bars and plates	ASTM A-36	f_y (psi) = 36,000
High strength bolts	ASTM A-325 A-193 B-7, or A-490	
Stainless steel	ASTM A-240 Type 304	
Insert plates (Auxiliary building)	ASTM A-516 Grade 70	f_y (psi) = 38,000

The materials and the quality control procedures have been described in paragraph 3.8.1.6.

The Category I structures listed in paragraph 3.8.4.1 are built of reinforced concrete and structural steel, using proven methods common to heavy industrial construction. No special construction techniques have been employed in the construction of these structures.

3.8.4.7 Testing and Inservice Surveillance Requirements

No structural preoperational testing of the Category I structures is planned. During the life of the plant, periodic inspections of the structures will be made to employ visual inspection for apparent structural deterioration such as large cracks and excessive deflection of structural members.

For the Category I structures required to be maintained as Category I, periodic inspections performed under the Structural Monitoring Program and Service Water Pond Dam Inspection Program (as applicable) are required license renewal aging management program activities for the period of extended operation (see chapter 18, subsections 18.2.10 and 18.2.3). The noncivil features of the outdoor tanks (e.g., fluid-retaining) are age-managed separately from these structural inspection programs as part of the associated fluid systems.

All seam and plug welds in the spent fuel pool liner plate were vacuum box tested upon completion of the welding. Where vacuum box testing was not possible, liquid penetrant testing was performed. The service water pond dam and spillway will be inspected during the period of extended operation on a periodic basis in accordance with NRC Regulatory Guide 1.127, Revision 1.

The spent fuel pool has a system that provides for leakage to be detected at any time in the life of the plant. This system consists of troughs under the liner plate which lead to a collection system where leakage can be observed.

3.8.5 FOUNDATIONS AND CONCRETE SUPPORTS

All Category I structures are founded, either directly or by means of caissons, on the Lisbon formation.

3.8.5.1 Description of the Foundations and Supports

A. Containment

The containment foundation is a conventionally reinforced circular concrete mat, 9 ft thick with a diameter of 146 ft 6 in. bearing directly on the Lisbon formation. The reactor cavity is located in the center of the mat and forms an integral part of the foundation. Figure 3.8-36 shows the relative position of the two containment foundations and the other Category I structure's foundations.

Figure 3.8-37 shows cross-sections of the containment base slab.

The internal structures that support the large equipment, such as steam generators and reactor coolant pumps, are anchored to the base slab in order to transfer the loads. Figure 3.8-6 shows a typical detail of anchorage to the base slab for the steam generator and reactor coolant pumps.

Figure 3.8-37 shows the reinforcing pattern at the junction of the base slab and containment wall.

B. Auxiliary Building

The Auxiliary Building foundation is a reinforced concrete slab 5 ft 0 in. thick, 430 ft long by 300 ft wide, bearing directly on the Lisbon formation. In the eastern section of the Auxiliary Building, specifically east of column line P, the structure is supported on spread footings 9 ft 0 in. x 9 ft 0 in. x 4 ft 0 in. which bear on the Lisbon formation. The loads are transmitted through cast-in place reinforced concrete columns 3 ft 6 in. diameter.

Refer to figure 3.8-36 for location of auxiliary building foundations in relation to other Category I structures.

Figure 3.8-38 shows typical base slab and foundation details.

C. Diesel Generator Building

The Diesel Generator Building foundation is a 4 ft 0 in.-thick reinforced concrete slab bearing on 5 ft 0 in.-diameter cast-in place caissons which transfer the loads to the Lisbon formation. Figure 3.8-39 shows details of caissons, caisson to base slab connection, and load transfer mechanism. Figure 3.8-36 shows the relative position of the diesel generator building foundation to the other Category I structures.

The diesel generator foundations are projected 4 1/2 in. above the base slab and form an integral part of it.

D. River Intake Structure

The river intake structure foundation is a 5-ft 0-in.- thick (3 ft 0 in. in the bay area) reinforced concrete slab which bears directly on the Lisbon formation.

The load transfer is done through bearing walls and columns which support the superstructure.

Refer to figure 3.8-36 for location of river intake structure foundation relative to other Category I structures.

E. Intake Structure at Storage Pond

The intake structure at storage pond foundation is a 3-ft-thick reinforced concrete slab bearing on 6 ft, 7 ft, and 10 ft-diameter cast in place caissons which transfer the loads to the Lisbon formation.

Refer to figure 3.8-36 for location of intake structure at the storage pond foundation relative to other Category I structures.

Figure 3.8-39 shows a typical detail of caisson and base slab.

Bearing walls and columns support the superstructure.

F. Electrical Cable Tunnel

The electrical cable tunnel foundation is a reinforced concrete slab bearing directly on the Moodys Branch formation for the most part. In areas where direct bearing was not possible, caissons anchored into the Lisbon formation were used.

G. Category I Outdoor Tanks

There are three Category I outdoor tanks:

1. Refueling water storage tank.

2. Reactor makeup water storage tank.

3. Condensate storage tank.

The foundations of these tanks are 4 ft 0 in.-thick reinforced concrete slabs bearing on compacted fill.

The tank foundations are physically separated from each other as shown in figure 3.8-36.

3.8.5.2 Applicable Codes, Standards and Specifications

The applicable codes, standards and specifications are discussed in the following subsections:

Containment	- 3.8.1.2
Internal Structures	- 3.8.3.2
Other Category I Structures	- 3.8.4.2

3.8.5.3 Loads and Loading Combinations

Containment foundation loads and loading combinations are discussed in subsection 3.8.1.3.

Foundation loads and loading combinations for other Category I structures are discussed in subsection 3.8.4.3.

3.8.5.4 Design and Analysis Procedures

Design and Analysis Procedures for the Containment including the base slab are discussed in Topical Report BC-TOP-5 Sections 6.0 and 7.0.

The basic techniques for analysis and design of the foundations for all other Category I structures are the conventional methods which involve simplifying assumptions such as are found in the theory of concrete structures. Stresses resulting from local moments, torques, concentrated reactions and uniform loadings are computed by these methods. These methods are further discussed in subsections 3.8.3.4 and 3.8.4.4.

3.8.5.5 Structural Acceptance Criteria

The foundations of all Category I structures are designed to meet the same structural acceptance criteria as the structures themselves. These criteria are discussed in subsections 3.8.1.5, 3.8.3.5 and 3.8.4.5.

The limiting conditions for the foundation medium together with a comparison of actual capacity and estimated structure loads are found in chapter 2.0, appendix 2B.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The foundations and equipment supports are built of reinforced concrete using proven methods for heavy industrial construction. The description of the materials and the quality control procedures, as well as special construction techniques for foundations, are the same as those discussed in paragraphs 3.8.1.6, 3.8.3.6, and 3.8.4.6 and chapter 17.0.

3.8.5.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance are not required and are not planned for foundations of structures or supports. A discussion of the test program which serves as the basis for the soils investigation and foundation evaluation may be found in chapter 2.0, appendix 2B.

For the period of extended operation, periodic inspections of the Category I structures required to be maintained as Category I (see paragraph 3.8.4.7) are credited license renewal aging management program activities for aging management of foundations and concrete supports.

3.8.6 MASONRY WALLS

As documented in SER NUREG 0117, Supplement 5 to NUREG 75/034 dated March 1981, the NRC acceptance criteria associated with the seismic design of masonry walls is contained in IE Bulletin 80-11 and the applicable requirements of GDC 2 and GDC 4. Additionally, as documented in NRC miscellaneous letter dated Oct. 24, 1984, the following three options are accepted by the NRC to seismically qualify masonry walls:

1. Reanalyze walls qualified by the energy-balance technique by linear elastic working stress approach as recommended in the staff acceptance criteria (SRP Section 3.8.4, Appendix A) and implement modifications to walls as needed.

2. Develop rigorous nonlinear time-history analysis techniques capable of capturing the mechanism of the walls under cyclic loads. Different stages of behavior should be accurately modeled; elastic uncracked, elastic cracked, and inelastic cracked with yielding of the central rebars. Then, a limited number of dynamic tests (realistic design earthquake motion inputs at top and bottom of the wall) should be conducted to demonstrate the overall conservatism of the analysis results. In this case, “as-built” walls should be constructed to duplicate the construction details of a specific plant.

3. For walls qualified by energy-balance technique, conduct a comprehensive test program to establish the basic non-linear behavioral characteristics of masonry walls (i.e., load-deflection hysteretic behavior, ductility rations, energy absorption and post yield envelopes) for material properties and construction details pertaining to masonry walls in question. The behavior revealed from tests should then be compared with that of elastic-perfectly-plastic materials for which the energy balance technique was originally developed. If there are significant differences, then the energy balance technique should be modified to reflect the actual wall behavior.

   Structural Monitoring Program inspections of seismically qualified masonry walls are credited license renewal aging management program activities for the period of extended operation (see chapter 18, subsection 18.2.10).

REFERENCES:

1. Welding Research Council (WRC) Bulletin No. 102, "State of Stress in a Circular Cylinder Shell with a Circular Hole," By Eringen, Naghdi and Thiel.

TABLE 3.8-1

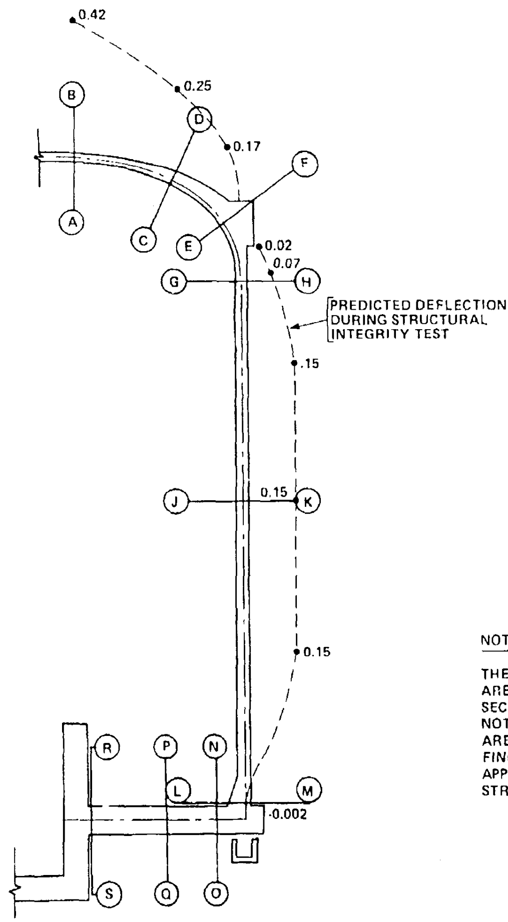
POST-TENSIONING SYSTEM - BBRV (170)

Designation	Wires
Ultimate capacity	2000 kips
Design capacity	1200 kips
Minimum tensile strength	240 ksi
Relaxation	8 1/2 percent @ 0.70 f
Ductility	4 percent[a]
End anchorage	Buttonhead
Anchor material	Head H.R. 4140 or 4142 Alloy Steel
	Bushing H.F.S.M. 4142 Tubing Shim ASTM A-36-70a .40/.50 Carbon Sheet Steel
Bearing plate	ASTM A-36-70a

a. When measured in a gauge length of 10 inches (for wire only).

TABLE 3.8-2 (SHEET 1 OF 8)

STRESS ANALYSIS RESULTS



KEY ELEVATION
(SHOWING LOCATION OF REFERENCE SECTIONS & PREDICTED DEFLECTION DURING STRUCTURAL INTEGRITY TEST)

STRUCTURAL DATA

SECTION	CONCRETE		REINFORCING STEEL		
	f'_c – PSI	t – in	TYPE	P_m – %	P_h – %
A B	5500	39	A615GR60	--- .271	--- .271
C D	5500	39		.214 .443	.107 .443
E F	5500	139		.094 .222	.030 .124
G H	5500	45		.096 .985	.347 .347
J K	5500	45		.033 .289	.073 .289
L M	5500	81		.299 .549	.193 .278
N O	5000	108		.370 .488	.231 .463
P Q	5000	108		.370 .488	.231 .463
R S	5000	108		.370 .488	.231 .463

NOTES ON VALUES IN TABLES

THE RESULTS SHOWN ON THE FOLLOWING SHEETS ARE FROM THE FINITE ELEMENT CRACKED SECTION ANALYSIS UNLESS OTHER WISE NOTED ON THE SHEET. EARTHQUAKE LOADS ARE ADDED TO THE RESULTS FROM THE FINITE ELEMENT ANALYSIS WHERE APPLICABLE AND THE ADDITIONAL SECTION STRESSES EVALUATED.

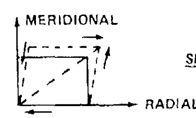
SIGN COVENTION

FORCES, STRESSES AND STRAINS:

+ TENSILE
– COMPRESSIVE

DEFLECTION: INWARD – | OUTWARD +

MOMENTS: + TENSION ON INSIDE FACE OF STRUCTURE
– COMPRESSION ON INSIDE FACE OF STRUCTURE



SHEAR: SHEAR FORCES SHOWN IN THE FIGURE OPPOSITE ARE POSITIVE. SHEAR FORCES OPPOSING THOSE IN THIS FIGURE ARE NEGATIVE.

TABLE 3.8-2 (SHEET 2 OF 8)

STRESS ANALYSIS RESULTS

NOTATION AND NOTES FOR STRESS AND STRAIN TABLES

NOTATION:

D = DEAD LOAD
F_F = MINIMUM GUARANTEED PRESTRESS LEVEL
F_I = INITIAL PRESTRESS LEVEL
P = INTERNAL PRESSURE
E = 1/2 SAFE SHUTDOWN EARTHQUAKE
E' = SAFE SHUTDOWN EARTHQUAKE
T_A = ACCIDENT TEMPERATURE
f'_c = COMPRESSIVE STRENGTH OF CONCRETE
t = THICKNESS OF SECTION
P_m = MERIDIONAL STEEL PERCENTAGE
P_h = HOOP STEEL PERCENTAGE

NOTES:

THE HIGHER ALLOWABLE CONCRETE STRESS IS FOR THE DOME AND WALL. THE LOWER ALLOWABLE CONCRETE STRESS IS FOR THE BASE SLAB ONLY.

DEFLECTIONS ARE MEASURED HORIZONTALLY FOR THE WALL AND VERTICALLY FOR THE BASE SLAB. DEFLECTIONS FOR THE DOME ARE THE RESULTANT OF THE VERTICAL AND HORIZONTAL COMPONENTS OF DEFLECTION AT THAT SECTION.

THE PREDICTED VALUES OF STRAIN IN CONCRETE AND REINFORCEMENT SHOWN IN THESE TABLES ARE THE VALUES RELATIVE TO THE ASSUMED REFERENCE TEMPERATURE OF 70° F AND ARE COMPUTED BY THE "FINEL" PROGRAM (CE-316-4) BASED ON THE 3-DIMENSIONAL STRESS-STRAIN-TEMPERATURE RELATIONSHIP.

THE CONCRETE STRAIN VALUES SHOWN IN THESE TABLES ARE THE MAXIMUM VALUES OF STRAIN AT EACH PARTICULAR SECTION; THEREFORE, THE CONCRETE STRAINS MAY NOT NECESSARILY EQUAL THE REINFORCEMENT STRAINS IN MAGNITUDE AT EACH SECTION. THE CONCRETE STRESSES SHOWN IN THESE TABLES ARE THE MAXIMUM VALUES OF STRESS FOR EACH PARTICULAR SECTION.

TABLE 3.8-2 (SHEET 3 OF 8)

STRESS ANALYSIS RESULTS
DEAD LOAD AND INITIAL PRESTRESS
$(D + F_I)$

PORTION	SECTION	CONCRETE		LINER		REINFORCING STRESS				SECTION RESULTANTS					SHEAR CAPACITY	DEFLECTION
						OUTSIDE		INSIDE								
		MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. FORCE K/FT.	HOOP FORCE K/FT.	MER. MOMENT K FT./FT.	HOOP MOMENT K FT./FT.	RADIAL SHEAR K/FT.		IN.
ALLOW.		−3300 −2250	−3300 −2250	STRAIN	GOVERNS	±20000	±20000	±20000	±20000	–	–	–	–	–	± k/FT	–
DOME	A B	−1690	−1631	−9150	−9064	−8120	−7730	–	–	−775	−753	11	7	6	98	−1.20
DOME	C D	−1760	−421	−9160	−3239	−4530	−1469	–	–	−644	−204	−176	−9	−51	81	−0.84
DOME	E F	−702	−318	−4090	−2234	−1270	−491	−3390	−1055	−931	−321	−472	−130	−29	150	−0.76
WALL	G H	−796	−864	−5230	−5588	−3530	−3890	–	–	−452	−472	−1	−28	33	63	−0.08
WALL	J K	−873	−1438	−5930	−9234	−3420	−7141	–	–	−490	−795	−18	−47	0	16	−0.18
WALL	L M	−621	−430	−4582	−3073	−3268	−1582	−2288	−1388	−538	−386	91	−90	−95	97	−0.04
BASE SLAB	N O	−818	−483	−6392	−3700	9770	3380	−4422	−1693	−141	−353	−865	−481	−23	293	−0.60
BASE SLAB	P Q	−332	−175	−2478	−1430	866	−124	−1853	−708	−139	−120	−887	−492	−39	293	−0.64
BASE SLAB	R S	−444	−272	753	281	−2076	−833	302	−24	−156	−258	430	305	−110	293	−0.61

TABLE 3.8-2 (SHEET 4 OF 8)

STRESS ANALYSIS RESULTS
DEAD LOAD, FINAL PRESTRESS, AND 115% DESIGN PRESSURE
$(D + F_F + 1.15P)$

PORTION	SECTION	CONCRETE		LINER		REINFORCING STRESS				SECTION RESULTANTS					SHEAR CAPACITY	DEFLECTION
						OUTSIDE		INSIDE								
		MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. FORCE K/FT.	HOOP FORCE K/FT.	MER. MOMENT K FT./FT.	HOOP MOMENT K FT./FT.	RADIAL SHEAR K/FT.		IN.
ALLOW.		−3300 −2250	−3300 −2250	STRAIN	GOVERNS	± 20000	± 20000	± 20000	±20000	–	–	–	–	–	± k/FT.	–
DOME	A B	−293	−279	−1650	−1442	−2430	−2253	–	–	−120	−111	14	15	1	98	−0.70
DOME	C D	−491	−39	−4870	−894	464	658	–	–	−117	6	−111	−23	−16	81	−0.54
DOME	E F	−617	−267	3010	−1074	−2257	−1060	2090	−1812	−431	−262	1050	90	−42	150	−0.69
WALL	G H	−186	−142	−1940	−1476	−1240	−885	–	–	−102	−77	−8	−11	−3	63	0.00
WALL	J K	−244	−71	−2390	−736	−2270	−64	–	–	−139	−38	−7	−7	0	15	0.02
WALL	L M	−823	−348	6381	453	−7632	−1581	4664	−1831	−208	−208	836	163	59	97	−0.05
BASE SLAB	N O	−17	−200	−400	−2540	−20	3260	−261	−2070	17	18	−41	−494	127	293	−0.55
BASE SLAB	P Q	−739	−661	−10360	−9280	19220	14840	−6450	−4930	−24	−25	−1167	−935	38	293	−0.71
BASE SLAB	R S	−101	−26	−1047	−216	302	52	−872	79	−63	−25	−148	−17	−169	293	−0.78

TABLE 3.8-2 (SHEET 5 OF 8)

STRESS ANALYSIS RESULTS
DEAD LOAD, FINAL PRESTRESS, DESIGN PRESSURE, AND THERMAL ACCIDENT
$(D + F_F + P + T_A)$

PORTION	SECTION	CONCRETE MER. PSI	CONCRETE HOOP PSI	LINER MER. PSI	LINER HOOP PSI	REINFORCING STRESS OUTSIDE MER. PSI	REINFORCING STRESS OUTSIDE HOOP PSI	REINFORCING STRESS INSIDE MER. PSI	REINFORCING STRESS INSIDE HOOP PSI	SECTION RESULTANTS MER. FORCE K/FT.	SECTION RESULTANTS HOOP FORCE K/FT.	SECTION RESULTANTS MER. MOMENT K FT./FT.	SECTION RESULTANTS HOOP MOMENT K FT./FT.	SECTION RESULTANTS RADIAL SHEAR K/FT.	SHEAR CAPACITY	DEFLECTION IN.
ALLOW.		−3300 −2250	−3300 −2250	STRAIN	GOVERNS	±30000	±30000	±30000	±30000	–	–	–	–	–	± K/FT	–
DOME	A B	−963	−905	−23900	−23875	13100	11942	–	–	−171	−165	−276	−266	0	98	−0.12
DOME	C D	−1280	−396	−19300	−19195	27197	23731	–	–	−164	−67	−355	−176	−19	81	0.47
DOME	E F	−871	−1721	−12200	−24366	−2530	12200	8360	−5513	−478	−126	520	−181	−47	150	0.18
WALL	G H	−1470	−1222	−24600	−23959	19400	18798	–	–	−139	−187	−521	−388	−7	63	0.32
WALL	J K	−1130	−662	−23400	−21384	14900	19291	–	–	−177	−113	−350	−274	0	15	0.38
WALL	L M	−373	−383	−126	−4237	−4145	11106	−2337	−2889	−243	−51	49	−331	34	97	0.10
BASE SLAB	N O	−328	−402	−4220	−5540	4230	9590	−2440	−3510	−88	−28	−481	−653	99	293	−0.54
BASE SLAB	P Q	−860	−721	−11420	−9880	16260	14150	−7510	−5380	−158	−76	−1351	−1012	32	293	−0.70
BASE SLAB	R S	−840	−29	−7370	−1170	1760	4240	−6620	1090	−379	70	−977	−43	−146	293	−0.75

TABLE 3.8-2 (SHEET 6 OF 8)

STRESS ANALYSIS RESULTS
DEAD LOAD, FINAL PRESTRESS, 150% DESIGN PRESSURE, AND THERMAL ACCIDENT
$(D + F_F + 1.5P + T_A)$

PORTION	SECTION	CONCRETE		LINER		REINFORCING STRESS				SECTION RESULTANTS					SHEAR CAPACITY	DEFLECTION
						OUTSIDE		INSIDE								
		MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. FORCE K/FT.	HOOP FORCE K/FT.	MER. MOMENT K FT./FT.	HOOP MOMENT K FT./FT.	RADIAL SHEAR K/FT.		IN.
ALLOW.		−4950 −4500	−4950 −4500	STRAIN	GOVERNS	±54000	±54000	±54000	±54000	–	–	–	–	–	± K/FT	
DOME	A B	−195	−142	−16200	−15482	19800	18299	–	–	−46	−40	−138	−125	−1	37	1.02
	C D	−574	−165	−18900	−18510	49000	26146	–	–	−67	−35	−260	−144	−15	37	1.23
	E F	−676	−1779	−5460	−23882	−3194	10945	13500	−7315	−178	−139	164	−1383	−30	255	0.66
WALL	G H	−891	−982	−22200	−22540	22800	18017	–	–	−48	−152	−375	−357	−18	107	0.35
	J K	−507	–	−20100	−14612	20400	24871	–	–	−85	−2	−239	−128	1	25	0.54
	L M	−786	−361	7657	−2049	−8218	10677	4435	−3581	−161	−58	626	−308	88	185	0.08
BASE SLAB	N O	−371	−448	−5360	−6630	9580	14930	−2770	−3910	18	58	−469	−670	153	497	−0.35
	P Q	−1012	−728	−13950	−10290	24030	16850	−8830	−5430	−82	−16	−1617	−1063	32	497	−0.71
	R S	−616	−41	−7090	−1260	1710	3840	−6380	910	−364	65	−941	−61	−193	497	−0.85

TABLE 3.8-2 (SHEET 7 OF 8)

STRESS ANALYSIS RESULTS
DEAD LOAD, FINAL PRESTRESS, 125% DESIGN PRESSURE, 125% OF 1/2 SAFE SHUTDOWN EARTHQUAKE, AND THERMAL ACCIDENT
($D + F_F + 1.25P + 1.25E + T_A$)

PORTION	SECTION	CONCRETE		LINER		REINFORCING STRESS				SECTION RESULTANTS					SHEAR CAPACITY	DEFLECTION
						OUTSIDE		INSIDE								
		MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. FORCE K/FT.	HOOP FORCE K/FT.	MER. MOMENT K FT./FT.	HOOP MOMENT K FT./FT.	RADIAL SHEAR K/FT.		IN.
ALLOW.		−4950 −4500	−4950 −4500	STRAIN	GOVERNS	±54000	±54000	±54000	±54000	–	–	–	–	–	± K/FT	–
DOME	A B	−869	−620	−21400	−20856	15600	14283	–	–	−125	−123	−227	−216	−1	37	0.84
DOME	C D	−1050	−336	−19300	−19204	45500	24493	–	–	−135	−65	−328	−169	−18	37	1.13
DOME	E F	−741	−1798	−6892	−23763	−1576	11274	7843	−6778	−483	−191	854	−1358	−17	255	0.68
WALL	G H	−1243	−1192	−24156	−23788	20448	16946	–	–	−116	−195	−494	−378	−13	107	0.64
WALL	J K	−928	−385	−24033	−19130	17404	21603	–	–	−171	−75	−314	−216	1	26	0.62
WALL	L M	−849	−374	1348	−4129	−6146	10739	−524	−3380	−278	−74	289	−351	64	166	0.14
BASE SLAB	N O	−333	−569	−4470	7770	5780	12800	−2480	−4960	−65	−55	−567	−789	145	497	−0.57
BASE SLAB	P Q	−1090	−876	−14900	−12350	24490	20030	−9510	−6630	−152	−91	−1720	−1218	48	497	−0.77
BASE SLAB	R S	−768	−109	−9020	−2620	4000	8690	−7950	540	−396	116	−1270	−165	−173	497	−0.83

TABLE 3.8-2 (SHEET 8 OF 8)

STRESS ANALYSIS RESULTS
DEAD LOAD, FINAL PRESTRESS, DESIGN PRESSURE, SAFE SHUTDOWN EARTHQUAKE, AND THERMAL ACCIDENT
$(D + F_F + P + E' + T_A)$

PORTION	SECTION	CONCRETE		LINER		REINFORCING STRESS				SECTION RESULTANTS					SHEAR CAPACITY	DEFLECTION
						OUTSIDE		INSIDE								
		MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. PSI	HOOP PSI	MER. FORCE K/FT.	HOOP FORCE K/FT.	MER. MOMENT K FT./FT.	HOOP MOMENT K FT./FT.	RADIAL SHEAR K/FT.		IN.
ALLOW.		−4950 −4500	−4950 −4500	STRAIN	GOVERNS	±54000	±54000	±54000	±54000	–	–	–	–	–	± K/FT	–
DOME	A B	−963	−905	−23900	−23875	13100	11942	–	–	−173	−169	−279	−269	0	37	0.49
DOME	C D	−1280	−396	−19300	−19195	42700	23731	–	–	−169	−73	−358	−176	−19	37	0.85
DOME	E F	−693	−1742	−9279	−24387	−1327	12221	6397	−5534	−487	−181	541	−184	−50	255	0.58
WALL	G H	−1493	−1258	−24584	−23995	19391	16832	–	–	−151	−203	−529	−390	−8	107	0.63
WALL	J K	−1182	−671	−23240	−21378	15145	19263	–	–	−217	−119	−352	−274	0	25	0.20
WALL	L M	−453	−383	−173	−4237	−4225	11106	−2431	−2889	−324	−59	−63	−338	42	165	−0.04
BASE SLAB	N O	−337	−472	−4290	−6410	3960	10270	−2510	−4100	−112	−82	−530	−759	126	497	−0.17
BASE SLAB	P Q	−1017	−858	−13720	−12000	21420	18840	−8870	−6400	−176	−109	−1609	−1176	49	497	−0.16
BASE SLAB	R S	−746	–	−8740	−860	3680	11200	−7720	2290	−396	84	−1222	−191	−147	497	−0.13

TABLE 3.8-3 (SHEET 1 OF 6)

CONTAINMENT STRAINS (X 10^{-6})
DEAD LOAD AND INITIAL PRESTRESS
($D + F_I$)

PORTION	SECTION	CONCRETE STRAIN		LINER STRAIN		REINFORCEMENT STRAIN			
						OUTSIDE		INSIDE	
		MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN
ALLOWABLE		(STRESS	GOVERNS)	–2000 +1000	–2000 +1000	(STRESS	GOVERNS)	(STRESS	GOVERNS)
DOME	AB	–280	–267	–230	–226	–280	–267	–	–
	CD	–347	–51	–259	1	–156	–51	–	–
	EF	–126	–39	–120	–39	–121	–17	–117	–36
WALL	GH	–128	–142	–128	–144	–122	–134	–	–
	JK	–119	–261	–119	–263	–118	–246	–	–
	LM	–113	–48	–79	–48	–113	–44	–79	–48
BASE SLAB	NO	–152	–58	–196	–74	337	117	–152	–58
	PQ	–64	–25	–71	–26	29	–4	–64	–25
	RS	–72	–28	23	2	–72	–29	10	–1

TABLE 3.8-3 (SHEET 2 OF 6)

CONTAINMENT STRAINS (X 10^{-6})
DEAD LOAD, FINAL PRESTRESS, AND 115% DESIGN PRESSURE
($D + F_F + 1.15P$)

PORTION	SECTION	CONCRETE STRAIN		LINER STRAIN		REINFORCEMENT STRAIN			
						OUTSIDE		INSIDE	
		MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN
ALLOWABLE		(STRESS	GOVERNS)	–2000 +1000	–2000 +1000	(STRESS	GOVERNS)	(STRESS	GOVERNS)
DOME	AB	–84	–78	–43	–34	–84	–78	–	–
	CD	–167	28	–145	29	16	23	–	–
	EF	–192	–64	114	–65	–180	–37	72	–62
WALL	GH	–51	–32	–53	–32	–43	–31	–	–
	JK	–78	–2	–75	–3	–78	–2	–	–
	LM	–260	–63	161	–65	–260	–55	166	–64
BASE SLAB	NO	9	–71	10	–88	–1	113	9	–71
	PQ	–216	–167	–294	–227	663	512	–216	–167
	RS	–30	3	–33	4	10	2	–30	3

TABLE 3.8-3 (SHEET 3 OF 6)

CONTAINMENT STRAINS (X 10^{-6})
DEAD LOAD, FINAL PRESTRESS, DESIGN PRESSURE, AND THERMAL ACCIDENT
($D + F_F + P + T_A$)

PORTION	SECTION	CONCRETE STRAIN		LINER STRAIN		REINFORCEMENT STRAIN			
						OUTSIDE		INSIDE	
		MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN
ALLOWABLE		(STRESS	GOVERNS)	–2000 +1000	–2000 +1000	(STRESS	GOVERNS)	(STRESS	GOVERNS)
DOME	AB	322	347	352	377	202	162	–	–
	CD	1220	564	295	546	1220	568	–	–
	EF	714	195	783	144	–436	196	636	157
WALL	GH	444	367	207	362	444	355	–	–
	JK	295	458	259	456	295	441	–	–
	LM	–399	139	444	136	–399	127	232	137
BASE SLAB	NO	176	140	172	116	151	336	176	140
	PQ	13	78	–49	28	571	498	13	78
	RS	42	311	19	304	81	167	42	311

TABLE 3.8-3 (SHEET 4 OF 6)

CONTAINMENT STRAINS (X 10^{-6})
DEAD LOAD, FINAL PRESTRESS, 150% DESIGN PRESSURE, AND THERMAL ACCIDENT
($D + F_F + 1.5P + T_A$)

PORTION	SECTION	CONCRETE STRAIN		LINER STRAIN		REINFORCEMENT STRAIN			
						OUTSIDE		INSIDE	
		MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN
ALLOWABLE		(STRESS	GOVERNS)	–5000 +3000	–5000 +3000	(STRESS	GOVERNS)	(STRESS	GOVERNS)
DOME	AB	540	563	564	594	433	381	–	–
	CD	1440	648	451	567	1440	652	–	–
	EF	920	152	1010	81	–466	152	813	95
WALL	GH	561	410	423	407	561	397	–	–
	JK	481	659	420	659	481	633	–	–
	LM	–539	121	615	110	–539	112	481	113
BASE SLAB	NO	165	130	143	89	336	520	165	130
	PQ	–53	78	–145	19	839	591	–53	78
	RS	48	305	26	299	80	153	48	305

TABLE 3.8-3 (SHEET 5 OF 6)

CONTAINMENT STRAINS (X 10^{-6})
DEAD LOAD, FINAL PRESTRESS, 125% DESIGN PRESSURE, 125% OF 1/2 SAFE SHUTDOWN EARHTQUAKE, AND THERMAL ACCIDENT
($D + F_F + 1.25P + 1.25E + T_A$)

PORTION	SECTION	CONCRETE STRAIN		LINER STRAIN		REINFORCEMENT STRAIN			
						OUTSIDE		INSIDE	
		MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN
ALLOWABLE		(STRESS	GOVERNS)	−5000 +3000	−5000 +3000	(STRESS	GOVERNS)	(STRESS	GOVERNS)
DOME	AB	406	428	434	458	289	243	–	–
	CD	1320	590	334	546	1320	595	–	–
	EF	1168	164	1530	100	−499	164	779	113
WALL	GH	480	372	310	367	480	360	–	–
	JK	374	540	328	540	374	521	–	–
	LM	−468	122	483	120	−468	115	301	122
BASE SLAB	NO	175	90	166	51	205	449	175	90
	PQ	−59	42	−154	−30	855	701	−59	42
	RS	4	293	−28	272	159	321	4	293

TABLE 3.8-3 (SHEET 6 OF 6)

CONTAINMENT STRAINS (X 10^{-6})
DEAD LOAD, FINAL PRESTRESS, DESIGN PRESSURE, SAFE SHUTDOWN EARHTQUAKE, AND THERMAL ACCIDENT
($D + F_F + P + E' + T_A$)

PORTION	SECTION	CONCRETE STRAIN		LINER STRAIN		REINFORCEMENT STRAIN			
						OUTSIDE		INSIDE	
		MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN	MER. IN/IN	HOOP IN/IN
ALLOWABLE		(STRESS	GOVERNS)	−5000 +3000	−5000 +3000	(STRESS	GOVERNS)	(STRESS	GOVERNS)
DOME	AB	322	347	352	377	202	162	–	–
	CD	1220	564	295	546	1220	568	–	–
	EF	1115	195	1485	144	–484	196	719	157
WALL	GH	445	368	208	363	445	356	–	–
	JK	295	457	259	456	295	441	–	–
	LM	–402	138	444	137	–402	128	232	138
BASE SLAB	NO	174	116	170	88	142	360	174	116
	PQ	–34	42	–117	–25	749	660	–34	42
	RS	13	352	–17	328	148	407	13	352

TABLE 3.8-4 (SHEET 1 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA

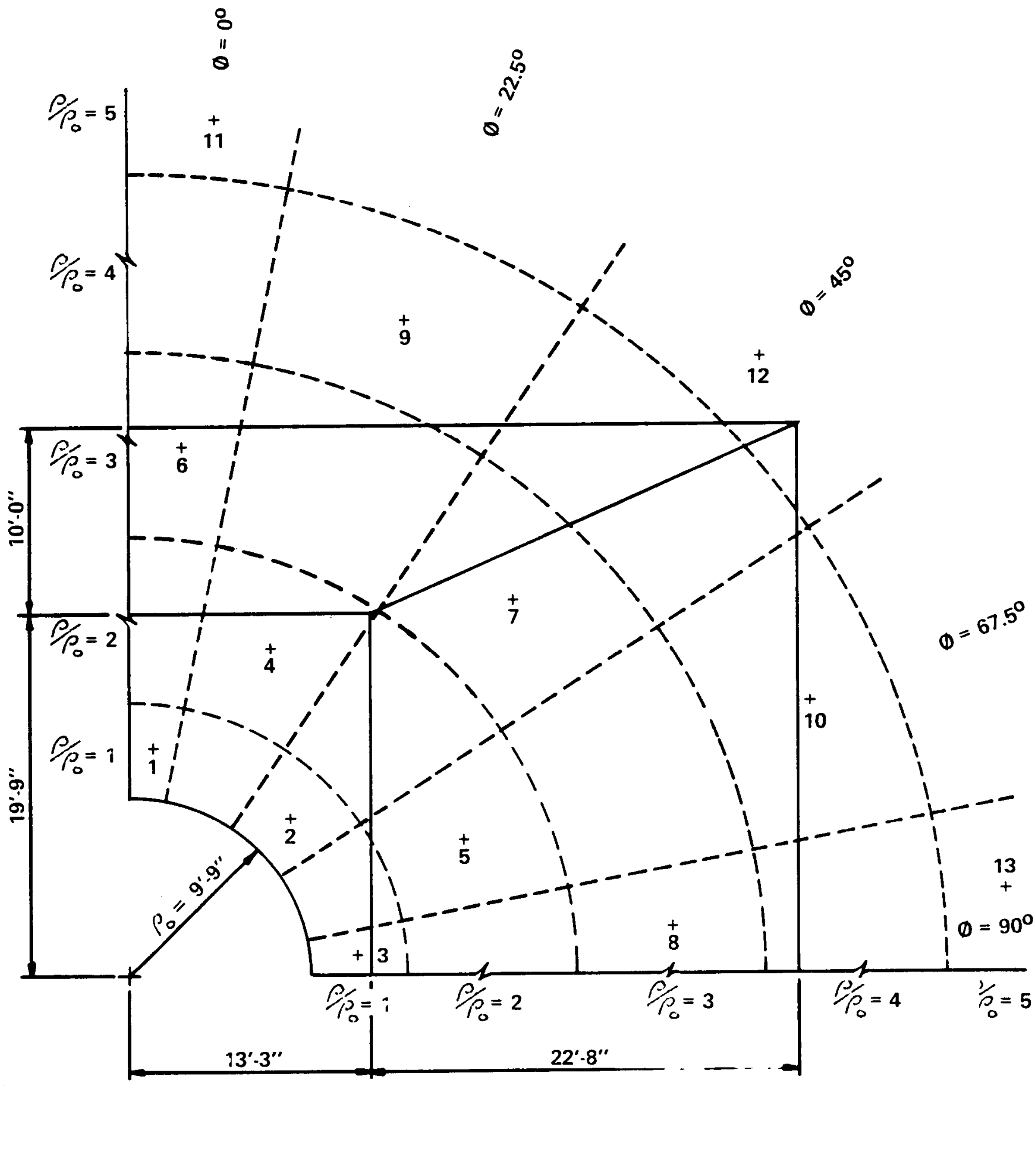


TABLE 3.8-4 (SHEET 2 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA

NOTATION AND NOTES FOR STRESS AND STRAIN TABLES

NOTATION:

D	=	DEAD LOAD
F_F	=	MINIMUM GUARANTEED PRESTRESS LEVEL
F_I	=	INITIAL PRESTRESS LEVEL
P	=	INTERNAL PRESSURE
E	=	1/2 SAFE SHUTDOWN EARTHQUAKE
E^I	=	SAFE SHUTDOWN EARTHQUAKE
T_A	=	ACCIDENT TEMPERATURE

NOTES:

THIRTEEN POINTS, POSITIONED RADIALLY ABOUT THE CENTER OF THE EQUIPMENT HATCH, ARE SELECTED FOR TABULATION. THIS COVERS THE LOCAL AREA WHERE STRESS CONCENTRATIONS DUE TO THE PRESENCE OF A LARGE OPENING CAN BE FELT.

EQUIPMENT HATCH POLAR COORDINATE SYSTEM IS USED FOR LOCATIONS 1, 2, & 3. RADIAL STRESS (MERIDIONAL IN THE TABLE) IN THESE LOCATIONS IS NEGLIGIBLE, AND THUS IS NOT TABULATED.

CONTAINMENT GLOBAL CYLINDRICAL COORDINATE SYSTEM IS USED FOR OTHER LOCATIONS.

FOR CONCRETE, ONLY COMPRESSIVE STRESS IS CONSIDERED.

THE LINER PLATE YIELDS IN SOME LOADING COMBINATIONS. STRAIN WILL GOVERN THE DESIGN. THEREFORE, MAXIMUM STRAIN IS ALSO TABULATED FOR THE LINER PLATE.

TABLE 3.8-4 (SHEET 3 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA
DEAD LOAD AND INITIAL PRESTRESS
(D + F_I)

LOCATION	CONCRETE STRESS		REINFORCING STRESS				LINER STRESS & STRAIN		
			OUTSIDE		INSIDE				
	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	STRAIN ($\times 10^{-6}$) IN/IN
ALLOWABLE	−3300		±20000				–		−2000 +1000
1	–	−3558	–	−1357	–	−10739	–	−22166	−741
2	–	−2879	–	−2077	–	−9784	-	−17936	−600
3	–	−1954	–	−2663	–	−7418	–	−12173	−407
4	−714	−2296	−906	1760	−4013	−12491	−4448	−14304	−447
5	−1141	−1282	−2316	−540	−6398	−6969	−7108	−7987	−217
6	−577	−2161	−3008	−4838	−3415	−11853	−3595	−13463	−425
7	−918	−1648	−2717	−3719	−5175	−9140	−5719	−10267	−303
8	−897	−1017	−3920	−5531	−5173	−6046	−5588	−6336	−173
9	−832	−1875	−4794	−7119	−4982	−10508	−5183	−11681	−354
10	−823	−1836	−6588	−10821	−5281	−6893	−5127	−6149	−346
11	−901	−1984	−4484	−6433	−5258	−10901	−5613	−12360	−374
12	−850	−1520	−5606	−9097	−5229	−7997	−5296	−7974	−279
13	−829	−1867	−7197	−10789	−5454	−6821	−5165	−5981	−164

TABLE 3.8-4 (SHEET 4 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA DEAD LOAD, FINAL PRESTRESS, AND 115% DESIGN PRESSURE ($D + F_F + 1.15P$)

LOCATION	CONCRETE STRESS		REINFORCING STRESS				LINER STRESS & STRAIN		
			OUTSIDE		INSIDE				
	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	STRAIN ($\times 10^{-6}$) IN/IN
ALLOWABLE	−3300		±20000				–		−2000 +1000
1	–	−852	–	17308	–	5410	–	−9099	−304
2	–	−913	–	5681	–	−1617	–	−9751	−326
3	–	−1380	–	−439	–	−7147	–	−14738	−493
4	−180	−257	−1695	2175	−154	−2073	−732	−2861	−78
5	−522	–	−466	6368	−4548	2856	−4965	892	−186
6	−370	−259	−3096	−2117	6587	−2444	8854	−139	−113
7	−280	−68	−2676	−353	−1899	−619	−1965	−726	−170
8	−363	–	186	3962	−2900	2612	−3412	2373	−130
9	−211	−173	−2024	−1150	−1846	−1572	−1997	−1848	−54
10	−292	−117	−2814	−1072	−2619	−170	−2703	−558	−86
11	−223	−190	−1575	−258	−2043	−1526	−2382	−2029	−65
12	−266	−96	−2541	−925	−2270	−915	−2435	−1004	−74
13	−265	−438	−3062	−1839	−2660	17562	2293	7978	−94

TABLE 3.8-4 (SHEET 5 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA DEAD LOAD, FINAL PRESTRESS, DESIGN PRESSURE, AND THERMAL ACCIDENT ($D + F_F + P + T_A$)

LOCATION	CONCRETE STRESS		REINFORCING STRESS				LINER STRESS & STRAIN		
	MER. STRESS PSI	HOOP STRESS PSI	OUTSIDE MER. STRESS PSI	OUTSIDE HOOP STRESS PSI	INSIDE MER. STRESS PSI	INSIDE HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	STRAIN ($\times 10^{-6}$) IN/IN
ALLOWABLE	−3300		±30000				–		−2000 +1000
1	–	−3560	–	17485	–	1414	–	−24000	−296
2	–	−3480	–	9392	–	−3505	–	−24000	−268
3	–	−4046	–	4080	–	−9330	–	−24000	−470
4	−2200	−2645	789	4112	−3455	−7083	−24000	−24000	196
5	−2945	−2248	4901	9576	−9340	−2832	−24000	−24000	92
6	−2121	−2657	2693	3036	−2255	−6630	−22652	−24000	185
7	−2583	−2459	1746	3699	−6352	−5047	−24000	−24000	237
8	−2816	−2524	7958	9148	−6905	−4353	−24000	−24000	159
9	−2783	−2567	3794	4717	−6963	−5056	−24000	−24000	174
10	−2699	−2531	2920	3230	−6474	−5097	−24000	−24000	201
11	−2755	−2553	4116	4124	−6682	−5054	−24000	−24000	183
12	−2704	−2591	2333	4667	−6476	−5253	−24000	−24000	204
13	−2661	−2383	2381	2498	−6136	−3976	−24000	−24000	203

TABLE 3.8-4 (SHEET 6 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA
DEAD LOAD, FINAL PRESTRESS, 150% DESIGN PRESSURE, AND THERMAL ACCIDENT
$(D + F_F + 1.5P + T_A)$

LOCATION	CONCRETE STRESS		REINFORCING STRESS				LINER STRESS & STRAIN		
			OUTSIDE		INSIDE				
	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	STRAIN ($x10^{-6}$) IN/IN
ALLOWABLE	–4950		±54000				–		–5000 +1000
1	–	–	–	21828	–	29609	–	9330	1286
2	–	–2876	–	16687	–	3056	–	–24000	–52
3	–	–3973	–	6479	–	–7813	–	–24000	–444
4	–1921	–	293	3583	–1041	3969	–24000	–24000	390
5	–2735	–	7716	17693	–7226	14380	–24000	–16883	730
6	–	–	3235	5525	21275	5889	1692	–13773	1155
7	–2266	–	1657	11431	–3693	8670	–24000	20835	559
8	–2720	–	19157	20520	–4295	160	–24000	–24000	342
9	–2530	–	8196	15006	–4141	5543	–24000	–24000	224
10	–2331	–2326	8182	27045	–2591	1054	–24000	–24000	317
11	–2515	–	9361	20807	–3797	7141	–24000	–24000	408
12	–2281	–2270	4791	19109	–2717	51	–24000	–24000	330
13	–2314	–	10763	22628	–1750	21204	–24000	–5433	1032

TABLE 3.8-4 (SHEET 7 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA DEAD LOAD, FINAL PRESTRESS, 125% DESIGN PRESSURE, 125% OF 1/2 SAFE SHUTDOWN EARTHQUAKE, AND THERMAL ACCIDENT ($D + F_F + 1.25P + 1.25E + T_A$)

LOCATION	CONCRETE STRESS		REINFORCING STRESS				LINER STRESS & STRAIN		
			OUTSIDE		INSIDE				
	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	STRAIN ($\times 10^{-6}$) IN/IN
ALLOWABLE	−4950		±54000				–		−5000 +1000
1	–	−3052	–	24321	–	7352	–	−24000	−115
2	–	−3203	–	12396	–	−669	–	−24000	−169
3	–	−4062	–	4155	–	−9376	–	−24000	−476
4	−2101	−2250	567	3997	−2598	−3610	−22439	−24000	329
5	−2890	−1947	5400	14150	−8817	261	−24000	20794	88
6	−1971	−2321	3799	4621	−856	−3618	−21050	−24000	294
7	−2487	−2226	1571	8251	−5561	−2495	−24000	−23774	254
8	−2807	−2472	11089	15546	−6321	−2883	−24000	−24000	158
9	−2674	−2424	4615	10341	−5954	−2848	−24000	−24000	202
10	−2568	−2606	3517	14733	−5332	−3489	−24000	−24000	238
11	−2651	−2479	5310	12891	−5637	−2762	−24000	−24000	214
12	−2562	−2672	2745	11830	−5299	−3716	−24000	−24000	249
13	−2526	−2456	3080	15021	−4950	−1972	−24000	−24000	257

TABLE 3.8-4 (SHEET 8 OF 8)

CONTAINMENT STRESSES IN EQUIPMENT HATCH AREA DEAD LOAD, FINAL PRESTRESS, DESIGN PRESSURE, SAFE SHUTDOWN EARTHQUAKE, AND THERMAL ACCIDENT ($D + F_F + P + E' + T_A$)

LOCATION	CONCRETE STRESS		REINFORCING STRESS				LINER STRESS & STRAIN		
			OUTSIDE		INSIDE				
	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	MER. STRESS PSI	HOOP STRESS PSI	STRAIN ($x10^{-6}$) IN/IN
ALLOWABLE	–4950		±54000				–		–5000 +3000
1	–	–3557	–	18296	–	1863	–	–24000	–295
2	–	–3488	–	9137	–	–3670	–	–24000	–271
3	–	–4112	–	3174	–	–10090	–	–24000	–494
4	–2237	–2638	780	4131	–3783	–7016	–23890	–24000	201
5	–3007	–2253	4609	9649	–9921	–2868	–24000	–24000	70
6	–2157	–2663	2309	2908	–2610	–6695	–23037	–24000	186
7	–2639	–2455	1573	3785	–6846	–5008	–24000	–24000	217
8	–2866	–2530	7053	9237	–7454	–4384	–24000	–24000	141
9	–2833	–2565	3425	4531	–7420	–5081	–24000	–24000	156
10	–2785	–2527	2714	3244	–7182	–5065	–24000	–24000	170
11	–2798	–2551	3752	4018	–7081	–5056	–24000	–24000	167
12	–2772	–2596	2060	4652	–7054	–5298	–24000	–24000	180
13	–2758	–2373	2135	2401	–6931	–3913	–24000	–24000	168

TABLE 3.8-5

AGGREGATE TESTS

ASTM No.	Title	Results to Be Achieved	Initial Test	User's Test	Daily Test
C-33	Gradation	To conform with spec	X		X
C-40	Organic impurities	To conform with spec	X		X
C-87	Mortar making properties	To conform with spec	X		
C-88	Soundness	To conform with spec	X	X	
C-117	Specific gravity and No. 200 sieve	Design mix calculations	X		
C-127	Specific gravity and absorption (fine aggregates)	Design mix calculations	X		
C-128	Specific gravity and absorption (fine aggregates)	Design calculations	X		
C-131	Los Angeles abrasion	To conform with spec	X	X	
C-136	Sieve analysis	To conform with spec	X		
C-142	Clay lumps	To conform with spec	X		
C-227	Potential reactivity (mortar bar)	To conform with spec	X		
C-289	Potential reactivity (chemical)	To conform with spec	X	X	
C-295	Petrographic	To conform with spec	X		

TABLE 3.8-6

CEMENT TESTS

ASTM No.	Type of Test	Initial Test	User's Test	Periodic Tests
C-109	Compressive strength	X	X	X
C-114	Chemical analysis	X	X	
C-115	Fineness-turbidimeter	X	X	
C-151	Autoclave expansion (Soundness)	X	X	

TABLE 3.8-7

FLY ASH TESTS

ASTM No.	Type of Test	Initial Test	User's Test	Periodic Tests
C-109	Compressive strength	X	X	X
C-114	Chemical analysis	X	X	
C-151	Autoclave expansion(Soundness)	X	X	
C-188	Specific gravity	X	X	
C-311	Sampling and testing	X	X	

TABLE 3.8-8

PRESTRESSING SEQUENCES

Phase	Hoop	Numbers of Tendons Dome	Vertical	Total	Description
1	57	-	-	57	Between the top of base slab and approximately 30 ft below the bottom of the ring grider.
2	6	-	-	63	Between approximately 30 ft below the bottom of the ring girder and the uppermost tendon.
3	-	-	33	96	Every fourth tendon of three vertical tendon groups at 120 degrees apart.
4	-	-	33	129	Repeat Phase 3 with the tendons immediately adjacent to the last tendons.
5	-	45	-	174	Stressing every other tendon on alternate sides of the center ones moving outward.
6	-	-	33	207	Continue stressing every fourth tendon of three vertical tendon groups at 120 degrees apart.
7	-	-	31	238	Stressing the remaining vertical tendons.
8	54	-	-	292	The remaining 50 percent of tendons specified in Phase 1.
9	18	-	-	310	The remaining 75 percent of the tendons specified in Phase 2.
10	-	48	-	358	Stressing the remaining out-most dome tendons toward the center of each group.
Total	135	93	130	358	

TABLE 3.8-9

CALCULATED RESULTS - INTERNAL STRUCTURES

DESCRIPTION OF MEMBER	LOCATION OF MEMBER	LOAD COMBINATION	TOTAL CALCULATED REINF. STEEL STRESS KSI	ALLOWABLE REINF. STEEL STRESS KSI	REMARKS
5 ft 0 in. concrete wall	Reactor cavity wall from EL 79 ft 0 in. to EL 102 ft 0 in.	1.0 D + 1.0 L + 1.0 To 1.0 D + 1.0 L + 1.0 To + 1.0 P	19.5 28.2	20 54	WSD USD
9 ft 8 in. concrete wall	Primary shield wall from EL 102 ft 0 in. to EL 129 ft 0 in.	1.0 D + 1.0 P + 1.25 E 1.0 D + 1.25 To + 1.25E 1.0 D + 1.25 To + 1.0E' + 1.0P	49.3	54	USD
2 ft 6 in. concrete wall	Secondary shield wall from EL 104 ft 0 in. to EL 125 ft 9 in.	1.0 D + 1.0 R	47.6	54	USD
3 ft 9 in. and 5 ft 0 in. concrete wall	Refueling cavity wall	1.0 D + 1.0 E'	33.9	54	USD
3 ft 3 in. concrete slab	Slab at EL 129 ft 0 in.	1.0 D + 1.0 L + 1.0 E 1.0 D + 1.0 L ± 1.0 E' + 1.0 P	25.4 43.0	32 54	WSD USD
3 ft 6 in. concrete wall	Secondary shield wall from EL 129 ft 0 in. to EL 152 ft 0 in.	1.0 D + 1.0 L 1.0 D + 1.0 L + 1.0 P 1.0 D + 1.0 L + 1.0 R	18.3 44.3 42.5	24 54 54	WSD USD USD
3 ft 0 in. concrete slab	Slab at EL 155 ft 0 in.	1.0 D + 1.0 L + 1.0 E 1.0 D + 0.1 L + 1.0 E' + 1.0 R	23.0 40.4	32 54	WSD USD
2 ft 0 in. and 3 ft 0 in. concrete wall	Secondary shield wall above	1.0 D + 1.0 P	44.8	54	USD

D = Dead load
T = Operating temperature load
E = 1/2 safe shutdown earthquake load
R = Pipe rupture load
L = Live load
P = Pressure load
E'= Safe shutdown earthquake load
WSD = Working stress design
USD = Ultimate strength design

TABLE 3.8-10 (SHEET 1 OF 2)

CALCULATED RESULTS - AUXILIARY BUILDING

Description Of Member	Location Of Member	Load Combination	Total Calculated Stress, Or Required Capacity	Allowable Stress, Or Maximum Capacity	Remarks
3-ft 6-in. diam. concrete column	Cols for cask crane along Col. line V from EL 100-ft 0-in. to 146-ft 6-in. south-east quadrant	D + L + E	P= 498 K	P_m = 1970 K	WSD
4-ft 6-in. diam. concrete column	Spent fuel pool support between Col lines O and T, and 2 and 9.8 – EL 100-ft 0-in.	D + L + E	P= 3153 K	P_m = 3730 K	WSD
2-ft 6-in. diam. concrete column	Cols supporting electric penetration rooms from EL 100-ft 0-in. to 175-ft 0-in.	D + L + E	P= 753 K	P_m = 947 K	WSD
2-ft 3-in. by 5-ft 6-in. concrete beam	Floor in demineralizer area at Col. line 14, between Col. lines M and N - EL 139-ft 0-in.	D + L + E + Pipe	M= 1885 ft-K	M_m = 2000 ft-K	WSD
2-ft 0-in. by 4-ft 0-in. concrete beam	Floor in hot machine shop between Col. Lines 18 and 19, and R and U- EL 155-ft 0-in.	D + L + E	M= 665.5 ft-K	M_m = 725 ft-K	WSD

D	=	Dead load	M	=	Maximum moment
L	=	Live load	P	=	Maximum load
E	=	Earthquake load	WSD	=	Working stress design
M	=	Missile load	H	=	Hydrostatic load
S	=	Surcharge load	W	=	Tornado load

TABLE 3.8-10 (SHEET 2 OF 2)

Description Of Member	Location Of Member	Load Combination	Total Calculated Stress, Or Required Capacity	Allowable Stress, Or Maximum Capacity	Remarks
2-ft 0-in. concrete slab	Floor in radwaste filter room between Col. Lines J and N, and 17 and 18 - EL 139-ft 0-in.	D + L + E + Pipe	M= 12.9 ft-K	M_m = 29.8 ft-K	WSD
2-ft 0-in. by 4-ft 0-in. concrete beam	Floor in hot machine shop between Col. Lines 18 and 19, and R and U - EL 155-ft 0-in.	D + L + E	M= 665.5 ft-K	M_m = 725 ft-K	WSD
2-ft 0-in. concrete slab	Floor in radwaste filter room between Col. Lines J and N, and 17 and 18 - EL139-ft 0-in.	D + L + E + Pipe	M= 12.9 ft-K	M_m = 29.8 ft-K	WSD
2-ft 0-in. concrete roof slab	Roof slabs	D + L + M	M= 43.91 ft-K	M_m = 59.8 ft-K	WSD
5-ft 0-in. concrete slab	Floor in cask wash area between Col. lines P and Q, and 7 and 9.3 - EL 139-ft 0-in.	D + L + 1000 K (Impact)	M= 80.2 ft-K	M_m = 143 ft-K	WSD
3-ft 6-in. concrete wall	Wall below EL 121-ft 0-in. Col. lines P northeast quadrant - east wall	D + L + H + S	M+ 235 ft-K	M_m = 274 ft-K	WSD
2-ft 0-in. concrete wall	Wall above EL 155-ft 0-in. southwest quadrant – west wall, typical	D + L + W_t + M_1	M= 20.5 ft-K	M_m = 36 ft-K	WSD

D = Dead load
L = Live load
E = Earthquake load
M = Missile load
H = Hydrostatic load

M = Maximum moment
P = Maximum load
WSD = Working stress design
S = Surcharge load
W = Tornado load

TABLE 3.8-11

CALCULATED RESULTS - DIESEL GENERATOR BUILDING

Description Of Member	Location Of Member	Load Combination	Total Calculated Stress, Or Required Capacity	Allowable Stress, Or Maximum Capacity	Remarks
Caissons (Typ.) 5-ft 0-in 54-ft 0-in. long	Top elevation 151-ft 0-in.	1.0 D + 0.5 L + 1.0 E	P_u = 330 K	P_u = 450 K	USD
			M_u = 2480 K	M_u = 3250 K	USD
Slab 4-ft 0-in. thick 33-ft 0-in. Span	Ground Slab on caissons EL 155-ft 0-in.	1.0 D + 0.5 L + 1.0 E	M_u = 277 K	M_u = 345 K	USD
Exterior walls 2-ft 6-in. thick, 20-ft 0-in. span	Exterior walls	1.0 D + 0.5 L + 1.0 W_T	M_u = 94 ft-K	M_u = 350 ft-K	USD
Interior walls 1-ft 6-in. thick, 20-ft 0-in. span	Interior walls	1.0 D + 0.5 L + 1.0 E	M_u = 31 ft-K	M_u = 40 ft-K	USD
Roof 2-ft 0-in. thick 33-ft 0-in. span	Roof EL 177-ft 0-in.	1.0 D + 0.5 L + 1.0 W_T	M = 78 ft-K	M_u = 175 ft-K	USD

D = Dead load
L = Live load strength
E = Earthquake load design
USD = Ultimate
W = Tornado wind load
P = Ultimate load
M = Ultimate moment

TABLE 3.8-12

CALCULATED RESULTS - RIVER INTAKE STRUCTURE

Description Of Member	Location of Member	Load Combination	Total Calculated Stress, Or Required Capacity	Allowable Stress, Or Ultimate Capacity	Remarks
Base slab 6 ft 0 in. thick	EL 61 ft 6 in.	D + L + H + S	M_b = 3859 ft-K	M_m = 4107 ft-K	WSD
			M_t = 2538 ft-K	M_m = 2590 ft-K	
Base slab Bay area 3 ft 0 in. thick	EL 64 ft 0 in.	D + L + H + S	M_b = 635 ft-K	M_b = 676 ft-K	WSD
Bay area 3 ft 0 in. thick			M_t = 267 ft-K	M_t = 436 ft-K	
Walls	Bays area	D + L + H + S	M = 380 ft-K	M_m = 443 ft-K	WSD
Walls 3 ft 0 in. thick	Exterior walls	D + L + H + S	M = 119 ft-K	M_m - 147 ft-K	WSD
Roof Slab 2 ft 0 in. thick	EL 128 ft 0 in	D + L	M = 43 ft-K	M_m = 45 ft-K	WSD

D = Dead load
L = Live load
H = Hydrostatic load
S = Surcharge load

Mb = Moment at bottom
Mt = Moment at top
Mm = Maximum moment
WSD = Working stress design

TABLE 3.8-13

CALCULATED RESULTS - INTAKE STRUCTURE AT STORAGE POND

Description Of Member	Location of Member	Load Combination	Total Calculated Stress, Or Required Capacity	Allowable Stress, Or Ultimate Capacity	Remarks
Base slab 3 ft 0 in. thick	EL 151 ft 6 in.	D + L + H + S	M = 43 ft-K	M_m = 72 ft-K	WSD
Exterior walls 4 ft 0 in. thick	See figure 3.8-28	D + L + H + S	M = 176 ft-K	M_m = 242 ft-K	WSD
Columns Typically 3 ft 0 in.	See figure 3.8-28	D + L	M = 59 ft-K	M_m = 381 ft-K	WSD
			P = 304 ft-K	P_m = 1414 ft-K	WSD
Floor slab 3 ft 0 in. thick	Operating floor	D + L	M_b = 61 ft-K M_t = 161 ft-K	M_m = 88 ft-K M_m = 176 ft-K	WSD
Roof slab 2 ft 6 in. thick	See figure 3.8-28	D + L + W	M = 59 ft-K	M_m = 74 ft-K	WSD

D = Dead load
L = Live load
H = Hydrostatic load
S = Surcharge load
W = Tornado wind load

M_b = Moment at bottom
M_t = Moment at top
M_m = Maximum moment
P_m = Maximum load
WSD = Working stress design

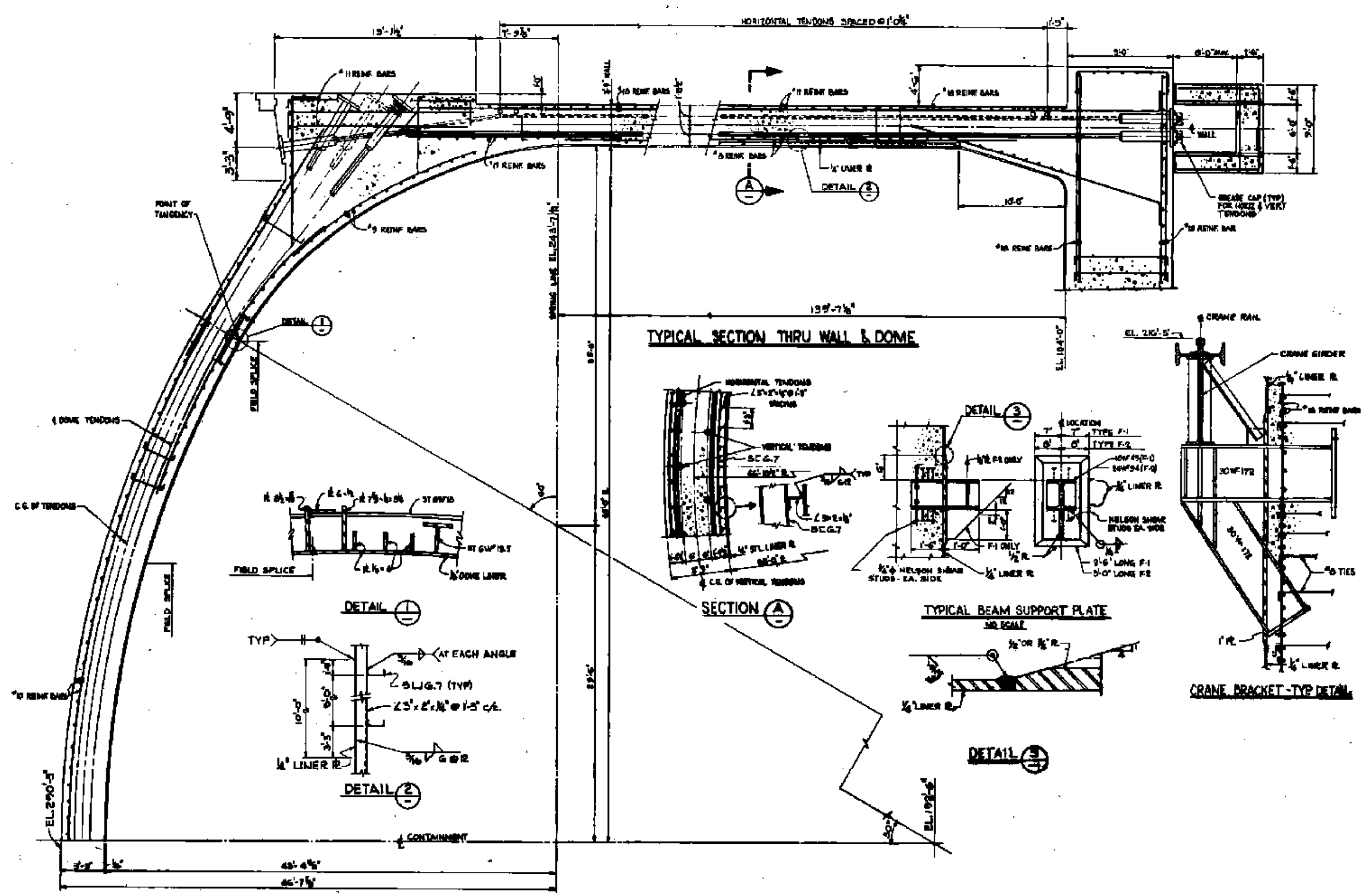
TABLE 3.8-14

CALCULATED RESULTS - ELECTRICAL CABLE TUNNELS

Description Of Member	Location of Member	Load Combination	Total Calculated Stress, Or Required Capacity	Allowable Stress, Or Ultimate Capacity	Remarks
Base slab 2-ft 0-in. thick	Electrical cable tunnels connecting intake structure with diesel generator building and auxiliary building	(D + L + E) 0.75	M + 50 ft-K	M_m = 60 ft-K	WSD
Walls 1-ft 6-in. thick	Electrical cable tunnels connecting intake structure with diesel generator building and auxiliary building	(D + L + E) 0.75	M = 32 ft-K	M_m = 34 ft-K	WSD
Roof slab 2-ft 0-in. thick	Electrical cable tunnels connecting intake structure with diesel generator building and auxiliary building	(D + L + E) 0.75	M = 36 ft-K	M_m = 60 ft-K	WSD

D = Dead load
L = Live load
E = Earthquake load

M = Maximum moment
WSD = Working stress design



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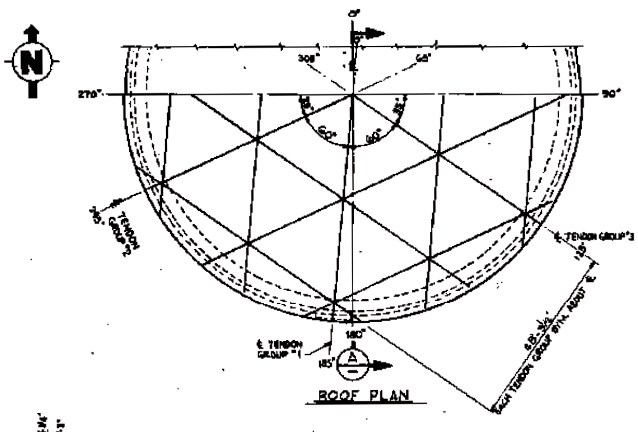
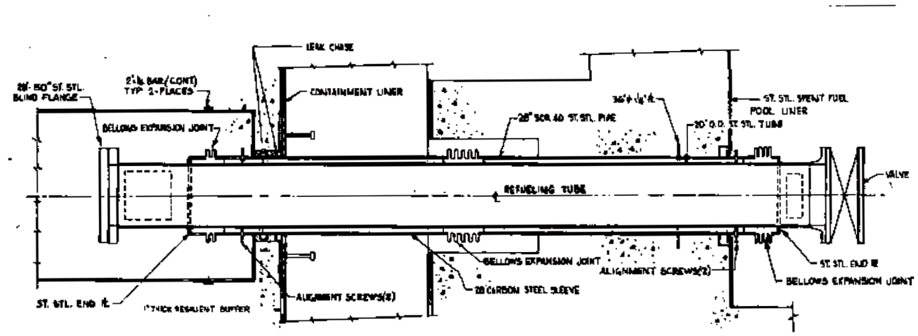
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT TYPICAL SECTIONS AND DETAILS

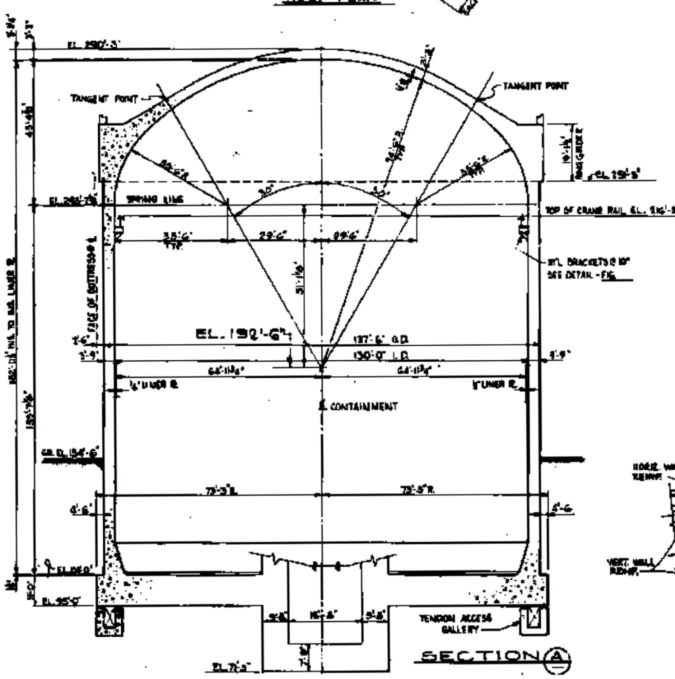
FIGURE 3.8-1



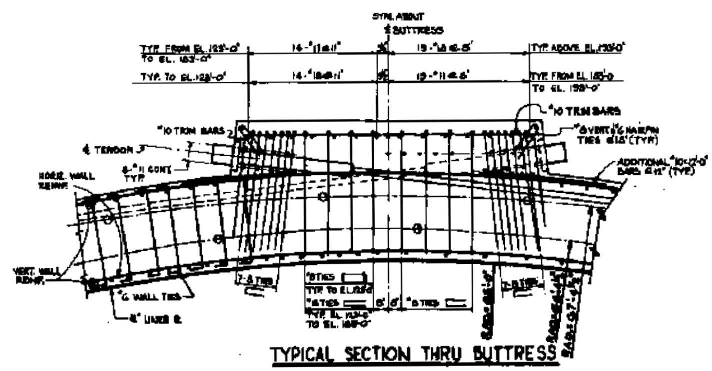
ROOF PLAN



PLAN - REFUELING TUBE



SECTION A



TYPICAL SECTION THRU BUTTRESS

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Energy to Serve Your World®

**JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2**

CONTAINMENT PLANS & SECTION

FIGURE 3.8-2

THIS FIGURE HAS BEEN DELETED

15

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT DETAILS OF EQUIPMENT HATCH

FIGURE 3.8-3

THIS FIGURE HAS BEEN DELETED

15

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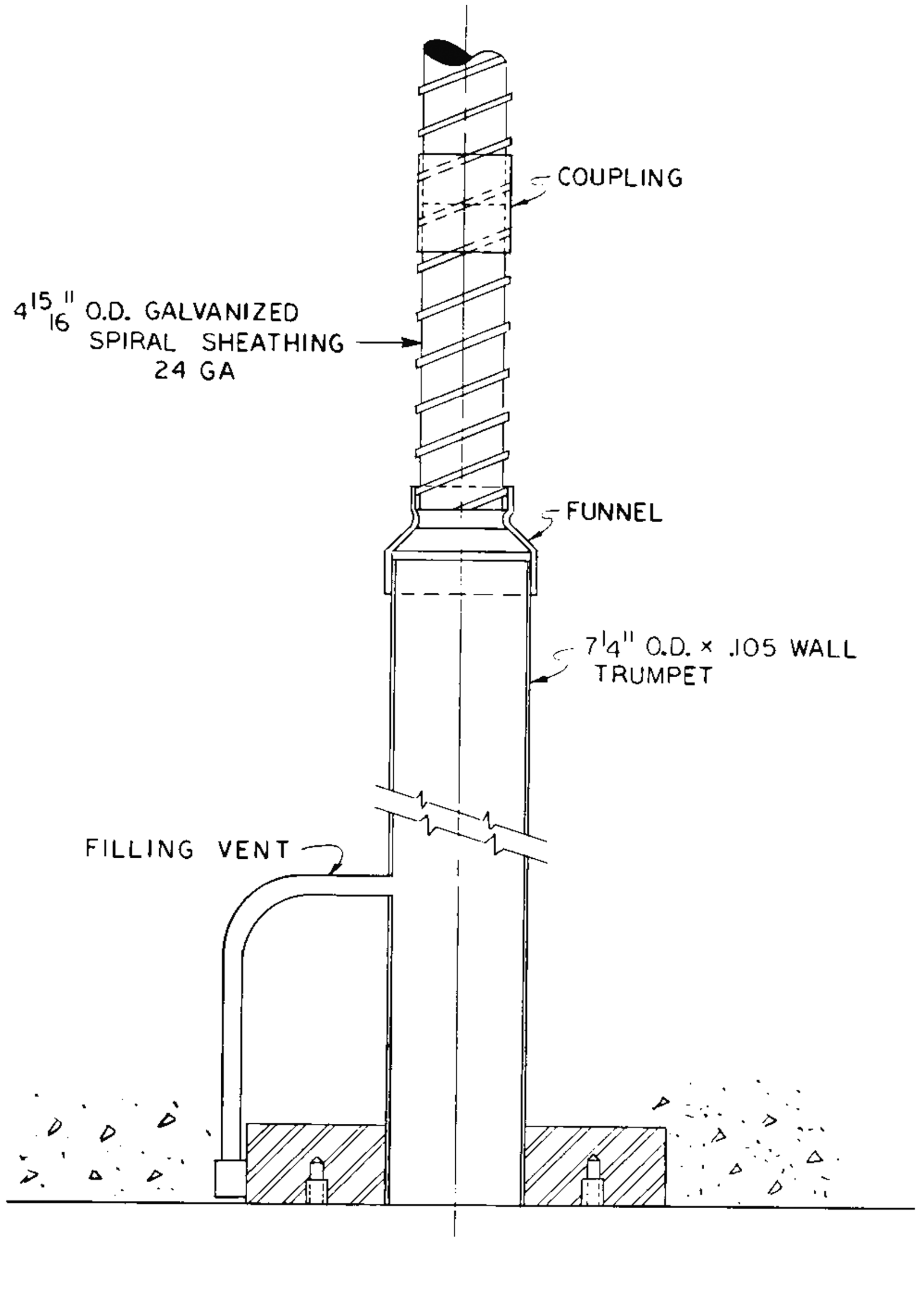
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT DETAILS OF PERSONNEL LOCK

FIGURE 3.8-4



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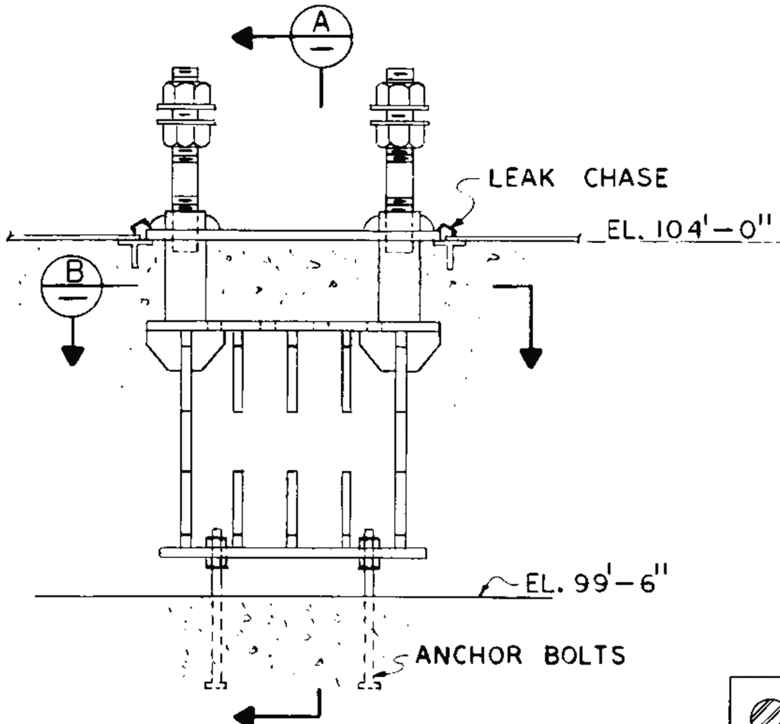
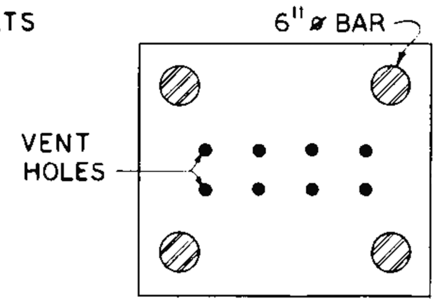
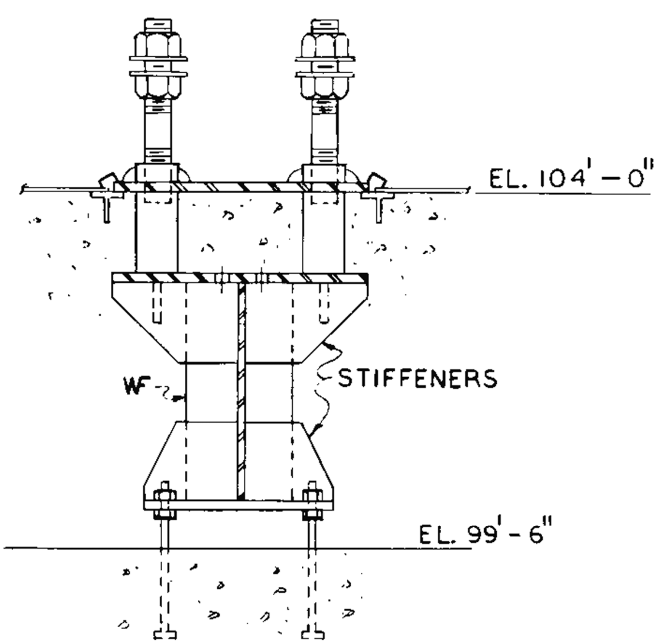
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SHEATHING AND TRUMPET DETAIL

FIGURE 3.8-5







SECTION B

SECTION A

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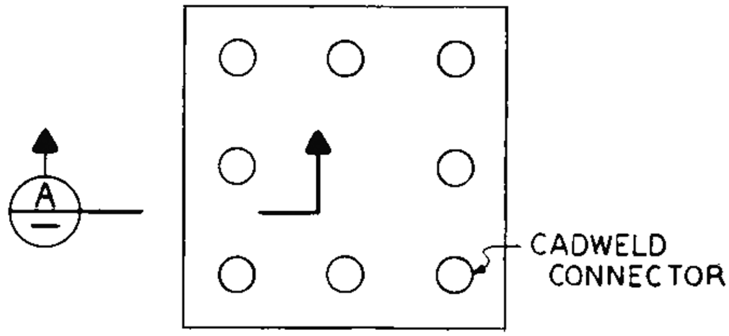
SOUTHERN COMPANY
Energy to Serve Your World®

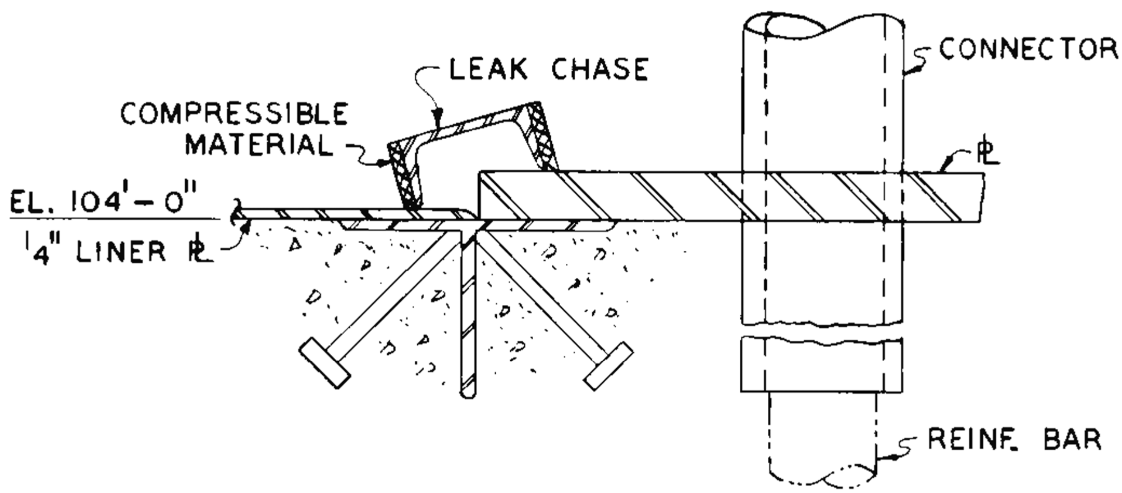
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

BASE DETAILS FOR STEAM GENERATOR AND REACTOR COOLANT PUMP FOUNDATIONS

FIGURE 3.8-6



PLAN



SECTION A

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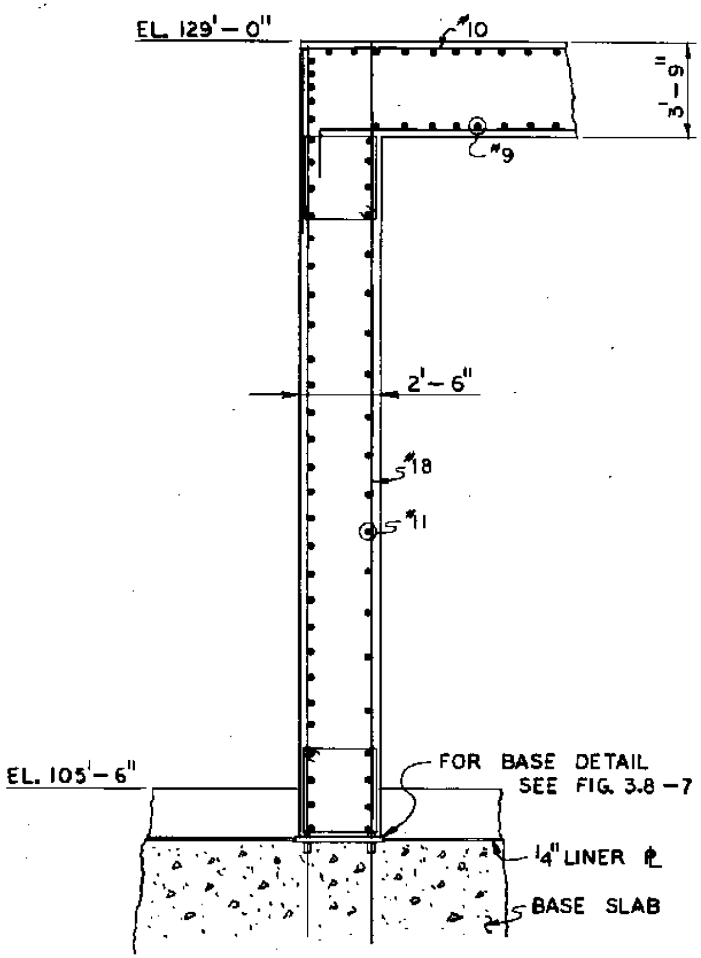
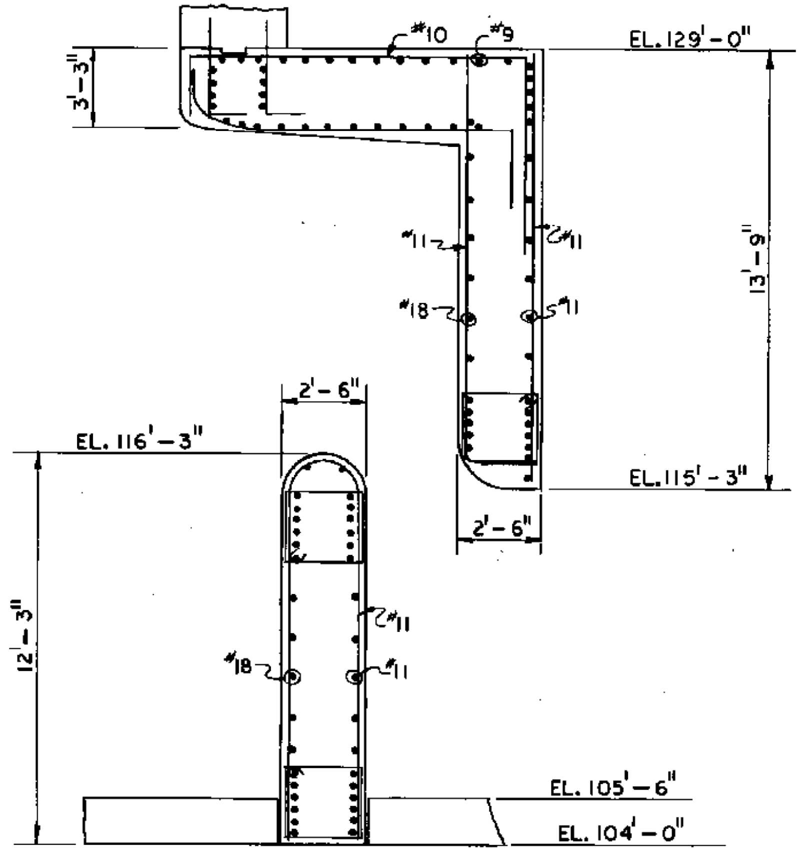
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	BASE DETAIL FOR SECONDARY SHIELD WALL
		FIGURE 3.8-7



FULL HEIGHT



PARTIAL HEIGHT

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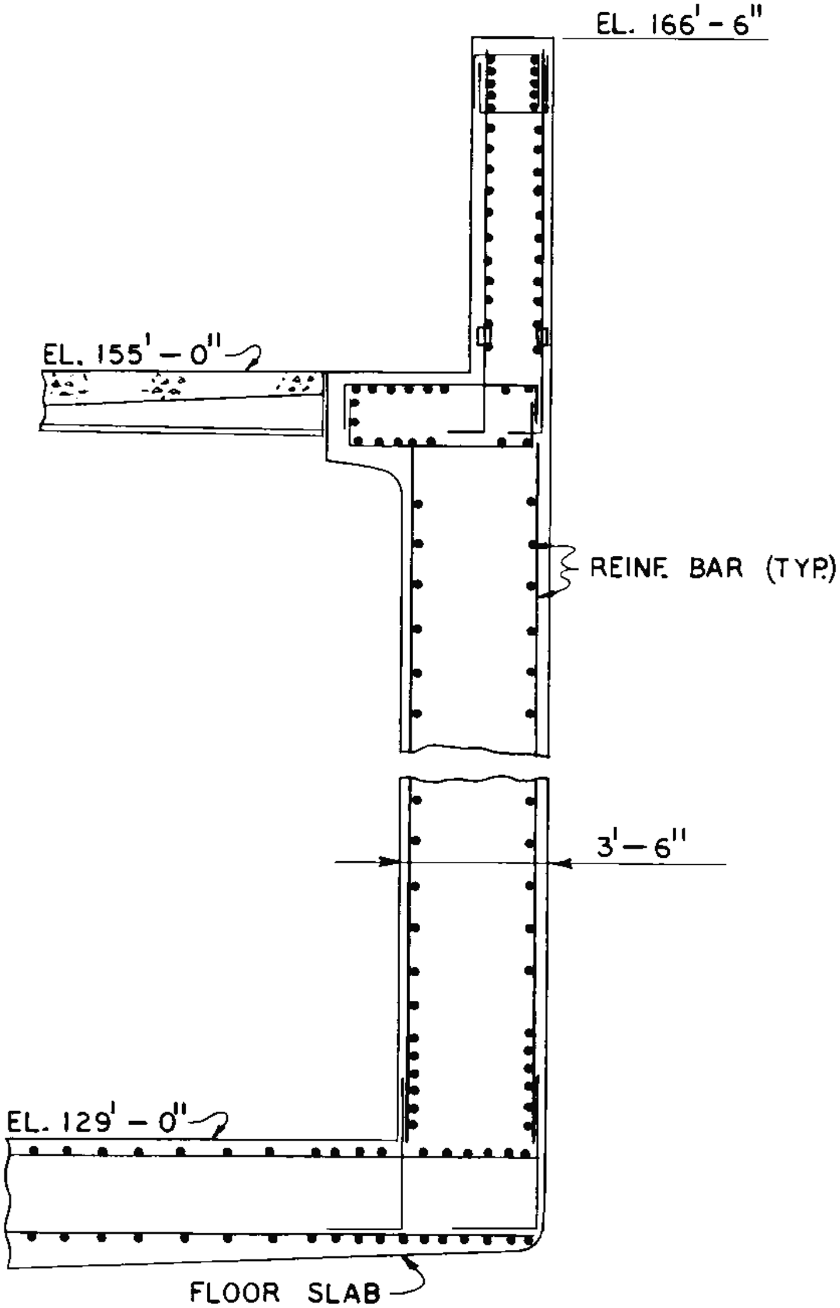
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SECONDARY SHIELD WALLS BELOW EL. 129'-0"

FIGURE 3.8-9



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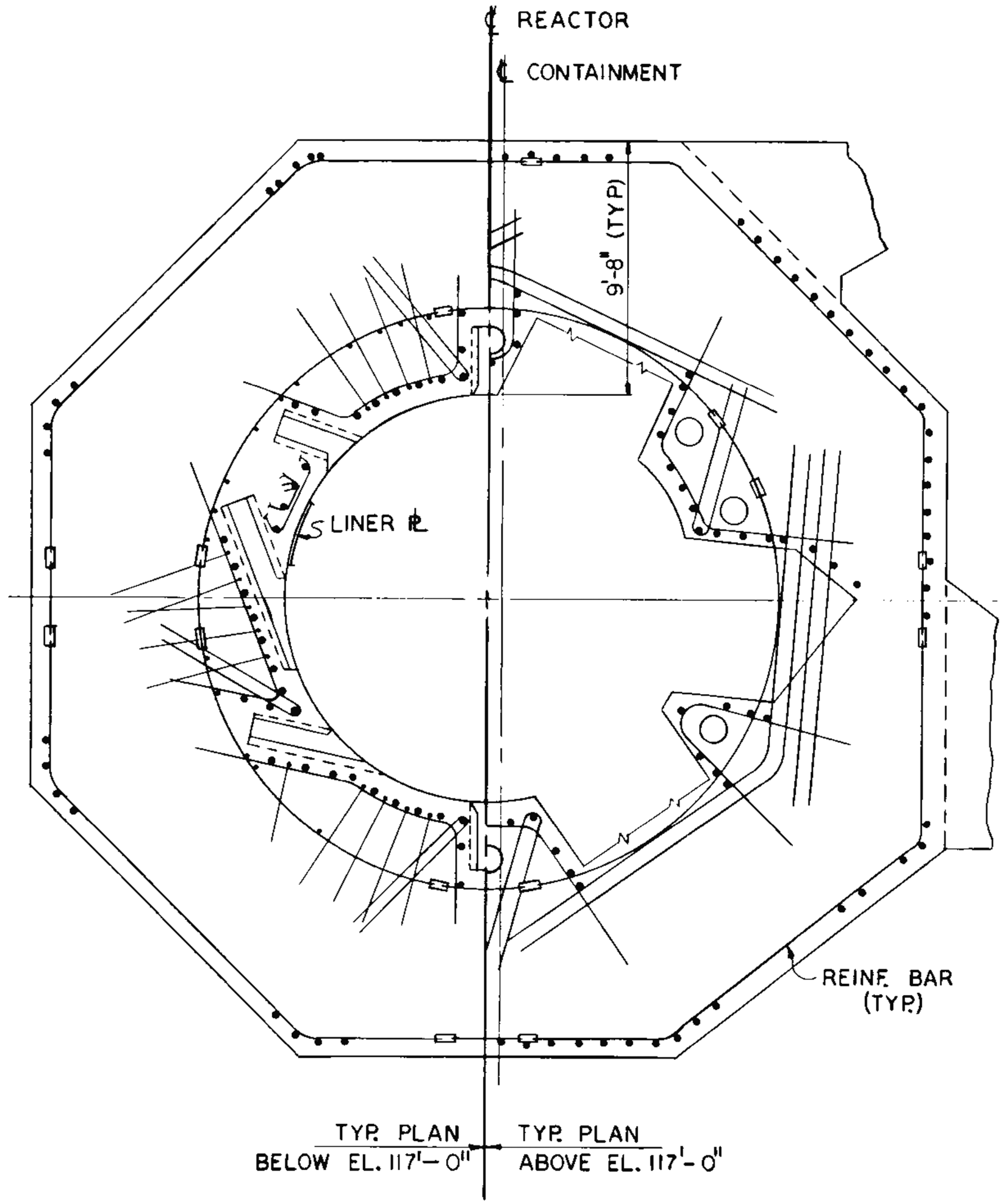
SECONDARY SHIELD WALL EL. 129'-0" TO 166'-6"

FIGURE 3.8-10



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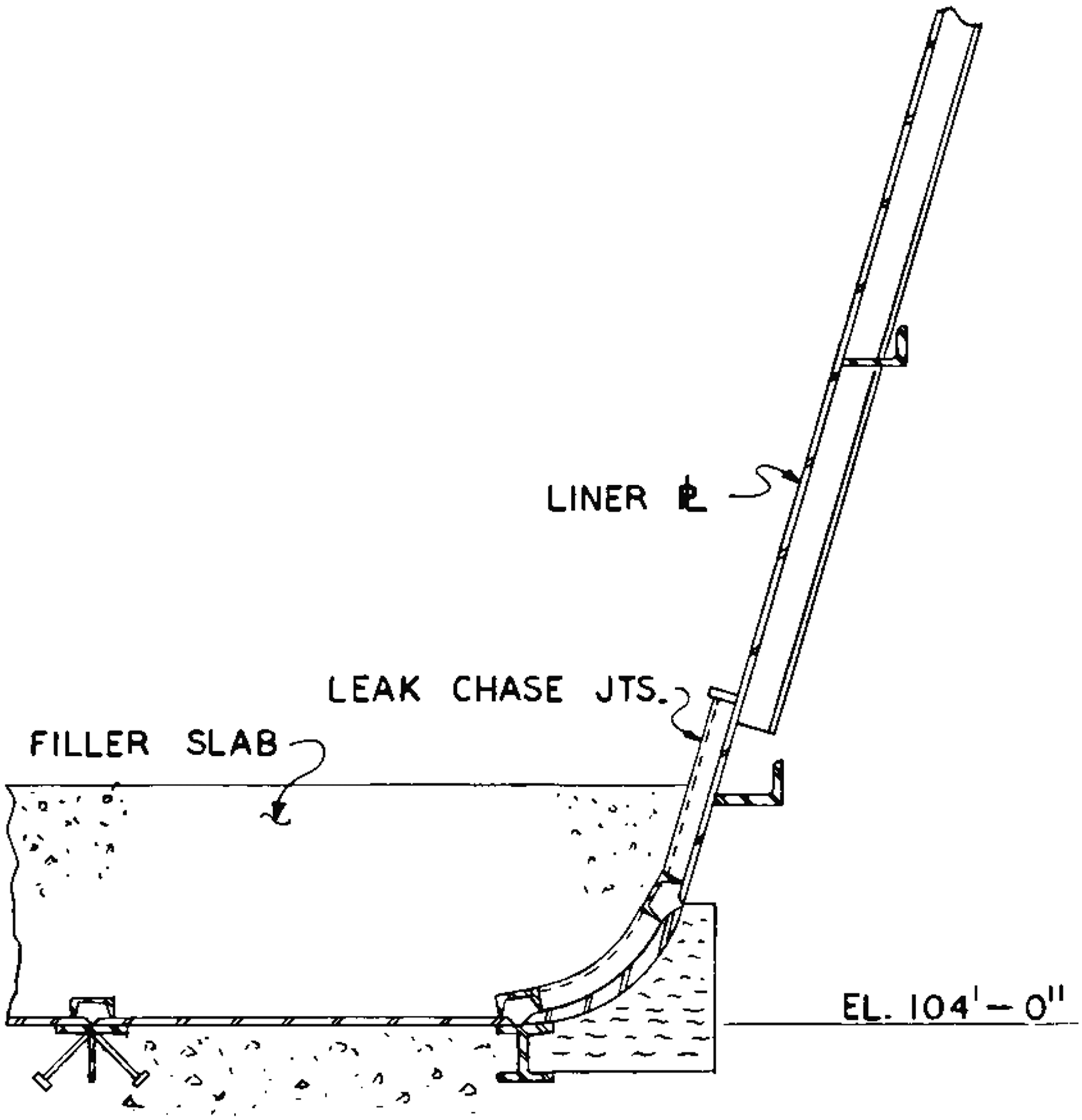
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRIMARY SHIELD WALL

FIGURE 3.8-11



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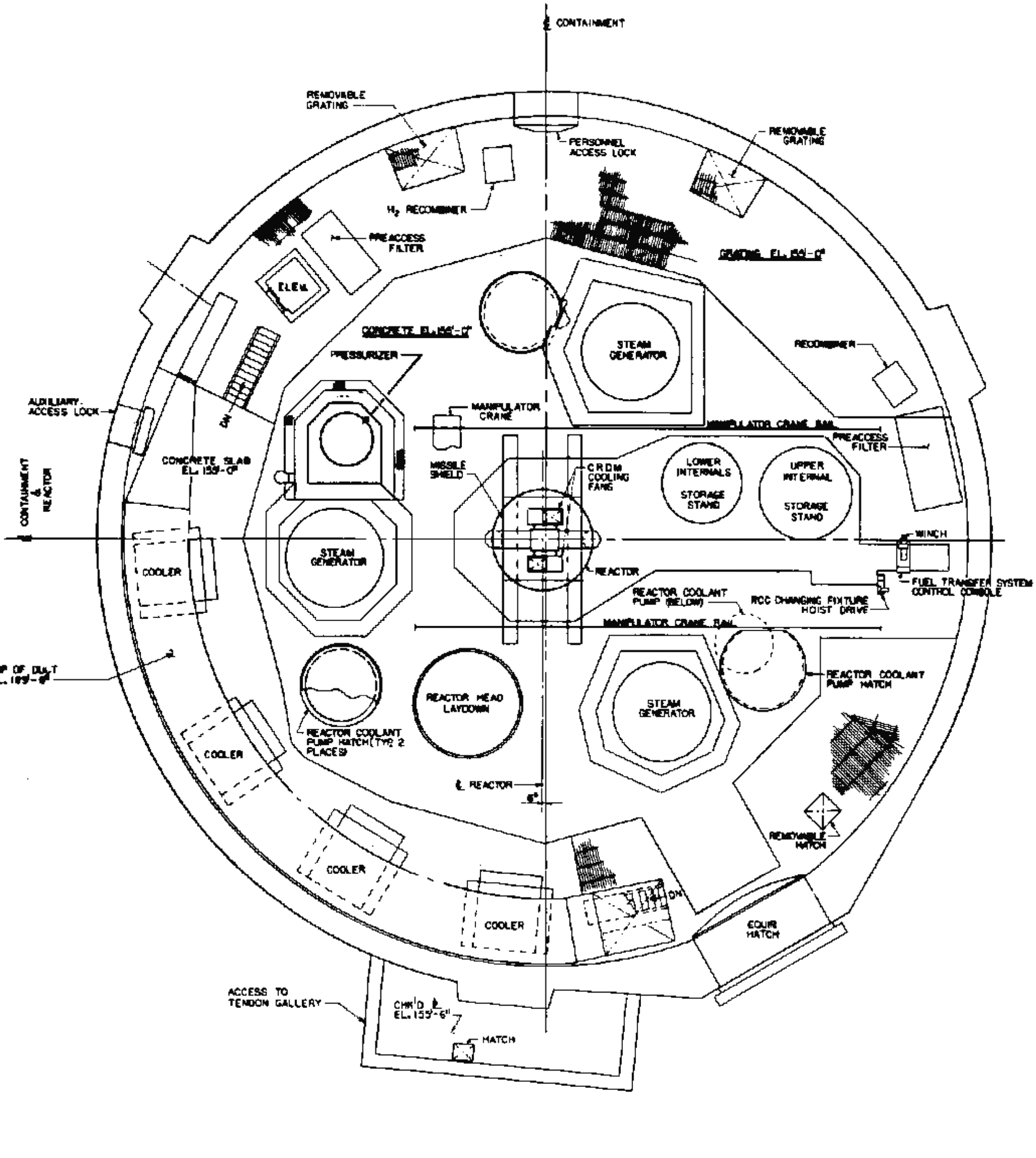
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DETAIL FOR BASE SLAB TO CYLINDER LINER JUNCTURE

FIGURE 3.8-12



PLAN AT ELEVATION 155'-0"

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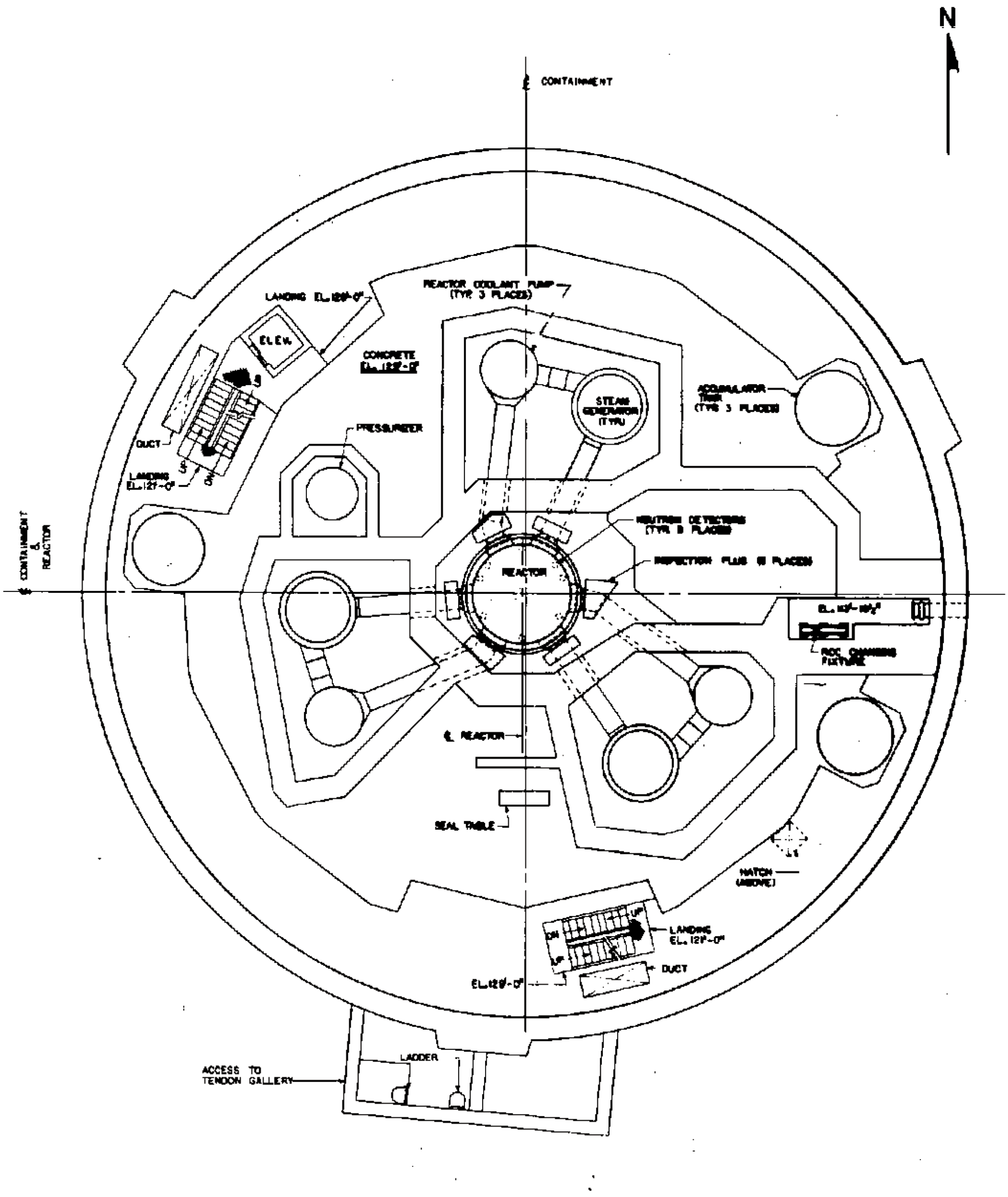
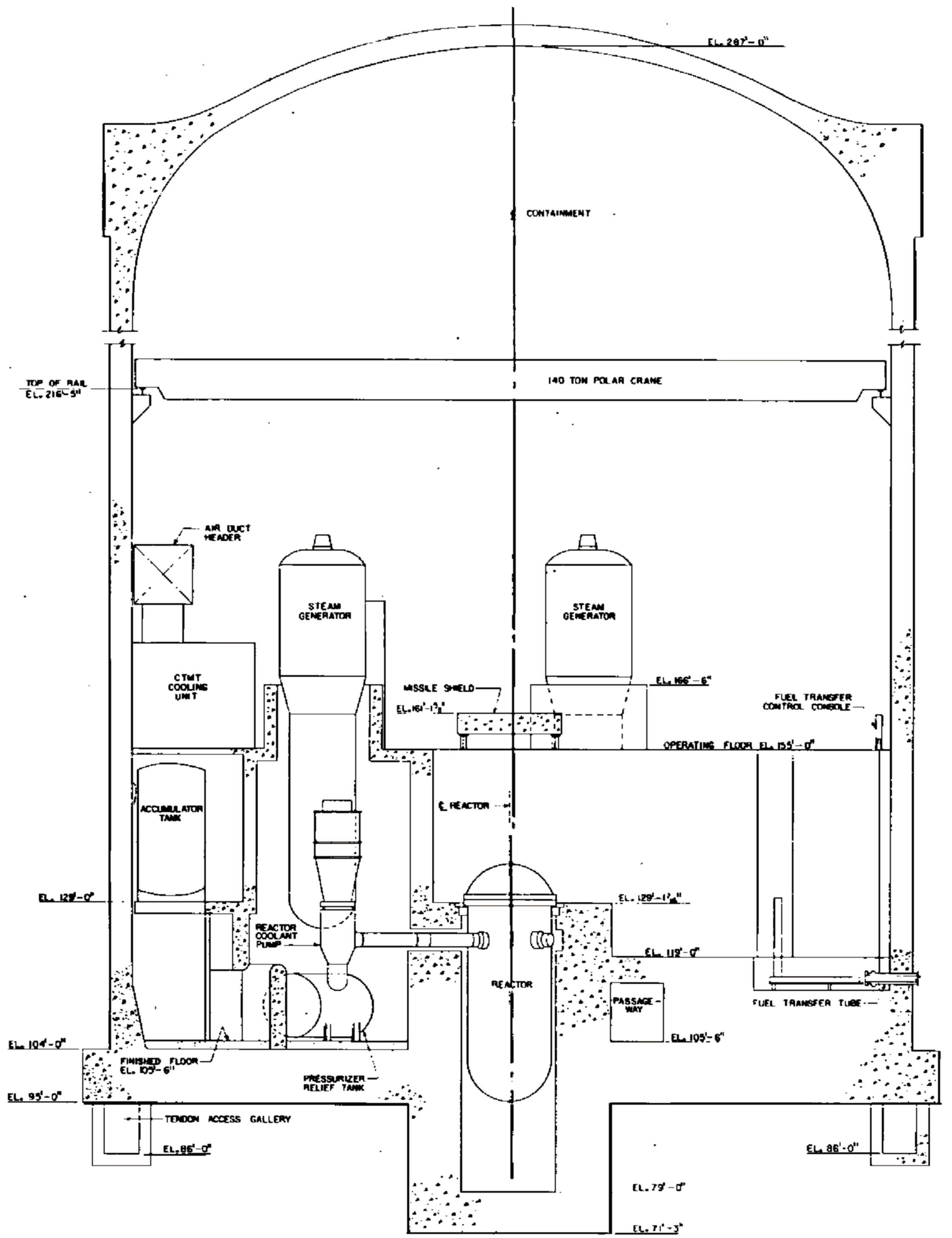
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TYPICAL PLANS CONTAINMENT FIGURE 3.8-13 (SHEET 1 OF 2)



PLAN AT ELEVATION 129'-0"

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TYPICAL PLANS CONTAINMENT FIGURE 3.8-13 (SHEET 2 OF 2)



SECTION LOOKING NORTH

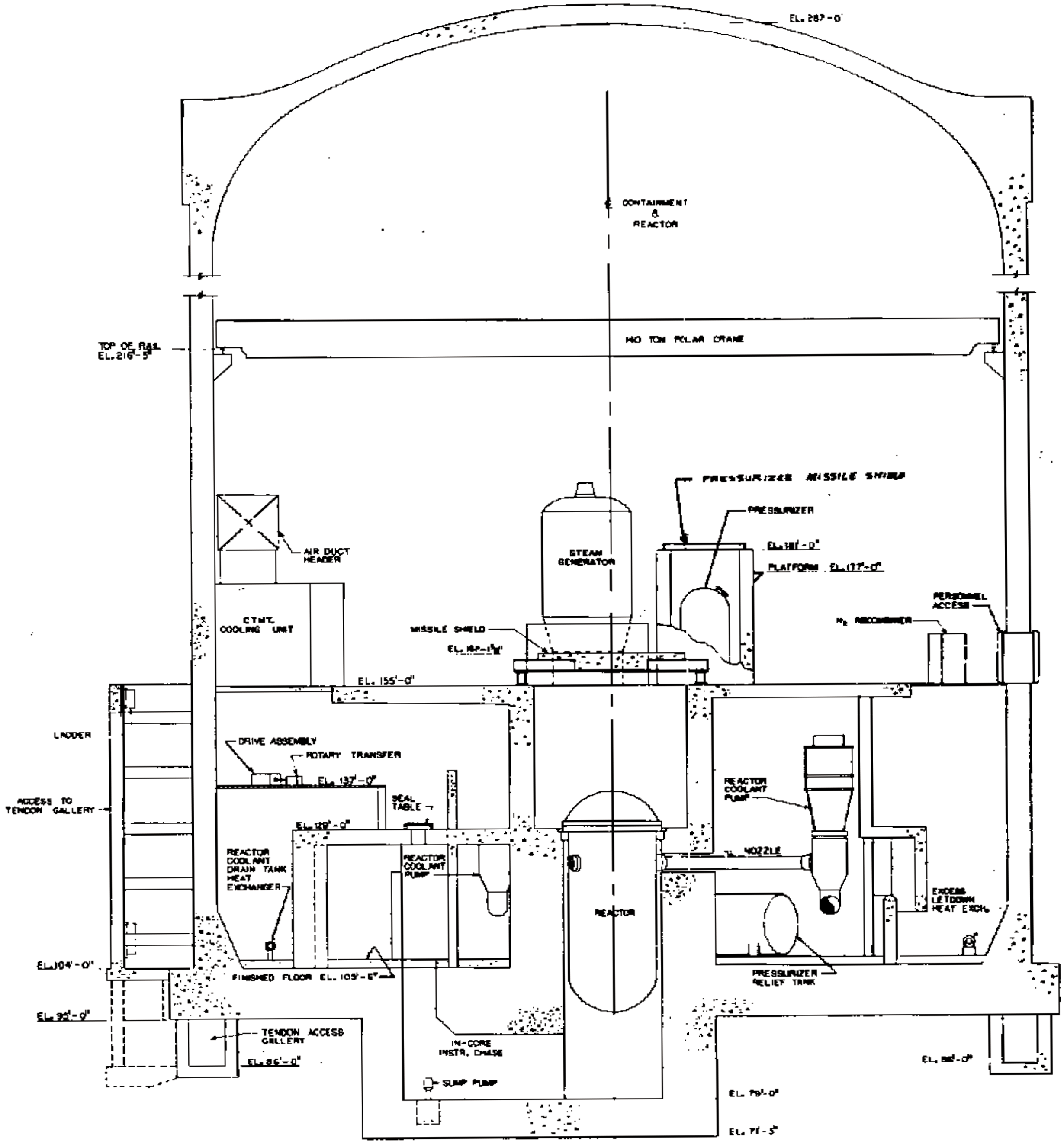
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

TYPICAL SECTIONS CONTAINMENT

FIGURE 3.8-14 (SHEET 1 OF 2)



SECTION LOOKING WEST

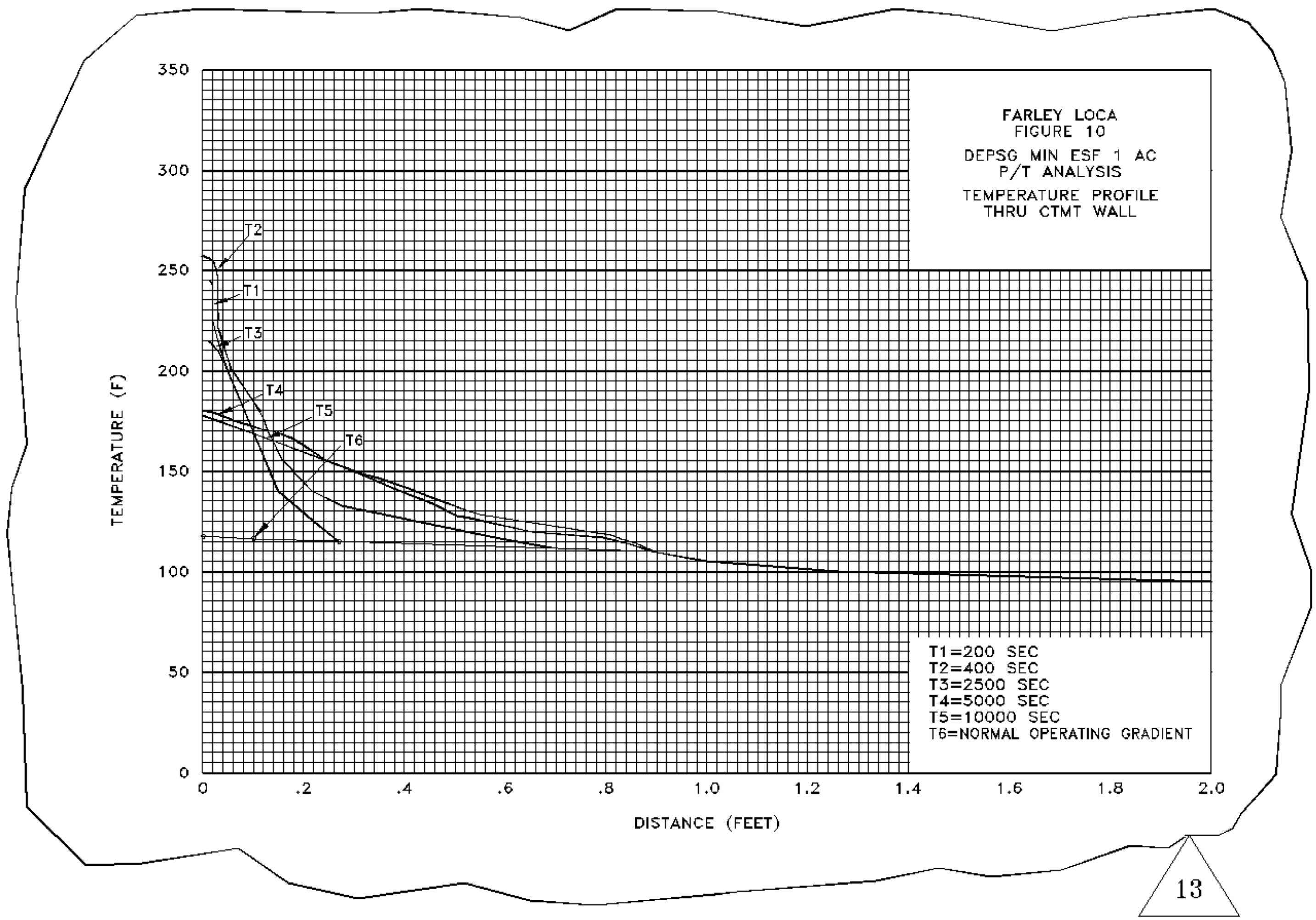
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

TYPICAL SECTIONS CONTAINMENT

FIGURE 3.8-14 (SHEET 2 OF 2)



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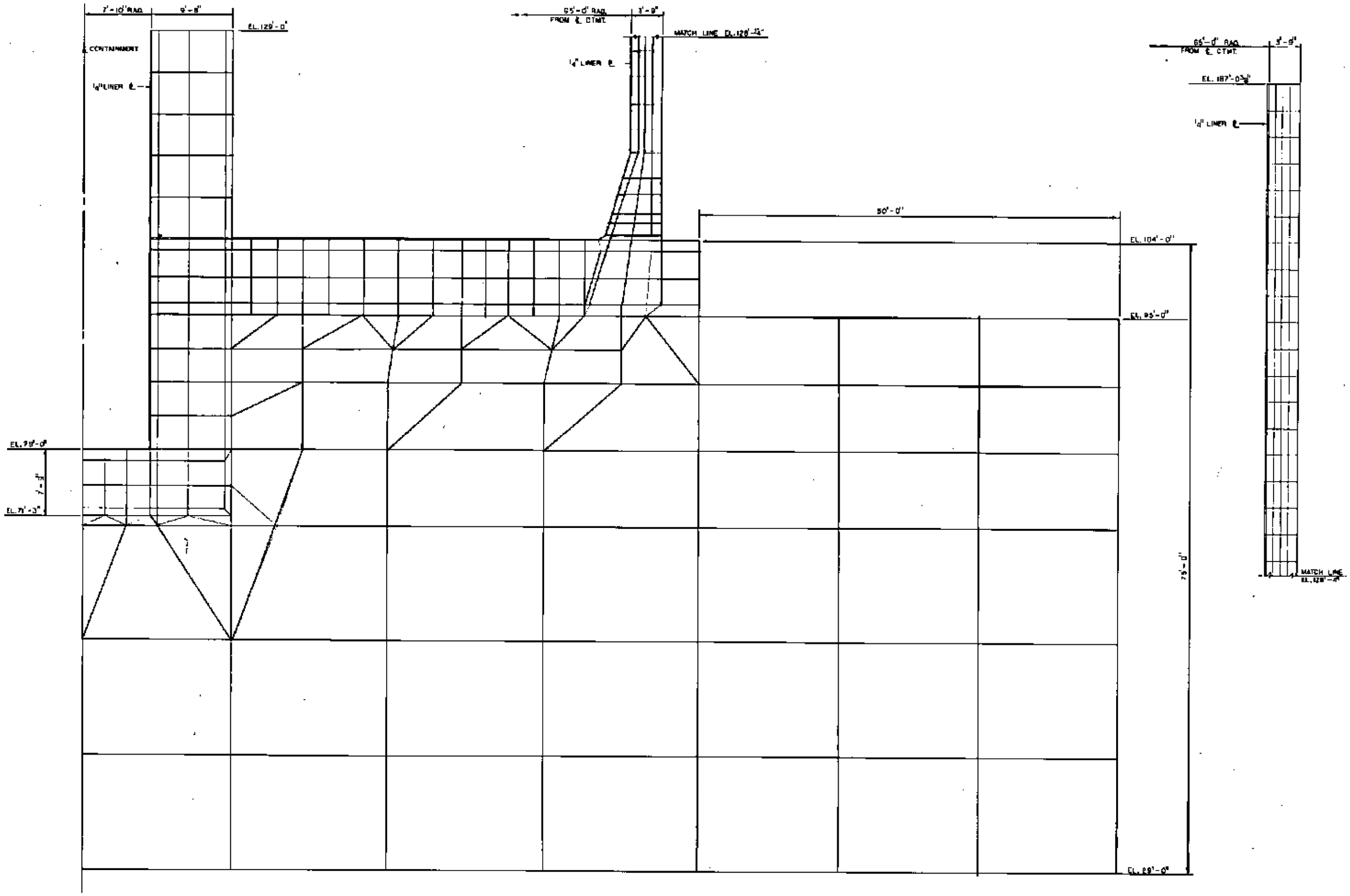
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

THERMAL GRADIENT ACROSS CONTAINMENT WALL

FIGURE 3.8-15



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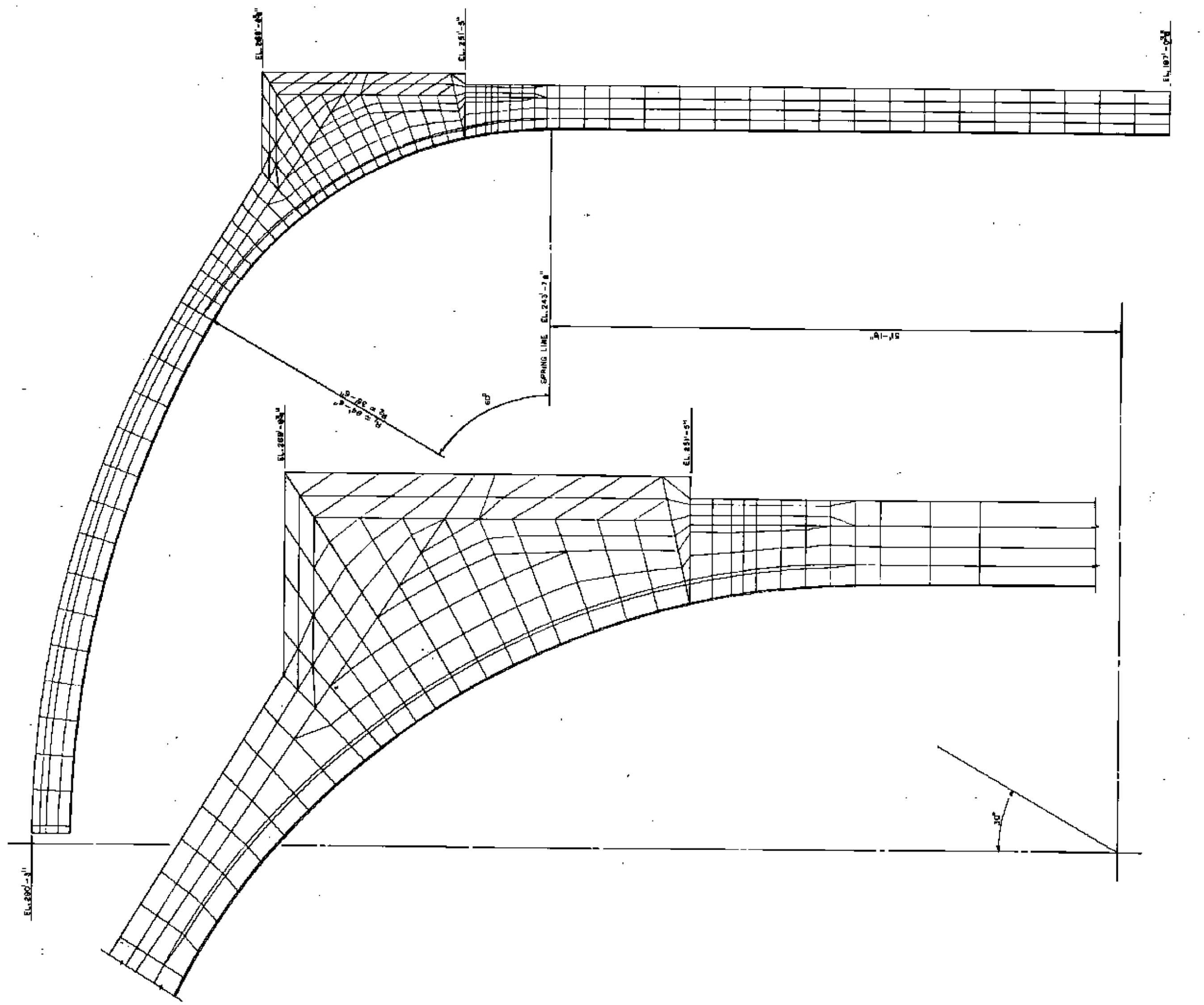
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FINITE ELEMENT MESH
BOTTOM HALF CONTAINMENT FOR AXISYMMETRIC LOADS

FIGURE 3.8-16



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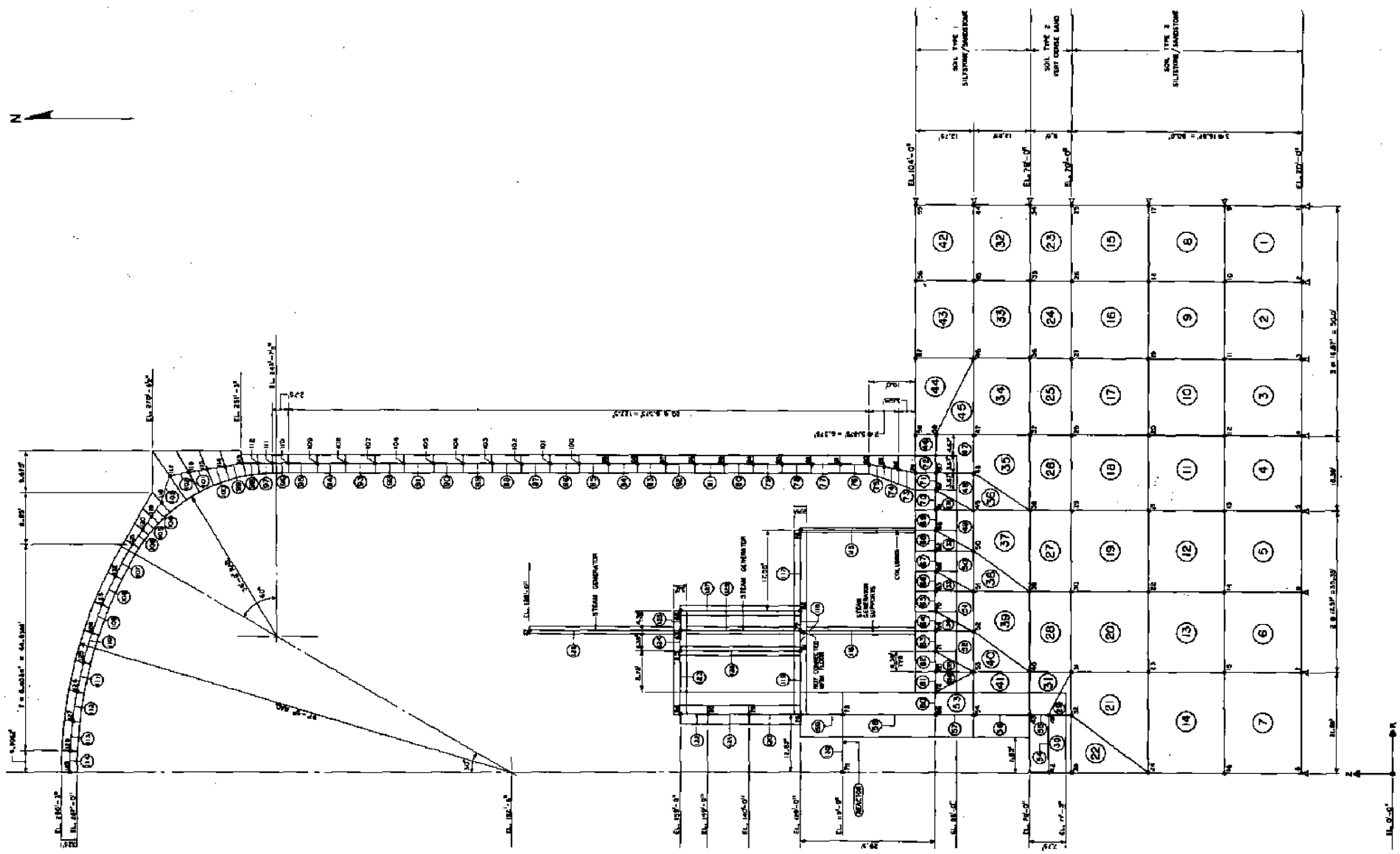
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FINITE ELEMENT MESH
TOP HALF CONTAINMENT FOR AXISYMMETRIC LOADS

FIGURE 3.8-17



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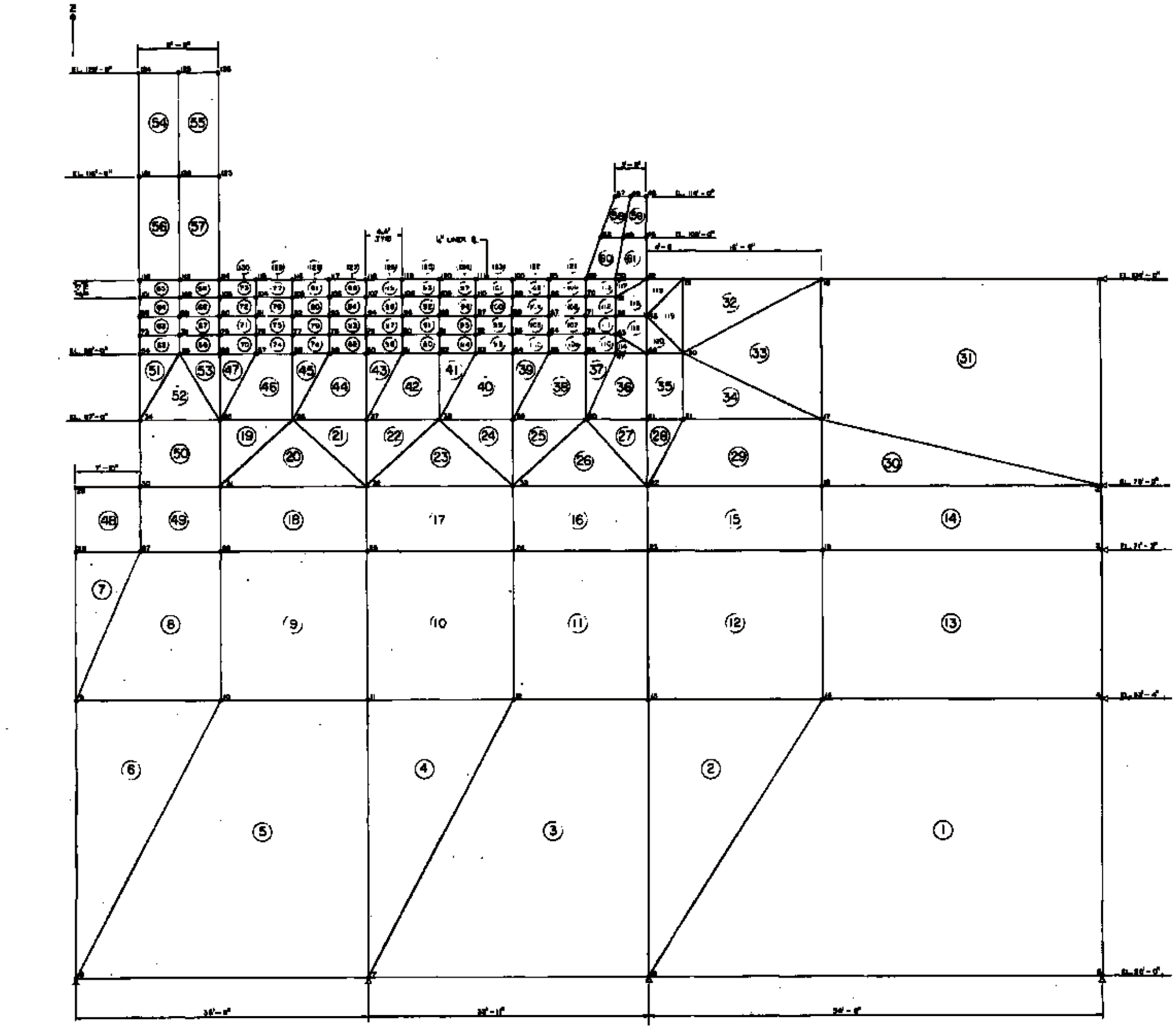
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MODEL OF CONTAINMENT FOR
FINITE ELEMENT ANALYSIS NON-AXISYMMETRIC LOADS

FIGURE 3.8-18



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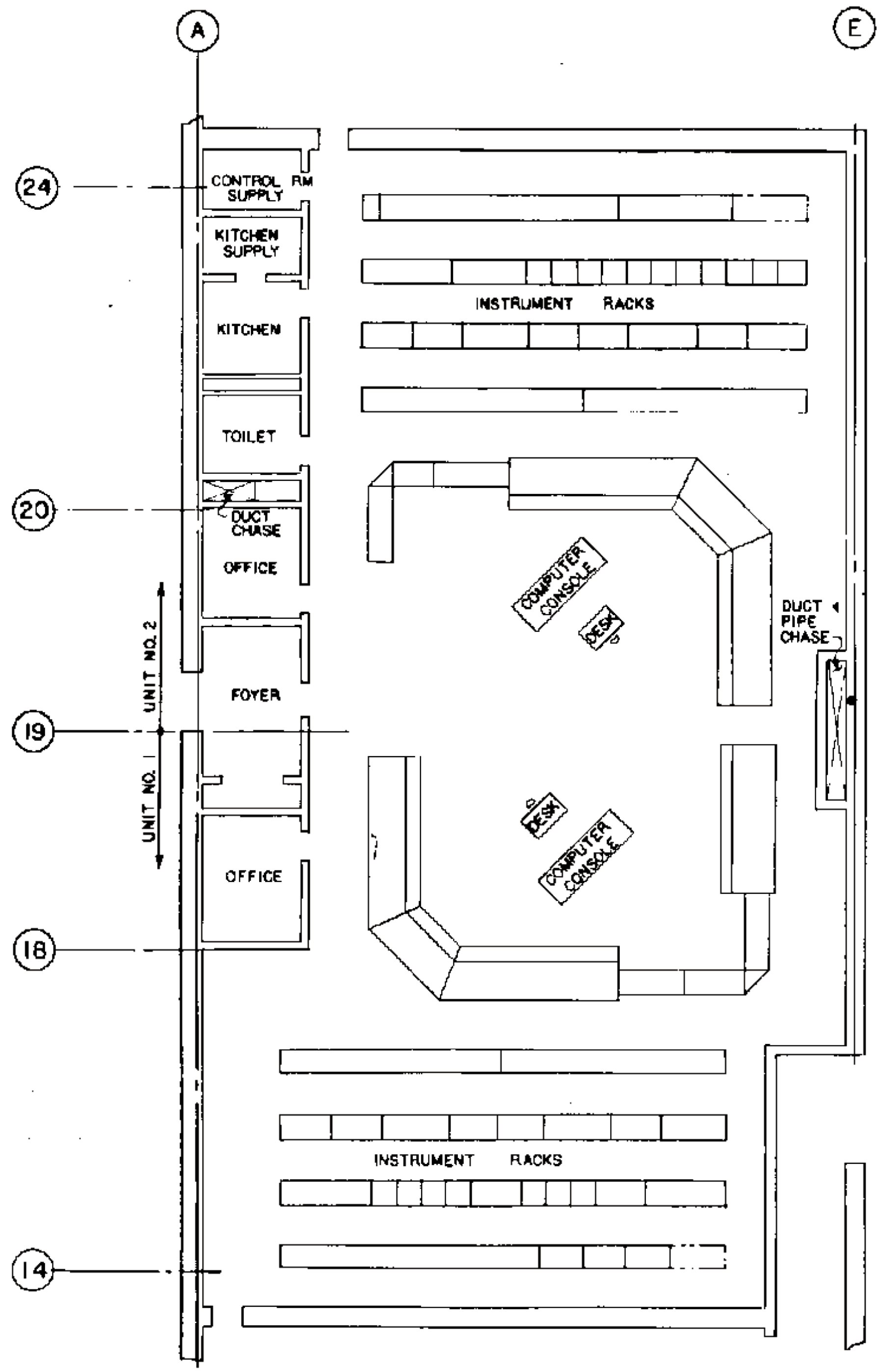
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

CONTAINMENT BASE SLAB FINITE ELEMENT MESH
NON-AXISYMMETRIC LOADS

FIGURE 3.8-19



CONTROL ROOM
PLAN AT EL.155'-0"

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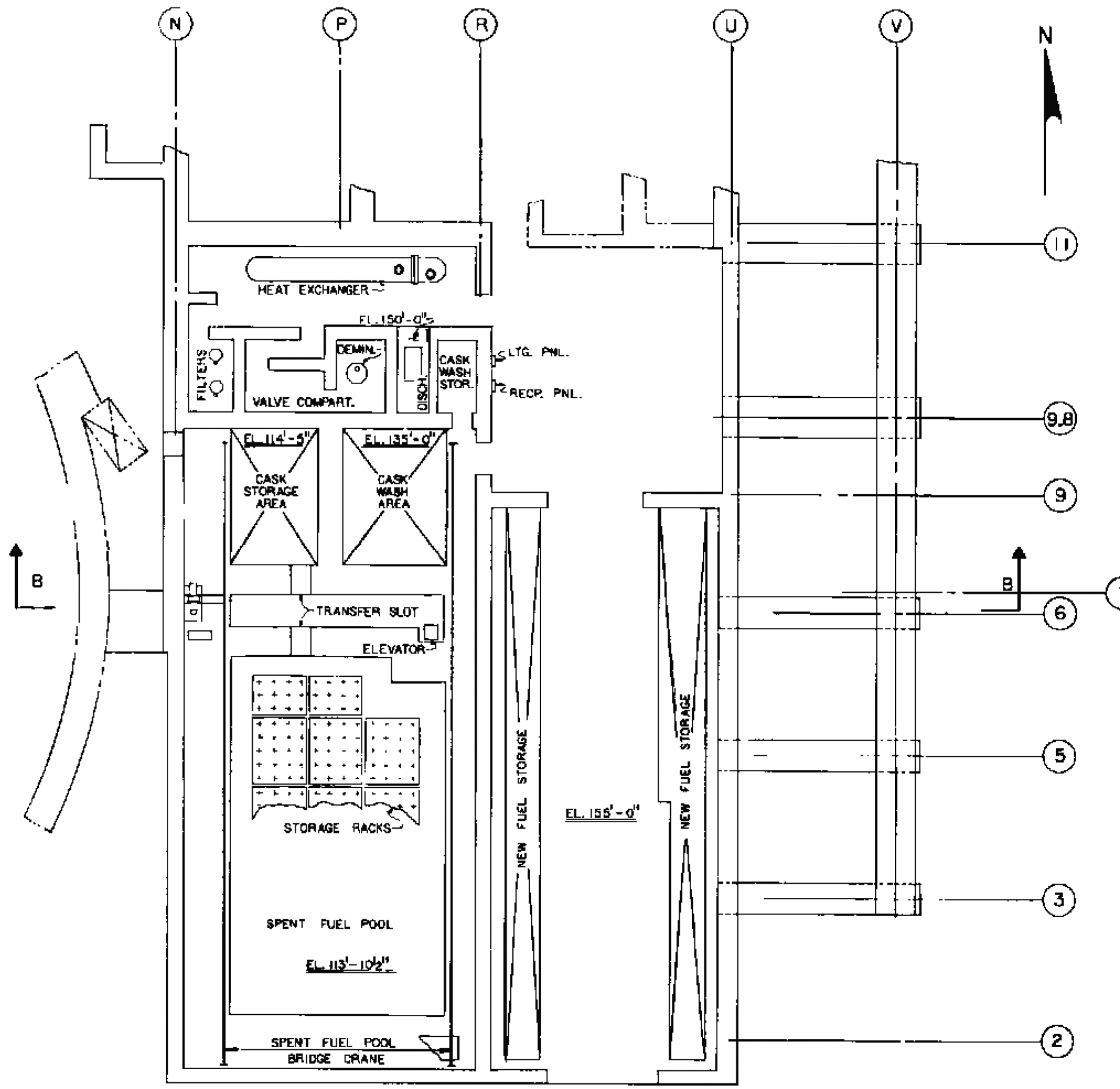
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING CONTROL ROOM &
SPENT FUEL POOL PLANS AT EL. 155'-0"

FIGURE 3.8-23 (SHEET 1 OF 2)



SPENT FUEL POOL
PLAN AT EL. 155'-0"

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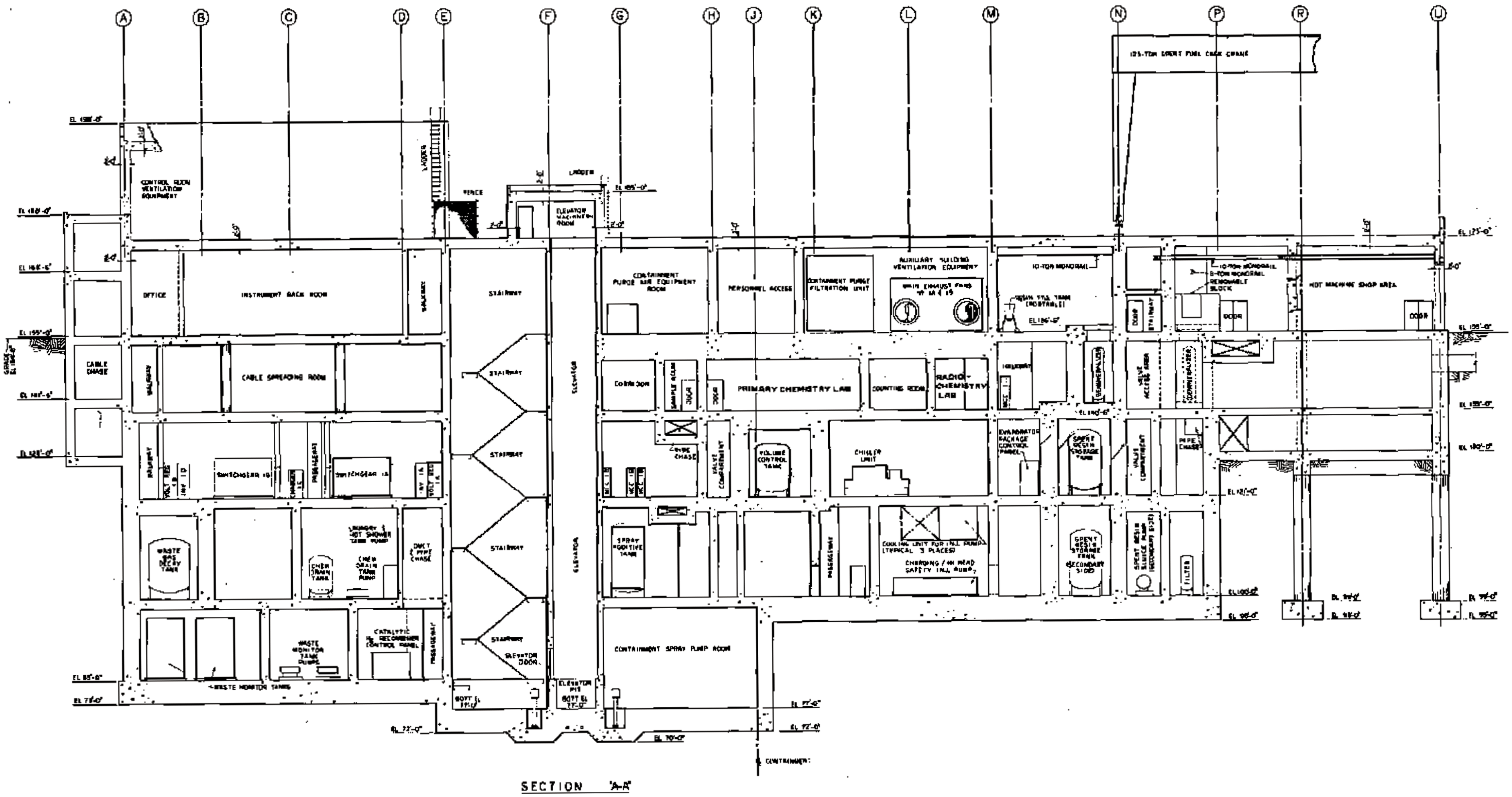
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING CONTROL ROOM & SPENT FUEL POOL PLANS AT EL. 155'-0"

FIGURE 3.8-23 (SHEET 2 OF 2)



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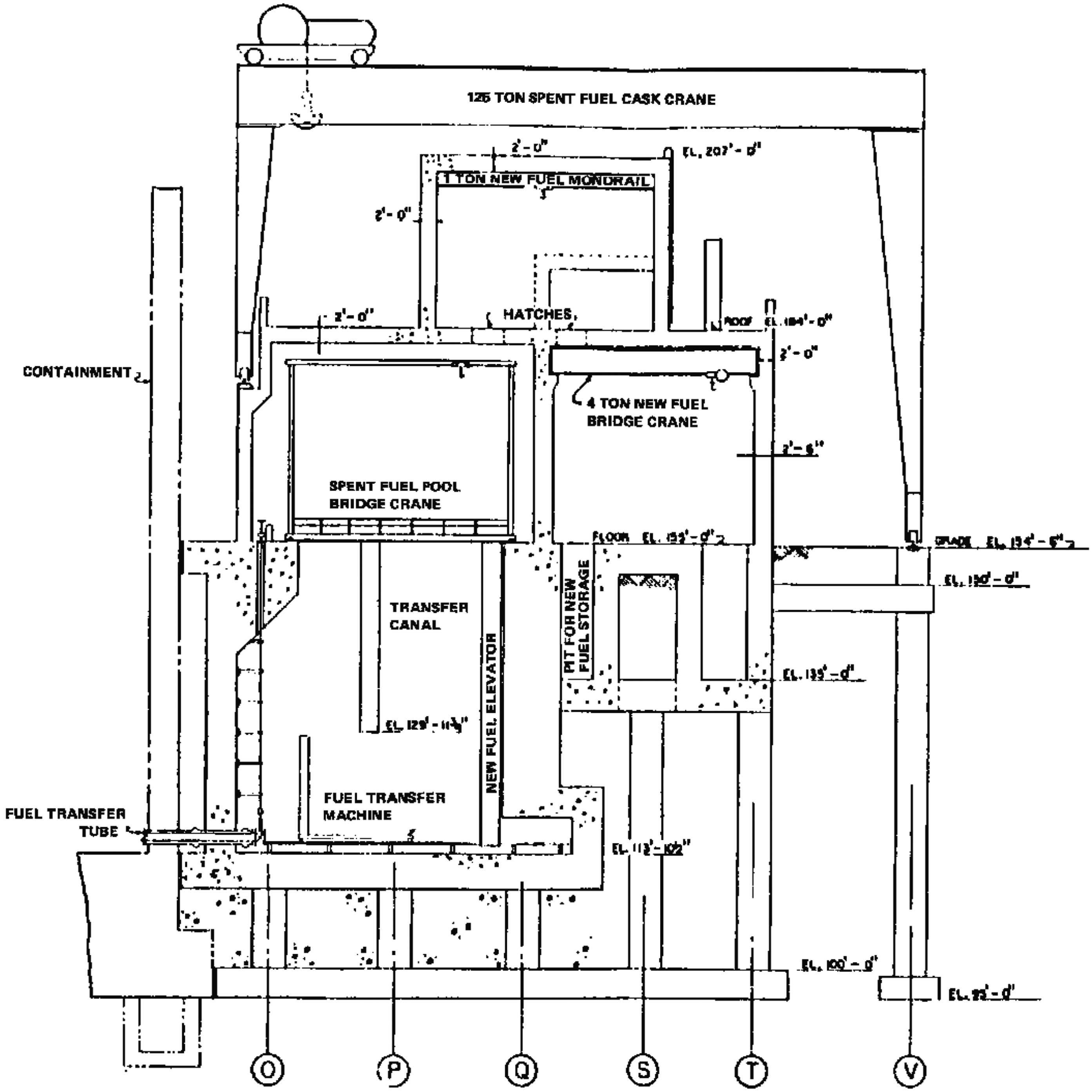
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING SECTION A-A

FIGURE 3.8-24



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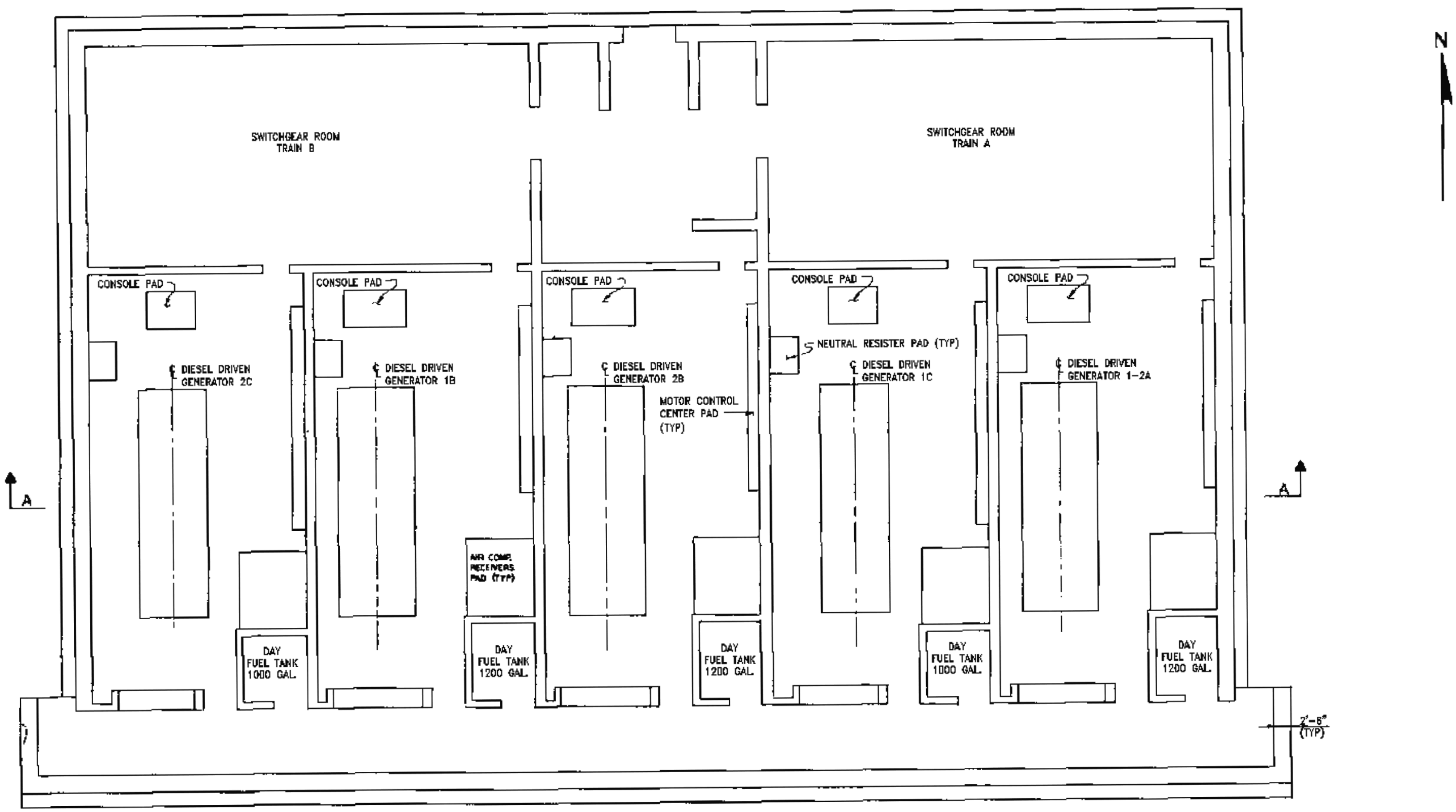
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING SECTION B-B

FIGURE 3.8-25



PLAN AT EL.155'-6"

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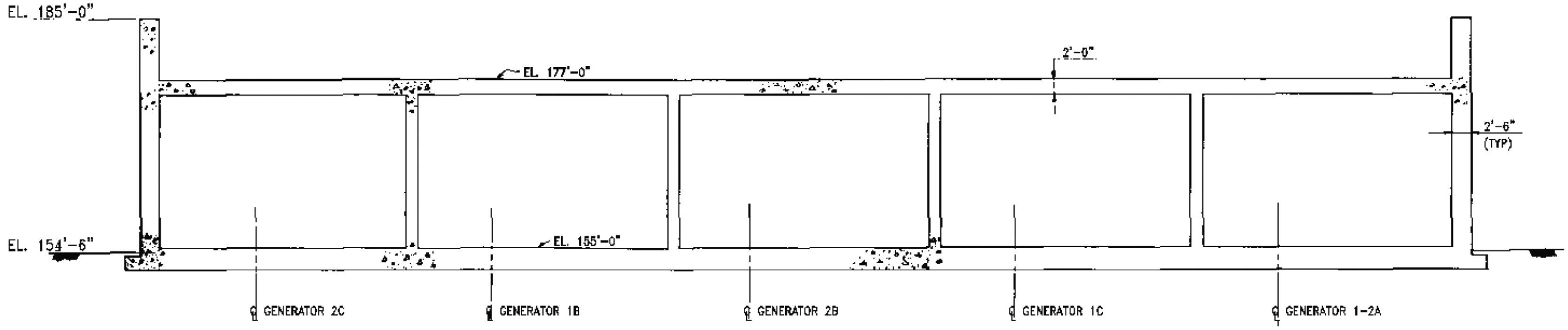
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DIESEL GENERATOR BUILDING PLAN AND SECTION

FIGURE 3.8-26 (SHEET 1 OF 2)



SECTION A-A

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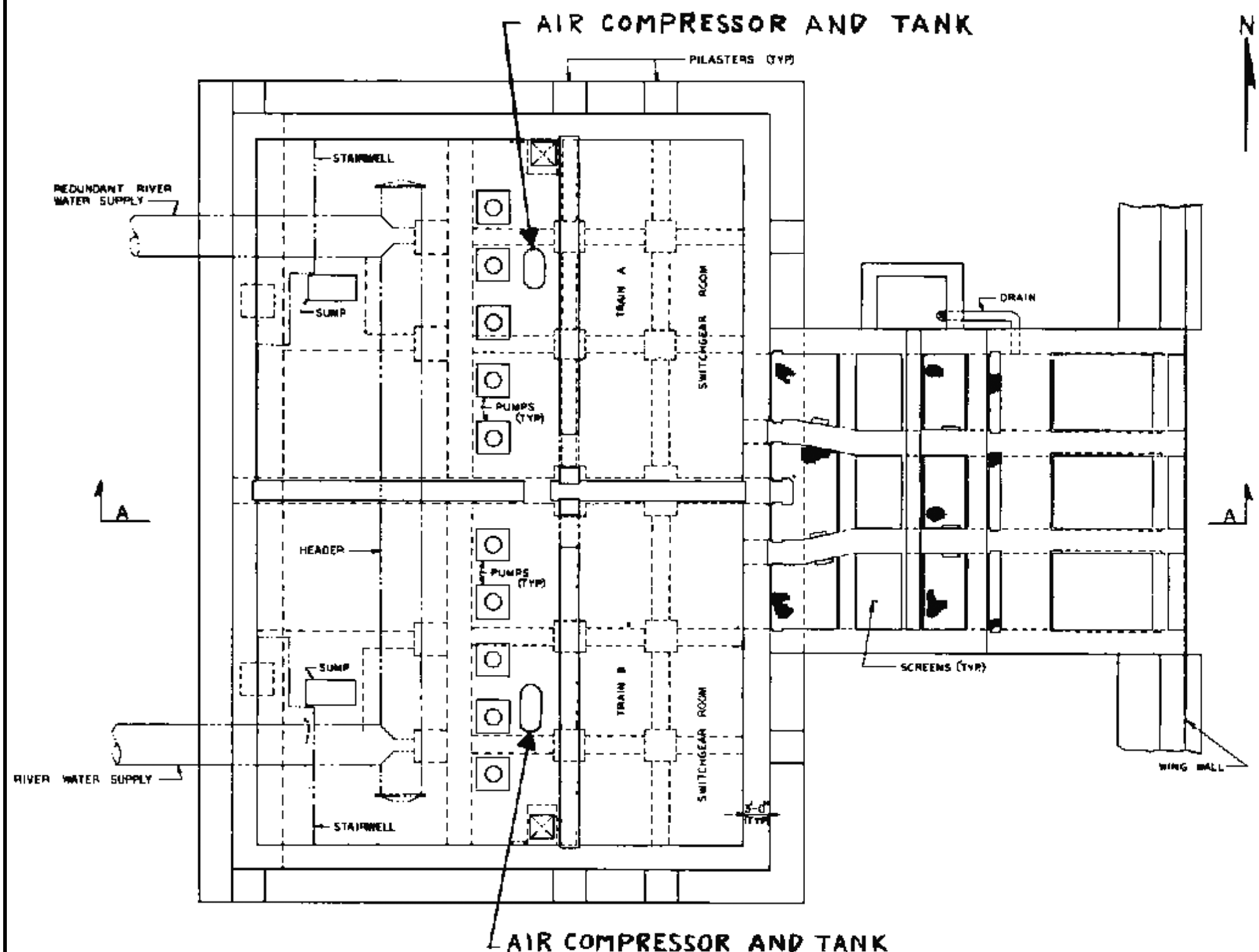
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DIESEL GENERATOR BUILDING PLAN AND SECTION

FIGURE 3.8-26 (SHEET 2 OF 2)



PLAN

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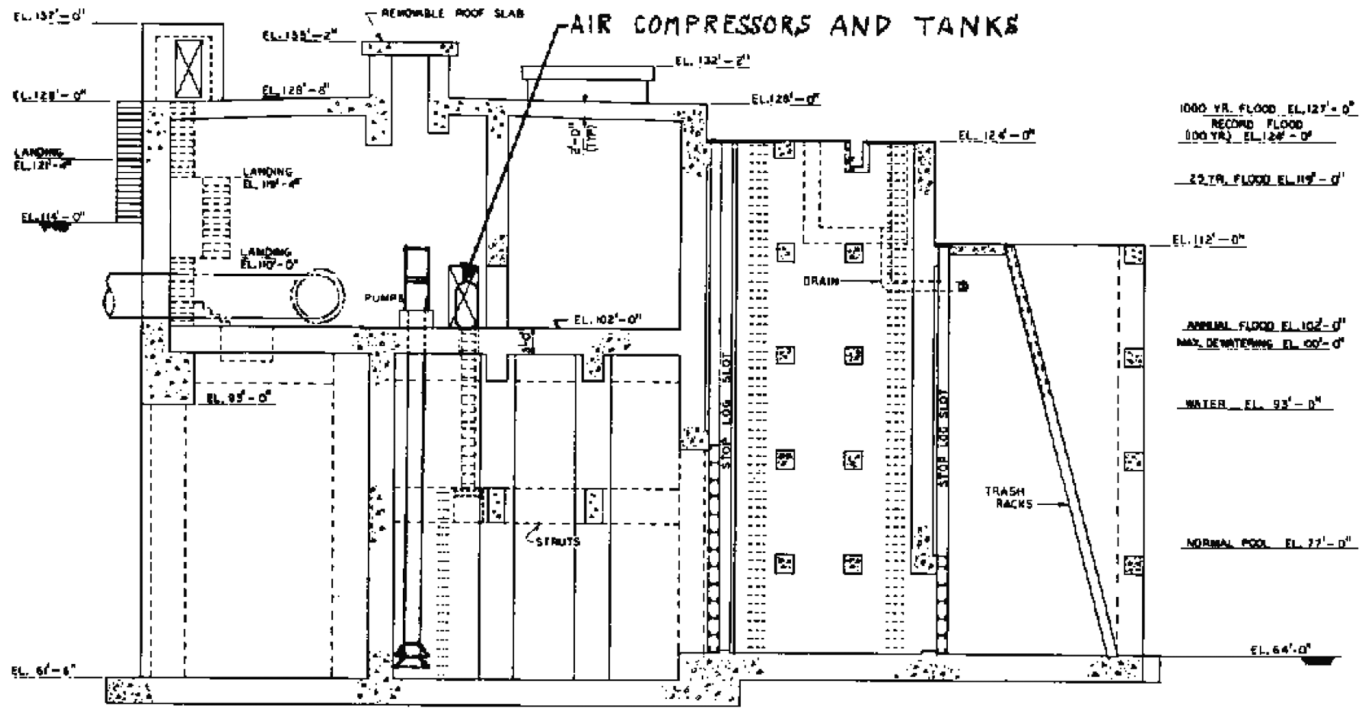
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RIVER INTAKE STRUCTURE PLAN AND SECTION

FIGURE 3.8-27 (SHEET 1 OF 2)



SECTION A – A

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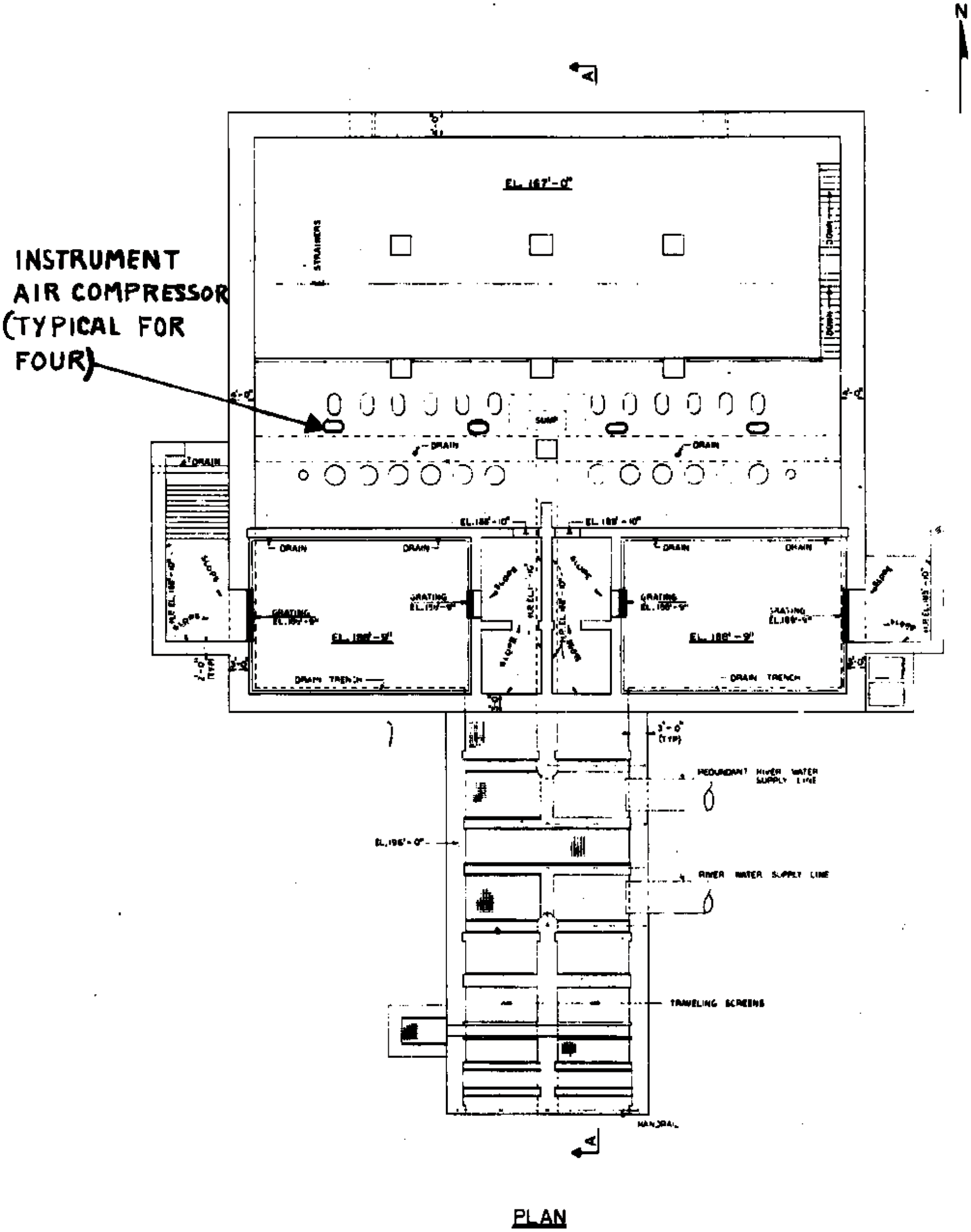
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	RIVER INTAKE STRUCTURE PLAN AND SECTION FIGURE 3.8-27 (SHEET 2 OF 2)



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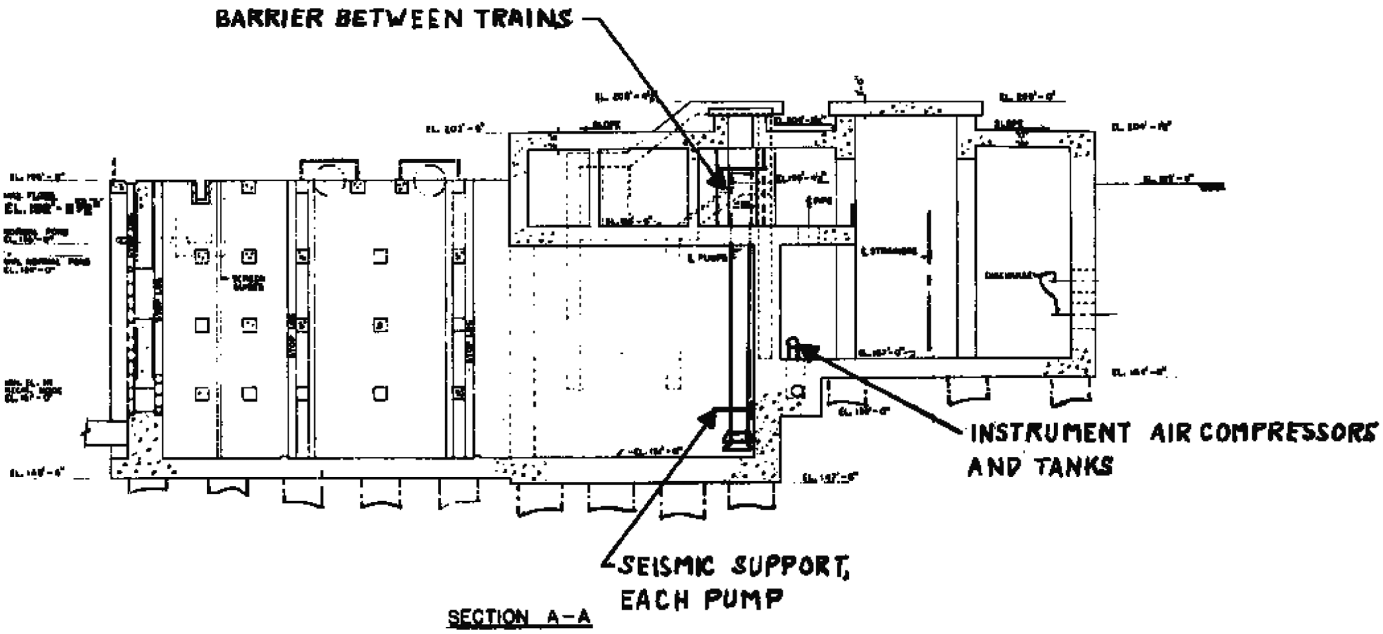
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

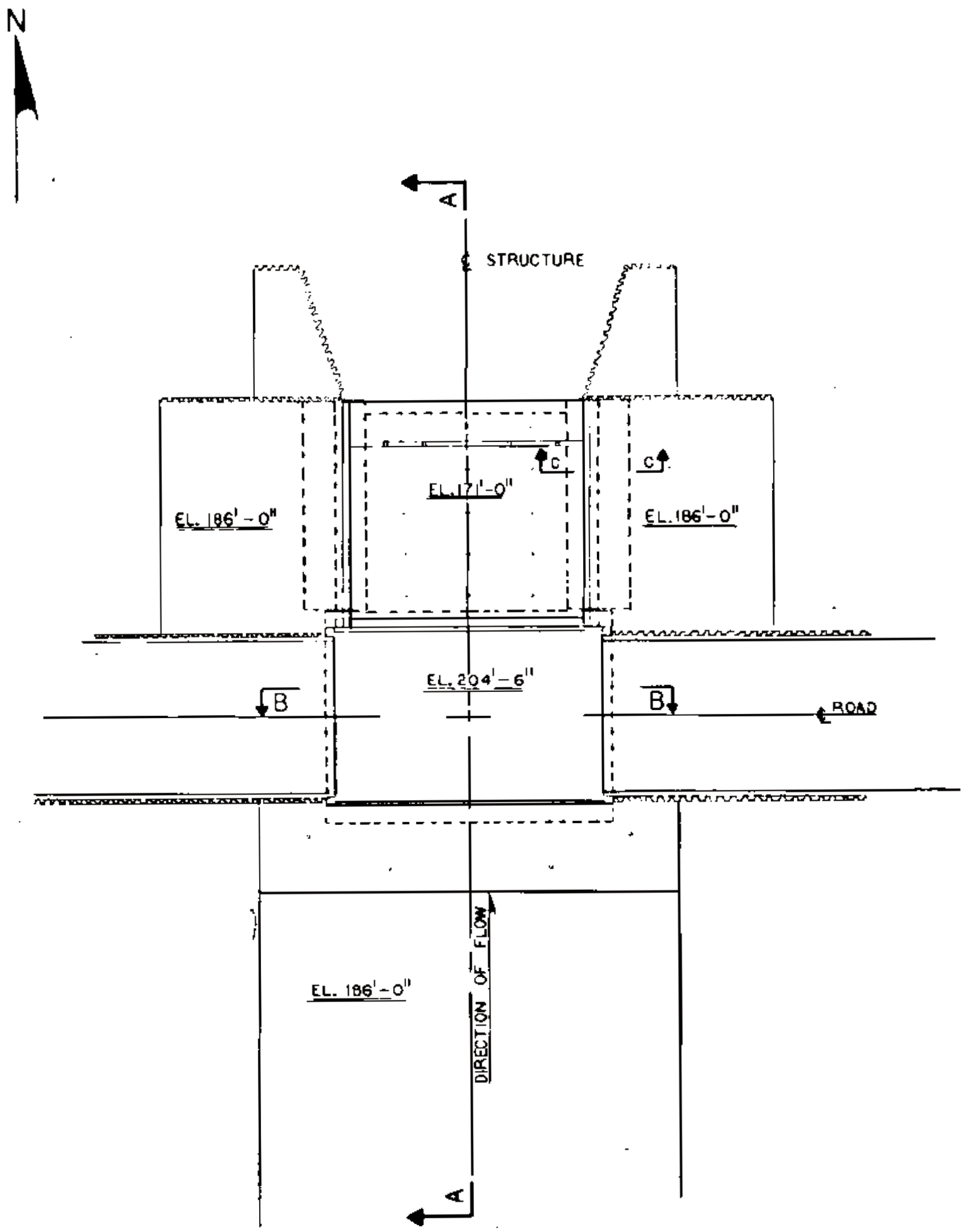
INTAKE STRUCTURE AT STORAGE POND
PLAN AND SECTION

FIGURE 3.8-28 (SHEET 1 OF 2)



BARRIER BETWEEN TRAINS
INSTRUMENT AIR COMPRESSORS AND TANKS
SEISMIC SUPPORT, EACH PUMP
SECTION A-A

REV 21 5/08



PLAN

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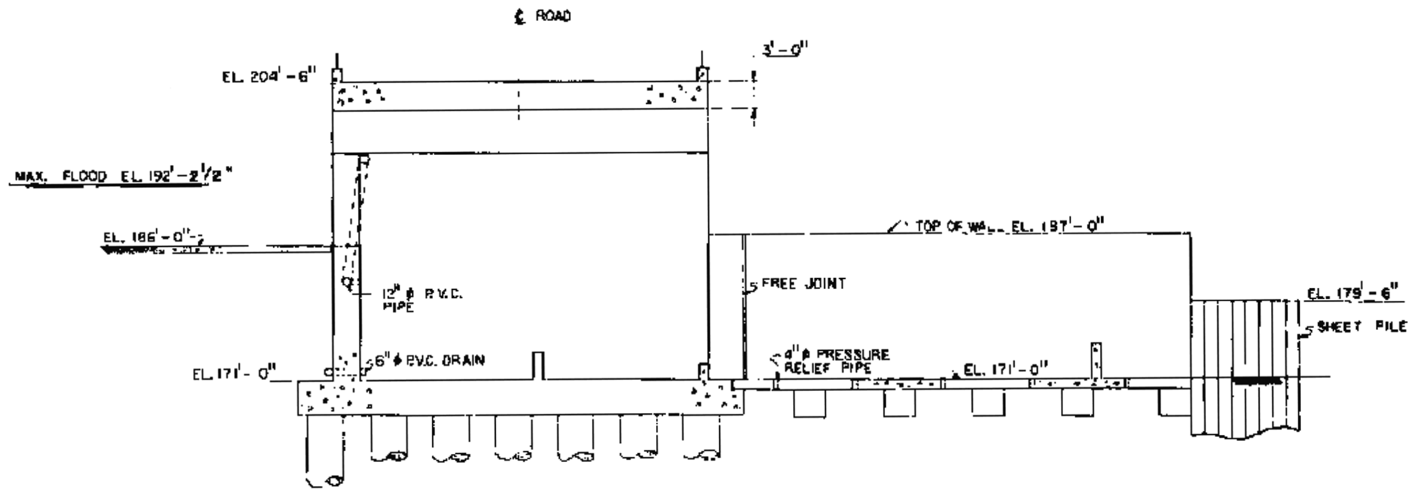
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE PLAN AND SECTIONS

FIGURE 3.8-29 (SHEET 1 OF 4)



SECTION "A-A"

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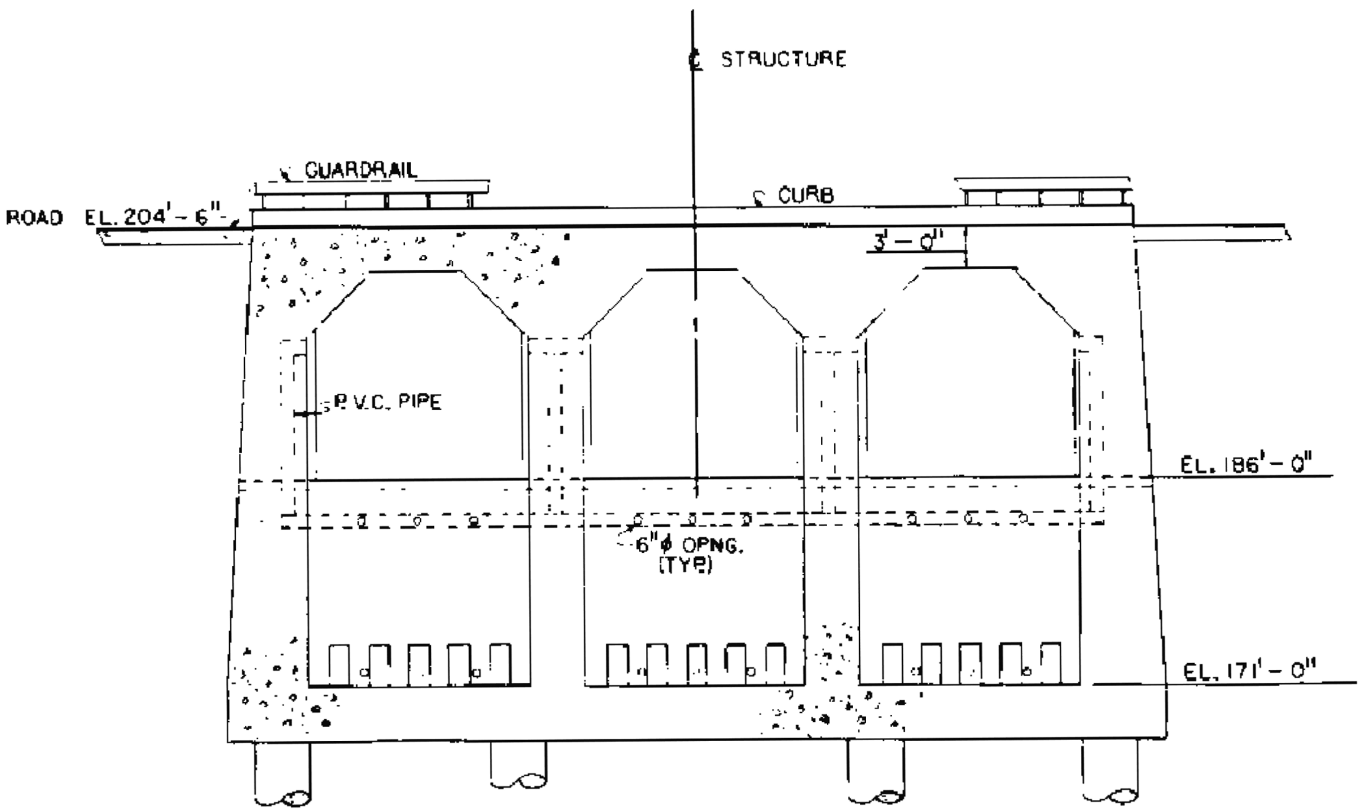
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE PLAN AND SECTIONS

FIGURE 3.8-29 (SHEET 2 OF 4)



SECTION "B-B"

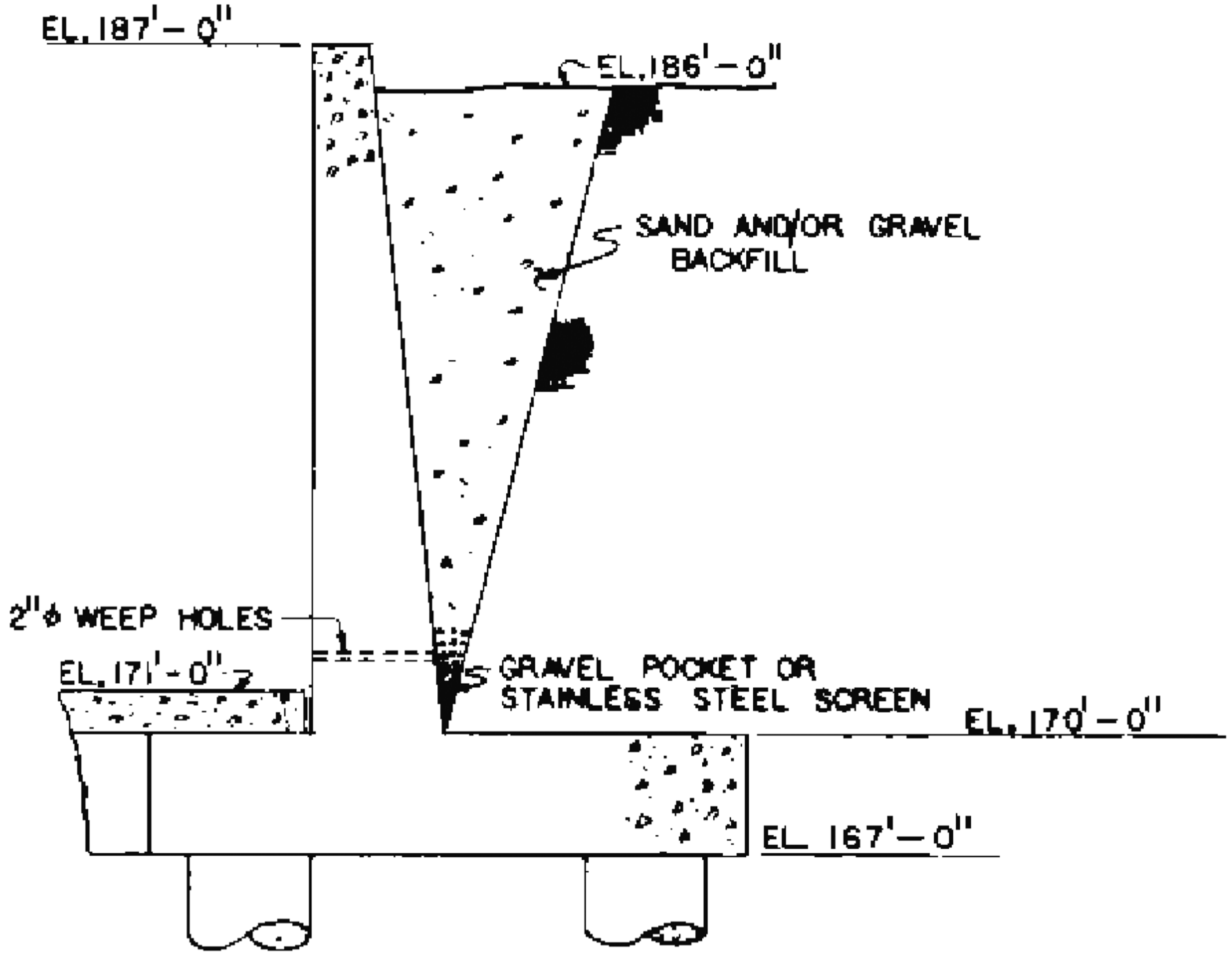
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE PLAN AND SECTIONS

FIGURE 3.8-29 (SHEET 3 OF 4)



SECTION "C-C"

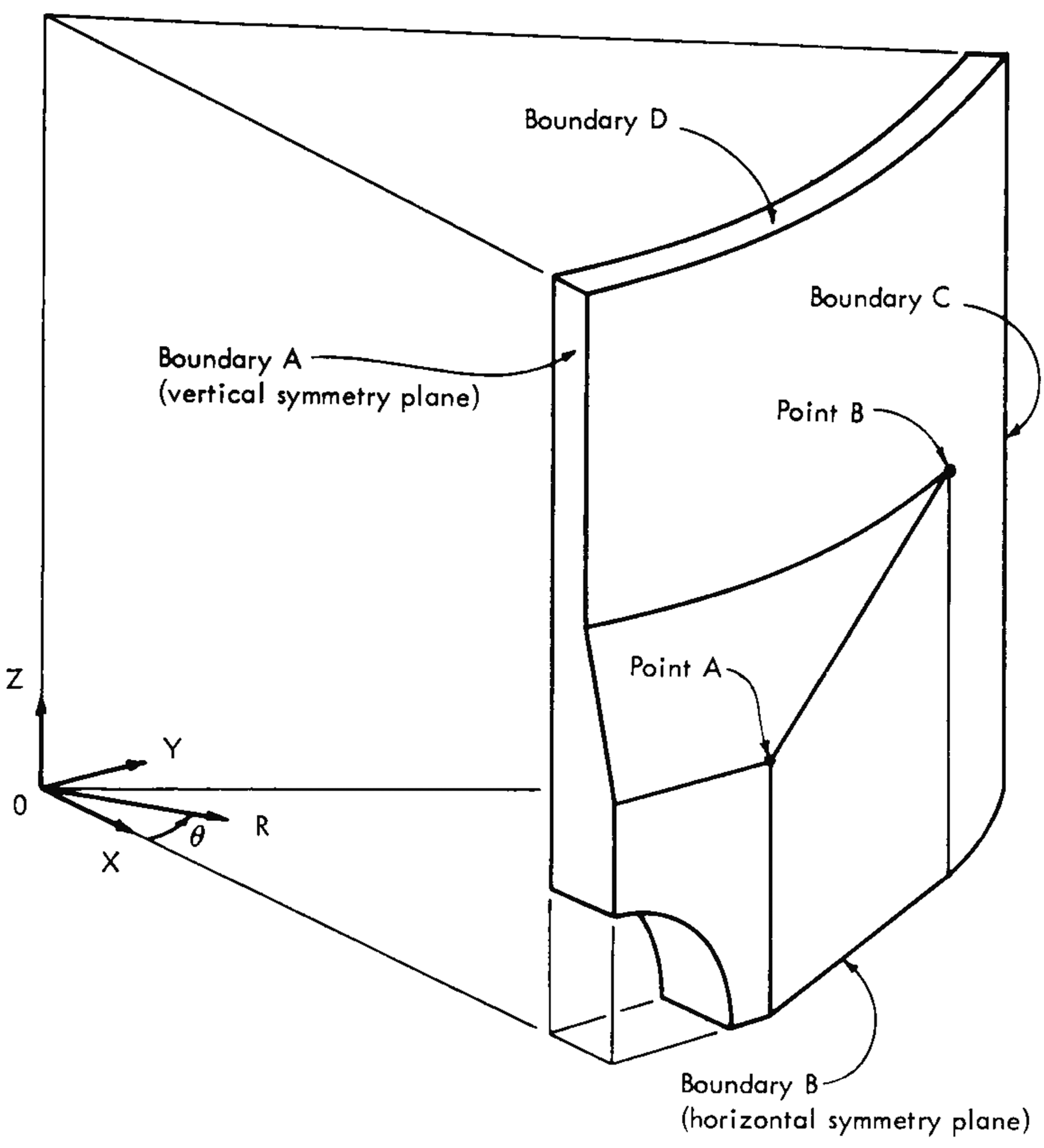
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POND SPILLWAY STRUCTURE PLAN AND SECTIONS

FIGURE 3.8-29 (SHEET 4 OF 4)



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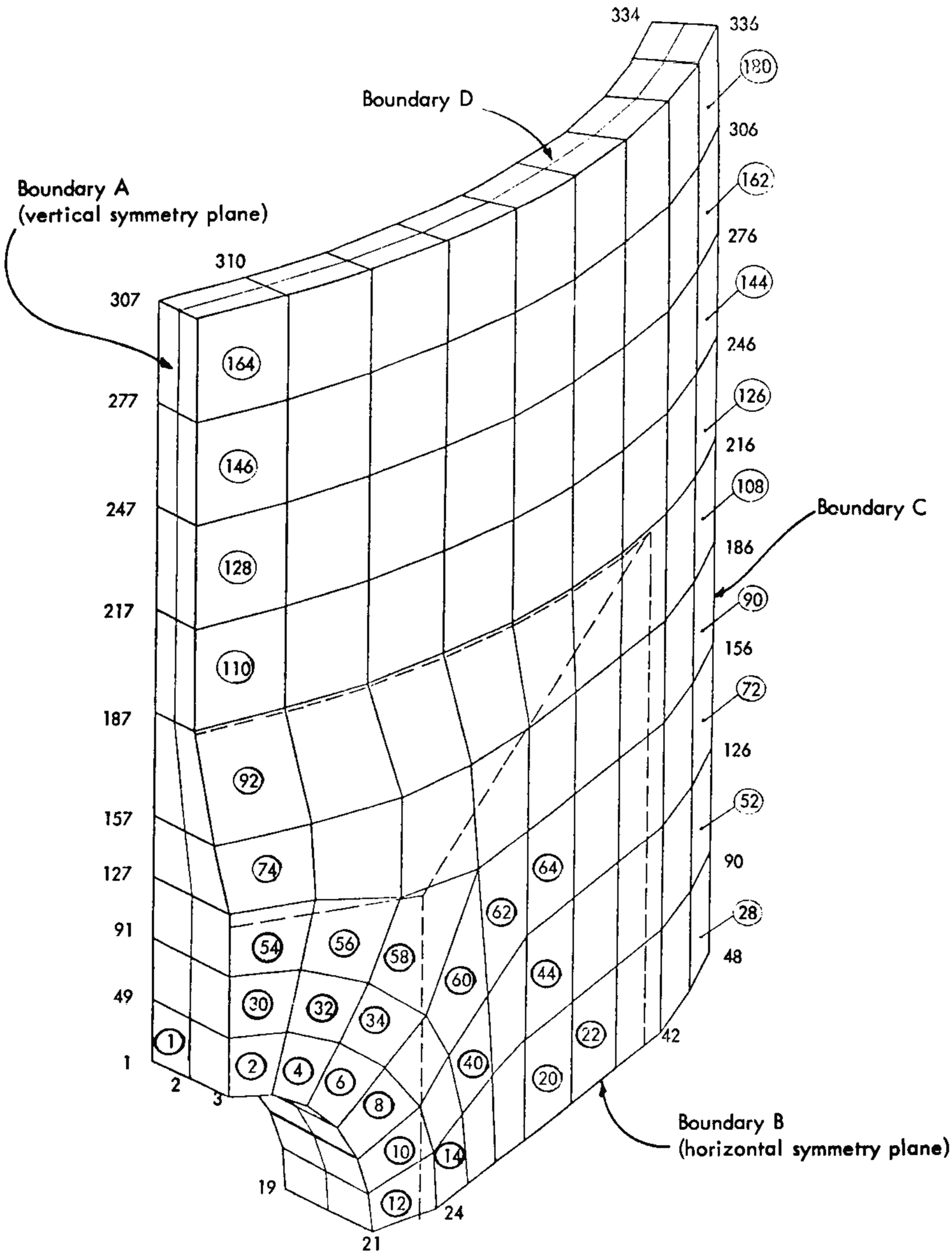
EQUIPMENT HATCH BOUNDARY LINES
FOR THE SAP ANALYSIS

FIGURE 3.8-30



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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

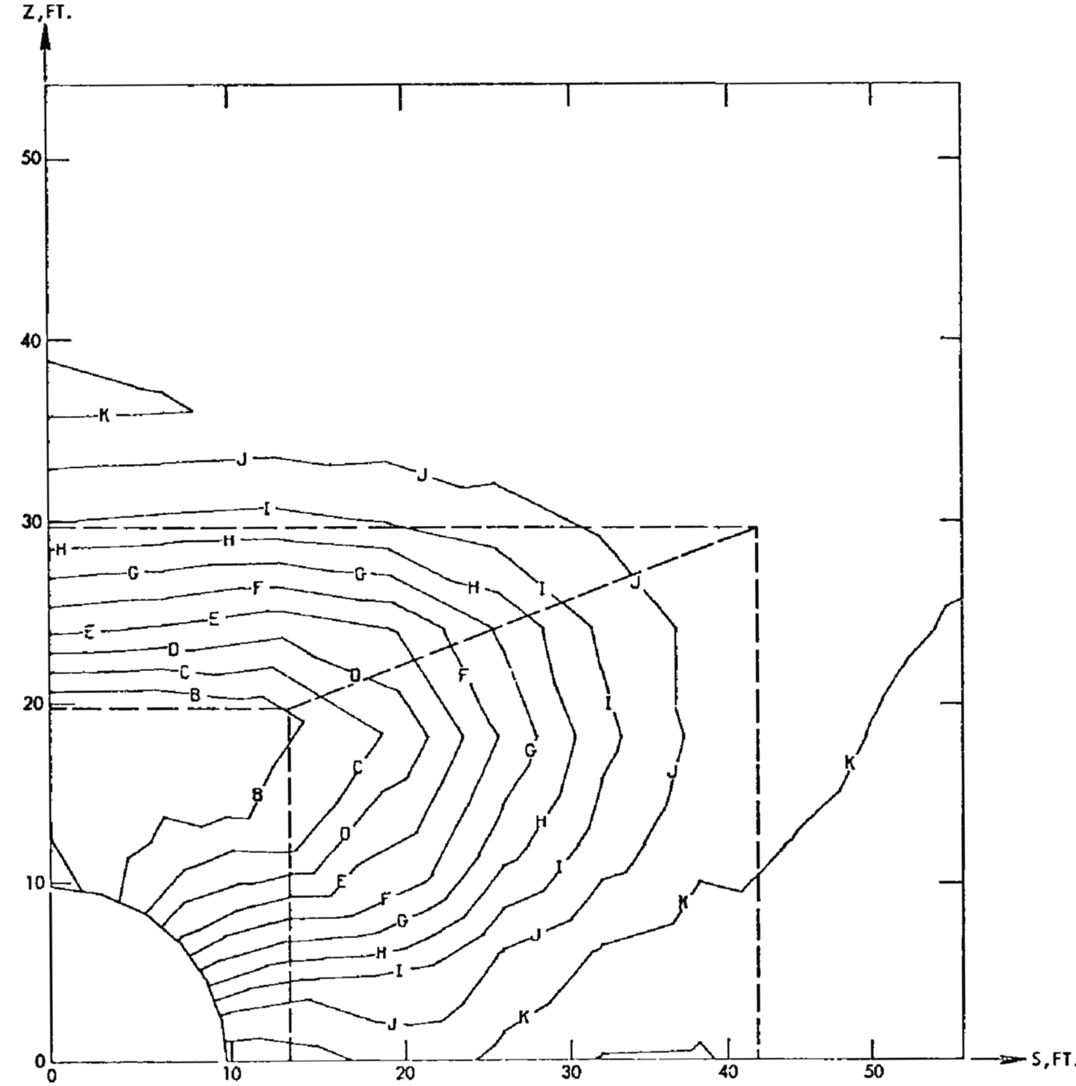
SAP FINITE ELEMENT MESH FOR THE EQUIPMENT HATCH

FIGURE 3.8-31

D+F+1.5P+T_A

VERTICAL STRESSES
OUTSIDE FACE

CONTOUR LEVELS		
A	20.0	KIPS/SQ.FT.
B	40.0	
C	60.0	
D	80.0	
E	100.	
F	120.	
G	140.	
H	160.	
I	180.	
J	200.	
K	220.	
L	240.	



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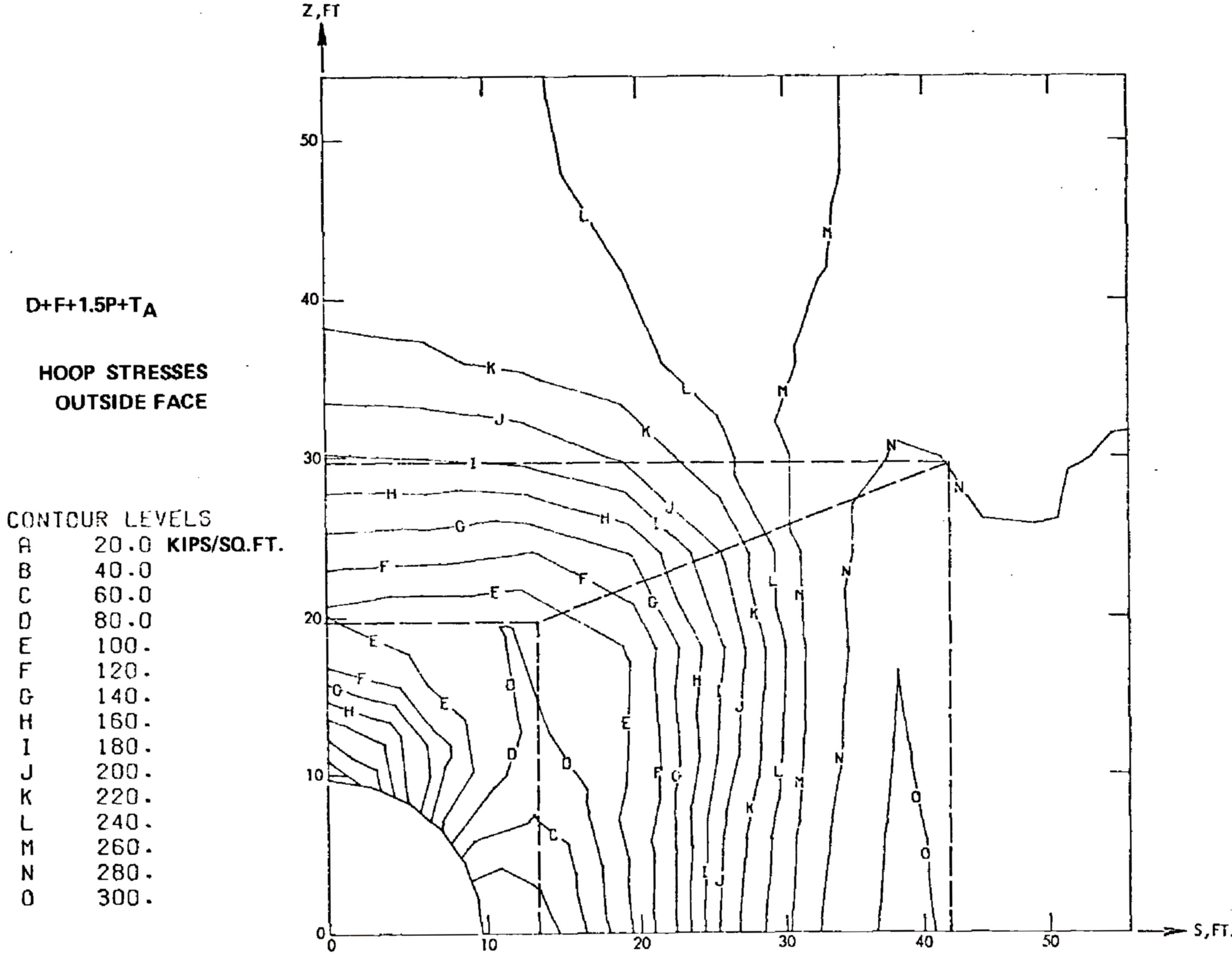
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SAP ANALYSIS OF EQUIPMENT HATCH

FIGURE 3.8-32



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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SAP ANALYSIS OF EQUIPMENT HATCH

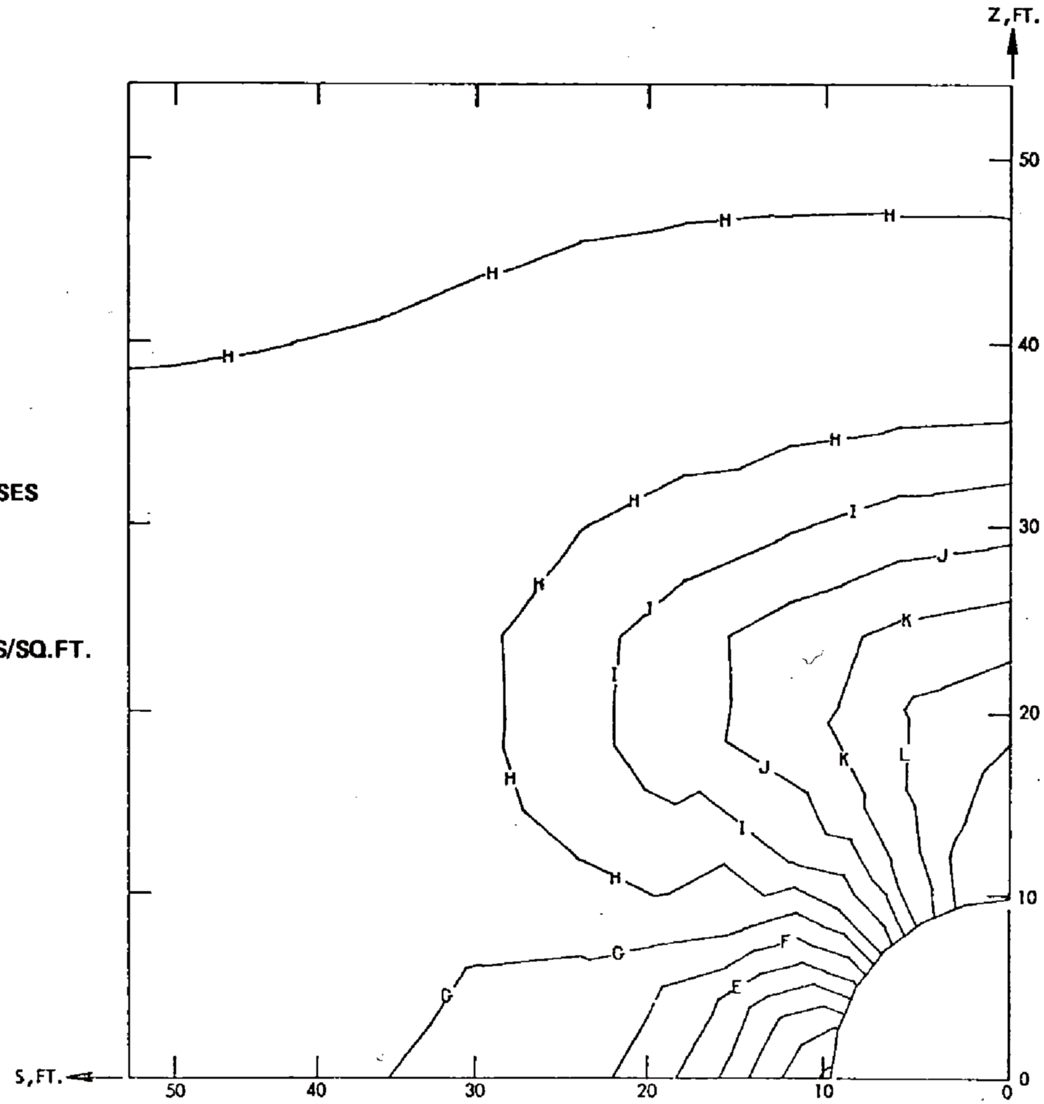
FIGURE 3.8-33

D+F+1.5P+T_A

VERTICAL STRESSES
INSIDE FACE

CONTOUR LEVELS

A	-700.	KIPS/SQ.FT.
B	-670.	
C	-640.	
D	-610.	
E	-580.	
F	-550.	
G	-520.	
H	-490.	
I	-460.	
J	-430.	
K	-400.	
L	-370.	
M	-340.	
N	-310.	



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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

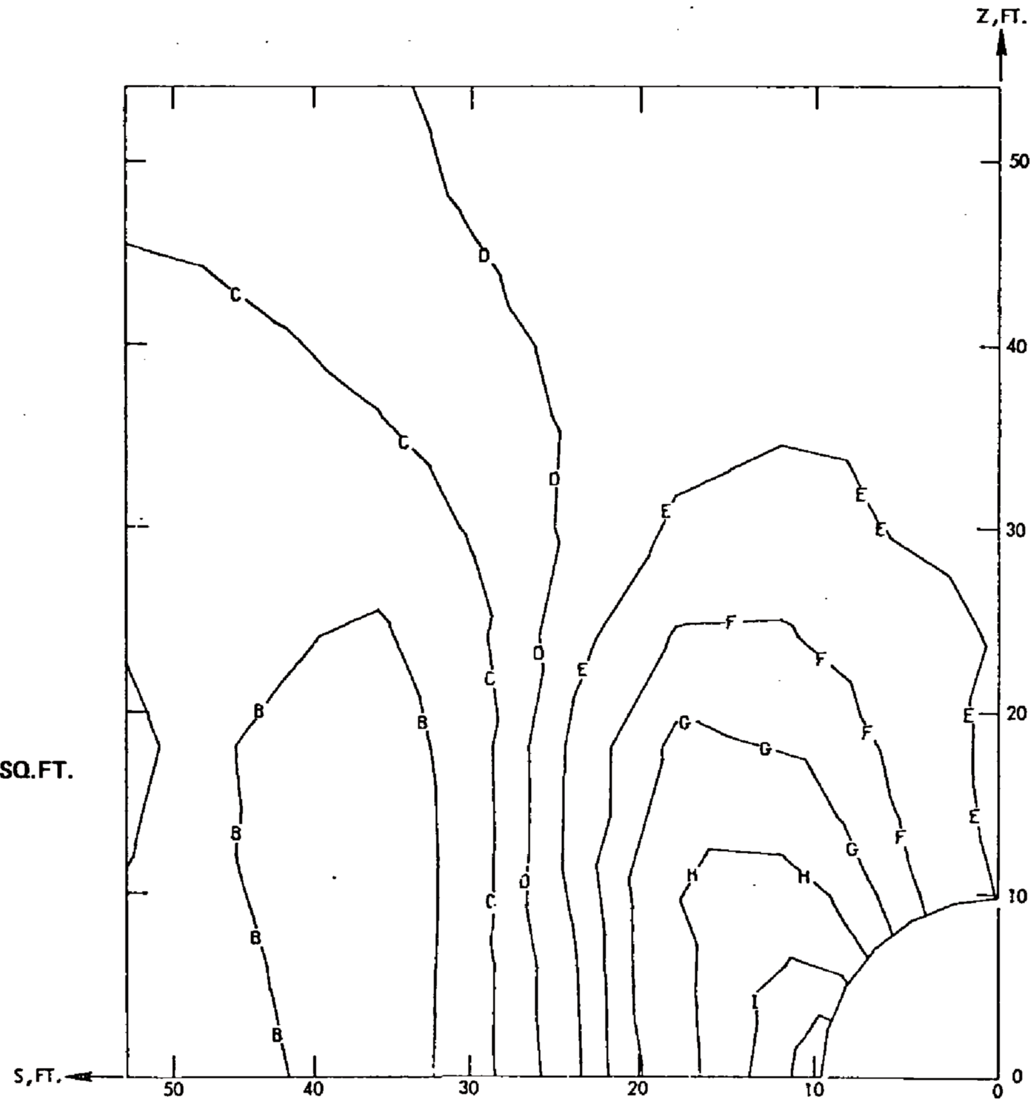
SOUTHERN COMPANY
Energy to Serve Your World®

SAP ANALYSIS OF EQUIPMENT HATCH

FIGURE 3.8-34

D+F+1.5P+T_A

HOOP STRESSES
INSIDE FACE



CONTOUR LEVELS

A	-500.	KIPS/SQ.FT.
B	-480.	
C	-460.	
D	-440.	
E	-420.	
F	-400.	
G	-380.	
H	-360.	
I	-340.	
J	-320.	

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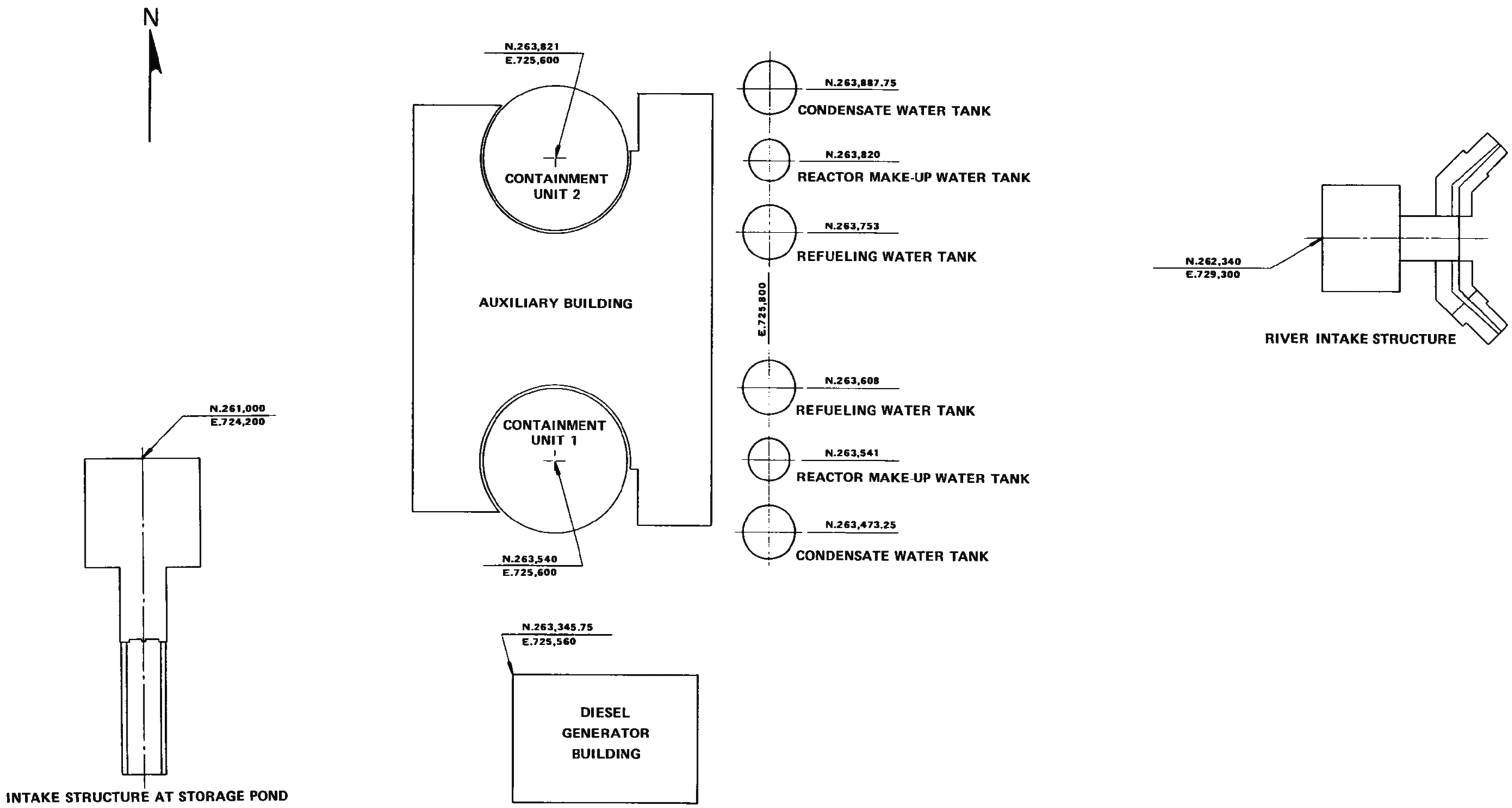
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SAP ANALYSIS OF EQUIPMENT HATCH

FIGURE 3.8-35



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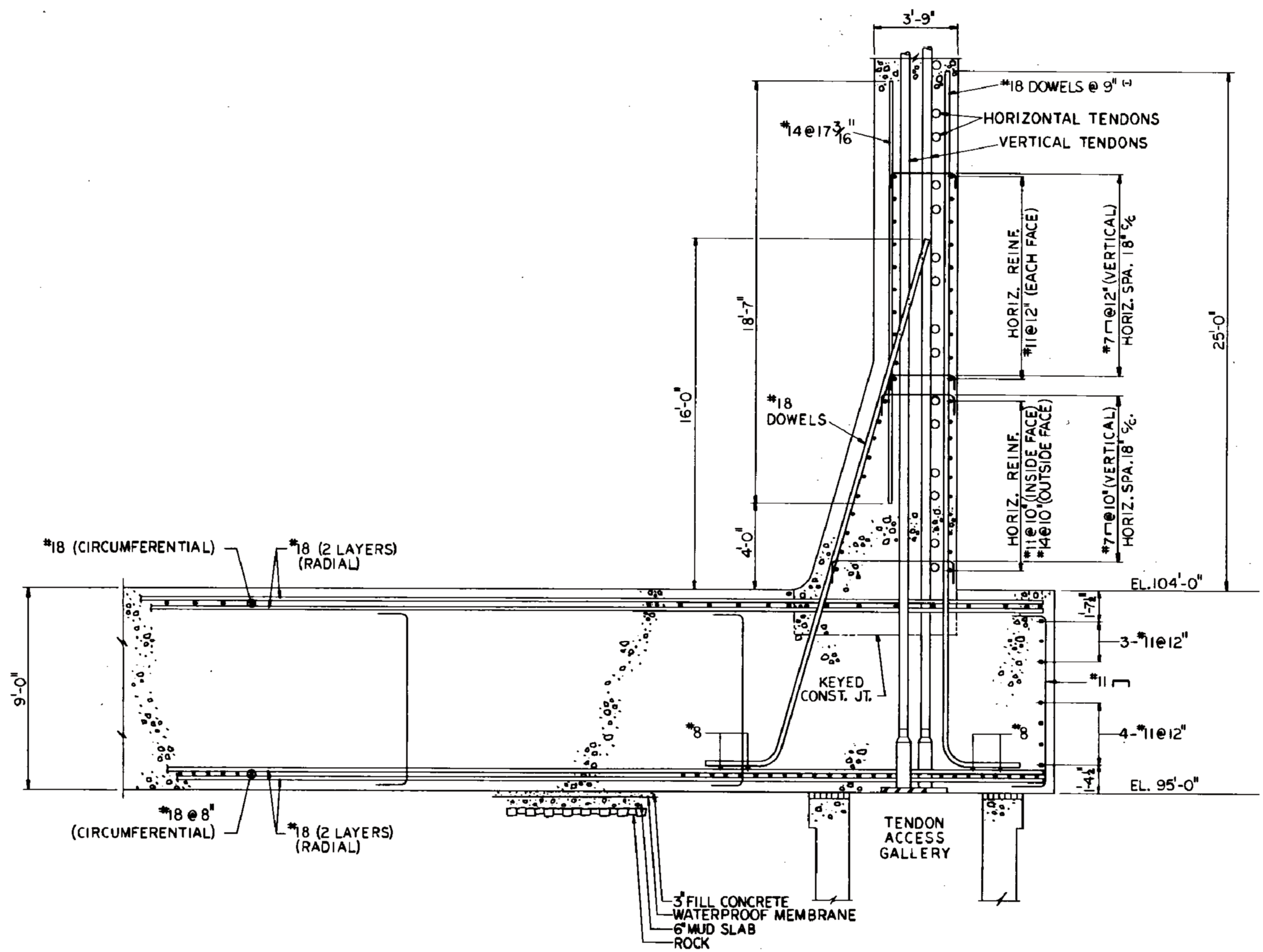
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOCATION PLAN – FOUNDATIONS
FOR CATEGORY 1 STRUCTURES

FIGURE 3.8-36



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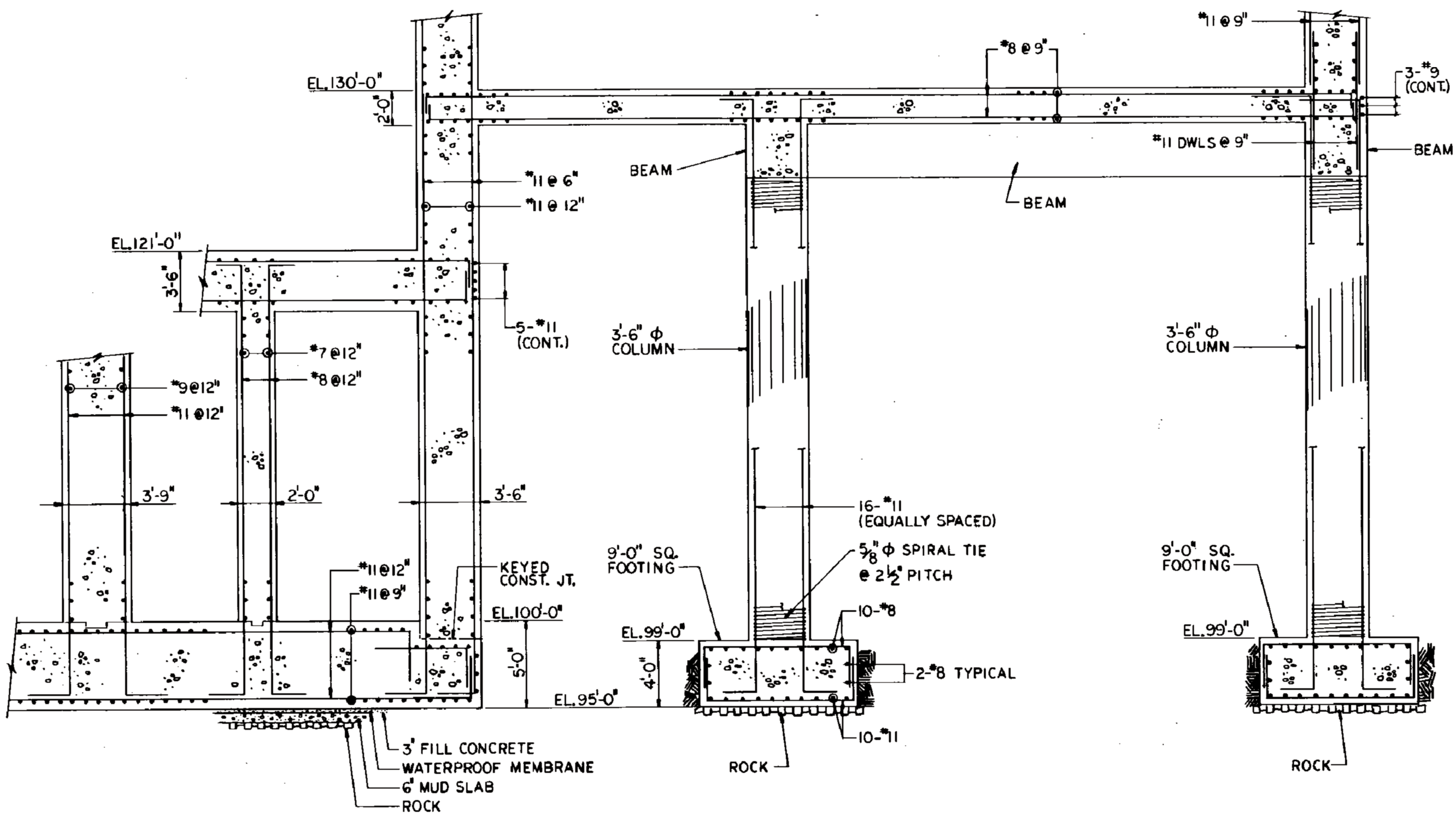
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT BASE SLAB DETAILS

FIGURE 3.8-37



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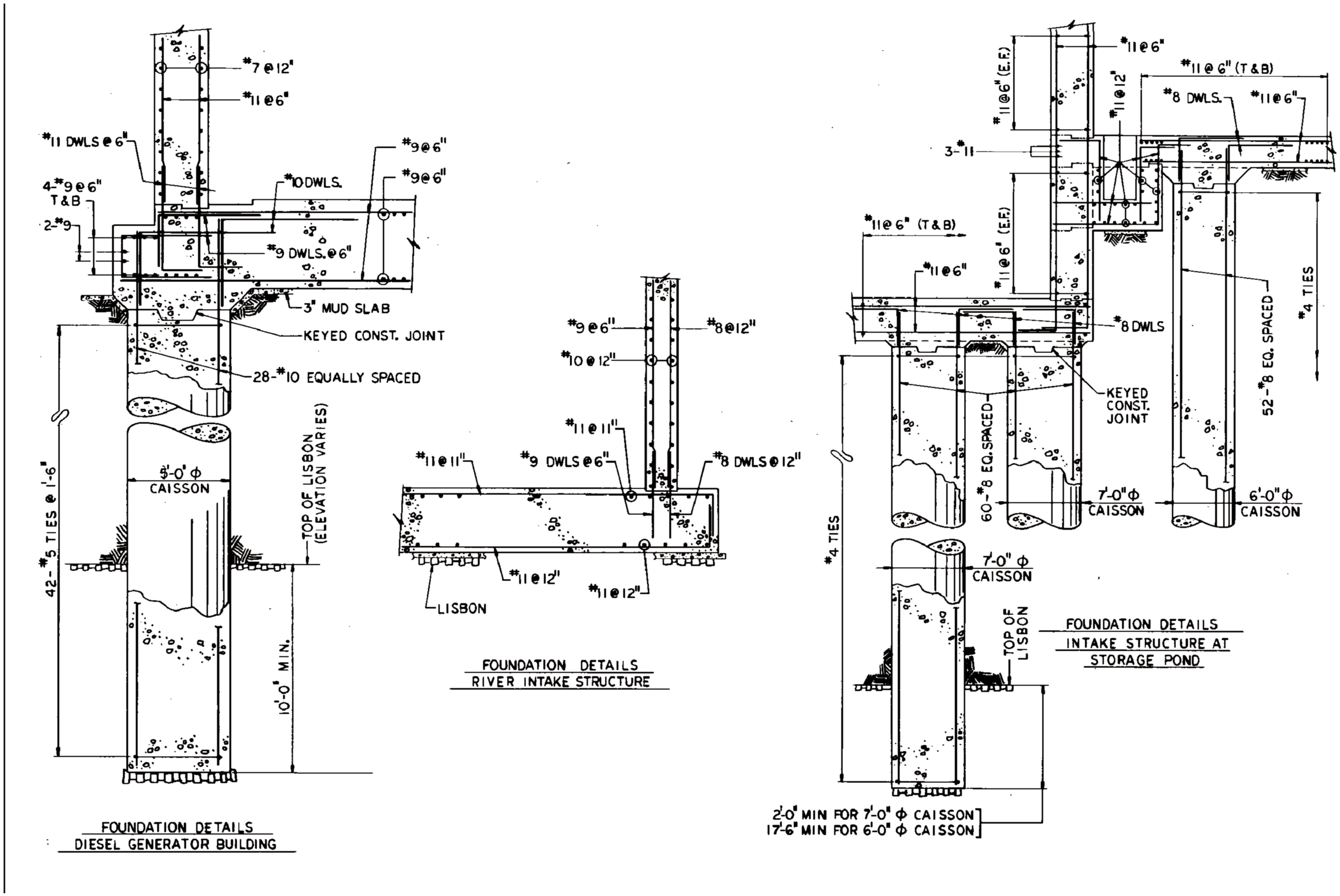
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING BASE SLAB DETAILS

FIGURE 3.8-38



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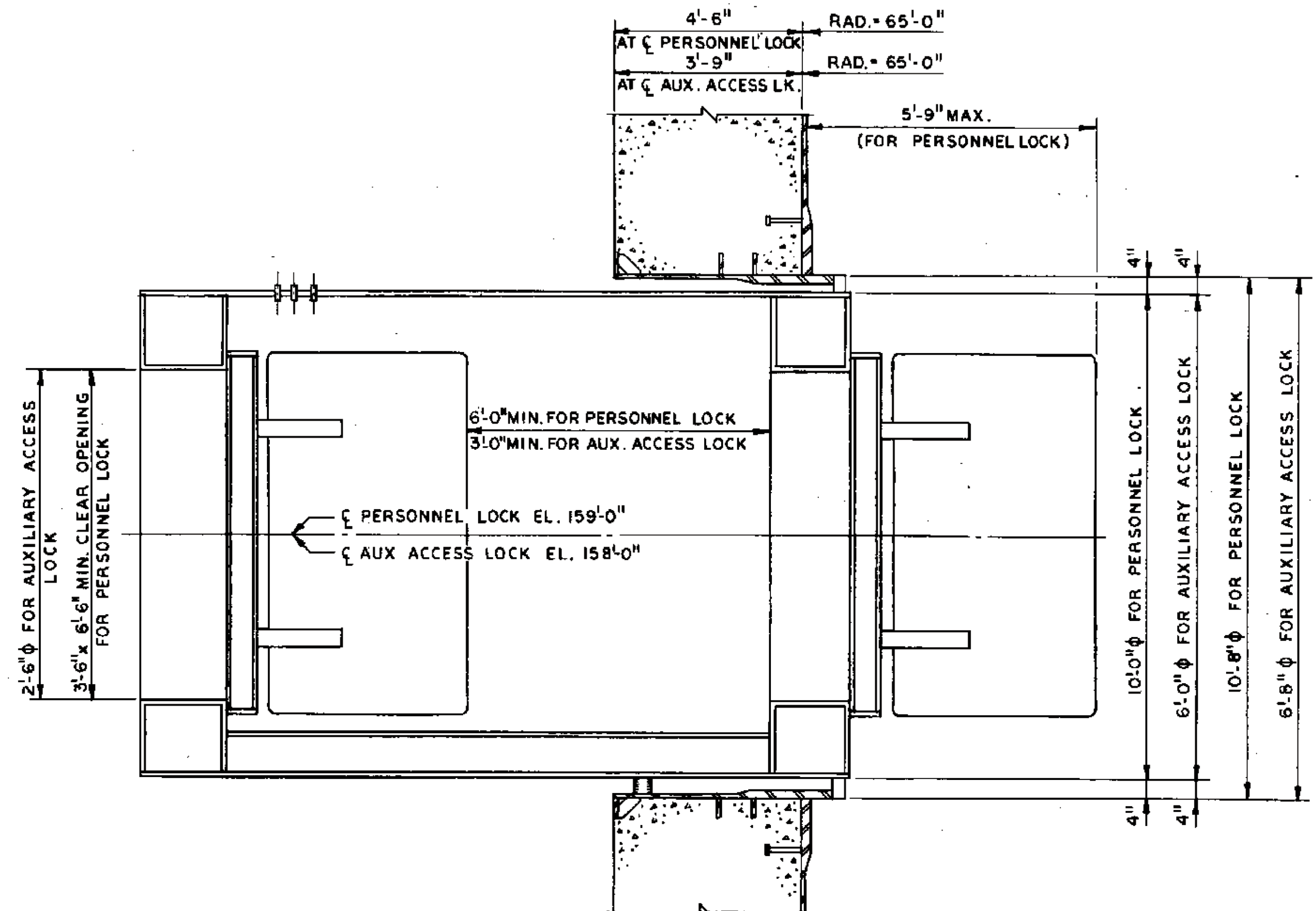
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FOUNDATION DETAILS FOR DIESEL GENERATOR BUILDING, RIVER INTAKE STRUCTURE, AND INTAKE STRUCTURE AT STORAGE POND

FIGURE 3.8-39



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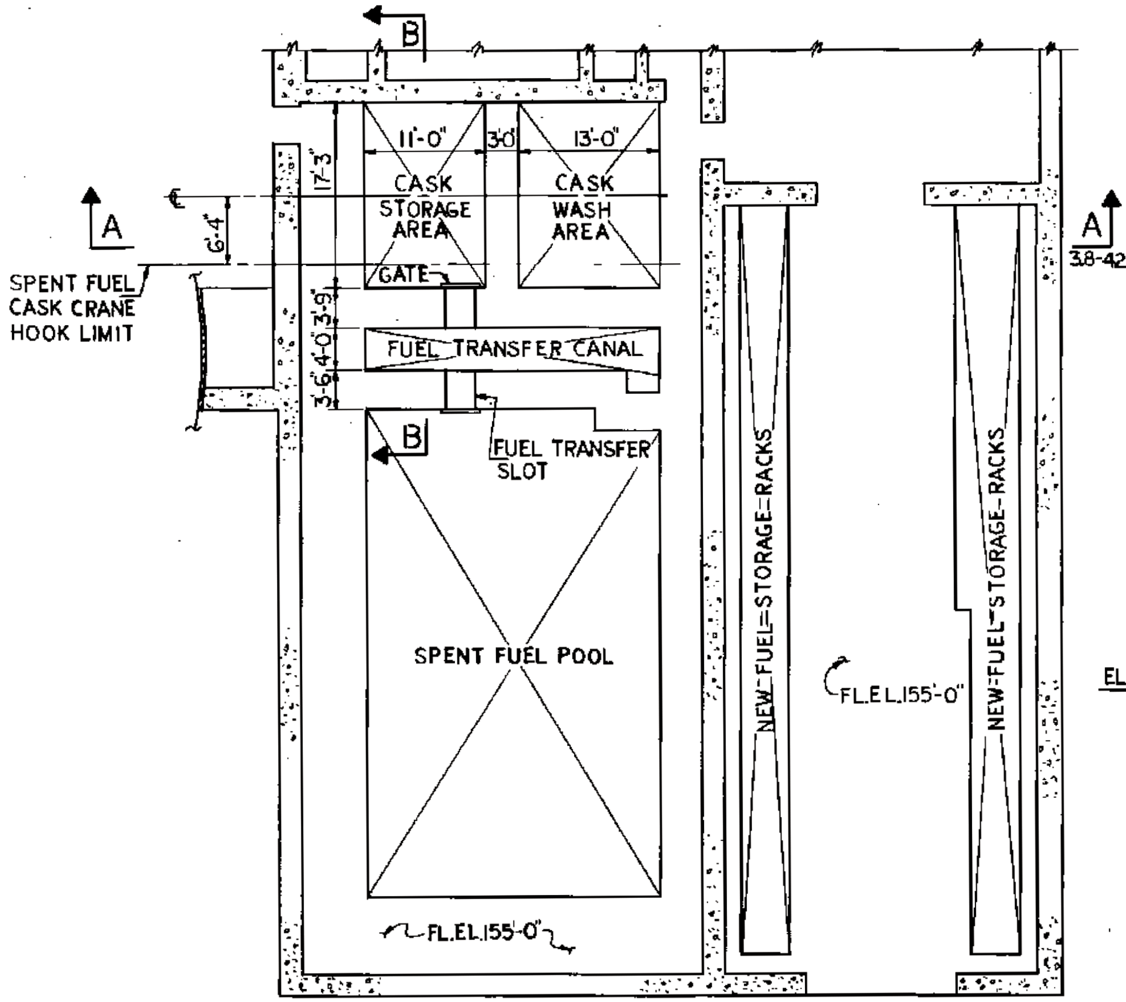
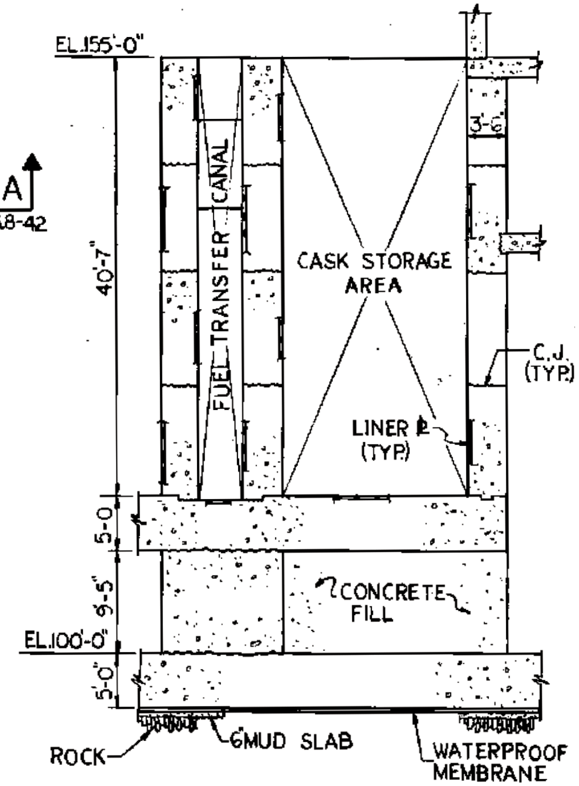
UNIT 1 AND UNIT 2

GEOMETRY OF PERSONNEL LOCK AND AUXILIARY ACCESS LOCK

FIGURE 3.8-40



PLAN AT EL155'-0"



SECTION B-B

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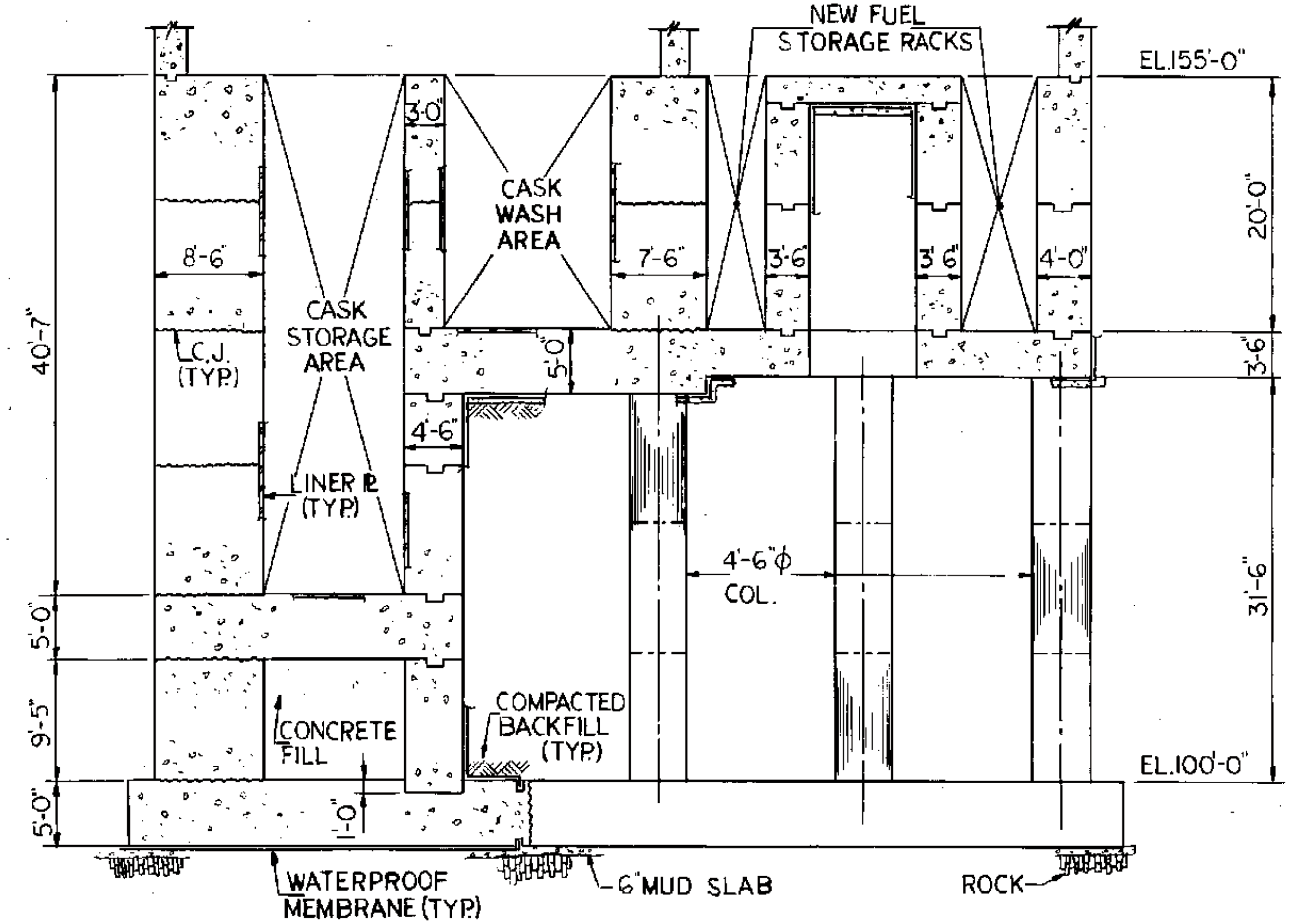
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

AUXILIARY BUILDING CASK WASH AND
CASK STORAGE AREAS PLAN AND SECTION

FIGURE 3.8-41



SECTION A-A

REV 21 5/08

SOUTHERN COMPANY
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**JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2**

AUXILIARY BUILDING CASK WASH AND
CASK STORAGE AREAS SECTION

FIGURE 3.8-42

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 DYNAMIC SYSTEM ANALYSIS AND TESTING

3.9.1.1 Vibration Operational Test Program

Piping vibration and thermal expansion tests were performed during the startup program to conform to Regulatory Guide 1.68 and as outlined in section 14.1. Criteria for the test satisfy the requirements of the applicable portions of ASME Section III Code for Class 1 and 2 components.

Farley Nuclear Plant systems included in this program are: reactor coolant system (RCS), power conversion system (PCS), emergency core coolant systems (ECCS), and chemical and volume control systems (CVCS). These system tests are integrally performed during the hot functional and power ascension test programs. These programs allow system operation at normal operating temperature and including flow modes, temperature plateaus, valve operations, pump starts and stops.

Criteria for these tests, including the basic monitoring locations and the type of monitoring, were coordinated with design groups and the test results were evaluated by the design groups for acceptability. If the acceptance criteria established by the design groups were not satisfied during these tests, then corrective measures were taken to achieve an acceptable system response. Further retests were performed as required to verify the acceptability of design following modifications.

3.9.1.2 Dynamic Testing Procedures

A description of the analyses or tests used in the design of safety related mechanical equipment such as pumps and heat exchangers to withstand seismic loadings is given in subsections 3.7.2 and 3.7.3.

Most of this mechanical equipment is isolated from the effects of the faulted plant condition and, therefore, will see negligible accident loadings. For equipment which is not isolated from the effects of the faulted plant condition, the dynamic accident loads are evaluated.

The tubes in the steam generator are subject to a possible flow induced vibration that does not exist in the primary coolant loop. This vibration could result from flow across the tubes due to vortex shedding. To ensure that no sympathetic vibration is generated by the vortex shedding, there is a wide frequency separation between the vortex frequency of the fluid and the beam frequency of the tube. Parallel flow vibration is analyzed using the correlations of Burgreen, and the amplitude of vibration is shown to be low enough that neither stress, banging, nor fatigue is a problem.

3.9.1.3 Dynamic System Analysis Methods for Reactor Internals

The reactor internals are modeled dynamically for loads produced by a pipe rupture of the largest branch lines attached to the main reactor coolant loop; the design basis accident (DBA), for both cold and hotleg breaks; response due to a safe shutdown earthquake (SSE); and for the most unfavorable combination of LOCA and SSE. Seismic analysis of the reactor vessel and its internals is described in subsections 3.7.2 and 3.7.3.

The upper internals support structure is made of two plates much like a sandwich. The upper support assembly is a plate reinforced by a weldment of a skirt and a grid of beams. The upper core plate is connected to the upper support assembly by hollow columns bolted to the plates. The guide tubes are pinned to the upper core plate and bolted to the upper support assembly. This structure compresses the fuel assembly springs during assembly and is subjected to vertical upward forces from these springs. During operation, normal and abnormal transverse flow drag forces are applied to the columns and guide tubes, and differential pressure exists across the horizontal plates. The forces on the columns and guide tubes vary with the distance from the outlet nozzles. Because of the complexity of the upper package geometry and loading conditions, the modeling of the reactor internals was performed by the structure and matrix displacement for each finite element. This finite element structural analysis computer program permits static elastic, non linear dynamic and plastic analysis. Descriptions of the techniques used to model the various parts of the internals are given in the following paragraphs.

The top structure, deep beam, and upper core plate have been modeled with flat shell elements, the support columns with "three dimensional" beam elements and the fuel assemblies and hold down springs with "three dimensional" spring elements. Because of symmetry, a one-eighth slice of the upper package has been modeled. The core plate is perforated and is modeled as a geometrically equivalent solid plate which has modified elastic constants according to the theory of perforated plates.

Columns of two different lengths are modeled, the long columns between the upper support and upper core plates and the short columns between the beam grid and the upper core plate.

Under the loads used for design, according to the operating condition under study, the previously described computer program provides stresses and deflections at all nodal points.

There is no change in the configuration of the reactor internals core support structures from the 15 x 15 fuel assembly configuration due to the incorporation of the 17 x 17 fuel assembly. The mechanical properties of the 17 x 17 fuel assembly, such as fuel assembly weight and beam stiffness, are similar to the 15 x 15 fuel assembly. Their input to the reactor internals core support structures is similar and the response of the total reactor internals core support structural model is also essentially similar.

3.9.1.3.1 Preoperational Tests

The program used to establish the integrity of reactor internals has involved extensive design analysis, model testing, and post hot functional inspection. Additionally, a full size reactor has been instrumented[(2)] to measure the dynamic behavior of a Farley size plant and has compared measurements with predicted values.

This program was instituted as part of a basic philosophy of instrumenting the internals of the "first-of-a-kind" of the current nuclear steam supply system designs for power plants. The magnitude of this test program was much greater than the intent of the philosophy, and was established as part of an extensive plan to develop theories and basic concepts related to internals vibration under various operating conditions.

Thus, not only is added assurance obtained that all of the hardware will operate in the manner for which it was designed, but these data also assist in the development of increased capability for the prediction of the dynamic behavior of pressurized water reactor (PWR) internals. The previous "first-of-a-kind" plants that were instrumented are R. E. Ginna (two loops), H. B. Robinson No. 2 (three loops) and Indian Point Unit II (four loops).

The H. B. Robinson No. 2 reactor has been established as the prototype for the Westinghouse three-loop plant internals verification program. Subsequent three-loop plants are similar in design. Past experience with other reactors indicates that plants of similar designs behave in a similar manner. For these reasons an instrumentation program was conducted on the H. B. Robinson No. 2 to confirm the behavior of the reactor internal components. The main objectives of this test were to increase confidence in the adequacy of the internals by determining stress or deflection levels at key locations.

In the final analysis, the proof that the internals are adequate, free from harmful vibrations, and have performed as intended is through component observations and examinations during service. With this thought, Robinson, the 3-loop prototype, was subjected to a thorough visual and dye penetrant examination by a qualified Westinghouse quality assurance engineer before and after the hot functional test. This inspection was in addition to the normal inspection of the internals in the shop, and before and after shipment. A visual inspection of the internals was also conducted during the Robinson unit's first refueling in March of 1973. This inspection was performed with the aid of television cameras and borescopes

Also, for the particular case of the three loop plants, the following operating experiences offer additional assurance of the adequacy of this design:

A. Southern California Edison's San Onofre plant is a three-loop plant with a slightly different design. This plant has been in operation since 1967 with no internals vibration problems. The internals have been inspected on various occasions.

B. H. B. Robinson No. 2, after completion of the hot functional inspection, has been at power operation since 1970 with no internals vibration problems.

C. Florida Power and Light's Turkey Point No. 3 and No. 4 have successfully completed the post hot functional inspection, with the results indicating no internals vibration problems.

D. Virginia Electric and Power Company's Surry No. 1 and No. 2 have also successfully completed the post hot functional inspection with similar results.

The only significant differences between the Farley plant internals and the Robinson Plant internals are the replacement of the annular thermal shield with neutron shield panels, and the substitution of 17 x 17 fuel assemblies for 15 x 15 assemblies. In addition, the Farley Unit 1 upper internals were modified after hot functional testing to add additional instrumentation to measure temperatures in the reactor vessel head plenum as described in subsection 4.4.5.4. The design of the special instrumentation stanchion has been reviewed analytically by Westinghouse, using very conservative assumptions for both flow loading and seismic loading under normal, upset, and faulted conditions. (See table 3.9-4.) With the conservative assumptions made, all stresses were found to be within ASME code allowable values as shown in table 3.9-5. No excitation is expected due to vortex shedding since the ratio of natural frequency to the shielding frequency is greater than 3.0.

In addition, the inclusion of the special stanchions and associated hardware increases the weight of the upper internals by approximately 0.5 percent. The design is such that it does not change the structural stiffness of the upper internals nor does it change the normal and upset forcing functions imposed on the internals. Consequently, a negligible effect on the internals vibratory response will be realized, and therefore, no additional preoperational testing is required.

The replacement of the thermal shield with segmented neutron shield panels results in a reduction of the flow induced vibrations of the reactor core structures. This conclusion was confirmed in tests with a 1/24th scale model.[(111)] The flow test was first conducted on a model with a thermal shield and indicated that the vibration levels of the internals were low and levels on the neutron shield panel were negligible. Appendix B of reference 1 presents the test results. Reference 12 justifies in more detail the comparison of the relative effects of replacing the annular thermal shield with neutron shielding pads.

The change to 17 x 17 fuel assemblies results in the use of newly designed guide tubes which are stronger and more rigid than the 15 x 15 guide tubes and hence will be less susceptible to flow induced vibration problems. The remainder of the core structure design has not been changed, and consequently remains identical to the prototype, which has been tested and proven to be well within design expectations and limits.

The Portland General Electric Company's Trojan plant internals were instrumented for strain measurements on the core barrel, and on the 17 x 17 guide tube subject to highest cross flow. The Trojan plant is the lead plant featuring neutron panels and 17 x 17 style internals. The data obtained in this program provides verification of Westinghouse analysis and scale model predictions of 17 x 17 and neutron panel behavior in a full size plant and is applicable to Farley Nuclear Plant.

The Three Loop Internals Assurance Program conducted on H. B. Robinson No. 2, supplemented by the Trojan data on neutron panels and 17 x 17, jointly satisfies the intent of Regulatory Guide 1.20.

The core support structures will receive, in addition to the testing discussed above, the normal pre- and post-hot functional testing examination for integrity per paragraph D, "Regulations for Reactor Internals Similar to the Prototype Design," of Regulatory Guide 1.20. This examination will include the points in figure 3.9-1, summarized as follows:

A. All major load bearing elements of the reactor internals relied upon to retain the core structure in place

B. The lateral, vertical, and torsional restraints provided within the vessel

C. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals

D. Those other locations on the reactor internals components that are similar to those which were examined on the prototype H. B. Robinson No. 2 design.

The inside of the vessel was inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign material were in evidence.

1. <u>Lower Internals</u>

   A particularly close inspection was made on the following items or areas, using a 5X or 10X magnifying glass where applicable. The locations of these areas are shown in figure 3.9-1.

   a. Upper barrel to flange girth weld.

   b. Upper barrel to lower barrel girth weld.

   c. Upper core plate aligning pin. Examined bearing surfaces for any shadow marks, burnishing, buffing, or scoring. Inspected welds for integrity.

   d. Irradiation specimen guide screw locking devices and dowel pins. Checked for lockweld integrity.

   e. Baffle assembly locking devices. Checked for lockweld integrity.

   f. Lower barrel to core support girth weld.

   g. Neutron shield panel screw locking devices and dowel pin cover plate welds. Examined the interface surfaces for evidence of tightness and for lockweld integrity.

   h. Radial support key welds.

   i. Insert screw locking devices. Examined soundness of lockwelds.

   j. Core support columns and instrumentation guide tubes. Checked all the joints for tightness and soundness of the locking devices.

k. Secondary core support assembling welds.

l. Lower radial support keys and inserts.
(Examined for any shadow marks, burnishing, buffing, or scoring. Checked the integrity of the lockwelds.) These members supply the radial and torsion constraint of the internals at the bottom relative to the reactor vessel while permitting axial growth between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing, or shadowing marks that would indicate pressure loading and relative motion between the two parts. Some scoring of engaging surfaces is also possible and acceptable.

m. Gaps at baffle joints. (Checked for gaps between baffle and top former, and at baffle to baffle joints.)

2. Upper Internals

A particularly close inspection was made on the following items or areas, using a magnifying glass of 5X or 10X magnification, where necessary.

The locations of these areas are shown in figure 3.9-1.

a. Thermocouple conduits, clamps, and couplings.

b. Guide tube, support column, and thermocouple column assembly locking devices.

c. Support column and conduit assembly clamp welds.

d. Upper core plate alignment inserts. Examined for any shadow marks, burnishing, buffing, or scoring. Checked the locking devices for integrity of lockwelds.

e. Connections of the support columns mixing devices and orifice plates to the upper core plate. Checked for tightness and lock device integrity.

f. Thermocouple conduit gusset and clamp welds.

g. Thermocouple end plugs. (Checked for tightness.)

h. Guide tube closure welds, tube transition plate welds and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals were subjected to a total operating time at greater than normal full flow conditions (three pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition there was some operating time with only one and two pumps operating.

When no signs of abnormal wear, no harmful vibrations are detected, or no apparent structural changes take place, the three-loop core support structures are considered to be structurally adequate and sound for operations.

3.9.1.4 Correlation of Test and Analytical Results

The dynamic behavior of reactor components has been studied, using experimental data obtained from operating reactors, along with results of model test and static and dynamic tests in the fabricators shops and at plant site. Extensive instrumentation programs to measure vibration on reactor internals (including prototype units of various reactors) have been carried out during preoperational flow tests, and reactor operation.

From scale model tests, information on stresses, displacements, flow distribution and fluctuating differential pressures is obtained. Studies have been performed(2) to verify the validity and to determine the prediction accuracy of models for determining reactor internals vibration due to flow excitation. Similarity laws need to be satisfied to ensure that the model response can be correlated to the real prototype behavior.

Vibration of structural parts during preoperational tests is measured using displacement gauges, accelerometers, and strain transducers. The signals are recorded with magnetic tape recorders. Onsite offsite signal analysis is done using hybrid real time and digital techniques to determine the approximate frequency and phase content. In some structural components the spectral content of the signals include nearly discrete frequency or very narrow band, usually due to excitation by the main coolant pumps and other components that reflect the response of the structure at a natural frequency to broad bands, mechanically or flow induced excitation. Damping factors are also obtained from wave analyses.

In general, the study follows two parallel procedures. Frequencies and spring constants are obtained analytically, and these values are confirmed from the results of the tests.
Damping coefficients are established experimentally, and forcing functions are estimated from pressure fluctuations measured during operation and in models. Once these factors are established, the response can be computed analytically. In parallel, the responses of important reactor structures are measured during preoperational reactor tests and the frequencies and mode shapes of the structures are obtained. Once all the dynamic parameters are obtained, as explained above, the forcing functions can be estimated. These two procedures are not independent; both are performed simultaneously and, when combined, they provide indications of the internals behavior during reactor operation. Internals behavior during reactor operation also is measured using mechanical devices and nuclear noise methods. The last method involves the frequency spectral analysis of signals from out-of-core ion chambers. Information is obtained on the frequency, amplitude, and damping of their vertical and lateral vibrations of the core, because relative motions of the core cause reactivity perturbations and fluctuations in the neutron flux signal level.

Some components, such as control rod guide tubes, fuel rods, and incore instrumentation tubes, are subjected to cross flow and parallel flow with respect to the axis of the structure. In these cases there are numerous theoretical and experimental studies directed toward establishing the response of the structure.[(2)] These studies also provide information on the added apparent mass of the water, which has the effect of decreasing the natural frequency of the component. For both cases, cross and parallel, the response is obtained after the forcing function and the damping of the system is determined.

Cross flow may excite the structure with periodic vortex shedding, which gives rise to a lateral oscillatory lift force perpendicular to the flow direction and a drag force in the flow direction. The dimensionless vortex shedding frequency, or Strouhal number S = fD/V, is a function of the Reynolds number and known for different cross sections. The structure is usually designed in such a manner that its natural frequency in water is considerably higher than the vortex shedding frequency so as to avoid coincidence. The lateral force per unit length is given by

$$F(x,t) = C_L\,[1/2\ P_f\ V(x)^2]D \cos \omega t$$

where C_L is the oscillatory lift coefficient including correlation length effects (C_L depends on the Reynolds number); P_f is fluid density; V is cross flow velocity; D is the characteristic diameter, and ω is the vortex shedding circular frequency. Data obtained from preoperational and shop tests are used to confirm the coefficients used.

The preoperational vibration monitoring test on H. B. Robinson No. 2, the three-loop prototype plant, has been completed. The pre- and postoperational flow test examination of the internals bas been completed indicating that all the components performed as predicted. No evidence of damage or incipient failure has been found.

The testing programs consisted of measurements of the stresses, deflections, and responses of select key points in the internals structures during hot functional and low power physics tests. The main purpose of this testing program was to ensure that no unexpected large amplitudes of vibration existed in the internals structure during operation. The tests were intended to provide data and results on indications of overall core support structure performance and to verify particular stress and deflection quantities.

3.9.1.5 Analysis Methods Under LOCA Loadings

The scope of the different dynamic analysis techniques and methods used to evaluate mechanical systems and components of the Westinghouse pressurized water reactor for loads produced by a double ended pipe rupture of the largest branch lines attached to the main coolant loop (LOCA) is very extensive.

A. Reactor Internals Analysis

Analysis of the reactor internals for blowdown loads resulting from a loss-of-coolant accident is based on the time history response of the internals to simultaneously applied blowdown forcing functions. The forcing functions are defined at points in the system where changes in cross section or direction of flow occur so that differential loads are generated during blowdown transient. The dynamic analysis can employ the displacement method, lumped parameters, and stiffness matrix formulations and assumes that all components behave in a non-linear manner, due to the presence of gaps at certain interfaces, such as gaps at the reactor vessel to core barrel flange, reactor vessel to upper support flange, lower radial keys, upper core plate alignment pins and core barrel outlet nozzles.

In addition, because of the complexity of the system and the components, it is necessary to use finite element stress analysis codes to provide more detailed information at various points.

A comprehensive explanation of all the techniques and analytical methods used cannot be included in the scope of the FSAR. The more important and relevant methods are presented as an overview in paragraph 3.9.1.3 and summarized in the following.

B. Blowdown Forces Due to Cold and Hot Leg Break

A digital computer program called MULTIFLEX(9), which is developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a loss-of-coolant accident, is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM(3) which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. This MULTIFLEX(9) code is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time.

Although spatially one dimensional conservation laws are employed, the code can be applied to describe three dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc., are considered.

The MULTIFLEX(9) code evaluates the pressure and velocity transients for a maximum of 2000 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the programs LATFORC and FORCE2, which utilize detailed geometric descriptions in evaluating the horizontal and vertical loadings on the reactor internals.

Each reactor component for which FORCE2 calculations are required is designated as an element and assigned an element number. Vertical forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element.

2. Flow stagnation on, and unrecovered orifice losses across the element.

3. Friction losses along the element.

Input to the code, in addition to the MULTIFLEX[9] pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

In addition to the vertical forces calculated by FORCE2, the horizontal forces on the vessel, core barrel, and thermal shield are calculated by LATFORC. The horizontal forces are calculated by summing the lateral force components around the vessel, core barrel, and thermal shield, based on the pressure differential across each section, multiplied by the area of each section. This is done at ten different elevations. The total lateral force is calculated by summing the forces over the ten elevations.

The mechanical analysis has been performed using conservative assumptions in order to obtain results with extra margin. Some of the most significant are:

1. When applying the hydraulic forces, no credit is taken for the stiffening effect of the fluid environment which will reduce the deflections and stresses in the structure.

2. The multi-mass model for the Reactor Pressure Vessel (RPV) system described below is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration of the system.

   The RPV system finite element model for the nonlinear time history dynamic analysis consists of three concentric structural sub-models connected by nonlinear impact elements and linear stiffness matrices. The first sub-model represents the reactor vessel shell and its associated components. The reactor vessel is restrained by six reactor vessel supports (situated beneath each nozzle) and by the attached primary coolant piping.

   The second sub-model represents the reactor core barrel assembly, lower support plate, tie plates, and the secondary support components. These sub-models are physically located inside the first, and are connected to them by stiffness matrices at the vessel/internals interfaces. Core barrel to reactor vessel shell impact is represented by nonlinear elements at the core barrel flange, upper support plate flange, core barrel outlet nozzles, and the lower radial restraints.

   The third and innermost sub-model represents the upper support plate assembly consisting of guide tubes, upper support columns, upper and lower core plates, and the fuel. The fuel assembly simplified structural model incorporated into the RPV system model preserves the dynamic characteristics of the entire core. For each type of fuel design, the corresponding simplified fuel assembly model is incorporated into the system model. The third sub-model is connected to the first and second by stiffness matrices and nonlinear elements.

The appropriate dynamic differential equations for finite element system model describing the aforementioned phenomena are formulated and the results are obtained using a general purpose finite element computer code which computes the response at each mode point of the RPV system model. The system model is excited by a set of time dependent horizontal and vertical forces generated by the LATFORC and FORCE2 programs.

The results from the computer program provide time history nodal displacements and nonlinear impact forces at various locations of the reactor vessel and reactor internals interfaces. The methodology used in the RPV system LOCA/seismic analyses is the NRC approved methodology (Reference 23).

C. Reactor Coolant Loop (RCL) Analysis

A flow diagram representing the procedure for the complex time history dynamic solution is shown in figure 3.9-2. The procedure for dynamic solution is iterative in nature since the definition of support stiffness matrices for dynamic behavior (to be incorporated in the reactor coolant loop (RCL) model) depends upon the response of the support points which is not known a priori.

The initial displacement configuration of the mass points is defined by applying the initial steady state hydraulic forces to the unbroken RCL model. For this calculation, the support stiffness matrices for the static behavior are incorporated into the RCL model. For dynamic solution, the unbroken RCL model is modified to simulate the physical severance of the pipe due to the postulated LOCA under consideration. The static support cases (i.e., steam generator columns and reactor coolant pump columns) are included in the dynamic model as stiffness matrices. Other supports such as tie rods, bumper blocks, and hydraulic snubbers, which go directly to ground, are represented in FIXFM by nonlinear elements which correctly define the restraint of the physical element. For supports which cannot be represented by nonlinear elements, the stiffness matrix for dynamic behavior is selected on the basis of anticipated displacement response at the support points.

The natural frequencies and normal modes for the modified RCL dynamic model are determined. The time history hydraulic forces at appropriate node points are combined to determine the forces and moments at structural lumped mass points of interest. After proper coordinate transformation to the RCL global coordinate system, the hydraulic forcing functions are stored on magnetic tape for later use as input to the FIXFM program.

The initial displacement conditions, natural frequencies, normal modes, and the time history hydraulic forcing functions form the input to the FIXFM program which calculates the dynamic time history displacement response for the dynamic degrees of freedom in the RCL model. The displacement response at support points is reviewed to validate the use of support stiffness matrices for dynamic behavior. If the calculated support point response does not match with the anticipated response, the dynamic solution is revised using a new set of support stiffness matrices for dynamic behavior. This procedure is repeated until a valid dynamic solution is obtained.

The time history displacement response from the valid solution is stored on magnetic tape for later use to compute the support loads and to analyze the RCL piping stresses.

The support loads, {F}, are computed by multiplying the support stiffness matrix, [K], and the displacement vector, {δ}, at the support point. The support loads are stored on magnetic tape for use in the support member evaluation. The time history displacement response from the FIXFM program is used as input to the WESDYN-2 program. The program treats this input as an imposed deflection condition on the RCL model and computes the time history of internal forces, deflections, and stresses at each end of the members of the RCL piping system. The results of this solution are stored on magnetic tape for later use in piping stress evaluation.

3.9.1.6 Analytical Methods for ASME Code Class 1 Components

No plastic instability allowable limits given in ASME Section III are used when dynamic analysis is performed. The limit analysis methods have the limits established by ASME Section III for normal, upset, and emergency conditions. For these cases, the limits are sufficiently low to assure that the elastic system analysis is not invalidated. For ASME Code Class I components, the stress limits for faulted loading conditions are specified in section 5.2. For ASME components other than Class 1 and components not covered by the ASME code, the stress limits for faulted loading conditions are specified in subsections 3.9.2 and 3.9.3, respectively. These faulted condition limits are established in such a manner that there is equivalence with the adopted elastic limits and consequently will not invalidate the elastic system analysis. Particular cases of concern are checked by readjusting the elastic system analysis.

3.9.2 ASME CODE CLASS 2 AND 3 COMPONENTS

The design loading combinations and design stress limits for ASME Code Class 2 and 3 components are given below.

3.9.2.1 Plant Conditions and Design Loading Combinations

ASME Code Class 2 and 3 components were not designed to specific plant conditions. However, the design loading combinations used for the design are given in subsection 3.9.2.2, below.

3.9.2.2 Design Loading Combination

Table 3.9-1 presents the various loading combinations for ASME Class 2 and 3 components.

3.9.2.3 Design Stress Limits

The design stress limits for the various design loading combinations are given in table 3.9-1.

Where no design stress limits were defined at the time of purchase, the design limits were specified by the vendors for the various ASME Class 2 and 3 components. These limits were based on having no gross deformation of the components that would render the components incapable of performing their intended safety functions.

For the ASME Class 2 and 3 piping, where no design stress limits were defined at the time of design, the stress allowables were based on having no gross deformation of the piping. The analysis also demonstrated that the piping would not transmit loads to the components connected to the piping which would exceed vendor allowable loads.

3.9.2.4 Analytical and Empirical Methods for Design of Pumps and Valves

The design methods for pumps and valves are described in table 3.9-1.

The methods used to assure operability are provided in subsection 3.9.4.

3.9.2.5 Design and Installation Criteria, Pressure-Relieving Devices

All overpressure relief valves and their connected piping (i.e. headers, header connections, and discharge piping) are designed to withstand the following conditions without exceeding the applicable code's primary stress allowable. The maximum loads due to valve discharge thrust internal pressure, deadweight, and earthquake are applied simultaneously. When more than one relief valve is attached to a piping system, the loads due to all relief valves discharging simultaneously are applied to the system along with the above mentioned primary loads. In addition, the loads from the most critical combination of valves discharging are applied. The local stresses in the main steam line outside the containment at the connection of the relief valves were computed as specified in "Welding Research Council Bulletin", No. 107, and held below the allowable stress level S_h defined in Section NC-3611.1-(b.4) of Section III, 1971 Edition, and modified according to Section NC-3612.3.

A static analysis was initially used in the analysis of safety and safety relief valves and a dynamic analysis performed to verify the adequacy of the design.

3.9.2.6 Stress Levels for Category I Components

Methods used to analyze Category I systems are discussed in subsections 3.9.1 and 3.9.2. Stress analysis results are documented in the applicable piping system stress calculations.

3.9.2.7 Field Run Piping System

Piping classified under ASME Section III, Classes 2 and 3, and analysis, was routed on the piping design drawings, but dimensioned in the field. Detail isometrics were prepared for those pipes dimensioned in the field and forwarded to the project engineer for review and analysis of seismic stress, thermal stress, shielding, and thermal insulation requirements as needed. The approved isometrics were then released for permanent installation.

3.9.2.8 Class 2 and 3 Component Supports

The stress limits used for ASME Class 2 and 3 component supports are identical to those used for the supported component. These allowed stresses are such that the design requirements for the components and the system structural integrity are maintained.

3.9.3 COMPONENTS NOT COVERED BY ASME CODE

Core and Internals Integrity Analysis (Mechanical Analysis)

The response of the reactor core and vessel internals under excitation produced by a simultaneous complete severance of a reactor coolant pipe and seismic excitation for a typical Westinghouse pressurized water reactor plant internals has been determined. The following mechanical functional performance criteria apply:

A. Following the design basis accident the basic operational or functional criterion to be met for the reactor internals is that the plant shall be shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This criterion implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

B. For large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. The subsequent refilling of the core by the emergency core cooling system uses borated water to maintain the core in a subcritical state. Therefore, the main requirement is to ensure effectiveness of the emergency core cooling system. Insertion of the control rods, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

C. The functional requirements for the core structures during the design basis accident are shown in table 3.9-2. The inward upper barrel deflections are controlled to ensure no contacting of the nearest rod cluster control guide tube. The outward upper barrel deflections are controlled in order to maintain an adequate annulus for the coolant between the vessel inner diameter and core barrel outer diameter.

D. The rod cluster control guide tube deflections are limited to ensure operability of the control rods.

E. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited to the value shown in table 3.9-2.

F. The reactor has mechanical provisions that are sufficient to maintain the design core and internals and to ensure that the core is intact with acceptable heat transfer geometry following transients arising from the design basis accident operating conditions.[4][8]

G. The core internals are designed to withstand mechanical loads arising from 1/2 SSE, SSE, and pipe ruptures.[4][5][6][8]

3.9.3.1 Faulted Conditions

The following events are considered in the faulted conditions category:

A. Loads produced by a double ended pipe rupture of the largest branch lines attached to the main coolant loop design basis accident, for both cases: cold and hot leg break. Branch line breaks, rather than main loop piping breaks are analyzed in accordance with the leak-before-break exemptions to GDC-4 discussed in FSAR chapter 3.6 references[3][4][5]. The methods of analysis adopted are related to the type of accident assumed (cold leg break or hot leg break).

B. Response due to an SSE.

C. Most unfavorable combination of a safe shutdown earthquake and a design basis accident. Maximum stresses obtained in each case are conservatively added using the square root of the sum of the squares method.

Maximum stress intensities are compared to allowable stresses for each of the above conditions. Elastic analysis on each component is performed on an elastic basis. For faulted conditions, stresses may be above yield in a few locations. For these cases only, when deformation requirements exist, a plastic analysis is independently performed to ensure that functional requirements are maintained (guide tubes deflections and core barrel expansion). The elastic limit allowable stresses are used to compare with the result of the analysis.

The above described analyses show that the stresses and deflections that would result following a faulted condition are less than those that would adversely affect the integrity of the structures. Also, the natural and applied frequencies are such that resonance problems should not occur.

3.9.3.2 Structural Response of Reactor Vessel Internals During LOCA and Seismic Conditions

3.9.3.2.1 Structural Model and Methods of Analysis

The response of reactor vessel internals due to an excitation produced by a complete severance of auxiliary loop piping is analyzed. With the acceptance of Leak-Before-Break (LBB) by USNRC, References[3][4][5] of Chapter 3.6, the dynamic effects of main coolant loop piping no longer have to be considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of auxiliary lines need to be considered, and consequently, the components will experience considerably less loads than those from the main loop line breaks.

Assuming that such a pipe break in cold leg occurs in a very short period of time (1 millisecond), the rapid drop of pressure at the break produces a disturbance that propagates through the reactor vessel nozzle into the down-comer (vessel and barrel annulus) and excites the reactor vessel and the reactor internals. The characteristics of the hydraulic excitation combined with those of the affected structures present a unique dynamic problem. Because of the inherent gaps that exist at various interfaces of the reactor vessel/reactor internals/fuel, the problem becomes that of nonlinear dynamic analysis of the RPV system. Therefore, nonlinear dynamic analyses (LOCA and seismic) of the RPV system include the development of LOCA and seismic forcing functions which are also discussed here.

3.9.3.2.2 Structural Model

The RPV system finite element model for the nonlinear time history dynamic analysis consists of three concentric structural sub-models connected by nonlinear impact elements and linear stiffness matrices. The first sub-model represents the Farley reactor vessel shell and its associated components. The reactor vessel is restrained by six reactor vessel supports (situated beneath each nozzle) and by the attached primary coolant piping.

The second sub-model represents the Farley reactor core barrel assembly, lower support plate, tie plates, and the secondary support components. These sub-models are physically located inside the first, and are connected to them by stiffness matrices at the vessel/internals interfaces. Core barrel to reactor vessel shell impact is represented by nonlinear elements at the core barrel flange, upper support plate flange, core barrel outlet nozzles, and the lower radial restraints.

The third and innermost sub-model represents the Farley upper support plate assembly consisting of guide tubes, upper support columns, upper and lower core plates, and the fuel. The fuel assembly simplified structural model incorporated into the RPV system model preserves the dynamic characteristics of the entire core. For each type of fuel design, the corresponding simplified fuel assembly model is incorporated into the system model. The third sub-model is connected to the first and second by stiffness matrices and nonlinear elements.

3.9.3.2.3 Analysis Technique

The Westinghouse Electric Computer ANalysis (WECAN) Computer Code, Reference[18], which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of discrete members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformations, the element global matrices and arrays are assembled into global structural matrices and arrays and used for dynamic solution of the differential equation of motion for the structure.

The WECAN Code solves equation of motions using the nonlinear modal superposition theory. Initial computer runs such as dead weight analyses and the vibration (modal) analyses are made to set the initial vertical interface gaps and to calculate eigenvalues and eigenvectors. The modal analysis information is stored on magnetic tapes and is used in subsequent computer runs which solve equations of motions. The first time step performs the static solution of equations to determine the steady state solution under normal operating hydraulic forces. After the initial time step, WECAN calculates the dynamic solution of equations of motions and nodal displacements, and the impact forces are stored on tape for post-processing.

The fluid-solid interactions in the LOCA analysis are accounted through the hydraulic forcing functions generated by MULTIFLEX Code, Reference(9). Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wave front with low pressure on one side and high pressure on the other.

Depressurization waves propagate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle. After a postulated cold leg break, the depressurization path for waves entering the reactor vessel is through the nozzle that contains the broken pipe and into the region between the core barrel and the reactor vessel (i.e., down-comer region). The initial wave propagates up, around, and down the down-comer annulus, then up through the region circumferentially enclosed by the core barrel, that is, the fuel region. In the case of a cold leg break, the region of the down-comer annulus close to the break depressurizes rapidly, but because of the restricted flow areas and finite wave speed (approximately 3000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and the reactor vessel. As the depressurization wave propagates around the down-comer annulus and up through the core, the core barrel differential pressure reduces, and similarly, the resulting hydraulic forces drop.

In the case of a postulated break in the hot leg, the wave follows a similar depressurization path, passing through the outlet nozzle and directly into the upper internals region, depressurizing the core, and entering the down-comer annulus from the bottom exit of the core barrel. Thus, after an RPV outlet nozzle break, the down-comer annulus would be depressurized with very little difference in pressure forces across the outside diameter of the core barrel. A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the down-comer annulus is not directly involved), and internal differential pressures are not as large as for a cold leg break of the same size. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX(9) computer code calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and early two-phase (saturated) blowdown regimes. The MULTIFLEX code employs the method of characteristics to solve the conservation laws, and it assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture. As mentioned earlier, the MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by a separate spring-mass oscillator system. A beam model of the core support barrel has been developed from the structural properties of the core barrel. In this model, the cylindrical barrel is vertically divided into equally spaced segments, and the pressure as well as the wall motions are projected onto the plane parallel to the broken nozzle. Horizontally, the barrel is divided into 10 segments; each segment consists of three separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible wall is determined by solving the global equations of motions for the masses representing the forced vibration of an undamped beam.

In order to obtain the response of reactor pressure vessel system (vessel/internals/fuel), the LOCA horizontal and vertical forces obtained from the LATFORC and FORCE2 Codes are applied to the finite element system model. The transient response of the reactor internals consists of time history nodal displacements and time history impact forces.

3.9.3.2.4 Seismic Analysis

The basic mathematical model for seismic analysis is essentially the same as the LOCA model except for some minor differences. In the LOCA model, the fluid-structure interactions are accounted through the MULTIFLEX Code; whereas, in the seismic model, the fluid-structure interactions are included through the hydrodynamic mass matrices in the down-comer region. Another modeling difference is the difference between loop stiffness matrices. The seismic model uses the unbroken loop stiffness matrix, whereas, the LOCA model uses the broken loop stiffness matrix. Except for these two differences, the RPV system seismic model is identical to that of LOCA model.

The horizontal fluid-structure or hydroelastic interaction is significant in the cylindrical fluid flow region between the core barrel and the reactor vessel annulus. Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel, thermal shield and the reactor vessel. The mass matrices for the hydroelastic interactions of two concentric cylinders are developed using the work of reference(19). The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel shell, thermal shield, and core barrel. The off-diagonal terms reflect the fact that all the water mass does not participate when there is no relative motion of the vessel and core barrel. It should be pointed out that the hydrodynamic mass matrix has no artificial virtual mass effect and is derived in a straight forward, quantitative manner.

The matrices are a function of the properties of two cylinders with the fluid in the cylindrical annulus, specifically, inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the reactor vessel and the core barrel allows inclusion of radii variations along their heights and approximates the effects of beam mode deformation. These mass matrices were inserted between the selected nodes on the core barrel, thermal shield, and the reactor vessel. The seismic evaluations are performed by including the effects of simultaneous application of time history accelerations in three orthogonal directions. The WECAN computer code is also used to obtain the response for the RPV system under seismic excitations.

3.9.3.3 <u>Results and Acceptance Criteria</u>

The reactor internals behave as a highly nonlinear system during horizontal and vertical oscillations of the LOCA forces. The nonlinearities are due to the coulomb friction at the sliding surfaces and due to gaps between components causing discontinuities in force transmission.

The frequency response is consequently a function not only of the exciting frequencies in the system but also of the amplitide. Different break conditions excite different frequencies in the system. This situation can be seen clearly when the response under LOCA forces is compared with the seismic response. Under seismic excitations, the system response is not as nonlinear as LOCA response because various gaps do not close during the seismic excitations.
The results of the nonlinear LOCA and seismic dynamic analysis include the transient displacements and impact loads for various elements of the mathematical model. These displacements and impact loads and the linear component loads (forces and moments) are then used for detailed component evaluations to assess the structural adequacy of the reactor vessel, reactor internals, and the fuel.

A. Structural Adequacy of Reactor Internals Components

The Farley reactor internal components are not ASME Code components. This is due to the fact that subsection NG of the ASME Boiler and Pressure Code edition applicable to Farley reactor internals did not include design criteria for the reactor internals since its design preceded subsection NG of the ASME Code. However, these components were originally designed to meet the intent of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with addenda through the Winter, 1971. As mentioned earlier, with the acceptance of Leak-Before-Break by USNRC, Reference(3)(4)(5) of Chapter 3.6, the dynamic effects of the main reactor coolant loop piping no longer have to be considered in the design basis analysis. Only the dynamic effects of the next most limiting breaks of the auxiliary lines (accumulator line and pressurizer surge or RHR lines) are considered. Consequently, the components experience considerably less loads and deformations than those from the main loop breaks which were considered in the original design of the reactor internals.

B. Allowable Deflection and Stability Criteria

The criteria for acceptability with regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be ensured. This implies that the deformation of reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the reactor internals are concerned principally with the maximum allowable deflections and stability of the components.
For faulted conditions, deflections of critical reactor internal components are limited to the values given in table 3.9-2. In a hypothesized vertical displacement of internals, energy absorbing devices limit the displacement to 1.25 inches by contacting the vessel bottom head.

<u>Core Barrel Response Under Transverse Excitations</u> - In general, there are two possible modes of dynamic response of the core barrel during LOCA conditions: a) during a cold leg break, the inside pressure of the core barrel is much higher than the outside pressures, this subjecting the core barrel to outward deflections, and b) during hot leg break, the pressure outside the core barrel is greater than the inside pressure thereby subjecting the core barrel to compressive loading. Therefore, this condition requires the dynamic stability check of the core barrel during hot leg break.

(1) To ensure shutdown and cooldown of the core during cold leg blowdown, the basic requirement is a limitation on the outward deflection of the barrel at the locations of the inlet nozzles connected to unbroken lines. A large outward deflection of the upper barrel in front of the inlet nozzles,

accompanied with permanent strains, could close the inlet area and restrict the cooling water coming from the accumulators. Consequently, a permanent barrel deflection in front of the unbroken inlet nozzles larger than a certain limit, called “no loss of function” limit, could impair the efficiency of the ECCS.

(2) During the hot leg break, the rarefaction wave enters through the outlet nozzle into the upper internals region and thus depressurizes the core and then enters the down-comer annulus from the bottom exit of the core barrel. This depressurization of the annulus region subjects the core barrel to external pressures, and this condition requires a stability check of the core barrel during hot leg break. Therefore, to ensure rod insertion and to avoid disturbing the control rod cluster guide structure, the barrel should not interfere with the guide tubes.

Table 3.9-2 summarizes the allowable and no loss of function displacement limits of the core barrel for both the cold leg and hot leg breaks postulated in the main line loop piping. With the acceptance of LBB, the reactor internal components such as core barrel will experience much less loads and deformations than those obtained from main loop piping.

Control Rod Cluster Guide Tubes - The deflection limits for the guide tubes (to be consistent with conditions under which the ability to trip has been tested), and for fuel assembly thimbles cross-section distortion (to avoid interference between the control rods and the guides) are given in table 3.9-2.

Upper Package - The local vertical deformation of the upper core plate, where a guide tube is located, shall be below 0.100 inch. This deformation will cause the plate to contact the guide tube since the clearance between the plate and the guide tube is 0.100 inch. This limit will prevent the guide tubes from undergoing compression. For a plate local deformation of 0.150 inch, the guide tube will be compressed and deformed transversely to the upper limit previously established. Consequently, the value of 0.150 inch is adopted as the “no loss of function” local deformation limit with an allowable limit of 0.100 inch. The deformation limits are given in table 3.9-2.

3.9.3.4 Method of Analysis

The internals structures are analyzed for loads corresponding to normal, upset, emergency, and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the stress analysis problem is large, requiring many different techniques and methods, both static and dynamic. The more important and relevant methods are presented in subsections 3.9.1 and 3.9.3.2.

3.9.3.5 Evaluation of Reactor Internals for Accumulator Line Cold Leg and Pressurizer Surge Line Hot Leg Breaks

This section contains an evaluation of the effects of a 90.75 in^2 accumulator line cold leg safe end break and a 103.87 in^2 pressurizer surge line hot leg safe end break on the reactor internals. Both breaks are assumed to have a break opening time of 1 millisecond.

The main operational requirement to be met is that the plant be shutdown and cooled down in an orderly fashion so that the fuel cladding temperature is kept within the specified limits. This implies that the deformation of the reactor internals must be kept sufficiently small to allow core cooling and assure effectiveness of the emergency core cooling system. A detailed description of LOCA methodology and the acceptance criteria for the components is given in subsections 3.9.3.2 and 3.9.3.3. Use of LBB methodology was approved for the pressurizer surge line (Reference 25).

3.9.3.6 Baffle-Former Bolt Replacement Analysis

In order to satisfy the concern of possible degradation in the reactor vessel baffle-former bolts due to long term irradiation, a selected number of bolts have been replaced in Units 1 and 2. To justify replacing a selected number of these bolts, an analysis was performed by Westinghouse to determine the acceptability of a bolt replacement pattern that would require a limited number of bolts to be replaced, while maintaining the functionality of the reactor vessel during normal and accident conditions.

This new reactor vessel analysis used for the baffle-former bolt replacement utilized sophisticated tools such as the computer application code MULTIFLEX Version 3.0. This code utilizes a detailed network to represent the vessel downcomer, and allow for vessel motion and for non-linear boundary conditions at the vessel and downcomer junctions at the radial keys and upper core barrel flange.

While MULTIFLEX Version 3.0 was used for the single-phase blowdown portion of the transient (first five hundred milliseconds), WCOBRA/TRAC was used for the two-phase portion. Loads from the two-phase portion were derived and compared to the loads obtained from the single-phase portion of the transient. Loads from the single-phase portion were determined to be the limiting faulted event conditions for this analysis.

The ANSYS computer program was used to develop a finite element model of the baffle-former region, which was used in LOCA, seismic, thermal growth, and flow induced vibration analyses. The modeled baffle plates and former plates were simulated by elastic plate elements, the bolts were simulated by pipe elements, and baffle-former, barrel-former, and baffle-fuel nozzle interfaces were simulated by gap elements.

A. Criteria

These analysis programs were used to calculate loads on the fuel assemblies during LOCA and seismic conditions (singularly and combined), and then evaluated to the following acceptance criteria:

1. Fuel rods must remain intact such that fuel pellets are not allowed to escape where they could achieve a configuration in which a core coolable geometry cannot be demonstrated.

2. Control rod guide tubes must not be deformed to the point at which control rod insertion cannot be demonstrated where it is credited in accident analysis consequences.

3. Fuel grid loads must be below allowable grid crush strength limits.

In addition to the above criteria, core bypass flow, fuel rod stability (momentum flux), high cycle fatigue, low cycle fatigue, and structure stress limits were factors that were also examined in the analysis.

B. Break Opening Time

For the project of replacing baffle-former bolts, the NRC-approved Westinghouse methodology to invoke the leak-before-break (LBB) concept, which allowed for a break opening time greater than 1 millisecond, was utilized. This methodology was based on the results of break opening experiments, calculations of break openings by Westinghouse and others, engineering practices of domestic and foreign nuclear suppliers, conservatism inherent within the computer analysis software, and regulatory considerations.

C. Acceptability

The final configuration of the baffle-former bolts replaced within the reactor was analyzed as being acceptable for the following reasons:

- The normalized fuel grid impact loads were found acceptable for both peripheral and interior fuel assemblies.

- There is adequate stress margin in maintaining the guide thimble structural integrity.

- Control rod insertability was maintained.

- Fuel rod integrity was maintained, and fuel rod fragmentation will not occur.

- Low-cycle fatigue for the design lifetime of the replaced baffle-former bolts was less severe than the fatigue of the original bolts.

- Momentum flux margin of safety was found to be acceptable with the installed baffle-former bolt pattern.

- The design core bypass flow can be maintained.

- High-cycle fatigue stresses were found to be below the bolt material endurance limits.

- Alternating stresses due to flow induced vibration were determined to be below bolt material allowables.

3.9.3.7 Heating, Ventilation, and Air-Conditioning (HVAC) Equipment

Table 3.9-3 presents a list of safety related heating, ventilation, and air-conditioning (HVAC) equipment, and the applicable standards and codes to which they are designed. This table also presents report numbers containing test results for the equipment.

3.9.4 OPERABILITY ASSURANCE

Equipment for the Farley Nuclear Plant was designed to comply with the intent of Regulatory Guide 1.48; i.e., it was designed/analyzed to ensure structural integrity and operability. However, the load combinations and stress limits that were used reflect NRC requirements that were in effect when the construction permit for this plant was issued and when the components were purchased and subsequently designed. Furthermore, the codes and procedures which were available when the components were purchased are based on conservative design requirements rather than detailed stress analysis and actual testing. These codes and procedures have been used by the nuclear industry for the design of components that are installed in plants that are presently operating.

3.9.4.1 ASME Code Class Valves

A tabulation of all active valves in the reactor pressure boundary whose operation is relied upon either to assure safe plant shutdown or to mitigate the consequences of a transient or accident is provided in table 5.2-8.

The requirements of the (draft) ASME Code for Pumps and Valves were adhered to in the design of Active Code Class 1 valves. For faulted conditions, stress intensities in the valves and extended structures were limited to 1.0 S_m for general membrane and 1.5 S_m for general membrane plus bending. These limits ensure that the valve stresses will remain within elastic limits and that no plastic deformation will occur.

The requirements of Section III of the ASME Boiler and Pressure Vessel Code were adhered to in the design of Code Class 1 manually operated globe valves and check valves, 2 in. in size or less.

Class 2 and 3 active valves were designed to the requirements of ANSI B16.5 Code. In addition, an analysis of the extended structure was performed with loads of 3.0 g in the horizontal and vertical directions, simultaneously for valves specified by Bechtel and Southern Company Services Specifications. For this analysis, stresses were limited to values that restrict the maximum stress in the extended structure. Deflections of the extended structure will thus be small and operability of the valves will not be impaired.

Prior to installation, the valves are subjected to shell hydrostatic tests, seat leakage tests, and functional tests. After installation, the valves undergo cold hydrostatic tests, hot functional tests to verify operation, and those under the Farley ISI program undergo periodic inservice inspection and operation to ensure the continued ability of the valves to operate.

3.9.4.2 ASME Code Class Pumps

Active pumps were designed in accordance with the ASME Code for Pumps and Valves or the ASME Boiler and Pressure Vessel Code for Nuclear Power Plants, depending on which code was in effect at the time the purchase order was issued. The stress levels in the pumps did not exceed those allowed by the applicable code. Forces resulting from seismic accelerations in the horizontal and vertical directions are included in the analysis of the pumps and their supports. The supports were designed to have natural frequencies in excess of 20 Hz.

The pumps are subjected to a series of tests prior to installation and after installation in the plant. In-shop tests include hydrostatic tests to 150 percent of the design pressure, seal leakage tests, net positive suction head (NPSH) tests to qualify the pumps for the minimum available NPSH, and functional performance tests. For the NPSH and functional performance tests, the pumps are placed in a test loop and subjected to operating conditions. After installation, the pumps undergo cold hydrostatic tests, hot functional tests to verify operation, and periodic inservice inspection and operation.

The above design procedures and qualification tests are, therefore, adequate to ensure the structural integrity and operability of the pumps and valves for this plant.

3.9.4.3 Qualification of Vital Appurtenances

The following typical appurtenances that were identified to be vital to the operation of active pumps and valves were qualified for operation during a seismic event by dynamic testing procedures as described below.

<u>Seismic Qualification Test of National ACME Company Snap-Lock Electric Switch No. D 2400X-2</u>

A seismic qualification test program of National ACME Snap-Lock electric switch No. D 2400X-2 was conducted by Fisher Controls Co. and reported in document No. 1529 dated 11/2/72. Testing was conducted with the switch assembly fastened to a metal plate which in turn was attached to a shaker table. All tests were conducted with the switch in an operating condition. The following is a summary of the test procedure and results:

Test Procedure

A. Conduct a continuous frequency sweep for each of the three axes, from 5 to 60 Hz at an acceleration level of 1.0g in no less than 31 seconds.

B. If the resonant frequency is less than 33 Hz, conduct a 4g 1-min dwell at the resonant frequency and at 10 and 33 Hz.

C. If the resonant frequency is greater than 33 Hz, conduct a 4g 1-min dwell at 10, 17, 25 and 33 Hz and at the resonant frequency if it is less than 60 Hz.

Test Results

The Snap-Lock electric switch performed satisfactorily with no malfunctions noted and meets or exceeds the specifications outlined in the test procedure.

Seismic Qualification of Valve Motor Operators

A seismic qualification test program of valve motor operators, manufactured by Limitorque Corporation, was conducted by Lockheed Electronics Company, Inc., Environmental Laboratory and reported in Report No. 2785-3-4785 dated 2/6/73; Report No. 2786-4786, Issue 2, dated 9/5/72; Report No. 2773C-4773, dated 5/3/72; and Report No. 2785-4-4785, dated 2/1/73. The test specimens were electrically monitored and operated during testing. The test procedure consisted of subjecting the specimen to the vibration test referenced in Limitorque Co. Purchase Order No. 600374, dated 6/2/72, and is summarized as follows for each of the three orthogonal axes:

a. Two exploratory scans were performed over the frequency range of 5 to 60 Hz at the amplitudes specified in Table 1.

TABLE 1

Freq. Range (Hz)	Vib. Amplitude (In., Double Amplitude)
5 - 33	.020 ± .004
34 - 50	.006 + .000 - .002
51 - 60	.004 + .000 - .002

b. Two 1-min dwells were performed at the resonant frequency at a nominal vibration input of 3 to 5.8 g's. The first minute of vibration was followed by one minute of rest.

Test results for the SMB-0/H3BC, SMB-0-25/H3BC, SMB-000-2/HOBC and SMB-3/H5BC valve operators indicated that the vibration test was completed with no visible evidence of any external damage or performance degradation. There were no resonances detected during the vibration test except for a resonance at 44 Hz in the Y axis, 46 Hz in the Z axis, and 39 Hz in the X axis for valve operator No. SMB-3/H5BC.

A review of the test data indicates that the valve motor operator performed satisfactorily when subjected to the dynamic environment.

Seismic Qualification of Solenoid Valves

A seismic qualification test program for the solenoid valves used on Westinghouse supplied air operated valves has been completed. The components tested were ASCO valve models 8300C58RV, 8300B64RU, and 831654. The test dynamic input forces, frequency limits etc., are discussed in reference 13.

Instrumentation and Control Panel (Series 7300, Westinghouse) – Balance of Plant

The test involved subjecting a 7300 Series nuclear power plant control system to seismic conditions for qualification and evaluation of performance. The seismic test was run in three parts for horizontal (front to back and side to side) and vertical conditions. These parts consisted of: Part 1 - Low Present Seismic, Part 2 - High Present Seismic, and Part 3 - High Future Seismic.

The control system tested included at least one of each type of printed circuit card used in all the various protection and safeguard actuation channels.

The equipment performed satisfactorily with no malfunctions noted and meets or exceeds the specifications outlined in the detailed procedures.

Class 2 and 3 Air-Operated Control Valves

The valve with diaphragm actuator was analyzed in accordance with the customer's specifications, following acceptable analytical methods and allowable stress limits as set forth in the appropriate design standards and codes.

The stress developed by gravity loads, operating loads, applicable temperature and pressure, combined with simultaneously applied horizontal and vertical seismic loads, shall not cause loss of function of this valve.

The analysis demonstrated that the design adequately satisfies the requirements of all the specifications.

Valve Motor Operators

Valve motor operators for applicant specified valves are seismically tested at g-loadings higher than those specified in the valve design specification. These tests demonstrate that the operators experience no physical damage as a result of the postulated seismic event, and that the activating mechanisms undergo no change in position during the test and remain operable after the test.

REFERENCES

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TABLE 3.9-1

DESIGN CRITERIA FOR ASME CLASS 2 AND 3 COMPONENTS

Loads	Vessel/ Tanks (Note 1)	Pumps (Note 1)		Valves		Piping
Pressure + Deadweight + Thermal (nozzle loads only)	ASME III/ASME VIII	ASME III/Performance Testing in accordance with standards of the Hydraulic Institute Procedures		ASME III/ANSI B 16.5		ASME III
Pressure + Deadweight + Thermal (nozzle loads only) + Transients (Note 2)	ASME III/ASME VIII	ASME III/Performance Testing in accordance with standards of the Hydraulic Institute Procedures		ASME III/ANSI B 16.5		ASME III
Pressure + Deadweight + SSE + Dynamic Effects (where applicable) (Note 6)	ASME III/ASME VIII (Note 3)	Structural Assured by integrity of connecting piping	Functional Rigid (f_n>20) within working conditions by dynamic analysis	Structural Assured by integrity of connecting piping	Functional Rigid (f_n>20) within working conditions by dynamic analysis	(Note 4)
		(Note 3)		(Note 5)		

1. Allowable nozzle loads are contained in the equipment specifications or specified by the vendor for pumps and tanks. The piping is designed so that the loads generated on the components nozzles are no greater than the allowable loads specified for that component. The allowables and stress calculations for the components are reviewed by the designer.

2. The transients considered in the piping analyses are:
   (1) a relief valve-closed system (transient).
   (2) a fast valve closure.
   (3) a relief valve-open system (sustained) +1/2 SSE.

   (The loads in the piping generated by the individual transients are considered in the design of the components, where applicable).

3. The design limits for tanks are specified by the vendors. These design limits are based on having no gross deformation of the components.

4. The design limits for piping are based on having no gross deformation of the piping.

5. ASME Class 2 and 3 valves are designed such that the section modules of the valves is greater than that of the pipe connected to the valve. For valves that do not meet the selection modules criteria, a stress analysis will be performed to verify adequacy.

6. The loads in the piping generated by the dynamic effects are considered in the design of the components.

TABLE 3.9-2

MAXIMUM DEFLECTIONS SPECIFIED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable[1] Limit(in.)	No Loss-of Function Limit(in.)
Upper Barrel, Expansion/Compression (to ensure sufficient inlet flow area/and to prevent the barrel from touching any guide tube to avoid disturbing the rod cluster control guide structure)[2,3]		
Radial Inward	4.1	8.2
Radial Outward	1.0	1.0
Upper Package, Axial Deflection (to maintain the control rod guide structure geometry)[2,3]	0.1	0.15
Rod Cluster Control Guide Tube Deflection As a Beam (to be consistent with conditions under which ability to trip has been tested)[3]	1.0	1.75
Fuel Assembly Thimbles Cross-Section Distortion (to avoid interference between the control rods and the guides)[3]	0.036	0.072

Notes:

1. The allowable limit deflection values given above correspond to stress levels for internals structure well below the limiting criteria given by the collapse curves in WCAP-5890 (Reference 20). Consequently, for the internals, the geometric limitations established to ensure safe shutdown capability are more restrictive than those given by the failure stress criteria.

2. See Reference 21.

3. See Reference 22.

TABLE 3.9-3 (SHEET 1 OF 2)

DESIGN CRITERIA FOR COMPONENTS NOT COVERED BY ASME CODE

Systems - Components	Controlling Standards and/or Codes	Test Report Number (type)
Auxiliary Building Ventilation System		
ES air cooling units:		
Heat exchangers	ARI Standard 410-64	CVI Seismic Analysis Report (Dynamic/Seismic Analysis)
Fan	AMCA Test Code 300-67, 211A-65	CVI Seismic Analysis Report (Dynamic/Seismic Analysis)
Penetration Room Filtration System		
Fans	AMCA Test Code 300-67, 211A-65	AAF Report on Seismic Analysis PEP-497 (Dynamic/Seismic Analysis)
Filters (HEPA and charcoal)	AACC CS-IT (HEPA), ORNL-NSIC-65 (charcoal)	AAF Report on Seismic Analysis PEP-497 (Dynamic/Seismic Analysis)
Ductwork	SMACNA High Velocity Duct Construction, 2nd Edition, 1969.	Stress Analysis
Containment Cooling System		
Fans	AMCA Test Code 300-67, 211A-65	AAF Report on Seismic Analysis, PEP-495 (Seismic-Dynamic Analysis)
Heat Exchangers	ARI Standard 410-64	AAF Report on Seismic Analysis, PEP-495 (Seismic-Dynamic Analysis)
Fusible linked plate	UL Standard of Safety UL555-1970, UL33-1968	AAF Report on Seismic Analysis, PEP-495 (Seismic - Dynamic Analysis)
Control Room Ventilation System		
Fans	AMCA Test Code 300-67, 211A-65	Joy Certification or Dynamic Analysis
Filters (HEPA and charcoal)	AACA, CS-IT (HEPA), ORNL, NSIC - 65 (charcoal)	AAF Report on Seismic Analysis PEP-497 (Dynamic - Seismic Analysis)

TABLE 3.9-3 (SHEET 2 OF 2)

Systems - Components	Controlling Standards and/or Codes	Test Report Number (type)
Air conditioning unit	ANSI B9-1971, Safety Code for Mechanical Refrigeration	AAF Report on Seismic Analysis PEP 648 (Seismic Dynamic Analysis)
Ductwork	SMACNA Low Velocity Duct Construction, 2nd Edition, 1969	Stress Analysis

TABLE 3.9-4

COMPARISON OF BEST ESTIMATE AND DESIGN VALUES OF PEAK SEISMIC ACCELERATIONS (g)

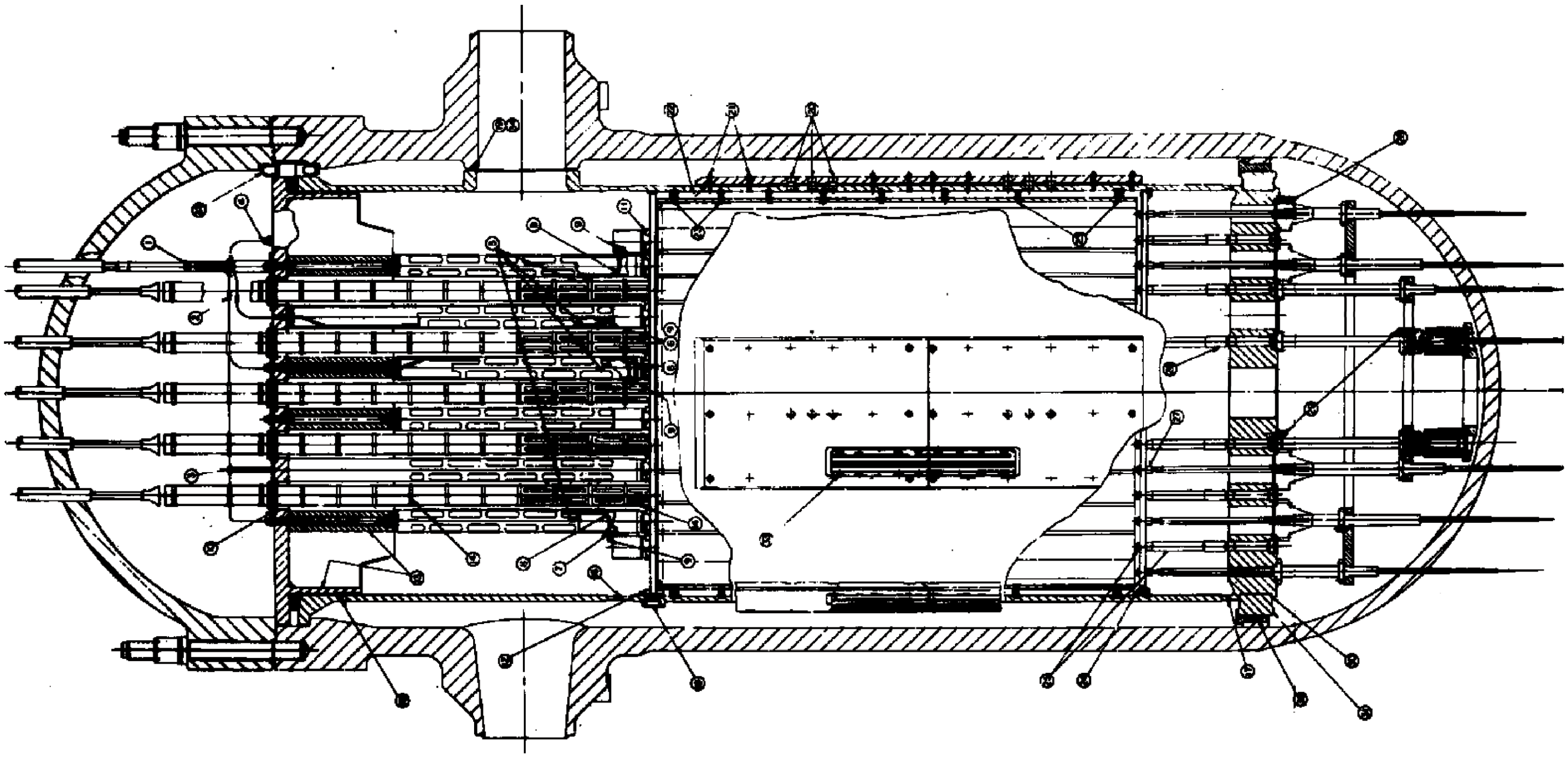
Earthquake Type	Maximum Acceleration (g)	
	Best Estimate Value	Design Value
Operational basis earthquake (OBE)	1.68	3.20
Safe shutdown earthquake (SSE)	2.20	4.20

TABLE 3.9-5

SUMMARY OF STRESS AND
MARGIN OF SAFETY TO CODE ALLOWABLES

Item	Maximum Stress ($lb/in.^2$)	Allowable Stress ($lb/in.^2$)	Margin of Safety
Stanchion	5,090.0	24,795.0	3.87
Stanchion bolts	44,412.0	45,000.0	0.013
Stanchion flange	12,397.5	24,795.0	1.0
Conduit tube cantilevered	1,590.0	24,795.0	14.70
Conduit tube attached to stanchion	1,375.0	24,795.0	17.0
Conduit tube support clip welds	104.0	9,918.0	Large

NOTE: Margin of safety = $\frac{\text{allowable stress}}{\text{maximum stress}} - 1.0$



	FEATURES TO BE EXAMINED
1	THE THERMOCOUPLE CONDUIT CLAMPS INSIDE THE THERMOCOUPLE COLUMN.
2	CONDUIT SWAGELOK FITTINGS, THEIR BANDINGS, AND THE TAB TYPE LOCKS
3	CLAMP ARRANGEMENTS AT THE MOUNTING BRACKET LOCATIONS.
4	PLUG TO CONDUIT WELD AT THE FOUR SUPPORT COLUMNS ADJACENT TO THE THERMOCOUPLE COLUMNS
5	ACCESSIBLE ANGLE CONDUIT CLAMPS INSIDE THE UPPER SUPPORT COLUMNS.
6	ACCESSIBLE WELD JOINTS AT THE THERMOCOUPLE STOP FOR THE SELF INSTRUMENTED COLUMNS.
7	WELD JOINTS ON ACCESSIBLE SUPPORT COLUMN AND MIXING DEVICE GUSSETS (THE THERMOCOUPLE SUPPORT HARDWARE)
8	INTEGRITY OF EXPOSED PORTION OF THE THERMOCOUPLE CONDUIT RUNS, AT ACCESSIBLE LOCATIONS (INSIDE SUPPORT COLUMNS LOWER END.)
9	[illegible] OF THE ACCESSIBLE PROTRUDING THERMOCOUPLE TIPS
10	THERMOCOUPLE COLUMN AND GUIDE TUBE SCREW LOCKING DEVICES.
11	ACCESSIBLE SUPPORT COLUMN, MIXING DEVICE, ORIFICE PLATE, AND CORE PLATE INSERT SCREW LOCKING DEVICES.
12	UPPER CORE PLATE INSERTS.
13	DEEP BEAM WELDS AT THE SKIRT AND AT THE OUTER HOLLOW ROUNDS.
14	ACCESSIBLE GUIDE TUBE WELDS.
15	UPPER BARREL TO FLANGE GIRTH WELD.
16	UPPER BARREL TO LOWER BARREL GIRTH WELD.
17	LOWER BARREL TO CORE SUPPORT GIRTH WELD.
18	UPPER CORE PLATE ALIGNING PIN WELDS AND BEARING SURFACES
19	OUTLET NOZZLE INTERFACE SURFACE CONDITION.
20	NEUTRON SHIELD PANEL DOWEL PIN COVER PLATE WELDS.
21	NEUTRON SHIELD PANEL SCREW LOCKING DEVICES
22	INTERFACE SURFACES AT THE SPACER PADS ALONG THE TOP AND BOTTOM ENDS OF THE NEUTRON PANELS
23	BAFFLE ASSEMBLY SCREW LOCKING ARRANGEMENTS AT THE TWO TOP AND THE TWO BOTTOM FORMER ELEVATIONS.
24	LOWER CORE PLATE TO CORE BARREL FLANGE SCREW LOCKING DEVICES ACCESSIBLE AT THE 0 90, 180, AND 270 AXES.
25	CORE SUPPORT COLUMNS AND THEIR SCREW LOCKING DEVICES
26	CORE SUPPORT COLUMN ADJUSTING SLEEVES.
27	ACCESSIBLE (2) INSTRUMENTATION GUIDE COLUMN LOCKING COLLARS NEAREST THE MANWAY.
28	LOCKING DEVICES AND CONTACT OF THE CRUCIFORM SHAPED BOTTOM INSTRUMENTATION GUIDE COLUMNS WHERE ATTACHED TO THE CORE SUPPORT AND TIE PLATES
29	LOCKING DEVICES OF THE SECONDARY CORE SUPPORT BUTT COLUMNS AT THE CORE SUPPORT AND AT THE TIE PLATE.
30	RADIAL SUPPORT KEY WELDS.
31	RADIAL SUPPORT KEY LOCKING ARRANGEMENTS AND BEARING SURFACES
32	HEAD AND VESSEL ALIGNING PIN SCREW LOCKING DEVICES AND BEARING SURFACES
33	IRRADIATION SPECIMEN GUIDE SCREW LOCKING DEVICES AND DOWEL PINS
34	VESSEL NOZZLE INTERFACE SURFACE CONDITION.
35	VESSEL CLEVIS LOCKING ARRANGEMENTS AND BEARING SURFACES

REV 21 5/08

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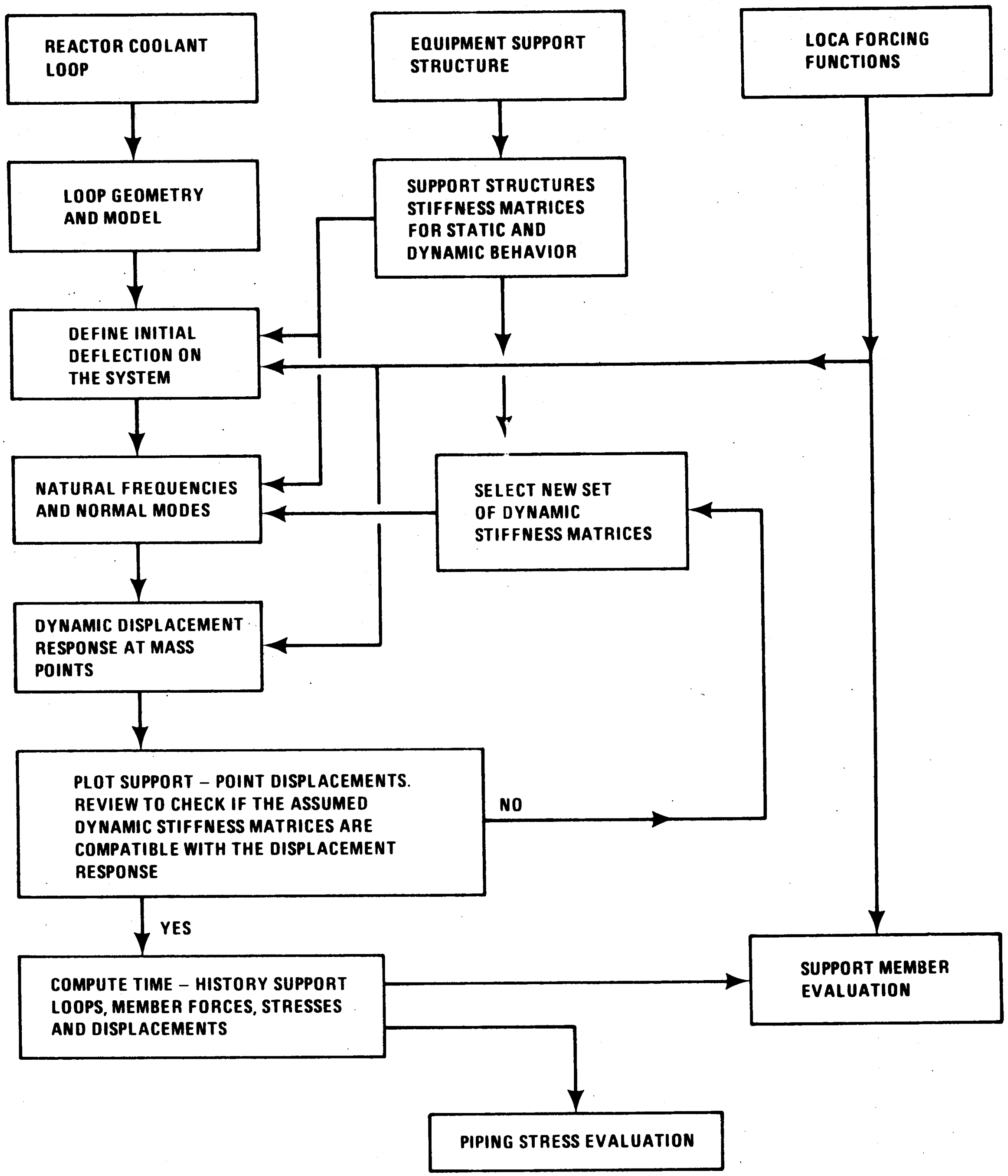
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

VIBRATION CHECKOUT
FUNCTIONAL TEST INSPECTION POINTS

FIGURE 3.9-1



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UNIT 1 AND UNIT 2

TIME-HISTORY DYMAMIC SOLUTION FOR LOCA LOADING

FIGURE 3.9-2

3.10 SEISMIC DESIGN OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Equipment and components of the reactor protection system and the engineered safety feature actuation system meet the seismic Category I requirements and are identified in section 3.2.

3.10.1 SEISMIC DESIGN CRITERIA

A. Category I Electrical Equipment

Category I electrical equipment has been designed to withstand, without exceeding normal allowable working stresses and without loss of function, the forces resulting from the 1/2 SSE caused by a horizontal ground acceleration of 0.05g and a vertical ground acceleration of 0.033g. The equipment is also designed to withstand, without exceeding 90 percent of the yield stresses, or without loss of function, the forces resulting from the safe shutdown earthquake (SSE) caused by a horizontal ground acceleration of 0.10g and a vertical ground acceleration of 0.067g.

The seismic response spectra, based on the synthesized time history spectra, have been developed for the specific equipment location and appropriate damping factors. A detailed discussion of the seismic design criteria is given in section 3.7. The electrical equipment under Category I was qualified in one of the following ways:

1. The natural frequencies of the equipment (as it would be installed in service) were determined in the horizontal and vertical directions based on a multi-degree of freedom lumped mass system. From the appropriate response spectra curve, the acceleration levels were selected corresponding to the natural frequency. Forces due to this acceleration level are used in the seismic analysis.

2. If it was not practical to calculate the natural frequency, the maximum acceleration of the spectra curves was used for seismic analysis.

3. Prototype equipment was subjected to a test demonstrating its ability to perform its intended function during and after SSE.

   When simulated seismic testing was not entirely practical, proof of performance was obtained by a combination of mathematical analysis and simulated testing.

The following items, which were part of the test reports submitted by the manufacturer, conform to the requirements of IEEE 344-1971.

1. Equipment identification.

2. Equipment specification.

3. Test facility:
   location.
   test equipment.

4. Test method.

5. Test data.

6. Results and conclusions (pertaining, in particular, to natural frequencies and maximum accelerations).

7. Signature of manufacturer's authorized representative and date.

For the analytical approach, the manufacturer was required to submit complete seismic design calculations in step-by-step form. Preference was given to actual testing. The manufacturer was required to furnish documentation justifying his selection of the analytical method over simulated testing. These requirements are in accordance with the stipulations of IEEE 344-1971.

Cable tray supports are designed using the appropriate instructure response spectra. The calculated stresses from dead load, live load, and earthquake loads are less than 50 percent of yield stress for the 1/2 SSE and 90 percent of yield stress for the SSE.

The location and performance requirements of class IE switchgear, motor control centers, and distribution panels are such that post accident conditions do not impose any additional stresses over and above those experienced due to a safe shutdown earthquake (SSE). The capability of the equipment to withstand seismic disturbances, established under nonaccident conditions, is considered adequate to meet the requirements during post accident operation.

B. <u>Category I Instrumentation and Control Equipment</u>

Equipment specifications for Category I instrumentation control equipment required that equipment be designed to withstand without loss of function the forces resulting from the 1/2 safe shutdown earthquake and the safe shutdown earthquake. The following procedures were used.

1. Equivalent static acceleration factors for the horizontal and vertical directions were provided to the equipment manufacturers.

2. The seismic response spectra and the appropriate damping factor, based on the synthesized time history spectra, were provided for the specific equipment location.

The equipment vendor was given two ways by which the equipment could be qualified: by dynamic analysis and/or by testing. The manufacturer was permitted to use:

1. Test reports of the particular component(s).

2. Performance data of equipment, with applicable supporting data, which under specified conditions has been subjected to equal or greater dynamic loads.

3. Analysis.

The choice of the method was based on the practicability of either method (test or analysis) for the size, type, shape, and complexity of the instruments or equipment, and reliability of results. Table 3.7-4 indicates the type of procedure, qualification method (test or analysis), as well as the applicable stress or information criteria.

Following submission of the results of the test or analysis, the methods, procedures, and results were examined for compliance with the specification requirements. Test and analytical procedures, as well as submitted reports, conform to the requirements of IEEE-344-1971.

Component Testing:

Testing of components, such as relays, was performed as part of the primary equipment being tested. The relays were tested in the energized state and the output contacts were monitored for continuity. A change in continuity would be indicative of a malfunction. Where a representative component was qualified by previous testing, the test results were reviewed and, if found acceptable, a certificate of conformance to the FNP seismic specification from the vendor was considered adequate in lieu of a repeat test.

C. Category I Equipment and Components

Seismic qualification for Category I equipment and components supplied by the NSSS vendor was originally described in WCAP-7817, "Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)" and its supplements 1, 2, 3, and 4. The AEC, in a letter dated January 12, 1973, indicated its acceptance of this Westinghouse report. However, in a letter dated December 23, 1974, additional information was requested.

In order to address these additional concerns, Westinghouse conducted a supplemental qualification program and submitted the results to the NRC. The NRC conducted a seismic audit of Westinghouse to evaluate the results of the supplemental program as well as on-site item by item inspection of equipment at the Salem, Farley, and Sequoyah plants.

For equipment to be tested after May 1974, and for equipment to be installed in plants having a construction permit docketed after October 1972, Westinghouse has committed to conduct seismic qualification testing in conformance with IEEE 344-1975. However, for equipment tested prior to May 1974, the following conclusion was drawn by the NRC in Section IV "Conclusion and Regulatory Position of Report on Seismic Audit of Westinghouse Electrical Equipment (TAR's 3678-1, 3683-1, 0706, 0921-1, 0788-2, 1111-2, and 3000-2)" dated August 26, 1976.

> "The Mechanical Engineering Branch, Division of Systems Safety has completed the seismic audit of Westinghouse electrical equipment tested prior to May 1974. Based on our evaluation of topical reports, inspection of equipment on the plant site, numerous meeting discussions, laboratory

> visits, and our evaluation of confirmatory retesting for equipment in question, we conclude that adequate assurance is achieved for this equipment to sustain seismic excitations to their designated SSE levels."

These test and analytical procedures, as well as the submitted reports, conform to the requirements of IEEE 344-1971. The seismic design criteria applicable to NSSS scope equipment are addressed in the Westinghouse generic qualification program.

3.10.2 SEISMIC ANALYSES, TESTING PROCEDURES AND RESTRAINT MEASURES

Table 3.7-4 indicates which of the methods given in subsection 3.10.1 have been used for seismic qualification of the equipment. The procedures for seismic qualification, either by analysis or by testing, are in conformance with the requirements of IEEE 344-1971.

The seismic design criteria discussed in subsection 3.10.1 form a part of all specifications for Category I equipment. Certification was obtained from each manufacturer to ensure that his equipment will perform without loss of function in accordance with the stipulations of the specifications.

For seismic qualification by analysis, the certification requires that the calculations have been checked by an engineer knowledgeable in the design of such equipment.

The specification includes applicable floor response spectra where the equipment is located, and the values indicated in the curves are used for seismic qualification of that equipment. The floor response spectra take into consideration the loading amplification of floors.

Equipment hold down details, specifically size and spacing of weld or bolt, are obtained from the manufacturer for designing the foundation to be compatible with the seismic withstanding capability of the equipment.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section provides information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features, the reactor protection systems, and other safety-related systems are designed to ensure acceptable performance during normal and design basis accident (DBA) environmental conditions.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Safety-related equipment which is required to function during and subsequent to a DBA, is identified in section 3.2 of the FSAR. Active pumps and valves are discussed in section 3.9 of the FSAR.

The original specifications for safety-related electrical equipment which is subject to a post DBA harsh environment and required to function during and subsequent to a DBA required qualification to IEEE 323-1971. Subsequently, the Farley Nuclear Plant Environmental Qualification (EQ) Program was implemented to comply with the requirements of NRC Inspection and Enforcement Bulletin (IEB) 79-01B, NUREG-0588, Revision 1, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, and 10 CFR 50.49. Based on the dates of the Farley plant operating licenses, Unit 1 was required to comply with the requirements of IEB 79-01B, which provides the NRC Division of Operating Reactors (DOR) Guidelines, and Unit 2 was required to comply with the requirements of NUREG-0588, Category II. The requirements set forth under these programs supplement the requirements of IEEE 323-1971. After implementation of these programs, 10 CFR 50.49 was issued and mandated environmental qualification requirements for safety related electrical equipment. Regulatory Guide 1.89, Revision 1, followed and established IEEE 323-1974 as an acceptable standard to comply with the requirements of 10 CFR 50.49. The provisions of 10 CFR 50.49 waive the need to requalify components previously qualified under the DOR Guidelines or NUREG-0588 unless the components are replaced. The replacement components must comply with the provisions of 10 CFR 50.49 unless there are sound reasons to the contrary. These reasons, when required, will be documented. Accordingly, the EQ program implements the requirements of 10 CFR 50.49 as documented in the EQ master lists and the associated EQ packages. The EQ packages document which version of the IEEE-323 standard was used for the qualification.

Normal operating environmental conditions are defined as conditions existing during routine plant operations. These environmental conditions, as listed in table 3.11-1, represent the normal, maximum, and minimum conditions expected during routine plant operations.

Accident environmental conditions are defined as those deviating significantly from the normal operating environmental conditions as a result of a DBA. These conditions are specified in table 3.11-1 for the postulated accident duration of 30 days. Compatibility of equipment with the specified environmental conditions is provided to fulfill the following design criteria:

A. For normal operation, systems and components required to mitigate the consequences of a DBA or to provide for safe shutdown are designed to remain functional after exposure to the environmental conditions listed in table 3.11-1.

Where possible, all safety-related systems and components are designed to withstand the maximum expected 40-year[a] integrated radiation dose at their respective locations within the plant. If it cannot be assured that equipment is designed for the 40-year[a] dose, a replacement maintenance program for that equipment is established. The replacement maintenance program ensures operational integrity of the equipment throughout the life of the plant.

B. In addition to the normal operation environmental requirements given in A. above, systems and components required to mitigate the consequences of a DBA or to provide for safe shutdown of the reactor are designed to remain functional after exposure to the following environmental conditions. Qualification time is based on the operating duration following a DBA.

1. Such components inside the containment are designed for the temperature, pressure, humidity, and chemical environment inside the containment after a design basis LOCA or main steam line break accident (MSLB).

2. Such components inside the containment which are required after a LOCA are designed for the post-LOCA radiation dose.

3. Such components outside the containment which are required to mitigate the consequences of a design basis LOCA are designed for the expected integrated accident radiation dose at the equipment location.

4. Such components outside the containment are designed for the temperature, pressure, and humidity environmental conditions resulting from a postulated high energy line break (HELB) in areas where such components are located.

---

a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. The EQ program is credited to continue to manage aging effects associated with EQ equipment for the period of extended operation (see chapter 18, subsections 18.3.1 and 18.4.4). Applicable EQ evaluations based on a 40-year design life were evaluated as time-limited aging analyses (TLAAs) for license renewal and will be revised as necessary to reflect the 60-year plant operating life before the units enter the period of extended operation.

3.11.2 QUALIFICATION TESTS AND ANALYSES

Qualification is based on simulated environmental testing where feasible. If qualification test data was inadequate, and if sufficiently reliable data and proven analytical methods were available, environmental adequacy was based on analysis.

Testing consists of simulation of actual physical conditions on an actual component or prototype, analyses, or a combination of tests and analyses, as applicable. Qualification testing is performed under conditions of temperature, pressure, humidity, chemistry, and radiation in excess of the design basis conditions. The testing period is sufficient to ensure the capability to function during and for the required interval after an accident (30 days).

3.11.2.1 Equipment Inside Containment

Equipment listed in table 3.2-1 is designed for 40 years[a] of operation in the most severe temperature, pressure, humidity, and radiation environment which exists at the equipment location during normal operation. In some cases, a 40-year[a] life under such conditions is not within the state-of-the-art; therefore, a replacement program is established to ensure continuous, reliable operation. Furthermore, the safety-related equipment listed in table 3.2-1 is designed to remain functional in the most severe temperature, pressure, humidity, radiation, and chemical environment which exists at the equipment location at the time it is required to perform after a design basis loss-of-coolant or main steam line break accident. Such equipment required after a design basis LOCA is also designed for the integrated radiation exposure after the LOCA. The temperature, pressure, radiation, and humidity environment inside the containment after such accidents is presented in table 3.11-1. The containment spray characteristics are given in subsection 6.2.2.

3.11.2.2 Equipment Outside Containment

Active safety-related equipment located outside the containment normally operates in ambient temperatures up to 104°F. Normal operating radiation environments are provided in table 3.11-1. The design environmental conditions, including cumulative radiation exposure, are also given in table 3.11-1.

a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. The EQ program is credited to continue to manage aging effects associated with EQ equipment for the period of extended operation (see chapter 18, subsections 18.3.1 and 18.4.4). Applicable EQ evaluations based on a 40-year design life were evaluated as time-limited aging analyses (TLAAs) for license renewal and will be revised as necessary to reflect the 60-year plant operating life before the units enter the period of extended operation.

3.11.2.3 Equipment Supplied by Bechtel and Southern Company Services

Descriptions of the qualification tests and analyses that have been performed on the components of safety-related systems are contained in the sections indicated below:

A. Containment isolation system in paragraph 6.2.4.4.

B. Containment cooling system in paragraph 6.2.2.4.2.

C. Penetration room filtration system in paragraph 6.2.3.4.2.

D. Control room ventilation system in paragraph 9.4.1.4.

E. Auxiliary feedwater system in subsection 6.5.4.

F. Component cooling system in subsection 9.2.2.

G. Service water system in subsection 9.2.1.

H. Diesel building ventilation system in subsection 9.4.7.

In the auxiliary building ventilation systems, 11 coolers and fan units are designated as engineering safeguards. They are:

A. High head injection pump rooms (3 cooling units).

B. Low head injection pump rooms (2 cooling units).

C. Auxiliary feed pump rooms (2 cooling units).

D. Containment spray pump rooms (2 cooling units).

E. Component cooling pump rooms (2 cooling units).

The test and analysis requirements for these cooling units are the same as required for the containment heat removal system, as given in paragraph 6.2.2.4.2.

The main steam isolation valves are safety-related components. They are hydrostatically tested in the manufacturer's facilities in accordance with the applicable code. Test and inspection requirements are contained in subsection 10.3.4.

3.11.2.4 Equipment Supplied By Westinghouse

Temperature in the control room and computer room is maintained for personnel comfort between 60 and 80°F, with the exact range in each room being controlled by a procedure. Design specifications for this equipment require that no loss of protective function should result when operating in temperatures up to 120°F and humidity up to 95 percent, which may occur upon the loss of air conditioning and/or the ventilation system. Thus there is a wide margin between the design limit and the normal operating environment for the protective equipment.

The normal operating temperature for the protective equipment in the containment will be maintained below 120°F, (except that for out of core neutron detectors the normal operating temperature will be maintained below 135°F). The protective equipment is designed for continuous operation within design tolerance in this environment.

The neutron detectors will be designed for continuous operation at 135°F (the normal operating environment is always designed to be below this value) and will be capable of operation at 175°F for 8 hours. The power range detector has been tested in temperatures in excess of 175°F for a period of 16 hours with negligible decrease in insulation resistance. The insulation resistance is the governing factor for severe environments.

Type testing has been performed on safety-related equipment required to operate in the post design basis accident environment. This testing has demonstrated that Westinghouse supplied safety-related equipment has been designed to complete its protective functions in the environments in which it must operate.

A reconfirmation program of this testing as described in letter NS-CE-692, Eicheldinger to Vassallo, July 10, 1975 has been completed for Farley.

Applicable subprograms for Westinghouse supplied safety-related electrical equipment on the Farley plant are:

<u>Subprogram</u>	<u>Equipment</u>
B	Process instrumentation and control equipment, Parts 1, 3.
C	Post accident hydrogen control system.
D	Valve motor operators, Parts 1, 2, 3.
	Solenoid valves, Part 1.

The electric hydrogen recombiners used for post-accident protection have been type tested to demonstrate their compatibility of design for post accident operation. This test series is documented in <u>WCAP 7820</u>, and <u>WCAP 7709-L</u> Supplements 1-4[1] which were accepted by the NRC in a letter from Vassallo to Eicheldinger dated May 1, 1975.

The safety-related motor-operated valves which are required to operate in the design basis accident environment are protected by Class H insulation. The insulation is used regardless of the brevity of time for which the valves must operate after the design basis accident.

The environmental qualification of the safety related motor operated valves is demonstrated in reference 2 and Westinghouse submittals NS-CE-692, Eicheldinger to Vassallo, dated July 10, 1975 and NS-CE-756, Eigheldinger to Vassallo, dated August 15, 1975. The operability under severe environmental conditions of Westinghouse supplied solenoid valves is documented in NS-CE-755, Eicheldinger to Vassallo, dated August 15, 1975.

Westinghouse furnished process instrumentation and control equipment which is located inside the containment and which must function in a post DBA environment, have been identified in responses to IEB 79-01B and NUREG 0588. With the exception of the pressurizer pressure channel installed in Unit 1, which was tested as described below, instruments for each of these applications have been tested by exposing them to a steam and chemical spray environment, as described in the test references. As a result of this testing, the Unit 1 pressurizer level, steam generator level (W/R and N/R) and RCS pressure (W/R) instruments were replaced with modified instruments during the first Unit 1 refueling. Modified transmitters for these applications have been installed for Unit 2.

The pressurizer pressure instruments have also been type tested for the design basis accident environment. This test consisted of exposing a pressurizer pressure instrument similar to the ones used at Farley to saturated steam at 60 psig and 300°F for a 2-hour period and then to 20 psig and 244°F for 22 hours. The proper performance of the tested instrument was verified by monitoring its output signal and then comparing it to a reference transmitter which was outside the test chamber at room conditions. A similar instrument was exposed to an integrated dose of 7.6×10^7 rads.

The pressurizer pressure instruments were not exposed to sodium hydroxide. The instruments' protective functions of guaranteeing engineered safety features (ESF) actuation will be completed prior to containment spray initiation which is the source of NaOH. ESF actuation will occur when the containment pressure reaches the high setpoints of the containment pressure instruments, while the sprays are not initiated until the high-high setpoints of the containment pressure instruments are reached. The effect of chemical sprays on the transmitters has been shown to be negligible in the short term as reported in NS-CE 719, Eicheldinger to Vassallo, dated July 25, 1975. Thus NaOH will not be part of the design basis accident environment prior to ESF actuation by the pressurizer instruments or by attainment of the high setpoints of the containment pressure instruments.

While the instruments which are typical of the ones used on FNP were being tested, their output voltages were monitored. The pressurizer pressure instruments had a change of 6.5 percent of span while being subjected to the DBA environment. For those instruments assumed to function in the safety analyses, the reactor protection system setpoints will be compatible with the recorded accuracies of environmental testing, normal operational accuracies, and the accident analyses.

Analyses have been performed which include taking into account those short-term environmental inaccuracies reported in letter NS-CE-792, Eichelding to Vassallo, dated October 1, 1975. The corresponding setpoint modifications have been provided in the Farley Technical Specifications. These analyses demonstrate that the design bases are still met for all chapter 15 analyses.

3.11.3 QUALIFICATION TEST RESULTS

A final rule on environmental qualification of electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, established the NRC acceptance criteria and specified the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment may be qualified to the criteria specified in either the DOR Guidelines, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," or NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," except for replacement equipment. Replacement equipment installed subsequent to February 22, 1983, must be qualified in accordance with the provisions of 10 CFR 50.49, using the guidance of Regulatory Guide 1.89, unless there are sound reasons to the contrary. (Reference NRC letter dated December 13, 1984)

In order to address the question for environmental qualification of electrical equipment for the Farley Nuclear Plant, Alabama Power Company organized a task force to review the qualification of electrical equipment. The equipment covered in this review included Class 1E equipment inside containment and Class 1E equipment outside containment which is required to mitigate a postulated accident and is subjected to a harsh environment. Harsh environment is defined as LOCA/MSLB inside the containment and HELB areas outside the containment. Additionally, this review addressed the effects of radiation on equipment outside the containment building during post-LOCA recirculation of containment sump fluids. This scope of review assured that equipment necessary to protect the public health and safety is capable of performing its function when subjected to a harsh environment.

This review of environmental qualification was based on IE Bulletin 79-01B - Environmental Qualification of Class 1E Equipment dated January 14, 1980 and the guidelines outlined for Category II plants as defined by NUREG 0588, which was issued to operating license applicants by NRC letter on February 5, 1980. The review was conducted by a task force composed of personnel experienced in reactor systems safety analysis and design, plant operations, emergency operating procedures, nuclear safety, and environmental qualification. A critical review of all documentation was conducted, using criteria established from IEB 79-01B and NUREG 0588, resulting in an auditable record with appropriate procedures documented to identify the specific equipment, the criteria used in reviewing the report, the reviewer, and the specific report reference.

As a part of this review effort, the task force reviewed the Plant Emergency Procedures to ensure that equipment required by the procedures that could be subjected to a harsh environment is qualified to operate for the time necessary to mitigate the particular accident.

The results of the Farley Nuclear Plant environmental qualification review for each item of safety-related electrical equipment subject to a harsh environment are documented in a submittals to the NRC dated July 30, 1980, for IEB 79-01B and September 15, 1980, as revised and amended. These submittals consisted of tabular listings of all such equipment and appropriate qualification-related data for each item in accordance with the NRC guidelines. Documentation was also provided, for a comparison, of the environmental qualification data against the requirements set forth in IEB 79-01B and NUREG 0588, on report evaluation sheets for each type of equipment to identify the degree to which the qualification complies with the

NRC staff position. Outstanding items were defined as being those for which discrepancies in meeting the guidelines of IEB 79-01B and NUREG 0588 have been identified. A summary of these discrepancies was provided as part of the submittals and included corrective actions and schedules together with justification for interim operation.

3.11.4 LOSS OF VENTILATION

The control room is provided with redundant air conditioning and filtration systems, as described in subsection 9.4.1. This ensures that there will be no loss of ventilation to the control and electrical equipment located within the control room.

All safeguard pumps and motors in the auxiliary building are located in rooms equipped with pump room coolers to provide adequate ventilation for the motors. These coolers are provided as a redundant system, so that if any one pump room cooler fails, the corresponding pump would be shut down, except for the component cooling pumps. An engineering analysis has been performed for all engineered safety feature pump rooms with room coolers. This analysis demonstrates that the equipment in the CCW pump rooms are capable of performing their specified function during the temporary unavailability of one or both room coolers with the plant encountering a design basis accident (DBA). The pump room cooler fan in a safeguard pump room is powered from the same emergency bus as the pump motor. Thus, no single active or passive failure can result in the loss of the safety function of both pumps of a redundant system.

REFERENCES

1. Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report," WCAP 7709-L Supplements 1-4 (Proprietary), October 1973, WCAP 7820 (Non-proprietary), October 1973.

2. Locante, J. and Igne, E. G., "Environmental Testing of Engineered Safety Features-Related Equipment (NSSS Non-standard Scope)," WCAP 7744, Volumes 1 and 2, September 1970.

TABLE 3.11-1 (SHEET 1 OF 5)

EQ PROGRAM ENVIRONMENTAL CONDITIONS

Environmental Conditions	Location		
	Inside Containment	Outside Containment	
		Aux Bldg. Main Steam Room	Aux Bldg. Other Areas
Normal	Category A	Category B	Category C
Pressure, (psig)	0	0	0
Temperature, (°F)	120[a]	104	104[j]
Humidity, (%)	60	50	50[m]
Post Accident Conditions For EQ DBA			
Pressure (psig)	52 [b]	1.3[c]	0[k]
Temperature (°F)	367 [d]	325[e]	104[l]
Humidity (%)	100	100	50[k][m]
Cumulative radiation dose (rads)[f,i]	1.89×10^{7}[g]	3.36×10^{6}	See sheets 2 and 3
Chemical additive	Trisodium phosphate with pH 10.5 2,500 ppm boron	NA	NA
Submergence	el 115'0" and below	el 130'5" and below[h]	NA

TABLE 3.11-1 (SHEET 2 OF 5)

Unit 1 Auxiliary Building
Harsh Environment
Room Radiation Doses (i)

Room Number	Room Description	Total Dose (rads)
111	Containment Spray Pump Room	1.34E6
113	Valve Encapsulation Room	4.04E5
120	Corridor	1.35E5
124	Valve Encapsulation Room	4.04E5
125	Containment Spray Pump Room	2.68E6
128	RHR Heat Exchanger Room	4.17E6
129	RHR Low Head Pump Room	3.52E6
131	RHR Low Head Pump Room	3.52E6
161	Corridor	2.99E5
162	Hallway	3.01E3
172	Hallway	2.78E6
173	Charging/SI Pump Room	2.15E6
174	Charging/SI Pump Room	2.17E6
175	Hallway	1.85E5
181	Charging/SI Pump Room	2.19E6
182	Containment Storage Room	1.24E6
184	Piping Penetration Room	4.76E6
218	BTRS Chiller Unit Room	5.53E3
223	Piping Penetration Room	6.05E6
317	Penetration and Filtration System and Equipment Room	6.90E6
322	Hallway	2.74E5
333	Electrical Penetration Room	5.97E5
334	Electrical Penetration Room	1.34E6
347	Electrical Penetration Room	1.03E6

TABLE 3.11-1 (SHEET 3 OF 5)

Unit 2 Auxiliary Building
Harsh Environment
Room Radiation Doses (i)

Room Number	Room Description	Total Dose (rads)
2111	Containment Spray Pump Room	1.44E6
2113	Valve Encapsulation Room	4.04E5
2120	Corridor	2.56E5
2124	Valve Encapsulation Room	4.04E5
2125	Containment Spray Pump Room	2.68E6
2128	RHR Heat Exchanger Room	4.33E6
2129	RHR Low Head Pump Room	3.43E6
2131	RHR Low Head Pump Room	3.43E6
2161	Corridor	1.96E5
2162	Hallway	3.54E3
2172	Hallway	2.78E6
2173	Charging/SI Pump Room	2.11E6
2174	Charging/SI Pump Room	2.11E6
2175	Hallway	5.25E5
2181	Charging/SI Pump Room	2.11E6
2182	Containment Storage Room	1.85E6
2184	Piping Penetration Room	4.75E6
2218	BTRS Chiller Unit Room	7.34E3
2223	Piping Penetration Room	5.75E6
2317	Penetration and Filtration System and Equipment Room	6.90E6
2322	Hallway	2.74E5
2333	Electrical Penetration Room	5.97E5
2334	Electrical Penetration Room	1.34E6
2347	Electrical Penetration Room	1.03E6

TABLE 3.11-1 (SHEET 4 OF 5)

NOTES:

a. Normal temperature for the out-of-core neutron detectors is 135°F. The neutron detectors are capable of operation at 175°F for 8 h.

b. Peak containment pressure is 52 psig for an MSLB inside containment and 44 psig for a LOCA. The LOCA/MSLB containment composite pressure profile is shown in figure 3.11-1 and covers the DBA duration of 30 days.

c. Peak pressure for an MSLB in the main steam room is 1.3 psig. The main steam room accident pressure profile is shown on FSAR figure 3J-15.

d. Peak containment atmosphere temperature is 367°F for an MSLB inside containment and 264°F for LOCA. The containment accident temperature profiles cover the DBA duration of 30 days. The LOCA/MSLB containment composite temperature profile is shown in figure 3.11-2 and the LOCA-specific profile is shown in figure 6.2-40. Qualification testing envelops either the figure 3.11-2 profile by itself or the figure 6.2-40 profile, with consideration of worst-case MSLB surface temperatures from the spectrum of MSLBs.

e. Peak main steam room atmosphere temperature for an MSLB in the main steam room is 325°F. The main steam room accident temperature profile is shown in figure 3J-10.

f. Radiation dose is cumulative for 40 years[(a)] and 30 days post accident. Individual components may be required to operate less than 30 days as specified in operating time of APC responses to IE Bulletin 79-01B and NUREG 0588.

   The operating licenses for both FNP units have been renewed ad the original licensed operating terms have been extended by 20 years. Radiation dose values used in EQ evaluations will be updated before the units enter the period of extended operation.

g. A radiation dose of 1.89×10^{7} rads is outside the secondary shield wall. The radiation dose for the out-of-core neutron detectors, inside the secondary shield wall, is 1×10^{10} rads. The area of the head of the narrow- and wide-range RCS RTDs from which the RTD pigtails emerge is exposed to a radiation dose of 4.11×10^{7} rads.

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a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. The EQ program is credited to continue to manage aging effects associated with EQ equipment for the period of extended operation (see chapter 18, subsections 18.3.1 and 18.4.4). Applicable EQ evaluations based on a 40-year design life were evaluated as time-limited aging analyses (TLAAs) for license renewal and will be revised as necessary to reflect the 60-year plant operating life before the units enter the period of extended operation.

TABLE 3.11-1 (SHEET 5 OF 5)

h. Submergence is considered below el 130 ft 5 in. for an MSLB in the main steam room.

i. The auxiliary building rooms listed on sheets 2 and 3 of table 3.11-1 are those rooms for which 1) the total integrated dose (TID) is greater than 1E3 rads, 2) in which there is a significant radiation level increase due to an accident, and 3) in which EQ equipment is located. In the context of this note, cable is not considered to be equipment. Therefore, there are no rooms listed on sheets 2 and 3 which have only EQ cable in them. The rooms having EQ cable but no other EQ equipment were not included in the list because a detailed calculation to arrive at a precise value was not performed. Instead, it was assumed that the total radiation dose is 1E8 rads. From the calculated values listed in sheets 2 and 3, this is clearly a conservative assumption.

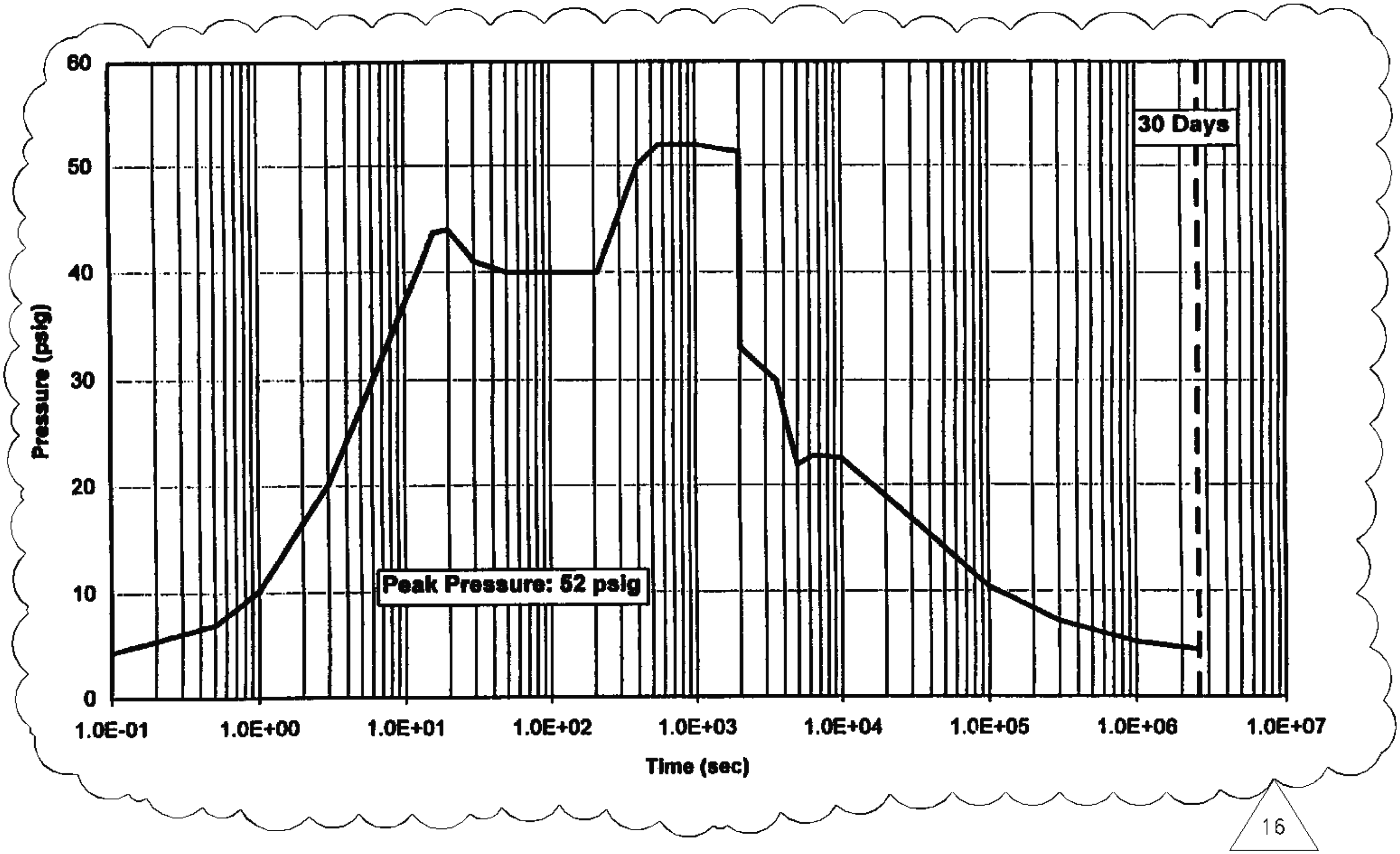
Per the recommendations of EPRI NP-2129, dated November 1981, the threshold values are taken to be 1E3 rads for electronic components, 1E4 rads for equipment utilizing Teflon, and 1E5 rads for all other electrical equipment. Thus, any room listed with a TID less than 1E5 rads is considered a harsh environment only if it has electronic equipment or equipment utilizing Teflon; any room listed with a TID less than 1E4 rads is considered harsh only if it has electronic equipment.

j. Temperatures above 104°F are evaluated to ensure that equipment operability is maintained and is appropriately documented.

k. Post-accident pressure and humidity for all rooms in the auxiliary building would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational transients.

l. For rooms not cooled by service water the post-accident temperature would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational transients. For rooms cooled by service water the post-accident temperatures are documented in table 9.4-6A. The electrical equipment in these rooms has been evaluated and found acceptable for operation at the anticipated temperatures. Therefore, the increase in the ambient temperature of these rooms during the 30-day post-accident period is not considered significant as defined in 10 CFR 50.49 Section c.(iii). The equipment in rooms cooled by service water is thus considered to be in the mild environment except for the rooms that have been previously identified harsh due to the post-accident radiation.

m. The humidity inside the residual heat removal (RHR) and containment spray (CS) encapsulation vessels is ≤ 100%, and the components inside the encapsulation vessels are evaluated to ensure equipment operability is maintained and appropriately documented.



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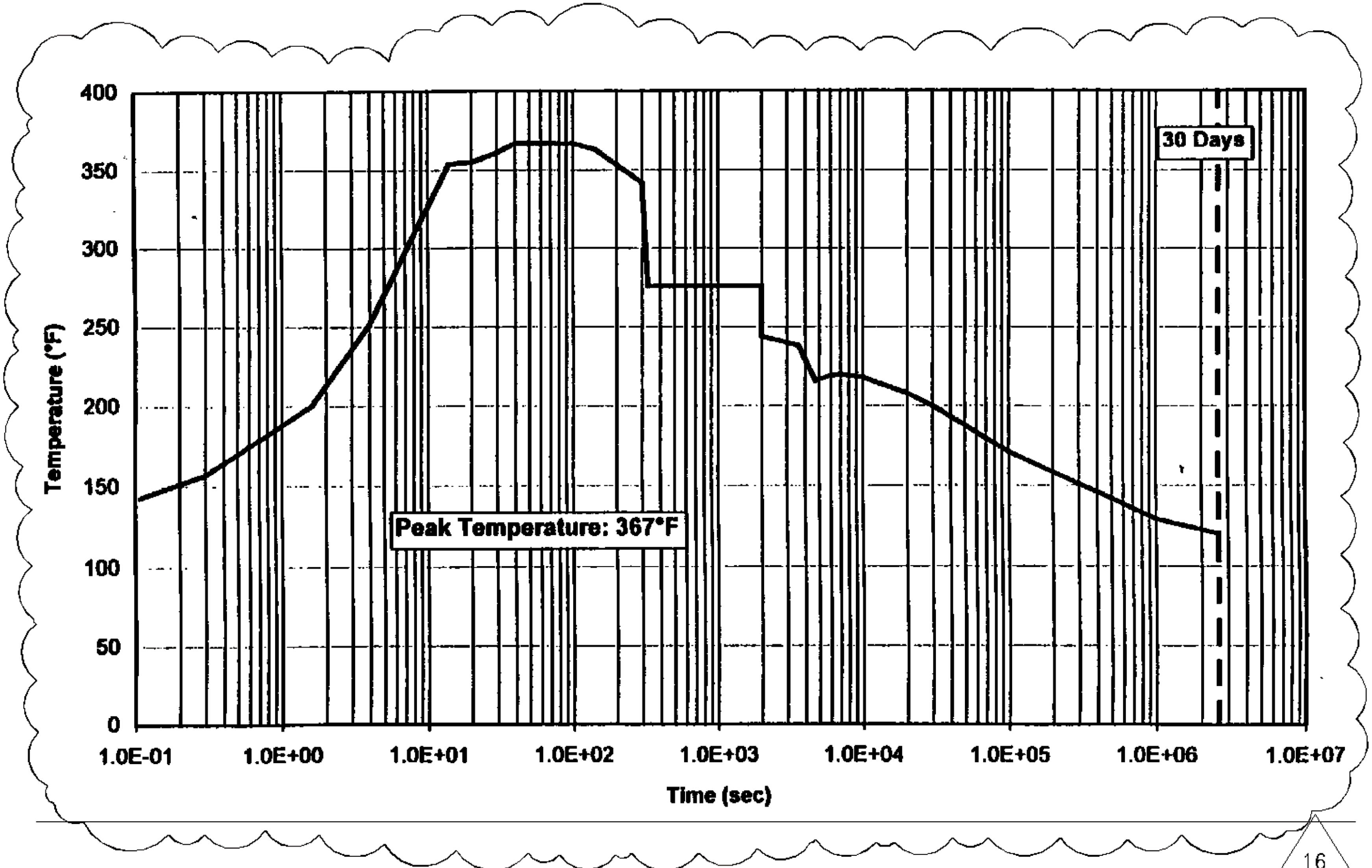
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FNP COMPOSITE LOCA/MSLB
CONTAINMENT PRESSURE PROFILES

FIGURE 3.11-1



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NUCLEAR PLANT
UNIT 1 AND UNIT 2

FNP COMPOSITE LOCA/MSLB
CONTAINMENT TEMPERATURE PROFILES

FIGURE 3.11-2

3A CONFORMANCE WITH NRC REGULATORY GUIDES

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APPENDIX 3A

CONFORMANCE WITH REGULATORY GUIDES

This appendix discusses the extent of conformance of the Farley Nuclear Plant with Division 1 NRC Regulatory Guides, which were issued through the end of August 1974 -- that is, through Regulatory Guide 1.88. Regulatory Guide 1.70 is discussed through 1.70.7. A description of Farley conformance with Regulatory Guides issued subsequent to Regulatory Guide 1.88 is included in this appendix. Specific revision numbers and dates of issue are identified in the title of each guide.

Page numbering for appendix 3A has been changed in Amendment 40 to reflect the appendix letter, division number, particular Guide and page number.

Example:	3A-1.3-1	(one page)
	3A-1.18-2	(two pages)

Regulatory Guide 1.1 - NET POSITIVE SUCTION HEAD FOR ECCS AND CONTAINMENT HEAT REMOVAL PUMPS (SAFETY GUIDE 1, 11/2/70)

CONFORMANCE

The NRC Regulatory Guide 1.1 states that the emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated loss-of-coolant accidents.

As discussed in subsection 6.3.2.14, the emergency core cooling and containment heat removal systems are designed to provide an available NPSH which is greater than pump vendor specified minimum NPSH requirements assuming either of the following conditions:

1. If expected pumped fluid temperatures are less than the saturation temperature at the Technical Specifications (TS) minimum operating containment pressure prior to the accident, NPSH is determined assuming no increase in containment pressure from that present prior to postulated loss-of-coolant accidents.

2. If expected maximum pumped fluid temperatures exceed the saturation temperature at the TS minimum operating containment pressure prior to the accident, NPSH is determined assuming the containment pressure is equal to the pumped liquid vapor pressure (saturated conditions).

Adequate net positive suction head is provided to both the RHR and containment spray pumps. In calculating the available NPSH during injection, no credit is taken for any water within the RWST and full penalty is taken for head losses based on actual piping layouts. During the recirculation mode, credit is taken for a minimum available water level above the top of the inlet to the containment spray pump suction piping and the RHR suction piping during recirculation following a LBLOCA event which generates bounding debris head losses. No credit is taken for any increase in containment pressure from a postulated accident.

Additional conservatism is introduced by assuming that all recirculated sump water is at saturation conditions for sump temperatures above the saturation temperature at the minimum operating containment pressure prior to the accident.

The methods utilized in calculating NPSH for the Farley Nuclear Plant, as described above and in subsection 6.3.2.14, are adequately conservative and meet Regulatory Guide 1.1 by ensuring adequate NPSH with adequate margin for the centrifugal charging, safety injection, residual heat removal, and containment spray pumps.

Regulatory Guide 1.2 - THERMAL SHOCK TO REACTOR PRESSURE VESSELS (SAFETY GUIDE 2, 11/2/70)

CONFORMANCE

The reactor pressure vessel design supported by current research programs in the area of fracture toughness of reactor vessel materials conforms to the intent of Regulatory Guide 1.2. Fracture toughness is discussed in subsection 5.2.4; the capability for annealing the reactor vessel is discussed in subsection 5.4.3.7.

Regulatory Guide 1.3 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS (Rev. 2, 6/74)

CONFORMANCE

The Guide is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.4 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS (Rev. 2, 6/74)

CONFORMANCE

The assumptions of the regulatory position of Regulatory Guide 1.4 are used with exceptions described in the analysis of the design basis loss-of-coolant accident in subsection 15.4.1.

Regulatory Guide 1.5 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS (SAFETY GUIDE 5, 3/10/71)

CONFORMANCE

The Guide is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.6 - INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS (SAFETY GUIDE 6, 3/10/71)

CONFORMANCE

As described in subsection 8.3.1.2(c), the applicant's design conforms to the regulatory position without exception.

Regulatory Guide 1.7 - CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT (SAFETY GUIDE 7, 3/10/71)

CONFORMANCE

The design guidance and assumptions for analysis of the regulatory position of Regulatory Guide 1.7 are used without exception for control of combustible gas concentrations in containment following a loss-of-coolant accident, as described in subsections 6.2.5, 7.6.4, and 15.4.1.

Regulatory Guide 1.8 - PERSONNEL SELECTION AND TRAINING (SAFETY GUIDE 8, 3/10/71)

CONFORMANCE

Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in 10 CFR 55, except for the Health Physics manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. Personnel who complete an accredited program which has been endorsed by the NRC shall meet the requirements of the accredited program in lieu of the above. Since the occurrence of the TMI-2 accident, additional shift manpower has been added and an upgrading process has been instituted for the training and qualification of operating personnel. These changes are in conformance with NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," as modified by NUREG-0737, "Clarification of TMI Action Plan Requirements." Details of the training program are presented in Section 13.2.

<u>Regulatory Guide 1.9</u> - SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES (SAFETY GUIDE 9, 3/10/71)

<u>CONFORMANCE</u>

The standby power system is discussed fully in subsection 8.3.1.2(d), AC Power Systems, and meets the recommendations of Regulatory Guide 1.9.

Regulatory Guide 1.10 - MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CONCRETE CONTAINMENTS (Rev. 1, 1/2/73)

CONFORMANCE

The recommendations of Regulatory Guide 1.10 were the basis for testing and inspecting all mechanical splices utilized at the facility. However, a disadvantage of using production splices for testing was that each production splice removed had to be replaced by two cadwelds, thus introducing additional splices into the structure. Further, production splices in hoop bars were not tested, since the test itself imposes bending in addition to tension and would not represent the actual stressing of the bars. Accordingly, splices in curved bars were tested by the "sister splice" method while straight bar splices were tested by either method.

Additional information is contained in appendix 3C and subsection 4.4.2.

Regulatory Guide 1.11 - INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT (SAFETY GUIDE 11, 3/10/71)

CONFORMANCE

Sensing lines penetrating containment are provided with isolation valves in accordance with NRC General Design Criteria 55 or 56 or meet regulatory position C.2 of Regulatory Guide 1.11. The sensing lines, configurations are discussed in subsection 6.2.4.

The containment pressure sensing lines for the Post-Accident Monitoring (PAM) and Engineered Safety Features Actuation (ESF)/Reactor Protection System (RPS) must be open to containment at all times. The transmitters are located on unvalved lines outside containment with remote seal sensors located inside containment. This arrangement is considered to be in compliance with General Design Criterion 56 as the sensors are located as close to containment boundary as practical and their pressure boundaries are either ASME Code Class 2 or are a double pressure boundary rated higher than the containment design pressure.

Regulatory Guide 1.12 - INSTRUMENTATION FOR EARTHQUAKES (Rev. 1, April 1974)

The Regulatory Guide 1.12 guidelines for instrumentation to monitor earthquakes has been replaced by EPRI Reports. The NRC has accepted the EPRI Reports listed in section 3.7.4 as an acceptable approach to meet the seismic monitoring requirements and determine plant action following an earthquake.

Regulatory Guide 1.13 - FUEL STORAGE FACILITY DESIGN BASES (SAFETY GUIDE 13, 3/10/71)

CONFORMANCE

Design of the wet spent-fuel facility complies with the recommendations of Regulatory Guide 1.13 in the manner described in subsections 9.1.2, Wet Spent-Fuel Storage; 9.1.3, Spent-Fuel Pool Cooling and Cleanup; 9.1.4, Fuel Handling and Spent-Fuel Cask Crane; section 3.5, Missile Protection; subsection 3.8.4, Design of Other Category I Structures; and section 9.4, Air Conditioning, Heating, Cooling and Ventilation.

Regulatory Guide 1.14 - REACTOR COOLANT PUMP FLYWHEEL INTEGRITY (SAFETY GUIDE 14, 10/27/71)

CONFORMANCE

This topic is addressed in subsection 5.2.6.

Regulatory Guide 1.15 - TESTING OF REINFORCING BARS FOR CONCRETE STRUCTURES (Rev. 1, 12/28/72)

CONFORMANCE

The methodology for testing reinforcing bars used in Seismic Category I concrete structures conforms with the recommendations of Regulatory Guide 1.15 as described in subsection 3.8.1.6.2.

Regulatory Guide 1.16 - REPORTING OF OPERATING INFORMATION (Rev. 1, 10/73)

CONFORMANCE

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the program for reporting of Farley, Units 1 & 2, operating information is in accordance with Generic Letter 97-02, “Revised Contents of the Monthly Operating Report,” dated May 15, 1997.

Reporting Requirements are contained within the Technical Specifications.

Regulatory Guide 1.17 - PROTECTION AGAINST INDUSTRIAL SABOTAGE (Rev. 0, 6/73)

CONFORMANCE

The requirements of the Regulatory Guide are met as described in section 13.7 and the Security Plan.

Regulatory Guide 1.18 - STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS (Rev. 1, 12/28/72)

CONFORMANCE

Regulatory Guide 1.18 establishes a systematic approach to testing wherein quantitative information is obtained concerning structural response to pressurization. The following discussion is provided to clarify the extent of compliance to this Regulatory Guide.

1. Reference: Paragraph C.1 of the Regulatory Guide

   A continuous increase in containment pressure, rather than incremental pressure increases, is considered acceptable, since data collection is made rapidly at each pressure datum. "Rapidly" is defined as requiring a time interval for data collection sufficiently short so that the change in pressure while the data are being collected would cause a change in the structural response of less than 5 percent of the total anticipated change. For example, assume a pressure datum at each 15-lb/in^2 interval, a test pressure of 60 lb/in^2, and a total expected strain of 200 microstrains (microinches/in.). The time interval for data collection, therefore, is required to be equal to, or less than, the time during which pressurization would create a 10-microstrain change.

   When the test pressure reached its maximum in the containment, a hold period of at least 1 hour, or of such duration as necessary for recording crack patterns, was provided.

2. Reference: Paragraph C.5 of the Regulatory Guide

   The design of the Farley Containment closely follows those of the Arkansas Nuclear One Unit 1 (Docket No. 50-313) and Millstone Unit 2 (Docket No. 50-336), and therefore is not a prototype containment. Consequently, this paragraph is not considered to be applicable.

3. Reference: Paragraph C.9 of the Regulatory Guide

   The structural integrity test was scheduled for periods in which extremely inclement weather is not forecast. However, due to the state-of-the-art of weather forecasting, and the time involved in the preparation and performance of the test, should snow, heavy rain, or strong wind occur during the test, it may be continued and the results considered valid unless evidence indicates otherwise. A retest will be made if the results are found to be invalid.

4. Reference: Paragraph C.10 of the Regulatory Guide

   Due to the amount of time involved in preparing for and performing the test, should the test pressure drop due to unexpected condition to or below the next lower pressure level, it is intended to continue the test, without a restart at atmospheric pressure, unless the structural response deviates significantly from that expected.

5. Reference: Paragraph C.12 of the Regulatory Guide

   It is believed that this paragraph applies to PSAR only. However, the type of information which will be included in the final test report will conform to Paragraph C.13 of the Regulatory Guide.

6. Reference: Appendix A.2.a of the Regulatory Guide

   The design of the Farley Containment closely follows those of the Arkansas Nuclear One Unit 1 and the Millstone Unit 2, and therefore is not a prototype containment. Consequently, this paragraph is not considered to be applicable.

7. Reference: Appendix A.2.g of the Regulatory Guide

   The design of the Farley Containment closely follows those of the Arkansas Nuclear One Unit 1 and the Millstone Unit 2, and therefore is not a prototype containment. Consequently, this paragraph is not considered to be applicable.

A description of the containment structural acceptance test is presented in appendix 3H.

Regulatory Guide 1.19 - NON-DESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS (SAFETY GUIDE 19, Rev. 1, 8/11/72)

CONFORMANCE

The recommendations of Regulatory Guide 1.19 have been met. See appendix 3G for details.

Regulatory Guide 1.20 - VIBRATION MEASUREMENTS ON REACTOR INTERNALS (SAFETY GUIDE 20, 12/29/71)

CONFORMANCE

This topic is addressed in subsection 3.9.1.3.1.

Regulatory Guide 1.21 - MEASURING AND REPORTING OF EFFLUENTS FROM NUCLEAR POWER PLANTS (SAFETY GUIDE 21, 12/29/71)

CONFORMANCE

The Farley units are in compliance with Regulatory Guide 1.21 with the following exceptions:

1. There are no continuous monitors on the turbine building drains. Grab samples are taken for composite prior to or during its discharge for each batch released.

2. Gamma spectroscopy measurements are used as the basis for estimating the quantity of low-level particulate activity released.

3. I-135 is not monitored in gaseous effluents due to its short half-life.

4. The steam jet air ejector is monitored by monthly grab samples in accordance with the Offsite Dose Calculation Manual.

Regulatory Guide 1.22 - PERIODIC TESTING 0F PROTECTION SYSTEM ACTUATION FUNCTIONS (SAFETY GUIDE 22, 2/17/72)

CONFORMANCE

The design recommendations of Regulatory Guide 1.22 have been met. Design of the protection system is discussed fully in chapter 7, Instrumentation and Controls. Some tests must be performed as sequential steps on isolated portions of the system so that an actual reactor scram does not occur as a result of the testing. Specific discussion is found in subsections 7.1.2, 7.2.3, and 7.3.2.

Regulatory Guide 1.23 - ONSITE METEROLOGICAL PROGRAMS (SAFETY GUIDE 23, 2/17/72)

CONFORMANCE

Meteorological programs are discussed in Section 2.3 in detail.

Regulatory Guide 1.23, which was issued in 1972, states in paragraph 4 of its Regulatory Position that temperature difference measurements should have an accuracy of ±0.1°C.

The thermistors used in the Farley tower temperature difference circuits were ordered in late 1970 and have given excellent service (well over 90-percent recovery) for the past 12 years. Inspection of the analog charts has shown very few questionable temperature difference readings or calibration or drift problems.

At the factory, each thermistor is checked against a calibration curve to ensure the accuracy is 0.15°C over its entire temperature range. Most of the measurements at the Farley site are taken over a small portion of this total range; therefore, the accuracy is better than 0.15°C. In view of the reliability and quality of the data recorded using this system, modifications are not necessary.

Regulatory Guide 1.23 makes no recommendation for maintaining the area surrounding the met tower free from obstructions. The Farley Nuclear Plant met tower will be maintained free from obstructions to wind flow in accordance with proposed Revision 1 to Regulatory Guide 1.23 as recommended by NUREG-0654, Revision 1, Appendix 2.

Regulatory Guide 1.24 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL-RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE (SAFETY GUIDE 24, 3/23/72)

CONF0RMANCE

The assumptions of the regulatory position of Regulatory Guide 1.24 are used without exception in the analysis of the potential radiological consequences of the failure of a radioactive gas storage tank in section 15.3.

Regulatory Guide 1.25 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS (SAFETY GUIDE 25, 3/23/72)

CONFORMANCE

The assumptions of the regulatory position of Regulatory Guide 1.25 as used in the analysis of the potential radiological consequences of a fuel handling accident, are discussed in subsection 15.4.5.

Regulatory Guide 1.26 - QUALITY GROUP CLASSIFICATION AND STANDARDS (SAFETY GUIDE 26, 3/23/72)

CONFORMANCE

Equipment classification and code requirements are given in subsection 3.2.2. The classification system of ANSI "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," August 1970 draft is an alternate acceptable method of meeting the intent of Regulatory Guide 1.26. Since there was no established commercial standard for pumps at the time of the license application, ASME Boiler and Pressure Vessel Code Subsection VIII, Division 1, and ANSI B31.1.0, Power Piping, represented related available standards that, while intended for other applications, were used for guidance and recommendations in determining quality group D pump construction requirements, such as allowable stresses, steel casting quality factors, wall thicknesses, materials compatibility and specifications, temperature pressure environment restrictions, fittings, flanges, gaskets, bolting, and installation procedures.

Regulatory Guide 1.27 - ULTIMATE HEAT SINK (Rev. 2, 1/76)

CONFORMANCE

The ultimate heat sink meets the recommendations of Regulatory Guide 1.27. Compliance is discussed fully in subsection 9.2.5.

[HISTORICAL] *[Regulatory Guide 1.28* - *QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION) (SAFETY GUIDE 28, 6/7/72)*

CONFORMANCE

This topic is addressed in section 17.1.]

Regulatory Guide 1.29 - SEISMIC DESIGN CLASSIFICATION (Rev. 1, 8/73)

CONFORMANCE

Design of structures, systems, and components complies with the recommendations of Regulatory Guide 1.29. Seismic design classification is discussed in subsection 3.2.1, Seismic Classification.

Regulatory Guide 1.30 - QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRIC EQUIPMENT (SAFETY GUIDE 30, 8/11/72)

CONFORMANCE

[HISTORICAL] *[This topic is addressed in chapter 17.]*

Regulatory Guide 1.30 provided NRC endorsement of ANSI N45.2.4 (IEEE 336-1971). The SNC Quality Assurance Topical Report (QATR) is based on NQA-1-1994 which incorporates IEEE 336-1985. Accordingly, the quality assurance requirements for the installation, inspection, and testing of instrumentation and electric equipment are described in the QATR.

Regulatory Guide 1.31 - CONTROL OF STAINLESS STEEL WELDING (Rev. 1, 6/73)

CONFORMANCE

For Westinghouse scope of supply Farley Nuclear Plant conforms to the intent of Regulatory Guide 1.31 as discussed in Subsection 5.2.5.5, paragraphs 3 through 7, and Subsection 5.2.5.7. The tests described in these sections are in accordance with the requirements of ASME III, NB2430. In general, production welds were not checked for delta ferrite measurements, but the combination of specifying ferrite content in material procurement together with ASME Code required examinations provide assurance of the integrity of stainless steel welds.

Outside of Westinghouse scope of supply, conformance is as follows:

1. As an equivalent alternate method to Regulatory Guide 1.31 for the prevention of potential fissures in austenitic stainless steel welds for service temperatures up to 650°F, all welding material pertaining to AWS classification ER308L or E308L has a ferrite content within the range of 8 to 25 percent, and all welding material pertaining to AWS classification ER309 or E309 has a ferrite content within the range of 5 to 15 percent. For bare electrode, rod, or wire filler metal used with gas metal arc welding or gas tungsten arc welding processes, the chemical analysis was performed on the electrode, rod, wire, or consumable insert, or an undiluted weld deposit made with the bare filler materials. For all other processes and filler metal, the chemical analysis was performed on an undiluted weld deposit. The delta ferrite content was determined by analysis of each heat or lot as applicable as required by the ASME Boiler and Pressure Vessel Code Subsection III, Subarticle NB-2400.

2. Control of the chemical analysis of the weld filler metal in this range ensured adequate ferrite in the weld deposit in order to prevent fissuring in austenitic stainless steel welds. This made it unnecessary to make magnetic measurements of welding procedure samples or production welds. Magnetic measurements were subject to mass effects and were unreliable on groove welds such as root passes on piping where the potential for fissuring may be highest. At best, the magnetic measurements on actual welds were only semiquantitative and are subject to a variety of interpretations. The details of a welding procedure have minor effects on the quantity of ferrite in a weld, but the degree of variation was well within the margin of error of measurement.

3. Meaningful control of ferrite was accomplished by control of weld filler metal based on chemical analysis with an adequate margin above the minimum required to prevent fissuring under all conditions. There was no technical basis to control the upper limit of ferrite to 15 percent because the more ferrite that exists up to the point that it becomes the continuous phase (about 40 percent), the more resistant the welds are to fissuring. For example, type 312 electrodes, nominally 29 percent chromium and 9 percent nickel, are noted for excellent resistance to fissuring, and the ferrite content is about 30 percent. Many austenitic stainless steel castings, particularly grades CF3A, CF8A, CF3MA and CF8MA, purposely contain up to 30 percent ferrite to prevent fissures, hot tears

and shrinkage. Weld metal of similar composition and ferrite content is also desirable.

4. The restriction of using only one heat of weld rod in a particular joint presented significant problems. Where a number of weldments require 600-700 lb of rod of one heat of weld per joint, setting aside sufficient rods from a separate heat for each joint was not considered necessary; chemical analysis of weld material provided required assurance for integrity of the welds.

Regulatory Guide 1.32 - USE OF IEEE STD 308-1971, "CRITERIA FOR CLASS IE ELECTRICAL SYSTEMS FOR NUCLEAR POWER GENERATING STATIONS" (SAFETY GUIDE 32, 8/11/72)

CONFORMANCE

1. Design of the electric power system complies with IEEE Standard 308-1971. The degree of conformance is discussed in subsection 8.3.1.2(e).

2. Availability of power from the transmission network is of preferred design as recommended by Regulatory Guide 1.32. The design is discussed in subsection 8.2.1.3, Compliance with NRC Design Criteria.

3. Battery charger capacity complies with that recommended by Regulatory Guide 1.32 and is discussed in subsection 8.3.2.1.2.

Regulatory Guide 1.33 - QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION) (SAFETY GUIDE 33, 11/3/72)

CONFORMANCE

[HISTORICAL] *[During original plant licensing, a 24-month review process for all safety-related and security procedures was developed to meet the intent of Safety Guide 33 and ANSI N18.7-1972. Since the procedural process has now matured and adequate programs to assure procedural revisions consistent with plant design, operational, and regulatory requirements are in place, this original commitment has been modified to require biennial quality assurance audits of the procedural development and maintenance program utilizing a representative sampling process. Therefore, the 24-month review process is no longer required.*

Conduct of operations of the Safety Review Board, including audits performed under the cognizance of the Safety Review Board, meet the requirements of ANSI N18.7-1976, Section 4, Review and Audit.

Regulatory Guide 1.33, Section 4 provides that the following program elements should be audited at the indicated frequencies: the results of actions taken to correct deficiencies that affect nuclear safety and occur in facility equipment, structures, systems, or method of operation - at least once per 6 months; the conformance of facility operation to provisions contained within the Technical Specifications and applicable licensing conditions – at least once per 12 months; and the performance, training, and qualifications of the facility staff – at least once per 12 months. Audit frequencies for each of these program elements are now established as at least once per 24 months.]

The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 and incorporates the applicable requirements of ANSI N18.7-1976. Accordingly, SNC complies with the applicable requirements of ANSI N18.7-1976 via compliance with the QATR without an explicit (or implied) commitment to ANSI N18.7-1976.

Regulatory Guide 1.34 - CONTROL OF ELECTROSLAG WELD PROPERTIES (Rev. 0, 12/28/72)

CONFORMANCE

Regulatory Guide 1.34 is not applicable to the Farley Nuclear Plant, inasmuch as electroslag welds were not used in the fabrication of core support structures nor in Class 1 and 2 vessels and components.

[HISTORICAL]

[Regulatory Guide 1.35 - *INSERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES (Rev. 2, 1/76)*

CONFORMANCE

The containment tendon surveillance program for the Farley containment prestressing system complies with Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," Rev. 2. The containment tendon surveillance program is discussed in subsection 3.8.1.]

Regulatory Guide 1.36 - NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL (Rev. 0, 2/23/73)

CONFORMANCE

Regulatory Guide 1.36 is not applicable for components within the reactor coolant pressure boundary, since only stainless steel metal reflective insulation was used on reactor coolant pressure boundary austenitic stainless steel piping and equipment.

For austenitic stainless steel piping and components outside the reactor coolant pressure boundary, Regulatory Guide 1.36 was followed.

Regulatory Guide 1.37 - QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS (Rev. 0, 3/16/73)

CONFORMANCE

[HISTORICAL] *[Preoperational cleaning and layup and associated activities involving the cleanliness of safety-related fluid systems were performed in accordance with this Regulatory Guide as discussed in chapter 17.]*

Regulatory Guide 1.37 provides NRC endorsement of ANSI N45.2.1. The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 which incorporates the requirements of ANSI N45.2.1. Accordingly, quality assurance requirements for cleaning of fluid systems and associated components are described in the SNC QATR.

Regulatory Guide 1.38 - QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER COOLED NUCLEAR POWER PLANTS (ANSI N45.2.2-1972)

CONFORMANCE

[HISTORICAL] *[Conformance with applicable sections of ANSI N45.2.2-1972 and hence with this Regulatory Guide is discussed in sections 17.1 and 17.2.*

Certification of involved personnel in accordance with a recommended practice such as SNT-TC-1A-1968 was considered to be excessive. Personnel involved in conducting activities governed by this ANSI standard were properly qualified by reason of experience and training.

Exception is taken to paragraph 3 of the Regulatory Guide. Tape used to secure caps to stainless steel pipe was "essentially chloride free and approved by the purchaser prior to use." Polyethylene was used in combination with a wooden plug to protect all nonflanged stainless steel pipe openings larger than 2 in.]

The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 which incorporates the requirements of ANSI N45.2.2. Accordingly, quality assurance requirements for packaging, shipping, receiving, storage, and handling are described in the SNC QATR.

Regulatory Guide 1.39 - HOUSEKEEPING REQUIREMENTS FOR WATER COOLED NUCLEAR POWER PLANTS (Rev. 0, 3/16/73)

CONFORMANCE

[HISTORICAL] *[The housekeeping requirements during the construction phase were established prior to issuance of Regulatory Guide 1.39. These requirements were structured to meet the standards of Nuclear Electric Insurance Limited (NEIL) and the Occupational Safety and Health Act (OSHA).*

Housekeeping activities during operation meet the requirements of ANSI Standard N45.2.3-1973, except with regard to the general fire protection guidelines of subdivision 3.2.3. The FNP fire protection program is described in Appendix 9B.]

The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 which incorporates the requirements of ANSI N45.2.3. Accordingly, housekeeping requirements are described in the SNC QATR.

Regulatory Guide 1.40 - QUALIFICATION TESTS OF CONTINUOUS DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER COOLED NUCLEAR POWER PLANTS (Rev. 0, 3/16/73)

CONFORMANCE

Continuous-duty, Class 1[a] motors installed within the containment were purchased on the basis that prototypes were type-tested to the requirements of IEEE Standard 334-1971, "IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations." Qualification reports from the vendors covering the tests were reviewed for full compliance. The tests covered, as far as practicable, any auxiliary equipment that was a part of the complete motor assembly. There was no special condition to meet for the auxiliary equipment, as was exampled by Regulatory Position 2.

a. In later IEEE standards, called Class 1E.

Regulatory Guide 1.41 - PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEMS TO VERIFY PROPER LOAD GROUP ASSIGNMENTS (Rev. 0, 3/16/73)

CONFORMANCE

This topic is discussed in subsection 14.1.3 and also in the conformance to Regulatory Guide 1.79.

Regulatory Guide 1.42 - INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHT WATER COOLED NUCLEAR POWER REACTORS (Rev. 1, 3/74)

CONFORMANCE

This topic is addressed in subsection 11.3.6.

Regulatory Guide 1.43 - CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS (Rev. 0, 5/73)

The Farley Units are in compliance with Regulatory Guide 1.43 as demonstrated by the following:

The reactor vessel bottom head, intermediate, and lower shell courses were fabricated from A-533 Grade B Class 1 plate material and clad by use of the 3-wire submerged arc process. The closure head is fabricated from an SA-508 Grade 3 Class 1 forging and is strip clad with the electroslag process and manual welding as dictated by the geometry. These materials and cladding processes are not restricted by Regulatory Guide 1.43 and, consequently, are in compliance with the Guide.

The closure head flange, vessel flange, nozzles, lower transition ring, and upper shell course were fabricated from A-508 Class 2 forging material and clad by use of either the single-wire or 3-wire submerged arc process or the manual metal arc process or combinations of these processes. Since these welding methods are not considered as high heat input processes, the above components are in compliance with the Guide.

Stainless steel weld cladding was applied to the steam generator channel head surface in contact with primary coolant. The head, including the nozzles and manway opening, are integrally forged SA-508 Class 3 material. The head hemispherical surface was clad by the shielded metal arc process or strip cladding submerged arc process with controlled dilution of the deposit, and the channel head nozzles and manway openings were clad by the shielded metal arc or tungsten arc welding process. Both processes are low heat input techniques. The material and the weld processes are not restricted by the Guide. Therefore, the steam generator stainless steel cladding meets the recommendations of the Guide.

Stainless steel weld cladding was applied to the pressurizer shell courses, heads, spray nozzle, and manway opening surfaces in contact with primary coolant. The pressurizer shell and upper and lower heads are constructed of SA-533 Grade A Class 2 material. The shell courses are clad by the plasma process. The heads are clad by the 2-wire series submerged arc process with controlled dilution of the deposit.

The pressurizer lower head nozzle and the upper head manway openings are constructed of SA-508 Class 2 material, use of which is restricted by the Guide.

However, the manual metal arc process, a low heat input process, was used to apply the weld cladding on these surfaces. Because of the low heat input processes and/or materials used for the stainless steel weld cladding, the intent of Regulatory Guide 1.43 is met for the pressurizer.

Regulatory Guide 1.44 - CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL (Rev. 0, 5/73)

CONFORMANCE

The Farley Nuclear Plant meets the intent of Regulatory Guide 1.44. It is Westinghouse practice to use processing, packaging and shipping controls, and preoperational cleaning to preclude adverse effects of exposure to contaminants on all stainless steel materials as recommended by Position 1 of the Regulatory Guide.

All austenitic stainless steel starting materials were procured from the raw material suppliers in the final heat-treated condition required by the respective ASME Code subsection II material specification for the particular type or grade of alloy, as recommended by the Guide, Regulatory Position 2.

Westinghouse met the intent of the remaining positions of Regulatory Guide 1.44 by using the most practicable and conservative methods and techniques to avoid partial or local severe sensitization. Methods and material techniques that were employed to avoid partial or local severe sensitization are discussed in subsection 5.2.5.

Moreover, Westinghouse technical background and service experience, as detailed in WCAP-7735 (reference 7 of section 5.2), support the conclusion that serious intergranular attack of sensitized stainless steel is unlikely in Westinghouse PWR nuclear steam supply systems, since water chemistry and contamination are kept under control.

Regulatory Guide 1.45 - REACTOR COOLANT PRESSURE BOUNDARY LEAK DETECTION SYSTEMS (Rev. 0, 5/73)

CONFORMANCE

The leakage detection system as described in subsection 5.2.7 meets the intent of Regulatory Guide 1.45. The instrumentation available provides approximately 1 gal/min RCS leak detection and a response time of approximately 1 hour under conditions described in subsection 5.2.7.

Calibration of the leakage detection system is performed during plant shutdown. Experience on similar equipment has shown calibration performed during plant shutdown is reliable and sufficient.

Although the leak detection system does not include a sump monitoring system as required by the Regulatory Guide, an acceptable alternative is the plant's condensate measuring system, as documented in NUREG-75/034, section 5.6.

Farley's reactor coolant pressure boundary is designed to withstand the design basis earthquake (DBE) loads; therefore, no failure in the system following such an event is expected. The present leakage detection system in FNP forewarns the operator of minor leakages that may develop during normal operation; however, the system neither performs a safeguard function nor is required to operate during or after a seismic event. Therefore, the system, except for the containment air particulate and gas monitors, is not seismically qualified.

The containment air particulate and gas monitor (channels R-11 and R-12) will be qualified to function following a safe shutdown earthquake as described in paragraph 11.4.2.2.3. The skid monitor control panel will provide an alarm signal to the control room on increasing radiation levels and a local indication.

Regulatory Guide 1.46 - PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT (Rev. 0, 5/73)

CONFORMANCE

Conformance is discussed in section 3.6.

Regulatory Guide 1.47 - BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS (Rev. 0, 5/73)

CONFORMANCE

Extensive control room indication is employed in the protection and engineered safety features systems to provide status and availability information to plant personnel, as discussed in chapter 7. Specifically, the existing design of the main control board provides adequate information to the operator on the component level. Component level instruments associated with the same system are grouped together on the main control board. This information is applied by the trained operator to establish the effects of component status, as it relates to overall system status and availability.

A system to provide indication on a system level was not a requirement when the plant was designed, and the specific recommendations of Regulatory Guide 1.47, particularly the requirement to provide automatic bypass indication at the system level, was not anticipated at the time of the design.

A manually operated light display board was subsequently installed in the main control room on a single panel to show those engineered safety features that are bypassed by deliberate operator action. These systems are train oriented and arranged on the board to indicate clearly which system in a single train is bypassed. The systems displayed on this board are as follows:

1. Containment spray.
2. RHR.
3. High-head safety injection.
4. Component cooling water.
5. Auxiliary feedwater.
6. Post-LOCA combustible gas control.
7. Main steam line isolation.
8. Containment isolation.
9. Safety-related heating, ventilation, and air conditioning (HVAC).
    a. Penetration room HVAC.
    b. Control room emergency HVAC.
    c. Containment cooling.
    d. Spent fuel pool.

10. Service water.

11. Emergency power.

These features coupled with the administrative procedures outlined in the technical specifications provide adequate insurance that systems important to safety are not inadvertently bypassed.

Regulatory Guide 1.48 - DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS (Rev. 0, 5/73)

CONFORMANCE

Sections 3.7, 3.9, and 5.2 include the design criteria for Seismic Category I fluid system components.

<u>Regulatory Guide 1.49</u> - POWER LEVELS OF NUCLEAR POWER PLANTS (Rev. 1, 12/73)

<u>CONFORMANCE</u>

The rated thermal power (i.e., core thermal power) level of the Farley Nuclear Plant is 2775 MWt, which is below the maximum limit of 3800 MWt recommended in Regulatory Guide 1.49.

Regulatory Guide 1.50 - CONTROL OF PREHEAT TEMPERATURE FOR WELDING OF LOW-ALLOY STEEL (Rev. 0, 5/73)

CONFORMANCE

Regulatory Guide 1.50 describes an acceptable method of implementing the requirements of 10 CFR 50 in regard to the control of welding for low-alloy steel components during initial fabrication.

The Farley Nuclear Plant conforms to Regulatory Guide 1.50 in the following manner:

1. Preheat for welding of low-alloy steel was controlled in accordance with the regulatory position of Regulatory Guide 1.50 except as described in sections 2 through 4 below.

2. The position of Regulatory Guide Part C, Paragraph I.a, was met when impact testing, in accordance with ASME Boiler and Pressure Vessel Code, Subsection III, Subarticle 2300, was required. When impact testing was not required, specifying a maximum interpass temperature in the welding procedure was not necessary in order to ensure that the other required mechanical properties of the weld were met.

3. Compliance with Regulatory Guide Part C, Paragraph 2, was required for pressure vessels with nominal thicknesses greater than 1 in. Maintaining preheat after welding until postweld heat treatment (PWHT) was not required for thinner sections, since experience indicated that delayed cracking in the weld or heat- affected zone (HAZ) was not a problem.

4. Usage of low-alloy steel in piping, pumps, and valves was minimal and was normally limited to Class 3 construction. When low-alloy steel piping, pumps, and valves were used, preheat was maintained until welding was complete, but not until PWHT was performed.

Regulatory Guide 1.51 - INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS (Rev. 0, 5/73)

CONFORMANCE

This topic is discussed in subsection 5.2.8.

Regulatory Guide 1.52 - DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (Rev. 0, 6/73)

DESIGN, INSPECTION, AND TESTING CRITERIA FOR AIR FILTRATION AND ADSORPTION UNITS OF POST ACCIDENT ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEMS IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (Rev. 3, 6/01)

CONFORMANCE

Regulatory Guide 1.52 provides detailed information on design, testing, and maintenance for air filtration and adsorption units of atmosphere cleanup systems in light-water-cooled nuclear power plants. Conformance with this guide is not complete because at the time of the design of the atmosphere cleanup systems, such as penetration room filtration system, the guide ORNL-NSIC-65, "Design, Construction and Testing of High Efficiency Air Filtration Systems for Nuclear Application," was the only available design reference. However, each engineered safety feature air filtration system is designed to safely and reliably mitigate the consequences of the postulated accidents.

The Farley Nuclear Plant conforms to the Regulatory Guide with the following exceptions:

1. Penetration Room Filtration Unit

   Reference Paragraph C.2.a of the Regulatory Guide. No demister was provided because the unit is located outside the containment and no entrained water droplets are anticipated. No high efficiency particulate air (HEPA) filters are provided downstream of the charcoal since radioactive fines carryover is very unlikely. This is true because the charcoal trays are pressure tested at high velocity in the manufacturer's shop prior to delivery, thereby removing fines. Also, during system operation, air is passing through the charcoal at a very low velocity.

2. Control Room Filtration, Recirculation, and Pressurization Units

   Reference Paragraph C.2.a of the Regulatory Guide. No demister was provided because the unit is located outside of any high humidity area and no entrained water droplets are anticipated. No electric heater was provided for the filtration and recirculation units since humidity control is unnecessary. No HEPA filters are provided downstream of the charcoal since radioactive fines carryover is very unlikely. This is true because the charcoal trays are pressure tested at high velocity in the manufacturer's shop prior to delivery, thereby removing fines. Also, during system operation, air is passing through the charcoal at a very low velocity.

3. All Engineered Safety Feature Filtration Units

Reference Paragraph C.2.b of the Regulatory Guide. No physical separation was provided for each filtration unit since there are no units located in areas where missiles are postulated.

As discussed above, each engineered safety feature (ESF) air filtration system is designed to safely and reliably mitigate the consequences of postulated accidents and includes the majority of the recommendations of Regulatory Guide 1.52, Revision 0. The testing of these systems is performed in accordance with the Plant Technical Specifications. Through license amendments adopted since the Technical Specifications were originally issued, surveillance testing associated with the ESF filtration systems is now conducted in accordance with the recommendations of Regulatory Guide 1.52, Revision 3 as discussed below.

Reference Paragraphs C.6.3 and C.6.4 of the Regulatory Guide. These paragraphs recommend an in-place leak test removal efficiency of 99.95 percent for HEPA filters and charcoal adsorbers in filter systems. As noted above, design and construction were completed prior to issuance of this guide, and conformance with this guide is not complete. Therefore, the in-place leak tests on HEPA filters and charcoal adsorbers are performed to demonstrate a 99.5 percent removal efficiency. These efficiencies were reviewed and approved by the NRC in License Amendment No. 46 for Unit 1 and License Amendment No. 37 for Unit 2.

Regulatory Guide 1.52, Revision 3 references ASME N510-1989 for testing air cleaning systems for Nuclear Power Plants. FNP has adopted ASME N510-1989 with errata dated January 1991 as the appropriate standard for guidance in testing ESF filtration units. The details of testing conformance to ASME N510-1989 are documented in FSAR tables 9.4-15 through 9.4-18. As stated in the NRC SER dated May 1, 1997, Farley Nuclear Plant is not required to perform all the acceptance tests as identified in ASME N510-1989. These type tests will be performed after major modification or major repair to the systems as identified in tables 9.4-15 through 9.4-18. Inspections following system modification or repair will be those inspections required on only those components affected by the modification or repair and not the complete system.

Post-Accident Containment Venting Filter Unit

The post-accident containment venting filter unit is not an ESF filtration system. This system was designed prior to issuance of Regulatory Guide 1.52, Rev. 0, therefore, conformance with this guide is not complete. Because of the time of the design and the system not being an ESF filtration system, the post-accident containment venting filter unit has unique design features which do not allow strict application of the design, maintenance, and testing requirements of this Regulatory Guide. Maintenance and periodic testing will be provided with guidance from ASME N510-1989 (see Section 6.2.5.4.2 for description of testing program). Design conformance is summarized as follows:

Reference Paragraph C.2.a of Regulatory Guide 1.52, Rev. 0. The post-accident containment venting filter unit is provided with an iodine adsorber and a HEPA filter downstream of the adsorber. The system components and arrangement are described in FSAR Section 6.2.5, and shown on drawings D-175019 and D-205019. No credit was taken for the system as an ESF filtration system in DBA analyses; therefore, redundant filter units are not installed. No demister was provided because the unit is located outside the containment and no entrained water droplets are anticipated. No prefilters or HEPA filters before the adsorbers are provided since this is a standby system and, if utilized, it is expected to have low operating time. No fan was provided since motive force is provided by containment pressure. No electric heater was provided since humidity is not to be controlled.

Regulatory Guide 1.53 - APPLICATION OF THE SINGLE FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS (Rev. 0, 6/73)

CONFORMANCE

The principles described in IEEE Trial Use Guide 379-72 were used in the design of the Westinghouse protection system. Although this guide had not been issued at the time of the design of the Farley Plant, the system does comply with the intent of this guide and the additional requirements of Regulatory Guide 1.53. The formal analyses required by the trial use guide have not been documented exactly as outlined although parts of such analyses are published in various documents, such as references 1, 2, and 3. Failure analysis results are given in tables 7.3-5 through 7.3-15. Failure analysis of the plant cooling water system is given in subsection 9.2.1.

1. W. C. Gangloff, "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486, May 1971.

2. W. C. Gangloff and W. D. Loftus, "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706, July 1971.

3. "Anticipated Transients Without Reactor Trip in Westinghouse Pressurized Water Reactors," WCAP-8096 April 1973.

Regulatory Guide 1.54 - QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER COOLED NUCLEAR POWER PLANTS (Rev. 0, 6/73)

CONFORMANCE

Regulatory Guide 1.54 (June 1973) and related ANSI Standard N101.4 (November 1972) postdate the construction permit for the Farley Nuclear Plant, which was issued in August 1972. Consequently, these requirements were not available for application to the nuclear steam supply system (NSSS) equipment for the FNP.

For Westinghouse scope of supply equipment, however, a process specification was applied to the NSSS equipment. This required that protective coatings for use on system components in the reactor containment be demonstrated to withstand the design basis accident conditions and meet all the criteria given in ANSI Proposed Standard N-101.2-1971, "Protective Coatings (Paints) For Light Water Nuclear Reactor Containment Facilities."

Regulatory Guide 1.54 and ANSI N101.4-1972 postdate the construction permit. Therefore, specifications and procedures relative to coatings on Category I structures did not reference the ANSI Standard. However, specifications and quality procedures used to control the application processes for these structures are such that they ensured proper application of these coatings. (See subsection 3.8.1.6.6.)

The protective coating systems specified for Seismic Category I structures are discussed in subsection 3.8.1.6.6, Interior Coating Systems.

Regulatory Guide 1.55 - CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (Rev. 0, 6/73)

CONFORMANCE

[HISTORICAL] *[Concrete placement in Seismic Category I structures was in accordance with Regulatory Guide 1.55, "Concrete Placement in Category I Structures," except as discussed below:*

1. *Regulatory Guide 1.55, Appendix A, Reference 11, "ACI/ASME Proposed Standard-Code for Concrete Reactor Vessels and Containments," and Reference 12, "ANSI N45.2.5-1972 (proposed) Supplementary QA Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," were not used since they had not received final approval by their sponsoring organizations.*

2. *Creep tests for concrete were performed for the containment structure only. Loss of prestress through creep was not applicable to non-prestressed structures.*

Concrete placement and testing are discussed in subsection 3.8.1.6.1, Reinforced Concrete.]

The requirements for concrete placement in Category I structures applicable to operation-phase activities are contained in ASME NQA-1-1994, as described in the SNC Quality Assurance Topical Report (QATR).

Regulatory Guide 1.56 - MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (Rev. 0, 6/73)

CONFORMANCE

Regulatory Guide 1.56 is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.57 - DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (Rev. 0, 6/73)

CONFORMANCE

Regulatory Guide 1.57 is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.58 - QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL (Rev. 1, 9/80)

CONFORMANCE

[HISTORICAL] [Personnel involved with examination of items on the site are qualified and certified in accordance with the requirements of ANSI N45.2.6-1978. Personnel performing inspection and testing were qualified for those specific tasks on the basis of experience and specific training (education or on-the-job-training or a combination of both). Compliance was verified by periodic audits by quality assurance personnel. However, two exceptions have been taken to Regulatory Guide 1.58, Revision 1, and ANSI N45.2.6-1978. A description and justification of these exceptions is presented below.

Regulatory Position C.2 of Regulatory Guide 1.58, Revision 1, endorses the 1975 edition of SNT-TC-1A as acceptable guidance for qualifications of nondestructive examination (NDE) personnel. In lieu of this, the version of SNT-TC-1A or other similar document used for qualifying personnel to perform nondestructive inspection, examination, or testing shall be in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where specific written relief has been granted by the NRC. The document used for the qualification of such personnel shall be specifically identified in the Inservice Inspection Program for FNP. In addition, FNP shall supplement these requirements by replacing the "shoulds" contained in SNT-TC-1A with "shalls" where they occurred in the 1975 version. A change to this paragraph shall be treated as a change to the FNP QAP in accordance with NRC SER dated March 17, 1998.

Subsection 2.3 of ANSI N45.2.6-1978 requires that the job performance of inspection, examination, and testing personnel be reevaluated at least every 3 years and that any person who has not performed inspection, examination, and testing activities in his qualified area for a period of 1 year shall undergo requalification in accordance with subsection 2.2. Inspection, examination, and testing activities are inherently integrated into the FNP staff's routine job responsibilities such that when a person holds a given position, he routinely performs those inspection, examination, and testing activities for which that position is responsible. Therefore, an annual demonstration of proficiency has no meaning under the FNP Quality Control Program. Moreover, the requalification per ANSI N45.2.6-1978, paragraph 2.2, is based on the individual's education and experience. Since each person's initial certification was also based on that individual's accumulated education and experience, and since an individual's accumulated education and experience cannot be revoked, there is no purpose in performing the requalification exercise simply because an individual did not perform or document an annual demonstration of proficiency in the inspection, examination, and testing activities for which he is certified (assuming, of course, that the individual's job performance of inspection, examination, and testing activities remained satisfactory). For these reasons an annual demonstration of proficiency in inspection, examination, and testing activities is fruitless and exception is taken to this aspect of ANSI N45.2.6-1978. The present practice, which is expected to be continued, of conducting job performance evaluations as a basis for recertification for inspection, examination, and testing personnel on a 2-year cycle satisfies the ANSI N45.2.6-1978 requirement that these evaluations be conducted at periodic intervals not to exceed 3 years.

On September 9, 1996, the NRC amended its regulations to incorporate by reference the 1992 edition with the 1992 addenda of Subsections IWE and IWL of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code with specified limitations in 10 CFR 50.55a. The new rules require certain containment liner and concrete inspections/examinations to be performed prior to September 9, 2001 and to be repeated on a regular basis thereafter. Containment repair and replacement requirements of the new rules including preservice examinations after repair or replacement were effective on September 9, *1996. Relief from this effective date until March 15, 1997 was requested by FNP. The 1992 edition with 1992 addenda of Section XI requires personnel performing NDE examinations to be qualified and certified using a written practice prepared in accordance with ANSI/ASNT CP-189. However, current certification based on SNT-TC-1A remains valid until recertification is required.]*

The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 which incorporates the requirements of ANSI N45.2.6. Accordingly, the requirements for qualification of inspection, examination, and testing personnel are described in the QATR.

Regulatory Guide 1.59 - DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS (Rev. 0, 8/73)

CONFORMANCE

The design of Category I structures for the protection of safety-related equipment from external flooding as discussed in subsection 3.4, Water Level (Flood) Design Criteria, complies with Regulatory Guide 1.59.

The material in the FSAR complies with Regulatory Position No. 1 of Regulatory Guide 1.59 as detailed in Appendix A and explained below.

A.1 Introduction

No comment.

A.2 Probable Maximum Flood (PMF)

Flood design is addressed in subsections 2.4.2.2 and 2.4.3.4.

A.3 Hydrologic Characteristics

A topographic map of drainage basin showing sub-basins and isohyetal pattern of PMP is shown on figures 2.4-3 and 2.4-12. A list of upstream river control structures is shown in figure 2.4-14. The historical flood profiles for 1929 and 1961 floods are shown on figure 2.4-10. Major storms and resulting floods were considered by the Corps of Engineers in the study for the spillway design flood for Walter F. George Project.

A.4 Flood Hydrograph Analysis

Analysis of observed hydrographs was done by the Corps of Engineers for Walter F. George Project. The results of this study are summarized in subsections 2.4.3.2 and 2.4.3.3.

A.5 Precipitation Losses and Base Flow

Precipitation losses are given in subsection 2.4.3.2. A base flow of approximately 2.5 $ft^3/s/mi^2$ drainage area was added to obtain the total discharge hydrograph from each of the inflow areas.

A.6 Runoff Model

The analysis of rainfall runoff records was done by the Corps of Engineers for the Walter F. George Project. The runoff model used for FNP is given in subsection 2.4.3.3.

A.7 Probable Maximum Precipitation Estimate

The results of the PMP estimates are given in subsection 2.4.3.1. The adjustment factors are shown on figure 2.4-2 and the isohyetal map used in study is shown on figure 2.4-3.

A.9 PMF Hydrograph Estimates

The antecedent reservoir level for Walter F. George Dam was taken at elevation 185, and the induced surcharge envelope shown on figure 2.4-65 was used in routing by the dam. As the highest power pool during summer months is at elevation 190, a routing of PMF was made using this initial pool elevation. The peak flood level at the FNP was elevation 144.3, 0.1 ft higher than the elevation used in chapter 2.

In this report, no antecedent storm before PMF was considered. A review of the flow record of four of the highest floods indicates that an average flow of 50,000 ft^3/s for the fifth day after the peak would be a reasonable flow to apply to peak PMF flow as the effect of an antecedent storm. This would increase the peak flow of PMF from 642,000 ft^3/s to 692,000 ft^3/s and raise the peak water surface elevation from 144.2 to 145.9, or 1.7 ft.

Regulatory Guide 1.59, paragraph A.12, states that a 40-mph wind would be an acceptable postulate. However, as stated in subsection 2.4.3.6, a 50-mph wind was used for wind wave activity. By using a 40-mph wind rather than 50-mph wind, a reduction of the runup of 1.9 ft could be made. The net change resulting from a peak flow of 692,000 ft^3/s and 40- mph wind would be a reduction of peak water surface elevation of about 0.2 ft.

A.10 Seismically Induced Floods

This study is given in subsections 2.4.4.1, 2.4.4.2, 2.4.4.3, and 2.4.4.4.

A.11 Water Level Determinations

The stage discharge relation at the site, as shown in figure 2.4-11, was determined by using the Corps of Engineers program HEC-2 as stated in subsection 2.4.3.5.

A.12 Coincident Wind Wave Activity

The studies made for wind waves in the river are stated in subsection 2.4.3.6. As stated under A.9 above, a 50-mph wind was used in the report when a 40-mph wind would be acceptable.

Regulatory Guide 1.60 - DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (Rev. 1, 12/73)

CONFORMANCE

Regulatory Guide 1.60 is intended to apply to nuclear power plants docketed after April 1, 1973; consequently, it was not applicable to the Farley plant.

The design response spectra for seismic design for Seismic Category I structures are discussed in subsection 3.7.1.1, Design Response Spectra, and subsection 3.7.1.2, Design Response Spectra Derivation.

Regulatory Guide 1.61 - DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (Rev. 0, 10/73)

CONFORMANCE

Regulatory Guide 1.61 is intended to apply to nuclear power plants docketed after April 1, 1973; consequently, it was not considered applicable to the Farley Plant.

The damping values for seismic design for Seismic Category I structures are discussed in paragraph 3.7.1.3, Critical Damping Values. However, as documented in table 3.7-1, Regulatory Guide 1.61 damping values are applied in the analysis of the reactor vessel head assembly structure. These values are endorsed by the NRC in the Standard Review Plan (NUREG-0800).

Regulatory Guide 1.62 - MANUAL INITIATION OF PROTECTION ACTIONS (Rev. 0, 10/73)

CONFORMANCE

The protection system for the Farley Nuclear Plant meets the intent of IEEE-279-71 as discussed in subsection 7.1.2.1. Regulatory Guide 1.62 presents an acceptable method for complying with the requirements of subsection 4.17 of IEEE-279- 71. The protection system does not, however, fully comply with Item 1 of Regulatory Guide 1.62. There are six manual, steam line isolation switches in the control room.

The switchover from injection to recirculation is performed at the component level following an accident when the refueling water storage tank (RWST) low level alarm is sounded.

Since there are two trains of safeguards, this system also has the ability to accept a single failure.

The Protection System complies with all other portions of Regulatory Guide 1.62.

Regulatory Guide 1.63 - ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR WATER COOLED NUCLEAR POWER PLANT (Rev. 0, 10/73)

CONFORMANCE

Each penetration assembly was designed to withstand, without loss of assembly integrity, the maximum short circuit current for a duration compatible with Insulated Power Cable Engineering Association (IPCEA) standards for the size of the applicable conductor. In addition, the circuits associated with the penetration assemblies are provided with overcurrent protection. Only Unit 2 power and control electrical containment penetrations are provided with primary and backup overcurrent protection to meet the single failure criteria. Those Unit 2 power and control electrical penetrations that are de-energized during normal plant operations and can be energized only under administrative control are excluded from the requirements of dual overcurrent protection to meet the single failure criteria.

Each penetration assembly is designed to withstand the maximum containment internal pressure of Item 2 under the Regulatory Position.

The penetration assemblies were installed, inspected, and tested in accordance with subsection 8.3.1.3, and particularly in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, Class MC vessels. In addition, leak tests and electrical tests to verify conductor continuity and insulation resistance were performed at the site.

Regulatory Guide 1.64 - QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS (Rev. 0, 10/73)[a]

CONFORMANCE

[HISTORICAL] *[The plant was designed with appropriate quality assurance provisions to meet the requirements of Appendix B to 10 CFR Part 50. This subject is discussed in Chapter 17.*

Regulatory Guide 1.64, dated October 1973, provides NRC endorsement of ANSI N45.2.11. The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 which incorporates the requirements of ANSI N45.2.11. Accordingly, quality assurance requirements applicable to design activities are described in the SNC QATR.

[HISTORICAL] *[a. Regulatory Guide 1.64, Rev. 0 applies to the design and construction phase of FNP. Additional Quality Assurance compliance with Regulatory Guide 1.64, Rev. 1 is given in Chapter 17.2.]*

Regulatory Guide 1.65 - MATERIAL AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS (Rev. 0, 10/73)

CONFORMANCE

Regulatory Guide 1.65 was published after procurement of the Farley Units 1 and 2 reactor vessel bolting material. However, this material meets the intent of the guide as follows.

All the reactor vessel closure stud bolts, nuts, and washers for the Farley Unit 1 Plant were fabricated from a single heat of SA540 Grade B24 Material.

The reactor vessel closure bolts for Unit 2 were machined from bars of SA540 Grade B24 material. The Unit 2 closure nuts and washers were machined from tubes of SA540 Grade B23 material.

The bolting material qualification tests were performed per the ASME Section III Code and Addenda (1970 Summer Addenda for both units) which required meeting an average of 35 ft-lb energy with no lateral expansion tests required and no maximum tensile strength limitation. All bolting material for both units met these ASME Code requirements.

Charpy tests were performed at 10°F on bolting material bar and tube specimens (three impact tests per bar or tube end tested) as required by the ASME Code referred to above. With the exception of one end of a single bar, all impact tests of the Unit 1 reactor vessel closure stud bar and tube material showed data equal to or greater than 45 ft-lb conforms with the guide position on Charpy impact energy. The single exception had three impact energy values of 39, 40, and 40 ft-lb on one end of the bar and 41, 48, and 50 ft-lb on the opposite end.

For the Unit 2 reactor vessel closure stud material, the required three impact tests on each end of each bar and tube that was tested showed Charpy impact energy values that ranged from a low of 39, 40, and 40 ft-lb to a high of 48, 50 and 51 ft-lb. Three of the bars and one of the tubes that were tested showed Charpy impact values that were less than the 45 ft-lb proposed by the guide.

For the bars and tubes showing 10°F impact data averaging below 45 ft-lb, Alabama Power Company believes that the intent of the guide is met, inasmuch as sufficient fracture toughness is obtained at the preload temperature or at the lowest service temperature (as specified by the Guide's reference to Appendix G to 10 CFR 50, Paragraph IV.A.4), both of which are significantly above the 10°F Charpy test temperature. Also, the impact energies at the preload or lowest service temperatures are higher than was obtained at the lower, actual Charpy test temperature.

For the bar material from which reactor vessel closure studs were made for Unit I, one end of one bar that was tested showed an ultimate tensile strength of 178,000 psi. Ultimate tensile strengths ranging from 157,000 psi to 170,000 psi were obtained for all the other bars, and the tubes that were tested, conforming with the guide position.

For Unit 2, the corresponding data showed a tensile strength of 178,000 psi reported for the end of one bar. Ultimate tensile strengths ranging from 155,000 psi to 167,500 psi were reported for all the other bars and tubes tested.

Alabama Power Company believes that the bar material tensile datum in excess of 170,000 psi meets the intent of the Guide, since Reference 2 of the Guide shows that the yield strength should be below 170,000 psi; the guide position concerning a maximum ultimate tensile strength should be based on the yield strength. Units 1 and 2 bar and tube bolting material yield strength data were below 170,000 psi.

The closure stud bolting material was procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. This strength level is compatible with Appendix G to 10 CFR 50 (July 1973, Paragraph I.C), although higher strength bolting materials are permitted in the ASME Code. Stress corrosion has not been observed in reactor vessel closure stud bolting manufactured from material of this strength level. Accelerated stress corrosion test data do exist for materials of 170,000 psi minimum yield strength exposed to marine water environments stressed to 75 percent of the yield strength (given in Reference 2 of the Guide). That these data show much more severe stress corrosion than is applicable because of the specified yield strength differences and the less severe environment to which Farley-reactor-vessel-closure-stud bolting is exposed is demonstrated by many reactor-years of satisfactory experience at nuclear power plants.

The reactor-vessel-stud-bolt materials for both plants were inspected to the requirements of the ASME Code Section III (through summer 1970) and meet the intent of Regulatory Guide C. 2 "Inspection."

Ultrasonic inspection of the Farley stud material was examined radially based upon the first back reflection from an indication-free area of the basic bar stock for each stud and of each bar from which the nuts and washers are machined per ASME Specification SA-388. The indication-free section is selected for calibration by a preliminary scan. Calibration for the axial examination conducted from the end faces of the studs and the bars for nuts and washers was established on a calibration block or standard per ASME Specification SA-388. The axial testing calibration block was manufactured from a representative, indication-free section of the stud, nut, and washer bar stock, based on a preliminary scan conducted for that purpose.

Magnetic particle examination of all exterior closure stud surfaces was conducted after threading, using continuous circular and longitudinal magnetization. The Farley Units reactor vessel design and Westinghouse practice and recommendations meet the recommendations of Regulatory Guide Positions 3, "Protection Against Corrosion," and 4, "Inservice Inspection."

Regulatory Guide 1.66 - NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS (Rev. 0, 10/73)

CONFORMANCE

Regulatory Guide 1.66 was published after procurement of tubular products for Farley Nuclear Plant Units 1 and 2. In addition, the licensee takes exception to Regulatory Guide 1.66. The Guide position concerning defect detection capability is impractical and the axial testing recommendations to meet the Guide position are technically unnecessary.

The Guide states that, "Nondestructive examination applied to tubular products used for components of the reactor coolant pressure boundary and other safety related systems. . . should be capable of detecting unacceptable defects regardless of defect shape, orientation, or location in the product." This is impractical to attain.

In addition, the guide position regarding ultrasonic angle beam scanning in the axial direction is technically unnecessary, since any flaws that developed from the processes employed in tubular product manufacture are invariably oriented in the axial direction. Any circumferential or transverse flaws that developed were mechanically induced surface defects detected by normal QC procedures. However, the nondestructive examinations performed on tubular products covered by the Guide (reactor vessel nozzles, control rod drive mechanism (CRDM) housings, core support columns) met the purposes of the Guide.

The tubular products used in the Farley units' equipment and systems, as listed above, were fabricated and inspected to high quality standards, suitable for nuclear equipment as were required by the applicable contemporary codes and standards. (See table 3.2-1.)

The examinations were performed to higher sensitivity levels than required by the codes, and included 100 percent volumetric nondestructive examination.

For the reactor vessel nozzle forgings, the ultrasonic examination included end to end axial testing, from the end faces, and angle beam testing in two circumferential directions. The angle beam testing was repeated to the fullest extent possible after machining. After heat treatment and prior to cladding, magnetic particle examinations were conducted over all nozzle surfaces.

The CRDM housings were ultrasonically angle beam tested in two circumferential directions, axially tested from the end faces, and radially tested from the circumferential surfaces of the bars or the tubular product machined from the bars.

The core support structure tubular products were ultrasonically tested axially from the end faces, and radially tested from the circumferential surfaces.

Regulatory Guide 1.67 - INSTALLATION OF OVERPRESSURE PROTECTION DEVICES (Rev. 0, 10/73)

CONFORMANCE[(a)]

All structures, systems, and components of overpressure protection systems important to safety, including the main steam safety and relief valves and associated piping and valve headers, were designed, analyzed, and qualified in accordance with the recommendations of this Guide.

Details for the pressurizer overpressure protection system are addressed in subsection 5.2.2.

a. The Winter 1978 Addenda to the 1977 Edition of the ASME Boiler and Pressure Vessel Code, Appendix O, Section III, Division 1, "Rules for Design of Safety Valve Installations," included requirements equivalent to the recommendations of Regulatory Guide 1.67. These changes to the Code were incorporated by reference to 10 CFR 50.55a on April 3, 1981. Subsequently, the NRC withdrew Regulatory Guide 1.67 on April 15, 1983.

However, the withdrawal of this Regulatory Guide does not alter any prior or existing licensing commitments based on its use. As noted above, Farley Nuclear Plant meets the recommendations of Regulatory Guide 1.67.

Regulatory Guide 1.68 - PREOPERATIONAL AND INITIAL STARTUP TEST PROGRAM FOR WATER COOLED POWER REACTORS (Rev. 0, 11/73)

CONFORMANCE

The conformance with Regulatory Guide 1.68 is discussed in chapter 14.

The standard Westinghouse nuclear steam supply system contains online analog protective circuits designed to provide continuous online protection against excessive power density and DNB. The plant process computer is not a part of this standard system but was purchased as an option. The plant process computer does not perform any safety-related function nor is it required for the operation of the plant; therefore, Alabama Power Company takes exception to Item D.1.r of Appendix A in Regulatory Guide 1.68.

Regulatory Guide 1.69 - CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS (Rev. 0, 12/73)

CONFORMANCE

This topic is addressed in subsection 12.1.2.1.

Regulatory Guide 1.70 - STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS (Rev. 1, 10/72)

CONFORMANCE

The FSAR for the Farley Nuclear Plant was prepared in accordance with Revision 1, October 1972. Recommendations for additional information that have been issued as new regulatory guides with numbers in the form 1.70.X dated up to and including an issue of August 1974 are addressed in this appendix. Some plant project drawings are included in the FSAR by reference to the drawing identification number (e.g., D-177024) in lieu of inclusion in the FSAR as a figure. Some pages and figures in the FSAR are referenced by the date of change or revision number or both in the lower right-hand corner per 10 CFR 50.71(e)(5).

Regulatory Guide 1.70.1 - ADDITIONAL INFORMATION - HYDROLOGICAL CONSIDERATIONS FOR NUCLEAR POWER PLANTS (Rev. 0, 12/73)

CONFORMANCE

Water quality is addressed in subsection 2.4.13.5.

The design bases for groundwater-induced hydrostatic loadings on safety-related structures are addressed in subsections 2B.4.4 and 2B.7.1 and section 2B.6. The design for hydrostatic loadings is based either on the normal groundwater level or on the design flood level, whichever is more critical for the particular structure. Dewatering during construction is not critical to the integrity of safety-related structures.

Groundwater conditions are addressed in subsection 2.5.4.6

A history of groundwater fluctuations beneath the site is provided in subsection 2.4.13.2.2. The water levels in the piezometers are shown on figures 2.4-26 through 2.4-60. Discussions of groundwater conditions during and after construction of the plant are included in subsections 2.4.13.1.3, 2.4.13.2.5, and 2B.4.4.

Regulatory Guide 1.70.2 - ADDITIONAL INFORMATION - AIR FILTRATION SYSTEMS AND CONTAINMENT SUMPS FOR NUCLEAR POWER PLANTS (Rev. 0, 12/73)

CONFORMANCE

B.(1) The analyses of the engineered safety features air filtration systems with respect to Regulatory Guide 1.52 are provided in the following sections:

a. Fuel handling building - subsection 9.4.2.2.2.

b. Control room - subsection 9.4.1.2.

c. Penetration room - subsection 6.2.3.2.2.

B.(2) The information requested dealing with the containment sumps and sump intake screens is provided in subsections 6.2.2.2.1 and 6.2.2.3.1.

Regulatory Guide 1.70.3 - ADDITIONAL INFORMATION - RADIOACTIVE MATERIALS SAFETY FOR NUCLEAR POWER PLANTS (Rev. 0, 2/74)

CONFORMANCE

The additional information described in the Regulatory Guide is provided in section 12.4.

Regulatory Guide 1.70.4 - ADDITIONAL INFORMATION - FIRE PROTECTION CONSIDERATIONS FOR NUCLEAR POWER PLANTS (February 1974)

CONFORMANCE

Several specific portions of this regulatory guide spell out information to be provided in FSARs. All required information is provided in the Fire Protection Program Reevaluation.

Regulatory Guide 1.70.5 - ADDITIONAL INFORMATION - WATER LEVEL (FLOOD) DESIGN FOR NUCLEAR POWER PLANTS (Rev. 0, 5/74)

CONFORMANCE

The additional information requested in the Regulatory Guide is provided in subsection 3.4.

Regulatory Guide 1.70.6 - ADDITIONAL INFORMATION QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION (Rev. 0, 6/74)

CONFORMANCE

Conformance with sections of Appendix B to 10 CFR Part 50, and hence with this Regulatory Guide, is discussed in subsection 17.1.1.

Regulatory Guide 1.70.7 - ADDITIONAL INFORMATION, GEOGRAPHY AND DEMOGRAPHY CONSIDERATIONS FOR NUCLEAR POWER PLANTS (Rev. 0, 8/74)

CONFORMANCE

Sufficient information is presented in section 2.1 to respond to the considerations of the Regulatory Guide, although the format is based on Revision 1 of the Standard Format and Content of Safety Analysis Report for Nuclear Power Plants.

Regulatory Guide 1.71 - WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY (Rev. 0, 12/73)

CONFORMANCE

The recommendation of the Regulatory Guide for limited accessibility qualification or requalification, in addition to ASME Section III and IX requirements, is an unduly restrictive requirement for shop fabrication, where the welder's physical position relative to the welds is controlled and does not present any significant problems. In addition, shop welds of limited accessibility were repetitive due to multiple production of similar components, and such welding was closely supervised.

Field welding procedures and personnel were qualified in accordance with the requirements of ASME Section IX. As far as is practicable, welds were located to provide physical and visual accessibility for welding and inspection. In the event that a weld must be located in an unfavorable position, considerations were given to the preparation of a mockup simulating the production weld, using welders qualified to Section IX.

The above practices with associated quality control will meet the intent of the Regulatory Guide.

Regulatory Guide 1.72 - SPRAY POND PLASTIC PIPING (12/73)

CONFORMANCE

Regulatory Guide 1.72 is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.73 - QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS (Rev. 0, 2/74)

CONFORMANCE

Westinghouse safety-related-active-valve-motor operators used inside containment are environmentally qualified by having passed a comprehensive testing program. The testing included heat, live steam, heat aging, shock and vibration, cycle life tests, radiation, and postaccident steam and chemical spray testing. Although the test sequences employed and documentation of the tests vary somewhat from requirements issued in 1974, these factors do not detract from the intent of the qualification, i.e., to provide assurance of operability under accident conditions.

Test results for Westinghouse supplied equipment are addressed in section 3.11.

Regulatory Guide 1.74 - QUALITY ASSURANCE TERMS AND DEFINITIONS (Rev. 0, 2/74)

CONFORMANCE

[HISTORICAL] *[The NRC Regulatory Guide 1.74 identifies forms and acceptable definitions that are important to the understanding of quality assurance requirements for the design, construction, and operation of nuclear power plant structures, systems, and components.*

Terms and definitions contained in Quality Assurance Procedures, References, or Reports are in accordance with ANSI N45.2.10-1973, an acceptable standard for use in describing and implementing quality assurance programs, as described in subsection 17.2.1.]

The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 which incorporates the requirements of ANSI N45.2.10. Accordingly, terms and definitions used in the quality assurance program are provided in the SNC QATR.

Regulatory Guide 1.75 - PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS (Rev. 0, 2/74)

CONFORMANCE

The extent of compliance of Farley design with the guide is given below:

Separation is provided to maintain independence between redundant safety-related circuits so that a failure in one circuit does not jeopardize the protective function of other safety-related circuits during and following any design basis event. Equipment and circuits requiring separation have the safety train designation assigned as part of the equipment and/or scheme cable number.

The Farley design is in compliance with the Regulatory Guide recommendations except for the separation recommendations between associated circuits. The "associated circuits" as defined in the guide, though uniquely identified, were not designed to meet the separation requirements of Class 1E circuits. On Farley, a non-Class IE circuit associated with Class 1E cables and equipment is assigned either an "X" or a "Y" scheme cable number in accordance with subsection 8.3.1.5. The routing procedures ensure that a non-Class 1E circuit routed with Class 1E circuits of one safety train will not be routed with a Class 1E circuit of the opposite train. The "X" or "Y" circuits, however, may run together in a non-Class 1E circuit raceway. Also, cables with "X" and "Y" scheme cable numbers are permitted to enter non-Class 1E equipment provided that they do not form an electrically continuous circuit within the equipment. Any exceptions are reviewed for acceptability on a case-by-case basis.

"Associated cables" of one train entering Class 1E equipment of the opposite train are run separately from other opposite train cables. In this way, a failure of a circuit does not defeat the protective function requirements of redundant Class 1E circuits. The cables used at Farley are of flame-retardant construction and have been manufactured to meet the quality assurance requirements of subsection 17B.1.2. In addition, power cables run in trays have interlocked armor that further prevents the propagation of fire.

The underlying philosophy of physical independence requirements for Farley is that fires are primarily caused by insulation failures due to overheating of cables. A fire in an "X" cable in an "A" train tray could affect the adjacent safety-related cables but only in the "A" train. Because of the flame- retardant qualities of cable insulation it is considered inconceivable that this fire would propagate to the non-Class 1E raceway carrying X and Y cables, then to an adjacent "Y" cable and further propagate via the "Y" cable to the "B" train safety-related cables and jeopardize both trains.

Specific areas of Farley design positions are given below:

A. Isolation Devices (Paragraph 3.8)

Interrupting devices actuated by fault current are isolation devices when justified by test or analysis.

B. Non-Class 1E Circuits (Paragraph 4.6)

Non-class 1E circuits are not separated by minimum separation distances, nor are non-class 1E circuits separated from associated circuits by minimum separation distances, nor are all non-class 1E circuits treated as associated circuits.

C. Routing of Class 1E Circuits (Paragraph 5.1.1.1)

Opposite sides of rooms or areas, if confined or otherwise incapable of dissipating heat from fires, are Class 1 separation if provided with fire protection or otherwise incapable of supporting combustion.

D. Cable Spreading Area and Main Control Room (Paragraph 5.1.3)

With regard to the requirement for a minimum of 1-in. separation between redundant Class 1E circuits and between Class 1E and non-Class 1E circuits, FNP requirements for cable routing meet the intent of this separation requirement as follows:

1. The outside edges of instrumentation and control conduits are separated by a minimum of 1/2 in. This separation is adequate since:

    a. These conduits contain low energy circuits which are provided with fuse/breaker interrupting devices that would preclude a fault in these cables from generating sufficient heat to ignite the cable insulation and jacketing material before being interrupted.

    b. All cables are IEEE 383 qualified which, even if faulted, will not sustain combustion.

2. For 600-V power circuits, spacing is no less than 1/4 of the largest conduit diameter. This separation is adequate since:

    a. The short-circuit coordination schemes preclude a fault in these cables from generating sufficient heat to ignite the cable insulation and jacketing material before being interrupted.

    b. IEEE 383 cables will not sustain fire propagation.

    Based on these separation requirements and the fault protection provided, an internally faulted cable would not produce sufficient heat energy to degrade cables in another conduit within the distances specified. Thus, these separation requirements meet the intent of this requirement.

E. General Plant Areas (Paragraph 5.1.4)

Solid enclosed raceways may be more detrimental than nonsolid raceways because of flue effects.

With regard to the requirement for a 1-in. separation between redundant Class 1E circuits and between Class 1E circuits and non-Class 1E circuits, the discussion provided above in response to paragraph 5.1.3 also applies to paragraph 5.1.4. Additionally, 4-kV power circuits are separated by a minimum of 1 conduit diameter spacing and all 4-kV power conduits are greater than 1-in. in diameter. This meets the 1-in. separation requirement for these cables.

F. Instrument Cabinets (Paragraph 5.7)

Separation requirements should not be the same for instrumentation racks and control boards because functional requirements are different. The IEEE draft criteria are adequate.

Regulatory Guide 1.76 - DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (Rev. 0, 7/74)

CONFORMANCE

The design basis tornado for the Farley Nuclear Plant is given in subsection 3.3.2, Tornado Loadings. The maximum rotational speed, maximum translational speed, and the rate of pressure drop differ somewhat from those specified in Regulatory Guide 1.76 as indicated below. However, the maximum windspeed, radius of maximum rotation, and the maximum pressure drop are in conformance with the Guide.

	Rotational Speed (mph)	Translational Speed (mph)	Rate of Pressure drop psi/s
Regulatory Guide 1.76	290	70	2.0
Farley	300	60	1.0

Regulatory Guide 1.77 - ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD ACCIDENT FOR PRESSURIZED WATER REACTORS (Rev. 0, 5/74)

CONFORMANCE

Methods for evaluating rod ejection accidents are in compliance with Regu1atory Guide 1.77. The methods employed to evaluate postulated rod ejection accidents are described in subsection 15.4.6.

Regulatory Guide 1.78 - EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (Rev. 1, 12/01)

CONFORMANCE

The design guidance and assumptions of Regulatory Guide 1.78 are used in the evaluation of control room habitability except as noted below:

a. Hazardous chemicals in the vicinity of the site are discussed in section 2.2.

b. Instead of release and transport models of paragraphs 3.2 and 3.3, the evaluation model used conforms to the guidance of NUREG-0570.

c. The design of the isolation system conforms to IEEE-279(1971) in lieu of IEEE-603 described in paragraph 4.2.

d. Analysis of a chlorine release from its storage locations onsite is discussed in section 2.2 and subsection 9.4.1. This discussion is provided for historical purposes only since all significant quantities of chlorine, i.e., single containers greater than 150 lb, have been removed from the plant site. Control of chlorine is in accordance with Regulatory Guide 1.95.

Regulatory Guide 1.79 - PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS (Rev. 0, 6/74)

CONFORMANCE

These tests were intended to evaluate the performance of the components of the emergency core cooling system (ECCS) to ensure that the ECCS accomplished its required function.

The preoperational test program covered the following tests:

1. Tests under ambient temperature conditions.

   The reactor vessel was open and flooded; thus the reactor coolant system (RCS) was essentially at atmospheric pressure.

2. Tests under hot operating conditions.

   The reactor vessel was closed and the reactor coolant system was at pressure and temperature conditions obtainable under preoperational testing.

3. ECCS component testing.

   1. Tests Under Ambient Temperature Conditions

      1.1 Integrated System Test

      The objective of this test was to ensure that the diesel generators had the capability to start and accelerate the engineered safety features (ESF) loads to rated speed without exceeding the specified safety limits.

      A safety injection signal concurrent with a loss of offsite power was simulated in one safety train, and it gave a signal to simultaneously start the diesel generators and shed the safety-related loads from the buses. After the diesel generators reached rated speed and voltage, the safety-related loads indicated below were automatically sequenced by the diesel generator sequencer. The response time from initiation of the safety injection signal to starting these loads was measured manually (with a stopwatch) and checked for acceptability.

      Safety-Related Loads

      a. High-head safety injection pump.
      b. Low-head safety injection (residual heat removal) pump.
      c. Component cooling water pump.
      d. Service water pumps.
      e. Auxiliary feedwater pump.

f. Containment spray pump.
g. Control room AC.
h. Containment coolers.
i. Battery charger.
j. River water pumps.[a]
k. Motor control centers associated with emergency power systems.
l. Safety-related valves.

For this test, all valves which, if operated, would have a detrimental effect on the subsequent commissioning of the plant, were blocked from operation. An example of this case was the accumulator isolation valve. Similarly, where full flow conditions could not be achieved, the pumps were operated on miniflow or on bypass. An example of the latter case was the containment spray pump.

The following tests were associated specifically with the ECCS:

1.2 High-Pressure Safety Injection Test – Flow Test

Fluid from the refueling water storage tank was injected into the open reactor vessel through various combinations of injection legs and pumps by operating the high head safety injection pumps. Flow was demonstrated to be within the design specifications. Response time data were obtained (manual measurement) for components under test to demonstrate that they met or exceeded the acceptance criteria.

1.3 Low-Pressure Safety Injection Test – Flow Test

The test under 1.2 above was repeated but with the operation of the residual heat removal pumps instead of the safety injection pumps.

1.4 Recirculation Test

This test, as required by paragraph C.3.b.(2), was not performed because, in view of the advanced stage of construction, substantial plant modifications would have been required to do the testing. This mode of operation was checked by analyses and operation of individual components, checked separately.

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a. Although river water pumps were originally tested as safety-related loads, they are not relied upon in any analysis and are therefore not automatically sequenced by the diesel generator sequencer.

1.5 Core Flooding - Flow Test

Each accumulator was filled and pressurized with the motor-operated isolation valve closed. The accumulator discharge was initiated by opening the accumulator isolation valves with the RCS at reduced pressure. Discharge flowrate was calculated from the change of accumulator pressure with time.

2. <u>Test Under Hot Operating Conditions</u>

2.1 High-Pressure Safety Injection - Flow Test

The capability of the high-head safety injection (HHSI) pumps to deliver emergency core cooling water from the refueling water storage tank to the RCS was checked by analysis. The operation of the safety injection check valves was shown to be satisfactory by brief operation of the HHSI pumps.

During this test, flow of auxiliary feedwater to the main feedwater system was blocked to avoid temperature and pressure transients. The pumps were started and run on recirculation. Flow from the auxiliary feedwater pumps was verified as part of feedwater system tests.

2.2 The HHSI pumps were used to produce flow through the check valves in the HHSI system.

The operation (partially open) of the check valves was verified by recording the flow in each branch line using installed orifices.

3. <u>ECCS Component Testing</u>

3.1 Accumulator Isolation Valve Test

The operation of the accumulator isolation valves was tested along with Test 1.5 above. Since the isolation valve operators are supplied from motor control centers that receive power from either normal or emergency power source, the test was performed using normal power only. The valve operation was initiated by the simulation of a safety injection signal.

3.2 Testing of Valves and Pumps of ECCS

Routine periodic testing of ECCS components is performed at power as discussed in subsection 6.3.4. Valves are operated through a complete cycle and their operation observed in the control room. The response times are measured manually.

Operation of pumps and motors are routinely checked at power by operation on miniflow or bypass.

3.3 Initiating Instrumentation

Safeguard system logic tests were performed in accordance with subsection 7.3.2.

3.4 Testing of Onsite Power System During Refueling Shutdown

The objective is to check the integrity of the onsite power system to start the safety related loads. Each safety train is tested in two steps so that the normal process of plant shutdown is not affected.

Step 1: A safety injection signal concurrent with a loss of offsite power is simulated in one safety train at the sequencer, which eventually loads the pump motors on the diesel generators in the manner of test 1.1. The safety injection signal is not transmitted to the safety-related valves, but the valves are positioned to ensure the normal shutdown procedure. The following pumps are operated on bypass: HHSI pumps and the containment spray pumps. This test provides a means of ensuring that the starting and operating of pumps and their response times (measured manually) are within acceptable limits.

Step 2: With the pump motor breakers in the test position, a safety injection signal is initiated that operates the safety-related valves. Indication in the control room provides a check of valve operation without simulating flow conditions. The response times, determined manually, for these valves are also checked for acceptability. Since the sequencing of motor control center loads was checked in step 1, the operation of the valves is tested with the diesel generator operating from step 1.

Periodic testing requirements are addressed in the Technical Specifications and the Technical Requirements Manual.

3.5 System Piping and Supports

The acceptability of system piping movements is discussed in subsection 3.9.1.

Regulatory Guide 1.80 - PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS (Rev. 0, 6/74)

CONFORMANCE

The Regulatory Guide addresses safety-related instrument air systems and thus is not applicable to the Farley instrument air system.

The acceptance test for the instrument air system generally included the recommendations of subsections CI through C7 of the Guide.

Operational tests of those air operated safety related valves required to assume the safe operating position upon 1oss of instrument air were included on an individual basis in the preoperational tests of the systems in which the valves are located.

Regulatory Guide 1.81 - SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTIUNIT NUCLEAR POWER PLANTS (6/74)

CONFORMANCE

Paragraph C.1

Each of the two units is provided with separate and redundant dc electrical systems. The sharing of the dc electrical systems is limited to control power requirements of components in the service water intake structure and diesel generators 1-2A, 1C, and 2C, which are shared between Unit 1 and Unit 2. The dc control power supplies to these diesels are mechanically interlocked so that only one source furnishes control power requirement at any time.[a] See drawings D-177082, D-177083, D-207082, and D-207083 for the dc distribution system for diesels. Details of the dc electrical system are discussed in subsection 8.3.2.

Paragraph C.2

The Onsite ac Power System is described in subsection 8.3.1, and, more specifically, the onsite emergency power system is described in 8.3.1.1.7. The details of conformance with this paragraph follow:

Item a. The sharing of onsite ac electrical systems is limited to two units.

Items b. and c. The sizing of diesels together with the control circuitry design is adequate considering a single failure capability to automatically supply the ESF loads for the accident unit and the safe shutdown requirements for the other unit. Details of diesel operation under various conditions are provided in paragraph 8.3.1.1.7 and take into consideration the most severe condition of a DBE on one unit and a failure of one diesel generator. A single failure includes a false or spurious accident signal in the non-accident unit.

Item d. The control circuits for shedding and loading the ESF loads are essentially separate in that each ESF 4160V bus is provided with its load sequencer. The interaction between control circuits for Unit 1 and 2 is limited to automatic starting signal from the 4160-V buses and instrumentation for the shared diesels. However, maintenance and testing of these starting signals in one unit will not prevent the diesel generator from supplying the minimum ESF loads on the other unit.

Item e. All diesel generators are controlled from the emergency power board common to both Units 1 and 2 and located in the control room. Coordination between unit operators is not necessary for meeting recommendations of Regulatory Positions 2b, 2c, and 2d.

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a. No common mode failures exist which could fail dc systems in both units.

Item f. Complete information in regard to the diesel generator, the load sequences, and the associated 4160-V breakers is displayed on the emergency power board in the control room

Item g. The design conforms to Regulatory Guides 1.6 and 1.9 as discussed in the appropriate areas of this appendix. Information regarding bypassed and inoperable systems is provided, although detailed conformance to Regulatory Guide 1.47 is not achieved. This is detailed in the discussion of Regulatory Guide 1.47 elsewhere in this appendix.

Paragraph C.3

This paragraph is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.82 - SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS (Rev. 3)

CONFORMANCE

Plant Farley complies with the regulatory positions on design criteria, performance standards, and analysis methods that relate to PWRs except, or as clarified in this section:

C.1.1.1.2: The sumps are located outside the missile barrier and are physically separated from each other by structural barriers to the extent practical. No additional protection from high-energy piping is required.

C.1.1.1.3: The sumps are located on the lowest possible floor elevation in the containment exclusive of the reactor vessel cavity. Each pump intake is protected by one strainer assembly. The sump strainer assemblies are not depressed below the floor. No trash rack is installed. Exception is based on:

- There are no high-energy line breaks postulated to occur near the strainers that would result in the required post-accident recirculation and there are no missiles generated in the vicinity of the strainer assemblies; therefore, there are no jet loads, no pipe whip restraint loads, and no missiles applicable to the strainer assemblies.

- The design of the stacked disk strainer prevents pieces of debris larger than 1.75 in. in diameter from reaching the perforated area due to the small slots between the strainer disks.

- The stress analysis results show the strainer assemblies can meet the design requirements of ASME Section III, subsections NC, ND, and NF. The loads used in the analyses include the pressure from debris, seismic loads and dead weight, and all other pertinent parameters.

Plant-specific transport analysis and testing has been performed.

C.1.1.1.4: The floor level in the vicinity of each coolant sump is sloped toward a drain through the missile barrier. In some cases the slope is away from the sump; in some cases it is towards the sump. Plant-specific transport analysis and testing has been performed. Therefore, floor slope and curb are not credited to limit debris transport.

C.1.1.1.6: The sumps are located outside the missile barrier and are physically separated from each other by structural barriers to the extent practical. No additional protection from high-energy piping is required. The strainers are designed to withstand loading for the largest postulated debris pieces and types resulting from a LOCA that requires post accident recirculation.

C.1.1.1.7: The strainers are designed to withstand loading for the largest postulated debris pieces and types. The stacked disc strainer design with perforated plate is self venting. A single solid cover is not practical.

C.1.1.1.12: Unit 2 has to replace the high-head safety injection throttle valves in order to comply with this position.

C.1.1.3: N/A

C.1.1.4: N/A

C.1.2: N/A

C.1.3.1.2: N/A

C.1.3.1.3: N/A

C.1.3.1.9: Minimum projected sump water levels for LBLOCA event which generates bounding debris head losses were used in applicable NPSH calculations. The NPSH calculations used a range of sump fluid temperatures between 120 ºF and 212 ºF. This temperature range covers the time period during the post-LOCA cooldown that results in the minimum post-LOCA containment sump NPSH margin.

C.1.3.2.1: Main steam and main feedwater line breaks were not evaluated since it is assumed that recirculation is not credited for these situations.

C.1.3.2.2: A zone of influence (ZOI) of 4 for acceptable coating has been used in the plant specific debris generation and transport analysis and testing. This is based on the guidance of WCAP-16568-P.

C.1.3.3.8: N/A

C.1.3.4.4: N/A

[HISTORICAL]
[CONFORMANCE (Prior to December 2007)

C. 1: The requirements of this paragraph are met. See appendix 6C for a further discussion.

C. 2: The sumps are located outside the missile barrier and are physically separated from each other by structural barriers to the extent practical. No additional protection from high-energy piping is required.

C. 3: The sumps are located on the lowest floor elevation in the containment exclusive of the reactor vessel cavity. Each sump intake is protected by two screens: an inner grating and a fine outer screen. The sump screens are not depressed below the floor elevation. Figure 6C-6 shows a containment sump intake and screen arrangement.

C. 4: *The floor level in the vicinity of each coolant sump is sloped toward a drain through the missile barrier. In some cases the slope is away from the sump; in some cases it is towards the sump.*

C. 5: *The requirements of this paragraph are met.*

C. 6: *An outer trash rack is provided.*

C. 7: *A vertically mounted fine screen is provided. The present effective sump screen height which was selected to ensure sump submergence during recirculation is such that a total effective sump peripheral length approximately 150 ft (corresponding to overall sump dimensions of approximately 40 ft x 40 ft, or equivalent) would be required to yield screen velocities of 0.2 ft/s for the sump serving both the ECCS and the containment spray systems, based on the assumptions of Regulatory Guide 1.82. Existing Farley containment layout precludes the location of sumps of these dimensions in protected locations. The present sumps were designed to yield low velocities of approach in the vicinity of the sump to promote the settling out of debris, and to yield negligible pressure drops through the sump screen. Materials inside containment that could cause sump screen blockage post-LOCA are eliminated or minimized by design. Present liquid velocities through the fine screen, based on the assumptions required by this guide, are between 1.16 and 2.17 ft/s*

C. 8: *A 3-ft-wide solid plate covers most of the top of the 5-ft-wide sump. (See figure 6C-6.) The top deck is designed to be fully submerged after a LOCA and completion of the safety injection.*

C. 9: *The recommendations of this paragraph are met.*

C.10: *The recommendations of this paragraph are met. Some nuclear fuel used at Farley Nuclear Plant may contain design features that provide for flow paths smaller than the inner containment sump screen. However, these flowpaths are evaluated as part of the fuel design and will provide adequate ECCS flow to ensure long term core cooling.*

C.11: *The recommendations of this paragraph are met, as there is a vortex breaker provided at each pump inlet. (See figure 6C-6.)*

C.12: *The recommendations of this paragraph are met by having the trash racks made of galvanized steel and the screens of stainless steel.*

C.13: *The recommendations of this paragraph are met by means of a removable 1/4-in. solid plate (see figure 6C-6) on the top of each sump.*

C.14: *The recommendations of this paragraph are met.]*

Regulatory Guide 1.83 - INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES (Rev. 1, 7/75)

CONFORMANCE

The Farley Nuclear Plant meets the intent of the Regulatory Guide. Surveillance requirements for the steam generator tubes are conducted in accordance with the Steam Generator Tube Surveillance Program, as required by the plant Technical Specifications.

Regulatory Guide 1.84 - CODE CASE ACCEPTABILITY ASME III DESIGN AND FABRICATION (Rev. 0, 6/74)

CONFORMANCE

Of the design and fabrication code cases used by the licensee, all were either annulled, adopted in later versions of the Code, or endorsed in the Regulatory Guide, except Code Case 1360. Tubes are not explosively welded to tube sheets in Code Class 1 components. However, in the absence of other references to explosive welding in Section III or Section XI of the ASME Code, this Code Case was cited as the basis for explosive tube plugging. This process was used only after installation of steam generators when they were found to have defective tubes.

ASME Code Case N-411 is used in the analysis of piping as accepted and endorsed in Rev. 28 of Regulatory Guide 1.84. See Section 3.7.1.3 for use of Code Case N-411.

Regulatory Guide 1.85 - CODE CASE ACCEPTABILITY ASME SECTION III MATERIAL (Rev. 0, 6/74)

CONFORMANCE

All the materials code cases that may have been used by the licensee were either annulled, adopted in later versions of the Code, or endorsed in the Regulatory Guide. Therefore, the Farley Nuclear Plant conforms to the Guide.

Regulatory Guide 1.86 - TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS (6/74)

CONFORMANCE

The termination of the operating license and the subsequent decommissioning of the Farley Nuclear Plant will be carried out in conformance with the regulations applicable at that time.

Regulatory Guide 1.87 - CONSTRUCTION CRITERIA FOR CLASS 1 COMPONENTS IN ELEVATED TEMPERATURE REACTORS (SUPPLEMENT TO ASME SECTION III CODE CASES 1592, 1593, 1594, 1595 AND 1596) (Rev. 0, June 1974)

CONFORMANCE

This guide is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.88 - COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS (Rev. 0, 8/74)

CONFORMANCE

[HISTORICAL] *[Compliance with ANSI N45.2.9-1974 provisions, which constitutes generally acceptable requirements for collection, storage, and maintenance of nuclear power plant quality assurance records, as stated in Regulatory Guide 1.88, is discussed in subsection 17.2.17.]*

The SNC Quality Assurance Topical Report (QATR) is based on ASME NQA-1-1994 which incorporates the requirements of ANSI N45.2.9. Accordingly, the requirements for collection, storage, and maintenance of quality assurance records are described in the QATR.

Regulatory Guide 1.95 - PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE (Rev. 0, 2/75)

CONFORMANCE

Chlorine storage locations onsite are discussed in section 2.2 and subsection 3.4.1. This discussion is provided for historical purposes only since all significant quantities of chlorine, i.e., single containers greater that 150 pounds, have been removed from the plant site. Control of chlorine is in accordance with Regulatory Guide 1.95.

Regulatory Guide 1.99, Rev.2 - RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS (MAY 1988)

CONFORMANCE

The methodology for determining the effect of neutron irradiation on the reactor vessel beltline materials conforms with the recommendations of Regulatory Guide 1.99, Revision 2, as described in subsection 5.2.4.3.

Regulatory Guide 1.108 - PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS (Rev. 1, 8/77)

CONFORMANCE

The conformance of diesel generator test frequencies to Regulatory Guide 1.108 is discussed in subsection 8.3.1.1.8.

Regulatory Guide 1.109 - CALCULATION OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10 CFR PART 50, APPENDIX I (Rev. 1, 10/77)

CONFORMANCE

Compliance with Regulatory Guide 1.109 is discussed in detail in paragraphs 11.2.8, 11.2.9, and 11.3.9.

Regulatory Guide 1.111 - METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASES FROM LIGHT-WATER-COOLED REACTORS (Rev. 1, 7/77)

CONFORMANCE

Compliance with Regulatory Guide 1.111 is discussed in detail in paragraph 2.3.5.2, and Table 2.3-17.

Regulatory Guide 1.112 - CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT-WATER-COOLED POWER REACTORS (Rev. O-R, 4/76)

CONFORMANCE

Compliance with Regulatory Guide 1.112 is discussed in detail in paragraph 11.1.1.2, and Table 11.1-7.

Regulatory Guide 1.113 - ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I (Rev. 1, 4/77)

CONFORMANCE

Compliance with Regulatory Guide 1.113 is discussed in detail in paragraph 11.2.8.

Regulatory Guide 1.127 - INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS (Rev. 1, 03/78)

CONFORMANCE

The service water pond dam and spillway inspections during the period of extended operation meet the intent of the guidance provided in NRC Regulatory Guide 1.127, Revision 1. See chapter 18, subsection 18.2.3.

Regulatory Guide 1.155 - STATION BLACKOUT (August 1988)

CONFORMANCE

Compliance with Regulatory Guide 1.155 is discussed in detail in paragraph 8.3.1.2.F.

Regulatory Guide 1.163 - PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM (September 1995)

CONFORMANCE

Farley Nuclear Plant has established a Containment Leakage Rate Testing Program to implement the requirements of 10 CFR 50 Appendix J, Option B, consistent with Regulatory Guide 1.163.

Regulatory Guide 1.163 endorses Nuclear Energy Institute (NEI) 94-01 Revision 0 dated July 26, 1995, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J”, with some exceptions. NEI 94-01 endorses ANSI/ANS - 56.8 - 1994, “Containment System Leakage Testing Requirements” for detailed descriptions of the technical methods and techniques for performing containment leakage tests with some exceptions. In addition, SNC maintains the option to use the Bechtel Topical Report BN-TOP-1, “Testing Criteria for Integrated Leak - Rate Testing of Primary Containment Structures for Nuclear Power Plants,” Revision 1, November 1972, method for performing Type A tests.

Regulatory Guide 1.182 - Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants (May 2000)

CONFORMANCE

FNP conforms to the guidance provided in this Regulatory Guide for complying with the provisions of 10 CFR 50.65(a)(4). This Regulatory Guide states that section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," dated February 11, 2000, of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provides methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.65(a)(4).

Regulatory Guide 1.190 - CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE (March 2001)

CONFORMANCE

Neutron fluence calculation procedures and dosimetry methods conform to the recommendations of Regulatory Guide 1.190.

Regulatory Guide 1.194 - ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL HABITABILITY ASSESSMENTS AT NUCLEAR POWER PLANTS (June 2003)

CONFORMANCE

The design guidance and assumptions of Regulatory Guide 1.194 are used with plant meteorological data for years 2000 through 2003 to calculate control room and technical support center X/Q for releases from the containment, containment equipment hatch, and plant vent. The plant meteorological data have been supplemented by a data set constructed to generate a 4.5-year data set with an overall frequency from the SSE, S, and SSW directions similar to that in the 1971 through 1975 data set presented in section 2.3.

Regulatory Guide 1.195 - METHODS AND ASSUMPTIONS FOR EVALUATING RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS AT LIGHT-WATER NUCLEAR POWER REACTORS (May 2003)

CONFORMANCE

The design guidance and assumptions of Regulatory Guide 1.195 are used as applicable in the evaluation of a fuel handling accident in the containment except:

> Dose conversion factors described in subsections 4.1.2 and 4.1.4 are not used. Instead, dose conversion factors from ICRP 30 are used as described in table 15B-1.

Regulatory Guide 1.196 - CONTROL ROOM HABITABILITY AT LIGHT-WATER NUCLEAR POWER REACTORS (May 2003)

CONFORMANCE

Regulatory Guide 1.196 provided detail guidance regarding control room habitability and the documentation of the control room habitability licensing bases as well as a program for testing the systems. FNP complies with this RG with the following exceptions:

1. FNP implemented a Technical Specification (TS) change that is an acceptable alternative to TS change identified in RG 1.196 which includes control room envelope (CRE) integrity testing and periodic assessments. These TS changes implement a Control Room Integrity Program.

2. FNP continues to use RG 1.52, Revision 0, for the design. FNP will update to Revision 3, dated June 2001, for testing only.

3. FNP uses RG 1.140 as information only for nonsafety-related air filtration systems.

4. FNP uses ASHRAE Guideline 1-1996 as reference only for maintenance programs for systems that handle hazardous chemicals and smoke challenges.

5. FNP uses RG 1.197, dated May 2003, for testing of the CRE. FNP has completed initial testing and will perform future testing of the Control Room Habitation System (CRHS) in accordance with RG 1.197, with the exceptions as defined in NEI 99-03, Rev. 1, Appendix EE, “ASTM E741 Exceptions.”

6. Surveys of hazardous chemicals will continue to be performed on a frequency of every 6 years. Offsite will be conducted in the year the tracer gas testing is scheduled, and the onsite will be offset by 3 years.

Regulatory Guide 1.197 - DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS (May 2003)

CONFORMANCE

Regulatory Guide 1.197 provides an approach acceptable to the NRC staff for measuring inleakage into the control room and associated rooms and areas at nuclear power reactors. The amount of inleakage is an input to the design of the control room, and periodic verification of the inleakage provides assurance that the control room will be habitable during normal and accident conditions. This guide provides guidance on methods acceptable to the staff for determining control room envelope (CRE) integrity for the purpose of confirming that the reactor meets GDC-19.

Farley Nuclear Plant conforms to this Regulatory Guide with the following exceptions:

The following exceptions as defined in NEI 99-03, Rev. 1, Appendix EE, “ASTM E-741 Exceptions.”

- These paragraphs may be totally excluded from implementation: 6.6.1, 6.6.2, 6.7,.6.7.1, 6.7.2, 8.5.4, 9.5.3, 9.5.4, 11.1.1, 12.3.2, 12.3.2.2, 13.2.1.2, 13.2.2, 13.4.2. Other editions are acceptable and may require similar exceptions.

- Use sections 1 through 5 only to define the test method and the equipment to be used.

- In section 8.5.3.1 a decay test using the regression method may be used to obtain confidence intervals as a part of the regression calculation.

- In section 9.2.1 the standard is not typically used when there is a nonsteady flow since such a test would only permit establishing bounds on the inleakage.

- Sections 9.2.3.1, 9.2.3.2, 9.2.3.3, and 9.2.3.4 are not typically used since makeup flowrate is typically used to estimate the anticipated concentration for an assumed tracer gas injection flowrate.

- Section 9.4.2 is not followed since a statistically significant number of samples are usually taken over 1 or 2 hours following the establishment of equilibrium.

- Sections 9.5.3.1 and 9.5.3.2 calculations are not used since the vendor demonstrates that concentration in CRE is not changing before making measurements designed to calculate total inleakage.

- Section 10 is not used in total.

- Section 11.1 is not used to measure indoor and outdoor temperatures or wind speed and direction, unless there is a direct need for the information.

- Section 15 is not used in total.

- Section 16 is not used in total. The vendor's report is to present the theory, data analysis, sampling locations, operating conditions, procedures, quality assurance records for the particular plant work order, data, calculations, and references.

- Section 17 is not used in total. The information is useful, but most vendors performing the test are highly experienced in many industrial settings and are familiar with these cautions and conditions. Uncertainty analysis or precision analysis may use the ANSI PT 19.1 Standard to calculate the 95-percent confidence intervals. The ANSI PT 19.1 Standard is not listed in Table D-1 since it is unrelated to the actual leakage determination.

[HISTORICAL] [APPENDIX 3B

CONTAINMENT PROOF TESTS

The basic purpose of a structural proof test is to substantiate that the containment can, in fact, carry the pressure load for which it is designed. By subjecting the containment to some degree of overpressure, the test can show that the containment has that degree of margin over design pressure and there would not be an incipient failure as might be the case if it were only tested at design pressure. Most previous steel and reinforced concrete containments and a number of the European prestressed concrete reactor vessels have been tested to 115 percent of design pressure. The FNP containment is proof tested at 115 percent of design pressure.

The prestressed containment relies mainly upon the tensile strength of the tendons for its ultimate strength. The secondary stresses of the containment are isolated from the tendons. At ultimate capacity of the containment, the secondary stresses and the thermal stresses are relieved by local cracking of the concrete and the tendons are generally subjected to internal pressure and dead load only.

In an evaluation of the containment overall margin of safety it is recognized that an exact determination is not a feasible requirement; however, it is possible to predict what is a reasonable value for the margin of safety.

As pointed out in Appendix 3D, the load factors associated with the pressure resulting from a LOCA will be the largest considered in the analysis and design of the containment. This is due to the fact that the degree of certainty for the magnitude of the pressure is less than that of the other loads. In view of this, the calculations to evaluate the margin of safety for the containment is based on the determination of what magnitude of pressure could be resisted at ultimate as a function of the design pressure. Consequently, the margin of safety is defined as a safety factor which, when multiplied by the design pressure, will result in the projected pressure that can be resisted by the ultimate capacity of the containment.

The calculations to determine this safety factor, which defines the margin of safety, are based on the strength of the prestressing steel, reinforcing steel, and the concrete. The additive strength that might be gained by the resistance afforded by the steel liner plate is not included because the leaktight membrane is not considered as a structural component member.

Limit strength calculations are developed to ascertain the minimum factor of safety. They are then compared to actual laboratory test results on prestressed concrete model structures for primary containments which are similar in form except for the end closure which in the model studies have been flat slabs.]

[HISTORICAL] [3C. MECHANICAL SPLICING REINFORCING BAR USING THE CADWELD PROCESS

[HISTORICAL] [APPENDIX 3C

MECHANICAL SPLICING REINFORCING BAR USING THE CADWELD PROCESS

3C.1 SCOPE

Mechanical splicing of deformed reinforcing bar for full tensile loading is accomplished with Cadweld connectors. The average tensile strength of the Cadweld joints is greater than the minimum tensile strength for the particular grade of reinforcing steel as specified in the appropriate ASTM standard. The minimum tensile strength of the splices exceeds 125 percent of the minimum yield strength for each grade of reinforcing steel as specified in the appropriate ASTM standard.

3C.2 PROCESS

All splices are made by the Cadweld process (Erico Products, Inc.), using clamping devices, sleeves, charges, and so on, as specified by the Cadweld Instruction Sheets for "T" series connections. "C" series and C-16 series materials are not permitted.

3C.3 QUALIFICATIONS OF OPERATORS

Prior to the production splicing of reinforcing bars, each operator or crew, including the foreman or supervisor for that crew, prepares and tests a joint for each of the positions used in production work. These splices are made and tested in strict accordance with the specification using the same ASTM grade and size of bar spliced in the production work. To qualify, the completed splices must meet the "Joint Acceptance Standards" for workmanship, visual quality, and minimum tensile strength. A list containing the names of qualified operators and their qualification test results is maintained at the jobsite.

3C.4 PROCEDURE

All joints are made in strict accordance with the manufacturer's instructions as presented in Erico Products Bulletin RB10M-670, 1970 Cadweld Rebar Splicing," plus the following additional requirements:

A. *A manufacturer's representative, experienced in Cadweld splicing of reinforcing bar, is present at the jobsite at the outset of the work to demonstrate the equipment and techniques used for making quality splices. He is also present for at least the first 50 production splices to observe and verify that the equipment is being used correctly and that quality splices are being obtained.*

B. *The splice sleeves, cartridges, asbestos wicking, ceramic inserts, and graphite parts are stored in a clean, dry area with adequate protection to prevent absorption of moisture.*

C. *Each splice sleeve is visually examined immediately prior to use to ensure the absence of rust and other foreign material on the inner surface.*

D. *The graphite molds are preheated with an oxyacetylene torch to 300°F minimum to drive off moisture immediately prior to use.*

E. *Bar ends to be spliced are in good condition with full-size, undamaged deformations. The bar ends are power brushed to remove all loose mill scale, rust, concrete and other foreign material. Prior to power brushing, all water, grease, and paint is removed by heating the bar ends with an oxyacetylene or propane torch.*

F. *A permanent line, marked 12 inches back from the end of each bar, serves as a reference point to confirm that the bar ends are properly centered in the splice sleeve.*

G. *Immediately before the splice sleeve is placed in final position, the previously cleaned bar ends are preheated with an oxyacetylene or a propane torch to insure complete absence of moisture.*

H. *Special attention is given to maintaining the alignment of sleeve and guide tube to insure a proper fill.*

I. *When the temperature is below freezing or the relative humidity is above 65 percent, the splice sleeve is externally preheated with an oxyacetylene or propane torch after all materials and equipment are in position.*

J. *The reinforcing bar deformations which become engaged in the Cadweld splice are not ground, flame cut, or altered in any way except for the longitudinal ribs, which are ground to a diameter not less than the other bar deformations.*

K. *An adequate escape route is provided for gases generated during the casting of horizontal splices. For splices in bars smaller than No. 11, this is done by inserting a hairpin piece of soft, twisted wire at the top of the splice between the rebar and the sleeve.*

L. *The packing material at the ends of the horizontal splices and at the top of the vertical splices is not hard packed. The material is firmly in place but loose enough to allow the escape of gases.*

3C.5 ONSITE USER TESTS

The onsite user test program for reinforcing steel splices is described below:

A. *Every operator is required to pass a qualification test.*

B. *All splices are visually inspected. As indicated in section 3C.7, unsatisfactory splices are replaced.*

C. *For each crew, after qualification, tests are made for each position as follows:*

Sister Splice Program

The following tensile program is used:

One out of the first lot of 10 production splices for each position, bar size and grade of bar.

One production splice and three "sister splices" from the next 90 splices, for each position, bar size, and grade of bar.

Three splices out of the next and subsequent lots of 100 splices for each position, bar size, grade of bar. One-fourth of these splices will be from production splices and three fourths from "sister splices."

A "sister splice" is defined as a 3-foot-long test bar spliced in sequence with, and in an otherwise identical manner as, the production splices.

3C.6 JOINT ACCEPTANCE STANDARDS

The following criteria are used for judging the acceptability of Cadweld joints:

A. Sound, nonporous filler metal must be visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed 1/4 inch from the end of the sleeve due to the packing material. This recess is not considered as poor fill.

B. Splices which contain slag or porous metal in the riser, tap hole, or at the ends of the sleeves (general porosity) are rejected. A single shrinkage bubble present below the riser is not detrimental and is distinguished from general porosity as described above.

C. The Cadweld splices, both horizontal and vertical, may contain voids at either or both ends of the Cadweld splice sleeve. At the end of the Cadweld splice sleeves, the acceptable size void for a No. 18 splice does not exceed 3 square inches per end of splice sleeve. The area of the void is assumed to be the circumferential length as measured at the inside face of the sleeve multiplied by the maximum depth of wire probe minus 3/16 inch.

D. The average tensile strength of the Cadweld joints shall be greater than the minimum tensile strength for the particular grade of reinforcing steel as specified in the appropriate ASTM standard. The minimum strength of the Cadweld joints must be greater than 125 percent of the specified minimum yield strength for the particular bar.

3C.7 REPAIRS

*Joints which do not meet the quality acceptance standards of section 3C.6 are rejected and completely removed. The bars are then rejoined with a new splice.**]***

APPENDIX 3D

JUSTIFICATION FOR LOAD FACTORS AND LOAD COMBINATIONS USED IN DESIGN EQUATIONS FOR THE CONTAINMENT

The load factors and load combinations in the factored load design equations represent the consensus of the individual judgments of a group of Bechtel engineers and consultants who are experienced in both structural and nuclear power plant design. Their judgment has been influenced by current and past practice, by the degree of conservativeness inherent in the basic loads, and particularly by the probabilities of coincident occurrences in the case of accident, wind, and seismic loads.

The following discussions explain the justification for the individual factors, particularly as they apply to containments.

A. Dead Load--Dead load in a large structure such as this is easily identified and its effect can be accurately determined at each point in the containment. For dead load in combination with accident and seismic or wind loads, a load factor of 1.0 is used for all load combinations.

B. Live Load--The live load that is present along with accident and seismic or wind loads produces a very small portion of the stress at any point. Also, it is extremely unlikely that the full live load is present over a large area at the time of an unusual occurrence. For these reasons, live loads are not included in the factored load design equations.

C. Seismic--The one-half safe shutdown earthquake (SSE) that has been selected is considered to be the strongest possible earthquake which could occur during the life of the plant. In addition to the one-half SSE, a safe shutdown earthquake which defines the maximum credible earthquake that could occur at the site, is considered in design. Category I structures are designed so that no loss of function results from the safe shutdown earthquake. Consequently, the probability of an SSE causing the loss-of- coolant accident (LOCA) is very small. For this reason, the two events, SSE and LOCA, are considered together, but at much lower load factors than those applied to the events separately. The earthquake load factors of 1.25 and 1.0 are conservative for one-half SSE and safe shutdown earthquake combination with the factored LOCA.

D. Wind--Loads are determined from the design tornado wind speed. Since the containment is designed for this extreme wind it is inconceivable that the wind would cause a LOCA. Therefore, wind loads are being considered with accident loads. A load factor of 1.0 is applied to the tornado load.

F. LOCA--The design pressure and temperature will be based on the operation of partial safeguards equipment using emergency diesel power.

European practice has been to use a load factor of 1.5 on the design pressure.[a] This factor is reasonable and is adopted for this design. The probabilities of a LOCA occurring simultaneously with a maximum wind or seismic disturbance are very small; therefore, a reduced load factor of 1.25 is used for the combination of events.

In all cases the design temperature is defined as that corresponding to the unfactored pressure. At 1.5 P the temperature is somewhat higher than the temperature at 1.0 P. It would be unrealistic to apply a corresponding temperature factor of 1.5 since this could occur only with a pressure much greater than a pressure of 1.5 P.

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a. Refer T. C. Waters and N. T. Barrett, "Prestressed Concrete Pressure Vessels for Nuclear Reactors," J. Brit. Nucl. Soc. 2, 1963.

[HISTORICAL] [APPENDIX 3E

JUSTIFICATION FOR CAPACITY REDUCTION FACTORS (ϕ - FACTORS) USED IN DETERMINING CAPACITY OF CONTAINMENTS

The ϕ-factors are provided to allow for variations in materials and workmanship. In the ACI Code, ϕ varies with the type of stress or member considered; that is, with flexure, bond or shear stress, or compression.

The ϕ-factor is multiplied into the basic strength equation or, for shear, into the basic permissible unit shear to obtain the dependable strength. The basic strength equation gives the "ideal" strength assuming materials are as strong as specified, as shown on the drawings, the workmanship is excellent, and the strength equation itself is theoretically correct. The practical, dependable strength may be something less since all these factors vary.

The ACI Code provides for these variables by using these ϕ-factors:

ϕ = *0.90 for concrete in flexure*
ϕ = *0.85 for diagonal tension, bond, and anchorage*
ϕ = *0.75 for spirally reinforced, concrete compression members*
ϕ = *0.70 for tied compression members*

ϕ is larger for flexure because the variability of steel is less than that of concrete. The ϕ values for columns are lower (favoring the toughness of spiral columns over tied columns) because columns fail in compression where concrete strength is critical. Also, it is possible that the analysis might not combine the worst combination of axial load and moment, and since the member is critical in the gross collapse of the containment, a lower value is used.

The additional ϕ values used represent Bechtel's best judgment of how much understrength should be assigned to each material and condition not covered directly by the ACI Code. The additional ϕ-factors have been selected based on material quality in relation to the existing ϕ-factors.

Conventional concrete design of beams requires that the design be controlled by yielding of the tensile reinforcing steel. This steel is generally spliced by lapping in an area of reduced tension. For members in flexure, ACI uses $\phi = 0.90$. The same reasoning is applied in assigning a value of $\phi = 0.90$ to reinforcing steel in tension, which now includes axial tension. However, the code recognizes the possibility of reduced bond of bars at the laps by specifying a ϕ of 0.85. For lap splices in the auxiliary building and structures other than the containment, a ϕ-factor of 0.90 is used because of the excessive lap used to splice. Mechanical and welded splices develop at least 125 percent of the yield strength of the reinforcing steel. Therefore, $\phi = 0.90$ is recommended for this type of splice.

The only significantly new value introduced is $\phi = 0.95$ for prestressed tendons in direct tension. A higher ϕ value than for conventional reinforcing steel is allowed because (1) during installation the tendons are each jacked to about 94 percent of their yield strength, so in effect each tendon has been proof tested, and (2) the method of manufacturing prestressing steel (cold drawing and stress relieving) ensures a higher quality product than conventional reinforcing steel.]

3.F COMPUTER PROGRAMS USED IN STRUCTURAL ANALYSES

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APPENDIX 3F

COMPUTER PROGRAMS USED IN STRUCTURAL ANALYSES

3F.1 INTRODUCTION

A number of computer programs are used in the structural analyses of the Category I structures. They are described and documented in this appendix. These computer programs are divided into four groups, as follows:

A. Computer programs used in the structural analyses by Bechtel Power Corporation.

B. Computer programs used in the seismic analyses by Bechtel Power Corporation.

C. Computer programs used in the seismic analyses by Southern Company Services, Inc. (SCS).

D. Computer programs used in the structural analyses by vendors and subcontractors.

   1. Chicago Bridge and Iron Company.

   2. Inland-Ryerson Construction Products Company.

   3. Whiting Corporation.

3F.2 COMPUTER PROGRAMS USED FOR THE STRUCTURAL ANALYSES BY BECHTEL POWER CORPORATION

Several computer programs are used by Bechtel Power Corporation in the structural analyses of the Category I structures. They are listed below and described and documented in the following sections:

A. Bechtel CE 316-4 Finite Element Stress Analysis (FINEL).

B. Bechtel CE 639-2 Forces and Pressures Acting on the Dome due to Prestressing of Tendons.

C. Bechtel CE 779 Structural Analysis Program (SAP).

D. Bechtel ME 620 Heat Conduction.

E. Axisymmetric Shell and Solid Computer Program (ASHSD).

A summary of these computer programs is presented in table 3F.1.

3F.2.1 CE 316-4 FINITE ELEMENT STRESS ANALYSIS (FINEL)

A. Description

The program performs the static analyses of plane or axisymmetric structures using the finite element method, in which a structure is idealized as an assemblage of finite elements. The finite elements are of either triangular or quadrilateral shape, connected at their corners (nodal points). The applied loads may be concentrated, uniformly distributed, or inertial, or may be temperature distributions. At boundaries, displacements may be forced.

The program develops the force displacement relationship (element stiffness matrix) for each individual element from its geometry and material properties. The element relationships are then assembled into an overall structure force displacement relationship (structure stiffness matrix). Equilibrium equations are developed for each degree of freedom at each nodal point in terms of the structure force displacement relationship, the unknown nodal point displacement, and the externally applied nodal point forces. Finally, these equations are solved simultaneously for the unknown nodal point displacements by a modified Gaussian elimination scheme. Once the nodal point displacements are known, element stresses are calculated.

B. Assumptions

The stress and the strain are assumed to be constant within each element.

C. Validation

The program has been verified by manual calculations. Document traceability is available at Pacific International Computing Corporation.

D. Extent of Application

The program is used to compute stresses in the containment structure due to gravity, pressure, and thermal loads.

3F.2.2 CE 639-2 FORCES AND PRESSURES ACTING ON THE DOME DUE TO PRESTRESSING OF TENDONS

A. Description

The program performs an analysis of forces and stresses that act on a dome due to prestressed tendons. The shape of the dome may be sphere-torus, sphere-cone, hemisphere, cone, or ellipsoid; and the tendons may be in one, two, or three directions. The program is capable of analyzing the prestress loss of the dome tendons caused by frictional resistance, and the variation of the prestressing forces due to seating of the anchorages. In addition, the program may be used in preparing prestressing sequences.

B. Validation

The necessary validation of CE 639-2 has been completed. The program was validated by an independent calculation performed in November 1974.

C. Extent of Application

The program is used to analyze the forces and pressures acting on the dome due to prestressing of tendons. It is also used in the preparation of the prestressing sequences.

3F.2.3 CE 779 STRUCTURAL ANALYSIS PROGRAM (SAP)

A. Description

The program performs the static and dynamic analyses of linear, elastic, three-dimensional structures using the finite element method. The finite element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved for either applied loads or temperature distributions. Concentrated loads, pressures or gravity loads may be applied. Temperature distributions are assigned as an appropriate uniform temperature change in each element. Prestressing may be simulated by using artificial temperature changes on rod elements.

Dynamic response routines are available for solving arbitrary dynamic loads or seismic excitations using either modal superposition or direct integration. The program can also perform response spectrum and time- history analyses.

B. Validation

The solutions to test problems have been demonstrated to be essentially identical to the results obtained using the ASKA program, which was developed by Prof. A. J. Argyris (Institut fur Statik und Dynamik, Stuttgart) and to the Chan and Fermin program. The test problem solutions have also been compared to, and found to be in agreement with, the solutions of the programs from the ASME Library of Benchmark Computer Problems and Solutions. Document traceability is available at Pacific International Computing Corporation.

C. Extent of Application

The program is used in the structural analysis of the containment shell at the region of the equipment hatch opening.

D. Reference

1. "A Refined Quadrilateral Element for Analysis of Plate Bending," Proc. (second) Conference on Matrix Methods in Structural Mechanics, Wright Patterson AFB, Ohio, 1968.

3F.2.4 ME 620 HEAT CONDUCTION

A. Description

The program performs the transient or steady-state temperature analyses of plane or axisymmetric solids. Regional temperature distributions may be determined due to prescribed boundary temperatures or boundary and internal heat fluxes.

The thermal analyses are carried out by the finite element technique coupled with a step-by-step time integration procedure.

B. Validation

The program has been verified by manual calculations. Document traceability is available at Pacific International Computing Corporation.

C. Extent of Application

The program is used to compute the temperature distribution for the containment at locations where the geometry of the structure is too complex for manual calculation, such as wall and base slab intersections, and wall and ring girder intersection, etc.

3F.2.5 AXISYMMETRIC SHELL AND SOLID COMPUTER PROGRAM (ASHSD)

A. Description

The program performs the static and dynamic analyses of linear, elastic, axisymmetric structures with axisymmetric or nonaxisymmetric loadings, utilizing the finite element technique. The program computes the element stresses and nodal displacements due to uniform, concentrated, or pressure loads, or temperature distributions, either over the surface area or through the wall thickness. Prestress forces may be simulated by applying the forces as equivalent concentrated temperature gradients.

B. Validation

The solutions of the program for various loadings have been demonstrated to be essentially identical to the results obtained by manual calculations and to those obtained from accepted experimental tests or analytical results published in technical literature. (See references 1 and 2.)

C. Extent of Application

The program is used in the analysis of the containment for nonaxisymmetric loadings.

D. References:

1. Ghosh, S. and Wilson, E. L., "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading", Report No. EERC 69-10, Univ. of California, Berkeley, Sept. 1969, pp 69-81.

2. "Topical Report on Dynamic Analysis of Reactor Vessel Internals under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 MWe Class Reactors", Combustion Engineering Report CENPD-42, Combustion Engineering, Inc., Nuclear Power Department, Combustion Division, Windsor, Conn., Appendix A.

3F.3 COMPUTER PROGRAMS USED IN THE SEISMIC ANALYSIS BY BECHTEL POWER CORPORATION

A number of computer programs are used by Bechtel Power Corporation in the seismic analyses of the Category I structures. They are listed below and described and documented in the following sections.

1. Bechtel CE 309 Structural Engineering Systems Solver (STRESS).

2. Bechtel CE 591 Spectra Analysis.

3. Bechtel CE 611 Time-History Response Analysis.

4. Bechtel CE 617 Modes and Frequencies Extraction.

5. Bechtel CE 641 Earthquake Response Spectrum Analysis of Structures.

6. Bechtel CE 655 Stress Dynamic Analysis.

7. Bechtel CE 785 Spectrum Suppressing.

8. Bechtel CE 786 Spectrum Raising.

9. Bechtel CE 791 Spectra Analysis.

10. Bechtel CE 792 Generation of Response Spectra from strong-motion Earthquake Records.

A summary of these computer programs is presented in table 3F.2.

3F.3.1 CE 309 STRUCTURAL ENGINEERING SYSTEMS SOLVER (STRESS)

A. Description

STRESS is a programming system for the solution of structural engineering problems. The system is capable of executing the linear, elastic, static analyses of two- and three-dimensional framed structures of the following types:

1. Plane truss.
2. Plane frame.
3. Plane grid.
4. Space truss.
5. Space frame.

The programming system was originally developed at Massachusetts Institute of Technology in 1964 and is now in the public domain.

B. Validation

The program has been verified by the ICES STRUDL II program. A sample problem of space frame analysis was run using the CE 309 program and the commercially available versions (Version 1 and Version 2) of the ICES STRUDL II program. The results from these runs were found to be identical. Document traceability is available at Pacific International Computing Corporation.

C. Extent of Application

The program is used to obtain the flexibility matrices of the Category I structures. The flexibility matrices are used in the dynamic analyses of the structures.

D. Reference

Fenjes, S.J., Logcher, R.D., and Mauch, S.P., Stress Reference Manual, The M.I.T. Press, Cambridge, Mass., 1964.

3F.3.2 CE 591 SPECTRA ANALYSIS

A. Description

The program computes and plots the response spectra for any ground excitation described in acceleration time coordinates, such as earthquakes, blasts, etc.

B. Validation

The solutions to the program have been verified to be essentially identical to the results obtained by manual calculations. Document traceability is available at Bechtel Power Corporation.

C. Extent of Application

The program is used to compute and plot the response spectra for the seismic analyses of Category I structures.

3F.3.3 CE 611 TIME-HISTORY RESPONSE ANALYSIS

A. Description

The program performs the response time-history analysis of a structure subjected to an earthquake motion using the modal superposition technique. The response is calculated in terms of displacement, velocity, and acceleration time-histories at selected points. Inertia forces, moments, and shears may be computed for cantilevered structures. Maximum response values and time of occurrence may be found.

B. Validation

The solutions of the program have been verified to be substantially identical to the results of manual calculations. Document traceability is available at Pacific International Computing Corporation.

C. Extent of Application

The program is used to generate the time histories at all Category I equipment locations in the structures.

D. References

1. Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill, 1964.

2. Hildebrand, F. B., Introduction to Numerical Analysis, McGraw-Hill, 1956.

3. Hurty, W. C., Rubinstein, M. F., Dynamics of Structures, Prentice Hall, Inc., 1964.

4. Kuo. S. S., Numerical Methods and Computers, Addison Wesley, 1965.

3F.3.4 CE 617 MODES AND FREQUENCIES EXTRACTION

A. Description

This program provides a means for obtaining the natural frequencies and mode shapes of structural models. Input to the program consists of the model's lumped masses and either the stiffness matrix or the flexibility matrix. If the flexibility matrix is entered, the program provides for automatic inversion to a stiffness matrix.

The program extracts eigenvalues and eigenvectors from the input, using the Jacobi diagonalization method by successive rotations.

B. Validation

The program has been verified by manual calculations. Document traceability is available at Bechtel Power Corporation.

C. Extent of Application

The program is used to obtain the mode shapes and natural frequencies of all Category I structures.

D. Reference

Crandall, S., Engineering Analyses, A Survey of Numerical Procedures, McGraw-Hill, 1966.

3F.3.5 CE 641 EARTHQUAKE RESPONSE SPECTRUM ANALYSIS OF STRUCTURES

A. Description

The program computes the response of a structure subjected to an earthquake motion, utilizing the response spectrum technique. The structure is defined in terms of natural frequencies, mode shapes, lumped weights, and elevations. The earthquake is described in terms of a response spectrum curve. The response values computed for each mode are combined, using the sum of the absolute values and the square root of the sum of the squares (SRSS).

B. Validation

The solutions to the problem have been verified to be substantially identical to the results obtained by manual calculations. Document traceability is available at Bechtel Power Corporation.

C. Extent of Application

The program is used to obtain the modal responses of all Category I structures.

3F.3.6 CE 655 STRESS-DYNAMIC ANALYSIS

A. Description

The program is used in conjunction with CE 309, STRESS, to obtain the flexibility matrix or stiffness matrix of a structure. The matrix is then used with programs such as CE 617, Modes and Frequencies Extraction, to evaluate the dynamic characteristics of the structure.

B. Validation

The program by itself does not have the capability to analyze a structure. It merely extracts the output from the CE 309 STRESS program to build up a flexibility/stiffness matrix for the same structure. Consequently, validation is not necessary.

C. Extent of Application

The program is used to extract a flexibility/stiffness matrix from CE 309 STRESS program.

3F.3.7 CE 785 SPECTRUM SUPPRESSING

A. Description

The program generates a synthetic time-history to fit closely a given response spectrum curve. This is accomplished by modifying a given accelerogram so that the acceleration response spectrum may be locally suppressed at any given frequency without significantly changing the remaining portions of the spectrum. The program computes the modified accelerogram and adjusts its maximum acceleration to match the specified value, plots the modified accelerogram, and calculates the modified accelerogram response spectrum.

B. Validation

The results of the spectrum computation of the program have been compared with those obtained by Bechtel programs CE 786 and CE 791. It was found that the spectra of the same motion as computed by the three different programs are essentially identical. Document traceability is available at Bechtel Power Corporation.

C. Extent of Application

The program is used to generate the synthetic time-history used in the seismic analyses of Category I structures.

3F.3.8 CE 786 SPECTRUM RAISING

A. The program generates a synthetic time history to fit closely a given response spectrum curve. This is accomplished by modifying a given accelerogram so that the acceleration response spectrum may be locally raised at any given frequency without significantly changing the remaining portions of the spectrum. The program computes the modified accelerogram and adjusts its maximum acceleration to match the specified value, plots the modified accelerogram, and calculates the modified accelerogram response spectrum.

B. Validation

The results of the spectrum computation of the program have been compared with those obtained by Bechtel programs CE 785 and CE 791. It was found that the spectra of the same motion as computed by the three different programs are essentially identical. Document traceability is available at Bechtel Power Corporation.

C. Extent of Application

The program is used to generate the synthetic time-history used in the seismic analyses of Category I structures.

3F.3.9 CE 791 SPECTRA ANALYSIS

A. Description

The program computes and plots the response spectra for any ground excitation described in acceleration time coordinates, such as earthquakes, blasts, etc.

B. Validation

The solutions to the program have been verified to be essentially identical to the results obtained by manual calculations. Document traceability is available at Bechtel Power Corporation.

C. Extent of Application

This program is used to compute and plot the response spectra for the seismic analyses of Category I structures.

3F.3.10 CE 792 GENERATION OF RESPONSE SPECTRA FROM STRONG-MOTION EARTHQUAKE RECORDS

A. Description

The program computes the response spectra at specified values of frequencies from the input time-history which are generated by CE 611, Time-History Response Analysis, and for given values of damping ratios.

B. Assumptions

The numerical method used for integration is based on the exact solution to the governing differential equation, assuming that the input acceleration time-history varies linearly between consecutive data points.

C. Validation

The solutions to the program have been demonstrated to be substantially identical to the results obtained by manual calculations. Document traceability is available at Bechtel Power Corporation.

D. Extent of Application

The program is used to generate response spectra at all equipment locations in Category I structures.

E. Reference

Nigam, N. C. and Jennings, P. C., "Digital Calculation of Response Spectra From Strong-motion Earthquake Records," C.I.T., 1968.

3F.4 <u>COMPUTER PROGRAMS USED IN SEISMIC ANALYSES BY SOUTHERN COMPANY SERVICES, INC.</u>

Several computer programs are used by Southern Company Services, Inc., (SCS) in the seismic analysis of the Category I structures. They are listed below and are described and documented in the following sections:

A. Program EN 3426, ICES STRUDL II.

B. Program EN 3423, Modal Analysis.

C. Program EN 8043, Response Spectra Plots.

D. Program EN 8045, Response Spectra.

E. Program EN 8046, Spectra Plotter.

A summary of these computer programs is presented in table 3F-3.

3F.4.1 EN 3426 ICES STRUDL II, THE STRUCTURAL DESIGN LANGUAGE

A. Description

This program computes the flexibility matrix of a mathematical beam model of a structure. This is accomplished by applying a unit load at one mass point and calculating displacements at all mass points. The procedure is repeated for all mass points. This matrix is used as input for program EN 3423 to generate mode shapes and frequencies.

B. Assumptions

The beam theory is used in the analytical procedure of the stiffness analysis. It assumes a linear, elastic, static, small-displacement analysis. Member properties are required, and the program treats the joint displacements as unknowns.

C. Validation

The solutions of the program have been proven to be substantially identical to the results obtained by another program, SAP IV. Document traceability is available at Southern Company Services, Inc. The STRUDL program is in the public domain and was originally issued March 1968. It was obtained from the Massachusetts Institute of Technology. The version used is ICES-VI-M2, STRUDL2-VI-MO, issued 1970.

SAP IV is a Structural Analysis Program that originated at the University of California, Berkeley. The original version was issued in September 1970 and is in the public domain. Southern Company Services, Inc., obtained SAP IV in October 1973 from the University of California.

The computer used is the IBM 370 155, together with its operation systems.

D. Extent of Application

The program is used to generate the flexibility matrix of all Category I structures.

E. References:

1. Fenves, S. J. and Branin, F., "Network - Topological Formulation of Structural Analysis", Journal of the Structural Division, ASCE, August 1963.

2. Fenves, S. J., Mauch, S. P., and Kinra, R. J., "Treatment of Releases and Constraints in the Network Formulation of Structural Analysis,"

Technical Report, Structural Research Series, No. 299, University of Illinois, Urbana, Ill., October 1965.

3F.4.2 EN 3423 MODAL ANALYSIS

A. Description

The flexibility matrix of the mathematical model is obtained from program EN 3426. A diagonal mass matrix is used. Basically, the program solves for the natural frequencies and mode shapes for the mathematical model. It calculates the composite damping as a percent of critical damping by the modal weighing method. From the ground spectra the spectral acceleration is obtained for each mode. The program then calculates the square root of the sum of the squares of deflections, shears, moments, inertial forces, and spectral accelerations for each mass point. These calculations are made for safe shutdown earthquake and one-half safe shutdown earthquake.

B. Assumptions

A three-dimensional structure is represented by a mathematical model of lumped masses with weightless elastic columns acting as spring restraints to obtain the response of the structure. The program solves for the natural frequencies and mode shapes of a freely vibrating, undamped linear elastic system.

C. Validation

The solutions of the program have been proven to be substantially identical to the results obtained by another program, SAP IV. Document traceability is available at Southern Company Service, Inc. SAP IV is a Structural Analysis Program that originated at the University of California, Berkeley. The original version was issued in September 1970 and is in the public domain. Southern Company Services, Inc., obtained SAP IV in October 1973 from the University of California.

The computer used is the IBM 370 155, together with its operation systems.

D. Extent of Application

The program is used to generate the response of the structure and obtain inertial forces that are applied to the original structure of all Category I structures.

E. References:

1. Invert matrix - Standard Gauss - Jordan Method.

2. Computation of eigenvalues and eigenvectors - Diagonalization method originated by Jacobi and adapted by Von Neumann.

3. Ralston, A. and Wilf, H. W., eds., Mathematical Methods for Digital Computers, John Wiley and Sons, N.Y., 1962, Chapter 7.

3F.4.3 EN 8043 RESPONSE FOR SPECTRA PLOTS

A. Description

The program uses the modal superposition method as a solution technique. Mode shapes, natural frequencies, participation factors, and viscous damping obtained from program EN 3423 are used as input. The program determines the response of each mode separately and then calculates the total response by superposition. The time-histories of displacement and acceleration of each mass point are calculated. The acceleration time-histories for desired mass points are stored on tape and used as input for program EN 8045.

B. Validation

The solutions of the program have been proven to be substantially identical to the results obtained by another program, DYNAL. DYNAL (Dynamic Analysis Computer Program) is in the public domain and has been in use since 1970. Document traceability is available at Southern Company Services, Inc.

C. Extent of Application

The program is used to calculate the deflections and accelerations of the structure. It is also used to generate the time-history acceleration at required mass points.

D. Reference

Scarborough, J. B., Numerical Mathematical Analysis, Johns Hopkins Press, Baltimore, 1956

3F.4.4 EN 8045 RESPONSE SPECTRA

A. Description

The time-history of floor acceleration generated in program EN 8043 is used as input. EN 8045 is then used in the generation of floor response spectra computed from the time-history motions at the floor desired. The floor response spectra give the maximum response of single degree of freedom bodies of different natural frequencies as a function of damping when these bodies are subjected to a floor time- history.

B. Validation

The solutions of the program have been proven to be substantially identical to the results obtained by another program, DYNAL. DYNAL (Dynamic Analysis Computer Program) is in the public domain and has been in use since 1970. Document traceability is available at Southern Company Services, Inc.

C. Extent of Application

The program is used to generate floor response spectra at all equipment locations.

D. Reference

Scarborough, J. B., Numerical Mathematical Analysis, Johns Hopkins Press, Baltimore, 1956.

3F.4.5 EN 8046 SPECTRA PLOTTER

A. Description

The program plots the floor response spectra. The maximum acceleration for a given frequency generated by program EN 8045 is used as input. The acceleration versus frequency is plotted on semilogarithmic, three- cycle paper.

B. Validation

The graph plotted by the program has been reproduced by a manual method. Both graphs were compared and found to be identical. Document traceability is available at Southern Company Services, Inc.

C. Extent of Application

The program is used to plot all floor response spectra.

D. Reference

Programming Calcomp Pen Plotters, California Computer Products, Inc., Anaheim, Calif., June 1968.

3F.4.6 PS+CAEPIPE

A. Description

The PS+CAEPIPE program is a finite element computer program which performs linear elastic analysis of piping systems using the stiffness method of finite element analysis; the displacements of the joints of a given structure are considered basic unknowns. The dynamic analysis by the modal synthesis method utilizes known maximum accelerations produced in a single degree of freedom model of a certain frequency. The principal program assumptions are as follows:

It is a linearly elastic structure. Simultaneous displacement of all supports is described by a single time-dependent function. Lumped mass model satisfactorily replaces the continuous structure. Modal synthesis is applicable. Rotational inertia of the masses has negligible effect.

B. Validation

The results obtained from pipe stress program PS+CAEPIPE have been compared with the following:

ASME Benchmark problem results, Pressure Vessel and Piping 1972 computer programs verification, American Society of Mechanical Engineers. Longhand calculations--PS+CAEPIPE is compatible with NRC Regulatory Guide 1.92. A synthesis of closely spaced modes is provided based on equation 4 of Regulatory Guide 1.92.

Verification problems were prepared by SST, Inc. and reviewed by SCS.

C. Extent of Application

The PS+CAEPIPE program is used to determine stresses and loads in the piping systems due to restrained thermal expansion, deadweight, seismic inertia and anchor movements, externally applied loads such as jet-loads, and transient forcing functions such as created by fast relief valve opening and closing, fast check valve closure after pipe breaks in main feedwater line, fast valve closure in main steam line, etc. PS+CAEPIPE analyzes piping systems in accordance with ANSI and ASME codes.

D. Reference

PS+CAEPIPE Program is a software licensed by SST System, Inc.

3F.5 COMPUTER PROGRAMS USED IN THE STRUCTURAL ANALYSES BY VENDORS AND SUBCONTRACTORS

A number of computer programs are used in the structural analyses of Category I structures by the following vendors and subcontractors:

A. Chicago Bridge and Iron Company.

B. Inland-Ryerson Construction Products Company.

C. Whiting Corporation.

The computer programs that they have used are listed, described, and documented in the following sections.

A summary of these programs is presented in table 3F-4.

3F.5.1 COMPUTER PROGRAM USED BY CHICAGO BRIDGE AND IRON COMPANY

The computer program described and documented in the following section is used by Chicago Bridge and Iron Company in the analyses of the equipment hatch and personnel lock for the containment structure.

3F.5.1.1 CB&I Program 7-81

A. Description

This Shells of Revolution program, which is based on the ASME paper "Analysis of Shells of Revolution Subjected to Symmetrical and Non-Symmetrical Loads" by A. Kalnins, is a standard computer program in the industry. The program computes the stresses and displacements in thin-walled, elastic shells of revolution when they are subjected to static edge loads, surface loads, or arbitrary temperature distribution over the surface of the shell. The geometry of the shell must be symmetrical; however, the shape of the median may be arbitrary. The shell wall may consist of four layers of different orthotropic materials, and the thickness and elastic property of each layer may vary along the median.

The program numerically integrates the eight ordinary first-order differential equations of the thin-shell theory derived by H. Reissner.

The CB&I version of the Shells of Revolution program incorporated modifications on the method of input and the format of output.

B. Validation

The results of the program were compared with those obtained by other shell programs, such as Seal and Cerl II, and were found to be in excellent agreement. Document traceability is available at Chicago Bridge and Iron Company.

C. Extent of Application

The program is used in the analyses of the equipment hatch and personnel lock for the containment structure.

D. Reference

Kalnins, A., "Analysis of Shells of Revolution subjected to Symmetrical and Non-Symmetrical Loads," presented at the Summer Conference of the Applied

Mechanics Division, Boulder, Colorado, June 9-11, 1964, of the American Society of Mechanical Engineers.

3F.5.2 COMPUTER PROGRAMS USED BY INLAND-RYERSON CONSTRUCTION PRODUCTS COMPANY

The computer programs described and documented in the following sections are used by Inland-Ryerson Construction Products Company in computing the geometry and prestress losses of the containment structure post-tensioning system.

A. Program WDINT.

B. Program POCKET.

C. Program NUFRCOHO.

3F.5.2.1 Program WDINT

A. Description

Program WDINT is a proprietary computer program of Inland-Ryerson Construction Products Company. The program deals with the spatial relationship between tendons of the containment structure post tensioning system, by the usual and familiar methods and formulas of three-dimensional analytic geometry.

Input parameters include the defined locations of the dome tendons, the desired locations of the vertical tendons, and the dimensions of both the dome and the vertical tendons.

The program examines each vertical tendon in turn for interference with any dome tendons. If an interference is detected, a new location for the vertical tendon is examined until the closest location to either side of the desired location which does not interfere with the dome tendons is found.

The output of the program provides the necessary information for detailing the tendon placement drawings.

B. Validation

The results of the program have been verified by manual calculations. Document traceability is available at Inland-Ryerson Construction Products Company.

C. Extent of Application

The program is used to detect interference between the vertical and the dome tendons of the containment structure post tensioning system.

3F.5.2.2 Program POCKET

A. Description

Program POCKET is a proprietary computer program of Inland-Ryerson Construction Products Company. The program deals with the spatial relationship between the containment structure dome tendon anchorages and the surrounding concrete surfaces, by the use of three-dimensional analytic geometry.

Input parameters include the locations and trajectories of the dome tendons and the geometry of the concrete surfaces at the anchorages.

The program computes the necessary dimensions and angles to define the locations, orientations, and dimensions of the pockets for the anchorage assemblies.

Output of the program is used to detail the pockets on the erection shop drawings.

B. Validation

The results of the program have been verified by manual calculations. Document traceability is available at Inland-Ryerson Construction Products Company.

C. Extent of Application

The program is used to compute the required angles and dimensions of the pockets at the containment structure ring girder for the dome tendon anchorage assemblies.

3F.5.2.3 Program NUFRCOHO

A. Description

Program NUFRCOHO is a proprietary computer program of the Inland-Ryerson Construction Products Company. The program deals with the prestress losses of the containment structure post tensioning system.

Input parameters include the geometry of the tendon trajectories, physical properties of materials, friction coefficients, and jacking characteristics.

The program computes the prestress losses for each tendon of the containment structure. For purposes of analysis, the geometry of each tendon trajectory is considered in segments, i.e. straight, curved, and transitional. The force at the end of the tendon is the jacking force, and the force at the remote end of the first segment is the jacking force reduced by friction. The force at the beginning of the second segment is then the force at the end of the first segment, etc., for the length of the tendon. The minimum force in the tendon is either at the fixed end (for tendons stressed from one end) or near the middle of the tendon (for tendons stressed from both ends). The location and magnitude of the minimum force for tendons stressed from both ends are found by computing, from each end, the point of intersection of the lines graphing the influences of each jack.

Output consists of data on elongations, forces, and stresses at various points along the tendon, and certain dimensional information.

B. Assumptions

a. The prestress elements behave in accordance with Hook's Law within the stress level to which they are subjected.

b. The force used in computing the elongation of any segment of a tendon is the average of the forces at the ends of the segment.

c. Friction acts in accordance with Coloumb's friction formula. In addition, the friction coefficients for use in Coloumb's formula are derived experimentally for each containment structure and are assumed to be the same for all similar tendons in the structure.

C. Validation

The results of the program have been verified by manual calculations. Document traceability is available at Inland-Ryerson Construction Products Company.

D. Extent of Application

The program is used to prepare field stressing cards, which provide the necessary information required during stressing of tendons, such as hydraulic pressures, pressure ranges, and elongations at various specified force levels.

E. Reference

American Concrete Institute, "Building Code Requirements for Reinforced Concrete," ACI 318-71.

3F.5.3 COMPUTER PROGRAMS USED BY WHITING CORPORATION

The computer program described and documented in the following section is used by Whiting Corporation in the seismic analysis of the polar crane inside the containment structure.

3F.5.3.1 STARDYNE DYNRE 4

A. Description

The STARDYNE Static and Dynamic Structural Analysis System, developed by Mechanics Research, Inc., Los Angeles, California, can perform static and dynamic analyses of complex three-dimensional structures using the finite element method. DYNRE 4 is one of the six major programs of the STARDYNE system and is used for computing the response of a general STARDYNE modeled structure to an arbitrarily oriented shock spectra input. The program includes two superposition techniques for displacements, velocities, and accelerations, as well as element loads and stresses. Shock input consists of user-furnished spectral values or averaged spectra, normalized to the 1940 El Centro earthquake (N-S component). The averaged spectra include data from the 1934 El Centro, 1940 El Centro, 1949 Olympia, and 1952 Taft earthquakes.

B. Validation

The program is a recognized program in the public domain and has had sufficient history of use to justify its applicability and validity.

C. Extent of Application

The program is used in the seismic analyses of the polar crane inside the containment.

D. Reference

"MRI/STARDYNE-Static and Dynamic Structural Analysis System-Theoretical Manual," Publication No. 866 16300, Control Data Corporation, June 15, 1973.

TABLE 3F-1

COMPUTER PROGRAMS USED FOR CATEGORY I STRUCTURAL ANALYSES BY BECHTEL POWER CORPORATION

Program	Title	Document Traceability	Program Capabilities
CE 316-4	FINEL	PICC[a]	Analyzes complex axisymmetric structures for both elastic and inelastic behavior by the finite element method
CE 639-2	-	BPC[b]	Performs analysis to determine the forces and stresses on a dome caused by prestressing of tendons, including the effects of friction losses
CE 779	SAP	PICC	Performs linear elastic analyses of three-dimensional structural systems
ME 620	Heat Induction	PICC	Determines the temperature distribution within a plane or axisymmetric body
ASHSD	-	PICC	Analyzes axisymmetric structures by the finite element method for axisymmetric and asymmetric static and dynamic loads

a. Pacific International Computing Corporation, San Francisco, California.

b. Bechtel Power Corporation, Gaithersburg, Maryland.

TABLE 3F-2 (SHEET 1 OF 2)

COMPUTER PROGRAMS USED IN SEISMIC ANALYSES BY BECHTEL POWER CORPORATION

Program	Title	Document Traceability	Program Capabilities
CE 309	STRESS	PICC[a]	Generates flexibility matrix or stiffness matrix for structural models
CE 591	Spectra Analysis	PICC	Calculates and plots response spectra for earthquake accelerogram
CE 611	Time-History Response Analysis	PICC	Computes time-history response for structures subjected to earthquake
CE 617	Modes and Frequencies Extraction	BPC[b]	Extracts modes and frequencies from stiffness or flexibility matrix and diagonal mass matrix
CE 641	Response Spectrum Analysis	BPC	Spectral response analysis of simple cantilever structures
CE 655	Stress Dynamic Analysis	PICC	Extracts flexibility or stiffness matrix from CE 309, "STRESS"
CE 785	Spectrum Suppression	BPC	Suppresses locally the response spectrum of a given accelerogram
CE 786	Spectrum Raising	BPC	Raises locally the response spectrum of a given accelerogram

TABLE 3F-2 (SHEET 2 OF 2)

Program	Title	Document Traceability	Program Capabilities
CE 791	Spectra Analysis	BPC	Calculated and plots response spectra for earthquake accelerogram
CE 792	Response Spectra	BPC	Computes response spectra at specified values of frequencies and damping ratios using CE 611, "Time-History Response Analysis"

a. Pacific International Computing Corporation, San Francisco, California.

b. Bechtel Power Corporation, Gaithersburg, Maryland.

TABLE 3F-3

COMPUTER PROGRAMS USED IN SEISMIC ANALYSES BY SOUTHERN COMPANY SERVICES, INC.

Program	Title	Document Traceability	Program Capabilities
EN 3426	ICES STRUDL II	SCS	Analyzes two- or three-dimensional framed structures and continuous mechanics problems. Flexibility or stiffness matrix may be generated.
EN 3423	Modal Analysis	SCS	Computes the weights of mass points, damping values, mode shapes, and frequencies from the flexibility matrix generated by EN 3426 and EN 3429
EN 8043	Response Spectra Plots	SCS	Computes the maximum response of a single degree of freedom oscillator for various frequencies and damping values
En 8045	Response Spectra	SCS	Computes the time acceleration of the desired mass points using the mode shapes and frequencies obtained from EN3423, "Modal Analysis," and the damping values of the structure
EN 8046	Response for Spectra Plots	SCS	Plots the maximum response computed by EN 8043, "Response Spectra Plots"
PS+CAEPIPE		SCS	Analysis of Piping System per ANSI and ASME Codes

TABLE 3F-4 (SHEET 1 OF 2)

COMPUTER PROGRAMS USED IN CATEGORY I STRUCTURAL ANALYSIS BY VENDORS AND SUBCONTRACTORS

Company	Program	Title	Document Traceability	Program Capabilities
CB&I[(a)]	7-81	Shells of Revolution	CB&I	Performs the analysis of shells of revolution subjected to symmetrical and nonsymmetrical loads
Inland-Ryerson[(b)]	WDINT	--	Inland-Ryerson	Computes the spatial relationship between tendons of the containment structure post-tensioning system
Inland-Ryerson	POCKET	--	Inland-Ryerson	Computes the spatial relationship between dome tendon anchorages and the surrounding concrete surfaces of the containment structure post-tensioning system
Inland-Ryerson	NUFRCOHO	--	Inland-Ryerson	Computes the prestress losses of the containment structure post-tensioning system

TABLE 3F-4 (SHEET 2 OF 2)

Company	Program	Title	Document Traceability	Program Capabilities
Whiting[c]	STARDYNE-DYNRE 4	-	Recognized Program in Public Domain	Computes responses caused by arbitrarily-oriented shock spectra

a. Chicago Bridge & Iron Company, Birmingham, Alabama.

b. Inland-Ryerson Construction Products Company, Melrose Park, Illinois.

c. Whiting Corporation, Harvey, Illinois.

3G QUALITY CONTROL PROCEDURES FOR FIELD WELDING AND NONDESTRUCTIVE EXAMINATIONS OF CONTAINMENT LINER PLATE

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[HISTORICAL] [QUALITY CONTROL PROCEDURES FOR FIELD WELDING AND NONDESTRUCTIVE EXAMINATIONS OF CONTAINMENT LINER PLATE

3G.1 SCOPE

These procedures outline the general quality control requirements for field welding of the containment steel liner plate, to ensure that all welding is performed in full compliance with the applicable job specifications.

3G.2 WELDING PERFORMED BY SUBCONTRACTOR

3G.2.1 WELDING PROCEDURES

All welding is performed in strict accordance with the applicable job specifications. All welding procedures, including the procedure qualification test records, used for the liner plate and penetrations are submitted to Southern Nuclear Operating Company (SNC) for approval. Production welding is not permitted without prior approval of those procedure and qualification records.

In all cases, the general contractor's welding inspector is responsible for ensuring that all subcontractors' welding is performed in accordance with the appropriate qualified welding procedures.

3G.2.2 WELDER QUALIFICATION

All welders and welding operators who perform welding under a code, standard, or specification that requires qualification of welders are tested and qualified accordingly prior to production welding. The subcontractor is responsible for testing and qualifying his own welders.

The general contractor's welding inspector is, in all cases, responsible for ensuring that the subcontractors' welders have successfully passed the necessary qualification tests, and that the subcontractor has the proper qualification test records for each qualified welder on file.

3G.3 NONDESTRUCTIVE EXAMINATIONS OF WELDMENTS

3G.3.1 VISUAL INSPECTION

The general contractor's welding inspector is responsible for carrying out the necessary surveillance to ensure that all welding meets the following requirements for visual inspection and general workmanship.

A. *Each weld is uniform in width and size throughout its full length. Each layer of weldment is smooth and free of cracks, pinholes, and undercut, and is completely fused to the adjacent weld beads and metals.*

In addition, the cover pass is free of coarse ripples, irregular surface, nonuniform bead pattern, high crown, and deep ridges or valleys between beads.

B. *Butt welds are of multipass construction, slightly convex, of uniform height, and have full penetration.*

C. *Butt welds made entirely from one side of the joint are welded against a backing strip.*

D. *Butt welds without backing strips are welded from both sides of the joints and are back-gouged to sound metal by arc-air, chipping, or grinding, before welding from the opposite side.*

E. *Fillet welds are of the specified sizes with full throat and legs of uniform length.*

F. *Where different base metal thicknesses are joined by welding, the finished joints are tapered no greater than one to four (1:4) between the thick and the thin sections.*

3G.3.2 RADIOGRAPHIC EXAMINATION

The general contractor's welding inspector is responsible for ensuring that all radiographic inspection is properly performed. He ensures that the correct radiographic techniques are followed and that the completed films are submitted to the licensee (SNC) for review, interpretation, and record.

The techniques for identification of radiographic examinations of welds are in accordance with Paragraph UW-51 of Section VIII of the 1968 ASME Code, using fine grain X-ray films.

The criteria for radiographic acceptance of welds are in accordance with Paragraph UW-52 of the 1968 ASME Code.

For quality control purposes for each welder's work, one spot radiograph 12 in. long is taken of the first 10 ft of welding completed on the liner in the flat, vertical, horizontal, and overhead positions. No further welding is permitted until this initial radiographic inspection has been completed and the welding is found to be acceptable by the general contractor's welding inspector.

Thereafter, a minimum of 2 percent of the welding is spot radiographed progressively as welding is performed, using 12- in.-long film, on a random basis as specified by the general contractor's welding inspector. This is done in such a manner that an approximately equal number of spot radiographs is taken from the work of each welder. Under conditions in which two or more welders make weld layers in a joint or on the two sides of a double-welded butt joint, one spot radiograph may represent the work of both welders.

If a radiograph discloses welding that does not comply with the minimum quality requirements, two additional spot radiographs 12 in. long are taken in the same weld seam at locations away from the original spot. The locations of these additional spots are determined by the general contractor's welding inspector. If the two additional spot radiographs show welding that meets the minimum quality requirements, the entire weld represented by the three radiographs is acceptable; however, the defective welding disclosed by the first of the three radiographs is removed and repaired by welding.

If either of the two additional spot radiographs shows welding that does not comply with the minimum quality requirements, the entire portion of the weld seam represented is rejected or, at the subcontractor's option, the entire weld seam represented is completely radiographed, and defective welding is corrected to meet the minimum quality requirements.

The rewelded joints or weld repaired areas are completely radiographed and must meet the minimum quality requirements.

3G.3.3 DYE PENETRANT EXAMINATION

The general contractor's welding inspector is responsible for ensuring that all dye penetrant inspection is properly performed. He sees to it that the correct techniques are followed, and that the results are properly interpreted.

Dye penetrant examination is used to examine 18 percent of all liner welds and 100 percent of all welds not vacuum box tested.

All dye penetrant inspection of welds is in accordance with Appendix VIII, "Methods for Liquid Penetrant Examination," of Section VIII of the 1968 ASME Code.

3G.3.4 MAGNETIC PARTICLE EXAMINATION

The general contractor's welding inspector is responsible for ensuring that all magnetic particle inspection is properly performed, that the correct techniques are followed, and that the results are properly interpreted.

Magnetic particle examination is used as an alternative to dye penetrant examination, at the subcontractor's option.

All magnetic particle inspection of welds is in accordance with Appendix VI, "Methods for Magnetic Particle Examinations," (dry particle) of Section VIII of the 1968 ASME Code.

3G.3.5 VACUUM BOX EXAMINATION

The general contractor's welding inspector is responsible for ensuring that all vacuum box inspection is properly performed, that the correct techniques are followed, and that the results are properly interpreted.

Vacuum box testing is used to examine 100 percent of all welds that must maintain leaktightness. Those welds that cannot be vacuum boxed because of configuration or space limitations are dye penetrant or magnetic particle examined.
The procedures for vacuum box inspection are as follows:

A. *The vacuum box is a portable unit and has a viewing window large enough to view the complete test area and to allow sufficient light to enter for proper examination. The box is capable of producing and holding a pressure differential of at least 8 psi.*

B. The leak detector solution is "Seam-Test Concentrate," manufactured by Winton Products Company, or "Test- Point" manufactured by United States Gulf Corporation.

C. Prior to testing, the test area is cleaned free of slag, scale, grease, paint, or any other materials that interfere with the testing procedures or the interpretation of the test.

D. Within 1 minute prior to testing, the leak detector solution is applied evenly over the entire test area.

E. The vacuum box then is put in place and evacuated to at least 5-psi pressure differential with respect to the atmospheric pressure.

The 5-psi minimum pressure differential is verified by a gage attached to the vacuum box unit and maintained for a minimum time of 20 seconds. The leak detector solution is continuously observed for bubbles that indicate leaks, from the time evacuation of the box is begun until 20 seconds after the required vacuum has been obtained.

F. All leaks regardless of size are repaired by completely removing the defect and rewelding the repaired area. A minimum of 2 in. on each side thereof is reinspected by vacuum box testing.

3G.3.6 HALOGEN LEAK INSPECTION (LEAK CHASE SYSTEM)

The general contractor's welding inspector is responsible for ensuring that all halogen leak inspection is properly performed, that the correct techniques are followed, and that the results are properly interpreted.

The halogen sniffer test is used at all welds that are fitted with a leak chase channel system. This system is installed over all welds that will not be accessible for inservice inspection during the plant's life. The halogen sniffer test is not used to examine the liner seam welds; it is used to verify the leaktightness of the liner to leak chase welds.

The procedure for halogen leak inspection of the leak chase system is as follows:

A. The standard high sensitivity industrial halogen leak detector is the General Electric Control Unit with a type H-2 gun probe. The refrigerant is R-12.

B. The test area is clean and dry, free of halides, cigarette smoke, welding smoke, and explosive vapors and is protected from sudden gusts of wind, foreign materials, or turbulent conditions.

C. The leak chase system test zone is pressurized with refrigerant to 15 psig. The pressure then is increased with air to 62 psig.

D. The halogen leak detector is calibrated against the halogen standard at the start of the testing and about every 2 hours during testing to determine that the instrument is capable of detecting the required minimum leakage of 1.0×10^{-5} *standard* cm^3/s.

E. The channel to liner plate weld joints are tested with the halogen leak detector at a sniffer rate of about 2-1/2 ft/min. with the halide gun held within 1/2 in. of the test surface.

F. All leaks detected at the established test sensitivity of 1.0 x 10^{-5} standard cm3/s are repaired and reinspected.

3G.7 PRESSURE DECAY TEST (LEAK CHASE SYSTEM)

The general contractor's welding inspector is responsible for ensuring that all pressure decay tests are properly performed, that the correct techniques are followed, and that the results are properly interpreted.

The pressure decay test is performed after the halogen sniffer test is conducted on the leak chase system. This is the test that verifies the leaktightness of all seam welds that are fitted with leak chases.

The procedure for pressure decay test is as follows:

A. After the halogen leak inspection of the leak chase system, the channel test zone is purged of the refrigerant a minimum of three times.

B. The channel test zone is pressurized to 62 psig and allowed to stand for 25 to 30 minutes prior to monitoring the pressure hold test.

C. Surface thermometers are installed on the channel test zone.

D. The pressure decay test is performed by holding the pressure for a minimum of 30 minutes.

E. The acceptance criterion is that the temperature corrected pressure loss is not greater than 2 psig per 25 to 30 min.

F. All channel test zones with unacceptable temperature- corrected pressure loss are repaired and retested.

3G.4 REPAIRS

It is the responsibility of the general contractor's welding inspector to determine that all weld defects are removed, repaired, and reinspected in accordance with the applicable job specifications.

3G.5 RECORDS

It is the responsibility of the general contractor's welding inspector to ensure that proper records of welding and nondestructive testing of the containment liner plate are kept on file.

3G.6 UNIT 2

Because the contract for Unit 2 was issued on February 1, 1972, some changes were made to the Unit 1 design documents to incorporate new or updated codes. The 1971 (with summer 1971 addenda) edition of the ASME Code was invoked, and sections of NRC Regulatory Guide No. 1-19 were included. Consequently, the following revisions to section 3.2 of appendix 3G are applicable to Unit 2 only:

A. *The criteria for radiographic acceptance of welds are in accordance with paragraph UW-51 of the 1971 ASME Code.*

B. *For quality control purposes the first 10 feet of welding is 100-percent radiographically examined for the flat, vertical, horizontal, and overhead positions of weld performed by each welder. No further welding is permitted until this initial radiographic inspection has been completed and the welding found to be acceptable.]*

[HISTORICAL] [3H. CONTAINMENT STRUCTURAL ACCEPTANCE TEST

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CONTAINMENT STRUCTURAL ACCEPTANCE TEST

3H.1 INTRODUCTION

The purpose of the containment structural acceptance test is to demonstrate that, when the containment is pressurized to the design loading, the deflections of the containment's structural elements and the cracks at the exterior surface concrete are within the acceptable limits. This confirms that the design and construction of the containment are adequate to withstand such pressure loading. The structural acceptance test will be performed in conjunction with the containment integrated leak rate test and will generally comply with Regulatory Guide 1.18 as discussed in appendix 3A.

A complete test procedure will be prepared and submitted to the NRC for review at least 90 days prior to conducting the structural acceptance test.

3H.2 TEST DESCRIPTION

3H.2.1 GENERAL

Prior to reactor fuel loading and operation, the integrity of the containment is demonstrated by a pressure proof test. The pressure test permits verification that the structural response due to the induced load is consistent with the predicted behavior. This is accomplished by measurements of the structure's deflections by the use of internally mounted taut wires.

3H.2.2 TEST PRESSURE

The pressure proof test is performed by subjecting the containment to a continuous increase in pressure from atmospheric pressure to 1.15 times the design pressure. Deflection measurements are recorded at atmospheric pressure, at 5 psi pressure increments during pressurization and depressurization cycles, and at the completion of depressurization. At maximum pressure level the pressure is held constant for at least 1 hour. Crack patterns are recorded at atmospheric pressure prior to the test, at maximum pressure, and following depressurization.

3H.2.3 DEFLECTION MEASUREMENTS

The structural deflections are measured with taut wire extensometers. Each extensometer consists of an invar wire spanning selected points, with one end (the dead end) fixed in position and the live end attached to a spring-loaded frame incorporating a linear potentiometer. The entire system spans the distance to be measured. The springs used are the "negator" type that apply an essentially constant force independent of extension. The springs selected apply a force of approximately 15 lb each, and they are used in matched pairs with a back-to-back mounting to avoid eccentricity. Accuracy of the extensometer is ±0.002 inch.

Radial deflections are measured along six equally spaced meridians at the spring line, at midheight of the cylinder, and at a point above the base slab at a height equal to three times the thickness of the wall at the location where the deflection is measured. The locations of these measurements are illustrated in figure 3H-1.

Vertical deflections are measured at the apex, at the spring line of the dome, and at three intermediate points. The locations of these measurements are shown in figure 3H-2.

The radial and tangential deflections of the containment wall adjacent to the equipment hatch are measured at 12 points as shown on figure 3H-3.

3H.2.4 CRACK PATTERNS

The patterns of cracks that exceed 0.01 inch in width at the exterior surface concrete are mapped near the base wall intersection, at midheight of the wall, at the springline of the dome, and around the equipment hatch. The crack patterns are mapped also at the intersection between a buttress and the wall, at the intersection between the top ring girder and the wall, and on the top shelf of the ring girder. At each location, an area of at least 40 ft^2 is mapped.

3H.2.5 STRAIN MEASUREMENTS

The Farley containment is similar to those of Turkey Point Unit 3 (Docket No. 50-250), Palisades Plant Unit 1 (Docket No. 50-255), and Point Beach Nuclear Power Plant Units 1 and 2 (Docket Nos. 50-266 and 50-301, respectively). The containments for both Turkey Point Unit 3 and Palisades Unit 1 were completely instrumented. The Turkey Point instruments provided approximately 400 strain measurements at 55 locations throughout the containment concrete and liner. In addition, about 55 taut wire measurements of structural deformation have been made. The Palisades instrumentation was comparable. The data from the instrumentation permitted detailed comparison between design calculations and observed response. The basic structural design and the accuracy of the calculation procedures used by Bechtel have therefore been verified by these tests. This verification is applicable to the Farley containment design.

Since the detailed confirmation of the design techniques has been made, strain gage instrumentation of the Farley containment is not required and it is concluded that no additional confirmation of design techniques is necessary.

3H.2.6 TESTING ENVIRONMENT

The structural acceptance test will be scheduled for periods in which extremely inclement weather is not forecast. However, due to the state of the art of weather forecasting, and the time involved in the preparation and performance of the test, should snow, heavy rain, or strong wind occur during the test, it may be continued and the results considered valid unless evidence indicates otherwise.

The environmental conditions during the test are measured to permit the evaluation of their contribution to the response of the containment. Atmospheric temperature, pressure, and humidity inside and outside the containment are monitored continuously during the test. In addition, the temperature inside and *outside the containment is measured at sufficiently long periods prior to the test to establish an average temperature of the wall for the evaluation of effects of temperature change on the deflection measurements.*

3H.2.7 CONDITIONS FOR REPEATING TEST

The test will be repeated under the following conditions:

A. *If the structural response deviates at any time during the test, up to a value that may jeopardize the containment integrity, the containment will be depressurized and the cause(s) for the deviation of response determined. If repair to the containment is necessary the test will be repeated.*

B. *If extremely inclement weather, such as snow, heavy rain, or strong wind, occurs during the test and the results of the test are found to be invalid, a retest will be performed.*

C. *If any significant modifications or repairs are made to the containment following the test, the test will be repeated.*

3H.3 STRUCTURE RESPONSE

The numerical values of the predicted structure response are established by the analytical techniques described in section 3.8.1 and appendix 3F. These values and the tolerances to be permitted in the acceptance test are developed as the result of the containment structural analysis and will be determined prior to the test.

3H.4 TEST REPORT

The following information will be included in the final test report:

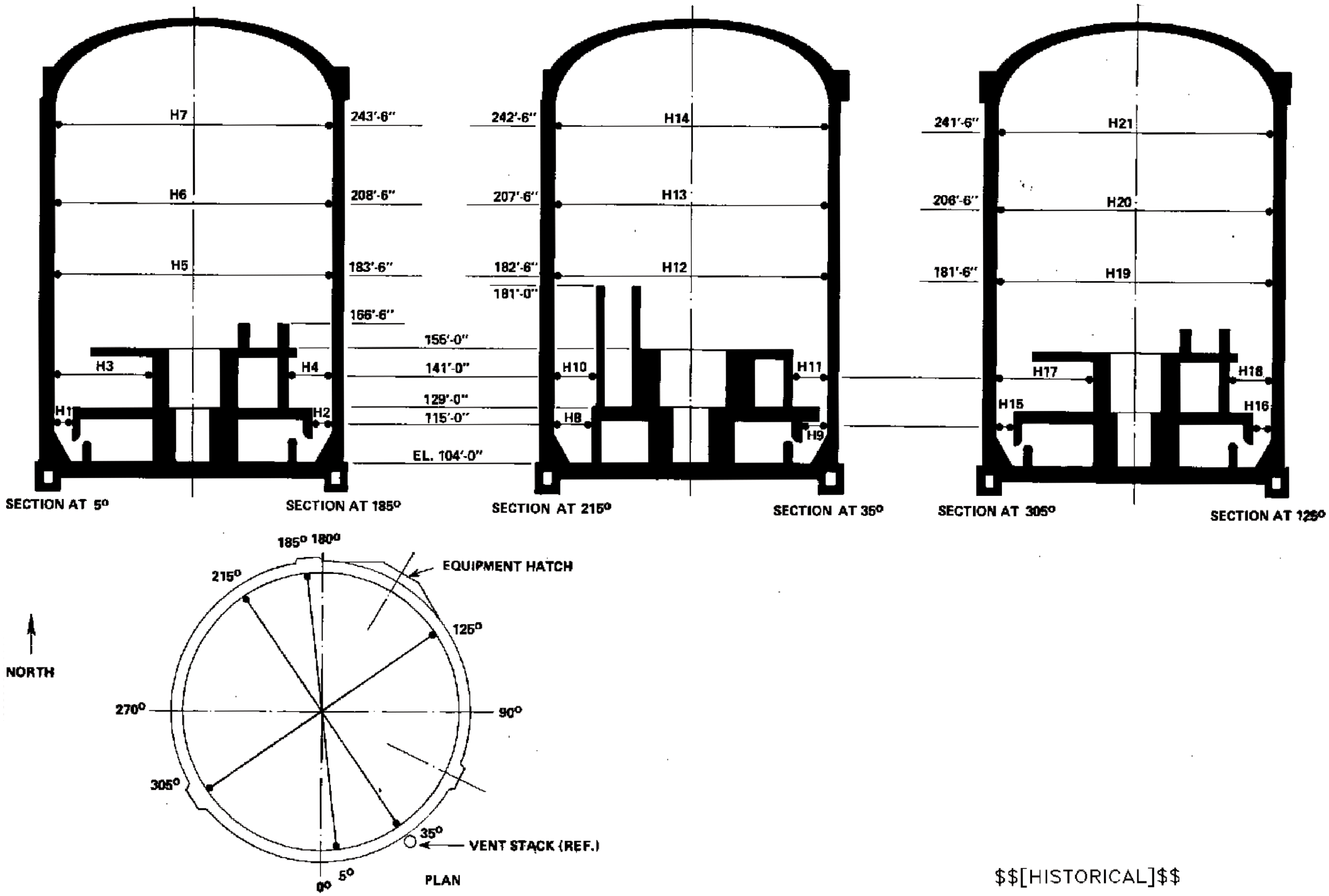
A. *A description of the test procedure and the taut wire system.*

B. *A comparison of the test measurements with the allowable limits (predicted response plus tolerance) for deflections and crack width.*

C. *An evaluation of the estimated accuracy of the measurements.*

D. *An evaluation of any deviations (i.e., test results that exceed the allowable limits), the disposition of the deviations, and the need for corrective measures.*

E. *A discussion of the calculated safety margin provided by the structure as deduced from the test results.* **]**



$$[HISTORICAL]$$

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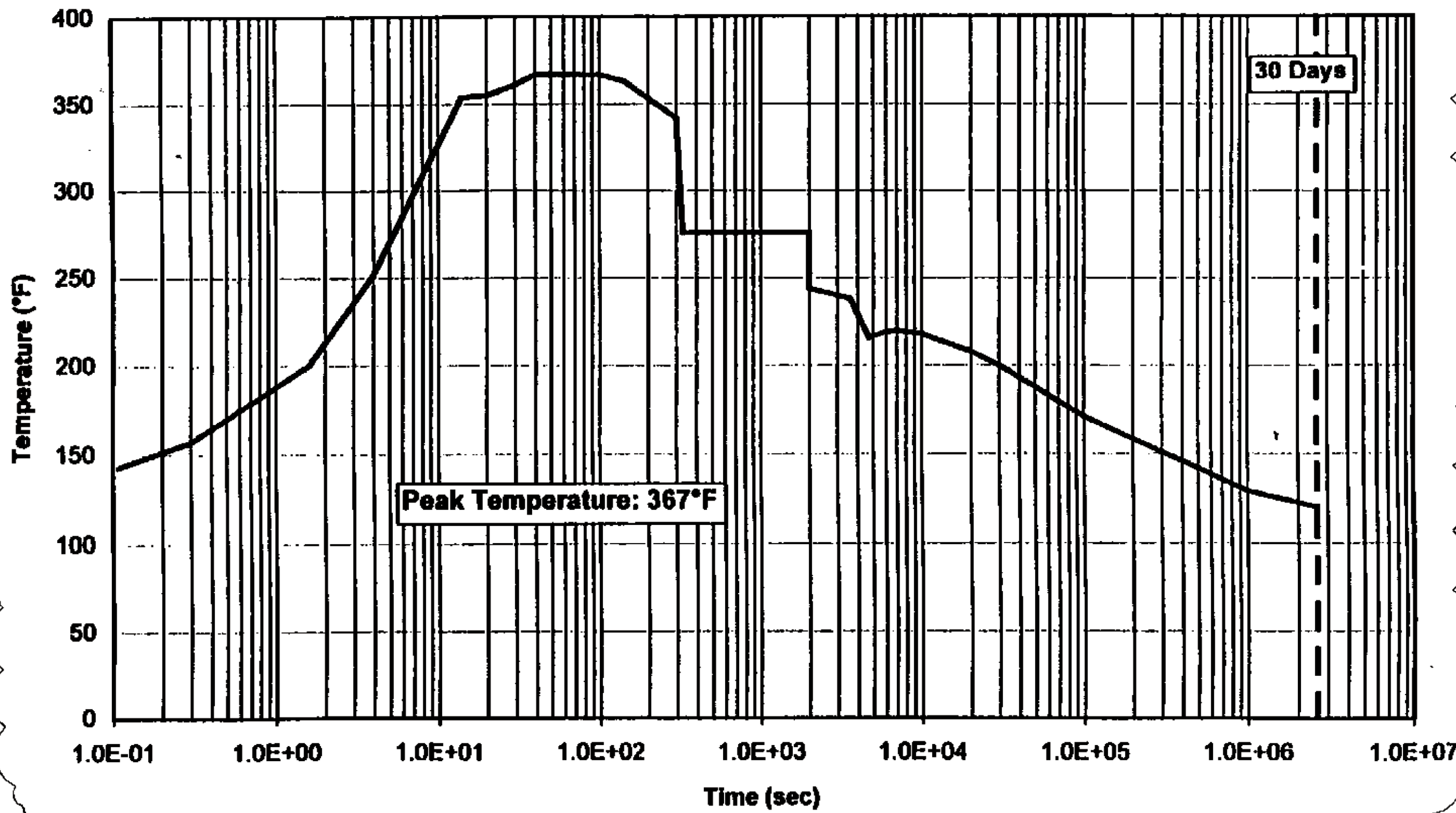
[TAUT WIRE DISPLACEMENT TRANSDUCER LOCATIONS

FIGURE 3H-1]



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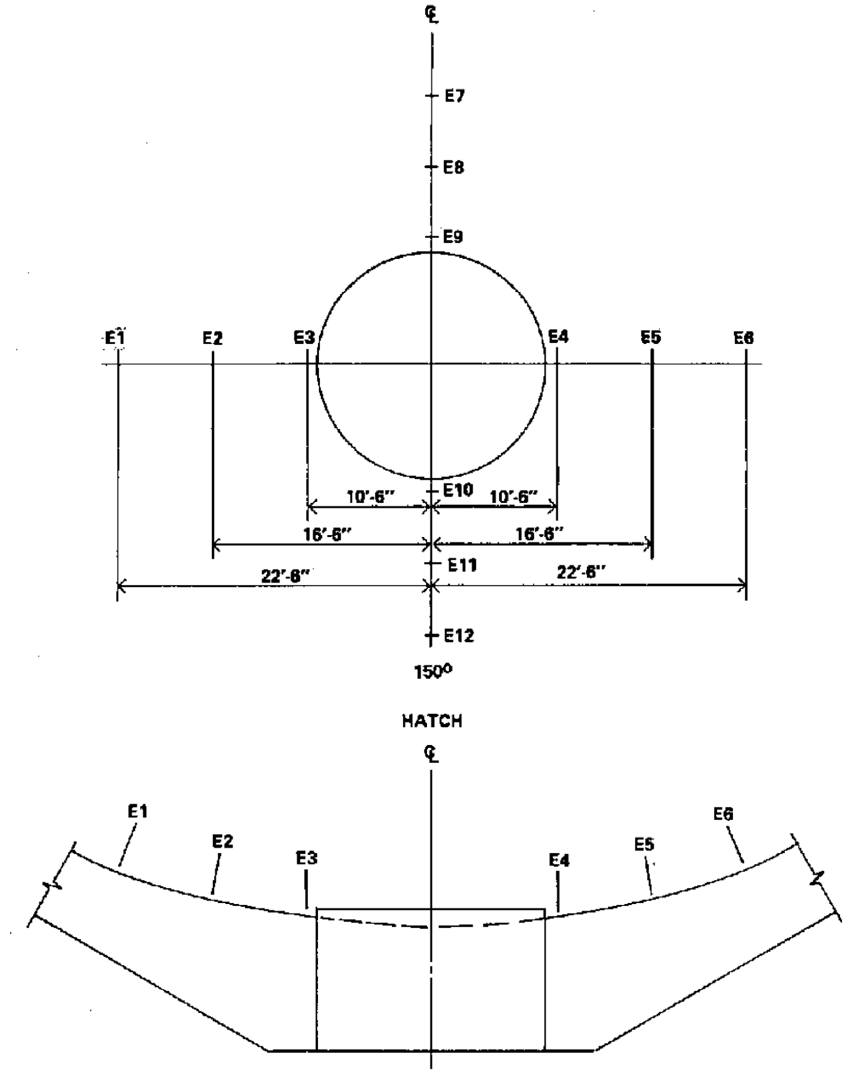
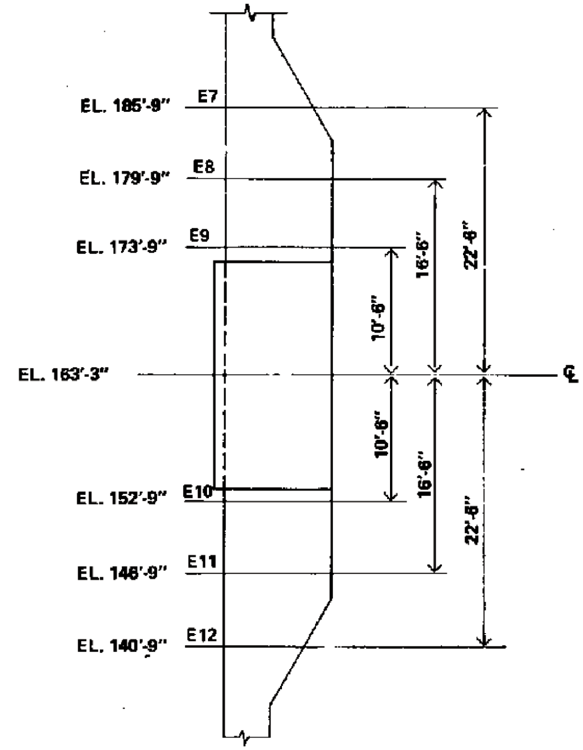
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	*[STRUCTURAL INTEGRITY TEST – DOME & VERTICAL DISPLACEMENTS MEASUREMENTS* *FIGURE 3H-2]*





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[STRUCTURAL INTEGRITY TEST – RADIAL DISPLACEMENT MEASUREMENTS AT EQUIPMENT HATCH

FIGURE 3H-3]

3I. LINER PLATE STABILITY

LIST OF FIGURES

APPENDIX 3I

LINER PLATE STABILITY

The stability of the liner plate was studied for the loading cases and deformations to which it may reasonably be subjected. The critical loading cases considered include the loss-of-coolant accident (LOCA) condition and the operating condition during the winter.

Two separate solutions of the plate stability were studied:

A. The plate as a compressed panel under biaxial compression, assuming that the channel and angle stiffeners are rigid in their attachment to the prestressed concrete containment and the liner.

B. The plate as a compressed panel under biaxial compression, assuming the panel to be a portion of a large cylinder with a flexible stiffener system.

Figure 3.8-1, Detail 2, illustrates the actual physical configuration of the stiffening system used for the liner plate. The channels function as horizontal stiffeners and the angles as vertical stiffeners.

For the solution, an initially deflected form for the liner plate is expressed in terms of a Fourier series of the form

$$\Delta = \sum_{n=0}^{\infty} \sum_{m=1}^{\infty} \Delta_{mn} \cos m\theta \sin \frac{n\pi\chi}{\ell}$$

where θ defines the central angle in a plan view of the cylinder, from a point on the circumference where there is zero radial deflection to the point on the circumference where there is maximum radial deflection; Δ defines the radial deflection and ℓ defines the unsupported length in the vertical direction.

Under normal operating conditions, the overall structural stability of the liner plate is maintained.

The most critical stress for the liner plate exists in the condition illustrated by figure 3I-2. In this condition, Panel 1 and Panel 3 have outward initial curvature and Panel 2 has inward initial curvature. When a load is applied parallel to the liner plate, Panels 1 and 3 bear against the concrete and Panel 2 deforms inward. If the load is primarily from concrete shrinkage creep, prestress, and thermal effects, the membrane stress (N/t) in Panels 1 and 3 tends to relax to a value of (N-ΔN/t) in Panel 2. The anchors between the panels with inward and outward curvature must restrain a force of ΔN for static equilibrium. Due to inward deformation, flexural stress also exists in Panel 2 and the anchors are subjected to the moment (M). (See figure 3I-2.)

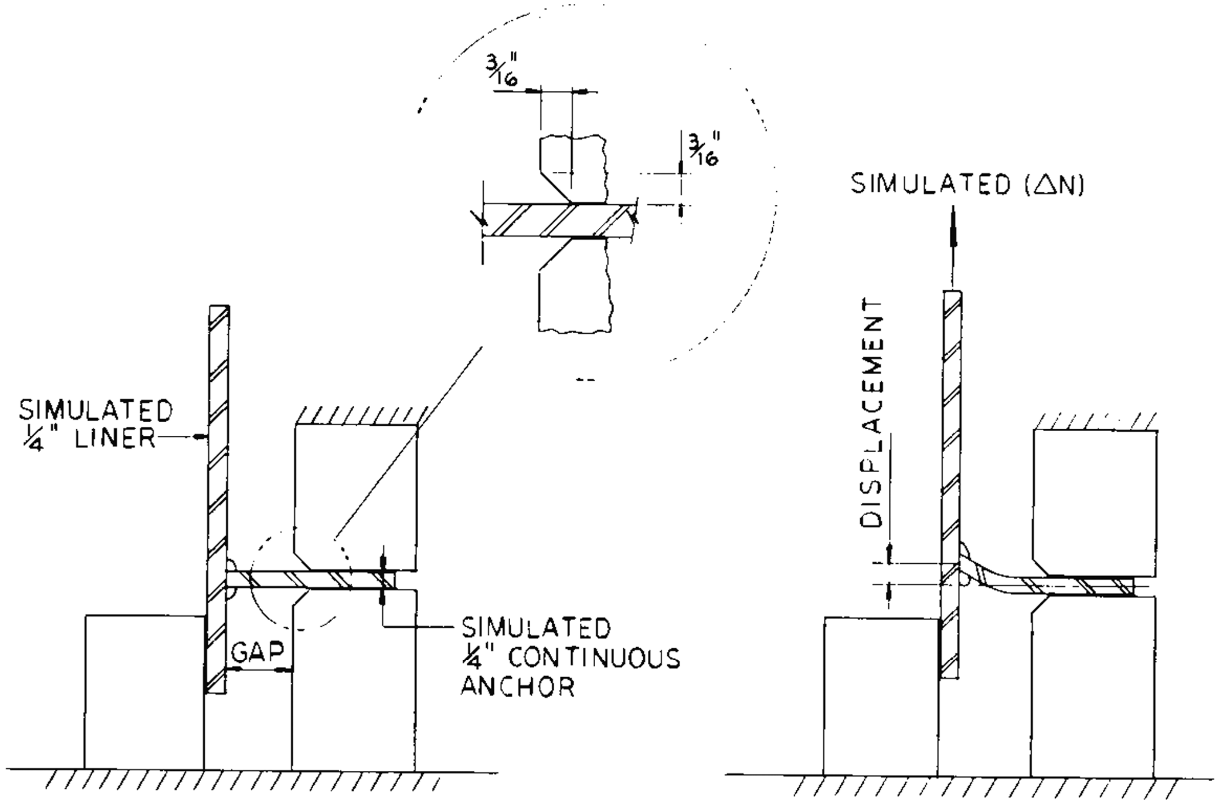
The maximum compressive strains are caused by accident pressure, thermal loading, prestress, shrinkage, and creep. The maximum calculated strains do not exceed 0.0025 in./in., and the liner plate always remains in a stable condition.

The anchorage has the capability of resisting the full force (ΔN) due to a theoretically fixed anchor, but in addition it has sufficient ductility to accept the 0.038-in. displacement without failure. The above displacement results from a uniform membrane strain of 0.0025-in./in. distributed over a 15-in. anchor spacing. Various patterns of welds attaching the angle anchors to the liner plate have been tested for ductility and strength when subjected to a transverse shear load such as ΔN and are shown in figure 3I-1.

Also of concern is the nature of the state of stress and behavior at the point of attachment between the stiffeners and the liner plate. Special tests[1] have been conducted on simulated models of the liner plate and vertical stiffener assembly to determine the shear capacity of the angle anchorage. The results of these tests and the various weld configurations are shown in figure 3I-1. Note in the test results that two different configurations of support were used for the simulated continuous anchor. The case in which the 0-in. gap was used simulates the expected condition that exists in the containment. The case with the 5/8-in. gap attempts to simulate the condition that might exist at an isolated location if the concrete were not in continuous contact with the anchor. Being guided in the proportioning of the liner plate stiffeners by the values of shear transfer for the case of the 5/8-in. gap will, in general, result in a margin of safety for progressive failure of anchors of approximately 2.7. The weld configuration shown in figure 3.8-1, Detail 2, is adequate to transfer all loads that are considered in the design on the containment between the liner plate and the stiffener-anchors.

REFERENCE

1. Liner Plate Anchorage Tests for Job No. 6600 Arkansas Nuclear One, Arkansas Power & Light Company; Job No. 6292 Rancho Seco Nuclear Station - Unit 1 Sacramento Municipal Utilities District; Job No. 6750 Calvert Cliffs - Units 1 and 2, Baltimore Gas & Electric Company.... Prepared by Bechtel Corporation, San Francisco, Calif. (April 18, 1969).



BEFORE LOAD

DURING LOAD

WELD CONFIGURATION	GAP (IN.)	ULTIMATE LOAD (K/IN)	ULTIMATE DISPLACEMENT (IN.)	LOCATION OF FAILURE
3/16	0	14.95	.14	LINER PLATE
3/16	5/8	5.56	.68	ANCHOR WELD
3/16 6-12	0	7.65	.18	ANCHOR WELD
3/16 6-12	5/8	2.93	.60	ANCHOR WELD
3/16 4-12	0	6.87	.18	ANCHOR WELD
3/16 4-12	5/8	2.46	.30	ANCHOR WELD

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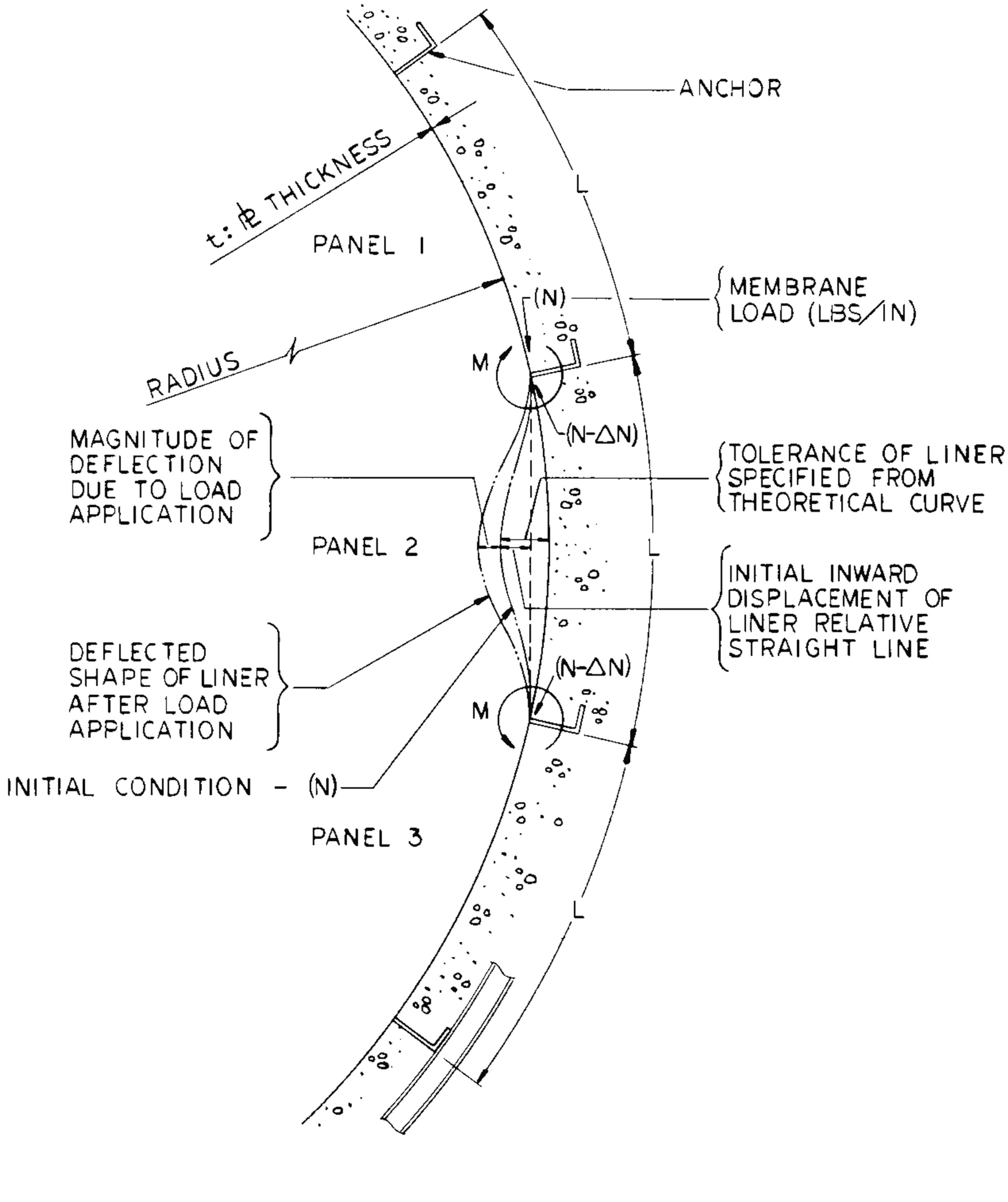
LINER – STIFFENER WELD TEST RESULTS

FIGURE 3I-1



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LINER PLATE LOADING CONDITIONS

FIGURE 3I-2

3J MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT PRESSURE AND TEMPERATURE ANALYSIS

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APPENDIX 3J

MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT PRESSURE AND TEMPERATURE ANALYSIS

3J.0 BACKGROUND

IE Information Notice 84-90, "Main Steam Line Break Effect on Environmental Qualification of Equipment," informed licensees of a concern with analyses of main steam line breaks (MSLB). IE Information Notice 84-90 stated that the assumption of large breaks bounding all others was not true for temperature effects and that smaller breaks would actually result in higher temperatures in the steam released from the break. The higher temperatures would result from steam remaining in the steam generator for longer periods than previously assumed due to the steam exiting the break at a slower rate. As steam generator water level decreases due to break flow, the secondary water level could allow tube bundle uncovery. Consequently, the temperature of the steam being generated would approach the reactor coolant system (RCS) temperature and become superheated before exiting the steam generator and the break in the main steam line.

In response to this issue, analyses were performed for a spectrum of break sizes using a more advanced computer code than was available at the time of the analyses presented in appendix 3K. The new analyses supersede the appendix 3K analyses with respect to the main steam valve room pressure and temperature response to postulated main steam line breaks.

3J.1 FNP APPLICABILITY

The maximum blowdown from any steam generator would be equivalent to that produced by a 1.069 ft^2 break due to the integral flow restrictors on the outlet nozzle for each steam generator. Since a MSLB downstream of the main steam isolation valves (MSIV) could result in blowdown from all three steam generators prior to MSIV closure, the maximum equivalent break size is 3.207 ft^2.

The MSLB for Farley Nuclear Plant (FNP) is not postulated for main steam piping outside containment up to and including the MSIVs. The basis for this position is discussed below. Paragraph 3.6.2.4 lists the specific location criteria for breakpoints in ASME Section III, Class 2 and 3 lines, with reference to appendix 3K. Appendix 3K describes the criteria for postulating pipe ruptures or cracks in high-energy lines outside containment and the methods for evaluating the effects of these breaks. Attachment A, part II of appendix 3K specifically addresses the postulated break and leakage locations in the main steam lines outside of containment and demonstrates that this piping conforms to Branch Technical Positions (BTP) APCSB 3-1 and MEB 3-1 of Standard Review Plan Sections 3.6.1 and 3.6.2, respectively. This is consistent with Section 3.6 to Supplement 1 of the NRC Safety Evaluation Report for the Joseph M. Farley Nuclear Plant - Units 1 and 2.

The only postulated break upstream of the MSIVs is the 3-in. diameter branch line to the turbine-driven auxiliary feedwater. This line is not part of the "no break zone" and a break must be postulated in this line as part of the FNP licensing basis. Thus, the only break upstream of the MSIVs which must be considered for FNP is the 3-in. branch line to the turbine-driven auxiliary feedwater pump.

3J.2 WESTINGHOUSE OWNERS GROUP EFFORTS

In response to the NRC concern, the Westinghouse Owners Group (WOG) formed the High-Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC) subgroup. The WOG determined mass/energy release data corresponding to a full spectrum of breaks (0.05 ft^2 to 4.6 ft^2 at 70-percent and 100-percent power). The results of the WOG analysis are presented in WCAP-10961 (reference 1).

In support of the Farley Nuclear Plant power uprating, the full spectrum of steamline breaks outside containment with superheated steam blowdown was reanalyzed. Mass/energy release data at 70-percent and 102-percent power were revised using FNP plant-specific assumptions including the increased power level. The revised blowdown analysis is presented in WCAP-14722 (reference 4) and supersedes the blowdown results documented in WCAP-10961.

In support of the Farley Nuclear Plant steam generator replacement, a limited spectrum of steam line breaks outside containment with superheated steam blowdown, based on the power uprating analysis, was reanalyzed. Mass/energy release data at 70-percent and 102-percent powers were again calculated using FNP plant-specific assumptions including those associated with the replacement steam generators. The revised blowdown analysis is presented in WCAP-15097 (reference 5) and supersedes the blowdown results documented in WCAP-14722.

MSIV closure time is a significant parameter in determination of the consequences of a postulated MSLB in the main steam lines because, for breaks downstream of the MSIVs, main steam line isolation terminates the blowdown. Additionally, if MSIV closure occurs prior to tube bundle uncovery, the transient will not result in any superheated blowdown. For the power uprating and steam generator replacement analyses, Westinghouse determined that the earliest actuation of MSIV closure for Farley Nuclear Plant is produced by a low-steam pressure signal for breaks outside containment.

Farley Nuclear Plant pressure transmitters are not located in an area where they would be subjected to the harsh environment during a steam line break. Thus, the Farley Nuclear Plant pressure transmitters can be assumed to operate with normal error allowances, and MSIV closure for Farley Nuclear Plant would occur much sooner than for a plant with pressure transmitters located in a harsh environment with corresponding environmentally-induced errors. Accordingly, Westinghouse provided the appropriate information in WCAP-10961 to determine the specific MSIV closure time corresponding to various break sizes for FNP. The following discussion addresses the specific low steam line pressure setpoint for FNP.

The FNP Technical Specification nominal trip setpoint for low-steam line pressure is 585 psig. FSAR paragraph 7.3.1.2 specifies a historical $\pm$ 4-percent actuation signal accuracy for a range of 0 to 1200 psig. Therefore, a 48-psi inaccuracy was conservatively applied to the nominal trip setpoint, which results in a safety analysis limit (SAL) of 537 psig for safety injection and main steamline isolation by low steam line pressure. Setpoint uncertainty calculations for these

ESFAS functions demonstrate adequate margin between the SAL and the corresponding nominal trip setpoint. In addition, the dynamic signal compensation specified in the Technical Specifications is explicitly modeled in the safety analysis and conservatively implemented by plant procedures. Therefore, FNP can be assured that the MSIV ESFAS actuation signal will close when steam pressure falls to 537 psig (551.7 psia). The power uprating and steam generator replacement analyses provide sufficient information to demonstrate that the FNP MSIVs will close prior to tube bundle uncovery for all breaks of 0.6 ft^2 and larger. Therefore, superheated blowdown will not occur for 0.6 ft^2 and larger breaks.

The cases analyzed for Farley Nuclear Plant in the power uprating and steam generator replacement analyses were the 3.2, 2.0, 1.4, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, 0.1, and 0.05-ft^2 breaks at 70-percent and 102-percent power. Due to the integral exit nozzle flow restrictors on each steam generator for FNP, only breaks of 3.207 ft^2 and smaller apply to Farley Nuclear Plant. See section 3J.4 for the discussion of the Model 54F replacement steam generators.

As discussed in section 3J.1, the 3-in. diameter branch line to the turbine-driven auxiliary feedwater pump is not considered a part of the "no break zone" and must be postulated to break. Westinghouse analyzed the 0.05-ft^2 break and presented the results in WCAP-15097 (reference 5). Because MSIV closure is not automatically initiated for this break, the results of the power uprating and steam generator replacement analyses for the time period analyzed (i.e., 0-1800 s) are applicable to a break either downstream or upstream of the MSIVs.

Due to the relatively low blowdown rate of the 0.05-ft^2 break, auxiliary feedwater flow is sufficient to delay tube bundle uncovery for nearly 1800 s. At this point, operator action to close the MSIVs is assumed and the transient for the downstream break is terminated. However, the break postulated upstream of the MSIV (a break in the branch line to the turbine-driven auxiliary feedwater pump) is not isolated by MSIV closure. Termination of this transient requires MSIV closure and termination of auxiliary feedwater flow to the steam generator with the faulted line. The steam blowdown through this break would then continue until steam generator dryout.

3J.3 APPLICATION OF WOG BLOWDOWN DATA

The results of the WOG HELB/SBOC analysis are presented in WCAP-10961. Blowdown data from WCAP-10961 was analyzed by Westinghouse to determine compartment temperatures using the COMPACT code. COMPACT is a multinode containment code developed by Westinghouse for analysis of containment and outside- containment compartment transients. This code models the mass transfer between an upper gaseous region and a lower sump region within each node. The code also models natural circulation flow induced by large temperature gradients, which promotes better compartment gas mixing and, hence, more uniform temperature distribution within the compartments.

The full spectrum of breaks for Farley Nuclear Plant was analyzed (0.05 ft^2 to 4.6 ft^2 at 70-percent and 102-percent power) even though WCAP-10961 indicated that only the 0.2 ft^2 and smaller breaks may produce superheated steam. This approach was taken to ensure a consistent methodology basis for all postulated steam line breaks in the main steam valve room (MSVR). Westinghouse determined that the results from the analysis of nine breaks would envelop the environmental conditions for all postulated breaks. The breaks downstream of the

MSIVs analyzed by Westinghouse were the 0.2-ft^2 break at 70-percent power and the 0.2, 0.4, 0.6, 0.8, 1.0, 1.1, and 4.6-ft^2 breaks at 102-percent power. Westinghouse also analyzed the 0.05-ft^2 break at 102-percent power upstream of the MSIVs. The results of these analyses are documented in WCAP-11652 (reference 2).

The break location was selected in the lower portion of the MSVR. The break compartment (Compartment 1) volume was selected to represent a volume occupied by high temperature steam exiting the break in the lower portion of the MSVR. The selection of break location and small break compartment volume ensures that the calculated gas temperatures in the MSVR would be conservative.

For all breaks downstream of the MSIVs, the mass and energy release data from WCAP-10961 were applied. For the 0.05-ft^2 break upstream of the MSIVs, the mass and energy releases were based on the 102-percent power Case 67 of Category 4 in WCAP-10961. These releases are utilized until 1800 s, when it is assumed that the operator takes action to close the MSIVs and isolate auxiliary feedwater to the faulted steam generator. The mass and energy releases following the operator actions assume a conservative linear blowdown until steam generator dryout and take no credit for continued cooldown of the RCS.

During winter months, plastic sheeting may be applied to the exterior of penthouse grating to prevent the freezing of MSVR instruments. Application of the plastic sheeting is discussed in appendix 3K. The plastic sheeting is installed such that the maximum pressure required to tear the sheeting away is 1.25 psig. The impact of the installation of plastic sheeting on the pressure and temperature transients was considered in the WCAP-11652 and WCAP-15560 analyses.

The results of the Westinghouse analyses demonstrate that the calculated environmental temperatures in the break compartment do not exceed 325°F for a wide range of break sizes and power levels. Since no credit was taken for heat removal by concrete and steel structural heat sinks, the results of the calculations are very conservative.

The results presented in WCAP-11652 demonstrated that for the spectrum of breaks analyzed in this WCAP, the peak temperatures produced by superheat are not a problem for FNP since the existing MSVR analysis peak temperature of 308°F and the existing environmental qualification temperature profile are greater than those produced by the 0.2-ft^2 and smaller breaks. However, the WCAP-11652 results indicate that the larger breaks, which do not produce superheated blowdown, result in peak temperatures which exceed the previous temperature profiles. The most limiting case identified by this Westinghouse analysis, with regard to peak temperature, is the 0.8-ft^2 break at 102-percent power, which yields a maximum temperature of slightly less than 325°F. Breaks larger than 0.8 ft^2 are less limiting due to rapid MSIV closure and the corresponding termination of blowdown. The results of this WCAP determined that the thermal transient induced by breaks smaller than 0.8 ft^2 were less limiting due to the reduction in mass flowrate and by the early onset of natural circulation.

The effects of long term steam releases, during postulated main steam line ruptures, on outside-containment equipment environmental qualification are presented in WCAP-15560. This analysis alters the assumptions that maximum superheated steam releases by maximizing the total mass and energy released over time into the MSVR. These analyses expand the power-level/break-area spectrum used in previous superheated steam mass and energy analyses and revise plant parameters to maximize the total energy released through each postulated steam line rupture. In effect, the analyses presented in WCAP-15560 revised the inputs for the initial steam generator inventory and main feedwater flowrates to develop bounding EQ requirements for the MSVR when used in combination with the prior superheated steam releases. The results of the steam line break analysis presented in WCAP-15560 represent the revised basis for the bounding EQ temperature envelope.

The combined temperature profile, presented in WCAP-15560, for all of the analyzed cases is shown in figure 3J-10. MSVR equipment which is required to be environmentally qualified has been reviewed against the temperature profile and found to be acceptable.

Although the pressure transient associated with the release of superheated steam was never considered to be an issue for NRC Information Notice 84-90, the WOG blowdown data was analyzed to determine the peak pressure which would result from the new analysis. Westinghouse analyzed the pressure transient associated with 4.6, 1.1, 0.2, and 0.05-ft^2 breaks of 102-percent power. These four breaks were determined by Westinghouse to envelope the pressure transient associated with the spectrum of breaks applicable to FNP because they include the two largest and two smallest flowrates. The analysis indicates the pressure transient resulting from these breaks is slight due to the large venting area available to the MSVR.

The composite pressure profile is shown on figure 3J-15. As indicated on the graph, MSVR pressure does not exceed 16 psia for any MSLB. The only discernible pressure increase is the 1.25-psi pulse required to clear the sheeting from the grating. The new maximum pressure of less than 16 psia is well below the peak pressure of 20.5 psia from the previous MSVR pressure analysis. A comparison of the mass and energy releases for uprated conditions (reference 4) to those of the previous evaluations (references 1 through 3) indicates that the blowdown for uprated conditions remains bounded by the previous analyses. Accordingly, the new pressure analysis did not impact MSVR equipment qualification or structural integrity.

3J.4 Discussion of Results for Model 54F Replacement Steam Generators

The original design basis analyses for main steam line breaks outside the containment were documented in WCAP-11652, Rev. 2 and in WCAP-14013. The spectrum of cases that was presented in these documents was used as the basis for determining the break spectrum for the Model 54F replacement steam generators (SG). Since maximizing the amount of superheat was the primary consideration, the following nine cases were studied (reference 6) for the impact of the Model 54F SGs on the main steam valve room (MSVR) post-accident temperature profile.

Case 1:	0.4 ft^2 break area at 102% power
Case 2:	0.3 ft^2 break area at 102% power
Case 3:	0.2 ft^2 break area at 102% power
Case 4:	0.1 ft^2 break area at 102% power
Case 5:	0.05 ft^2 break area at 102% power
Case 6:	0.3 ft^2 break area at 70% power
Case 7:	0.2 ft^2 break area at 70% power
Case 8:	0.1 ft^2 break area at 70% power
Case 9:	0.05 ft^2 break area at 70% power

There are two cases (Case 5 and Case 9) that are upstream of the MSIV for the Farley units, which means that the break releases cannot be terminated by MSIV closure. These cases model a very small break (0.05 ft^2) in which the releases continue for close to one hour, until the faulted SG is emptied. Because of the relatively low flow rates, these cases are among the least limiting in terms of the peak compartment temperature. However, the releases last for the longest time, and thus these cases define the compartment temperature envelope as it returns to normal temperatures.

The case which yields the most limiting compartment temperature is Case 1. The peak temperature in the MSVR for this case is 320.11°F. This is less than the maximum temperature of approximately 325°F that occurs for the original Model 51 steam generators. Figure 3J-16 provides a comparison of Cases 1 through 5 at 102% power to the Equipment Temperature Envelope. It can be seen that all of these cases are within the temperature limit. Figure 3J-17 provides a similar comparison with Cases 6 through 9 at 70% power. This comparison also shows that the cases for the model 54F result in compartment temperatures that are within the temperature envelope.

Thus, the model 54F replacement steam generators do not impact the MSVR equipment qualification or the structural integrity.

REFERENCES

1. WCAP-10961, Revision 1, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," Proprietary, October 1985.

2. WCAP-11652, Revision 2, "Joseph M. Farley Nuclear Station Units 1 and 2 Main Steam Valve Room Temperature Response to Superheated Steam Releases," Proprietary, June 1988.

3. WCAP-14013, "Joseph M. Farley Nuclear Station Units 1 and 2 Main Steam Valve Room Temperature Response to Superheated Steam," March, 1994.

4. WCAP-14722 (Proprietary), “Farley Nuclear Plant Units 1 and 2 Power Uprate Project NSSS Engineering Report,” November 5, 1997.

5. WCAP-15097 (Proprietary), “Farley Nuclear Plant Units 1 and 2 Replacement Steam Generator Program NSSS Engineering Report,” November 1998.

6. Westinghouse Letter, ALA-98-233, Rev. 1, Southern Nuclear Operating Company, Joseph M. Farley Nuclear Plant, Units 1 and 2, “Main Steam Valve Room Analyses for Main Steam Line Breaks, Revision 1, “November 17, 1998.

7. WCAP-15560, “Joseph M. Farley Nuclear Station Units 1 and 2 Main Steam Valve Room Analysis for Steam Line Breaks Outside Containment”, February 2001.