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	3.8-168	02-01	May 2002		3.8-30a	0	August 1984
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	3.8-55	0	August 1984		3.9-18	02-01	May 2002
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	3.8-57	0	August 1984		3.9-20	RN09-022	November 2011
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	3.8-64	0	August 1984		3.9-27	02-01	May 2002
	3.8-65	0	August 1984		3.9-28	02-01	May 2002
	3.8-66	0	August 1984		3.9-29	02-01	May 2002
	3.8-67	02-01	May 2002		3.9-30	RN09-022	November 2011
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	3.8-69	0	August 1984		3.9-32	02-01	May 2002
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	3.8-71	0	August 1984		3.9-34	02-01	May 2002
	3.8-72	0	August 1984		3.9-35	02-01	May 2002
	3.8-73	RN11-027	May 2013		3.9-36	02-01	May 2002
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	3.9-8	02-01	May 2002		3.9-44	02-01	May 2002
	3.9-9	02-01	May 2002		3.9-45	02-01	May 2002
	3.9-10	02-01	May 2002		3.9-46	02-01	May 2002
	3.9-11	02-01	May 2002		3.9-47	02-01	May 2002
	3.9-12	02-01	May 2002		3.9-48	02-01	May 2002
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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

The chapter includes identification, description and discussion of the principal architectural and engineering design of those structures, components, equipment and systems important to safety.

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

This section discusses briefly the extent to which the design criteria for the plant structures, systems and components important to safety meet the NRC "General Design Criteria for Nuclear Power Plants" specified in Appendix A to 10 CFR 50. A summary is provided to show how the principal design features meet each criterion and identify any exceptions.

3.1.1 SUMMARY DESCRIPTION

The Virgil C. Summer Nuclear Station is designed, constructed and operated to comply with South Carolina Electric and Gas Company's understanding of the intent of the NRC's "General Design Criteria for Nuclear Power Plants," Appendix A to 10 CFR 50. This section outlines the philosophy that will be adopted in meeting the criteria. Detailed evaluations of compliance with the various General Design Criteria are incorporated in applicable sections of the Final Safety Analysis Report as referenced herein.

3.1.2 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.2.1 Overall Requirements

Criterion 1 - Quality Standards and Records

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

Discussion

South Carolina Electric and Gas Company and Westinghouse, with its subcontractors, maintains, either in their possession or under their control, a complete set of records of the design, fabrication, construction and testing of safety components. Recognized codes and standards, when used, are identified and evaluated to assure their applicability, adequacy, and sufficiency in keeping with the required safety function.

The quality assurance program conforms with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Plants." This program is discussed in Chapter 17. Chapter 14 describes the initial test program to assure performance of installed equipment commensurate with the importance of the safety function.

Criterion 2 - Design Bases for Protection Against Natural Phenomena

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

Discussion

The natural phenomena and their magnitude are selected in accordance with their probability of occurrence at this specific site. The criteria adopted in the design of affected structures, systems and components, also depend on the likelihood of the natural phenomenon under consideration. 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 the historical data. The natural phenomena postulated in the design are presented in Chapter 2. The design criteria for the structures, systems and components affected by each natural phenomenon are presented in the sections listed below. These sections also identify which combinations of natural and plant originated accidents are considered in the design.

The design criteria developed meet the requirements of Criterion 2.

For further discussion, see the following sections:

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		Section
1.	Meteorology	2.3
2.	Hydrologic Engineering	2.4
3.	Geology, Seismology, and Geotechnical Engineering	2.5
4.	Classification of Structures, Components and Systems	3.2
5.	Wind and Tornado Loadings	3.3
6.	Water Level (Flood) Design	3.4
7.	Missile Protection	3.5
8.	Seismic Design	3.7
9.	Design of Category I Structures	3.8
10.	Mechanical Systems and Components	3.9
11.	Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment	3.10
12.	Environmental Design of Mechanical and Electrical Equipment	3.11

Criterion 3 - Fire Protection

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

Discussion

Fires in the plant are prevented or mitigated by the use of noncombustible and fire retardant materials. Redundant safety class equipment is separated by 1 or 2 hour fire barriers or adequate clear space separation based upon the fire hazard. Safety-related ventilation systems are designed and arranged to ensure that no single fire occurrence will jeopardize plant safety. In addition, special attention is given to the safety-related electrical systems by providing items such as metal cabinets, metal wireways and fire retardant insulation.

Cabling within trays is suitably derated and cable tray loading is designed to minimize internal heat buildup. Cable trays are suitably separated to avoid the loss of redundant channels of protection cabling should fires occur. The arrangement of equipment in protection channels assigned to separate cabinets provides physical separation and minimizes the effects of a possible fire.

Combustible supplies such as logs, records, manuals, etc., are limited in such areas as the Control Room to amounts required for current operation, thus minimizing the effect of a fire. No explosive gases or flammable liquids exist in these areas and therefore no explosion hazard exists.

The Plant Fire Protection System includes the following provisions:

1. Automatic fire detection equipment in those areas where fire danger is greatest.

2. Physical barriers and fire rated walls.

3. Automatic extinguishing systems for areas of highest fire loading as well as manually operated fire extinguishers for all areas.

4. Fire alarms and detection devices are connected to the Station Communication System in the Control Room.

The Westinghouse supplied equipment is designed to minimize the probability and effect of fires and explosions. Noncombustible and fire resistant materials are used in this equipment wherever practical.

The requirements of the National Fire Protection Association, the American Insurance Association, the Nuclear Energy-Liability Property Insurance Association (NE-LPIA), and the applicable local codes and regulations are observed in the design and installation of Westinghouse supplied equipment.

The design of the Fire Protection System thus meets the requirements of Criterion 3.

For further discussion, see the following sections:

		Section
1.	Instrumentation and Controls	7.0
2.	Electric Power	8.0
3.	Separation Criteria	8.3.1
4.	Fire Protection System	9.5.1
5.	Conduct of Operations	13.0

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Criterion 4 - Environmental and Missile Design Bases

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

Discussion

Structures, systems and components important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents (LOCA).

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

The electrical equipment, instrumentation and cables for protection of engineered safety feature systems which are located inside the Reactor Building are discussed in the sections listed below indicating the design requirements in terms of the time which each must survive the extreme environmental conditions following a loss of coolant accidents or a main steam line break.

The design of these structures, systems, and components meets the requirements of Criterion 4.

For further discussion, see the following sections:

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		Section
1.	Meteorology	2.3
2.	Hydrologic Engineering	2.4
3.	Geology, Seismology, and Geotechnical Engineering	2.5
4.	Classification of Structures, Components and Systems	3.2
5.	Wind and Tornado Loadings	3.3
6.	Water Level (Flood) Design	3.4
7.	Missile Protection	3.5
8.	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping	3.6
9.	Seismic Design	3.7
10.	Design of Category I Structures	3.8
11.	Mechanical Systems and Components	3.9
12.	Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment	3.10
13.	Environmental Design of Mechanical and Electrical Equipment	3.11
14.	Integrity of Reactor Coolant Pressure Boundary	5.2
15.	Engineered Safety Features	6.0
16.	Instrumentation and Controls	7.0
17.	Electric Power	8.0
18.	Main Steam System	10.3

Criterion 5 - Sharing of Structures, Systems, and Components

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

Discussion

No evaluation is required since this is a 1 unit installation and there are no shared facilities. This criterion does not apply.

3.1.2.2 Protection by Multiple Fission Product Barriers

Criterion 10 - Reactor Design

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

Discussion

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

Preclude significant fuel damage during normal core operation and operational transients (Condition I)[1] or any transient conditions arising from occurrences of moderate frequency (Condition II)[1].

Ensure return of the reactor to a safe state following a Condition III[1] event with only a small fraction of fuel rods damaged although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV)[1].

Chapter 4 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. This information supports the accident analyses of Chapter 15 which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.

Criterion 11 - Reactor Inherent Protection

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

Discussion

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

These reactivity coefficients are discussed in Section 4.3.

Criterion 12 - Suppression of Reactor Power Oscillations

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

Discussion

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

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

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

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

Xenon stability control is discussed in Section 4.3.

Criterion 13 - Instrumentation and Control

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

Discussion

Plant instrumentation and control systems are provided to monitor significant variables in the reactor core, Reactor Coolant System, and containment over their anticipated range for all conditions to the extent required. The installed instrumentation provides continuous monitoring, warning, and initiation of safety functions, by the use of instrumentation and control provided.

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

1. Reactor power level (manual or auto by controlling thermal load).

2. Reactor coolant temperature (manual or auto by rod control cluster assembly (RCCA) motion, in sequential groups).

3. Reactor coolant pressure (manual or auto by heaters and spray in the pressurizer).

4. Reactor coolant water inventory, as indicated by the water level in the pressurizer (manual or auto charging flow).

5. Reactor axial power balance.

6. Reactor Coolant System boron concentration (manual or auto makeup of charging flow).

7. Steam generator water inventory on secondary side (manual or auto feedpump flow through feedwater control valves).

The Reactor Control System is designed to automatically maintain a 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 a transient caused by a load change.

The Reactor Protection System trip setpoints are selected so that anticipated transients do not cause a departure from nucleate boiling ratio (DNBR) of less than the safety limit.

Proper positioning of the control rods is monitored in the control room by bank arrangements of individual meters for each rod cluster control assembly (RCCA). A rod deviation alarm alerts the operator of a deviation of 1 rod cluster control assembly from its bank position. There are also insertion limit monitors with visual and audible annunciation to avoid loss of shutdown margin. Each full length rod cluster control assembly is provided with an indication of positioning at the bottom of its travel. This condition is also alarmed in the control room. Four(4) excore long ion chambers also detect asymmetrical flux distribution indicative of rod misalignment.

Movable incore flux detectors and fixed incore thermocouples are provided as operational aids to the operator. Chapter 7 contains further details on instrumentation and controls. Section 7.5 details the information available to the operator for the performance of required safety functions. Information regarding the Radiation Monitoring System provided to measure environmental activity and alarm high levels is contained in Section 11.4.

Overall reactivity control is achieved by the combination of soluble boron and rod cluster control assemblies. Long term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short term reactivity control for power changes is accomplished by the Reactor Control System which automatically moves rod cluster control assemblies. This system uses input signals including neutron flux, coolant temperature, and turbine load.

These systems are described in Chapters 6, 7, 8, 9, 11 and 12.

Criterion 14 - Reactor Coolant Pressure Boundary

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

Discussion

The Reactor Coolant System boundary is designed, fabricated and erected to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. See Sections 3.9 and 5.2 for details. Reactor Coolant System 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. The system is protected from overpressure by means of pressure relieving devices as required by applicable codes.

Means are provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary with indication in the Control Room. See Section 5.2 for details.

The Reactor Coolant System boundary has provision for inspection, testing and surveillance of critical areas to assess the structural and leaktight integrity. See Section 5.2 for details. For the reactor vessel, a material surveillance program conforming to applicable codes is provided. See Section 5.4 for details.

Criterion 15 - Reactor Coolant System Design

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

Discussion

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

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

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

Multiple safety and relief valves are provided for the Reactor Coolant System. These valves and their setpoints satisfy ASME criteria for overpressure protection. The ASME criteria are satisfactory based upon a long history of industry use. Chapter 5 discusses the reactor coolant system design.

Transient analyses are included in Reactor Coolant System design which conclude that design conditions are not exceeded during normal operation. Protection and control setpoints are based upon these transient analyses. The design margin includes the effects of thermal lag, coolant transport times, pressure drops, system relief valve characteristics and instrumentation and control response characteristics.

Criterion 16 - Containment Design

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

Discussion

A steel lined prestressed concrete reactor building is provided. This structure encloses the entire Reactor Coolant System and is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the simultaneous occurrence of a double ended rupture of the largest pipe in the Reactor Coolant System and the safe shutdown earthquake (SSE). Should such an event occur, the engineered safety features (ESF) serve to cool the reactor core and return the containment to near atmospheric pressure. The containment systems and their associated engineered safety features are designed to ensure the functional capability of preventing the uncontrolled release of radioactive material and that the design conditions important to safety remain inviolate for as long as postulated accident conditions require.

For further discussion, see the following sections:

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		Section
1.	Design of Category I Structures	3.8
2.	Engineered Safety Features	6.0

Criterion 17 - Electrical Power Systems

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

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

Electric power from the transmission network to the Onsite Electric Distribution System shall be supplied by 2 physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One (1) of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

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

Discussion

Onsite and offsite power systems each independently provide the total power requirements to perform the functions required of the safety-related systems. The onsite power required to operate Engineered Safety Features System equipment is supplied by two 100% capacity diesel generators. The offsite power required to operate safety-related systems is supplied by 2 independent sources, one from the 230 kV system and 1 from the 115 kV system. Each source will supply total power requirements for either 1 or both of the redundant and independent power distribution systems for the Engineered Safety Features Systems.

Each Engineered Safety Features power supply bus is normally connected to physically and electrically independent power supplies. Electric power from the transmission network to the substation is provided by a sufficient number of independent lines to minimize the likelihood of simultaneous failure.

Two (2) onsite independent battery systems provide control power for the redundant and independent power distribution systems for Engineered Safety Features Systems as well as control power for the onsite power sources. The reactor protective instrumentation is powered from 4 independent 120 volt nominal ac vital buses which provide uninterrupted power from single phase inverters. Each bus is supplied by its own associated inverter. Each Class 1E battery system supplies power to 2 static inverters.

These systems are designed in accordance with IEEE-308[2].

For further discussion, see the following sections:

		Section	
1.	General Plant Description	1.2	RN 01-113
2.	Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment	3.10	
3.	Environmental Design of Mechanical and Electrical Equipment	3.11	
4.	Offsite Power Systems	8.2	
5.	Onsite Power Systems	8.3	

Criterion 18 - Inspection and Testing of Electric Power Systems

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

Discussion

The ESF power supply buses and associated diesel generators are arranged for periodic, independent testing of each system. These tests, performed periodically in accordance with the Technical Specifications, prove the operability of the Emergency Power Supply System under conditions as close to design as practical, thus permitting assessment of the continuity of the system and condition of the components.

The design of the standby power systems provides testability in accordance with the requirements of Criterion 18.

For further discussion, see the following sections:

		Section	
1.	Onsite Power System	8.3	RN 01-113
2.	Initial Test Program	14.0	
3.	Technical Specifications		RN 99-136

Criterion 19 - Control Room

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

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

Discussion

Safe occupancy of the Control Room under normal, abnormal, and accident conditions is assured by the design. The Control Room is located in a Seismic Category I structure. Adequate shielding is provided to maintain tolerable radiation levels in the Control Room in the event of a design basis accident. Redundant equipment is provided in the Control Room Ventilation System which permits recirculation of Control Room air through HEPA and charcoal filters. This equipment also permits control room air to be drawn from outside through roughing and HEPA filters and to be discharged outside or for the use of various combinations of outside and recirculated air. Radiation and smoke detectors are provided for the Control Room Ventilation System. Excessive concentrations of any one of these contaminants causes an alarm in the Control Room. High radiation places the Control Room Ventilation System in the emergency filtration mode and causes an alarm in the Control Room.

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The Control Room includes the following: substation control panel, electrical relay panels and control panels which contain those instruments and controls necessary for operation of the station functions, such as the reactor and its auxiliary systems, engineered safety features, turbine generator, steam and power conversion systems, station electrical distribution boards, and heating, ventilating and air conditioning systems.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

In the unlikely event that occupancy of the Control Room is restricted, a local control room evacuation panel and manual operation of critical components are used to effect cold shutdown from outside the Control Room.

By use of appropriate procedures and equipment, the unit can also be brought to cold shutdown conditions. For this operation, it is assumed that offsite power will be available.

In accordance with the implementation of the alternative source term for the Virgil C. Summer Nuclear Station, the above dose criteria is replaced by the 5 rem total effective dose equivalent (TEDE) acceptance criterion provided in 10 CFR 50.67(b)(2) for a loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, fuel handling

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accident (FHA), steam generator tube rupture (SGTR), reactor coolant pump locked rotor accident (RCPLRA) and the control rod ejection accident (CREA). RN 12-034

For further discussion, see the following sections: RN 01-113

		Section
1.	General Plant Description	1.2
2.	Control Room Diffusion Estimates	2.3.4.3
3.	Control Building Design	3.8
4.	Habitability Systems	6.4
5.	Instrument and Controls	7.0
6.	Shutdown from Outside Control Room	7.4
7.	Air Conditioning, Heating, Cooling and Ventilation Systems	9.4
8.	Fire Protection System	9.5.1
9.	Radiation Shielding	12.1
10.	Ventilation	12.2
11.	Control Room Dose	15.4

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3.1.2.3 Protection and Reactivity Control Systems

Criterion 20 - Protection System Functions

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

Discussion

A fully automatic protection system with appropriate redundant channels is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of IEEE-279[3] and trial use guide IEEE-379[4]. The Reactor Protection System automatically initiates reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

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

The Engineered Safety Features Actuation System automatically initiates emergency core cooling and other safeguards functions by sensing accident conditions, using redundant analog channels measuring diverse variables. Manual actuation of safeguards may be performed where time is available for operator action but is not relied upon to satisfy this criterion. The Engineered Safety Features Actuation System automatically trips the reactor upon manual or automatic safety injection (S) signal generation. See Section 7.5 and Section 7.1.2.1.5 for additional details. RN 99-101

The response and adequacy of the protection systems are analyzed for all environmental conditions specified by ANS N18.2[1], through Condition IV.

Criterion 21 - Protection System Reliability and Testability

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

Discussion

The Protection System is in accordance with IEEE-279[3]. It provides high functional reliability and adequate independence, redundancy, and testability commensurate with the safety functions of the system. All actuation circuitry is provided with a capability of online testing. This extends to the final actuating device except where operational safety requirements prohibit actual operation of the device; e.g., turbine trip, steam line isolation, etc.

The Reactor Protection System is designed for high functional reliability by providing electrically isolated and physically separated, redundant analog channels and two separate trip logic trains. This assures that no single failure will result in the loss of any protection function. Except for certain defined backup trip functions detailed in Section 7.2, the redundancy and independence provided in the Reactor Protection System allows individual channel test of channels required at power operation to be made during power operation without negating reactor protection or the single failure criterion. This testing will determine failures and losses of redundancy that may have occurred. This arrangement also permits removal of a channel from service while still maintaining the high reliability of the protection function. Details of the Protection system design and testing provisions are contained in Chapter 7.

There are 2 series connected circuit breakers, 1 breaker for each trip logic train, which supply all power to the full length rod drive mechanisms. A reactor trip train supplies a signal to the undervoltage coil of its respective trip breaker and opening of either train breaker will trip the reactor.

The Engineered Safety Features Actuation System is in accordance with IEEE-279[3]. It also utilizes redundant analog channels measuring the same parameter and redundant logic trains, either of which will actuate safety injection and/or reactor building spray.

The Engineered Safety Features Actuation System is testable at power with certain exceptions as detailed in Section 7.3. As with the components of the Reactor Protection System, both physical and electrical separation is practiced for the Engineered Safety Features Actuation System to provide a high degree of availability for its safety function.

Criterion 22 - Protection System Independence

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

Discussion

Protection System components are designed and arranged so that the environment accompanying any emergency situation in which the components are required to function does not result in loss of the safety function. Various means are used to accomplish this. Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. The general conclusion is that diverse protection functions would automatically terminate an accident before intolerable consequences could occur.

Automatic reactor trips occur as listed in Table 7.2-1. RN 99-050

Regarding the ESF actuation system, a safety injection signal can be obtained manually or by automatic initiation from any one of the following diverse sets of signals: RN 99-050

1. Low pressurizer pressure.

2. High Reactor Building pressure (Hi-1). RN 99-050

For a steam break accident, diversity of safety injection is provided by:

1. Low steam line pressure. RN 99-050

2. High steam line differential pressure.

3. For a steam break inside the Reactor Building, high Reactor Building pressure (Hi-1) provides an additional parameter for generation of the safety injection signal. RN 99-050

All of the above sets of signals are redundant and physically separated and meet the intent of the criterion.

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

The design criteria for instrumentation are given in Section 7.2 and qualification of the instrumentation is outlined in Sections 3.10 and 3.11.

Criterion 23 - Protection System Failure Modes

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

Discussion

The Protection System is designed with 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 de-energize-to-trip principle so that a loss of power or disconnection of the channel causes that channel to go into its tripped mode. In addition, a loss of power to the full length Rod Cluster Control Assembly drive mechanisms causes the Rod Cluster Control Assembly to insert by gravity into the core. See Section 7.2 for details.

With regard to engineered safety features, should a loss of the preferred offsite power source occur, onsite diesel generators are available to power emergency loads, with the station batteries being used to supply instrumentation power only for that period of time required for the diesel to start. See Section 8.3 for details. A loss of power to one train of safety injection equipment does not affect the ability of the other train to perform its function.

Criterion 24 - Separation of Protection and Control Systems

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

Discussion

Protection and control channels in the facility protection systems will be designed in accordance with the IEEE-279[3].

The reactor protection system itself is designed to maintain separation between redundant protection channels and protection logic trains. Separation of redundant analog channels originates at the process sensors and continues along the wiring route and through reactor building penetrations to analog protection racks and terminating at the Reactor Protection System logic racks. Isolation of wiring is achieved using separate wireways, cable trays, conduit runs and reactor building penetrations for each redundant channel. Analog equipment is separated by locating components associated with redundant functions in different protection racks. Each redundant protection channel set is energized from a separate ac power feed.

The redundant (2) reactor trip logic trains are physically separated from one another. The Reactor Protection System is comprised of identifiable channels which are physically separated and electrically isolated.

Each trip circuit is designed so that the trip occurs upon de-energization of the circuit; an open circuit or loss of power to a channel will, therefore, result in that channel going into its trip mode. Redundant protection channels are provided to prevent a single failure from defeating a protection function. Redundancy provides reliability and independence of operation. Channel independence is carried throughout the system from the sensor to the logic interface. In some cases, however, it is advantageous to employ control signals derived from individual protection channels through isolation amplifiers contained in the protection channel. As such, a failure in the control circuitry does not adversely affect the protection channel.

The electrical supply and control conductors for redundant or backup circuits have such physical separation as is required to assure that no single credible event will prevent operation of the associated function by reason of electrical conductor damage. Critical circuits and functions include power, control and analog instrumentation associated with the operation of Reactor Protection, Engineered Safety Features, Reactor Shutdown and Residual Heat Removal Systems.

See Sections 7.1 and 8.3 for more details.

Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions RN 01-113

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

Discussion

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

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

Criterion 26 - Reactivity Control System Redundancy and Capability

Two (2) independent reactivity control systems of different design principles shall be provided. One (1) of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One (1) of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Discussion

Two (2) reactivity control systems are provided. These are rod cluster control assemblies (RCCA's) and chemical shim (boric acid). The rod cluster control assemblies are inserted into the core by the force of gravity.

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

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

Details of the construction of the rod cluster control assembly are presented in Chapter 4. Operation is discussed in Chapter 7. The means of controlling the boric acid concentration is described in Chapter 9. Performance analyses under accident conditions are included in Chapter 15.

Criterion 27 - Combined Reactivity Control Systems Capability

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

Discussion

The plant is provided with means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Sections 4.3 and 9.3. Combined use of the Rod Cluster Control System and the Chemical Shim Control System permits the necessary shutdown margin to be maintained during long term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full-out upon trip for this determination.

In the event of a loss of coolant accident, the Safety Injection System is actuated and concentrated boric acid is injected into the cold legs of the Reactor Coolant System. This is in addition to the boric acid content of the accumulators which is passively injected due to a decrease in system pressure. See Section 6.3 for further details.

Criterion 28 - Reactivity Limits

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

Discussion

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

The maximum positive reactivity insertion rates for the withdrawal of rod cluster control assemblies and the dilution of the boric acid in the reactor coolant system are limited by the physical design characteristics of the rod cluster control assemblies and of the Chemical and Volume Control System. Technical Specifications on shutdown margin and on rod cluster control assembly insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in Section 4.3. The capability of the Chemical and Volume Control System to avoid an inadvertent excessive rate of boron dilution is discussed in Section 9.3.

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

Criteria 29 - Protection Against Anticipated Operational Occurrences

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

Discussion

The protection and reactivity control systems are designed to assure extremely high reliability in performing their required safety functions in any anticipated operational occurrence. Likely failure modes of system components are designed to be safe modes. Equipment used in these systems is designed, constructed, operated, and maintained with a high level of reliability. Loss of power to the Protection System results in a reactor trip. Details of system design are covered in Chapter 7. Also refer to responses to General Design Criteria 20 through 25.

3.1.2.4 Fluid Systems

Criterion 30 - Quality of Reactor Coolant Pressure Boundary

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

Discussion

Reactor coolant pressure boundary components are designed, fabricated, inspected and tested in conformance with the ASME Code, Section III. All components are classified in accordance with ANS N18.2[1] and are accorded the quality measures appropriate to the classification. The design bases and evaluations of reactor coolant pressure boundary components are discussed in Chapter 5.

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 leak off between double gaskets. Leakage inside the Reactor Building is drained to the Reactor Building sump where it is monitored.

Leakage is also detected by measuring the airborne activity within the Reactor Building and activity of manual samples of the condensate drained from the Reactor Building and recirculation units. Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank and coolant drain collection tanks make available an accurate indication of integrated leakage. The Reactor Coolant Pressure Boundary Leakage Detection System is discussed in Section 5.2.7.

Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

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

Discussion

Close control is maintained over material selection and fabrication for the Reactor Coolant System to assure that the boundary behaves in a nonbrittle manner. The Reactor Coolant System materials which are exposed to the coolant are corrosion resistant stainless steel or inconel. The reference temperature (RTNDT) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests. Materials testing is consistent with the intent of 10 CFR 50, Appendices G and H. These tests ensure the selection of materials with adequate toughness properties and margins.

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

1. Ultrasonic Testing

In addition to code requirements, the performance of a 100% volumetric ultrasonic test of reactor vessel plate for shear wave and a post hydrostatic test ultrasonic map of welds in the pressure vessel are required. Also, Westinghouse requires cladding bond ultrasonic inspection to more restrictive requirements than Code to preclude interpretation problems during inservice inspection.

2. Radiation Surveillance Program

In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E-185[5], and the requirements of 10 CFR 50, Appendix H.

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

Heatup and cooldown rates during plant life are predicted using conservative values for the change in ductility transition temperature due to irradiation.

Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

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

Discussion

Provision has been made in the Reactor Coolant System design for adequate inspection, testing and surveillance during the service lifetime. Necessary accessibility has been factored into the design. The vessel inspection program will conform to ASTM E-185[5]. These provisions are discussed in detail in Section 5.2.

Criterion 33 - Reactor Coolant Makeup

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

Discussion

The Chemical and Volume Control System provides a means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below a preset level. High pressure centrifugal charging pumps are provided which are capable of supplying the required makeup and reactor coolant pump seal injection flow when power is available from either onsite or offsite electric power systems. These pumps also serve as high head safety injection pumps. Details of system design are included in Chapters 6 and 9. Details of the electric power systems are presented in Chapter 8.

Criterion 34 - Residual Heat Removal

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

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

Discussion

The Residual Heat Removal (RHR) System, in conjunction with the Steam and Power Conversion System, is designed to transfer the fission product decay heat and other residual heat from the reactor core within acceptable limits.

Suitable redundancy is accomplished with the two residual heat removal pumps, located in separate compartments with means available for draining and monitoring of leakage, the 2 heat exchangers and the associated piping, cabling, and electric power source. The residual heat removal system is able to operate on either onsite or offsite electrical power systems.

The Residual Heat Removal System is able to accommodate a single failure (see Section 3.1.3). During the injection phase, no single active failure prevents the accomplishment of Residual Heat Removal System objectives. During the recirculation phase, but not in the injection phase, the Residual Heat Removal System can accommodate one active or passive failure. One active or passive failure in the systems required for long term Residual Heat Removal System operation does not prevent the accomplishment of Residual Heat Removal System objectives.

Details of the system design can be found in Section 5.5.7.

Criterion 35 - Emergency Core Cooling System

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

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

Discussion

An Emergency Core Cooling System (ECCS) is provided to cope with any loss of coolant accident due to a pipe rupture. Abundant cooling water is available to the core at a rate sufficient to maintain the core geometry and to assure that clad metal-water reaction is limited to less than 1%. The design is adequate to ensure performance of the required safety functions assuming a single failure and that electrical power is available from either the Offsite or Onsite Electrical Power System. Details of the capability of the system are discussed in Section 6.3. An evaluation of the adequacy of the system functions is presented in Chapter 15.

Criterion 36 - Inspection of Emergency Core Cooling System

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

Discussion

Equipment design facilitates access to the critical parts of the reactor vessel internals, injection nozzles, pipes, and valves for visual inspection and for nondestructive inspection where such techniques are desirable and appropriate. The design enables compliance with ASME Code, Section XI requirements.

Components located outside the Reactor Building are accessible for leaktightness inspection during normal operation of the plant.

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 Section 6.3.

Criterion 37 - Testing of Emergency Core Cooling System

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

Discussion

Components of the system are accessible for leaktightness inspection during periodic tests.

Active components of the Emergency Core Cooling System may be individually actuated on the normal power source at any time during plant operation to demonstrate operability.

Tests may be performed during shutdown to demonstrate proper automatic operation of the Emergency Core Cooling System.

Active components are identified in Section 3.9.2. Inservice testing is discussed in Section 3.9.4. The details of these tests are included in Section 6.3. Emergency power details are included in Chapter 8.

Criterion 38 - Containment Heat Removal

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

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

Discussion

Two systems based on different principles are provided to remove heat from the Reactor Building following an accident to maintain the pressure below the Reactor Building design pressure. The Reactor Building spray and the Reactor Building cooling units are each independently capable of removing sufficient energy to maintain the pressure below the Reactor Building design pressure. Each of these systems consists of redundant components supplied from separate power buses. No single failure can cause a loss of more than half of the installed 200% cooling capacity. These systems are described in Section 6.2.

Criterion 39 - Inspection of Containment Heat Removal System

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

Discussion

The Reactor Building Heat Removal Systems consist of the Reactor Building Spray System and the Reactor Building Cooling Units. The Reactor Building Cooling Units, the Reactor Building sump and Reactor Building spray pumps, are located so that the visual inspection of these items is possible during normal plant operation. The spray rings and nozzles of the Reactor Building Spray System are located under the dome of the Reactor Building. An air connection is provided on the supply piping to the spray rings for testing the spray nozzles. Functional operability of each nozzle is tested by blowing air or smoke into the spray rings and observing tell-tale devices such as streamers or balloons.

For further discussion, see Section 6.2.2.

Criterion 40 - Testing of Containment Heat Removal System

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

Discussion

The Reactor Building Heat Removal Systems have the capability of being periodically tested as follows:

1. Reactor Building Cooling Units

    a. The Reactor Building cooling units are used during normal operation and can be individually tested for emergency operation.

    b. The cooling coil service water valves can be operated through their full travel.

    c. The service water pumps can be tested for automatic operation.

    d. The service water booster pumps can be tested for automatic operation.

2. Reactor Building Spray System

   a. The operation of the spray pumps can be tested by recirculation to the refueling water storage tank through a test line.

   b. The Reactor Building Spray System valves can be operated through their full travel.

Active components are identified in Section 3.9.2. Inservice testing is discussed in Section 3.9.4. The Reactor Building cooling units and Reactor Building Spray System are discussed in Sections 6.2 and 9.2.

Criterion 41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for Onsite Electric Power System operation (assuming offsite power is not available) and for Offsite Electric Power System operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Discussion

The following systems are designed to clean up the reactor building atmosphere after a postulated loss of coolant accident (LOCA):

1. The Reactor Building Spray System sprays a sodium hydroxide (NaOH) solution into the Reactor Building to remove elemental iodine. The power supply consists of 2 independent subsystems each supplied from separate buses. Either subsystem alone can provide the iodine removal capacity for which credit is taken in Chapter 15. No single active failure will cause both subsystems to fail to operate. System design is discussed in Section 6.2.

2. The post accident hydrogen removal system is also designed with redundancy of vital components so that a single active failure will not prevent timely operation of the system. This system is described in Section 6.2.5.

3. The recirculation system HEPA filters are capable of filtering the full post loss of coolant accident recirculation air flow for which credit is taken in Chapter 15. This system is discussed in Sections 6.2.2 and 6.5.1.

Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

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

Discussion

The Reactor Building atmosphere cleanup systems, with the exception of the spray headers and nozzles, are designed and located such that they can be inspected periodically as required. The spray headers and nozzles can be air tested as described in the discussion of Criterion 39.

Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

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

Discussion

The Reactor Building Atmosphere Cleanup System can be tested as follows:

1. Reactor Building Spray System

    a. The operation of the spray pumps can be tested by recirculation of borated refueling water to the refueling water storage tank through a test line.

    b. The system valves can be operated through their full travel.

    c. The system is checked for leaktightness during testing.

2. Post Accident Hydrogen Removal System

   a. The post accident hydrogen recombiners are periodically tested to verify operation of the control system and functional performance of the heaters at the required temperature level. These tests are performed during normal plant operation from the recombiner control panels.

   b. The alternate purge line can be operated to test the full operational sequence.

3. Reactor Building Cooling Unit HEPA Filters

   a. The operation of the HEPA filter bypass dampers can be functionally tested.

   b. The HEPA filters are tested for through leakage during system testing.

Identification of active components is presented in Section 3.9.2. A further discussion of inservice testing is provided in Section 3.9.4.

Criterion 44 - Cooling Water

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

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

Discussion

The systems provided to transfer heat from items important to safety to the service water pond consist of systems identified as: Service Water, Component Cooling Water and Chilled Water.

Redundancy is provided by the creation of the Seismic Category I service water pond which functions as the ultimate heat sink in the unlikely event of main dam failure, by the installation of three 100% capacity pumps and by the provision of features of redundancy and isolation capability.

The cooling water systems provided to transfer heat produced under operating and accident conditions from the plant to the environment are as follows: service water pond, Service Water System, Chilled Water System, and Component Cooling Water System. The service water pond is the ultimate heat sink. The Service Water System transfers heat from the component cooling heat exchangers to the service water pond. The Component Cooling Water System and the Chilled Water System are intermediate, closed cycle cooling systems. The Component Cooling Water System is used to isolate service water (water from the service water pond) from normally radioactive streams that initially carry heat to be rejected to the environment.

The service water pond is an impounded portion of Monticello Reservoir and is enclosed by dams designed to withstand the effects of the SSE and by natural features. For additional details, see Sections 2.4 and 9.2. RN 99-155

The Service Water System, Component Cooling Water System and Chilled Water System have 2 independent subsystems supplied with electric power from separate buses. These subsystems are operable from either Offsite or Onsite (emergency diesel generators) Electric Power Systems. Therefore, for each system, the subsystems are totally redundant and the availability of the minimum engineered safety features requirements is ensured assuming a single failure.

Criterion 45 - Inspection of Cooling Water System

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

Discussion

Important Cooling Water System components are accessible for required periodic inspection. These components have suitable manholes, handholes or inspection ports to allow for periodic inspection.

Criterion 46 - Testing of Cooling Water System

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

Discussion

Redundancy and isolation are provided to allow periodic pressure and functional testing of the systems as a whole, including the functional sequence that initiates system operation, and also including transfer between the normal and diesel power sources. At least one of the redundant systems is in service during normal operation.

Identification of active components is presented in Section 3.9.3. A further discussion of inservice testing is provided in Section 3.9.4.

3.1.2.5 Reactor Containment

Criterion 50 - Containment Design Basis

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

Discussion

The design of the Reactor Building is based upon the containment design basis accident which assumes the double ended rupture of a main steam line inside the Reactor Building and the worst single active failure. The maximum pressure and temperature determined for the design basis accident are 53 psig and 372.7°F. The super heated temperatures within the Reactor Building are limited in magnitude and duration via closure of the main steam isolation valves and spray actuation. Following spray actuation, the Reactor Building remains saturated in the long term and below the Reactor Building design temperature of 283°F. The pressure differential between 53 psig and the reactor building design pressure of 57 psig, and the long term temperature of less than 283°F, provide ample margin to allow for increase energy sources as the result of degraded performance of emergency core cooling systems.

For further discussion, see the following sections:

RN 01-113

		Section
1.	Classification of Structures, Components and Systems	3.2
2.	Wind Design Criteria	3.3
3.	Missile Protection Criteria	3.5
4.	Criteria for Protection Against Dynamic Effects Associated with a Loss of Coolant Accident	3.6
5.	Seismic Design	3.7
6.	Design of Reactor Building	3.8
7.	Containment Functional Design	6.2.1
8.	Reactor Building Heat Removal System	6.2.2
9.	Accident Analyses	15.0

Criterion 51 - Fracture Prevention of Containment Pressure Boundary

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

Discussion

The Reactor Building liner material has a maximum nil ductility transition temperature of at least 30°F below the minimum service temperature.

Ferritic materials exposed to the external environment have been selected so that their temperatures under normal operating and testing conditions are 30°F or more above nil ductility transition temperature (see Section 3.8).

Criterion 52 - Capability for Containment Leakage Rate Testing

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

Discussion

The Containment System is designed and constructed and the necessary equipment is provided to permit periodic integrated leak rate tests during the plant lifetime. The testing program satisfies the requirements of Appendix J to 10 CFR 50.

The provisions for testing and the test program satisfy the requirements of Criterion 52.

For further discussion, see the following sections:

		Section	
1.	Concrete Reactor Building (Equipment Hatch and Personnel Airlocks)	3.8.1	RN 01-113
2.	Containment Functional Design	6.2.1	98-01
3.	Containment Leakage Testing	6.2.6	
4.	Technical Specifications	16.0	

Criterion 53 - Provisions for Containment Testing and Inspection

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

Discussion

The program defining, and the means for performing, individual leakage rate tests on applicable penetrations in accordance with Appendix J to 10 CFR 50 are presented in Section 6.2.6. The program defining the lifetime inservice tendon surveillance is presented in Chapter 16. This program provides physical evidence that the structural integrity of the Reactor Building has been maintained.

The provisions made for penetration testing satisfy the requirements of Criterion 53.

For further information, see the following sections:

		Section	RN 01-113
1.	Containment Functional Design	6.2.1	98-01
2.	Containment Leakage Testing	6.2.6	
3.	Technical Specifications		RN 99-136

Criterion 54 - Piping Systems Penetrating Containment

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

Discussion

The required isolation and testing capabilities are provided in all piping systems penetrating containment. Test connections are provided as required to enable periodic leak rate determination for individual valves and other isolation barriers. Means are provided for demonstrating the operability of remotely operated isolation valves or other isolation barriers. The Engineered Safety Features Actuation System test circuitry provides the means for testing isolation valve operability. This is discussed in Section 6.2.

For further discussion, see the following sections: RN 01-113

		Section
1.	Containment Functional Design	6.2.1
2.	Containment Isolation Systems	6.2.4
3.	Containment Leakage Testing	6.2.6

Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

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

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

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

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

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

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

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

Discussion

The boundary for the Reactor Coolant System is defined in accordance with Section 4 of ANS N18.2[1]. The entire Reactor Coolant System as defined above, is located within the Reactor Building. Thus, this criterion does not apply to Westinghouse pressurized water reactors. However, the specific valving arrangements specified in Criterion 55 have been adhered to in the design of this plant with the specific exceptions listed and justified in Chapter 6.

Criterion 56 - Primary Containment Isolation

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

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

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

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

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

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

Discussion

Each line that connects directly to the Reactor Building atmosphere and penetrates containment is provided with containment isolation valves, except where it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable. Details are provided in Sections 5.5 and 6.2 and on the flow diagrams included in Chapters 6 and 9.

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 shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Discussion

Each line that penetrates containment and is not connected directly to the Reactor Building atmosphere and is not part of the reactor coolant pressure boundary has at least one isolation valve located outside containment near the penetration or is a closed system outside containment. Details are provided in Sections 5.5 and 6.2 and in the flow diagrams included in Chapters 6 and 9.

3.1.2.6 Fuel and Radioactivity Control

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

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

Discussion

Waste handling systems have been incorporated in the plant design for retention and/or processing of radioactive wastes resulting from normal operation. Controls and monitoring are provided to ensure that Appendix I to 10 CFR 50 is satisfied. The plant is also designed such that radioactive releases during accidents will not exceed the limits of 10 CFR 100.11 or 10 CFR 50.67. RN 12-034

Chapter 11 describes the Radioactive Waste Processing System, design criteria, and amounts of estimated releases of radioactive effluents to the environment. Chapter 5 and Sections 6.2, 12.1 and 12.2 describe the containment system which forms a barrier to the escape of fission products should a loss of coolant occur. Chapter 6 describes the engineered safety features for control of reactivity and Reactor Building pressure.

Criterion 61 - Fuel Storage and Handling and Radioactivity Control

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

Discussion

The Spent Fuel Cooling System, Fuel Handling System, Radioactive Waste Processing Systems and other systems that contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions.

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

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

3. Individual components which contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems or are vented to the Gaseous Waste Processing System. Details of the ventilation systems are presented in Section 9.4.

4. The Spent Fuel Cooling System provides cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed with redundancy and testability to assure continued heat removal. The Spent Fuel Cooling System is described in Section 9.1.3.

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

Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

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

Discussion

Criticality in new and spent fuel storage areas is prevented by physical separation of fuel assemblies and the presence of borated water in the spent fuel pool. The fuel storage racks are constructed so that fuel assemblies may be inserted in prescribed locations only. These have a minimum center to center spacing in both directions to ensure subcriticality even if assemblies are immersed in unborated water. Criticality prevention and criticality considerations are discussed in Sections 9.1 and 4.3, respectively.

Criterion 63 - Monitoring Fuel and Waste Storage

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

Discussion

Monitoring systems are provided to cause alarms when excessive temperature or low water level occurs in the spent fuel pool. Appropriate safety actions are initiated by operator action as outlined in Section 9.1.3.

Radiation monitors and alarms are provided as required to warn personnel of impending excessive levels of radiation or airborne activity. The radiation monitoring system is described in Sections 11.4, 12.1.4, and 12.2.4.

Criterion 64 - Monitoring Radioactivity Releases

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

Discussion

The Reactor Building atmosphere is continually monitored during normal and transient plant operations, using the Reactor Building particulate, gaseous, and iodine radiation monitors. Under accident conditions, when temperature and pressure permit, samples of the Reactor Building atmosphere can be obtained to provide data on existing airborne radioactivity concentrations within the Reactor Building. In addition, a Reactor Building high range area gamma monitor is used to monitor the potential gamma radiation dose which may result from the postulated accident. Radioactivity levels contained in the plant effluent discharge paths and in the environs are monitored during normal and accident conditions by the plant radiation monitoring system, described in Sections 11.4 and 12.2.4, and the health physics program described in Section 12.3.

3.1.3 SINGLE FAILURE CRITERION

The Engineered Safety Features systems are designed to tolerate a single failure (see General Design Criteria 34 and 35) during the period of recovery following an incident without loss of their protective function.

During the short term immediately following the incident, this single failure is limited to a failure of an active component to complete its function as required. Should the failure occur during the long term rather than the short term period following the incident, the failure definition is expanded such that the systems designs will tolerate either an active failure or a passive failure without loss of their protective function.

3.1.3.1 Definitions

1. Engineered Safety Features

   Engineered safety features are provided for sensing the incident occurrence, initiating protective action and completing the necessary protective function to remain within NRC specified criteria for radioactivity release from the plant site.

   The engineered safety features are as follows:

   a. The containment

   b. The Reactor Building Heat Removal Systems (Reactor Building Spray System and Reactor Building Cooling Units)

   c. The Reactor Building Air Purification and Cleanup Systems

   d. The Containment Isolation System

   e. The Combustible Gas Control System

   f. The Emergency Core Cooling System

   g. Habitability systems

   h. Fission Product Removal and Control Systems

Engineered Safety Features Support Systems include Component Cooling, Service Water, Chilled Water, related heating, ventilating and cooling equipment, and electric power supply associated with the required fluid or steam system. Compressed air is not required. For systems failure analysis, the effect of a single failure is considered on the total of the engineered safety features and the related service system required for reactor protection following the incident. As an example, if the Emergency Core Cooling System is required and a diesel generator failure is postulated, that would be the active failure for which the consequences are analyzed. The consequences would include other failures specifically caused by the diesel failure. No further active or passive failures of the systems are considered either for the short or long term.

2. Period of Recovery

   The period of recovery is the time necessary to bring the plant to cold shutdown and regain access to faulted equipment. The recovery period is the sum of the short and long term periods defined below.

3. Incident

   An incident is any natural or accidental event of infrequent occurrence and the related consequences which affect plant operation and require the use of Engineered Safety Features systems. Such events, which are analyzed independently, are not assumed to occur simultaneously and include loss of coolant accident, steam line ruptures, steam generator tube ruptures, etc. Loss of offsite power may be an isolated occurrence or may be concurrent with any event requiring Engineered Safety Features systems use.

4. Short Term

   The short term is the time immediately following the incident during which automatic actions are performed, system responses are checked, type of incident is identified and preparations for long term recovery operation are made. In the event of a loss of coolant accident, the period of the injection mode of operation during Emergency Core Cooling System operation is the basis for the short term period.

5. Long Term

   The long term is the remainder of the recovery period following the short term. In comparison with the short term, where the main concern is to remain within NRC specified site criteria, the long term period of operation involves bringing the plant to cold shutdown conditions where access to the Reactor Building can be gained and repair effected.

6. Active Failure

   The failure of a powered component, such as a piece of mechanical equipment, component of the Electrical Supply System or instrumentation and control equipment, to act on command to perform its design function constitutes an active failure. Examples include failure of a valve to move to its correct position, failure of an electrical circuit breaker or relay to respond, failure of a pump, fan or diesel generator to start, etc.

   Consideration of equipment moving spuriously from the proper safeguards position, such as a motor operated valve inadvertently shutting, is specifically excluded.

7. Passive Failure

   The structural failure of a static component which limits the effectiveness of that component in carrying out its design function constitutes a passive failure. When applied to a fluid system, this means a break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm. Such leak rates are consistent with limited cracks in pipes, sprung flanges, valve packing leaks or pump seal failures.

3.1.4 REFERENCES

1. American Nuclear Society, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANS N18.2, as discussed in Appendix 3A, Regulatory Guide 1.26.

2. Institute of Electrical and Electronics Engineers, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Systems," IEEE-308-1971.

3. Institute of Electrical and Electronics Engineers, "IEEE Criteria for Nuclear Power Plant Generating Station Protection Systems," IEEE-279-1971.

4. Institute of Electrical and Electronics Engineers, "Trial Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems," IEEE-379-1972.

5. American Society for Testing and Materials, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," ASTM E-185, 1970.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

Certain structures, components and systems of a nuclear plant are considered important to safety because they perform safety functions required to avoid or mitigate the consequences of abnormal operational transients or accidents. This section discusses the classification of structures, components and systems according to the importance of the safety function they perform. In addition, design requirements are placed upon such equipment to ensure the proper performance of safety actions, when required.

3.2.1 SEISMIC CLASSIFICATION

Structures, components and systems are designated as Safety Class 1, 2a, 2b, 3 or Non-Nuclear Safety (NNS); Class 1E and/or Seismic Category I. These items are listed in Tables 3.2-1, 3.2-2 and 3.10-1. The Seismic Category I structures are designed to withstand the effects of the safe shutdown earthquake (SSE). The design of safety class structures, components and systems to resist the earthquake and other loads is based upon levels of material stress or load factors, as applicable. Such design yields margins of safety appropriate for the earthquake. The margin of safety provided for safety class structures, components and systems with respect to the SSE ensures that safety functions are not jeopardized.

Additionally, some components, the function of which does not require designation as safety class, but the failure of which under seismic conditions might be unacceptable, have been considered with regard to Seismic Category I requirements.

The designation of structures, components and systems as Seismic Category I is in conformance with the recommendations of Regulatory Guide 1.29 (see Appendix 3A) for balance of plant. Nuclear Steam Supply System fluid system components important to safety are classified in accordance with the August 1970 Draft of ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", except that components of the accumulator subsystem are classified in accordance with the 1973 version of N18.2, as finally accepted by ANSI, and components of the Liquid and Gaseous Waste Processing Systems and the Boron Recycle System are classified in accordance with Regulatory Guide 1.143.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS

3.2.2.1 Mechanical Components

Mechanical components are classified as Safety Class 1, 2a, 2b, or 3, in accordance with their importance to nuclear safety. This importance, as established by class assignment, is considered in design, materials, manufacture or fabrication, assembly, erection, construction and operation.

These safety classes are in conformance with ANS N18.2. This classification system differs from the system of Quality Group Classification defined in Regulatory Guide 1.26 (see Appendix 3A), but meets the intent of Regulatory Guides 1.26 and 1.143.

Safety classes are defined as follows:

1. Safety Class 1

   Safety Class 1 (SC-1) applies to Reactor Coolant System components, the failure of which could cause a Condition III or Condition IV loss of reactor coolant as defined by ANS N18.2.

2. Safety Class 2

   Safety Class 2 (SC-2) applies to those components of safety systems required to fulfill a system function. Safety Class 2 is subdivided into Safety Class 2a and Safety Class 2b. A safety system (in this context) is any system that functions to shutdown the reactor, cool the core, cool another safety system or the containment and contains, controls or reduces radioactivity released in an accident.

   a. Safety Class 2a

      Safety Class 2a (SC-2a) applies to containment and to components of those safety systems, or portions thereof, through which reactor coolant water flows directly from the Reactor Coolant System or the Reactor Building recirculation sumps.

   b. Safety Class 2b

      Safety Class 2b (SC-2b) applies to all other components of Safety Class 2.

3. Safety Class 3

   Safety Class 3 (SC-3) applies to components not classified as Safety Class 1 or Safety Class 2 and:

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

   b. That provide or support any safety system function,

      * The Liquid and Gaseous Waste Processing Systems and the Boron Recycle System are designated non-nuclear safety consistent with Regulatory Guide 1.143.

c. That control airborne radioactivity released outside the reactor containment, or

d. That remove decay heat from spent fuel.

4. Non-Nuclear Safety

Those structures, components and systems which do not fall into the above safety class categories are designated as Non-Nuclear Safety (NNS).

3.2.2.2 Electrical Components

Electrical components are classified as either Class 1E, as defined in IEEE-380 [1], or as NNS. RN 01-113

3.2.2.3 Structures

Structures are classified as Seismic Category I or non seismic category.

3.2.3 QUALITY ASSURANCE CLASSIFICATION

All structures, components and systems classified as Seismic Category I, including all safety class items, are Quality Assurance Class 1.

3.2.4 CORRELATION OF SAFETY CLASSES WITH INDUSTRY CODES

Table 3.2-3 provides a correlation between safety class and industry codes and standards applicable to mechanical components.

3.2.5 EQUIPMENT AND STRUCTURE LISTS

Table 3.2-1 provides a listing of major mechanical components and indicates the safety class, code class, seismic category and quality assurance class of each. Table 3.2-2 provides a listing of structures and indicates the classification of each. Table 3.10-1 provides a listing of Seismic Category I electrical components.

3.2.6 REFERENCE

[1] Institute of Electrical and Electronics Engineers, "Definitions of Terms Used in IEEE Standards on Nuclear Power Generating Stations," IEEE-380-1975.

TABLE 3.2-1

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
MECHANICAL COMPONENTS (by system)								RN 98-103
REACTOR COOLANT SYSTEM								
Reactor Vessel	NSSS	1	ASME III	1	Y	1	1	
Full Length CRDM Housing (48)	NSSS	1	ASME III	1	Y	1	2	98-01 RN 16-003
Reactor Coolant Pump Assemblies (3)	NSSS	1	ASME III	1	Y	1	-	
Reactor Coolant Pump Casings (3)	NSSS	1	ASME III	1	Y	1	2	
Reactor Coolant Pump Internals (3)	NSSS	1	ASME III	1	Y	1	2	
Reactor Coolant Pump Motors (3)	NSSS	2b	NEMA 20	-	Y	1	2	
Steam Generator, tube side (3)	NSSS	1	ASME III	1	Y	1	2	
Steam Generator, shell side (3)	NSSS	2a	ASME III	1	Y	1	2,3	
Pressurizer	NSSS	1	ASME III	1	Y	1	2	
Reactor Coolant Thermowell	NSSS	1	ASME III	1	Y	1	4	
Reactor Coolant Piping and Fittings	NSSS	1	ASME III	1	Y	1	1,5	
Surge Pipe and Fittings	NSSS	1	ASME III	1	Y	1	1	RN 98-103
Relief Valves (3), Safety Valves (3), Block Valves (3)	NSSS	1	ASME III	1	Y	1	1	
Valves to Reactor Coolant System Boundary	NSSS/BOP	1	ASME III	1	Y	1	6	
Piping to Reactor Coolant System Boundary	NSSS/BOP	1	ASME III ANSI B31.1	1	Y	1	5,6,29	
Pressurizer Relief Tank (PRT)	NSSS	NNS	ASME VIII	-	N	-	-	RN 16-003
Fuel Assemblies	NSSS	NA	-	-	Y	1	-	
CETNA Assembly	NSSS	1	ASME III	1	Y	1	42	RN 16-003

98-01
RN
16-003

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
CHEMICAL AND VOLUME CONTROL SYSTEM								
Regenerative Heat Exchanger	NSSS	2a	ASME III	2	Y	1	1	
Letdown Heat Exchanger, tube side	NSSS	2a	ASME III	2	Y	1	1	
Letdown Heat Exchanger, shell side	NSSS	2b	ASME III	3	Y	1	1,7	
Mixed Bed Demineralizers (2)	NSSS	3	ASME III	3	Y	1	4	
Cation Bed Demineralizer	NSSS	3	ASME III	3	Y	1	4	
Reactor Coolant Filter	NSSS	2a	ASME III	2	Y	1	1	
Volume Control Tank	NSSS	2a	ASME III	2	Y	1	1	
Charging Pumps, centrifugal (3)	NSSS	2a	ASME III	2	Y	1	1,7,8a,8c,8d	
Seal Water Injection Filters (2)	NSSS	2a	ASME III	2	Y	1	1	
Letdown Orifices (3)	NSSS	2a	ASME III	2	Y	1	4	RN 98-103
Excess Letdown Heat Exchanger, tube side	NSSS	2a	ASME III	2	Y	1	1	
Excess Letdown Heat Exchanger, shell side	NSSS	2b	ASME III	3	Y	1	1	
Seal Water Return Filter	NSSS	2a	ASME III	2	Y	1	1	
Seal Water Heat Exchanger tube side	NSSS	2a	ASME III	2	Y	1	1	
Seal Water Heat Exchanger, shell side	NSSS	2b	ASME III	3	Y	1	1,7	
Boric Acid Tanks (2)	BOP	2b	ASME III	3	Y	1	9	
Boric Acid Filter	NSSS	2b	ASME III	3	Y	1	4	
Boric Acid Transfer Pumps (2)	NSSS	2b	ASME III	3	Y	1	1,8b	
Boric Acid Transfer Pump Bypass Orifices (2)	BOP	2b	ASME III	3	Y	1	-	
Boric Acid Blender	NSSS	2b	ASME III	3	Y	1	1	
Resin Fill Tank	BOP	NNS	ASME VIII	-	N	-	9	
Boric Acid Batching Tank	NSSS	NNS	ASME VIII	-	N	-	9	
Boric Acid Batching Tank Agitator	NSSS	NNS	-	-	N	-	-	
Alternate Seal Injection Pump (1)	NSSS	2a	ASME III	2	Y	1	39	RN 11-027

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Chemical Mixing Tank	NSSS	NNS	ASME VIII	-	N	-	6,9	
Chemical Mixing Tank Orifice	BOP	NNS	No Code	-	N	-	-	
Reactor Coolant Pump Seal Bypass Orifices (3)	NSSS	2a	ASME III	2	Y	1	4	RN 14-043
Chemical and Volume Control System Piping	BOP	2a	ASME III	1,2	Y	1	5	
Chemical and Volume Control System Valves	NSSS/BOP	2a/2b/3	ASME III	2,3	Y	1	1,2,4,5,10	
BORON THERMAL REGENERATION SYSTEM								RN 98-103
Moderating Heat Exchanger, tube side	NSSS	3	ASME III	3	Y	1	4	
Moderating Heat Exchanger, shell side	NSSS	3	ASME III	3	Y	1	4	
Letdown Chiller Heat Exchanger, tube side	NSSS	3	ASME III	3	Y	1	4	
Letdown Chiller Heat Exchanger, shell side	NSSS	NNS	ASME VIII	-	N	-	-	
Letdown Reheat Heat Exchanger, tube side	NSSS	2a	ASME III	2	Y	1	1	
Letdown Reheat Heat Exchanger, shell side	NSSS	3	ASME III	3	Y	1	1	
Thermal Regeneration Demineralizers (4)	NSSS	3	ASME III	3	Y	1	4	
Boron Thermal Regeneration Chiller	NSSS	NNS	ASME VIII	-	N	-	7	
Boron Thermal Regeneration Chiller Surge Tank	NSSS	NNS	ASME VIII	-	N	-	6	RN 99-053
Boron Thermal Regeneration Chiller Pumps (2)	NSSS	NNS	No Code	-	N	-	-	
Boron Thermal Regeneration System Piping	BOP	3	ASME III	3	Y	1	5	
Boron Thermal Regeneration System Valves	NSSS/BOP	3	ASME III	3	Y	1	4,5	
BORON RECYCLE SYSTEM								RN 98-103
Recycle Holdup Tanks (2)	BOP	NNS	ASME III	3	Y	1	31,35	
Recycle Evaporator Feed Pumps (2)	NSSS	NNS	ASME III	3	Y	1	1,31	
Recycle Evaporator Feed Demineralizers (2)	NSSS	NNS	ASME III	3	Y	1	4,31	
Reactor Grade Water System Demineralizers (3)	BOP	NNS	ASME VIII	-	N	-	-	RN 07-037

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
								RN 98-103
Recycle Evaporator Feed Filter	NSSS	NNS	ASME III	3	Y	1	4,31	
Recycle Holdup Tank Vent Ejector	BOP	NNS	ASME III	3	Y	1	31,35	02-01
Recycle Evaporator **–No Longer In Service–**	NSSS	NNS	ASME III	3	Y	1	4,7,31	RN
Recycle Evaporator Concentrates Sample Cooler**–No Longer In Service–**	BOP	2b	ASME III	3	N	-	7	07-037
Recycle Evaporator Pumps and Valves	NSSS	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	4,7,31	
Recycle Evaporator Piping**–No Longer In Service–**	NSSS	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	4,7,31	RN 07-037
Recycle Evaporator Condensate Demineralizer	NSSS	NNS	ASME VIII	-	N	-	-	
Recycle Evaporator Condensate Filter	NSSS	NNS	ASME VIII	-	N	-	-	
Recycle Evaporator Concentrate Filter	NSSS	NNS	ASME VIII	-	N	-	-	
Recycle Evaporator Reagent Tank	NSSS	NNS	ASME VIII	-	N	-	6	
Sample Vessel (2)	BOP	NNS	ASME VIII	-	N	-		
Boron Recycle System Piping	BOP	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	5,31,35	
Boron Recycle System Valves	NSSS/BOP	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	4,5,31	RN 98-103
SAFETY INJECTION SYSTEM								
Accumulators (3)	NSSS	2a	ASME III	2	Y	1	1	
Protective Chambers on Recirculation Sump Isolation Valves	BOP	2a	ASME III	MC	Y	1	-	
Hydrostatic Test Pump	NSSS	NNS	-	-	N	-	-	
Safety Injection System Piping	BOP	2a	ASME III	2	Y	1	5	
Safety Injection System Valves	NSSS/BOP	2a	ASME III	2	Y	1	4,5	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
RESIDUAL HEAT REMOVAL SYSTEM								
Residual Heat Removal Low Head Safety Injection Pumps (2)	NSSS	2a	ASME III	2	Y	1	1,7,8a,8c	
Residual Heat Exchangers, tube side (2)	NSSS	2a	ASME III	2	Y	1	1,8c	
Residual Heat Exchangers, shell side (2)	NSSS	2b	ASME III	3	Y	1	1	
Residual Heat Removal System Piping	BOP	2a	ASME III	2	Y	1	5	
Residual Heat Removal System Valves	NSSS/BOP	2a	ASME III	2	Y	1	1,5	RN 98-103
LIQUID WASTE PROCESSING SYSTEM								
Reactor Coolant Drain Tank	NSSS	NNS	ASME VIII	-	N	-	31	
Reactor Coolant Drain Tank Pumps (2)	NSSS	NNS	ASME III	3	N	-	31	
Reactor Coolant Drain Tank Heat Exchanger, tube side	NSSS	NNS	ASME VIII	-	N	-	31	
Reactor Coolant Drain Tank Heat Exchanger, shell side	NSSS	2b	ASME III	3	Y	1	1,31	
Waste Holdup Tank	NSSS	NNS	ASME III	3	Y	1	4,31	
Waste Evaporator Feed Pump	NSSS	NNS	ASME III	3	Y	1	4,31	
Waste Evaporator Feed Filter – **No Longer In Service**								RN 03-038
Waste Evaporator – **No Longer In Service**								
Waste Evaporator Concentrates Sample Cooler	BOP	2b	ASME III	3	N	-	7	
Waste Evaporator Pumps and Valves	NSSS	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	4,31	RN 98-103
Waste Evaporator Piping	NSSS	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	4,31,35	
Waste Evaporator Condensate Demineralizer	NSSS	NNS	ASME VIII	-	N	-	31	
Waste Evaporator Condensate Filter	NSSS	NNS	ASME VIII	-	N	-	31	
Duratek Demineralizers	BOP	NNS	ASME VIII	-	N	-	-	RN 03-038

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Waste Evaporator Condensate Tank	NSSS	NNS	ASME VIII	-	N	-	31	
Waste Evaporator Condensate Tank Pump	NSSS	NNS	ASME III	3	N	-	31	
Chemical Drain Tank	NSSS	NNS	ASME VIII	-	N	-	6,31	
Chemical Drain Tank Pump	NSSS	NNS	ASME III	3	N	-	31	
Spent Resin Storage Tank	NSSS	NNS	ASME III	3	Y	1	4,31	
Spent Resin Sluice Pump	NSSS	NNS	ASME III	3	Y	1	4,31	
Spent Resin Sluice Filter	NSSS	NNS	ASME III	3	Y	1	4,31	
Laundry and Hot Shower Tank	NSSS	NNS	ASME VIII	-	N	-	9,31	
Laundry and Hot Shower Tank Pump	NSSS	NNS	ASME III	3	N	-	11,31	
Laundry and Hot Shower Tank Strainer	NSSS	NNS	ASME VIII	-	N	-	31	RN
Laundry and Hot Shower Tank Filter	NSSS	NNS	ASME VIII	-	N	-	6,31	98-103
Floor Drain Tank	NSSS	NNS	ASME VIII	-	N	-	9,31	
Floor Drain Tank Filter	NSSS	NNS	ASME VIII	-	N	-	31	
Floor Drain Tank Strainer	NSSS	NNS	No Code	-	N	-	6,31	
Floor Drain Tank Pump	NSSS	NNS	No Code	-	N	-	31	
Waste Monitor Tanks (2)	NSSS	NNS	ASME VIII	-	N	-	9,11,31	
Waste Monitor Tank Pumps (2)	NSSS	NNS	ASME III	3	N	-	11,31	
Waste Monitor Tank Demineralizer	NSSS	NNS	ASME VIII	-	N	-	31	
Waste Monitor Tank Filter	NSSS	NNS	ASME VIII	-	N	-	31	
Drumming Header Strainer	NSSS	NNS	ASME VIII	-	N	-	6,31	
Waste Evaporator Concentrates Holdup Tank	BOP	NNS	ASME VIII	-	N	-	31	
Waste Evaporator Concentrates Transfer Pumps	BOP	NNS	ASME III	3	N	-	31	
Waste Evaporator Reagent Tank – **No Longer In Service**								RN 03-038
Excess Waste Holdup Tank	BOP	NNS	ASME VIII	-	N	-	31	RN 98-103

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes
Decontamination Pit Collection Tank	BOP	NNS	ASME VIII	-	N	-	31
Excess Liquid Waste Pumps (2)	BOP	NNS	No Code	-	N	-	31
Excess Liquid Waste Filters (2)	BOP	NNS	ASME VIII	-	N	-	31
Excess Liquid Waste Demineralizers (2)	BOP	NNS	ASME VIII	-	N	-	31
Excess Liquid Waste System Piping and Valves	BOP	NNS	ANSI B31.1 / ASME III	-/3	N/Y	-/1	31
GASEOUS WASTE PROCESSING SYSTEM							
Waste Gas Decay Tanks (8)	NSSS	NNS	ASME III	3	Y	1	1,31,32,35
Gas Decay Tank Drain Pump	NSSS	NNS	ASME III	3	N	-	1,11,31
Catalytic Hydrogen Recombiners (2)	NSSS	NNS	ASME III	3	Y	1	1,4,7,12,31,35
Waste Gas Compressor (2)	NSSS	NNS	ASME III	3	Y	1	1,7,31,35
Waste Gas Compressor Pumps and Valves (2)	NSSS/BOP	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	1,5,31,35
Waste Gas Compressor Piping (2)	BOP	NNS	ANSI B31.1 / ASME III	-/3	N/Y	1	1,5,31,35
REACTOR BUILDING SPRAY SYSTEM							
Reactor Building Spray Pumps (2)	BOP	2a	ASME III	2	Y	1	7
Sodium Hydroxide Storage Tank	BOP	2b	ASME III	3	Y	1	-
Protective Chambers for Recirculation Sump Isolation Valves	BOP	2a	ASME III	MC	Y	1	-
Reactor Building Spray System Piping	BOP	2a	ASME III	2	Y	1	5
Reactor Building Spray System Valves	BOP	2a	ASME III	2	Y	1	5

RN 98-103

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
SPENT FUEL COOLING SYSTEM								
Refueling Water Storage Tank	BOP	2a	ASME III	2	Y	1	-	
Spent Fuel Cooling Pumps (2)	BOP	2b	ASME III	3	Y	1	-	
Spent Fuel Cooling Heat Exchangers (2)	BOP	2b	ASME III	3	Y	1	-	
Spent Fuel Purification Pump	BOP	NNS	No Code	-	N	-	-	
Spent Fuel Purification Demineralizer	BOP	NNS	ASME VIII	-	N	-	-	
Spent Fuel Purification Filter (2)	BOP	NNS	ASME VIII	-	N	-	-	RN
Spent Fuel Cooling System Piping	BOP	2b	ASME III	3	Y	1	5	98-103
Spent Fuel Cooling System Valves	BOP	2b	ASME III	3	Y	1	5	
NUCLEAR SAMPLING SYSTEM								
Residual Heat Removal Sample Cooler	BOP	2b	ASME III	3	Y	1	-	
Pressurizer Sample Cooler	BOP	2b	ASME III	3	Y	1	-	
Reactor Coolant Sample Coolers (2)	BOP	2b	ASME III	3	Y	1	-	
Steam Generator Blowdown Sample Coolers (3)	BOP	2b	ASME III	3	Y	1	-	
Volume Control Tank Gas Space Sample	BOP	NNS	ASME III	-	N	-	-	
Sample Sink	BOP	NNS	--	-	N	-	-	
Steam Generator Blowdown Sample Piping	BOP	2a	ASME III	2	Y	1	5	
Steam Generator Blowdown Sample Valves	BOP	2a	ASME III	2	Y	1	5	02-01
Nuclear Sampling System Piping	BOP	3	ASME III	3	Y	1	5	
Nuclear Sampling System Valves	BOP	3	ASME III	3	Y	1	5	RN
Reactor Coolant Sampling Delay Coils (2)	BOP	2a	ASME III	2	Y	1	-	98-103
CVCS Sampling Delay Coils (2)	BOP	3	ASME III	3	Y	1	-	
Flush Water Storage	BOP	NNS	--	-	N	-	-	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Flush Water Pumps (2)	BOP	NNS	--	-	N	-	13	
Auxiliary Sample Coolers (2)	BOP	NNS	--	-	N	-	13	
Auxiliary Sample Cooler Chiller	BOP	NNS	--	-	N	-	13	
Vacuum Pump	BOP	NNS	--	-	N	-	13	
Flush/Dilution Water Storage Tank	BOP	NNS	--	-	N	-	-	
Waste Pump	BOP	NNS	--	-	N	-	13	
Flush & Dilution Water Pump	BOP	NNS	--	-	N	-	13	
NUCLEAR SAMPLE PANEL								
Sample Panel Enclosure	BOP	NNS	--	-	N	-	-	RN 98-103
Pressurized Sample Flask	BOP	NNS	ASME VIII	-	N	-	-	
Gas Expansion Flask	BOP	NNS	ASME VIII	-	N	-	-	
Vacuum Expansion Flask	BOP	NNS	ASME VIII	-	N	-	-	
Auxiliary Gas Expansion Flask	BOP	NNS	ASME VIII	-	N	-	-	
Auxiliary Vacuum Expansion Flask	BOP	NNS	ASME VIII	-	N	-	-	
Liquid Sample Circulation Pump	BOP	NNS	--	-	N	-	13	
Liquid Sample Dilution Flask	BOP	NNS	ASME VIII	-	N	-	-	
Nuclear Sample Panel Valves	BOP	NNS	--	-	N	-	5,13	
Nuclear Sample Panel Tubing	BOP	NNS	--	-	N	-	5,13	
POST ACCIDENT HYDROGEN REMOVAL SYSTEM								
Electric Hydrogen Recombiner	NSSS	2b	--	-	Y	1	13	02-01
Piping and Valves	BOP	2a/NNS	ASME III/-	2/-	Y/N	1/-	5	RN 98-103
1000CC Collection Flasks (2)	BOP	NNS	ASME VIII	-	N	-	-	
Particulate Paper & Silver Zeolite Cartridge (2)	BOP	NNS	--	-	N	-	13	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Air Ejector (2)	BOP	NNS	--	-	N	-	13	
Purge Pressure Blower (2)	BOP	NNS	--	-	N	-	13	
Hydrogen Analyzer (2)	BOP	3	--	-	Y	1	-	
REACTOR MAKEUP WATER SYSTEM								
Reactor Makeup Water Storage Tank	BOP	2b	ASME III	3	Y	1	-	
Reactor Makeup Water Pumps (2)	BOP	2b	ASME III	3	Y	1	-	RN
Reactor Makeup Water System Piping	BOP	2b/NNS	ASME III/ ANSI B31.1	3/-	Y/N	1/-	-	98-103
Reactor Makeup Water System Valves	BOP	2b/NNS	ASME III/ ANSI B16.5	3/-	Y/N	1/-	-	
COMPONENT COOLING SYSTEM								
Component Cooling Pumps (3)	BOP	2b	ASME III	3	Y	1	-	
Component Cooling Heat Exchangers (2)	BOP	2b	ASME III	3	Y	1	-	
Booster Pumps (3)	BOP	2b	ASME III	3	Y	1	-	
Surge Tank	BOP	2b	ASME III	3	Y	1	11	
Component Cooling Drain Tank	BOP	NNS	AWWA D-100	-	N	-	-	98-01
Component Cooling Drain Tank Pump	BOP	NNS	HIS	-	N	-	-	
Chemical Injection Tank	BOP	NNS	-	-	N	-	13	
Chemical Injection Tank Agitator	BOP	NNS	-	-	N	-	13	
Chemical Injection Pump	BOP	NNS	-	-	N	-	13	RN
Component Cooling System Piping	BOP	2b	ASME III	3	Y	1	5	98-103
Component Cooling System Valves	BOP	2b	ASME III	3	Y	1	5	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes
SERVICE WATER SYSTEM							
Service Water Pump A	BOP	2b	ASME III	3	Y	1	-
Service Water Pump B	BOP	2b	ASME III	3	Y	1	-
Service Water Pump C	BOP	2b	ASME III	3	Y	1	36
Booster Pumps (2)	BOP	2b	ASME III	3	Y	1	-
Traveling Screens (3)	BOP	2b	ASME III	3	Y	1	14
Trash Racks (3)	BOP	2b	No Code	-	Y	1	-
Service Water System Piping	BOP	2b	ASME III	3	Y	1	5
Service Water System Valves	BOP	2b	ASME III	3	Y	1	5
MAIN STEAM SYSTEM							
From Steam Generators to Main Steam Isolation Valves							
Flow Elements	BOP/NSSS	2a	ASME III	2	Y	1	-
Piping	BOP	2a	ASME III	2	Y	1	5
Valves	BOP	2a	ASME III	2	Y	1	5
Components Downstream of Main Steam Isolation Valves	BOP	2a/NNS	ASME III/ ANSI B31.1	2/-	Y/N	1/-	15
FEEDWATER SYSTEM							
From Containment Isolation Valves to Steam Generators							
Piping	BOP	2a	ASME III	2	Y	1	5
Valves	BOP	2a	ASME III	2	Y	1	5
Components Upstream of Containment Isolation Valves	BOP/NSSS	2a/NNS	ASME III/ ANSI B31.1	2/-	Y/N	1/-	15

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TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
EMERGENCY FEEDWATER SYSTEM								
Pumps, motor driven (2)	BOP	2b	ASME III	3	Y	1	-	
Pump, turbine driven	BOP	2b	ASME III	3	Y	1	-	
Emergency Feedwater System Piping	BOP	2b	ASME III	3	Y	1	5	
Emergency Feedwater System Valves	BOP	2b	ASME III	3	Y	1	5	
Emergency Feedwater Control Valves Air Volume Tanks	BOP	-	ASME VIII	-	Y	1	24	RN 98-103
CONDENSATE SYSTEM								
Condensate Storage Tank	BOP	2b	ASME III	3	Y	1	-	
Remainder of Condensate System	BOP	NNS	ANSI B31.1	-	N	-	5	
STEAM GENERATOR BLOWDOWN SYSTEM								
Piping and Valves from Steam Generators to Containment Isolation Valves	BOP	2a	ASME III	2	Y	1	-	
Remainder of Steam Generator Blowdown System	BOP	NNS/QR	ASME VIII/ ANSI B31.1	-	N	-	-	RN 04-026
NUCLEAR BLOWDOWN PROCESSING SYSTEM								
Holdup Tank	BOP	NNS	ASME VIII	-	N	-	-	
Monitor Tank	BOP	NNS	ASME VIII	-	N	-	-	RN 98-103
Blowdown Spent Resin Storage Tank	BOP	NNS	ASME VIII	-	N	-	-	
Primary Demineralizers (2)	BOP	NNS	ASME VIII	-	N	-	-	
Polishing Demineralizers (2)	BOP	NNS	ASME VIII	-	N	-	-	
Demineralizer Inlet Filter	BOP	NNS	ASME VIII	-	N	-	-	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Spent Resin Sluice Filter	BOP	NNS	ASME VIII	-	N	-	-	
Demineralizer Outlet Filter	BOP	NNS	ASME VIII	-	N	-	-	
Holdup Tank Transfer Pumps (2)	BOP	NNS	HIS	-	N	-	-	
Monitor Tank Transfer Pump	BOP	NNS	-	-	N	-	13	
Spent Resin Sluicing Pump	BOP	NNS	-	-	N	-	13	
Piping	BOP	NNS	ANSI B31.1	-	N	-	-	
Valves	BOP	NNS	ANSI B16.5	-	N	-	-	
CRDM COOLING WATER								RN 17-029
CRDM Cooling Coils	BOP	NNS	ANSI B31.1	-	Y	-	41	
Remaining of CRDM Cooling Water System	BOP	NNS	ASME VIII/ ANSI B31.1	-	N	-	41	RN's 16-003 98-103
VENTILATION EQUIPMENT								
Reactor Building Cooling Units (4)	BOP	2b	ASME III	3	Y	1	16,17,25,27	
Reactor Building Charcoal Cleanup System Fans (2)	BOP	NNS	-	-	Y	1	18	
Reactor Building Purge Valves	BOP	2a	ASME III	2	Y	1	-	
Auxiliary Building Charcoal Exhaust System Fans (4)	BOP	NNS	-	-	N	-	13	
Fuel Handling Building Exhaust Fans	BOP	3	-	-	Y	1	18	
Spent Fuel Pool Supply Fan	BOP	NNS	-	-	N	-	13	
Residual Heat Removal/Spray Pump Room Cooling Unit	BOP	2b	ASME III	3	Y	1	17,18	
Charging Pump Room Cooling Unit	BOP	2b	ASME III	3	Y	1	17,18	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Auxiliary Building Charcoal Filter System Plenum (2)	BOP	NNS	-	-	N	-	13	
Fuel Handling Building Charcoal Filter Plenum (3)	BOP	3	-	-	Y	1	13,27	
Intermediate Building Pump Area Cooling Units (2)	BOP	2b	ASME III	3	Y	1	17,18	RN 98-103
Control Room Emergency Filtering System Fans	BOP	2b	-	-	Y	1	18	
Control Room Normal Supply Units	BOP	2b	ASME III	3	Y	1	17,18	
Relay Room Cooling System Units	BOP	2b	ASME III	3	Y	1	17,18	
Battery Room Air Supply Fans	BOP	2b	-	-	Y	1	18	
Battery Room Exhaust Fans	BOP	2b	-	-	Y	1	18	
Safety Class Water Chillers	BOP	2b	ASME III	3	Y	1	-	RN 99-053
Safety Class Chilled Water Pumps	BOP	2b	ASME III	3	Y	1	-	
Reactor Building Charcoal Cleanup Plenums	BOP	NNS	-	-	N	-	26	
Control Room Emergency Filter Plenums	BOP	2b	-	-	Y	1	13,27	
Service Water Pumphouse Supply Fans (2)	BOP	2b	-	-	Y	1	18	
ESF Switchgear Room Cooling Units (2)	BOP	2b	ASME III	3	Y	1	17,18	RN 98-103
Speed Switch Room Cooling Units (2)	BOP	2b	ASME III	3	Y	1	17,18	
Motor Control Center 12-28 Cooling Units (2)	BOP	2b	ASME III	3	Y	1	17,18	
Switchgear 63-01 Cooling Unit (1)	BOP	2b	ASME III	3	Y	1	17,18	
Duct Work, Dampers (including Control Room Outside Air Intake Isolation Valves), Supports for Safety Related Ventilation Systems	BOP	2b	-	-	Y	1	38	RN 09-025

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
FIRE PROTECTION SYSTEM								RN 98-103
Fire Pumps (2)	BOP	NNS	-	-	N	-	30	
Jockey Pump	BOP	NNS	-	-	N	-	30	
Fire Protection Piping and Fittings	BOP	NNS	-	-	N	-	30	
Fire Protection Valves	BOP	NNS	-	-	N	-	30	
Fire Protection Containment Isolation Valves	BOP	2a	ASME III	2	Y	1	37	02-01
Diesel Generator Building Ventilation Fans	BOP	2b	-	-	Y	1	18	
Emergency Feedwater Pump Room Ventilation Fans	BOP	2b	-	-	Y	1	18	
Piping and Ductwork for above Systems	BOP	2b/NNS	ASME III/ None	3/None	Y/N	1/-	19	
DIESEL GENERATOR FUEL OIL SYSTEM								RN 98-103
Fuel Oil Transfer Pumps (4)	BOP	2b	ASME III	3	Y	1	-	
Fuel Oil Day Tanks (2)	BOP	2b	ASME III	3	Y	1	-	
Fuel Oil Storage Tanks (2)	BOP	2b	ASME III	3	Y	1	-	
Fuel Oil Piping and Valves for Diesel Generators	BOP	2b	ASME III	3	Y	1	-	
DIESEL GENERATOR COOLING WATER SYSTEM								
Intercooler Pump	BOP	2b	-	-	Y	1	-	
Intercooler Heat Exchanger	BOP	2b	ASME VIII	-	Y	1	36	
Intercooler Thermostatic Valve	BOP	2b	ASME III	3	Y	1	-	RN 06-010
Jacket Water Heat Exchanger	BOP	2b	ASME VIII	-	Y	1	36	
Jacket Water Pump	BOP	2b	-	-	Y	1	-	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Jacket Water Motor Driven Auxiliary Pump	BOP	2b	ASME III	3	Y	1	-	
Jacket Water Expansion Tank	BOP	NNS	ASME VIII	-	Y	1	-	
Jacket Water Heater	BOP	2b	-	-	Y	1	-	
Jacket Water Thermostatic Valve	BOP	2b	ASME III	3	Y	1	-	
Cooling Water System Piping and Manual Valves	BOP	2b	ASME III	3	Y	1	-	
DIESEL GENERATOR STARTING SYSTEM								
Air Start Distributors	BOP	2b	-	-	Y	1	-	RN 98-103
Main Air Start Control Valves	BOP	2b	-	-	Y	1	-	
Air Filters	BOP	2b	-	-	Y	1	-	
Air Tanks	BOP	2b	ASME III	3	Y	1	-	
Shutdown Solenoids Valve	BOP	2b	ASME III	3	Y	1	-	
Fuel Rack Shutdown Cylinder	BOP	2b	-	-	Y	1	-	
Air Tank (accumulator)	BOP	2b	ASME III	3	Y	1	-	
Shuttle Valve	BOP	2b	ASME III	3	Y	1	-	
Air Start Solenoid Valves	BOP	2b	ASME III	3	Y	1	-	
Barring Gear Interlock	BOP	2b	-	-	Y	1	-	
Air Compressors	BOP	NNS	-	-	N	-	-	
Air Dryers	BOP	NNS	-	-	N	-	-	RN 04-039
DIESEL GENERATOR LUBRICATION SYSTEM								
Main Oil Pump, Engine Driven	BOP	2b	-	-	Y	1	-	RN 98-103
Thermostatic Valve	BOP	2b	ASME III	3	Y	1	-	
Main Lube Oil Strainer	BOP	2b	ASME III	3	Y	1	-	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
Auxiliary Motor Driven Oil Pump	BOP	2b	-	-	Y	1	-	RN
Auxiliary Strainer	BOP	2b	ASME III	3	Y	1	-	98-103
Electric Heater	BOP	2b	-	-	Y	1	-	
Oil Filter	BOP	2b	-	-	Y	1	-	
Motor Driven Rocker Prelube Pump	BOP	2b	-	-	Y	1	-	RN
Lube Oil Cooler	BOP	2b	ASME VIII	-	Y	1	36	06-010
Lube Oil Piping and Valves	BOP	2b	ASME III	3	Y	1	-	
DIESEL GENERATOR COMBUSTION AIR INTAKE AND EXHAUST SYSTEM								
Intake Filter/Silencers	BOP	2b	-	-	Y	1	-	
Exhaust Muffler	BOP	2b	-	-	Y	1	-	
Ducting	BOP	2b	-	-	Y	1	-	
REACTOR VESSEL OR CORE-RELATED								
Reactor Vessel Head and Shell Insulation	NSSS	QR	-	-	N	-	33	
Irradiation Sample Holder	NSSS	2a	-	-	Y	1	4,20	RN
Irradiation Samples	NSSS	NNS	-	-	N	-	1	98-103
Reactor Vessel Stud Tensioners	NSSS	NNS	-	-	N	-	6	
Reactor Vessel Support Shoes & Shims	NSSS	1	-	-	Y	1	4	
CRDM Dummy Can Assemblies	NSSS	NNS	-	-	N	-	6	RN
Reactor Vessel Core Support Structures & Internal Structures	NSSS	2a	-	-	Y	1	1,20	16-003
Reactor Vessel Internals Package	NSSS	2a	ASME III	2	Y	1	2	
Control Rod Guide Tubes	NSSS	2a	-	-	Y	1	1,2	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
RCC Full Length Assembly	NSSS	2a	-	-	Y	1	1,2	RN 16-003
Burnable Poison Rod Assembly	NSSS	NNS	-	-	N	-	1	
Primary & Secondary Sources	NSSS	NNS	-	-	N	-	1	
FUEL HANDLING AND STORAGE EQUIPMENT								
Reactor Cavity Seals	NSSS	NNS	-	-	N	-	6	
Cable Reels	NSSS	NNS	-	-	N	-	6,21	
Load Cell	NSSS	NNS	-	-	N	-	-	
Stud Hole Plug Handling Fixture	NSSS	NNS	-	-	N	-	21	
Stud Hole Plugs	NSSS	NNS	-	-	N	-	21	
Fuel Handling Machine	NSSS	3	-	-	Y	1	4	RN 98-103
Refueling Machine	NSSS	NNS	-	-	N	-	-	
New Fuel Storage Racks	NSSS	2b	-	-	Y	1	4	
Burnable Poison Assembly Rack Inserts	NSSS	NNS	-	-	N	-	6,21	
Spent Fuel Storage Racks	NSSS	2b	-	-	Y	1	4	
Upper Internal Storage Stand	NSSS	NNS	-	-	N	-	21	
RCC Changing Fixture	NSSS	NNS	-	-	N	-	-	
CRD Shaft Handling Fixture	NSSS	NNS	-	-	N	-	21	
CRD Shaft Unlatching Tool, Full Length	NSSS	NNS	-	-	N	-	6,21	
New Fuel Elevator Winch	NSSS	NNS	-	-	N	-	21	
New Fuel Elevator	NSSS	NNS	-	-	N	-	21	
New Fuel Assembly Handling Fixture	NSSS	NNS	-	-	N	-	6,21	
Spent Fuel Assembly Handling Tool	NSSS	3	-	-	Y	1	4	

TABLE 3.2-1 (Continued)
MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes	
								RN 98-103
Polar Crane	BOP	-	-	-	Y	1	23,24	
IHA Lift Rig Components	NSSS	1/NNS	ASME III	NF	Y/N	1/-	43	RN 16-003
CRDM Seismic Support Assembly (SSA)	NSSS	2a	ASME III	NF	Y	1	22	
Reactor Vessel Internals Lifting Device	NSSS	NNS	-	-	N	-	-	RN 98-103
Irradiation Sample Handling Tool	NSSS	NNS	-	-	N	-	6,21	
Burnable Poison Handling Tool	NSSS	NNS	-	-	N	-	6,21	RN 00-003
New RCC Handling Fixture	NSSS	NNS	-	-	N	-	6,21	
Crane Scales	NSSS	NNS	-	-	N	-	6,21	
Primary Source Installation Guide	NSSS	NNS	-	-	N	-	6,21	
Stud Tensioner Handling Device	NSSS	NNS	-	-	N	-	6,21	RN 98-103
RCC Thimble Plug Tool	NSSS	NNS	-	-	N	-	6,21	
Fuel Transfer System								
Fuel Transfer Tube and Flange	NSSS	2a	ASME III	MC	Y	1	4	
Track Upenders and Conveyor - Winches and Car	NSSS	QR	-	-	Y	-	4	RN 99-065
Control Panels and Hydraulic Units	NSSS	NNS	-	-	N	-	-	
Refueling Cavity Seal Ring	NSSS	NNS	-	-	N	-	6	RN 98-103
Neutron Detector Positioning Device	NSSS	2a	-	-	Y	1	4	
CRD Shaft Storage Racks	NSSS	NNS	-	-	N	-	-	
Lower Internals Storage Stand	NSSS	NNS	-	-	N	-	-	
Stud Nut Washer and Carrier	NSSS	NNS	-	-	N	-	-	
Irradiation Tube End Plug Seat Jack	NSSS	NNS	-	-	N	-	6	
Long Handle Tool Storage Rack	NSSS	NNS	-	-	N	-	6	
Fuel Handling Building Crane	BOP	-	-	-	Y	1	24	
Lift Yoke for HI-TRAC VW	BOP	-	ASME III	NF	Y	1	24	
Lift Yoke Extension for HI-TRAC VW	BOP	-	ASME III	NF	Y	1	24	RN 13-018
Cask Pedestal for HI-TRAC VW	BOP	-	ASME III	NF	Y	1	24	
VECASP for HI-TRAC VW	BOP	-	ASME III	NF	Y	1	40	

TABLE 3.2-1 (Continued)

MECHANICAL EQUIPMENT CLASSIFICATION

Component	Scope	ANS Safety Class	Code	Code Class	Seismic Category I	QA Class	Notes
INCORE INSTRUMENTATION							
Instrument Conduit and Couplings	NSSS	1	ASME III	1	Y	1	1
Seal Table Assembly	NSSS	1	ASME III	NF	Y	1	1
Flux Thimble Assemblies	NSSS	2	-	-	Y	1	1

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TABLE 3.2-1 NOTES

MECHANICAL EQUIPMENT CLASSIFICATION

Certain pressure retaining parts exist within a system which do not fall within the normal definition of pipes, pumps, or valves. In some cases these are not commercially available fabricated from materials conforming to ASTM specifications allowed in ANSI B31.1, ASME III or ASME VIII; in addition, instrumentation is specifically excluded from ASME III. Examples of these items are strainers, sight glasses, level switches, pressure transmitters, thermowells, etc. These items are specified and procured in a manner which ensures that these components are comparable with the remainder of the systems to preclude their structural failure under operating, accident, or test conditions. Also, certain consumable products used in conjunction with safety related equipment are considered safety related. Examples of these items are weld rod, diesel fuel oil, boric acid, lithium hydroxide, etc.

The NRC has granted relief per 10CFR50.55(a)(3) to licensees that choose to use the guidance provided in Generic Letter 89-09 (issued May 8, 1989) for procuring replacements that are not currently available in full compliance with the stamping and documentation requirements of Section III of the ASME Boiler and Pressure Vessel Code. The replacements meet all other applicable requirements of Section III (including third party inspection by an Authorized Nuclear Inspector) endorsed by NRC regulations and are procured in accordance with the SCE&G Quality Assurance Program Description. Replacements procured per this relief are designated by Note (36). RN 15-012

1. Meets "Quality Control System Requirements," Westinghouse QCS-1, which satisfies the requirements of 10CFR50, Appendix B.

2. Meets the quality assurance program of one of the Westinghouse NES Manufacturing Divisions, and is in accordance with 10CFR50, Appendix B.

3. As permitted by Paragraph NA-2134 of the ASME Code, Section III, this component is upgraded from the minimum required Code Class 2 to Code Class 1.

4. Meets "Quality Requirements for Manufacturer of Nuclear Plant Equipment," Westinghouse QCS-2, which satisfies the requirements of 10CFR50, Appendix B.

5. The classifications shown are for the predominant portion of the system. There may be portions that are classified higher or lower. Safety class boundaries are shown on applicable system diagrams. Seismic category, code, and QA classes for other safety classes are consistent with those other safety classes.

TABLE 3.2-1 NOTES (Continued)

6. Access for inspection and test is required by Westinghouse; however, no formal quality program approval is required.

7. Portions of equipment containing Component Cooling Water are Safety Class 2b, Code Class 3.

8. Services required to support a safety or other necessary function:

   a. Emergency power - automatic loading.
   b. Emergency power - manual loading.
   c. Component Cooling Water.
   d. Service Water.

9. Not stamped.

10. Designed in accordance with the Draft of the ASME Code for Pumps and Valves for Nuclear Power Plants, November, 1968. Ordered before the 1971 edition of the ASME Code, Section III became mandatory.

11. Outside the jurisdiction of ASME but designed, fabricated, and tested according to the ASME Code.

12. In the catalytic hydrogen combiner package, the control panel and gas analyzer shall be Non-Nuclear Safety, with no code requirements.

13. No applicable code. Built to supplier's standards.

14. ASME Code, Section III, applies to screen wash header only.

15. Main Steam and Feedwater piping, excluding branch lines, between the associated isolation valves and the wall between the Intermediate Building and Turbine Building satisfies all requirements, except for stamping, of the ASME Code, Section III, Code Class 2.

16. No code. The fans and motors are specifically designed for operation in the containment atmosphere under both normal operating and post LOCA conditions.

17. Code and Code Class apply to unit coils.

18. No code. The fans are designed and manufactured in accordance with the intent of ANS Safety Class 2b.

TABLE 3.2-1 NOTES (Continued)

19. No code. Ductwork is designed to withstand expected pressures and shocks for the section of the plant in which it is located.

20. Any reactor vessel internal, the single failure of which could cause release of a mechanical piece having potential for direct damage (as to the vessel cladding) or flow blockages, shall be classified to a minimum of Safety Class 1. 02-01

21. Failure can cause no nuclear safety problem, although an economic loss may result.

22. Portions which transmit loading from CRDM seismic supports are Safety Class 2a.

23. Applicable code is Crane Manufacturers Association of America, Specification No. 70 for Electric Overhead Traveling Cranes.

24. Equipment is not ANS Safety Class but is safety related.

25. Supports for Reactor Building Cooling Units are safety related.

26. The supports which attach the plenums to the Reactor Building wall are seismically qualified and safety related.

27. HEPA filters are an integral part of the Reactor Building Cooling Units, the Fuel Handing Building charcoal filter plenum, and the control room emergency filter plenums.

28. Portions of this system are used for post accident sampling. The Post Accident Sampling System is not safety related. Procedures associated with post accident sampling are under the pertinent requirements of the Quality Assurance Program.

29. Refer to Table 5.2-1 for information regarding the design code addenda.

30. See the response to NRC Questions 421.77 and 421.78a.

31. Equipment in this system was originally supplied to ASME III code. This system has subsequently been downgraded to NNS. Future equipment procurement and modifications will meet or exceed the code and quality requirements of Regulatory Guide 1.143, Revision 1, with exceptions noted in FSAR Appendix 3A.

32. The seismic design criteria given in FSAR Section 3.7 are applicable.

TABLE 3.2-1 NOTES (Continued)

33. Quality Related (QR) is a V. C. Summer classification used to impose controls for procurement, installation, or repair of that equipment.

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35. Equipment in this system was originally supplied to ASME III code. This system was then downgraded to NNS in accordance with Regulatory Guide 1.143, Revision 1, with exceptions noted in FSAR Appendix 3A. Subsequently, the portions of this system which may have the potential of carrying combustible concentration gas mixtures were upgraded to a Quality Related Classification (including seismic design and falldown protection) as defined by QRP-1.

36. Meets the requirements for ASME Section III component replacements pursuant to the 10CFR50.55(a)(3) relief granted in Generic Letter 89-09.

37. Includes valves XVG06772, XVG06773, XVG06797, & XVC06799 only. 02-01

38. Control Room Outside Air Intake Isolation Valves XVB00003A,B-AH and XVB00004A,B-AH are ANS safety class 2b butterfly valves. These valves serve the function of air flow isolation in the Control Room HVAC system which is a ducted ventilation system. This function does not necessitate the imposition of ASME code requirements. RN 09-025

39. The Alternate Seal Injection (ASI) pump would have technically been NSSS scope, but was added to the station per internal SCE&G modification ECR50780C. The ASI pump and interconnecting piping are ASME Code Class 2 for pressure boundary only. The actual function of supplying flow to the reactor coolant pump seals is NNS. While not required, the ASI pump was conservatively procured as an active pump and its motor as 1E. RN 11-027

40. Equipment is not ANS Safety Class but is quality related per TRP-43. RN 13-018

41. The CRDM Cooling Water System has been placed in a lay-up condition per ECR 50897. RN 16-003

42. CETNA Assembly consists of various pressure retaining subcomponents

43. ANSI N14.6 and NUREG 0612 codes also apply. Refer to IHA Certified Design Spec., AREVA 08-9241459

TABLE 3.2-2

CLASSIFICATION OF STRUCTURES

	Seismic Category I	Non-Seismic Category	Method of Tornado Missile Protection (1)	Tornado Missile (2) Grade to 30 Feet	Tornado Missile (2) Above 30 Feet	
Reactor Building, Liner, Penetrations and Hatches (5) (6)	X		A,C	1,2,3,4,5,6,7	3,5,6,7	
Reactor Building Interior Structures (5) (6) (7)	X		Reactor Building	Reactor Building	Reactor Building	RN 01-113
Control Building (5) (6)	X		A,B,C,D,E	1,2,3,4,5,6,7	3,5,6,7	
Auxiliary Building (5) (6)	X		A,B,D,E	1,2,3,4,5,6,7	3,5,6,7	
Fuel Handling Building (5) (6)	X					
a. General structures for fuel pools and protective barriers for equipment (see Figures 3.8-58, 3.8-59, 3.8-60)			A,B,C,D,E	1,2,3,4,5,6,7	3,5,6,7	
b. Steel superstructure (3) (see Figures 3.8-58, 3.8-59, 3.8-60)			(4)	(4)	(4)	
Intermediate Building (5) (6)	X		A,B,C,D,E	1,2,3,4,5,6,7	3,5,6,7	
Diesel Generator Building (5) (6)	X		A,B,C,D,E	1,2,3,4,5,6,7	3,5,6,7	
Service Water Intake, Pumphouse and Discharge Structures (5) (6)	X		A,B,C	1,2,3,4,5,6,7	Not applicable	RN 01-113
Service Water Pond Dams	X		Not applicable	Not applicable	Not applicable	
Supports for Safety Class Components	X		A,B,C,D,E	1,2,3,4,5,6,7	3,5,6,7	
Turbine Building		X				
Substation Structure and Control Houses		X				RN 12-001
Water Treatment Building		X				
Circulating Water Intake and Discharge Structure		X				
Service Building		X				
Auxiliary Boiler House		X				
Warehouse(s)		X				
Guard House(s)		X				
Monticello Reservoir Dams		X				
Jetty		X				
Sanitary Waste Facility		X				
Industrial Waste Facility		X				
Containment Access Runway		X				

TABLE 3.2-2 (Continued)

CLASSIFICATION OF STRUCTURES

NOTES:

1. Method of tornado missile protection is as follows:

   A. Reinforced concrete walls.
   B. Reinforced concrete slabs.
   C. Reinforced concrete barriers.
   D. Orientation.
   E. Probability studies, probability criteria $< 10^{-7}$.

2. Numbers correspond to tornado missiles identified in Table 3.5-5.

3. Refer to discussion of probability study, response to Question 010.7.

4. Steel frame is designed to maintain its integrity under tornado missile impact.

5. Biological shielding and missile barrier structural components related to Category I structures identified in Table 3.2-2 are classified as either safety-related and meet Seismic Category I requirements or as Quality Related designed for the Safe Shutdown Earthquake.

6. Fire stops for electrical penetrations are safety related.

7. The supports for the Reactor Building charcoal cleanup filter plenums which attach the plenums to the Reactor Building wall are seismically qualified and safety related.

TABLE 3.2-3

SUMMARY OF CODES AND STANDARDS FOR COMPONENTS

Component	Safety Class 1	Safety Class 2a	Safety Classes 2b and 3	Non-Nuclear Safety Class	
Pressure Vessels	ASME Code, Section III, Class 1 (1)(1A)	ASME Code, Section III, Class 2 (2)	ASME Code, Section III, Class 3 (3)	ASME Code, Section VIII, (4) Division 1	RN 16-003
Piping	ASME Code, Section III, Class 1 (1)	ASME Code, Section III, Class 2 (2)	ASME Code, Section III, Class 3 (3)	ANSI B31.1.0 (5)	
Pumps and Valves	ASME Code, Section III, Class 1 (1)	ASME Code, Section III, Class 2 (2)	ASME Code, Section III, Class 3 (3)	Pumps: Hydraulic Institute Standards or Manufacturer's Standards Valves: ANSI B16.5	RNs 98-098 01-113
Storage Tanks (0-15 psig)	-	ASME Code, Section III, Class 2 (2)	ASME Code, Section III, Class 3 (3)	API 620	
Storage Tanks (atmospheric)	-	ASME Code, Section III, Class 2 (2)	ASME Code, Section III, Class 3 (3)	AWWA D-100 or API 650	

TABLE 3.2-3 (Continued)

SUMMARY OF CODES AND STANDARDS FOR COMPONENTS

NOTES TO TABLE 3.2-3

Specific editions and addenda of codes are dependent upon contract award dates for individual components.

1. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," including revisions to the following paragraphs published in the 1971 edition: NB-2510, NB-2541, NB-2553, NB-2561; except as noted in Table 3.2-1 for specific items in the Reactor Coolant System.

    a. Replacement Reactor Vessel Closure Head designed to ASME Section III, 2007 Edition through 2008 Addenda Rules for Construction of Nuclear Power Plant Components and Application Codes Cases. Reconciled to the Code of Record. RN 16-003

2. ASME, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," including a or b, below:

    a. Revisions to the following paragraphs published in the 1971 edition: NC-2510, NC-2571, NC-2573.

    b. The 1974 edition with all applicable addenda.

3. ASME, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," including a or b, below:

    a. Revisions to the following paragraphs published in the 1971 edition: ND-2510, ND-2571.

    b. The 1974 edition with all applicable addenda.

4. ASME, Boiler and Pressure Vessel Code, Section VIII, "Pressure Vessels."

5. American National Standards Institute, ANSI B31.1.0, "Power Piping Code," 1967 issue with addenda through 1972. RN 98-098

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

3.3.1.1 Design Wind Velocity

A design wind velocity of 100 mph was used in the analysis of Seismic Category I structures. This value represents the maximum wind velocity at the site for an altitude of 30 feet above grade and for a 100 year recurrence interval. Sections 2.3.1 and 2.3.2 provide the basis for this selection of design wind velocity.

The wind velocity profile for the Reactor Building was computed on the basis of Table 1a of ASCE Paper No. 3269 [1]. The 140 mph velocity at the top of the Reactor Building, which is 167 feet above grade, was conservatively applied for the full height of the structure as shown by Figure 3.3-2. 02-01

Reference [1] recommends a gust factor of 1.1 for structures similar in size to the Reactor Building. However, due to the conservative application of the design wind velocity over the Reactor Building height, the design gust factor was taken as 1.0.

The wind velocity profile for other Seismic Category I structures was computed based upon Reference [1], using equation (2) with x equal to 0.20. These velocities were multiplied by a gust factor of 1.1, based upon the recommendations of Reference [1]. The resulting wind velocity profiles are shown by Figure 3.3-1. These velocities are somewhat conservative since equation (2) of Reference [1] applies to "coastal areas" and the plant is located in an "inland area," as defined by Reference [1].

The conservative wind velocity profiles result in "actual gust factors used" as given below:

REACTOR BUILDING

Height Above Grade (Feet)	Percent of Total Exposed Height	Wind Velocities (mph): Required By Ref. [1] Table 1a	Wind Velocities (mph): Used in Design	Actual Gust Factor Used	Stated Design Gust Factor
0-50	30	100	140	1.40	1.0
50-150	60	120	140	1.17	1.0
150-167	10	140	140	1.0	1.0

Other Seismic Category 1 Structures

Height Above Grade (Feet)	Percent of Total Exposed Height	Wind Velocities (mph) Required By Ref. [1] Table 1a	Wind Velocities (mph) Used in Design*	Actual Gust Factor Used	Stated Design Gust Factor
0-30	30	100	110	1.10	1.1
30-50	20	100	122	1.22	1.1
50-99	50	120	140	1.17	1.1

* $V_Z = 1.1\ [V_{30}(Z/30)^{0.20}]$

3.3.1.2 Determination of Applied Forces

The effective pressures resulting from the design winds were computed using Equation (6) and the shape coefficients from Reference [1]. Resulting pressures for the Reactor Building and other Seismic Category I structures are shown by Figures 3.3-1 and 3.3-2.

3.3.2 TORNADO LOADINGS

3.3.2.1 Applicable Design Parameters

The design parameters applicable to the design basis tornado are as follows:

1. Rotational wind speed of 290 mph.

2. Translational wind speed of 70 mph.

3. Atmospheric pressure drop of 3 psi at the rate of 2 psi/sec.

3.3.2.2 Determination of Forces on Structures

Forces upon structures resulting from tornado winds were determined as follows:

1. The effective pressures on the Reactor Building due to a 360 mph wind were computed using the same shape coefficients as were used for design wind analysis [1]. These pressures were combined with the 3 psi atmospheric pressure and are shown in Figure 3.3-3. For all other Seismic Category I structures, the design pressures are as follows: 02-01

a. Maximum positive pressure on each building wall was obtained by applying a shape coefficient of 0.9 to the 332 lb/ft^2 dynamic pressure, neglecting the atmospheric pressure drop. This resulted in a positive pressure of approximately 300 lb/ft^2.

b. Maximum negative pressure (suction) on each building wall and roof was obtained by applying a shape coefficient of -0.7 to the 332 lb/ft^2 dynamic pressure and superimposing on this value the 3 psi suction due to atmospheric pressure drop. This resulted in a negative pressure (suction) of approximately 664 lb/ft^2.

The design tornado wind velocity of 360 mph was assumed to be constant with height. 02-01

2. No reduction in tornado wind pressure due to venting was assumed.

3. Structures were designed to resist the effects of tornado generated missiles described in Section 3.5.1.4. In the design it is assumed that these missiles can occur simultaneously with the tornado loads described in item 1, above.

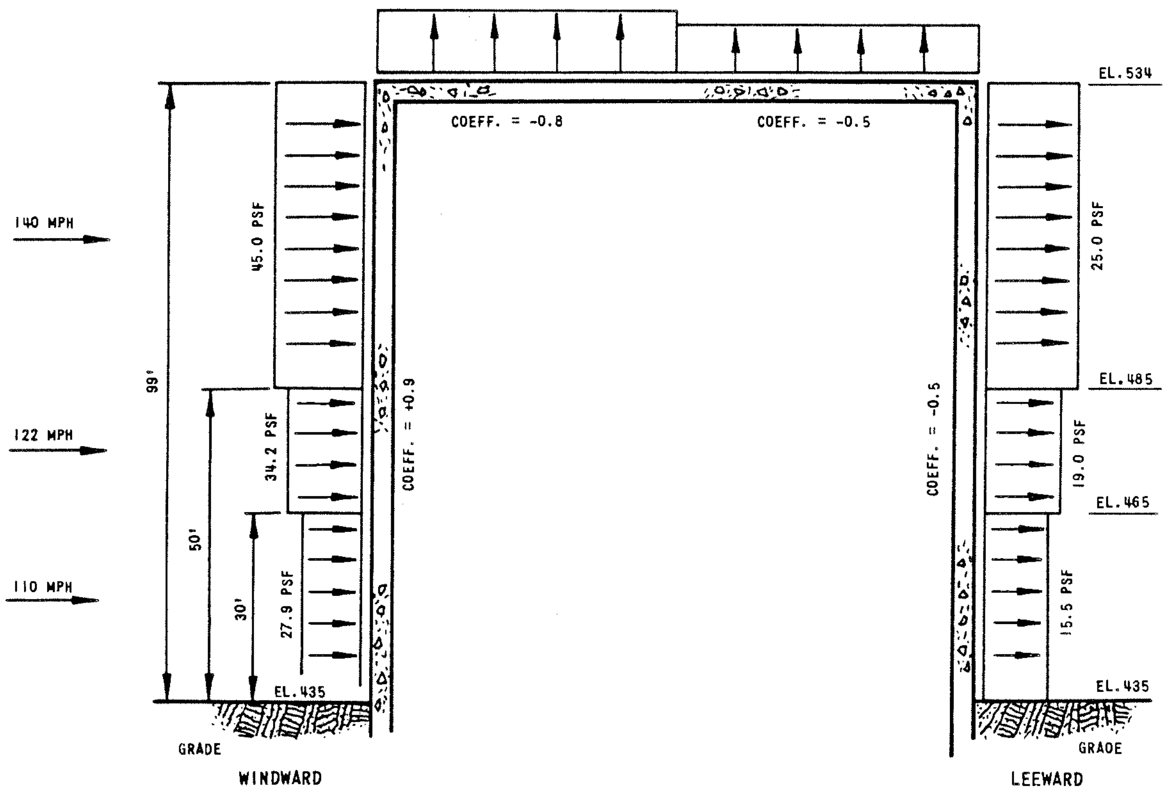
4. General stability of the entire structure was checked using net pressures of wind, tornado, and other loads as described in Section 3.8. However, individual elements are designed for the maximum effect in either direction.

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

Structures not designed to resist tornado loadings are either located so that their failure does not affect structures designed for tornado loads or are designed not to collapse. The metal siding and roofing of the Fuel Handling Building will blow off under tornado loading. The structural steel frames supporting the overhead traveling crane will resist the tornado and wind loads. Non-Seismic Category I structures may lose parts or portions but such parts or portions are less serious missiles than those described in Section 3.5.1.4.

3.3.3 REFERENCE

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

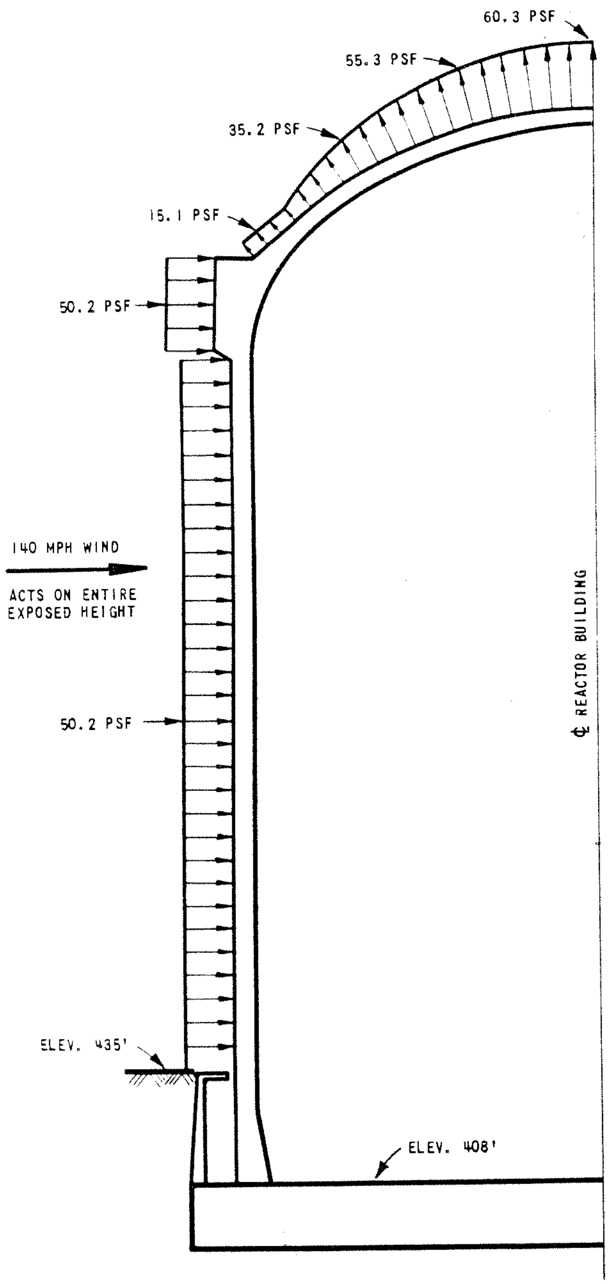


ELEVATION OF BUILDING

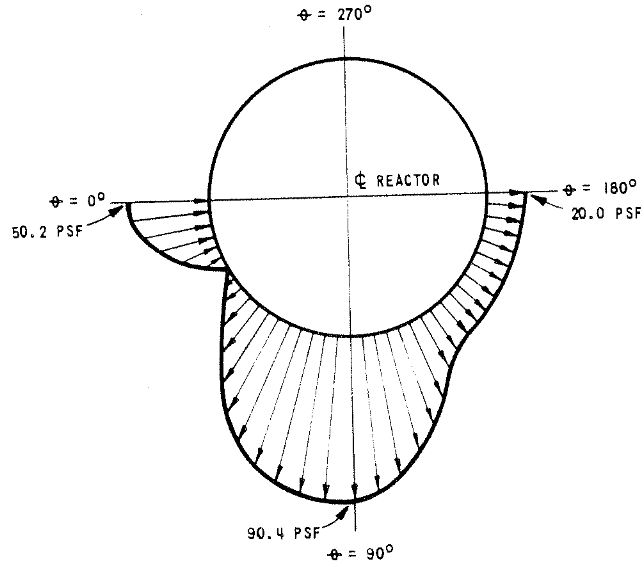
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Wind Pressures for Other Seismic Category I Buildings

Figure 3.3-1



EFFECTIVE PRESSURES FOR 100 MPH DESIGN WIND

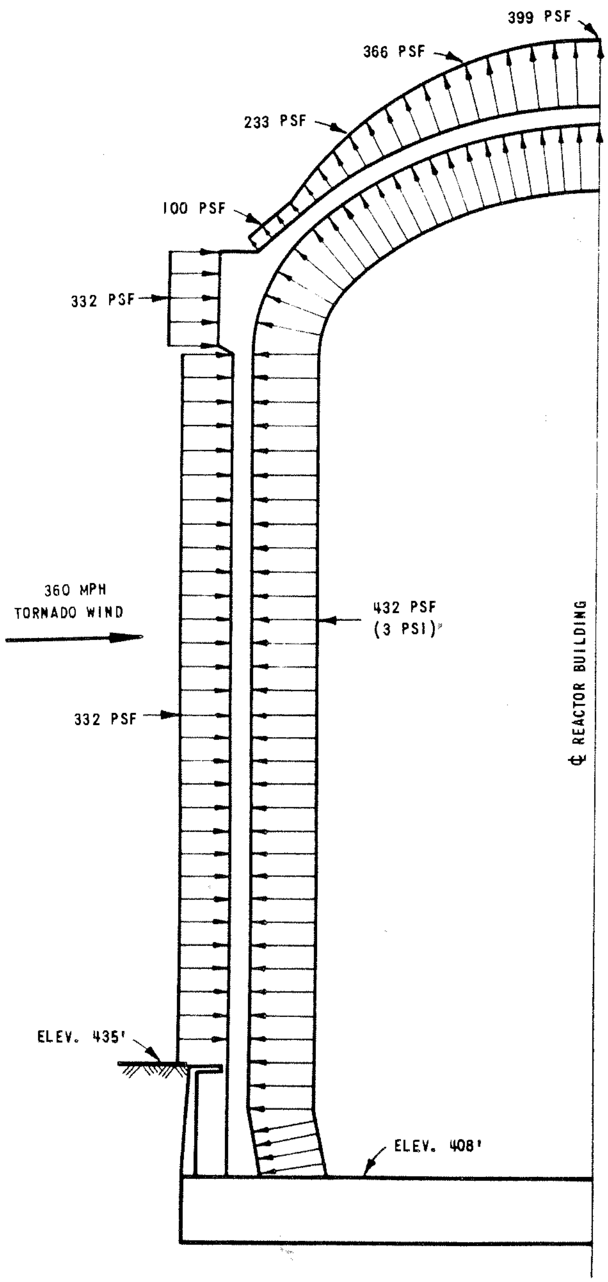


DISTRIBUTION OF EFFECTIVE PRESSURE AROUND CIRCUMFERENCE

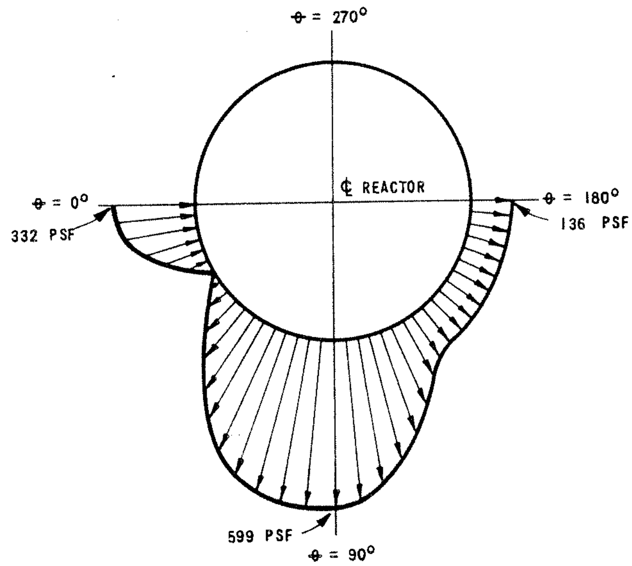
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Wind
Pressures

Figure 3.3-2



EFFECTIVE PRESSURES FOR 360 MPH TORNADO WIND VELOCITY



DISTRIBUTION OF EFFECTIVE PRESSURE AROUND CIRCUMFERENCE

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Tornado Pressures

Figure 3.3-3

3.4 WATER LEVEL (FLOOD) DESIGN

All Seismic Category I structures are designed for a maximum flood water level elevation of 437'-6" at the berm and a high water table ground water level elevation of 420' ± 3'.

3.4.1 FLOOD PROTECTION

3.4.1.1 Safety-Related Systems and Components Protected Against Floods

All safety-related systems and components are protected against surface flooding by grading the site to carry surface water away from structures housing these systems and components. See Section 2.4 for details. Systems located below grade are protected as described in Sections 3.4.1.2 and 3.4.1.3.

3.4.1.2 Structures that House Safety-Related Equipment

The portions of Seismic Category I structures located below finished grade are protected on their outside surfaces by a continuous waterproofing membrane. In addition, the auxiliary building mat is protected by a waterproof membrane on the bottom surface.

Below grade penetrations for conduit and piping are provided with waterproofing membrane covers. No personnel or equipment hatches are located in outer walls below grade.

3.4.1.3 Means of Providing Flood Protection for Vulnerable Equipment

In the event that leakage occurs, additional flood protection is provided for safety class components, equipment, and systems located below grade by sloping of floors to sumps and pumping of any water from these sumps.

3.4.1.4 Procedures Required and Implementation Times for Cold Shutdown for Flood Conditions

Special procedures for use in the event of flooding are not required. See Section 2.4.14.

3.4.1.5 Safety-Related Systems or Components Capable of Normal Function While Flooded

Safety-related electrical cabling in underground duct runs is capable of fulfilling its normal function when completely or partially flooded.

3.4.2 ANALYSIS PROCEDURES

Seismic Category structures are designed for buoyancy. No Seismic Category I structures become unstable with respect to uplift or overturning due to load combinations including the design water level. Load combinations are presented in Section 3.8. The plant site is protected against potential floods up to elevation 438.0' (for details see Section 2.4.10). Therefore, dynamic effects of flooding are not applicable and were not considered.

3.5 MISSILE PROTECTION

3.5.1 MISSILE SELECTION AND DESCRIPTION

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Missile Selection

Rotating and high pressure system components located inside the following buildings and areas are examined to identify and classify potential missiles:

1. Auxiliary Building
2. Intermediate Building
3. Penetration access areas
4. Fuel Handling Building
5. Diesel Generator Building
6. Service Water Pump House

Pumps located outside the Reactor Building are evaluated for missiles associated with potential failure due to overspeed. The maximum no-load speed of these pumps is equivalent to associated motor operating speed. No pipe break or other single failure in a pump suction line results in pump speeds exceeding the no-load speed. Pump casings are designed to contain impeller fragments should an impeller fail. Therefore, missiles generated by pumps outside the Reactor Building are not postulated.

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

Valves with threaded stems and backseats are not postulated as potential missile sources because of the unlikelihood of coincident failure of both the threaded stem and the backseat. Valve bonnets are not postulated as potential missile sources when the allowable stress for bonnet retaining ring material is less than 20 percent of the material yield strength.

Bolted valve bonnets are not postulated as missiles since the safety factor for the bolting is greater than 4. Valves outside the Reactor Building were reviewed and none were postulated as missiles.

(The main feedwater check valve bonnets were previously postulated as missiles. These valves were replaced, because of feedwater water hammer concerns, with those of another design which meet the criteria described above).

Non-seismically supported piping and components were considered as a source of gravity missiles.

High pressure compressed gas containers are postulated as missile sources due to rupture and rocketing. Discussion of this analysis is provided in Section 3.5.1.1.3. RN 01-006

3.5.1.1.2 Missile Protection Methods

The design basis for protection of safety-related systems and components against postulated missiles outside the Reactor Building is to provide for continued safe operation or shutdown under operating conditions, including transients and accidents.

Protection of safety-related systems and components from postulated missiles is accomplished by one, or more, of the following methods:

1. Compartmentalization

   Equipment is enclosed in missile protected compartments.

2. Barriers

   Barriers are erected to stop missiles either at the source or at equipment locations.

3. Separation

   Redundant components of vital systems are separated by one or a combination of the following methods:

   a. Components are located within separate cubicles.

   b. Adequate spatial separation is provided between redundant components and electric circuits.

   c. Physical barriers are installed, such as concrete block, concrete or steel walls.

4. Equipment Design

   Structures or components can, by virtue of design, withstand impact of postulated missiles without loss of function.

5. Strategic Orientation

   Equipment or components are so oriented that postulated missile paths are directed away from equipment and components requiring protection.

6. Distance

   Equipment is located out of range of postulated missiles.

Safety-related instrument and control components and instrument impulse lines outside the Reactor Building, which are required for safe plant shutdown, are not in the paths of postulated missiles.

A seismic induced physical interactions program ensured that safety-related systems and components required for safe plant shutdown would not be prevented from performing their intended safety functions as a result of physical interactions with non-safety related structures, systems, and components. This program consisted of seismically supporting such items as non-safety cable trays, HVAC ducts, and electric cabinets in areas containing safety-related components required for safe plant shutdown. Plant walkdowns were then carried out to identify for future evaluation other possible missile sources such as non-safety piping, tanks, pumps, motors, and light fixtures; which, if their supports failed, could impact on safety-related systems and components required for safe plant shutdown. Additional supports were provided where required.

Safety related systems and components outside the Reactor Building that are required for safe plant shutdown under all plant conditions are listed in Table 3.5-2.

3.5.1.1.3 Missiles Generated by Components Containing Compressed Gas

Missiles are postulated to result from rupture of a gas cylinder, rocketing of a gas cylinder due to failure of the valve, and explosion of a hydrogen storage tank. The events just listed bound the spectrum of missile-generating accidents associated with compressed gas containers. None of these events is considered to be a credible occurrence; however, without supporting probabilistic data, the potential impacts of these events on safety-related equipment has been examined. Components containing compressed gas are installed at various locations outside of buildings housing safety-related equipment, with any exceptions identified and analyzed on a case-by-case basis. Section 3.5.1.4 demonstrates that components located outside of Seismic Category I structures need not be protected from design basis missiles. Therefore, missiles generated by compressed gas components are normally only analyzed for their ability to penetrate Seismic Category I structures. Postulated missiles generated by compressed gas components inside of Seismic Category I structures are specifically analyzed to demonstrate postulated failures associated with compressed gas components have no adverse affects on safety-related equipment inside of Seismic Category I structures.

RN 01-006

RN 01-006 11-041

Reference [4] provides formulas for determining the penetration of concrete by high velocity missiles and the mass and velocity of fragments produced by explosions. The penetration of structural concrete by a solid, cylindrical missile of high strength steel is estimated by:

$$X_f = (0.162 \times 10^{-5})\, W_f^{0.4}\, v_s^{1.8}\, K$$

Where:

X_f = penetration depth (in)

W_f = fragment weight (oz)

v_s = striking velocity (ft/sec)

K = hardness factor (1 for armor piercing steel)

Note: the equation listed above is based on a concrete compression strength of 5000 psi. According to Reference [4], this equation can be corrected for other strengths by multiplying X_f by the square root of the ratio of 5000 psi to the strength of interest. A compression strength of 3000 psi was used in this analysis, which is listed in Section 3.8.4.1 as the minimum compressive strength of Seismic Category I structures. RN 01-006

The maximum fragment weight resulting from an explosion of a cylindrical metal casing is given by Reference [4] as:

$$W_f = \left(M_a \ln\left(8 \frac{wc}{M_a^2} \right) \right)^2$$

Where:

$$M_a = 0.3 t^{\frac{5}{6}} d^{\frac{1}{3}} \left(1 + \frac{t}{d} \right)$$

and

W_f = fragment weight (oz)

W_c = casing weight (lb)

t = casing thickness (in)

d = casing diameter (in)

The explosion or rupture is conservatively assumed to occur at point blank range from the target structure, and drag effects are ignored. The striking velocity, then, is equal to the maximum fragment velocity resulting from the explosion. The maximum fragment velocity is given by Reference [4] as:

$$v_o = G\sqrt{\frac{\frac{W}{W_c}}{1+0.5\frac{W}{W_c}}}$$

Where:

v_o = initial fragment velocity (ft/sec)

W = explosive yield (lb-TNT)

W_c = casing weight (lb)

G = Gurney velocity for TNT (6490 ft/sec)

For explosions occurring more than 20 feet from the target structure, it is reasonable to calculate a reduced striking velocity. Reference [4] gives this velocity as: RN 01-006

$$v_s = v_o\, e^{-0.004 R_f W_f^{-\frac{1}{3}}}$$

Where:

v_s = striking velocity (ft/sec)

v_o = initial fragment velocity (ft/sec)

R_f = distance traveled by the fragment (ft)

Three bounding cases were analyzed: nitrogen cylinder rupture, hydrogen storage tank explosion and nitrogen cylinder rocket. Detailed analyses of these cases is found in Reference [18]. The results of these analyses are summarized below.

Rupture of a standard high-pressure nitrogen cylinder may result in a release of energy equivalent to 1 lb-TNT. An equivalent TNT explosion could potentially generate a fragment missile of up to 5 ounces, traveling at an initial velocity of 585 ft/sec. Such a missile would penetrate less than 0.4 inches into a concrete structure.

Explosion of one of the bulk hydrogen storage tanks may result in a release of energy equivalent to 274 lb-TNT. A 70 ounce, 1526 ft/sec fragment missile could be generated by such an explosion. Since hydrogen tanks are not located within 275 feet of any Seismic Category I structure, the striking velocity is reduced to 1168 ft/sec. Such a missile could penetrate up to 3.8 inches into a concrete structure. All Seismic Category I structures at this site are 2 feet thick, double re-enforced; therefore, this missile may result in spalling and cracking of the concrete, but will not penetrate and damage components housed within. RN 01-006 12-023

Assuming that an accident were to occur (such as breaking the valve after falling over) that resulted in a high-pressure nitrogen cylinder rocketing, the cylinder could reach an impulse of 95,500 lbm-ft/sec. The striking velocity for a 100 lb cylinder would be 955 ft/sec. Conservatively assuming the cylinder missile weight to be 2240 ounces (140 lb), the resulting penetration depth would be 10.6 inches. Since a gas cylinder is a thin-walled vessel and not a solid projectile, much of the impact energy would be dissipated by vessel deformation, resulting in less actual penetration. Cylinder impact may result in cracking and spalling of the structure, but will not penetrate and damage components housed within. RN 01-006

3.5.1.2 Internally Generated Missiles (Inside Containment)

3.5.1.2.1 Missile Selection

Systems and components located inside the Reactor Building are examined to identify and classify potential missiles.

Non-Safety related systems inside containment are low energy systems. Therefore, generation of missiles from these systems need not be postulated.

Catastrophic failure of the reactor vessel, steam generator, pressurizer, reactor coolant pump casings, and piping resulting in the generation of missiles is not postulated because massive and rapid failure of these components is unlikely due to: the material characteristics; inspections; quality control during fabrication, erection, and operation; conservative design; and prudent operation as applied to the particular component.

Valves within the reactor coolant pressure boundary have been examined to identify potential missiles. As a result of this review, there are no credible failures that could result in missile formation. Therefore, valves are not considered a credible source of missiles.

Motor operated and air operated valves contain design features which effectively preclude the ejection of valve stems. Valves are designed against bonnet to body connection failure and subsequent bonnet ejection by means of the following:

1. Compliance with the ASME Code, Section III.

2. Control of load during tightening of bonnet to body studs.

Reactor coolant pressure containing parts are designed in accordance with the requirements of the ASME Code, Section III, Class 1. The complete valves are hydrostatically tested in accordance with the ASME Code, Section III. Valve bodies and bonnets are also volumetrically and surface tested to verify soundness.

In the special case of those valves located on the top of the pressurizer which extends above the operating deck, certain missiles, although also considered incredible, are postulated and protection is provided, due to the greater potential for damage to the Reactor Building liner, engineered safety features pipes, and components located outside the secondary shield wall. For this case, the pressurizer and associated piping and valves are enclosed in a separate concrete cubicle designed to contain the postulated pressurizer valve missiles listed in Table 3.5-3.

Postulated missiles generated inside the Reactor Building are listed in Table 3.5-3. Missile numbers in the following paragraphs are assigned for ease of reference to Table 3.5-3.

Missile 1 consists of three resistance temperature detectors located on the hot legs of reactor coolant piping. Rupture locations are postulated around the weld between the boss and the piping. A 10 degree expansion angle jet is postulated for these jet propelled missiles. Application of the modified Petry formula [1] results in estimated penetration of 0.01 inches into the 3 foot, 6 inch thick concrete shield wall. 02-01

Missile 2 consists of three resistance temperature detectors located on the cold legs of reactor coolant piping. Rupture locations are postulated around the weld between the boss and the piping. A 10 degree expansion angle jet is postulated for these jet propelled missiles. Application of the modified Petry formula [1] results in estimated penetration of 0.01 inches into the 3 foot, 6 inch thick concrete shield wall. 02-01

Missile 3 consists of three 6 inch safety valves located in the pressurizer compartment and on the top of the pressurizer. A tearing between bonnet and valve body is postulated for these jet propelled missiles. A velocity of 180 ft/sec is estimated as a result of application of analytical methods. Application of the modified Petry formula [1] results in estimated penetration of 2.1 inches into the 2 foot thick concrete slab.

Missile 4 consists of two 4 inch diaphragm operated control valves located in the pressurizer compartment and on the side of pressurizer. Rupture locations are postulated between the drive shaft and vee ball. It is postulated that the valve drive shaft and diaphragm actuator are forced out by the 2300 psig reactor coolant pressure and form a jet propelled missile. A velocity of 130 ft/sec is estimated as a result of application of analytical methods. Application of the modified Petry formula [1] results in estimated penetration of 0.5 inches into the pressurizer compartment wall for Valve 1-PCV-444D and 2.66 inches into the wall corbal concrete missile shield provided for Valve 1-PCV-444C.

Missile 5 was a control rod drive mechanism (CRDM) housing plug located in the reactor compartment on the top of reactor. The top plug on the CRDM Housing has been eliminated by the new design of the CRDM Travel Housings. The end of the travel housing is an integral forging of the overall Travel Housing, thus eliminating the end Cap Housing Plug. This change was made with the Replacement Reactor Vessel Closure Head per ECR 50868. RN 16-003

Missile 6 is a control rod drive shaft located in the reactor compartment on the top of reactor. After a postulated failure of the top of the Travel Housing, the drive shaft is postulated to be pushed out of the core by the 2300 psi differential pressure across the drive shaft. The drive shaft and control rod cluster, latched together, are assumed to be fully inserted when the accident starts. After approximately 12 feet of travel, the control rod cluster 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. It is assumed that the control rod cluster is completely stopped by the upper support plate. The drive shaft continues to be accelerated until its top hits the missile shield. A velocity of 180 ft/sec is estimated as a result of the application of analytical methods with a velocity of 185 ft/sec use for conservation. Missile 6 travels approximately 50 inches from the top of the CRDM Travel Housing to the bottom of the Integrated Head Assembly (IHA) mounted missile shield. The application of the BRL Formula[2] and the modified Petry Formula[1] determine the penetration of the missile into the steel plate that makes up the IHA mounted missile shield. Conservative application of this criteria identifies a penetration of approximately 1.357 inches into the 2 inch thick missile shield. The missile can originate from any of the 48 CRDM locations on the Replacement Head. Various locations were evaluated to ensure structural integrity of the missile shield and the IHA are maintained. RN 16-003 RN 16-003

Missile 7 consists of two instrument wells located on the wall of the pressurizer. Failure of the weld between the instrument well and the pressurizer wall is postulated. A velocity of 100 ft/sec is estimated. Application of the modified Petry formula [1] results in estimated penetration of 3 inches into the 3 foot thick pressurizer compartment wall.

Missile 8 consists of 78 pressurizer heaters located on the bottom shell of the pressurizer. It is postulated that the pressurizer heaters could loosen and become jet propelled missiles. A velocity of 55 ft/sec is estimated. Application of the modified

Petry formula [1] results in penetration of 0.144 inches into the 3 foot, 6 inch thick concrete floor.
Within the Reactor Building there are no gravity missiles which could constitute a threat to safety class equipment or systems. Gratings, handrails and stair treads inside the Reactor Building are fastened in place by welds, pins or clamps. Overhead light fixtures, located above safety related equipment, are installed by means of hangers capable of withstanding seismic forces.

Non-Seismically supported piping and components were considered as a source of gravity missiles.

3.5.1.2.2 Missile Protection Methods

The design basis is that postulated missiles generated within the Reactor Building in coincidence with a loss of coolant accident, do not cause loss of function of any redundant engineered safety feature. Protection of safety related equipment and redundant components of vital systems from postulated missiles is accomplished by one, or more, of the methods described in Section 3.5.1.1.2.

Thicknesses of barriers, including walls, slabs and specially designed barriers, which protect safety class equipment or systems satisfy the criteria discussed in Section 3.5.3. Thus, scabbing or the generation of secondary missiles from the nonimpacted face of such a barrier is precluded. Concrete fragments ejected from the impacted face (spalling effect), if any, will have energies too low for consideration as missiles due to the small weight and velocity of such fragments. Fragments and the initial missile constitute no threat as gravity missiles to safety class equipment or systems as secondary missiles during the drop following impact.

Safety related instrument and control components and instrument impulse lines inside the Reactor Building, which are required for safe plant shutdown, are not in the paths of postulated missiles.

A seismic induced physical interactions program ensured that safety-related systems and components required for safe plant shutdown would not be prevented from performing their intended safety functions as a result of physical interactions with non-safety related structures, systems, and components. This program consisted of seismically supporting such items as non-safety cable trays, HVAC ducts, and electric cabinets in areas containing safety related components required for safe plant shutdown. Plant walkdowns were then carried out to identify for future evaluation if other possible missile sources such as non-safety piping, tanks, pumps, motors, and light fixtures; which, if their supports failed, could impact on safety related systems and components required for safety plant shutdown. Additional supports were provided where required.

Safety related structures, systems, and components inside the Reactor Building which are required for safe shutdown of the plant under all operating conditions are listed in Table 3.5-4.

3.5.1.3 Turbine Missiles

RN 11-015

Turbine missiles are discussed in Reference [19].

3.5.1.4 Missiles Generated by Natural Phenomena

Potential tornado missiles which are assumed to be generated by the design basis tornado, described in Section 3.3., are listed in Table 3.5-5. The properties of the missiles, needed to determine penetrations, are also indicated by Table 3.5-5.

The utility pole and the automobile missiles are limited to elevations of 30 feet above grade within one-half mile of the plant structures. Hence, these missiles are not postulated as missiles on roofs which are at least 30 feet above grade.

Virgil C. Summer Nuclear Station components important to safety are, with certain exceptions listed in Table 3.5-6, housed within Seismic Category I structures that are designed to withstand the effects of the design basis tornado, including tornado missiles. The exceptions include safety class components located wholly, or partially, outside of Seismic Category I structures. These components, however, are not necessary to ensure the following:

1. Integrity of the reactor coolant pressure boundary.

2. Long term capability to shut down the reactor and maintain safe shutdown conditions.

3. Capability to prevent accidents that could result in potential offsite exposures that are a significant fraction of the guideline exposures of 10 CFR 100.11 or 10 CFR 50.67. RN 12-034

The safety related components listed in Table 3.5-6 are discussed in the following paragraphs.

The locations of the refueling water storage tank (RWST), sodium hydroxide (NaOH) storage tank, makeup water storage tank (MWST), and the condensate storage tank (CST) are shown by Figures 1.2-1, 1.2-4, 1.2-5, and 1.2-10. Emergency feedwater (EFW) pump suction line and EFW pump recirculation line piping is also connected to the condensate storage tank and partially located outdoors. RN 07-032

The diesel generator combustion air intakes are protected by missile resistant structural labyrinths and therefore are considered to be indoors. A short segment of the diesel generator exhaust pipes are exposed on the roof of the Diesel Generator Building. The general arrangement of the diesel generator intakes and exhausts are shown by Figures 1.2-13 and 1.2-14.

The diesel generator day tank vents and the diesel generator fuel oil storage tank vents extend outside the tornado protected structures. The location of the diesel generator fuel oil storage tanks is shown in Figure 1.2-1.

A portion of the EFW turbine exhaust steam stack is not physically protected from tornados since the stack extends through the roof of the structure in which it is located. The general arrangement of this stack is shown in Figures 1.2-11,1.2-13, and 1.2-26.

Portions of the main steam (MS) system power operated relief valve discharge stacks are not physically protected from tornados since the stacks extend through the roofs of the structures in which they are located. The MS power operated relief valve discharge stacks are not classified as safety related but provide the capability to allow for a controlled cooldown of the primary system. The general arrangement of these stacks is shown in Figures 1.2-5,1.2-12,1.2-13, and 1.2-26.

The control room air intake ductwork penetrates the wall and roof of the Control Building and the general arrangement is shown in Figure 1.2-26. The Control Building exhaust air relief heads are located on the Control Building roof and are not provided with tornado missile shields. The general arrangement for these components is provided in Figures 1.2-16 and 1.2-26. The chilled water expansion tanks are located on the Control Building roof underneath the tornado missile shield provided for the Control Building outside air openings. Based on the structure and component arrangement the expansion tanks are partially exposed to a potential tornado missile strike. The general arrangement of the chilled water expansion tanks is shown in Figures 1.2-16 and 1.2-26. RN 07-032

The safety class, seismic category, quality assurance class, and code designation of systems and components; and the seismic category of structures; are provided in Tables 3.2-1 and 3.2-2, respectively. The location of equipment listed in Table 3.2-1 is shown by layout drawings, Figures 1.2-1 through 1.2-29. Those safety related items required for integrity of the reactor coolant pressure boundary, safe shutdown, or to prevent accidents which could result in potential offsite radiation exposure are either located within buildings designed for tornado missile protection, are provided with missile shields, or are located external to tornado missile proof structures. Those components located external to tornado missile proof structures are listed in Table 3.5-6. The refueling water storage tank, sodium hydroxide storage tank, reactor makeup water storage tank, and condensate storage tank including the outdoor portion of the EFW suction line are designed to mitigate the consequences of a design basis accident. However, these tanks and components would either be emptied during the injection phase of the post accident sequence or alternate sources of makeup water have been provided as discussed in Section 3.5.3. Therefore, these tanks are not required as part of the long term Emergency Core Cooling System (see Appendix 3A discussion on Regulatory Guide 1.117) and do not require protection from tornado missiles.

The probability of tornado missiles striking the remaining components listed in Table 3.5-6 is evaluated and included in the strike probability computation. Protection of these outdoor components against tornado missiles is demonstrated by strike probability computations which indicate that the tornado missile strike probability for all missiles and targets is less than 1.0 x 10^{-7}. Therefore, no additional physical missile protection is provided. RN 07-032

Systems and components listed in Table 3.2-1 required to ensure the integrity of the reactor coolant pressure boundary and to maintain safe shutdown conditions or to provide the capability to prevent accidents which could result in exceeding offsite radiation exposure limits, are provided tornado missile protection through location within Seismic Category I structures, missile barriers (see Table 3.5-7), or probability analysis. The total probability per year of potential tornado missiles, generated by the design basis tornado, striking external safety related targets or penetrating a critical opening and striking critical components beyond the exposed walls and roofs of Seismic Category I structures is determined. Tornado missile protection is provided for the missiles listed in Table 3.5-5 when the total probability, P_T, per year per missile satisfies the following equation: RN 07-032

$$P_T = \sum_{i=1}^{i=n} P_1 x P_2 x P_3 x P_4 x P_5 x P_6 x P_7 x P_8 \leq 1.0 x 10^{-7}$$

where:

n = Total number of external targets and openings on the exposed surfaces of Seismic Category I structures. RN 07-032

P_1 = Probability of the design basis tornado striking the reactor site = 1.0 x 10^{-3} (see Reference [14])

P_2 = Probability of a missile being acted upon by a maximum wind over a sufficiently long distance, conservatively assumed to be 1.0.

P_3 = Probability of an object maintaining an orientation inside the tornado which exposes its maximum cross-sectional area to the full force of the wind. Since missiles tend to tumble, this probability will be quite low; a conservative estimate is 1.0 x 10^{-1} (see Reference [15]).

P_4 = Probability of a missile being hurled to the exact location of the specific target. Missiles of the type being considered could land anywhere within the area confined by the width of tornado damage path. This area is about

$$\frac{\pi(500)^2}{4} = 196{,}350\ \text{ft}^2$$

compared to the area of the external target or the opening, A_T. Therefore, the probability of the geometric center of the object being hurled to the external target or opening is:[15] RN 07-032

$P_4 = A_T/196{,}350$

P_5 = Probability of the missile being in an orientation that will allow it to penetrate the target opening. Since the missile would have no preferred angle of impact on the target opening, the probability of impacting within the critical angle required to cause penetration is dependent upon the length of the missile and the general dimensions of the opening. Therefore the probability can be expressed as: RN 07-032

$$P_5 = \frac{(2\theta_1)}{(180)} \frac{(2\theta_2)}{(180)}$$ RN 01-113

Where:

θ_1 = Arcsin x/lm,

θ_2 = Arcsin y/lm,

x = One dimension of the opening,

y = Other dimension of the opening,

lm = Length of the missile, or
P_5 = 1.0 for external targets. RN 07-032

P_6 = Probability of a missile impacting a target opening of specific dimension and the ability of the missile to pass through the target opening by nature of the frontal area of the missile.

$$P_6 = \frac{A_T - A_M}{A_T}$$

Where:

A_T = Area of target

A_M = Frontal area of missile

If A_T is greater than A_M, $P_6 = 1.0$.
If A_T is less than A_M, $P_6 = 0.0$, or
$P_6 = 1.0$ for external targets. RN 07-032

P_7 = Probability that a missile will pass around structures or barriers to external targets or openings on the exposed surfaces or walls of Seismic Category I structures. The directional probability density per unit solid angle can be expressed as:

$$P_7 = \frac{1}{2}\left(\frac{\theta_H}{180} \quad x \quad \frac{\theta_V}{180}\right) \leq 0.5$$

Where θ_H and θ_V are the angles in the horizontal and vertical planes, respectively, through which the missile must pass to strike the opening.

P_8 = Probability of a missile striking the exact location of components required to ensure the integrity of the reactor coolant pressure boundary and to maintain safe shutdown conditions or to provide the capability to prevent accidents which could result in exceeding offsite radiation exposure limits after the missile has passed through the opening on the exposed surface or wall of a Seismic Category I structure. The directional probability density per unit solid angle per component can be determined using the following expression:

$$P_8 = \frac{\theta_H}{B_H} \quad x \quad \frac{\theta_V}{B_V}$$

Where θ_H and θ_V are the angles in the horizontal and vertical planes, respectively, through which the missile must pass to strike the critical component and B_H and B_V are the preferred angles of impact through which the missile must pass to penetrate through the opening in the horizontal and vertical planes, respectively. The angles B_H and B_V are defined by the wall or roof thickness and the physical dimensions of the opening.

P_8 is evaluated for all critical components located behind the subject opening.

$P_8 = 1.0$ for external targets. RN 07-032

3.5.1.5 Missiles Generated by Events near the Site

3.5.1.5.1 Introduction

RN 01-006

In Section 2.2.3, it was determined that there were no design basis accidents external to the site (events with an annual probability of occurrence greater than 10^{-7}) which would generate high energy missiles and fragments impacting the Reactor Building walls. The events analyzed include transportation, (highway, railway and air) industrial (including pipelines) and military proximity to the site.

3.5.1.5.2 Penetration by Onsite Missiles

Compressed gas containers stored or installed onsite are considered potential sources of high velocity fragment missiles. Evaluation of these missiles are included in Section 3.5.1.1.3. RN 01-006

3.5.1.5.3 Pipeline Explosions

The nearest pipeline to the site containing natural gas is located about 13,000 feet from the Reactor Building. A surface ignition of the natural gas (methane) carried in this line would not produce a detonation resulting in high energy fragments. However, if leaking gas were to permeate the soil, it could produce a cratering explosion, throwing soil up and away from the leak point. The detonation speeds of the exploding volume would be reduced below those of ordinary gas mixtures. The maximum range of soil and small rocks in the pipeline cover are estimated at 500 feet, which is the minimum safe distance recommended for separating personnel from ditching explosions using TNT[5]. Thus, no detrimental effects on plant structures or personnel are considered credible.

3.5.1.5.4 Offsite Hazards Less Likely Than 10^{-7} Per Year

3.5.1.5.4.1 Industrial Facilities

Although a storage site for explosives was identified (see Section 2.2.2) about 2.5 miles from the reactor, the maximum range of high energy fragments from detonation of these explosives would be approximately 1 mile. This range is determined by extrapolating the velocity decay to 0.1 ft/sec, a value which is below the damage threshold for structures. Thus, no detrimental effects on plant structures or personnel are considered credible.

3.5.1.5.4.2 Transportation Facilities

The probability of a rail car accident involving ordinance and explosive materials on the rail line near the site was determined in Section 2.2.3 to be less than 10^{-7} per year. In 1975, no traffic of this type was carried by rail within 100 miles of the site by the Southern Railway (which operates the lines near the site).

3.5.1.5.4.3 Military Activities

Military activities generating missiles, fragments and shells within 30 miles of the site are discussed in Section 2.2.2.

3.5.1.6 Aircraft Hazards

The analysis of potential aircraft hazards is discussed in Section 2.2.2.5. Results of this analysis indicate an extremely low probability that an aircraft failure in the vicinity of the site would result in a plant failure. Based upon this extremely low probability, discussed in Section 2.2.2.5, the Virgil C. Summer Nuclear Station design does not consider aircraft impact.

3.5.2 SYSTEMS TO BE PROTECTED

Safety class systems and components and Seismic Category I structures whose failure could result in radiological consequences offsite or that are required to shut down the plant and maintain it in a safe condition are listed in Tables 3.5-2 and 3.5-4. Separation and independence of these systems, components, and structures are demonstrated by the system drawings presented in the applicable sections which discuss those systems.

3.5.3 BARRIER DESIGN PROCEDURES

The exposed walls and roofs of Seismic Category I structures have minimum concrete thicknesses of 24 inches and are reinforced each way, each face with a minimum of number 9 bars at 12 inches, center to center. Based upon the results of tests at Sandia Laboratories [6,7,8] and by Calspan Corporation[9], these structural elements are more than adequate to resist impact of the postulated tornado missiles.

The Fuel Handling Building exterior walls are not designed to prevent the penetration of tornado missiles. However, the walls of the spent fuel pool and safety dam components are protected and designed to prevent penetration, spalling or overall failure due to impact of tornado generated missiles.

Entry of tornado generated missiles into the top of the spent fuel pool (28 feet above grade) is very unlikely.

The Fuel Handling Building framework is designed with sufficient redundancy that failure of any structural element due to missile impact does not cause a general failure. The framework is also designed to resist tornado generated pressure forces. The metal wall and roof panels are designed to collapse under tornado pressure loads. However, due to their light weight and large area, these panels will not cause damage to the spent fuel elements should they fall into the spent fuel pool.

The loss of function of the refueling water storage tank, sodium hydroxide (NaOH) storage tank, makeup water storage tank and condensate storage tank will not affect the capability to shut down the reactor and maintain it in a safe shutdown condition (hot standby) since either the tanks will be emptied during the injection phase or alternate sources of makeup water are provided. These alternate makeup water sources are protected against tornado missiles or are housed within Seismic Category I structures, designed to withstand the effects of the design basis tornado, including tornado missiles. RN 07-032

The alternate source of borated makeup water for the primary system is the boric acid tanks; the alternate source for the secondary system is the Service Water System (see Section 10.4.9.2).

Loss of function of the diesel generator combustion air intakes and exhausts is not anticipated. The intakes are protected by missile resistant structural labyrinths (see Figure 9.5-10). Only a short section of each diesel generator exhaust pipe is exposed above the Diesel Generator Building roof. The probability of a tornado missile impacting the short exposed length of pipe has been determined and included in the tornado missile strike probability computation which concluded that protection is provided based on a low strike probability. Table 3.5-6 has been revised to indicate that only the short sections of exhaust piping are outdoors. The intakes are no longer located outdoors. RN 07-032

Loss of function of the Control Building outside air and relief openings is not anticipated. The outside air openings are protected by missile shields (see Figure 1.2-26). The probability of impact of a tornado missile on the exhaust air relief heads has been included in the tornado missile strike computation which concluded that protection is provided based on a low strike probability. In addition, the Control Building Ventilation System can be operated satisfactorily in the recirculation mode even assuming loss of function of the outside air and relief openings. RN 07-032

Barrier requirements for internally generated missiles are checked on a case by case basis using either the modified National Research Defense Committee (NRDC) formula [10], for concrete barriers, or the BRL formula [11], for steel barriers.

Effective loads due to impact of these missiles are derived by idealizing the target as an equivalent single-degree-of-freedom structure.

Ductility ratios for reinforced concrete structures are in accordance with Reference [12]. Barriers of steel are designed with the ductility ratio for flexure not to exceed 20 for plates, walls, or slabs and 1.3 for columns.

For missiles considered to be nondeformable, such as steel missiles, the investigation encompasses the cases of no penetration and penetration. This procedure is in accordance with that outlined by Williamson and Alvy [13].

In the case of no penetration, momentum transfer principles are used to calculate an equivalent static load. This load is shown to be less than the load that the structure can withstand.

For the case of penetration, energy balance techniques are used to define a triangular load-time history. The required ductility factor for the target structure is estimated using standard analytical procedures and is shown to be within the permissible value.

The above approaches are found to be conservative when considering the impact of deformable missiles. A significant portion of the kinetic energy of such a missile is dissipated in the deformation of the missile and this fact is taken into account. The impact of the missile upon a rigid barrier is studied and a reaction-time history for the barrier is developed. An idealized version of this time history is then treated as the loading in the target structure response analysis. The corresponding ductility factor is shown to be within the permissible value.

Wall and roof concrete thicknesses and concrete strengths at the specified age for each exterior tornado missile barrier are summarized by Table 3.5-7. In addition, Table 3.5-7 provides a list of components protected by exterior tornado missile barriers. These components are required to ensure the integrity of the reactor coolant pressure boundary and to maintain safe shutdown conditions or to provide the capability to prevent accidents which could result in exceeding offsite radiation exposure limits.

The minimum compressive strength of structural concrete for the exposed walls and roofs of Seismic Category I structures is 3000 psi in 28 days. The thicknesses of exposed walls and roofs of Seismic Category I structures are defined earlier in this section. The safety class, seismic category, quality assurance class, and code designation of systems and components are provided in Table 3.2-1. The seismic category of structures is presented in Table 3.2-2. The location of equipment listed in Table 3.2-1 is shown by Figures 1.2-1 through 1.2-29.

3.5.4 REFERENCES

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19. V. C. Summer Nuclear Station, "Main Turbine Missile Probability and Reliability Reports," VCSNS Technical Report TR03880-002, dated July 12, 2011. RN 11-015

TABLE 3.5-2

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Chemical and Volume Control System		9.3.4		
Letdown Heat Exchanger	Auxiliary Building		None	Compartmentalization by concrete wall
Reactor Coolant Filter	Auxiliary Building		None	Compartmentalization by concrete wall
Volume Control Tank	Auxiliary Building		None	Concrete wall
Seal Water Return Filter	Auxiliary Building		None	Compartmentalization by concrete wall
Seal Water Heat Exchanger	Auxiliary Building		None	Compartmentalization by concrete wall
Charging Pumps	Auxiliary Building		None	Physical separation and compartmentalization by concrete wall
Seal Water Injection Filters	Auxiliary Building		None	Physical separation and compartmentalization by concrete wall
Boric Acid Tanks	Auxiliary Building		None	Concrete wall
Boric Acid Transfer Pumps	Auxiliary Building		None	Physical separation and compartmentalization by concrete wall

TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Boric Acid Filter	Auxiliary Building		None	Concrete wall
Boric Acid Blender	Auxiliary Building		None	Concrete wall
Chemical and Volume Control System Piping[1]	Auxiliary Building Penetration Access Area		None	Physical separation
Safety Injection System		6.3		
Safety Injection System Piping	Auxiliary Building Penetration Access Area, Fuel Handling Building		None	Physical separation
Residual Heat Removal System		5.5.7		
RHR Pumps	Auxiliary Building		None	Physical separation and compartmentalization by concrete wall
Residual Heat Exchangers	Auxiliary Building		None	Physical separation and compartmentalization by concrete wall
Residual Heat Removal System Piping	Auxiliary Building Penetration Access Area		None	Physical separation

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TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Reactor Building Spray System		6.2.3		
Reactor Building Spray Pumps	Auxiliary Building		None	Compartmentalization and physical separation by concrete wall
Sodium Hydroxide Storage Tank	Auxiliary Building		None	Compartmentalization by concrete wall
Reactor Building Spray System Piping[(1)]	Auxiliary Building Penetration Access Area		None	Physical separation
Spent Fuel Cooling System		9.1.3		
Refueling Water Storage Tank	Auxiliary Building		None	Compartmentalization by concrete wall
Component Cooling System		9.2.2		
Component Cooling Pumps	Intermediate Bldg.		None	Separation by distance
Component Cooling Heat Exchangers	Intermediate Bldg.		None	Physical separation RN 01-113

TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection	
Booster Pumps	Intermediate Bldg.		None	Separation by distance	
Surge Tank	Intermediate Bldg.		None	Compartmentalization by concrete wall	
Component Cooling System Piping(1)	Auxiliary Building Penetration Access Area, Intermediate Bldg.		None	Physical separation	RN 01-113
Main Steam System		10.3			
Main Steam System Piping(1)	Penetration Access Area, Intermediate Bldg.		None	Strategic orientation	RN 01-113
Feedwater System		10.4.7			
Feedwater System Piping(1)	Penetration Access Area, Intermediate Bldg.		None	Strategic orientation	RN 01-113
Emergency Feedwater System		10.4.9			
Emergency Feedwater Pumps	Intermediate Bldg.		None	Compartmentalization and physical separation by concrete wall	

TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Emergency Feedwater System Piping(1)	Penetration Access Area Intermediate Bldg.		None	Strategic orientation
Reactor Makeup Water System		9.2.7		
Reactor Makeup Water Storage Tank	Auxiliary Building		None	Compartmentalization by concrete wall
Reactor Makeup Water Pumps	Auxiliary Building		None	Physical separation and compartmentalization by concrete wall
Reactor Makeup Water System Piping(1)	Auxiliary Building		None	Physical separation
Service Water System		9.2.1		
Service Water Pumps	Service Water Pumphouse		None	Physical separation and compartmentalization by concrete wall
Service Water Booster Pumps	Intermediate Bldg.		None	Compartmentalization by concrete wall
Service Water System Piping(1)	Intermediate Bldg. Penetration Access Area, Service Water Pumphouse		None	Physical separation

TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Chilled Water System		9.4.7		
Chilled Water Pumps	Intermediate Bldg.		None	Compartmentalization by concrete wall
HVAC Mechanical Chillers	Intermediate Bldg.		None	Compartmentalization by concrete wall
Chilled Water System Piping(1)	Intermediate Bldg. Auxiliary Building Control Building		None	Physical separation
Emergency Diesel Generator Services System		9.5.4, 9.5.5, 9.5.6, 9.5.7, 9.5.8		
Fuel Oil Transfer Pumps	Diesel Generator Building		None	Physical separation by concrete wall
Diesel Generator Coolers	Diesel Generator Building		None	Physical separation by concrete wall
Air Receiver Tanks	Diesel Generator Building		None	Physical separation by concrete wall
Fuel Oil Day Tanks	Diesel Generator Building		None	Physical separation by concrete wall

TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Emergency Diesel Generator Services System Piping(1)	Diesel Generator Building		None	Physical separation by concrete wall
Emergency Diesel Generators	Diesel Generator Building	8.3	None	Physical separation by concrete wall
Electrical Components		8.0		
120V Inverters, Nuclear Steam Supply System	Control Building		None	-
120V, a-c, Vital Distribution Panels	Control Building		None	-
Battery Bus Panels	Intermediate Bldg.		None	Physical separation by concrete wall
DC Distribution Panels	Intermediate Bldg.		None	Physical separation by concrete wall
125V Batteries	Intermediate Bldg.		None	Physical separation by concrete wall
Battery Chargers	Intermediate Bldg.		None	Physical separation by concrete wall

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TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Diesel Generator Control Cubicles	Diesel Generator Building		None	Physical separation by concrete wall
Component Cooling Pump Speed Switches	Intermediate Building		None	Physical separation by concrete wall
Service Water Pump Speed Switches	Service Water Pumphouse		None	Physical separation by concrete wall
Component Cooling Pump Transfer Switch	Intermediate Bldg.		None	Concrete wall
Charging Pump Transfer Switch	Auxiliary Building		None	Concrete wall
Engineered Safety Features Motor Control Centers	Intermediate Bldg.			
1DA2X			None	Concrete wall
1DB2X			None	Concrete wall
Engineered Safety Features Motor Control Centers	Auxiliary Building			
1DA2Y			None	Concrete wall
1DB2Y			None	Concrete wall

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TABLE 3.5-2 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES OUTSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against	Provisions for Protection
Engineered Safety Features Motor Control Centers	Diesel Generator Building		None	Physical separation by concrete wall
Engineered Safety Features Motor Control Centers	Service Water Pumphouse		None	Physical separation by concrete wall
7.2 KV Switchgears	Intermediate Bldg.		None	Concrete wall
7.2 KV Switchgears	Service Water Pumphouse		None	Physical separation by concrete wall
480V Engineered Safety Features Unit Substations	Intermediate Bldg.			
1DA1			None	Concrete wall
1DA2			None	Concrete wall
480V Engineered Safety Features Unit Substation 1DB1	Auxiliary Building		None	Concrete wall
480V Engineered Safety Features Unit Substation	Service Water Pumphouse		None	Physical separation by concrete wall

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(1) Piping includes pipes, fittings and valves.

TABLE 3.5-3

POSTULATED MISSILES INSIDE THE REACTOR BUILDING

Missile Number	Description	Tag Number	Elevation and Location	Weight (lbs)	Impact Area (in^2)	Effect	
1	9 RTD's of Reactor Coolant Hot Legs	N/A	436'-9" Inside Secondary Shield Wall	11	3.14	Will hit operating floor at elevation 459'-6", and penetrate 0.01 inches into 3'-6" thick concrete	
2.	9 RTD's of Reactor Coolant Cold Legs	N/A	436'-9" Inside Secondary Shield Wall	11	3.14	Will hit operating floor at elevation 459'-6", and penetrate 0.01 inches into 3'-6" thick concrete	
3	3 Six Inch Safety Valves	1-8010A 1-8010B 1-8010C	480'-2-1/2" Pressurizer Compartment	350	38.5	Will penetrate 2.1 inches into 2 foot thick concrete slab at elevation 486'-6"	
4	2 Four Inch Diaphragm Operated Control Valves	1-PVC-444C	464'-6" Pressurizer Compartment	200	50.0	Will penetrate 2.66 inches into concrete missile shield provided on wall.	02-01
		1-PVC-444D	464'-6" Pressurizer Compartment	200	50.0	Will penetrate 0.5 inches into 1.5 foot thick concrete wall.	02-01
5							RN 16-003

TABLE 3.5-3 (Continued)

POSTULATED MISSILES INSIDE THE REACTOR BUILDING

Missile Number	Description	Tag Number	Elevation and Location	Weight (lbs)	Impact Area (in^2)	Effect	
6	Control Rod Drive Shaft	N/A	451'-2" Reactor Compartment	120	2.41	Will penetrate approximately 1.357 inches of the 2 inch steel plate missile shield	RN 16-003
7	2 Instrument Wells of Pressurizer	N/A	1 at 477'-3¼" 1 at 444'-8⅜" Pressurizer Compartment	5.5	1.35	Will penetrate 3 inches into 3 foot thick pressurizer compartment concrete wall	RN 01-113
8	78 Pressurizer Heaters	N/A	438'-3" Pressurizer Compartment	15	2.4	Will penetrate 0.144 inches into concrete floor at elevation 437'-6"	

TABLE 3.5-4

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES INSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against [1]	Provisions for Protection
Reactor Building Liner	Reactor Building Wall	3.8.1	1,2,3,4,5,6	Primary and secondary shield walls and missile shield
Reactor Vessel Supports	Reactor Building	5.5.14	None	High strength of support material and strategic orientation
Steam Generator Supports	Reactor Building	5.5.14	None	High strength of support material and strategic orientation
Reactor Coolant Pump Supports	Reactor Building	5.5.14	None	High strength of support material and strategic orientation
Pressurizer Support	Reactor Building	5.5.14	None	High strength of support material and strategic orientation
Reactor Coolant System		5.0		
Reactor Vessel	Reactor Compartment		1,2,5,6	Primary shield wall and missile shield

TABLE 3.5-4 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES INSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against (1)	Provisions for Protection
Reactor Coolant Pumps	Inside Secondary Shield Wall		1,2	Physical separation and secondary shield wall
Steam Generators	Inside Secondary Shield Wall		1,2	Physical separation and secondary shield wall
Pressurizer	Inside Secondary Shield Wall		3,4	Secondary shield wall
Reactor Coolant Piping (2)	Reactor Building		1,2,3,4	Physical separation and secondary shield wall
Chemical and Volume Control System		9.3.4		
Regenerative Heat Exchanger	Reactor Building		None	Distance and strategic orientation
Excess Letdown Heat Exchanger	Reactor Building		None	Distance and strategic orientation
CVCS Piping(2)	Reactor Building		1,2	Physical separation and secondary shield wall

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TABLE 3.5-4 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES INSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against [1]	Provisions for Protection
Safety Injection System		6.3		
Accumulators	Reactor Building		1,2	Secondary shield wall
Safety Injection System Piping [2]	Reactor Building		1,2	Separation and secondary shield wall
Residual Heat Removal System Piping [2]	Reactor Building	5.5.7	1,2	Physical separation and secondary shield wall
Reactor Building Spray System Piping [2]	Reactor Building	6.2.3	1,2,3,4	Physical separation and secondary shield wall
Main Steam System Piping [2]	Reactor Building	10.3	1,3	Physical separation and secondary shield wall
Feedwater System Piping [2]	Reactor Building	10.4.7	1,2	Physical separation and secondary shield wall
Service Water System Piping [2]	Reactor Building	9.2.1	None	Physical separation
Reactor Building Cooling Units	Reactor Building	6.2.2	None	Secondary shield wall and strategic orientation

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TABLE 3.5-4 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES INSIDE THE REACTOR BUILDING TO BE PROTECTED

System, Component or Structure	Location	FSAR Section	Postulated Missiles to be Protected Against [(1)]	Provisions for Protection
Electrical Components		8.0		
Electrical Containment Penetrations	Reactor Building	6.2.4	1,2,3,4	Physical separation and secondary shield wall
Power, Control and Instrument Cable	Reactor Building	8.0	1,2,3,4	Physical separation and secondary shield wall
Cable Trays	Reactor Building	8.0	1,2,3,4	Physical separation and secondary shield wall
Post Accident Hydrogen Recombiners	Reactor Building	6.2.5	None	Physical separation and secondary shield wall

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(1) Missile numbers refer to missiles described in Table 3.5-3.

(2) Piping includes pipe, fittings and valves.

TABLE 3.5-5

POTENTIAL TORNADO MISSILES

	Tornado Missile	Geometric Properties [1]	Weight (lb)	Velocity (mph)
1.	Utility Pole [2]	L=35 ft, D=14 in	1880	150
2.	Compact Auto [2]	A=6.25 ft^2	2000	150
3.	Wood Plank	A=4 in x 12 in L=12 ft	108	300
4.	Passenger Auto [2]	A=20 ft^2	4000	50
5.	3 inch Schedule 40 Pipe, piece	A=9.62 in^2 L=10 ft	75.8	100
6.	Wood Pole, piece	D=8 in L=12 ft	209	225
7.	Steel Rod	D=1.0 in L=3.0 ft	8	216

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(1) L = length, D = diameter, A = minimum cross sectional area

(2) Impact point will not be more than 30 feet above grade

TABLE 3.5-6

SAFETY RELATED COMPONENTS LOCATED OUTDOORS

Component	Location Shown by Figure
Refueling Water Storage Tank	1.2-4, 1.2-5, 1.2-10
Reactor Makeup Water Storage Tank	1.2-4, 1.2-5
Sodium Hydroxide (NaOH) Storage Tank	1.2-4
Condensate Storage Tank (CST) (including Emergency Feedwater suction line at CST)	1.2-1
Emergency Feedwater Pump Recirculation Line (at CST)	1.2-1
Emergency Feedwater Turbine Exhaust Steam Stack	1.2-11,1.2-13,1.2-26
Diesel Generator Exhaust Stacks	1.2-13,1.2-14,1.2-26
Diesel Generator Day Tank Vents	1.2-12
Diesel Generator Fuel Oil Storage Tank Vents	1.2-1
Control Room Outside Air Intake Ductwork	1.2-26
Control Building Exhaust Air Relief Heads	1.2-16, 1.2-26
Chilled Water Expansion Tanks	1.2-16, 1.2-26
Main Steam System Power Operated Relief Valve Discharge Stacks*	1.2-5, 1.2-12, 1.2-13, 1.2-26

RN 07-032

* The main steam system power operated relief valve discharge stacks are not classified as safety related but provide the capability to allow for a controlled cooldown of the primary system.

TABLE 3.5-7

PHYSICAL PROPERTIES AND FUNCTION OF TORNADO MISSILE BARRIERS

Structure	Fig. No. & Grid Location	Thickness (ft) Wall	Thickness (ft) Roof	Concrete Strength	Components Protected Tag No	Components Protected Equipment
Control Complex	1.2-26 F-13, F-14	2.0	2.0	3000 psi in 28 days	Following equipment, related solenoids, and limit switches:	
					XDP-19A, 19B	Control Room Outside Air Intake Shutoff Damper
					XDP-21A, 21B	Control Room Relief Vent Control Damper
					XDP-22A, 22B	Control Room Return Duct Control Damper
					XDP-23A, 23B	Control Room Emergency Filter Shutoff Damper
					XDP-24A, 24B	Control Room Emergency Fan Discharge Shutoff Damper
					XDP-45	Controlled Access Outside Auxiliary Damper
					XDP-39A, 39B	Computer Room Outside Air Intake Control Damper
					XDP-96	Instrument Repair Room Outside Air Intake Control Damper
					XDP-105A, 105B	Control Room AHU Face and Bypass Control Damper
					XDP-106	Controlled Access Outside Air Intake Manual Isolation Damper

TABLE 3.5-7 (Continued)

PHYSICAL PROPERTIES AND FUNCTION OF TORNADO MISSILE BARRIERS

Structure	Fig. No. & Grid Location	Thickness (ft) Wall	Thickness (ft) Roof	Concrete Strength	Components Protected Tag No	Components Protected Equipment
Control Complex (Cont'd)					XDP-112A, 112B	Computer Room Outside Air Intake Control Damper
					XDP-113A, 113B	Relay Room Cooling System Return Air Damper
					XDP-129	Instrument Room Outside Air Damper
					XDP-133A, 133B, 133C, 133D	Control Complex Equipment Room General Area Relief Air Damper
					XDP-234A, 234B	Control Complex Equipment Room General Area Redundant Relief Air Damper
					XFN 30A, 30B	Control Room Emergency Filtering System Fan
					XFN 32A, 32B	Control Room Normal Supply Fan
					XFN 36A, 36B	Relay Room Cooling System Fan
					XAH 12A, 12B	Control Room Normal Supply Air Handling Unit
					XAH 113A, 113B	Relay Room Air Handling Unit
					XAH 29A, 29B	Control Room Emergency Filter Plenum
					XFL 56A, 56B	Relay Room Air Handling Unit Filter Box
					6490A, 6490B	Valve
					6412A, 6412B	Valve

TABLE 3.5-7 (Continued)

PHYSICAL PROPERTIES AND FUNCTION OF TORNADO MISSILE BARRIERS

Structure	Fig. No. & Grid Location	Thickness (ft) Wall	Thickness (ft) Roof	Concrete Strength	Components Protected Tag No	Components Protected Equipment
Control Complex (Cont'd)					-	Circuits and raceways north of column line F.2 and east of column line 13.9
					XTK-174A, 174B	Chilled Water (VU) Expansion Tanks
					LT-9004	Chilled Water (VU) Expansion Tank A Level Transmitter
					LT-9006	Chilled Water (VU) Expansion Tank B Level Transmitter
					ILT-9004-HR-VU	Chilled Water (VU) Expansion Tank A LT-9004 High Root Valve
					ILT-9006-HR-VU	Chilled Water (VU) Expansion Tank A LT-9006 High Root Valve
Diesel Generator Building	1.2-12 F-3, G-3	2.0		3000 psi in 28 days	-	Diesel Generator A and Auxiliaries
					-	Diesel Generator B and Auxiliaries
					XTK 20A, 20B	Diesel Generator Fuel Oil Day Tank
					XTK 9A, 9B, 9C, 9D	Diesel Generator Air Receiver
					XTK 114A, 114B	Diesel Generator Jacket Water Head Tank
					XPP 4A, 4B	Diesel Generator Fuel Oil Transfer Pump
					XCX 5201, 5202	Diesel Generator Control Cubicle
Fuel Handling Building	1.2-6 D-7	1.0	1.0	5000 psi in 90 days	-	Service Water Lines for Reactor Building Cooling Units
					-	Cable Trays for Hydrogen Recombiner Power Cables and for Reactor Building Cooling Unit Power Cables
					-	Pressure Transmitter PT-951

TABLE 3.5-7 (Continued)

PHYSICAL PROPERTIES AND FUNCTION OF TORNADO MISSILE BARRIERS

Structure	Fig. No. & Grid Location	Thickness (ft) Wall	Thickness (ft) Roof	Concrete Strength	Components Protected Tag No	Components Protected Equipment
Intermediate Building	1.2-12 C-6	2.0	2.0	3000 psi in 28 days	-	Main Steam Line C
					-	Emergency Feedwater Line C
					-	Leak Rate Piping at Penetration No. 216
					-	Pressure Transmitter No. 953
Intermediate Building	1.2-13 C-6	2.0		3000 psi in 28 days	-	Reactor Building Equipment Access Hatch
Service Water Pumphouse	1.2-24 F-7	2.0		3000 psi in 28 days	XPP 39A, 39B, 39C	Service Water Pump
					XRS 2A, 2B, 2C	Service Water Traveling Screen
					MPP 39A, 39B, 39C	Service Water Pump Motor
					1063, 1064, 4616	Cable Tray
					-	Small Service Water Piping
Service Water Pumphouse	1.2-24 G-11	2.0	2.0	3000 psi in 28 days	Following equipment, related solenoids, limit switches, and associated cabinets:	
					XDP 71A, 71B	Service Water Pumphouse Building Exhaust Air Control Damper
					XDP 73A, 73B	Service Water Pumphouse Building Recirculation Control Damper
					XDP 74A, 74B	Service Water Pumphouse Building Fan Inlet Isolation Damper
					XES 2003B	Service Water Pump Special Switch
					XFN 80A, 80B	Service Water Pumphouse Building Supply Fan

TABLE 3.5-7 (Continued)

PHYSICAL PROPERTIES AND FUNCTION OF TORNADO MISSILE BARRIERS

Structure	Fig. No. & Grid Location	Thickness (ft) Wall	Roof	Concrete Strength	Components Protected Tag No	Equipment
Service Water Pumphouse (Cont'd)					XSW1EB	7.2kV Switchgear, Bus 1EB
					XSW1EB1	Engineered Safety Features 480 Volt Unit Substation, Bus 1EB1
					XMC1EB1X	Engineered Safety Features Motor Control Center 1EB1X
					DPN1HB3	D-C Distribution Panels 1HB3
					1060, 1063	Cable Tray
					4610, 4615	Cable Tray
					4616, 4627	Cable Tray
					5223	Cable Tray

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

Protection of structures, systems, and components important to safety from the dynamic effects of piping failure is provided in accordance with the requirements of 10 CFR 50, Appendix A, General Criterion 4. Subsequent to the General Design Criterion 4 final rule change [16], postulated breaks in the reactor coolant loop piping, except for branch line connections, have been eliminated for V. C. Summer. The dynamic effects of the postulated breaks at six terminal ends in the cold, hot, and crossover legs, the steam generator inlet elbow, and the loop closure weld in the crossover leg were eliminated from the structural design basis by application of leak-before-break methodology, as presented in Reference [17] and updated by Reference [19]. Approval of the elimination 02-01 of the V. C. Summer reactor coolant loop piping breaks is given in the Summer Safety Evaluation Report, dated January 11, 1993 [18]. To provide the high margins of safety required by General Design Criterion 4, the nonmechanistic pipe rupture design basis is maintained for containment design and ECCS analysis, and the postulated pipe ruptures are retained for electrical and mechanical equipment environmental qualification.

This section describes the design bases and design measures employed to protect the containment, reactor coolant pressure boundary, and other essential systems and equipment from the effects of postulated pipe ruptures. This protection is provided inside and outside of containment and considers jet blowdown from postulated breaks, as well as pipe whip and reactive forces.

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS

3.6.1.1 Design Bases

3.6.1.1.1 Essential Systems Outside Containment

Essential piping systems required for postulated piping failures outside of containment are as follows:

1. Main Steam System up to and including containment isolation valves.

2. Feedwater System up to and including containment isolation valves.

3. Emergency Feedwater System.

4. Chemical and Volume Control System.

5. Residual Heat Removal System.

6. Safety Injection System.

7. Steam Generator Blowdown System up to and including containment isolation valves.

8. Service Water System.

9. Chilled Water System.

10. Reactor Makeup Water System.

11. Component Cooling Water System.

3.6.1.1.2 Essential Systems Inside Containment

Essential piping systems required for postulated piping failures inside containment are as follows:

1. Reactor Coolant System.

2. Main Steam System.

3. Feedwater System.

4. Chemical and Volume Control System.

5. Residual Heat Removal System.

6. Safety Injection System.

7. Steam Generator Blowdown System.

8. Service Water System.

9. Reactor Building Spray System.

10. Steam Generator Sampling System.

3.6.1.1.3 Criteria for Protection Against Postulated Pipe Breaks in Reactor Coolant System Piping

A loss of reactor coolant accident (LOCA) is assumed to occur for a Reactor Coolant System branch line break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3.6-1) on outgoing lines 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 does not result in an uncontrolled loss of reactor coolant if either of the two valves in the line close.

Accordingly, both of the automatic isolation valves must be suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint does not jeopardize the integrity and operability of the valves. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6-1), a LOCA 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 purposes of these criteria when the inside diameter is greater than 4 inches up to the largest connecting line. Rupture of these lines results in a rapid blowdown from the Reactor Coolant System and basic protection is provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the reactor coolant system are defined as "small" if they have an inside diameter equal to or less than 4 inches. This size is such that Emergency Core Cooling System analyses using realistic assumptions show that no fuel cladding damage is expected for a break area of up to 12.5 in^2 which corresponds to a 4 inch inside diameter piping.

Engineered safety features provide for core cooling and boration, reactor building temperature and pressure reduction, and activity confinement in the event of a LOCA or steam or feedwater line break accident. This ensures that the public is protected in accordance with 10 CFR 50.67 guidelines. These safety systems are designed to provide protection for a Reactor Coolant System pipe rupture of a size up to and including double ended severance of a Reactor Coolant System main loop. RN 12-034

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

1. Minimum performance capabilities of engineered safety features systems are not reduced to less than those required to protect against the postulated break.

2. Containment leak tightness is not decreased to less than the design value if the break leads to a loss of reactor coolant. The containment is here defined as the Reactor Building liner and penetrations, the steam generator shell, the steam generator steam side instrumentation connections and the steam, feedwater, blowdown, and steam generator drain pipes within the Reactor Building.

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

   a. A pipe break which does not directly result in a loss of reactor coolant will not cause a loss of reactor coolant or steam or feedwater line break, through consequential damage.

b. A Reactor Coolant System pipe break does not cause a Steam or Feedwater System pipe break and vice versa.

Criteria relative to large and small branch line breaks are as follows:

1. Large Branch Lines

   Large branch line piping is restrained to satisfy the following criteria:

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

   b. Propagation of the break in the affected loop is permitted to occur but does not exceed 20% of the area of line which initially ruptured. This criterion is voluntarily applied so as not to substantially increase the severity of the loss of coolant. Two exceptions to this criterion are permitted:

      (1) The postulated rupture of the 12 inch accumulator injection lines is permitted to result in rupture of the 6 inch safety injection lines.

      (2) A postulated rupture of reactor coolant loop 2, 6 inch cold leg injection line is permitted to result in a rupture of the 3 inch normal charging line. This results in an additional break area equal to 25% of the area of the line which initially ruptured.

   c. Where restraints on the lines are necessary to prevent impact on and subsequent damage to neighboring equipment or piping, restraint type and spacing are chosen such that either a plastic hinge on the pipe at the two support points closest to the break is not formed or, if a plastic hinge is formed, piping does not impact essential equipment.

2. Small Branch Lines

   In the unlikely event that one of the small pressurized lines should fail and result in a LOCA, the piping is restrained or arranged to satisfy the following criteria in addition to the criteria previously discussed for large branch lines:

   a. Break propagation is limited to the affected leg (i.e., propagation to the other leg of the affected loop and to other loops shall be prevented).

   b. Propagation of the break in the affected leg is permitted but is limited to a total break area of 12.5 in^2 (4 inch inside diameter). The exception to this criterion is when the initiating small break is in 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 other loops is prevented.

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

3.6.1.2 Description

3.6.1.2.1 High Energy Systems

High energy systems, systems with normal operating temperatures in excess of 200°F or normal operating pressures above 275 psig, are as follows: RN 01-113

1. Reactor Coolant System.

2. Main Steam System.

3. Feedwater System.

4. Emergency Feedwater System.

5. Chemical and Volume Control System.

6. Safety Injection System.

7. Steam Generator Blowdown System.

8. Auxiliary Steam System.

9. Main Steam Dump System to atmosphere, up to control valves.

10. Main Steam Drains System.

11. High energy systems with lines all 1 inch and smaller. No breaks are postulated in the following systems:

    a. Nuclear Sampling System.

    b. NSSS Carbon Dioxide and Nitrogen Supply Systems.

12. High energy systems remotely located in the Turbine Building and not considered for pipe rupture.

    a. Extraction Steam System.

    b. Condensate System.

    c. High Pressure Heater Drip, Vent and Relief System.

    d. Low Pressure Heater Drip, Vent and Relief System.

    e. Feedwater Pump Startup Drain System.

    f. Miscellaneous Steam Drain System.

    g. Auxiliary Boiler Chemical Feed System.

    h. Turbine Cycle Sampling System.

    i. Main Steam Dump System to condensers, up to control valves.

    j. Turbine Generator Electrohydraulic Fluid Control System.

3.6.1.2.2 Moderate Energy Systems

Moderate energy systems, systems with normal operating temperatures less than or equal to 200°F and normal operating pressures equal to or less than 275 psig, include all piping systems not listed in Section 3.6.1.2.1.

3.6.1.2.3 High Energy Systems Enclosed in Compartment to Protect Nearby Essential Systems and Components

Main steam and feedwater lines are enclosed in the penetration access areas at floor elevation 436' and on the upper level of the Intermediate Building to protect nearby essential systems and components.

The high and moderate energy lines outside of containment are enclosed in structures and compartments. The pressure buildups due to postulated pipe break within the enclosures are calculated to verify the adequacy of the structural design. The postulated break of the 32 inch main steam line produces the most adverse effects in the east and west penetration access areas, and Intermediate Building, while the 30 inch main steam line produces the most adverse effects in the Turbine Building. The FLASH-2 [1] and FLASH-4 [2] computer codes, which perform mass and energy balance calculations for specified control volumes and flow paths on a time step basis, were used to determine the blowdown flow rates and enthalpies for these ruptures. These results were then used as input for FLASH-4 [2] and CONTEMPT [4] calculations to determine the pressure responses of the compartment. In calculating these pressure responses, the following assumptions were made:

1. As mass and energy are introduced into a control volume, thermodynamic equilibrium is assumed throughout the node.

2. No heat transfer occurs between the accident environment and the surrounding structures and equipment.

3. Relief and vent areas are multiplied by a conservative 0.6 discharge coefficient.

4. The models used for subcompartment analysis in the Intermediate and Turbine Buildings are shown by Figures 3.6-1a through 3.6-1e and Tables 3.6-0 through 3.6-0g. Intermediate Building models were analyzed using the FLASH-4 [2] computer code. The Turbine Building was analyzed using the CONTEMPT [4] computer code to determine loadings on the Intermediate Building wall. Appropriate Tables and Figures are as follows:

    a. Intermediate Building - West Penetration Access Area

        (1) Nodalization model - Figure 3.6-1a, Table 3.6-0.

        (2) Mass/energy release data - Table 3.6-0b.

    b. Intermediate Building - East Penetration Access Area

        (1) Nodalization model - Figure 3.6-1b, Table 3.6-0a.

        (2) Mass/energy release data - Table 3.6-0b.

    c. Intermediate Building - Subcompartment Containing Main Steam and Feedwater Headers

        (1) Nodalization model - Figure 3.6-1c, Table 3.6-0c.

        (2) Mass/energy release data - Table 3.6-0e.

    d. Intermediate Building - Outside Subcompartment Containing Main Steam and Feedwater Headers

        (1) Nodalization model - Figure 3.6-1d, Table 3.6-0d.

        (2) Mass/energy release data - Table 3.6-0e.

e. Turbine Building - Loading on Intermediate Building Wall

(1) Nodalization model - Figure 3.6-1e, Table 3.6-0f.

(2) Mass/energy release data - Table 3.6-0g.

5. Initial conditions:

A steam-water mixture with a density equal to dry air at 120°F was used as the initial condition in FLASH-4 pressure response calculations. Initial conditions for CONTEMPT pressure response calculations are presented in Table 3.6-0f.

Since both the 30 inch and 32 inch main steam lines are at an elevation between 436' and 463' elevation slabs, each structure is analyzed assuming that the mass and energy released are introduced into the compartments between these two elevations. The results of these studies are shown by Figures 3.6-2 through 3.6-16. Table 3.6-0h presents a cross reference between the results shown by Figures 3.6-2 through 3.6-16 and the models listed in Item 4, above. The peak differential pressures, design and calculated, are listed in Table 3.6-0i. The analytical methods used to determine subcompartment pressure and temperature responses are the same for subcompartments inside or outside containment.

Where piping systems have been enclosed in compartments to protect essential systems in other compartments, the pressure versus time curves, Figures 3.6-2 through 3.6-16, are used to determine the peak pressure to be applied as loads to the compartments walls or slabs.

This is done by considering the pressure versus time as a forcing function and calculating the resistance function of the structural elements. From this a dynamic load factor is determined and applied to the peak differential pressure to arrive at an equivalent static load. The equivalent static load is applied to the structure in the load combination described in Section 3.8.4.

For postulated ruptures of high and moderate energy lines outside containment, there is the possibility that adverse environmental conditions might result. To ensure proper design of safety-related equipment to withstand such an environment, it is necessary to calculate the extent to which the environment would be effected by such ruptures. These calculations are performed using the FLASH-2 [1] and FLASH-4 [2] computer code results for the mass and energy releases from these ruptures as input to the MNODE [3] and CONTEMPT [4] computer codes. Where it is desirable to calculate the environmental response to a postulated rupture in a few interconnected volumes, the MNODE computer code is used. However, when a one-node study of the resulting environmental conditions is sufficient, the CONTEMPT program is used. Both of these programs give the pressure and temperature of each control volume directly. Humidity must be calculated. This is easily performed using the following equation:

$$\text{Relative humidity} = \frac{P_v}{P_g}$$

Where:

P_g = Saturation pressure corresponding to control volume temperature.

P_v = Partial pressure of vapor which:

a. For CONTEMPT results - can be taken directly from the output.

b. For MNODE results - calculated using the control volume vapor mass and temperature in the ideal gas law equation. RN 01-113

The environmental conditions of the various compartments of the penetration access areas and the Intermediate Building are studied to determine how they are affected by postulated ruptures of the 32 inch diameter main steam line, the 4 inch main steam line to the emergency feedwater pump turbine and the 3 inch steam generator blowdown line.

Results of these calculations for a postulated main steam line rupture in the penetration rooms and Intermediate Building are presented in Table 3.6-1.

Reactor Building temperature and pressure versus time curves used to determine the effect on the structure of accident temperature during a pipe break are discussed in Section 6.2.1.

No postulated rupture of a high energy line will have any adverse effect on the control room.

Environmental qualification of equipment is discussed in Section 3.11.

3.6.1.3 Safety Evaluation

Failures which could affect the ability to bring the plant to a safe shutdown condition are analyzed in Chapter 15. These analyses include consideration of the occurrence of a single active component failure in required systems concurrent with postulated pipe rupture except as noted below in Section 3.6.2.1.1.1, for APCSB 3-1, paragraphs B.3.b and B.3.d, for an environmentally-induced failure which would not of itself result in protective action. The pipe rupture analysis required for safe plant shutdown under the applicable criteria is rendered inoperable as a consequence of postulated pipe rupture.

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

This section discusses the following:

1. The design bases for determining the location of postulated cracks in piping inside and outside of containment.

2. Procedures used to define the jet thrust reaction at the break or crack location.

3. The jet impingement loadings on adjacent safety-related structures, systems, and components.

3.6.2.1 <u>Criteria Used to Define Break and Crack Location and Configuration</u>

3.6.2.1.1 High Energy System Piping Outside Containment

Breaks are postulated to occur in ASME Code, Section III, Class 2 and 3 piping and branch runs at the following locations:

1. At terminal ends. The terminal end for piping which penetrates containment is the pipe to penetration weld (see Figure 3.8-15) outside the Reactor Building.

2. At intermediate locations selected by either one of the following criteria:

    a. At each pipe fitting.

    b. At each location where the stresses exceed 0.8 ($1.2S_h + S_a$). Stresses are determined under the combination of loadings associated with the OBE and the nominal and upset plant condition loadings. RN 01-113

Specific non-nuclear safety class piping has been classified as Quality Related piping for the purpose of minimizing postulated pipe break locations. This Quality Related piping is designed in accordance with Code requirements. A rigorous analysis of this Quality Related piping is performed and breaks are then postulated based on the criteria noted above for Code Class piping. 02-01

Breaks in non-nuclear safety class piping are postulated to occur at the following locations in each piping or branch run:

1. At terminal ends.

2. At each intermediate pipe fitting, welded attachment, and valve.

Circumferential breaks are postulated to occur in fluid system piping and branch runs with nominal pipe size in excess of 1 inch.

Where a pipe elbow break location is selected without benefit of stress calculations, the pipe-to-elbow weld that joins the elbow to the shorter, straight, piping run is considered as the location of the break. Where break locations are selected in full size branch connection tees without benefit of stress calculations, the two pipe-to-tee welds that join the tee to the shorter, straight, piping runs are considered to be break locations. Where break locations are selected in reduced size branch connection tees without benefit of stress calculations, the pipe to tee weld that joins the tee to the shorter, straight, main piping run and the pipe to tee weld that joins the tee to the branch piping run are considered to be break locations.

Longitudinal breaks are postulated to occur in high energy system piping at the location of each postulated circumferential break except at the terminal ends under conditions discussed below:

1. Longitudinal breaks are not postulated to occur in high energy system piping and branch runs of nominal 3 inch pipe size and smaller.

2. Longitudinal breaks are postulated to occur in addition to, but not concurrently with, circumferential breaks.

3. Longitudinal breaks are not postulated to occur at terminal ends if the system piping at the terminal ends contains no longitudinal pipe welds.

4. Longitudinal breaks are assumed to result in an axial pipe split without pipe severance. Splits are oriented at two diametrically opposed points on the circumference of the pipe or fitting such that a jet reaction results that is normal to the plane formed by two of the applicable orthogonal axes, x, y, and z, of the piping configuration.

3.6.2.1.1.1 Conformance to Branch Technical Positions APCSB 3-1 [13] and MEB 3-1 [14]

An analysis has been performed which demonstrates that acceptable protection against the effects of piping failures outside containment has been provided. This analysis satisfies the intent of the guidelines of Branch Technical Positions (BTP) APCSB 3-1 and MEB 3-1. Since these positions were published a considerable period of time after the Virgil C. Summer Nuclear Station pipe rupture analysis had commenced, certain requirements could not be followed.

However, in certain respects, the design and analyses to cope with postulated pipe rupture for the Virgil C. Summer Nuclear Station are more stringent than required by the previously referenced BTPs. Specific differences are as follows:

1. APCSB 3-1:
   Paragraph B.3.b(1): Offsite power was assumed to be unavailable for all postulated piping failures.

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Criteria stated in the previously referenced BTPs with which the analysis does not fully comply and the alternative approaches are as follows:

1. APCSB 3-1:

   a. Paragraph B.2.c(1): The fluid system piping between containment isolation valves is not designed to the stress limits specified in Paragraphs B.1.b or B.2.b of BTP MEB 3-1. Breaks or cracks, as appropriate, are postulated in these portions of the fluid system piping in accordance with the criteria stated in Section 3.6.2.1.1.

   b. Paragraph B.2.d(2): For these portions of fluid system piping identified in Paragraph B.2.c, the inservice examination will be that required by the ASME Code, Section XI.

   c. Paragraph B.3.b and B.3.d: The effects of an environmentally - induced failure caused by a leak or rupture which would not of itself result in protective action may include a loss of redundancy in the protective function, but not a loss of the protective function, as permitted by BTP-APCSB 3-1, Appendix B, paragraph 11.b. [15]. In these cases, plant shutdown is required. The use of Appendix B in lieu of BTP-APCSB 3.1 is permitted by the implementation schedule of paragraph B.4.c, since the V. C. Summer construction permit is dated March 1973. Other criteria of BTP-APCSB 3.1 for single failure analyses are met, including the paragraph B.3 criteria for evaluating effects of cracks in moderate energy lines. RN 98-014 RN 01-113

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3.6.2.1.2 High Energy System Piping Inside Containment

As indicated in Section 3.6, dynamic effects resulting from postulated breaks in the reactor coolant loop piping (i.e., the six terminal ends in the cold, hot, and crossover legs, a split in the steam generator inlet elbow, and the loop closure weld in the crossover leg) were eliminated from the structural design basis for V. C. Summer. Reactor coolant loop branch line connection (i.e., accumulator connection, pressurizer surge line, residual heat removal, etc.) breaks are postulated. These postulated break locations and methods that are used to determine them are described in Reference [5].

Breaks are postulated to occur in ASME Code, Section III, Class 1, piping, other than piping discussed in Reference [5], at the following locations in each piping or branch run:

1. At the terminal ends.

2. At any intermediate location between terminal ends where the primary plus secondary stress intensities (circumferential or longitudinal) derived on an elastically calculated basis under loadings associated with specific seismic events and normal and upset operational plant conditions exceed $2.4S_m$.

3. At any intermediate location between terminal ends where the cumulative usage factor, U, derived from the piping fatigue analysis under the loadings associated with specified seismic events and normal and upset plant operational conditions exceeds 0.1.

Breaks are postulated to occur in ASME Code, Section III, Class 2 and 3 piping at the following locations in each piping or branch run:

1. At the terminal ends. The terminal end for piping which penetrates containment is the pipe to penetration weld (see Figure 3.8-15) inside the Reactor Building.

2. At intermediate locations selected by either one of the following criteria:

    a. Each pipe fitting.

    b. Any location where either the circumferential or longitudinal stresses, derived on an elastically calculated basis under loadings associated with specified seismic events and normal and upset operational plant conditions exceeds 0.8 ($1.2\ S_h + S_a$).

The following types of breaks are postulated to occur at locations previously identified for ASME Code, Section III, Class 1, 2, and 3 piping:

1. Circumferential breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

2. Longitudinal breaks in piping runs and branch runs of 4 inch nominal pipe size and large except as discussed in item 3, below.

3. Longitudinal breaks are not postulated to occur at terminal ends if system piping at the terminal ends dose not contain longitudinal pipe welds.

Where break locations are selected without benefit of stress calculations, breaks are postulated to occur at the piping welds to each fitting or valve.

Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are oriented at two diametrically opposed points on the circumference of the pipe or fitting such that a jet reaction results that is normal to the plane formed by two of the applicable orthogonal axes, x, y, and z, of the piping configuration.

3.6.2.1.3 (Section Deleted)

3.6.2.1.4 Moderate Energy System Piping

Leakage cracks are postulated to occur in the following moderate energy piping system locations where the maximum stress range exceeds 0.4 (1.2 S_h + S_a):

1. In system piping located within structures and compartments containing required systems and components. Cracks are postulated to occur individually at locations appropriate to the formation of a basis for provision of maximum required protection against spray and flooding and resultant hazard or environmental conditions.

2. In system piping and branch runs with nominal pipe size larger than 1 inch.

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

3.6.2.2 <u>Analytical Methods to Define Forcing Functions and Response Models</u>

3.6.2.2.1 Reactor Coolant Loop Piping Branch Line Connections

Following is a summary of the methods used to determine the dynamic response of the reactor coolant loop associated with postulated pipe breaks at the loop piping branch nozzle. Detailed descriptions of the methods are given in Reference [5].

1. Time Functions of Jet Thrust Force on Ruptured and Intact Loop Piping

   To determine the thrust and reactive force loads to be applied to the reactor coolant loop during the postulated LOCA resulting from a break at the branch line nozzle, it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the reactor coolant loops as a result of a postulated LOCA. These forces result from the transient flow and pressure histories in the Reactor Coolant System. The calculation is performed in two steps. The first step is calculation of the transient pressure, mass flow rates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, and calculates the time history of forces at appropriate locations in the reactor coolant loops.

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

The blowdown hydraulic analysis was required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the Reactor Coolant System. The MULTIFLEX computer code [6] was developed with a capability to provide this information. RN 09-022

The MULTIFLEX computer code performs a comprehensive space-time dependent analysis of a LOCA and is designed to treat all phases of the blowdown. The stages are as follows: RN 09-022

a. A subcooled stage where the rapidly changing pressure gradients in the subcooled fluid exert the influence upon the Reactor Coolant System internals and support structures. RN 09-022

b. A two phase depressurization stage.

c. The saturated stage.

The MULTIFLEX code employs a one dimensional analysis in which the entire Reactor Coolant System is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coastdown and cavitation, core and steam generator heat transfer, including the W-3 DNB correlation, in addition to the reactor kinetics are incorporated in the code. RN 09-022

The MULTIFLEX computer program [7] was developed to compute the transient (blowdown) hydraulic loads resulting from a LOCA. RN 09-022

The blowdown hydraulic loads on primary loop components are computed from the following equation which includes both static and dynamic effects:

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

Where:

F = Force, lb_f,

A = Aperture area, ft^2,

P = System pressure, lb_f/in^2,

$\dot{m}$ = Mass flow rate, lb_m/sec,

ρ = Density, lb_m/ft^3,

g = Gravitational constant = 32.174 $ft\text{-}lb_m/lb_f - sec^2$,

A_m = Mass flow area, ft^2

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

a. Blowdown hydraulic information.

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

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

2. Dynamic Analysis of the Reactor Coolant Loop Piping Equipment Supports and Pipe Whip Restraints.

The dynamic analysis of the reactor coolant loop piping for the LOCA loadings is described in Section 5.2.1.10.

3.6.2.2.2 Balance of Plant Piping

By using the transient response of a piping system to a postulated rupture, the resulting blowdown thrust and jet forces are calculated as functions of time. For the secondary systems this information is calculated using the FLASH-2 [1] and FLASH-4 [2] computer codes. In performing these calculations, the following assumptions are made for both circumferential and longitudinal ruptures:

1. For feedwater line ruptures, it is assumed that the break area takes 1 millisecond to be fully opened, while for the main steam lines, it is assumed that the rupture is instantaneous.

2. A discharge coefficient of 1.0 for the escaping fluid.

3. For Reactor Coolant System and Feedwater System piping ruptures, the plant is at 100% power.

4. For Main Steam System piping breaks, the plant is in the hot standby condition.

Using these results, the blowdown forces after the wave propagation period are calculated using the JIP computer code [8]. The techniques used by this program depend upon the fluid stagnation conditions present in the piping system. However, in general, the thrust is calculated from the following equation as presented by Moody [9],

$$\frac{T}{A_E} = P_E - P_\infty + \frac{G_E^2 V_{ME}}{g_c} \qquad (3.6\text{-}1)$$

Where:

T	=	Thrust,
A_E	=	Exit (break) area,
P_E	=	Exit pressure,
P_∞	=	Environmental pressure,
G_E	=	Exit mass flow rate per unit area,
V_{ME}	=	Exit momentum specific volume, defined below,
g_c	=	Gravitational constant

The momentum specific volume, V_M is defined by the following equation:

$$V_M = \left[XV_g + (1-X)KV_f\right]\left(X + \frac{1-X}{K}\right) \qquad (3.6\text{-}2)$$

$$K = (V_g/V_f)^R \qquad (3.6\text{-}3)$$

Where:

X = Quality,
V_g = Vapor specific volume,
V_f = Liquid specific volume,
K = Vapor to liquid velocity ratio,
R = Velocity ratio exponent

Evaluation of P_E, G_E, and V_{ME} is discussed below for the various stagnation conditions:

1. Cold Water (i.e., T < 212°F)

The thrust is calculated assuming the exit pressure, P_E, is equal to the environmental pressure, P_∞. The mass flow rate, G_E, is then calculated from the Bernoulli equation.

$$G_E = \left[\frac{2g_c(P_o - P_E)}{V_f}\right]^{1/2} \qquad (3.6\text{-}4)$$

Where:

V_f = Liquid specific volume at calculated exit pressure,
P_o = Stagnation pressure

The thrust is then calculated using Equation (3.6-1) above, assuming that V_{ME} is equal to V_f.

2. Subcooled Water

Thrust for subcooled water at stagnation conditions is calculated by the use of the Henry-Fauske model [10]. Solution of the transcendental expressions of the Henry-Fauske model gives predictions for the exit pressure, P_E, and mass flow rate, G_E. The quantity V_{ME} is calculated assuming that the exit quality is equal to the stagnation quality. The thrust is calculated using Equation (3.6-1) where a unity velocity ratio at the exit is assumed, in Equation (3.6-2).

3. Water-Steam Mixtures

Water-steam mixtures, or two phase flow, are handled by a combination of the Moody model [9,11] and the Henry-Fauske model. For stagnation qualities greater than 2%, the Moody model is used and the mass flow rate and exit conditions are calculated with the velocity ratio exponent, R, equal to 1/3. The thrust is then calculated using Equation (3.6-1). For stagnation quality less than or equal to 2% [12] the Henry-Fauske model is used and the thrust is calculated using a unity velocity ratio.

4. Steam

Saturated and superheated steam is analyzed as a perfect gas with k, the ratio of specific heats, nominally equal to 1.3 and the gas constant equal to 85.76 ft-lb_f/°F-lb_m. For frictionless flow, the thrust reduces to a theoretical maximum as shown in Equations (3.6-5) and (3.6-6) below:

$$\frac{T}{A_E} = \left[(1+k)\frac{(2)}{k+1}\frac{k}{k-1}\right]P_o - P_\infty \qquad (3.6\text{-}5)$$

$$\approx 1.26 P_o - P_\infty \qquad (3.6\text{-}6)$$

Where:

k = C_P/C_v = 1.3 for steam

Typical results of thrusts versus time for postulated main steam and feedwater breaks are shown by Figures 3.6-17 and 3.6-18.

In certain cases, for small lines instead of the above, the blowdown force is represented by a steady-state function equal to KpA.

Where:

K = Thrust coefficient,
p = System pressure prior to pipe break,
A = Pipe break area

The K values used are 1.26 for steam-saturated water and 2.0 for subcooled, nonflashing water.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

In addition to pipe restraints, barriers and layout are used to provide protection from pipe whip, blowdown jet, and reactive forces.

One of the barriers utilized for protection against pipe whip is the steam generator compartment wall which serves as a barrier between the reactor coolant loops and the Reactor Building liner. In addition, the refueling cavity walls and the steam generator compartment wall enclose each reactor coolant loop in a separate compartment, thereby preventing an accident, which may occur in any loop, from affecting another loop or the Reactor Building liner. The portion of the steam and feedwater lines within the containment are routed behind barriers which separate these lines from all reactor coolant piping. These barriers are designed to withstand loadings caused by jet forces and pipe whip impact forces.

Engineered safety features, except Emergency Core Cooling System lines which must circulate cooling water to the reactor vessel and engineered safety features instrument lines, are located outside the steam generator compartment wall. Emergency Core Cooling System lines which penetrate the steam generator compartment wall are routed around and outside the wall so that they penetrate the wall in the vicinity of the loop to which they connect.

Each individual postulated break is reviewed to determine which essential components and piping, if any, are in the projected path of either the pipe whip or the jet.

In each case where essential components are in the path of either the pipe whip or the jet from a postulated break, such components are moved, where practical, to unaffected locations. Components that cannot be relocated are protected from pipe whip by restraining the pipe in which a break is postulated to occur and from jet impingement by installation of shields.

In reviewing the mechanical aspects of these lines, it has been demonstrated, through tests performed by the nuclear steam supply system manufacturer [5], that lines hitting equal or larger size lines of the same schedule do not cause failure of the line being hit; e.g., a 1 inch line, should it fail, does not cause subsequent failure of a 1 inch or larger size line. The converse, however, is assumed to be probable; i.e., a 4 inch 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 inch or 2 inch lines.

Piping, that can be damaged and is determined to be in the projected path of a postulated whipping pipe and that cannot be relocated, is protected by restraint of the postulated whipping pipe. Piping in the projected path of a jet resulting from a postulated break is either rerouted or shielded. When rerouting or shielding is not practical, the piping system design specification identifies jet impingement loadings of safety-related piping, as well as the allowable stress limits and loading combination requirements.

The piping system is then analyzed using the equivalent static load method or, in some cases, a detailed time history method of analysis to determine the jet impingement effect upon the piping. The resulting loads are then combined and compared to the allowable stress limits as defined in the design specification. In the event that compliance with the specified stress limits cannot be demonstrated, shielding of the piping is provided.

3.6.2.3.1 Restraint Criteria

Where pipe restraints are employed they are designed using the principles of the equations of motion (dynamic analysis) or those of energy balance. Since the forces due to dead, live, seismic, and thermal loads are considered negligible or self relieving, only the dynamic effect of the whipping pipe is considered in the design.

The equation of motion methods (dynamic analysis) employ the use of the computer programs DYREC (S061), DYNAL (S085), and nonlinear numerical integration techniques. See Section 3.8.4.4 for a description of computer codes.

The program uses the thrust versus time data described previously. The pipe and the restraint are modeled as lump mass systems. Nodal masses and element spring properties are determined and gaps are input. Using direct numerical integration of the equations of motion, the dynamic response of the pipe and restraint are calculated at specific time points.

The following criteria were used for material properties:

1. Minimum yield strength of pipe steel is reduced in accordance with operating temperature.

2. Minimum tensile strength of the pipe material, as listed in the material specification, is used as the ultimate strength of the pipe. Refinement of assumed ultimate strength for changes due to operating temperatures would not result in a substansive change in pipe restraint design or reactions.

3. Ultimate tensile strain of both piping and restraint material is one half of guaranteed minimum percent elongation. Ultimate shear strain is equal to guaranteed percent elongation.

4. Minimum values of yield strength, ultimate strength, and modulus of elasticity for pipe are taken from the ASME Code. Values for restraint material are taken from the applicable ASTM specification.

5. A 10% increase in material properties is applied to allow for strain rate effect.

Acceptability of the restraint design was based upon the results of the dynamic analysis. Neither the pipe nor the restraint stresses and strains exceed the following limits:

1. Tensile strains are limited to 50% of the assumed ultimate tensile strain. This is equal to 0.25 times percent elongation.

2. Bending and axial tensile stresses are limited to the values at the above strain limit as determined appropriate from the stress/strain, moment/curvature, or P/Δ curves.

3. Shearing strains are limited to 50% of the assumed ultimate shear strain. This is equal to 0.5 times percent elongation.

4. Shearing stresses are limited to the value at the above strain limit as determined from the shear/shear strain curve.

Restraints, connections, anchorages and the supporting structure are designed for the maximum reactions obtained from the dynamic analysis.

In the energy balance method, an amplification factor of 1.2 is applied to the peak thrust force to determine the force on the pipe.

Initial gap and the kinetic energy of the pipe are considered in balancing the internal work with the external.

Pipe and restraint properties as described for the dynamic analysis, are used with the exception that no strain rate effect is assumed.

Acceptable restraints design limits are the same as for the dynamic analysis.

Restraint connections, anchorages and the supporting structure are designed for 15% more than the maximum reactions from the energy balance analysis.

3.6.2.3.2 Jet Impingement

1. Balance of Plant Piping

   Blowdown forces obtained as described in Section 3.6.2.2 are used in the jet impingement analysis. It is assumed that the total jet impingement force is equal to the thrust calculated at the break [9] and is uniformly distributed across the cross sectional area of the jet at any particular location.

   For both circumferential and longitudinal ruptures, the configuration of the break areas is assumed to be circular in nature. These break areas, i.e., for the longitudinal rupture and for each end of the circumferential rupture, are assumed to be equal to the flow area of the pipe in the vicinity of the postulated rupture [12].

Calculation of the jet expansion profile is accomplished assuming a constant 10 degree half angle of expansion for the escaping fluid. Due to these assumptions, the area of the jet as a function of distance from the break point can be expressed by the following equation:

$$A_j(x) = A_E\left(1 + \frac{2x}{D_E \tan 10^\circ}\right)^2 \qquad (3.6\text{-}7)$$

Where:

$A_j(x)$ = The jet area as a function of distance from the break,
x = The distance from the break,
A_E = Exit or break area,
D_E = Equivalent diameter of break area (equals the pipe diameter in these cases)

In some instances where the postulated break is very close to structures, a more detailed blowdown analysis may be performed to evaluate jet impact forces. This detailed analysis may prove beneficial if the fluid properties support the jet expansion profile as presented by Moody [9]. In such cases, calculation of the jet profile may be performed by the computer code JIP [8], which evaluates the fluid properties and determined jet profiles and pressures in accordance with the following expansion model.

The Moody expansion model estimates fully expanded, one-dimensional asymptotic jet properties for jet expansion calculations. Diffusion, friction, turbulent momentum, and energy exchange, and heat transfer effects with the environment are not considered. Current usage of this model conservatively assumes that the expansion takes place over the first five equivalent diameters from the break [12]. The area ratio existing at five diameters can be calculated using the following equation [9]:

$$\frac{A_\infty}{A_E} = \frac{{G_E}^2 \ W_{M\infty}}{g_c \ (T / A_E)} \qquad (3.6\text{-}8)$$

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Where:

A_∞ = Area at 5 diameters,
A_E = Exit or break areas,
G_E = Exit mass flow rate per unit area,
g_c = Gravitational constant,
$V_{M\infty}$ = Environmental momentum specific volume, discussed below,
T = Total thrust at the break

The environmental momentum specific volume, $V_{M\infty}$, is defined by Equation (3.6-2) of Section 3.6.2.2. However, the velocity ratio used in evaluating $V_{M\infty}$ is different for jet impingement than for thrust. As concluded in Reference [9], a unity velocity ratio predicts the two-phase jet force better than a velocity ratio, $(V_g/V_f)^{1/3}$, as used in the Moody thrust evaluation. Therefore, a unity velocity ratio is used for determination of the Moody area ratio at five diameters. For calculation of areas before the full expansion at five diameters is reached, an additional assumption is made, i.e., that the jet area increases uniformally from zero to five diameters [12] . After the Moody area ratio is determined and the break geometry specified, the equivalent Moody angle can be determined.

The jet expansion profile calculations used by JIP [8] are discussed for cold water, subcooled water, and water-steam and steam conditions in the following paragraphs:

a. Cold and Subcooled water

Subcooled water is treated with a 10 degree half angle and a Moody expansion angle, if applicable. If the subcooled thrust is calculated from the Bernoulli equation, as discussed in Section 3.6.2.2, a jet expansion half angle of 10 degrees is used. However, if the thrust is calculated using the Henry-Fauske theory [10], as explained in Section 3.6.2.2, this uniform expansion is not always used. Henry-Fauske gives predictions for exit pressure and mass flow rate. In addition, an equilibrium exit quality is calculated. If the equilibrium exit quality is greater than or equal to 0.1% (approximately equal to a volume breakdown of 10% steam and 90% saturated water at the exit), the Moody expansion model, as described earlier, is used. Subsequent to jet expansion out to five diameters, the area is held constant until the 10 degree half angle is applicable. The overall jet geometry combining the Moody expansion profile with the 10 degree half angle is shown in Figure 3.6-19. If the calculated exit quality is less than 0.1%, the uniform 10 degree half angle expansion is used. the application of this jet profile is limited, however, since the pressure will rapidly drop to the saturation value.

b. Water-Steam and Steam

For water-steam and steam conditions, the Moody expansion model is directly applicable. Therefore, the jet geometry for water-steam, saturated steam, and superheated steam is depicted in Figure 3.6-19.

NUREG/CR-2913 [20] contains a model that has been developed for predicting two phase, water jet loadings on targets. The model was developed using advanced two dimensional computational techniques to solve the governing equations of mass, momentum, and energy. The application of this model results in a series of charts of the target load and pressure distributions. These charts were developed for a wide range of pressure and temperatures that completely cover the range of interest in pressurizer water reactors. In lieu of the methodology described previously, the charts contained in NUREG/CR-2913 are used to determine the effective jet pressure at varying distances from the jet origination.

RN 03-022

2. Reactor Coolant Loop Piping

The methods described below are used in the design and verification of the adequacy of reactor coolant loop components and supports.

The design basis postulated pipe rupture locations for the main reactor coolant loop piping are determined using the criteria given in Section 3.6.2.1.2 and Reference [5]. These design basis ruptures are used as the rupture locations for consideration of jet impingement effects on primary equipment and supports.

The dynamic analysis as discussed in Section 5.2, is used to determine maximum piping displacements at each design basis rupture location. These maximum piping displacements are used to compute the effective rupture flow area at each location. These areas and rupture orientations are then used to determine the jet flow patterns and to identify any primary components and supports which are potential targets for jet impingement.

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

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

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

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

$F_j = K_o P_o A_{mB} RS$

Where:

F_j = jet impingement load on target,
K_o = dimensionless jet thrust coefficient based on initial fluid conditions in broken loop,
P_o = initial system pressure,
A_{mB} = calculated maximum break flow area,
R = fraction of jet intercepted by target,
S = target shape factor

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

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

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

$F_j = KPA_B RS$

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

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipes are used as shields as described in sections 3.6.2.1.4 and 3.6.2.5.1 and are designed consistent with the moderate energy piping postulated to crack. 99-01

3.6.2.5 Material to be Submitted for the Operating License Review

3.6.2.5.1 Location and Orientation of Design Basis Breaks

The locations and orientations of the postulated design basis breaks are determined by stress analysis. Break locations and orientations for postulated reactor coolant piping branch line connection breaks are discussed below. Break locations for all other high energy system piping are postulated at terminal ends according to Sections 3.6.2.1.1 and 3.6.2.1.2. Break orientations for lines outside containment are discussed in Sections 3.6.2.1.1 and 3.6.2.1.2.

Cracks are postulated in moderate energy system piping in the vicinity of essential components. Components located in the vicinity of moderate energy system piping were relocated, if practical, or were specified as spray proof. In instances where neither relocation nor specification as spray proof is possible, shields are provided.

Due to the elimination of postulated breaks in the reactor coolant loop piping, as described in Section 3.6, jets are no longer considered in the analysis of the reactor coolant loop piping supports.

Table 3.6-2 and Figure 3.6-51 identify the design basis break locations and orientations for the main reactor coolant loop.

3.6.2.5.2 Restraint and Shield Location and Design Information

The E-303-300 series drawings includes a summary of the protection of essential equipment, located outside containment, from pipe whip and jet impingement resulting from postulated rupture of high energy system piping.

Reference should be made to E-303-300 series drawings for protection of essential equipment for the plant. Pipe whip restraints associated with the reactor coolant loop are discussed in Section 5.5.14.

3.6.2.5.3 Analytical Results

The analytical results show that no essential system or component is rendered incapable of performing its necessary functions as a result of any postulated pipe rupture.

The methods and analysis procedures used to determine jet impingement loads associated with the rupture of the main reactor coolant loop are discussed in Section 3.6.2.3.2.

3.6.2.5.4 Interface Responsibilities

The interface responsibility between Westinghouse and the Balance of Plant Supplier for the design of the component supports and the reactor coolant pressure boundary, which is not part of the reactor coolant loop is as follows:

Westinghouse

1. Design and evaluation of all primary equipment supports,

2. Design of the pressurizer surge line piping and fittings,

3. Analysis of all Class 1 branch lines attached to the reactor coolant loop piping (except for the pressurizer spray line) for deadweight, pressure, thermal, seismic, and effects of pipe rupture in the reactor coolant loop piping,

4. Design and analysis of All Class 1 reactor coolant loop branch line nozzles, including the pressurizer spray nozzle,

5. Provide the Balance of Plant supplier with loop displacements due to all loading conditions at branch line nozzle locations.

Balance of Plant Supplier

1. Design of all Class 1 branch lines except for pressurizer surge line,

2. Analysis of all Class 1 branch lines not within Westinghouse's scope for deadweight, pressure, thermal, seismic, and effects of pipe rupture in the reactor coolant loop piping,

3. Evaluation of all Class 1 branch lines for effects of pipe rupture in auxiliary Class 1 branch lines,

4. Supply Westinghouse with loads at the reactor coolant loop and line branch nozzles.

3.6.3 REFERENCES

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2. Porsching, T. A., et al., "FLASH-4: A Fully Implicit Fortran-IV Program for the Digital Simulation of Transients in a Reactor Plant," WAPD-TM-840, March 1969.

3. Huang, Y. S., Kowal, G. M., and Savery, C. W., "Subcompartment Analysis of High-Energy Pipe Ruptures," Nuclear Technology, Vol. 27, November 1975.

4. Richardson, L. C., et al., "CONTEMPT A Computer Program for Predicting the Containment Pressure-Temperature Response to a Loss of Coolant Accident," IDO-17220, June 1976.

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

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7. "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252 Rev. 1 (Non-Proprietary), May 1977.

8. Webb, S. W., Gilbert Associates, Inc., "JIP Thrust and Jet Impingement Analysis Code," Revision 2, December 1975.

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10. Henry, R. E. and Fauske, H. K., "The Two-Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices, and Short Tubes," ASME Paper 70-WA/HT-5, May 1971.

11. Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," ASME Paper 64-HT-35, February 1965.

12. "Design for Pipe Break Effects," Bechtel Report BN-TOP-2, Revision 2, May 1974.

13. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to USNRC Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," November 24, 1975.

14. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," Revision 2, June 1987, attached to Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," June 19, 1987. RN 02-016

15. Letter of A. Giambusso, USNRC, to Applicants and Licensees, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," December 12, 1972; attached, with Errata of January 1973, as Appendix B to BTP-APCSB 3-1 (Reference [13], above).

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

17. Schmertz, J.C., Swamy, S. A., and Lee Y. S., "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Virgil C. Summer Nuclear Power Plant," Westinghouse Report WCAP-13206, April 1992.

18. Nuclear Regulatory Commission Safety Evaluation Report, "Request to Use Leak-Before-Break Technology for Reactor Coolant System Piping - Virgil C. Summer Nuclear Station, Unit No. 1 (TAC No. M83971)," January 11, 1993.

19. D. C. Bhowmick, J. F. Petsche, and S. A. Swamy, "Technical Justification For Eliminating Large Primary Loop Pipe Rupture As The Structural Design Basis For The Virgil C. Summer Nuclear Power Plant," Westinghouse Report WCAP-13206, Revision 1, June 2000. 02-01

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TABLE 3.6-0

INTERMEDIATE BUILDING - WEST PENETRATION ACCESS AREA NODALIZATION MODEL

Control Volume Number	Control Volume Identification	Volume (ft^3)	Area (ft^2)	Height (ft)	Bottom (ft)
1	Intermediate Building	134,000	5570	24	436
2	Intermediate Building	50,000	2070	24	436
3	Intermediate Building	168,000	7650	22	412
4	Environment	1. + 8	1. + 7	10	458
5	East Penetration Access Area	72,200	2887	25	436
6	West Penetration Access Area	63,500	2900	22	412

Junction Number	From C.V.	To C.V.	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Friction (K)
1	1	2	448	550	.013	1.5
2	1	3	448	525	.0074	1.5
3	1	4	448	350	.00787	1.5
4	2	4	448	350	.0079	1.5
5	2	4	448	350	.0079	1.5
6	2	4	448	350	.0079	1.5
7	2	4	448	350	.0079	1.5
8	5	1	448	12.5	.25	1.5
9	5	1	448	335	.096	1.5
10	5	6	448	50	.048	1.5

TABLE 3.6-0a

INTERMEDIATE BUILDING - EAST PENETRATION ACCESS AREA
NODALIZATION MODEL

Control Volume Number	Control Volume Identification	Volume (ft^3)	Area (ft^3)	Height (ft)	Bottom (ft)	
						02-01
1	Intermediate Building	134,000	5570	24	436	
2	Intermediate Building	50,000	2070	24	436	02-01
3	Intermediate Building	168,000	7650	22	412	
4	Environment	1. + 8	1. + 7	10	458	
5	East Penetration Access Area	63,400	2643	24	436	02-01
6	East Penetration Access Area	58,100	2643	22	412	

Junction Number	From C.V.	To C.V.	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Friction (K)
1	1	2	448	550	.013	1.5
2	1	3	448	525	.0074	1.5
3	1	4	448	350	.00787	1.5
4	2	4	448	350	.0079	1.5
5	2	4	448	350	.0079	1.5
6	2	4	448	350	.0079	1.5
7	2	4	448	350	.0079	1.5
8	5	1	448	12	.23	1.5
9	5	1	448	410	.067	1.5
10	5	6	448	45	.05	1.5
11	6	3	430	11	.24	1.5

TABLE 3.6-0b

INTERMEDIATE BUILDING - PENETRATION ACCESS AREAS
MASS/ENERGY RELEASE DATA

Time (sec)	Mass Release Rate (lbm/sec)	Enthalpy (Btu/lbm)
0.0	2.202 + 4	1186
.00375	1.586 + 4	1198
.0113	1.079 + 4	1203
.0238	9.592 + 3	1204
.0463	1.184 + 4	1202
.108	9.358 + 3	1204
.413	7.009 + 3	1203
.875	8.312 + 3	1203
1.95	7.838 + 3	1203

TABLE 3.6-0c

INTERMEDIATE BUILDING - SUBCOMPARTMENT CONTAINING
MAIN STEAM AND FEEDWATER HEADERS
NODALIZATION MODEL

Control Volume Number	Control Volume Identification	Volume (ft^3)	Area (ft^2)	Height (ft)	Bottom (ft)	
						02-01
1	Intermediate Building	134,000	5570	24	436	
2	Intermediate Building	50,000	2070	24	436	02-01
3	Intermediate Building	168,000	7650	22	412	
4	Environment	1. + 8	1. + 7	10	458	

Junction Number	From C.V.	To C.V.	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Friction (K)	
							02-01
1	1	2	448	550	.013	1.5	
2	1	3	448	525	.0074	1.5	
3	1	4	448	350	.00787	1.5	
4	2	4	448	372	.0079	1.5	

TABLE 3.6-0d

INTERMEDIATE BUILDING - OUTSIDE SUBCOMPARTMENT CONTAINING MAIN STEAM AND FEEDWATER HEADERS NODALIZATION MODEL

Control Volume Number	Control Volume Identification	Volume (ft^3)	Area (ft^2)	Height (ft)	Bottom (ft)	02-01
1	Intermediate Building	134,000	5570	24	436	
2	Intermediate Building	60,500	2520	24	436	
3	Intermediate Building	168,000	7650	22	412	
4	Environment	1. + 8	1. + 7	10	458	

Junction Number	From C.V.	To C.V.	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Friction (K)	02-01
1	1	2	448	580	.0107	1.5	
2	1	3	448	525	.0074	1.5	
3	1	4	448	350	.00787	1.5	
4	2	4	448	1372	.0060	1.5	

TABLE 3.6-0e

INTERMEDIATE BUILDING - MAIN STEAM AND FEEDWATER HEADERS
INSIDE AND OUTSIDE HEADER SUBCOMPARTMENT
MASS/ENERGY RELEASE DATA

Time (sec)	Mass Release Rate (lbm/sec)	Enthalpy (Btu/lbm)
0.0	2.202 + 4	1186
.00625	1.900 + 4	1193
.0163	1.538 + 4	1198
.0313	1.354 + 4	1200
.0413	1.409 + 4	1199
.0563	1.555 + 4	1197
.188	1.404 + 4	1198
.363	1.248 + 4	1200
.488	1.151 + 4	1201
.725	1.034 + 4	1201
.975	9.880 + 3	1201
1.43	9.232 + 3	1202

TABLE 3.6-0f

TURBINE BUILDING NODALIZATION MODEL

Volume, ft^3	3×10^6
Initial Temperature, °F	80
Initial Relative Humidity, %	60
Relief Area, ft^2	570
Discharge Coefficient	0.6

TABLE 3.6-0g

TURBINE BUILDING
MASS/ENERGY RELEASE DATA

Time (sec)	Mass Release Rate (lbm/sec)	Enthalpy (Btu/lbm)
0.0	1.65 + 4	1197
.005	1.65 + 4	1197
.04	1.06 + 4	1203
.1	9.71 + 3	1204
.2	9.09 + 3	1204
.4	8.32 + 3	1203
.6	7.52 + 3	1203
.8	7.15 + 3	1203
1.	6.87 + 3	1203
1.6	6.55 + 3	1203
2.1	6.36 + 3	1203
3.1	1.04 + 4	717
5.1	1.57 + 4	559
7.1	1.63 + 4	574
10.1	1.57 + 4	594
15.0	8.59 + 3	623
20.0	5.36 + 3	675

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Amendment 02-01

TABLE 3.6-0h

INTERMEDIATE AND TURBINE BUILDING SUBCOMPARTMENT PRESSURE RESPONSE CROSS REFERENCE

Analysis	Applicable Control Volumes	FSAR Figure
West Penetration Access Area	5; 6	3.6-2
	5-6	3.6-3
	5-1; 5-3	3.6-4
	1-6; 3-6	3.6-5
East Penetration Access Area	5; 6	3.6-6
	5-6	3.6-7
	5-1; 5-3	3.6-8
	1-6; 3-6	3.6-9
Intermediate Building (Inside compartment containing main steam and feedwater headers)	2; 1 2-1 2-3; 2-1	3.6-10 3.6-13 3.6-15
Intermediate Building (Outside compartment containing main steam and feedwater headers)	1 1-3 1-4; 1-2	3.6-11 02-01 3.6-12 3.6-14
Turbine Building (Loading on Intermediate Building wall)	1	3.6-16

TABLE 3.6-0i

INTERMEDIATE AND TURBINE BUILDING DIFFERENTIAL PRESSURES

Subcompartment	Calculated Pressure (psi)	Design Pressure (psi) (Including Dynamics Effect)
West Penetration Access Area	5.0	6.5
East Penetration Access Area	4.8	5.9
Intermediate Building (Inside compartment main steam and feedwater headers)	3.0	4.0
Intermediate Building (Outside compartment containing main steam and feedwater headers)	2.2	4.0
Turbine Building (Loading on intermediate building wall)	2.3	4.0

TABLE 3.6-1

ENVIRONMENTAL CONDITIONS RESULTING FROM POSTULATED PIPE RUPTURES OUTSIDE OF CONTAINMENT

Large Line Break Conditions [(1)]

Area	Temperature (°F)	Relative Humidity (%)
East and West Penetration Access Areas, Floor Elevation 412’	220	100
East and West Penetration Access Areas, Floor Elevation 436’	220 [(2)]	100
West Penetration Access Area, Floor Elevation 463’	220	100
Intermediate Building, Floor Elevation 412’	212	100
Intermediate Building, Floor Elevation 436’	212 [(3)]	100

Small Line Break Conditions [(4)]

Area	Temperature (°F)	Relative Humidity (%)
East and West Penetration Access Areas and Intermediate Building, All Elevations	200	100

(1) Conditions last for 3 minutes, then return to 100°F in about 30 minutes.

(2) Temperature of 320°F for 30 seconds.

(3) Temperature of 283°F for 4 seconds.

(4) Conditions last for 3 hours, then return to 100°F in about 30 minutes.

NOTE: Details of the main steam line break HELB/SBOC (High Energy Line Break / Superheated Blowdown Outside Containment) environmental conditions are provided in Section 3.11.2.2.2.2. 99-01

TABLE 3.6-2

POSTULATED BREAK LOCATIONS FOR THE LOCA ANALYSIS OF THE PRIMARY COOLANT LOOP (1) (3)

	Location of Postulated Rupture	Type	Break Opening Area (2)
1.	Residual Heat Removal (RHR) Line/Primary Coolant Loop Connection	Guillotine (viewed from the RHR line)	Cross-sectional flow area of the RHR line
2.	Accumulator Line/Primary Coolant Loop Connection	Guillotine (viewed from the accumulator line)	Cross-sectional flow area of the accumulator line
3.	Pressurizer Surge Line/Primary Coolant Loop Connection	Guillotine (viewed from the pressurizer surge line)	Cross-sectional flow area of the pressurizer surge line

02-01

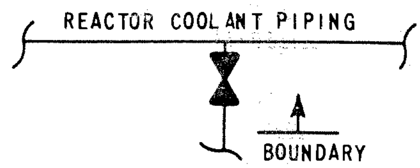
(1) Refer to Figure 3.6-51 for location of postulated breaks in reactor coolant loop.

(2) Less break opening area will be used if justified by analysis, experiments or considerations of physical restraints such as concrete walls or structural steel.

(3) Elimination of the dynamic effects of postulated pipe ruptures in the reactor coolant loop piping at the terminal ends, steam generator inlet elbows, and crossover leg closure welds have been eliminated as allowed by the revised General Design Criterion 4 (Section 3.6).

CASE I

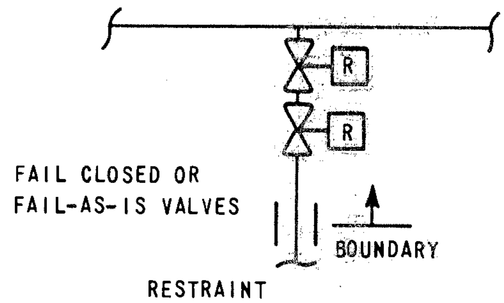
OUTGOING LINES WITH NORMALLY CLOSED VALVE



NOTE: PRESSURIZER SAFETY VALVES ARE INCLUDED UNDER THIS CASE.

CASE II

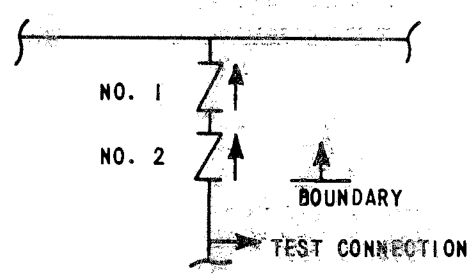
OUTGOING LINES WITH NORMALLY OPEN VALVES



NOTE: THE REACTOR COOLANT PUMP NO. 1 SEAL IS ASSUMED TO BE EQUIVALENT TO FIRST VALVE

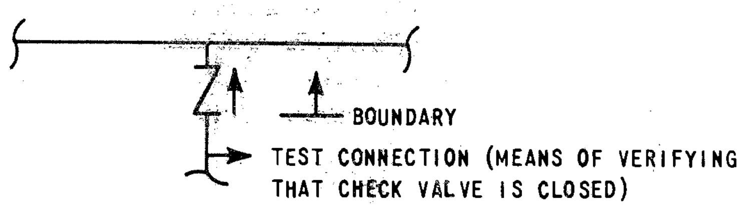
CASE III

INCOMING LINES NORMALLY WITH FLOW



CASE IV

INCOMING LINES NORMALLY WITHOUT FLOW



CASE V

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

SOUTH CAROLINA ELECTRIC & GAS CO.
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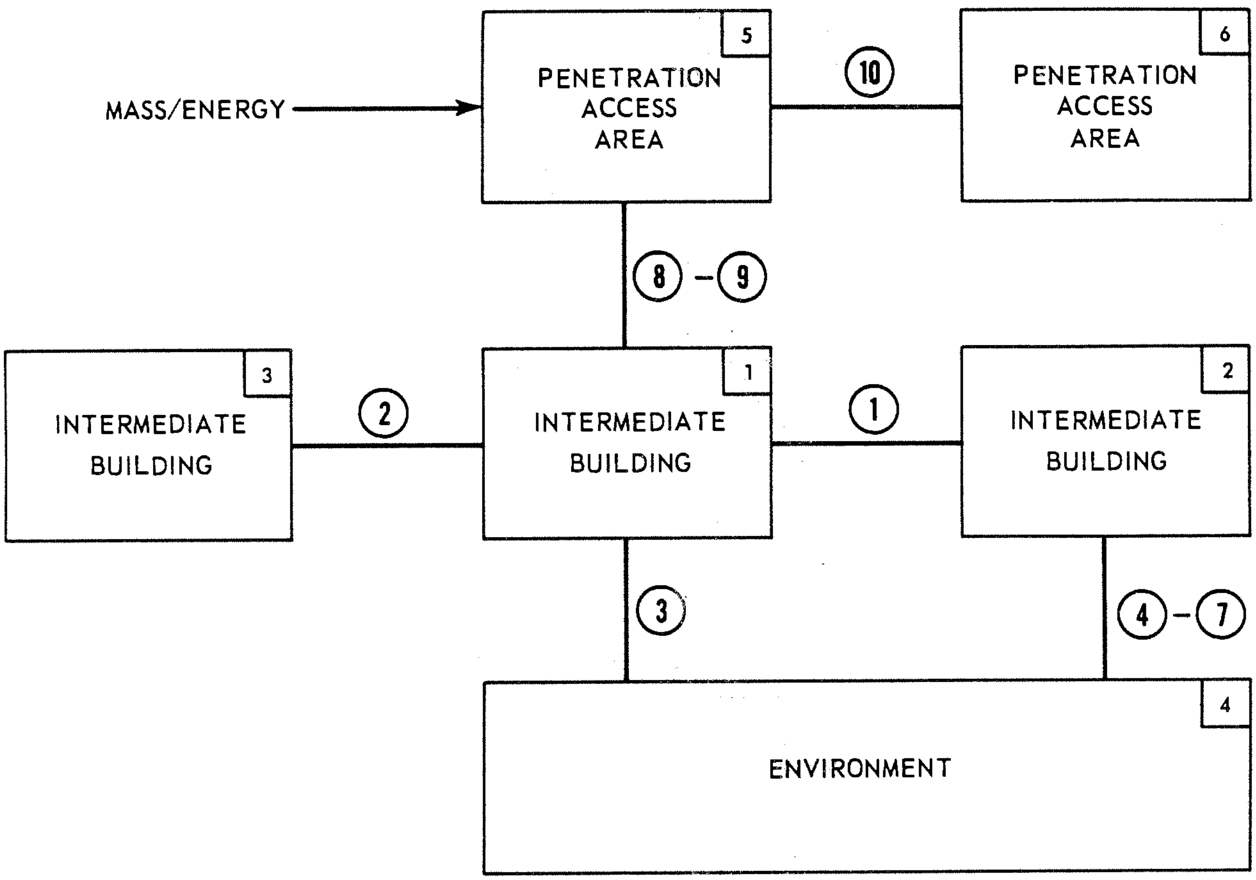
Loss of Reactor Coolant Accident
Boundary Limits

Figure 3.6-1



LEGEND:

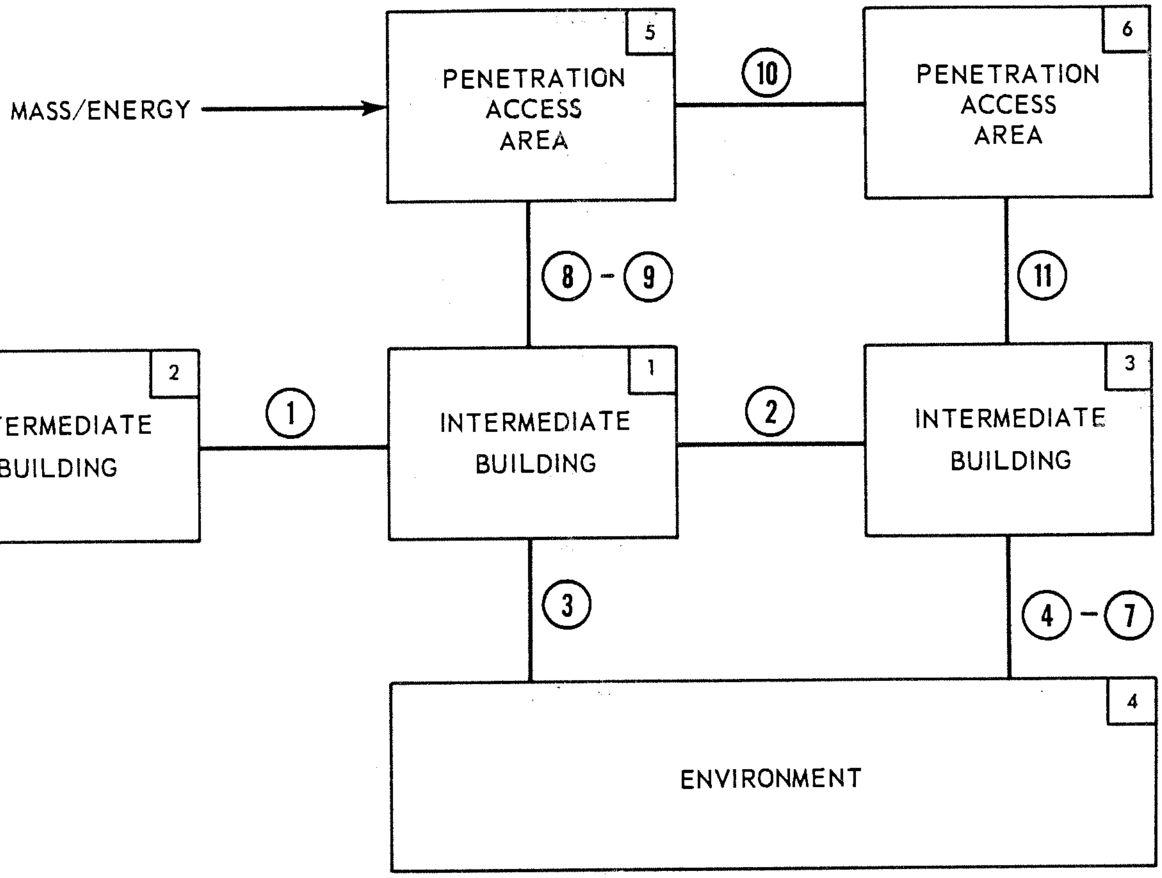
N – CONTROL VOLUME NUMBER

M – JUNCTION NUMBER

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Intermediate Building - West Penetration Access Area - Nodalization Model

Figure 3.6-1a

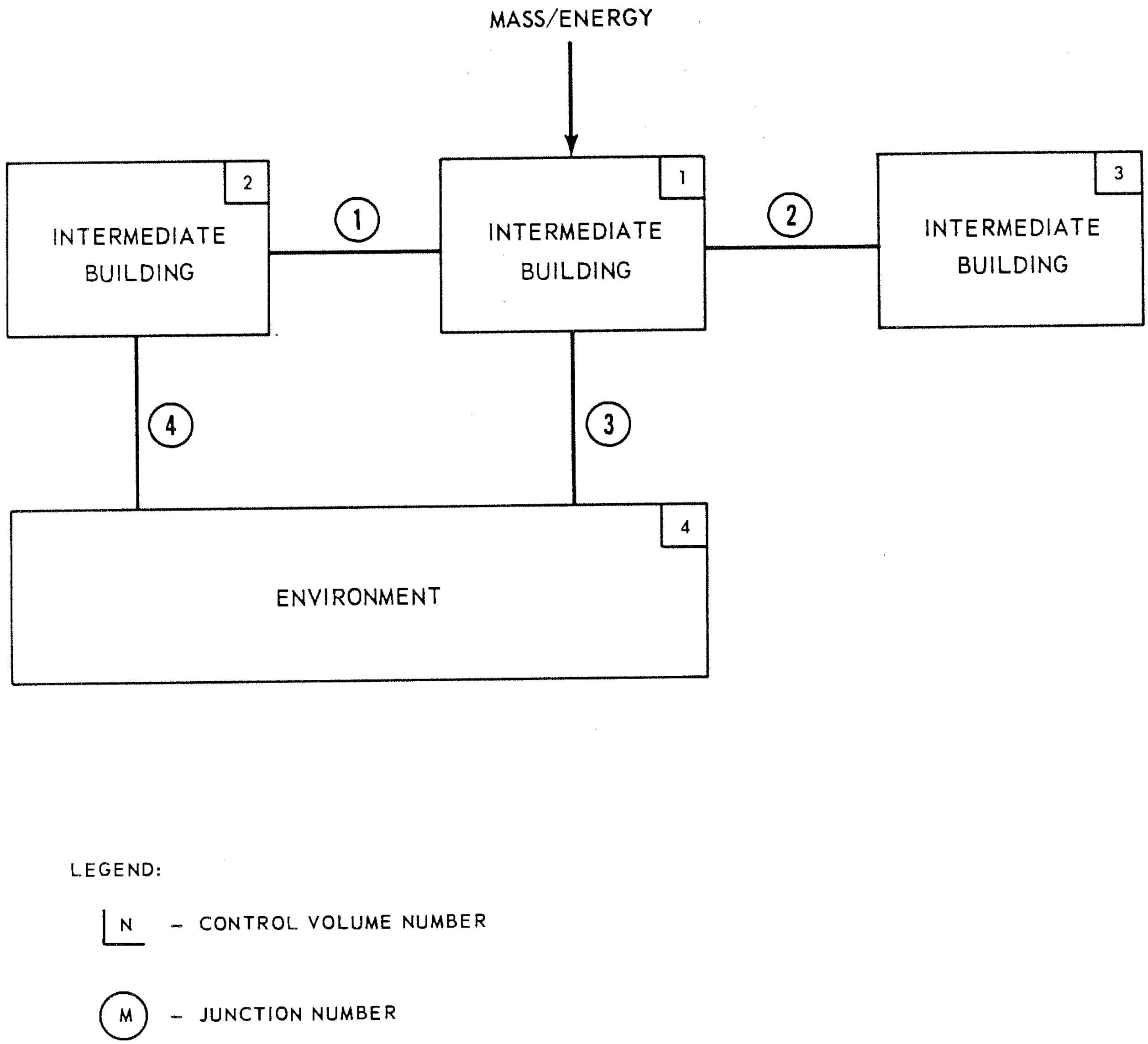


LEGEND:

N – CONTROL VOLUME NUMBER

M – JUNCTION NUMBER

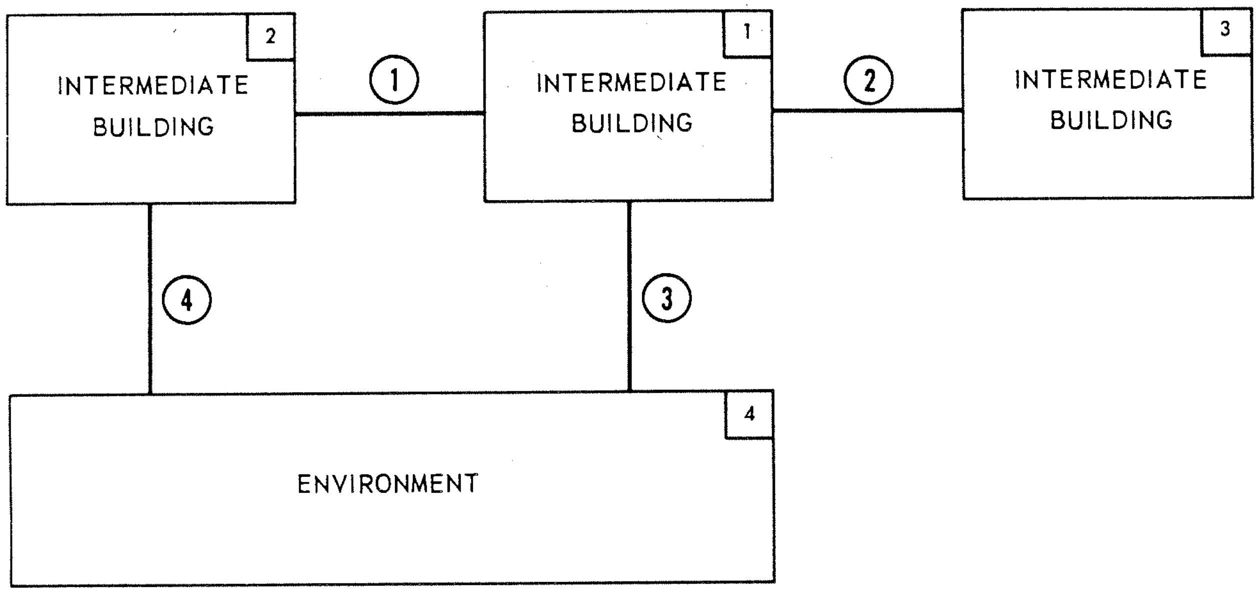
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Intermediate Building - East Penetration Access Area - Nodalization Model Figure 3.6-1b



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Intermediate Building - Subcompartment Containing Main Steam and Feedwater Headers - Nodalization Model

Figure 3.6-1c



LEGEND:

N – CONTROL VOLUME NUMBER

M – JUNCTION NUMBER

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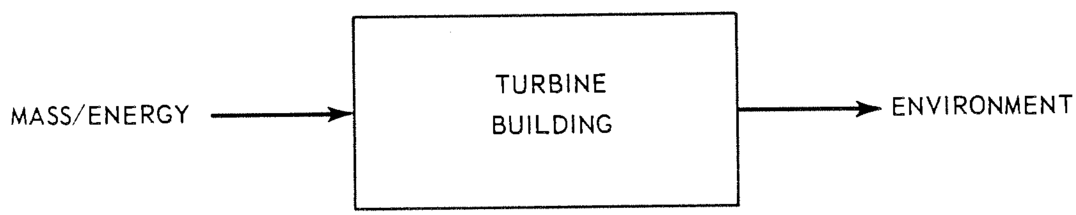
Intermediate Building - Outside Subcompartment Containing Main Steam and Feedwater Headers - Nodalization Model

Figure 3.6-1d



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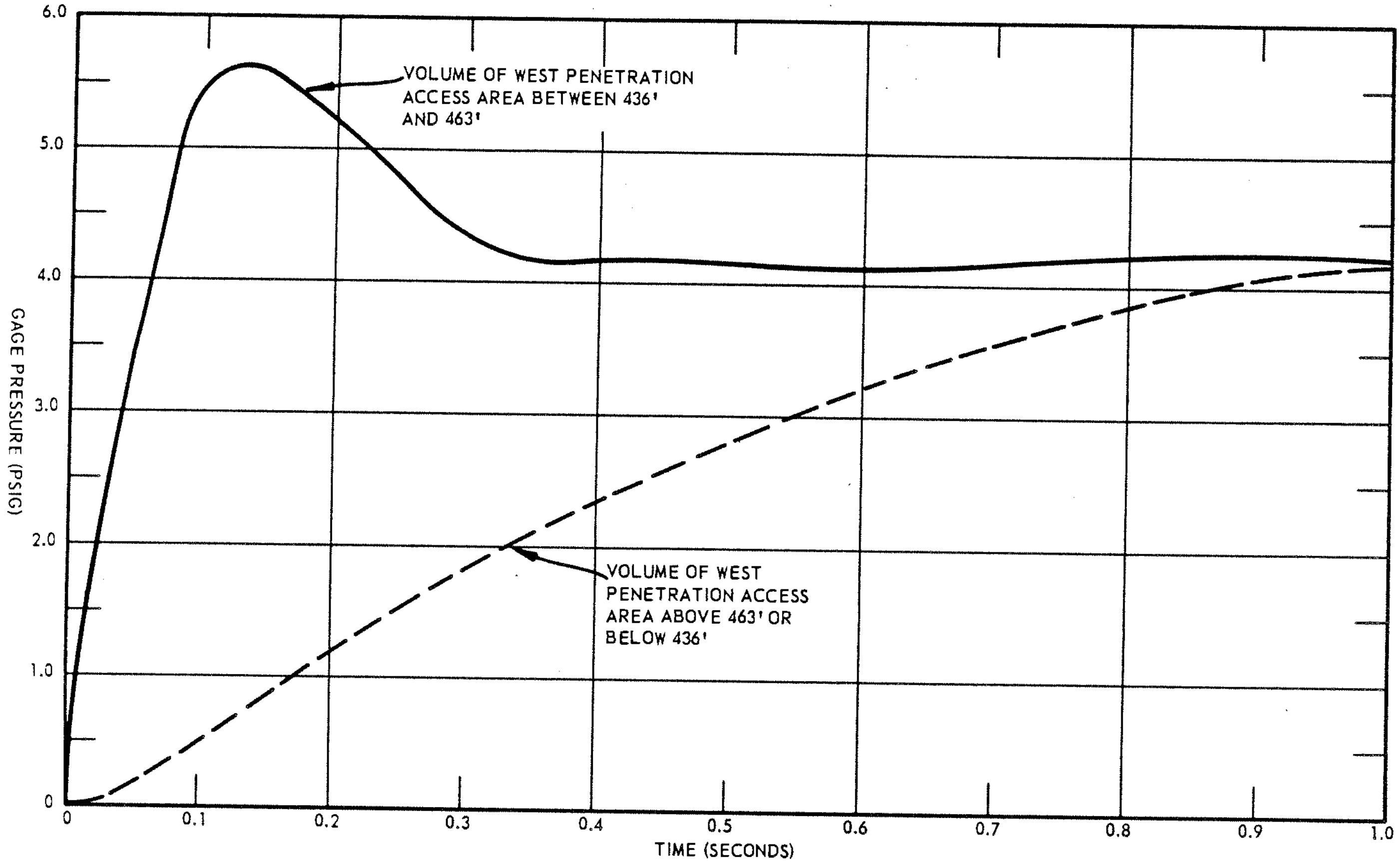
Turbine Building - Nodalization Model

Figure 3.6-1e



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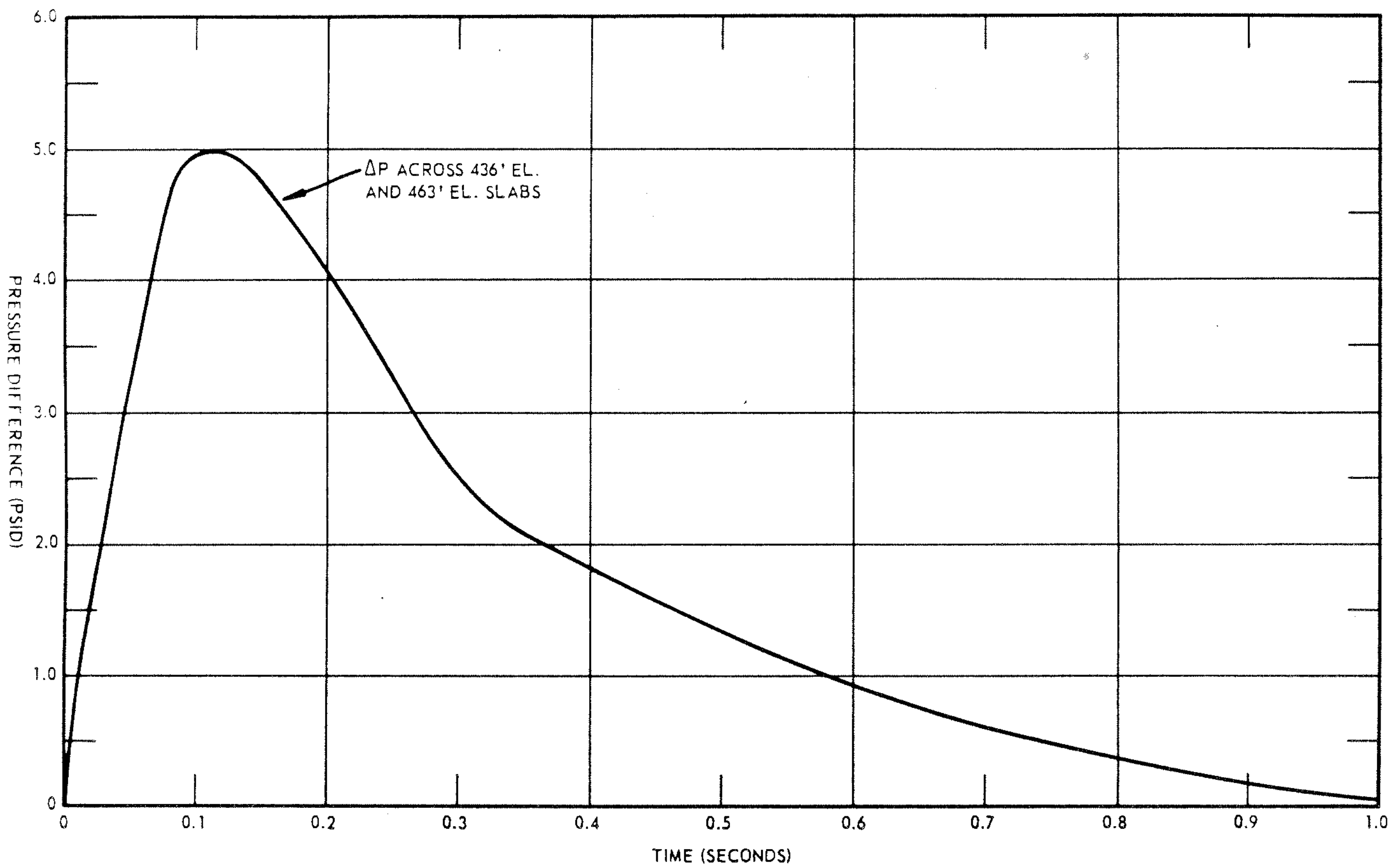
Gage Pressure versus Time Following a Main Steam Line Rupture in the West Penetration Access Area

Figure 3.6-2



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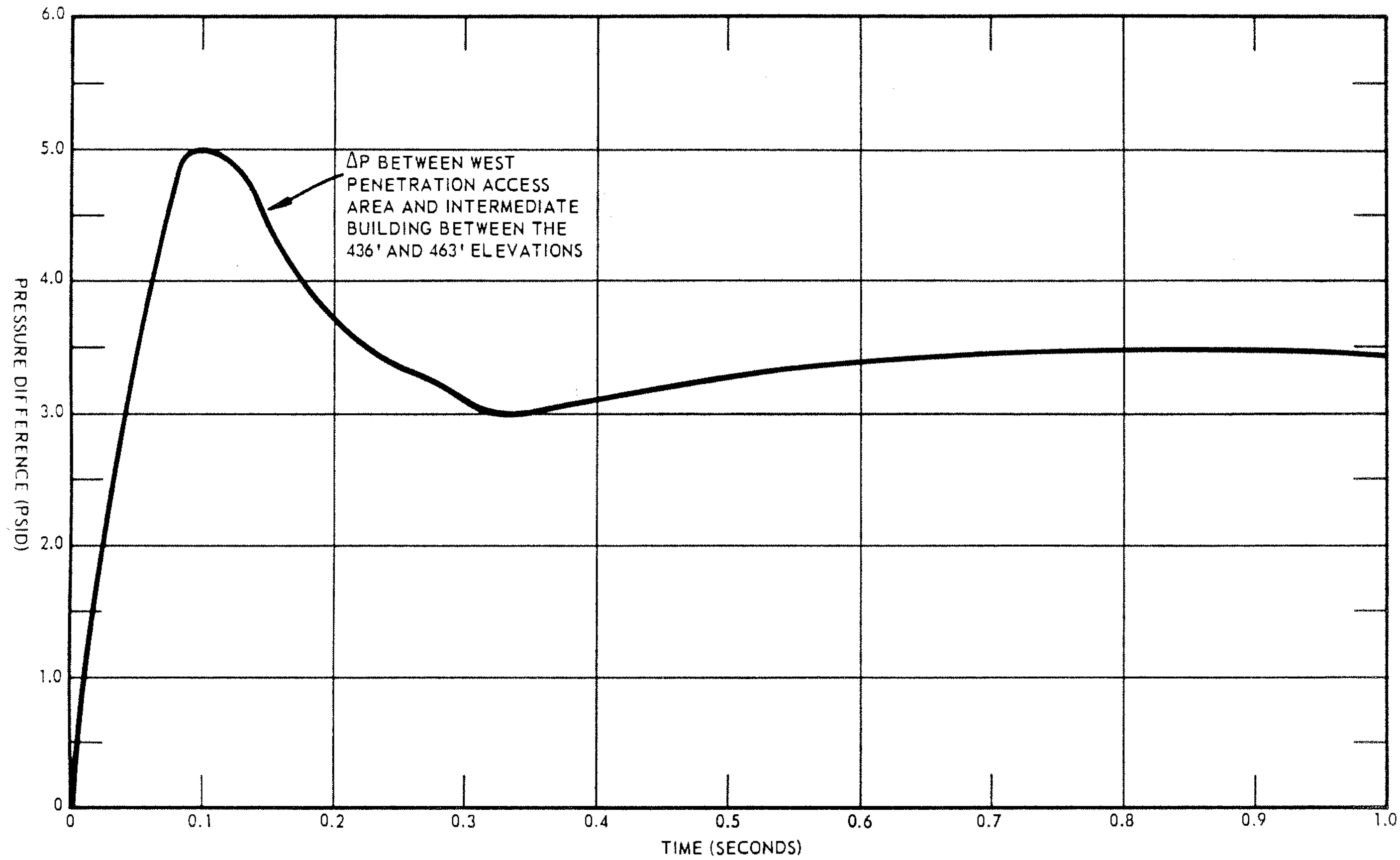
Differential Pressure versus Time Following a Main Steam Line Rupture in the West Penetration Area

Figure 3.6-3



SOUTH CAROLINA ELECTRIC & GAS CO.
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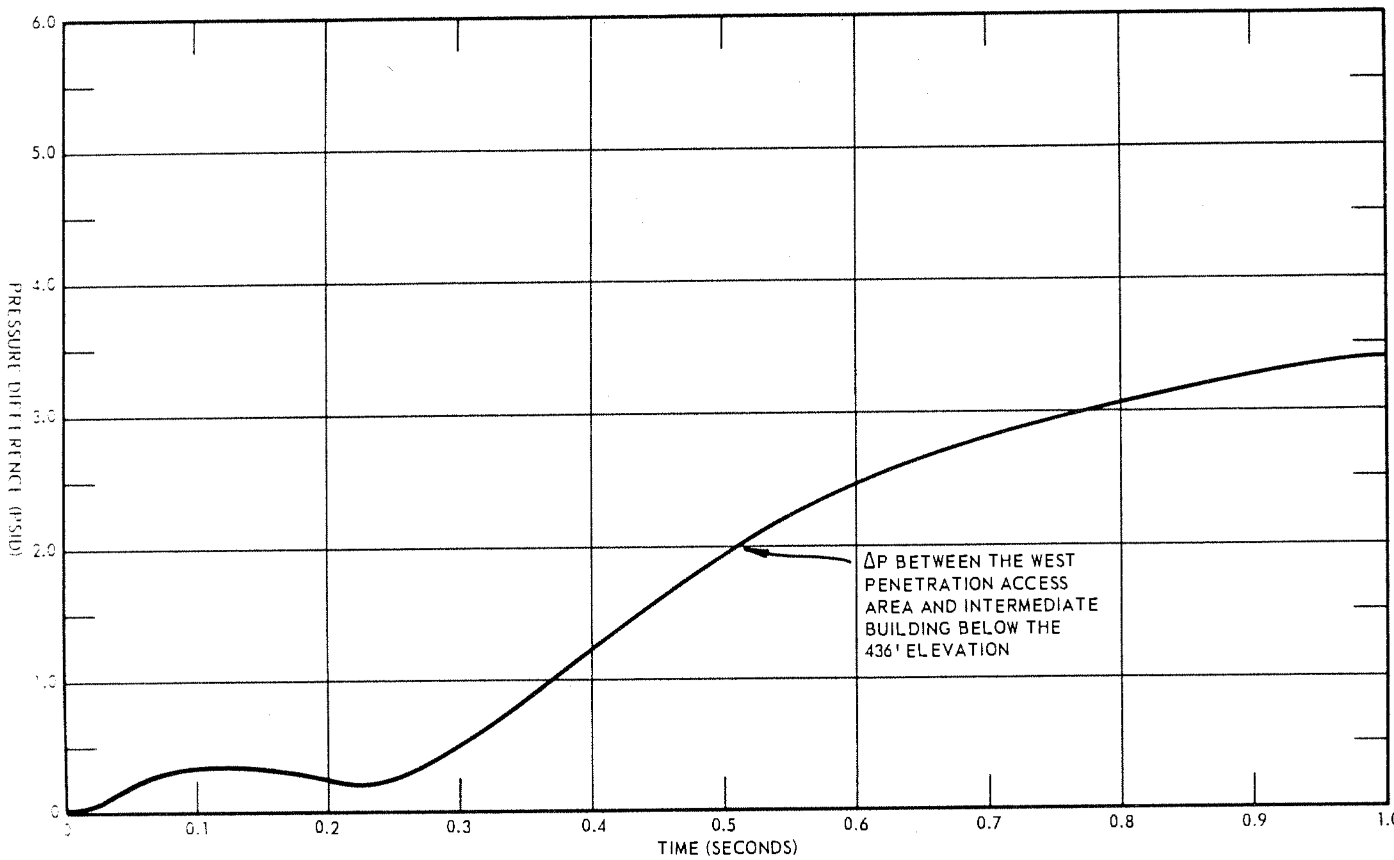
Differential Pressure versus Time Following a Main Steam Line Rupture in the West Penetration Area

Figure 3.6-4



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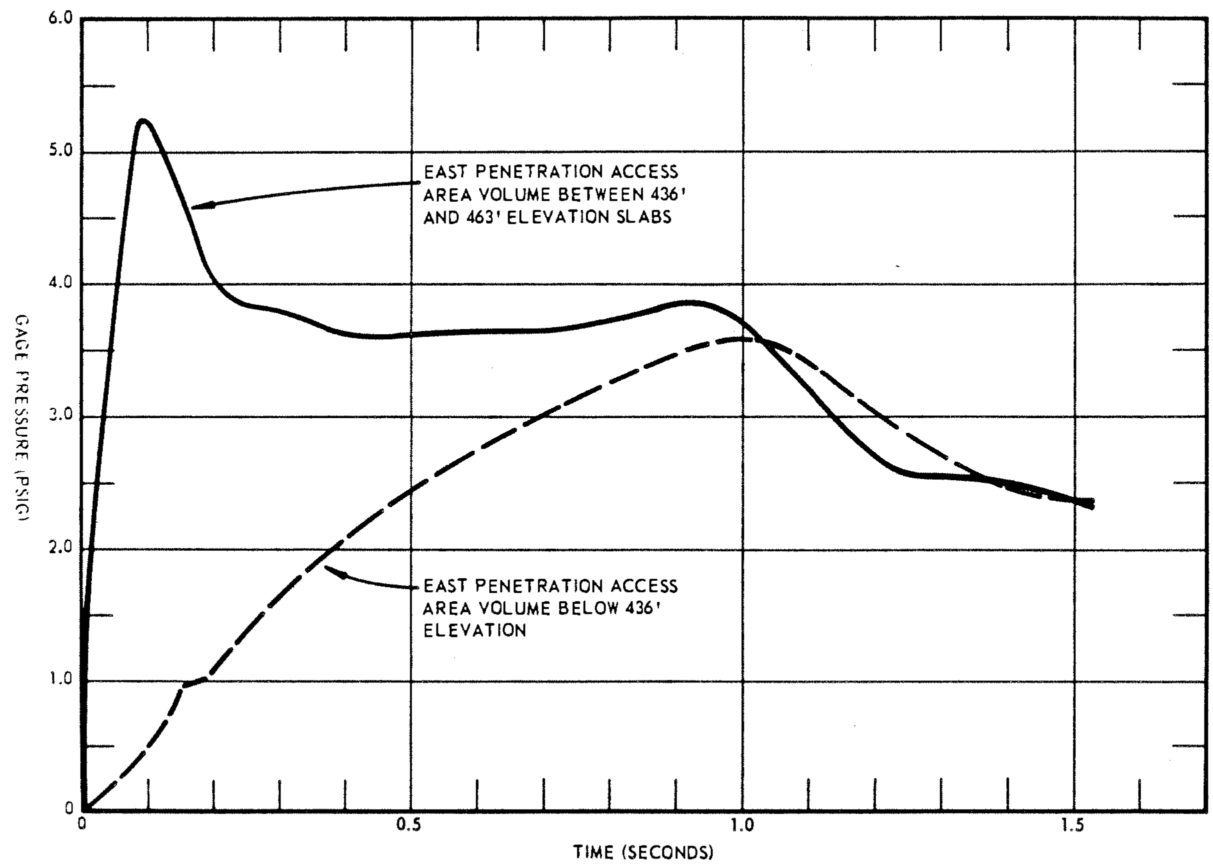
Differential Pressure versus Time Following a Main Steam Line Rupture in the West Penetration Access Area

Figure 3.6-5



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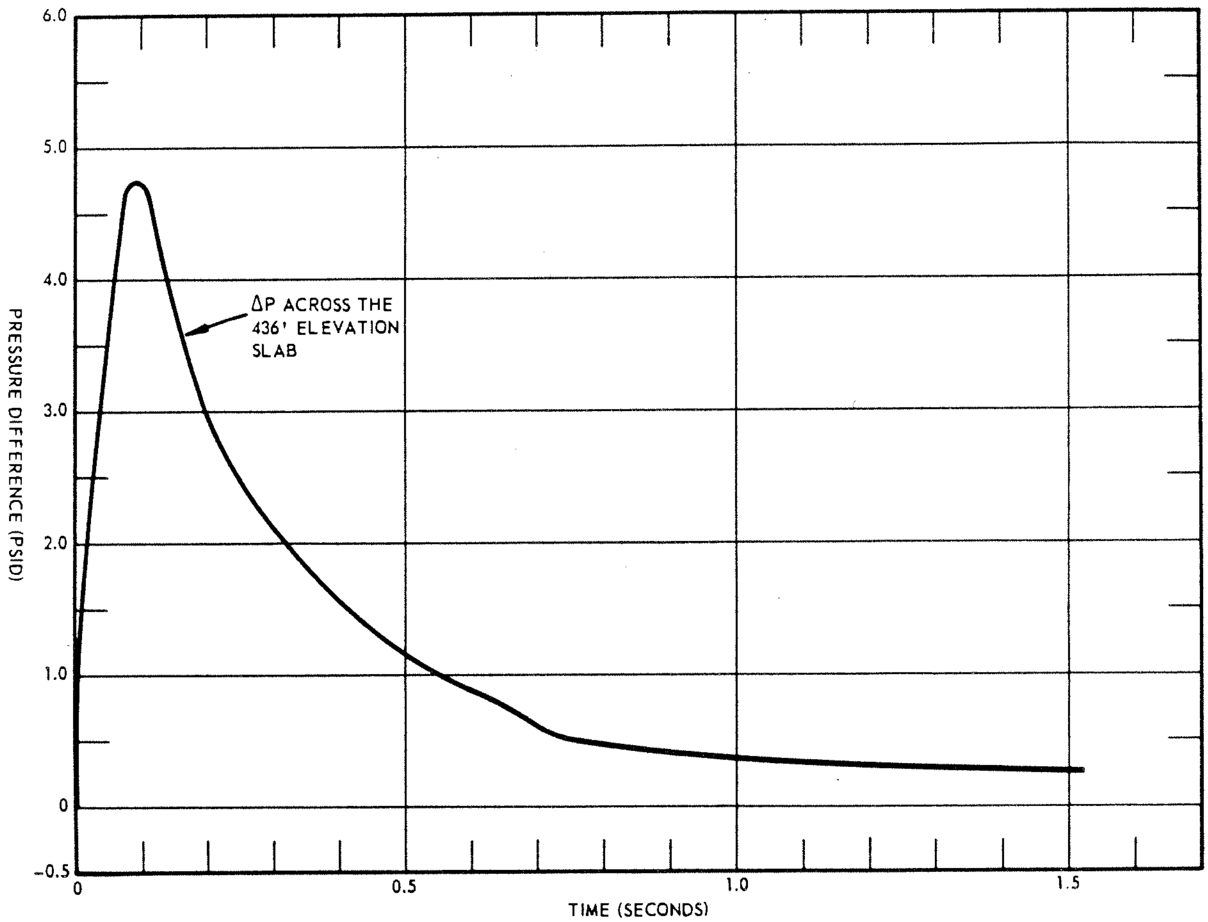
Gage Pressure versus Time Following a Main Steam Line Rupture in the East Penetration Access Area

Figure 3.6-6



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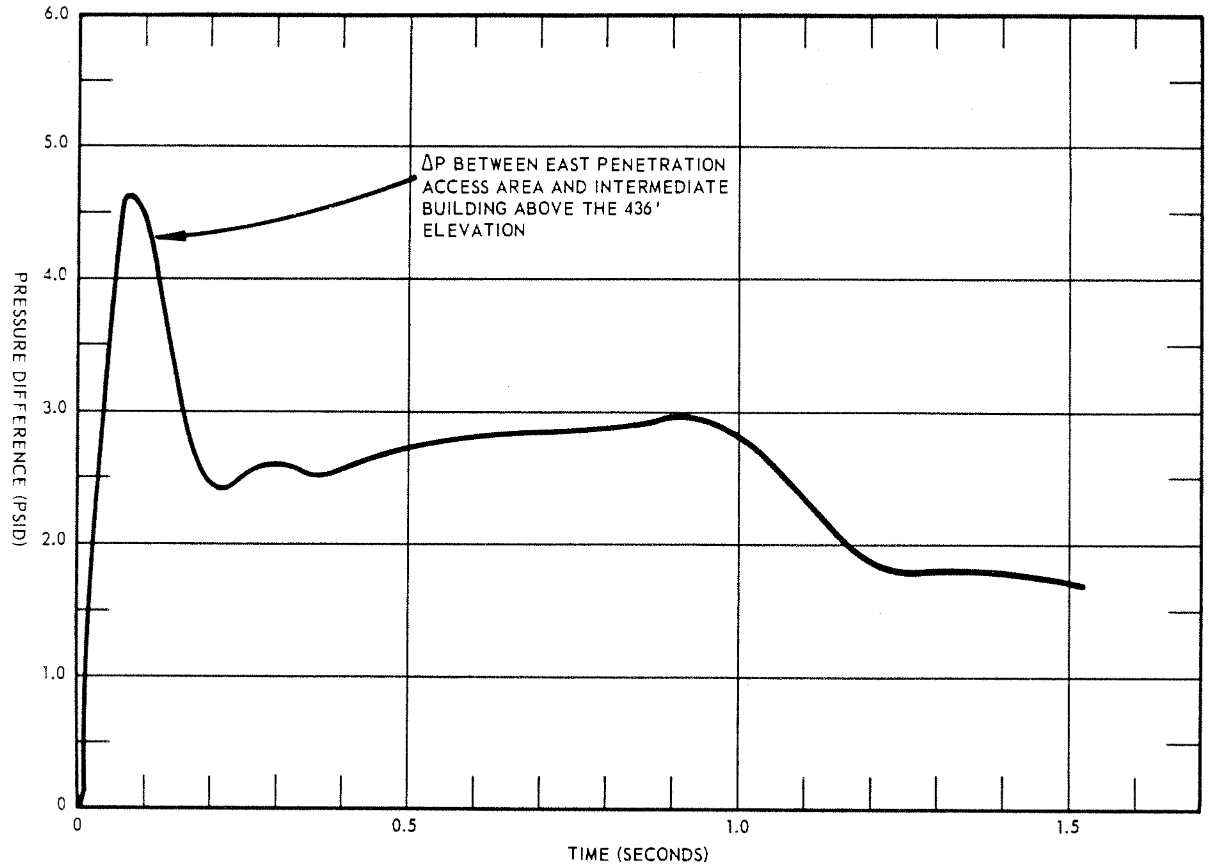
Differential Pressure versus Time Following a Main Steam Line Rupture in the East Penetration Access Area

Figure 3.6-7



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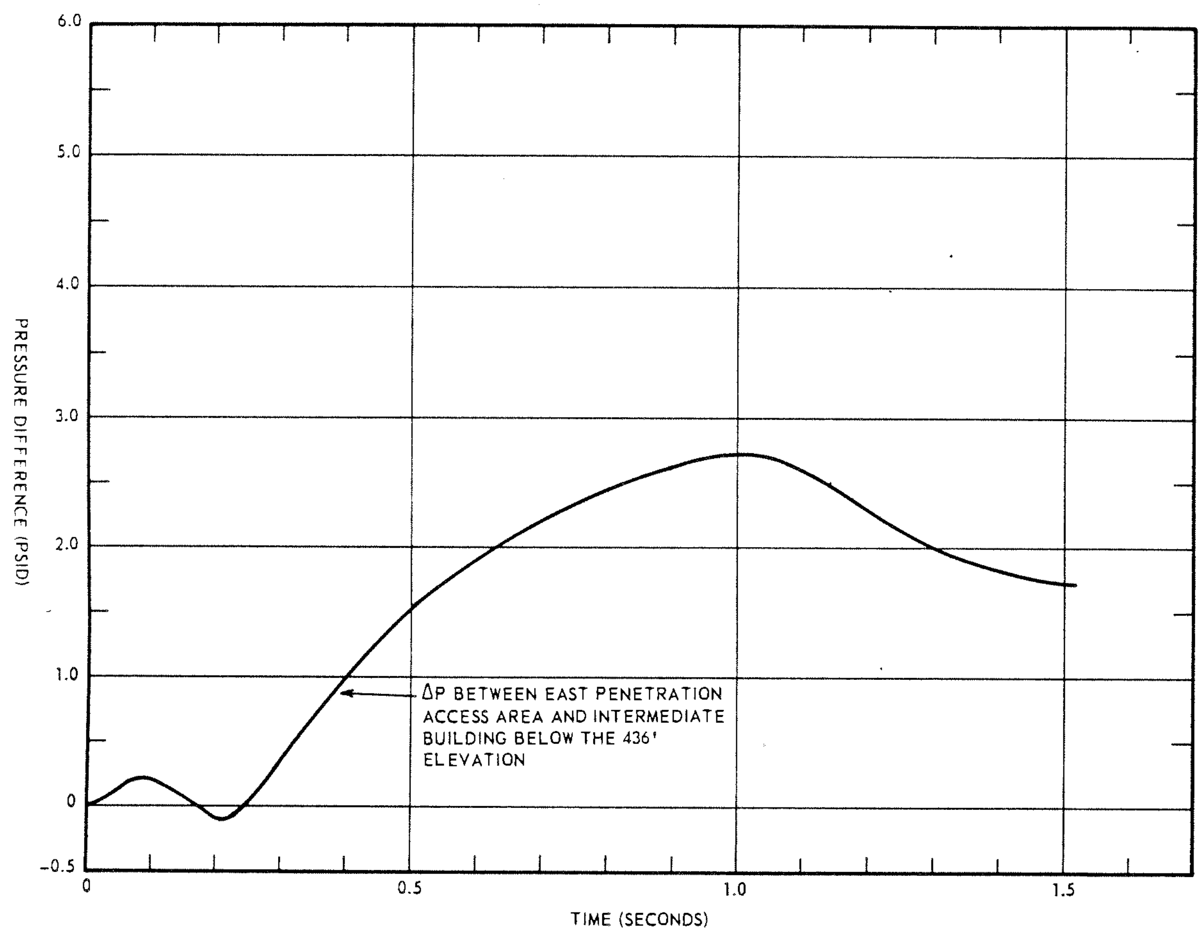
Differential Pressure versus Time Following a Main Steam Line Rupture in the East Penetration Access Area

Figure 3.6-8



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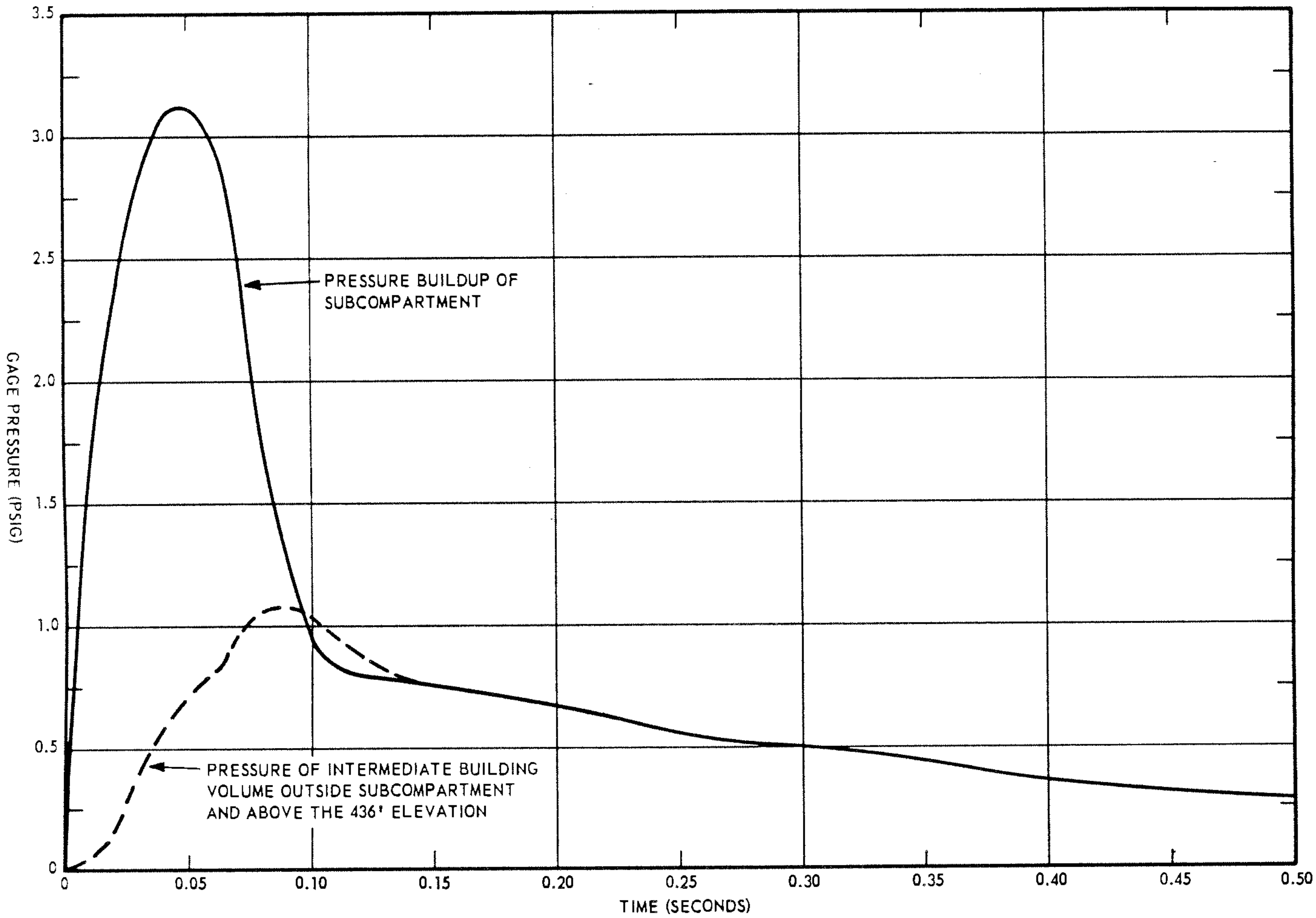
Differential Pressure versus Time Following a Main Steam Line Rupture in the East Penetration Access Area

Figure 3.6-9



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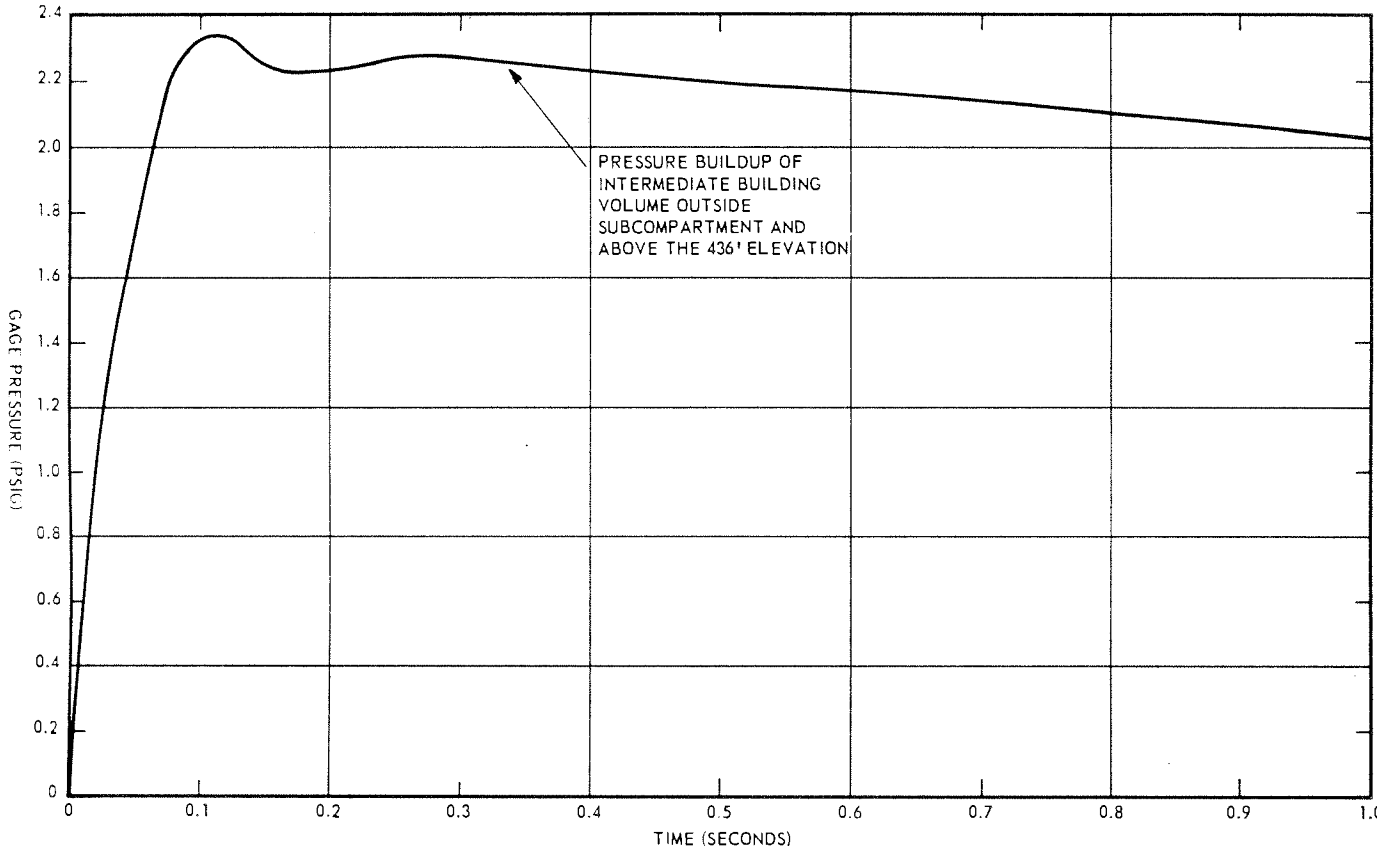
Gage Pressure versus Time Following a Main Steam Line Rupture in the Intermediate Building Sub-Compartment Containing the Main Steam and Feedwater Headers
Figure 3.6-10



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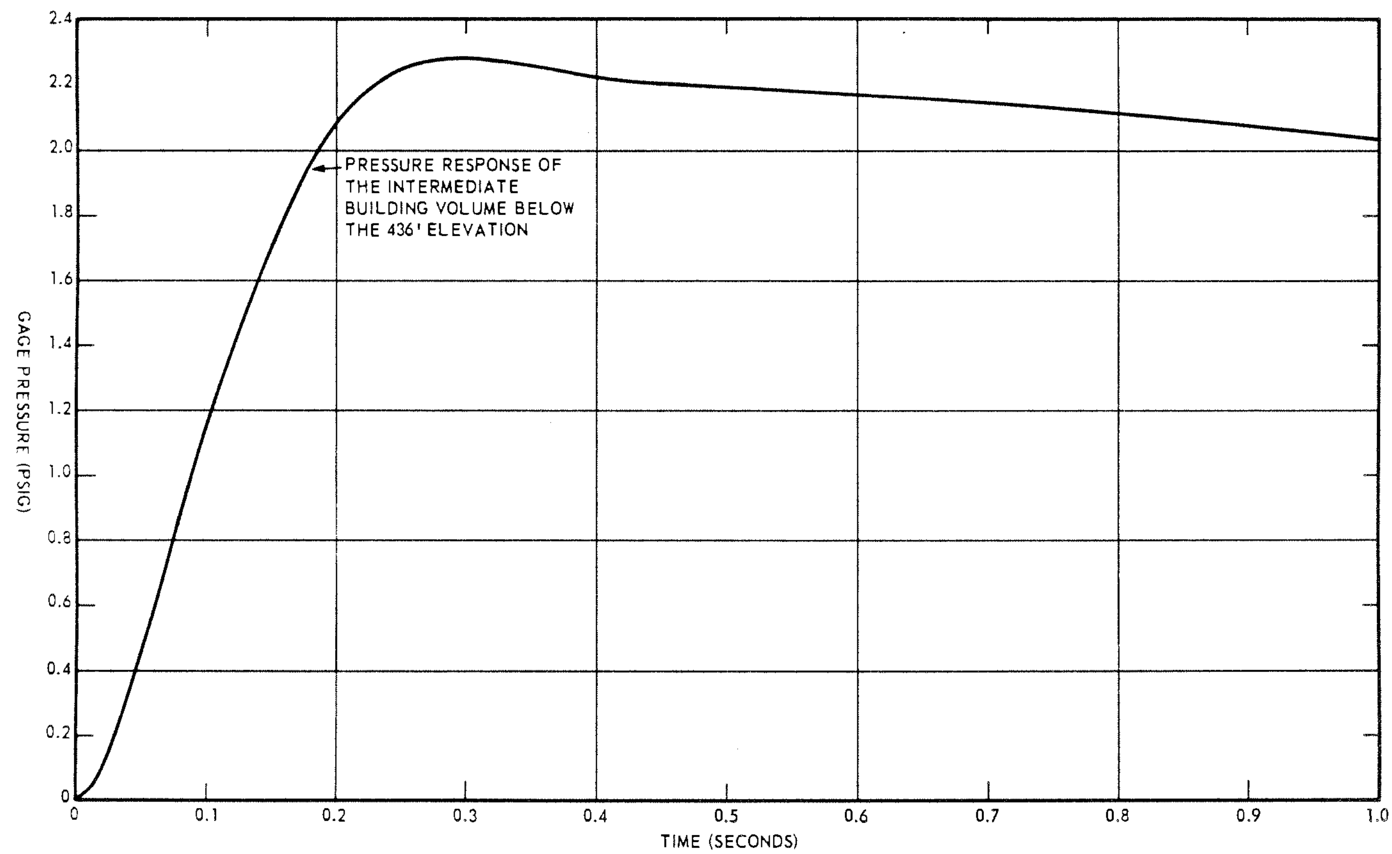
Gage Pressure versus Time Following a Main Steam Line Rupture in the Intermediate Building Outside the Sub-Compartment Containing the Main Steam and Feedwater Headers
Figure 3.6-11



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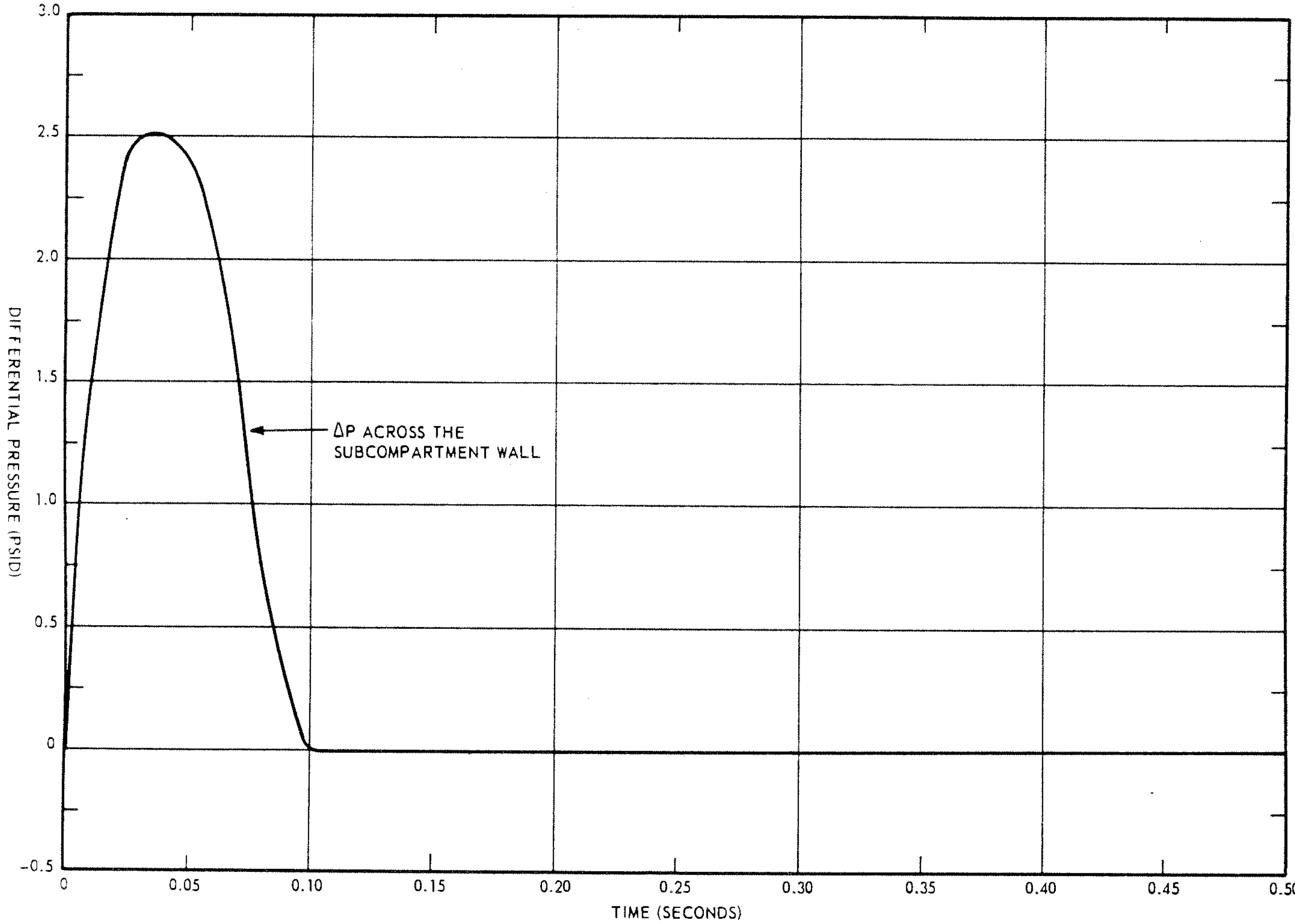
Gage Pressure versus Time Following a Main Steam Line Rupture in the Intermediate Building Outside the Sub-Compartment Containing the Main Steam and Feedwater Headers

Figure 3.6-12



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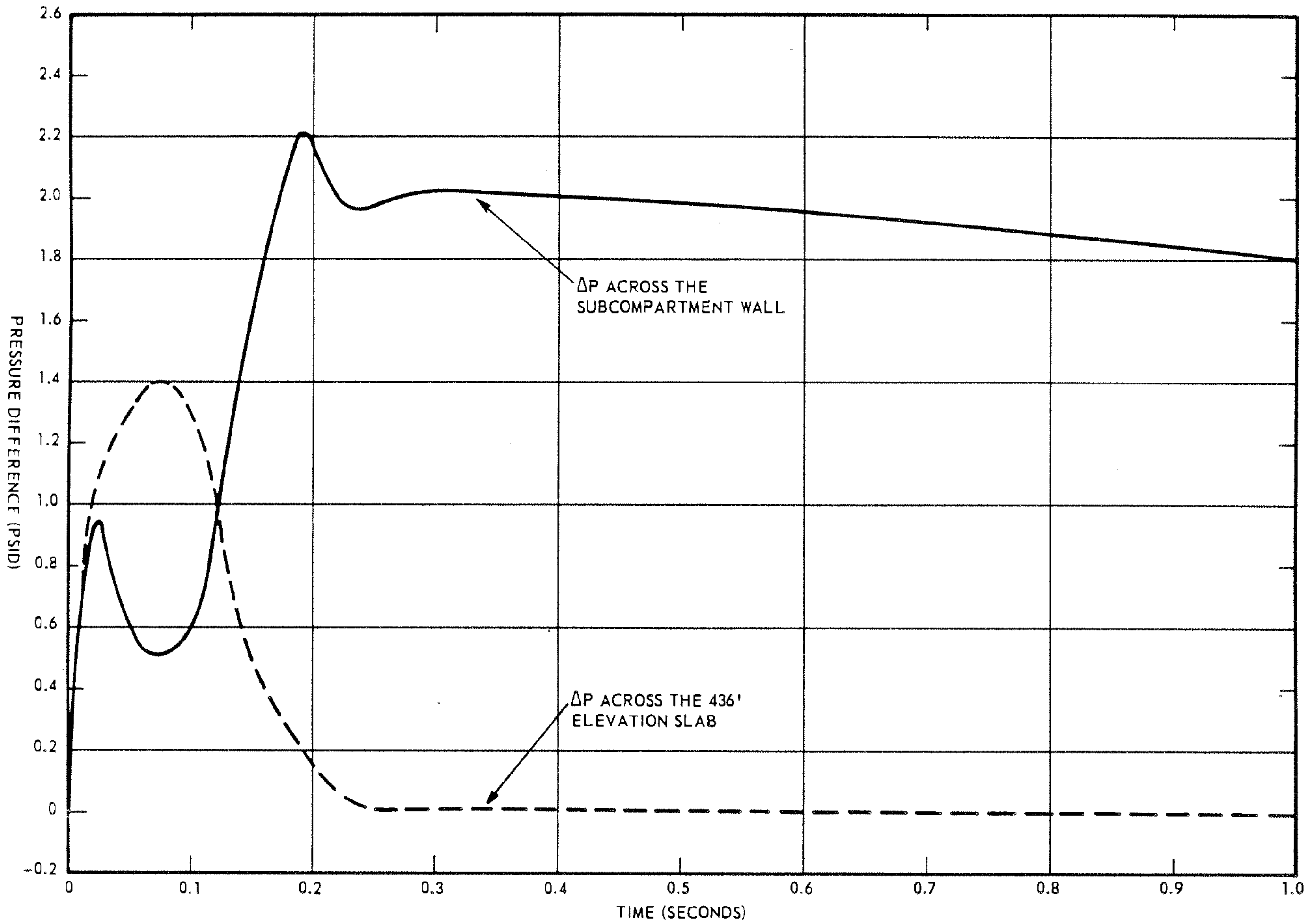
Differential Pressure versus Time Following a Main Steam Line Rupture in the Intermediate Building Sub-Compartment Containing the Main Steam and Feedwater Headers
Figure 3.6-13



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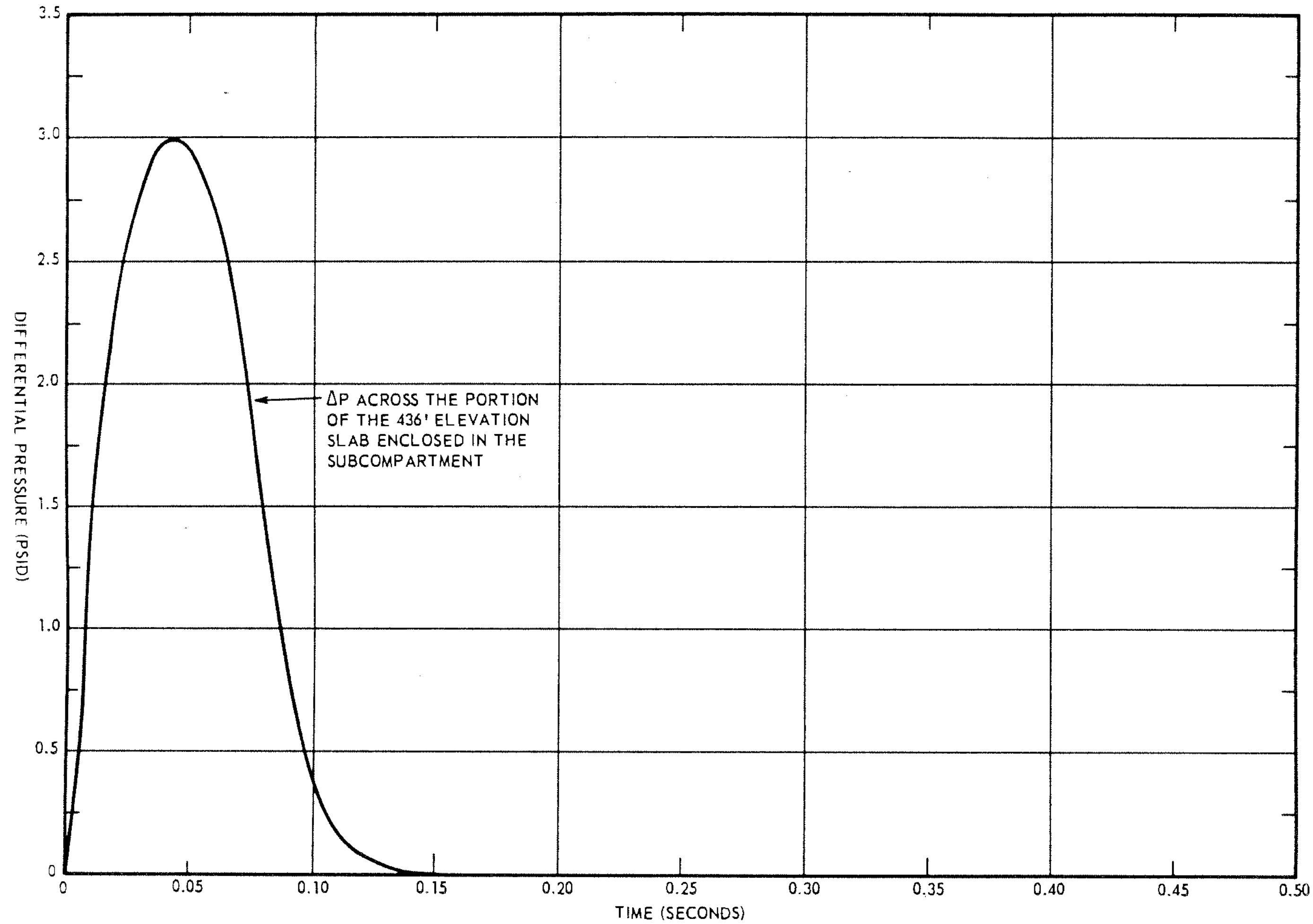
Differential Pressure versus Time Following a Main Steam Line Rupture in the Intermediate Building Outside the Sub-Compartment Containing the Main Steam and Feedwater Headers
Figure 3.6-14



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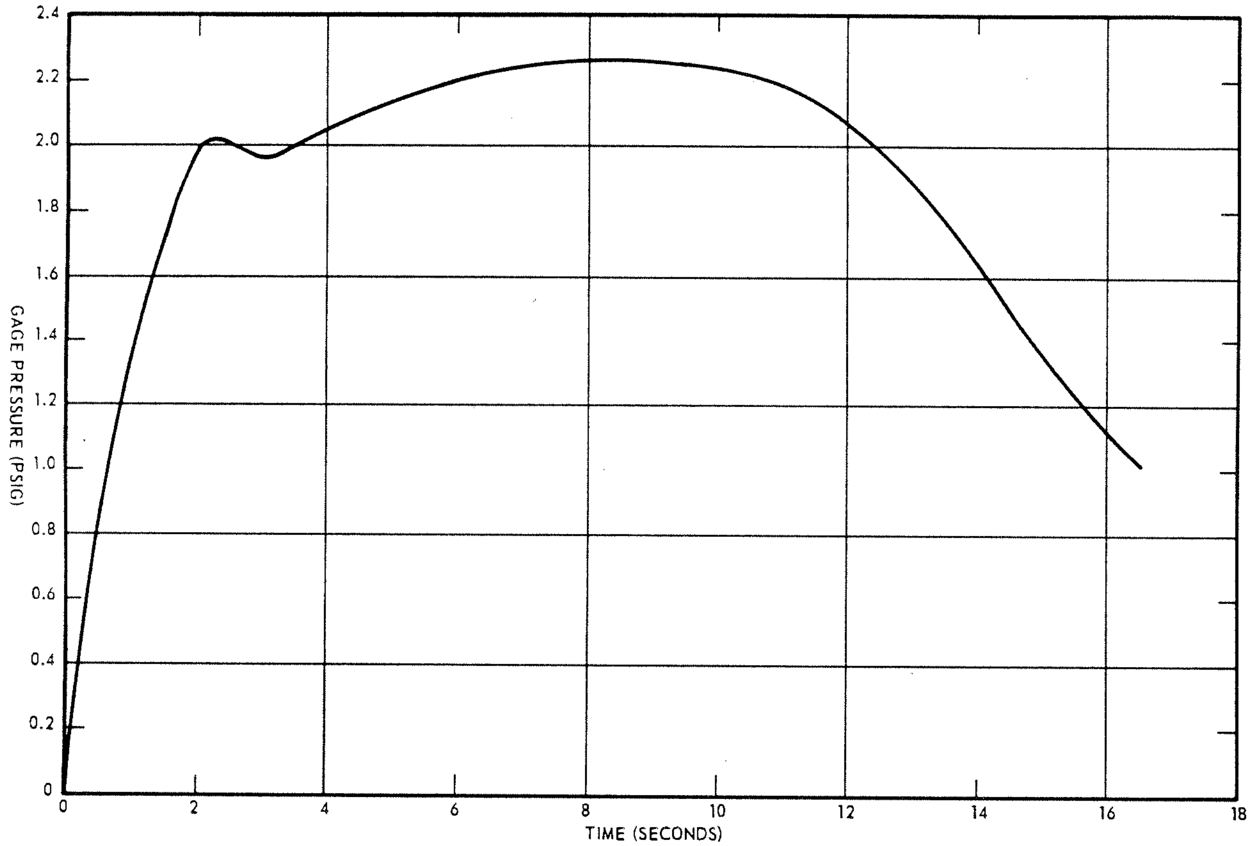
Differential Pressure versus Time Following a Main Steam Line Rupture in the Intermediate Building Sub-Compartment Containing the Main Steam and Feedwater Headers
Figure 3.6-15



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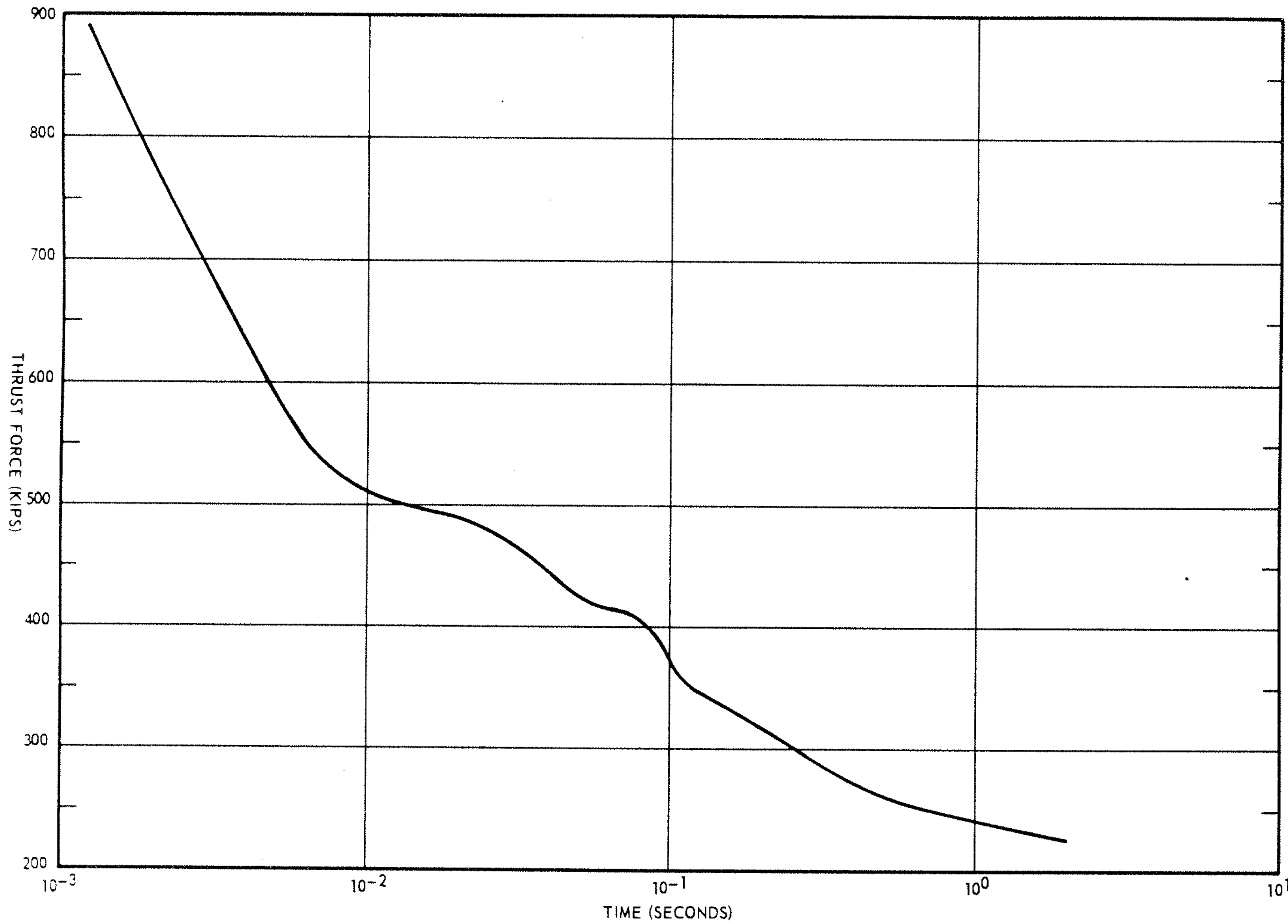
Gage Pressure versus Time Following a Main Steam Line Rupture in the Turbine Building

Figure 3.6-16



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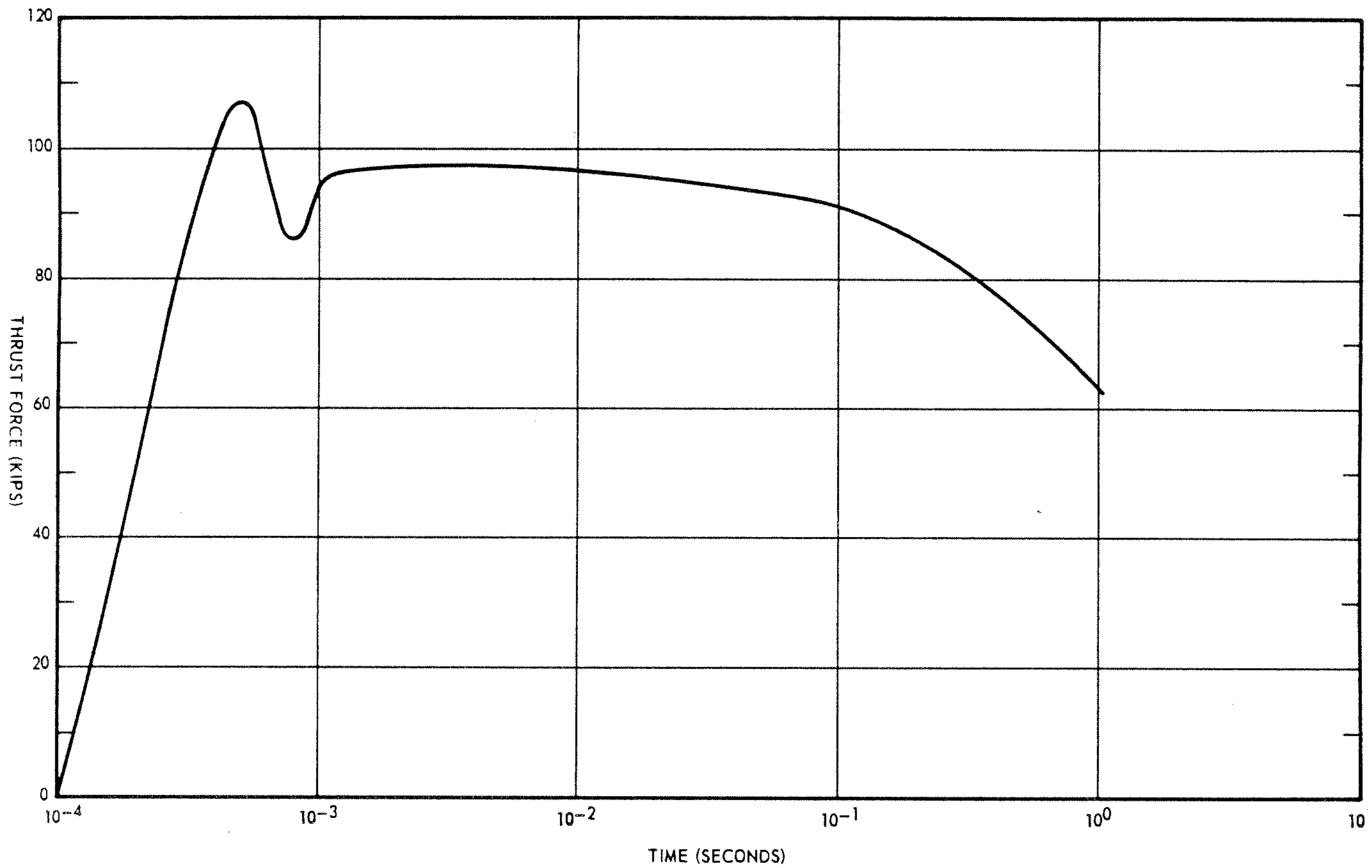
Thrust Force versus Time for Steam Generator Side of a Main Steam Line Break at the Containment Wall Penetration

Figure 3.6-17



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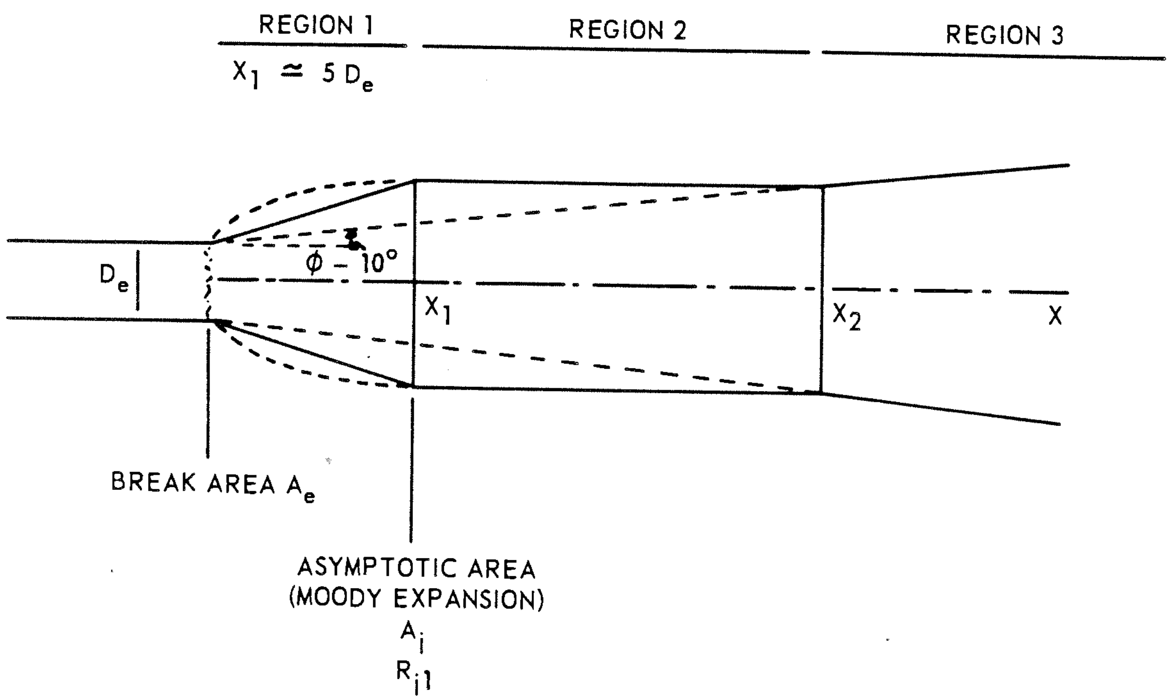
Thrust Force versus Time for Pump Side of Feedwater Line Break at Loop A or B Secondary Shield Penetration

Figure 3.6-18



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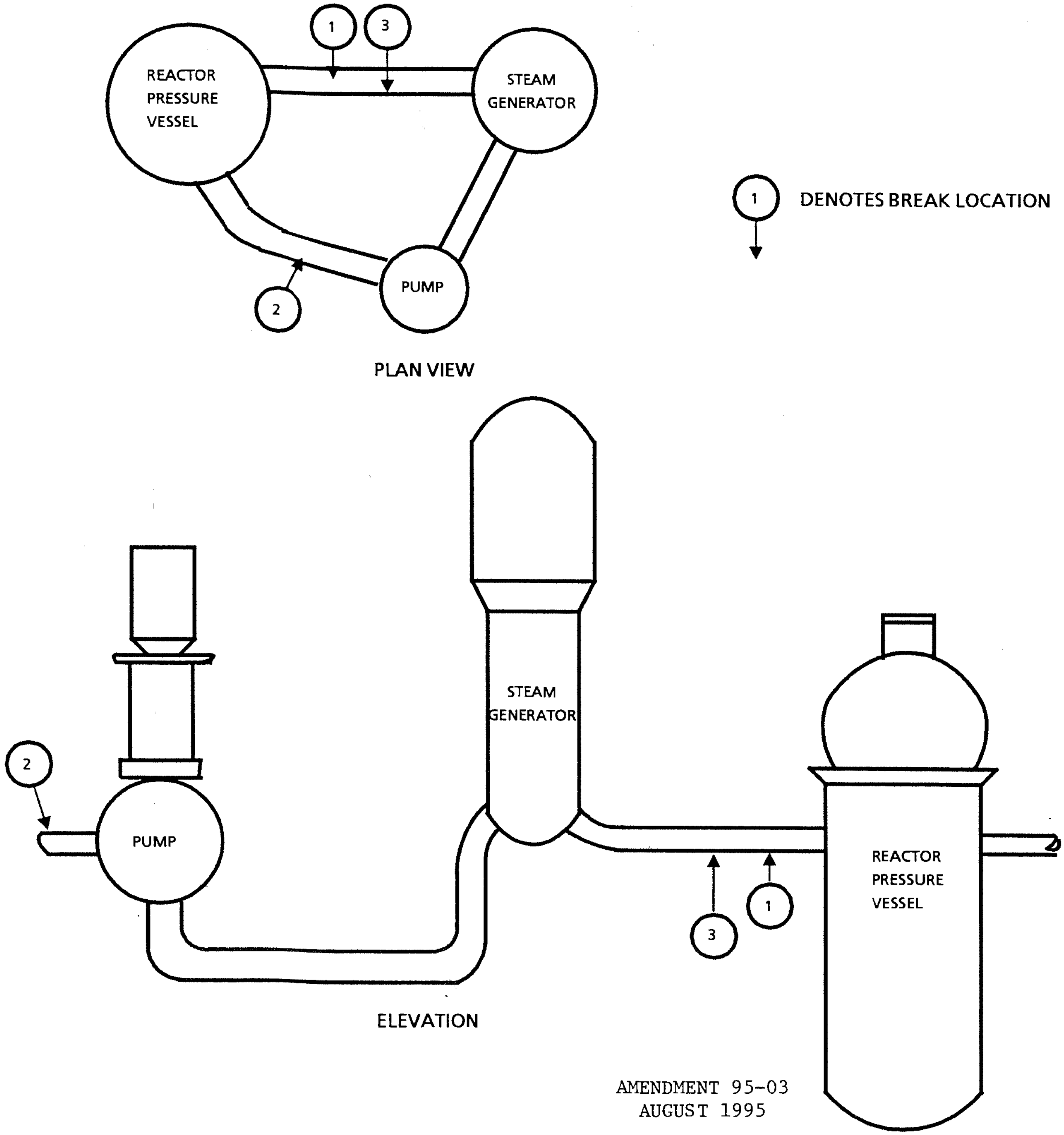
Fluid Jet Geometry

Figure 3.6-19

Figures 3.6-20 through 3.6-50 were deleted
by Amendment 1 dated August, 1985
(RN 01-071)



AMENDMENT 95-03
AUGUST 1995

SOUTH CAROLINA ELECTRIC & GAS CO.
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Location of Postulated Breaks in Main Reactor Coolant Loop

Figure 3.6-51

3.7 SEISMIC DESIGN

In addition to the steady-state loads imposed on the system under normal operating conditions, the design of equipment, equipment supports, and Seismic Category I structures requires that consideration also be given to abnormal loading conditions such as earthquakes. Seismic loading, are considered for earthquakes of two magnitudes: safe shutdown earthquake (SSE) and operating basis earthquake (OBE). The SSE is defined as the maximum vibratory ground motion at the plant site that can reasonably be predicted from geologic and seismic evidence. The OBE is that earthquake which, considering the local geology and seismology, can be reasonably expected to occur during the plant life.

Aside from these two major earthquakes, the effects of reservoir induced seismicity were reviewed in a separate study which is summarized in Section 3.7.1.5. Specific details on input, analysis methods and procedures, etc. for reservoir induced seismicity are not included within Section 3.7 but are contained in Reference [26].

For the OBE loading condition, the Nuclear Steam Supply System and safety-related balance of plant equipment and structures are designed to be capable of continued safe operation. The design for the SSE is intended to assure:

1. That the integrity of the reactor coolant pressure boundary is not compromised;

2. That the capability to shutdown the reactor and maintain it in a safe condition is not compromised; and

3. That the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.11 and 10 CFR 50.67 is not compromised. RN 12-034

It is necessary to ensure that required Seismic Category I structures, systems, and components do not lose their capability to perform their safety function. Not all components have the same functional safety requirements. For example, a charging pump must retain its capability to function normally during the SSE. Therefore, the deformation in the pump must be restricted to appropriate limits to assure its ability to function. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

This Section is presented in the following subsections:

3.7.1 Seismic Input

3.7.2 Seismic System Analysis

3.7.3 Seismic Subsystem Analysis

3.7.4 Seismic Instrumentation Program

3.7.5 Seismic Design Control

The seismic requirements for safety-related instrumentation and electrical equipment are covered in Section 3.10. The safety class definitions and classifications are given in Section 3.2.

3.7.1 SEISMIC INPUT

3.7.1.1 Design Response Spectra

Design response spectra for the horizontal component of the SSE, formerly the design basis earthquake, and OBE are applied in the design of Seismic Category I structures, systems, and components according to preliminary results of studies conducted by N. M. Newmark and J. A. Blume, consultants to the AEC. These studies were performed prior to and the response spectra were developed before issuance of Regulatory Guide 1.60. Regulatory Guide 1.60 is discussed in Appendix 3A.

Separate design response spectra are specified at rock and soil foundation elevations. Horizontal spectra for structures founded upon rock like media are presented as Figures 3.7-1 and 3.7-2 for the SSE and OBE, respectively. Design response spectra for structures founded upon soil are presented as Figures 3.7-3 and 3.7-4 for the SSE and OBE, respectively.

The vertical component spectra used are two thirds of the horizontal components in all frequency ranges and occur simultaneously.

The maximum horizontal ground acceleration for the SSE is 0.15g at the competent rock foundation elevation and 0.25g for the soil foundation. For the OBE, the maximum horizontal ground accelerations used are 0.1g (rock) and 0.15g (soil). Seismological and geological background data pertaining to the plant site are presented in Section 2.5.

The design response spectra are similar to the previous NRC minimum criteria (see Figure 17 of Reference [20]). The differences between the design response spectra and the current NRC recommended spectra of Regulatory Guide 1.60 have been investigated for all damping values considered. It was found that the design response spectra are, in general, enveloped by the recommended spectra throughout the entire frequency range and are apparently less conservative in the range of frequencies below about 0.55 Hz. In the latter region, the difference is of no concern, in general, for Seismic Category I structures, systems, and components. For the region from about 2 to 9 Hz, where the predominant frequencies of the Seismic Category I structures lie, the differences between the spectra are relatively small. Specifically, in the frequency range from about 4.6 to 6.8 Hz, the two sets of spectra are practically the same for the 2, 5, and 7% curves. In the frequency ranges from about 2 to 4.6 Hz and from about 6.8 to 9 Hz, the maximum differences with respect to the design spectra are about 22, 20, 18, 18, and 23%, respectively, for 0.5, 2, 5, 7, and 10% critical damping values. In the ranges of frequencies from about 0.55 to 2 Hz and from about 9 to 33 Hz, the differences between the spectra are again relatively small, except possibly for the spectrum curves of 7 and 10% critical damping.

The differences between the two sets of response spectra stated above might result in an underestimate of the responses of Seismic Category I structures should the damping values of Regulatory Guide 1.61 be used. However, with the use of more conservative (i.e., lower) damping values than those of Regulatory Guide 1.61, the two sets of spectra would lead to very nearly the same results, as is shown in Section 3.7.1.3.

As indicated in Section 3.7.1.4, some of the Seismic Category I structures are founded upon competent soils. Others are founded upon rock by means of caissons or lean concrete. The foundation elevation for each of the Seismic Category I structures is given in Tables 3.7-5 and 3.7-6. In the seismic analyses, with soil structure interaction effects included, the design spectra are applied at the various foundation locations of Seismic Category I structures. For the Reactor Building, Auxiliary Building, and Control Building, which are founded upon lean concrete bearing on rock, the design spectra in Figures 3.7-1 and 3.7-2 are applied in the respective lumped-mass-spring soil-structure interaction system at the foundation, as listed in Table 3.7-5. For the Diesel Generator Building, Fuel Handling Building, and Intermediate Building, which are founded upon caissons, the effective design spectra are applied throughout the elevation of the caissons in the lumped-mass-spring soil-structure interaction systems. Such effective design spectra are determined based upon equivalent energy input accounting for the design spectra at rock elevation (Figures 3.7-1 and 3.7-2) and the design spectra at soil elevation (Figures 3.7-3 and 3.7- 4), as described in Section 3.7.2.4. For the Service Water Pumphouse, which is founded on competent soils, the design spectra (Figures 3.7-1 and 3.7-2) are applied at the base rock elevation, at 350 feet, in the finite element soil-structure interaction system (see Section 3.7.2.1.1). Since no structural response amplification was expected from the service water intake structure and service water discharge structure, design spectra were not used for these two structures. Instead, effective seismic loads, based upon the maximum ground accelerations, were conservatively used for design as is described in Section 3.8.4.4.8 and 3.8.4.4.9, respectively.

3.7.1.2 Design Time History

A synthetic earthquake ground motion time history was developed to simulate the SSE and OBE, based upon the design response spectra discussed in Section 3.7.1.1, for the time history analysis of structures, components, and equipment.

The time history is compatible with the corresponding response spectra. The "valleys" in the unsmoothed response spectra generated from the time history do not fall below the corresponding smoothed response spectra presented as Figures 3.7-1 through 3.7-4 for rock and soil foundation elevations. Figures 3.7-5 through 3.7-8 present comparisons between the smoothed response spectra and spectra derived from the synthetic earthquake time history for 2%, 5%, 7%, and 10% of critical damping, respectively.

The synthetic earthquake time history was developed with an equal time interval of 0.01 second. The period intervals at which the unsmoothed response spectra values were calculated ranged from 0.001 second to 0.2 second. A total of 108 points were used in generating each spectrum. The closest period intervals were used in the most critical portions of each spectrum. In Table 3.7-0, the frequency intervals used for calculating the unsmoothed response spectra (see Figures 3.7-5 and 3.7-8) from the synthetic earthquake time history are given.

3.7.1.3 Critical Damping Values

3.7.1.3.1 Balance of Plant Scope

The specific percentage of critical damping values used for Seismic Category I structures, systems, and components are presented in Table 3.7-1 and 3.7-2. The values were obtained from Reference [1]. These values are equivalent to or provide less damping than the current values in Regulatory Guide 1.61 (see Appendix 3A). For example, the damping values provided in Table 3.7-1 for reinforced concrete structures are 2 and 5% critical for stress levels no more than about half the yield point and at or just below the yield point, respectively. The corresponding values in Regulatory Guide 1.61 are 4 and 7% critical for OBE and SSE, respectively. The damping values for stress levels beyond the yield point, as shown in Table 3.7-1, were not used in the analysis. A comparison of the design spectrum of 2% (Section 3.7.1.1) and the recommended spectrum of 4% in Regulatory Guide 1.60 indicates that, except in the range of frequencies below about 0.55 Hz, the design spectrum envelops the recommended spectrum with significant margin, especially in the frequency range from about 0.55 to 9 Hz. The 5% design spectrum is also compared with the 7% recommended spectrum with similar conclusion obtained.

In the case of prestressed concrete structures, such as the Reactor Building shell, the corresponding damping values required in Table 3.7-1 are equivalent to those in Regulatory Guide 1.61 which are 2 and 5% critical for OBE and SSE respectively. In actual design of the Reactor Building shell, however, a damping value of 2% was used for both OBE and SSE. This, coupled with the only slight unconservatism of the design spectra in comparison to the recommended spectra (see Section 3.7.1.1), suffices to assure the adequacy and conservatism of the structural responses under both OBE and SSE.

Based upon the above, the use of more conservative damping values, together with the design spectra, has provided a measure of the seismic responses of Seismic Category I structures, systems, and components which is equivalent to or more conservative than would be provided by using the damping values in Regulatory Guide 1.61 combined with the recommended spectra in Regulatory Guide 1.60.

For the soil-structure interaction system, the hysteretic damping values for the foundation media were taken to be 5% for the rocking mode and 10% for the swaying mode, regardless of the type of foundation media and frequency range. For conservatism, the radiation (viscous) damping due to wave propagation from a structure into the half-space, which is frequency dependent, was not considered.

In a typical classical modal analysis, different damping has been specified for each mode of the vibratory system. The method of weighted modal damping as described in Reference [2] was used to calculate the damping for each mode in the dynamic analysis of the soil-structure interaction system.

3.7.1.3.2 Components and Equipment Provided by the NSSS Vendor

The damping values given in Table 3.7-3 are used for the systems analysis of Westinghouse equipment. These are consistent with the damping values recommended in Regulatory Guide 1.61 except in the case of the primary coolant loop system components and large piping (excluding reactor pressure vessel internals) for which the damping values of 2% and 4% are used as established in testing programs reported in Reference [3]. The damping values for control rod drive mechanisms (CRDM) and the fuel assemblies of the Nuclear Steam Supply System are in conformance with the values for welded and/or bolted steel structures as listed in Regulatory Guide 1.61.

Tests on fuel assembly bundles justified conservative component damping values of 7% for OBE and 10% for SSE to be used in the fuel assembly component qualification. Documentation of the fuel assembly tests is presented in Reference [4].

The damping values used in component analysis of CRDM and their seismic supports were developed by testing programs performed by Westinghouse. These tests were performed during the design of the CRDM support; the support was designed so that the damping in Table 3.7-3 could be conservatively used in the seismic analysis. The CRDM support system is designed with plates at the top of the mechanism and gaps between mechanisms. These are encircled by a box section frame which is attached by tie-rods to the refueling cavity wall. The test conducted was on a full size CRDM complete with rod position indicator coils, attachment to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support. The internal pressure of the CRDM was 2250 psi and the temperature on the outside of the pressure housing was 400°F.

The program consisted of transient vibration tests in which the CRDM was deflected a specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variations in the drive shaft axial position, upper seismic support clearance, and initial deflection amplitude was investigated.

The upper support clearance had the largest effect on the CRDM damping with the damping increasing with increasing clearance. With an upper clearance of 0.06 inches, the measured damping was approximately 8%. The clearances in a typical upper seismic CRDM support are a minimum of 0.10 inches. The increasing damping with increasing clearances trend from the test results indicated that the damping would be greater than 8% for both the OBE and the SSE based on a comparison between typical deflections during these seismic events to the initial deflections of the mechanisms in the test. Component damping values of 5% are, therefore, conservative for both OBE and SSE.

The Replacement Service Structure, Integrated Head Assembly (IHA), was designed similarly to the original CRDM Seismic Support Assembly. Damping values used for the original design were applied to the IHA with the CRDM Support System designed with plates at the top of the CRDM mechanisms with gaps between the plates that approximate the original design. RN 16-003

These damping values are used and applied to CRDM component analyses by response spectra techniques. Where time history analyses are used, damping in the pressure housing is set at the levels stated in Table 3.7-3 for welded steel structures and impact damping is applied at the contact points in the upper seismic supports.

3.7.1.4 Supporting Media for Seismic Category I Structures

The Seismic Category I service water intake structure, Service Water Pumphouse, and service water discharge structure are supported by mats founded upon competent soil. Other Seismic Category I structures are founded upon rock by means of caissons or mats. For the Reactor Building, Auxiliary Building, and Control Building, soils below planned foundation grade have been excavated and replaced with lean concrete bearing on rock.

Dynamic engineering properties of the soil and bedrock at the site have been evaluated for use in foundation interaction analyses. These properties, presented in Table 3.7-4, were developed using field geophysical and geologic data and static and dynamic laboratory test data. These values are appropriate for design at strain levels corresponding to the SSE and OBE. The shear module and subgrade module developed from laboratory and geophysical data have been reduced using factors related to rock quality designation (RQD). More detailed information concerning the foundation media is provided in Section 2.5.

Descriptions of the supporting media for Seismic Category I structures are presented in Table 3.7-5.

The foundation elevation and the foundation type for the Seismic Category I structures are presented in Table 3.7-5. Foundation material data for those structures supported on caissons is presented in Table 3.7-6.

Cross sections illustrating building foundation elevations, general subsurface conditions, and surface topography are presented in Figures 2.5-77 through 2.5-82. The locations of the sections are shown on Figures 2.5-46 and 2.5-47. The approximate dimensions, including height and top elevation, as well as the embedment depth of the Seismic Category I structures, as previously discussed, are presented in Table 3.7-7.

3.7.1.5 Effects of Reservoir Induced Seismicity

In addition to the seismic design data and procedures for the normal tectonic earthquake as presented within Section 3.7 herein, the effects of Reservoir Induced Seismicity were fully investigated in a separate study completed in 1983.

Shortly after filling of Monticello Reservoir in late 1977, a series of small earthquakes began occurring in the vicinity of the Virgil C. Summer Nuclear Station. These events, termed Reservoir Induced Seismicity (RIS), are still continuing to date but much less frequently than during the initial two years after filling the reservoir. The largest of the RIS events which have occurred are local magnitude M_L 2.8 microearthquakes and are characterized by free field motions of very short duration, high frequency, and high peak accelerations.

As a result of this RIS phenomena, both the Advisory Committee on Reactor Safeguards (ACRS) and the Atomic Safety and Licensing Board (ASLB) expressed concerns on the impact that these small earthquakes would have on plant equipment and components required for the shutdown and continued removal of residual heat. The ACRS concern was with the largest postulated earthquake which might occur from Reservoir Induced Seismicity. The ASLB concern dealt with the shallow RIS events recorded at Monticello Reservoir. To address the ACRS and ASLB concerns, programs were set up to develop and establish envelope spectra for both the ACRS and ASLB type of events.

For the ACRS it was determined that the earthquake could be adequately represented by a 4.5 M_L event of normal tectonic depth anchored to a zero period acceleration of .22g. This response spectrum was considered by the NRC as conservative for the necessary evaluation. Comparing the ACRS response spectrum with the SSE response spectrum and considering the dominant frequencies of the structures, it was concluded that only buildings on rock had to be evaluated for the ACRS commitment. Buildings on rock include the Reactor Building, Auxiliary Building, Control Building, and Intermediate Building. The Intermediate Building, although on caissons, is so restrained that it was considered as a building on rock. Four Oroville aftershocks of magnitude 4.5 were used at nine varied time increments to generate 36 separate time history components for the ACRS. These components were applied to the seismic models of the appropriate plant structures to generate the necessary floor response spectra. This floor response spectra was required for use in the subsequent ACRS equipment margin study for safe shutdown equipment and components.

For the ASLB program, the envelope spectra for RIS events, as recorded by the USGS accelerograph located near the site, was used as a basis to further determine the response spectra which should be applied to the specific foundation types of the nuclear plant structures (i.e. rock, caissons, soil). The development of this specific response spectra included a program of explosion test experiments and resulted in a reduced envelope spectra for the three individual foundation types on rock, caissons, or soil.

This reduced enveloped spectra, as approved by the NRC, was compared to the SSE response spectra at the foundation level. It was concluded that for buildings on rock, the SSE foundation response spectra exceeded the ASLB reduced envelope spectra, except for the frequency region higher than 16 Hz. For buildings on caissons or on soil, the SSE response spectra exceeded the ASLB reduced response spectra essentially at all frequencies. Therefore only the structures on rock were required to be evaluated for the ASLB criteria.

The ASLB reduced envelope spectra was applied to the seismic models of the four buildings on rock to generate floor response spectra curves. These floor response spectra were the bases for performing the subsequent ASLB equipment margin study for the safe shutdown equipment and components.

Because of the similarities of the ACRS and the ASLB review procedures, it was determined that the equipment margin study could be concurrently performed for both the ACRS and ASLB criteria. This study was divided into two categories; one for active components and one for passive components.

Active components are defined as those components which require actuation or moving parts to perform their function for shutdown and residual heat removal. Passive components are defined as those components which do not require actuation or moving parts to perform this same function, but are required to maintain their structural integrity.

For the active equipment margin study, separate design margins were determined for both the ACRS and ASLB. The active components which were originally qualified by tests were evaluated for margin by comparison of the ACRS and ASLB floor response spectra and the original test response spectra (TRS).

For those active components originally qualified by analyses, the original design spectrum or ZPA value was compared to the ACRS and ASLB floor response spectra to determine the resulting margin. Results of the active equipment margin study showed that all active equipment and components possessed sufficient margin in original design to equal or exceed that required by the ACRS and ASLB criteria.

Passive components were generically addressed and qualified by using inelastic response spectra and ductility demand criteria as a more appropriate and accurate technique of measuring real damage potential. Results of the analysis for passive components showed that the SSE is a relatively more severe design criterion than either the ACRS and ASLB spectrum. Maximum ductility demands were well below the minimum capacity of the passive components, proving that neither the ACRS or the ASLB criteria pose a significant seismic risk to the passive components in the station.

Thus the equipment margin studies showed that both active and passive equipment and components possessed adequate design margins to withstand the affects of any imposed reservoir induced seismicity.

The overall conclusion of the program is that the ACRS and ASLB criteria do not affect the safe shutdown and residual heat removal equipment in the plant and that no plant modifications are required. Based on these positive results, both the ACRS requirement and the ASLB License Condition 2.C[25] are satisfied with no further actions required.

Detailed input, data, and conclusions for the ACRS and ASLB seismic programs are presented in the report, "Seismic Confirmatory Program, Equipment Margin Study, V. C. Summer Nuclear Station Unit 1, OL No. NPF-12, November 1983". (Reference [26])

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 Balance of Plant Scope

Seismic analyses of the Seismic Category I structures are performed using the method of normal mode. The design response spectra or synthetic time history, described in Section 3.7.1.2, are used as the input motion.

For the Reactor Building, including the interior concrete structure, the flexibility matrix of the structure system is first formulated using the STRUDL[5] computer program (see Section 3.8.4.4). This flexibility matrix is then combined with the mass matrix of the structure system to solve for the natural frequencies and mode shapes by the Jacobian diagonalization routine.

Each individual modal response is obtained by the numerical integration method of Nigam and Jennings[6]. The total structural response is taken as the superposition of the responses of all significant modes. Significant modes for structural response are defined using the dominant modal participation factor as a base. Any mode with a modal participation factor greater than 10% of the dominant mode is considered a significant mode. Since all other modes lie within frequencies well above the rigid frequency and have small participation factors with response spectrum values much less than that of the dominant mode, they contribute less than 10% of the total response. For example, the participation factors of the Reactor Building, as shown in Table 3.7-7c, indicate that the significant modes are first through seventh mode. The frequency for the seventh mode is 45.87 cps. beyond which there is very little energy input in the time history.

Floor response spectra are calculated from the resulting floor acceleration time history using the method of Nigam and Jennings[6].

For the remaining Seismic Category I structures, except the service water intake structure, Service Water Pumphouse, and service water discharge structure, the seismic analysis is performed using the computer program DYNAL[7] (see Section 3.8.4.4). The Householder-Ortega-Wielandt method, described in Reference [7], is used in the modal analysis to obtain the natural frequencies and mode shapes for the structure system. Structural responses are obtained by superposition of modal responses of all significant modes. The time histories of structural responses at mass points of interest are used to generate the floor response spectra.

Because of the embedment condition and the underlying foundation characteristics, the Service Water Pumphouse and associated supporting foundation media are analyzed for the soil-structure interaction effect (see Section 3.7.2.4) using the computer program FLUSH[8] (see Section 3.8.4.4). This program uses the plane-strain finite element method and is capable of taking into account the three-dimensional characteristics of the soil-structure system by use of viscous boundaries on planar sides of the plane-strain model. Additional features of FLUSH are discussed in Reference [8] and Section 3.7.2.4. The analytical procedure used is two-dimensional. Material damping is included in stiffness matrices formed from complex modules of foundation media. The structural response is obtained by applying the method of complex response. This includes solving the displacements in the frequency domain through Gaussian elimination, followed by the inverse Fast Fourier Transform for the displacements in the time domain.

Typical lumped mass mathematical models used for seismic analysis are shown by Figure 3.7-9 for the Reactor Building which is founded upon rock. Figure 3.7-10 depicts a typical lumped mass mathematical model for the Intermediate Building which is supported by caissons seated in rock. The finite element model of the Service Water Pumphouse and its supporting medium is presented by Figure 3.7-11.

The mass points of a building are, as a rule, chosen at the major floor elevations of the building. In the case of the Reactor Building interior concrete structure, typical mass points are also selected at the supports of safety class equipment, such as the reactor vessel, steam generators, pressurizer, and reactor coolant pumps. The number of degrees of freedom is the same as the total number of modes. Since, for the Virgil C. Summer Nuclear Station Seismic Category I structure, the total number of modes is always greater than the number of significant modes and the significant modes contribute to more than 90% of the total response, the number of degrees of freedom selected is adequate.

For Seismic Category I structures founded upon rock (see Figure 3.7-9), the effects of foundation torsion, rocking, and translation with respect to the supporting media are represented by the effective linear foundation springs derived from the vibration of the rigid disc sitting upon an elastic half- space as described in Section 3.7.2.4. The stiffness members between floors are connected at stiffness centers, and the mass center and the stiffness center of each floor are connected by a rigid link to simulate torsional effects. The torsional spring is also attached to the bottom of the mat at the same point as the rocking and translational springs. The induced torsions, transverse shears, and bending moments at every section of the model are used in the design of the buildings. A typical torsional model is illustrated by Figure 3.7-11a. The torsional response is presented by Table 3.7-7b.

For Seismic Category I structures supported on caissons (see Figure 3.7-10), the effective stiffness of the surrounding soil medium is represented by lateral soil springs attached to the caissons in addition to the previously discussed foundation springs. The lateral soil springs are calculated in accordance with the method proposed by Penzien[9] (see Section 3.7.2.4).

The concrete structure element is assumed to have linear elastic properties.

3.7.2.1.2 Components and Equipment Provided by the NSSS Vendor

Those Seismic Category I components and systems that must remain functional in the event of the SSE are identified by applying the criteria of Section 3.2.1.

In general, the dynamic analyses are performed using a modal analysis plus either the response spectrum analysis or integration of the uncoupled modal equations as described in Sections 3.7.2.1.2.3 and 3.7.2.1.2.4 respectively, or by direct integration of the coupled differential equations of motion described in Section 3.7.2.1.2.5.

3.7.2.1.2.1 Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dashpots suitable for mathematical analysis. The essence of this step is to select a model so that the displacements obtained will be a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure or component. Some typical modeling techniques are presented in Reference [10].

Equations of motion consider the multi-degree-of-freedom system shown in Figure 3.7-12. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r \ddot{y}_r + \sum^{i} c_{ri} u_i + \sum^{i} k_{ri} \dot{u}_i = 0 \qquad (3.7\text{-}1)$$ RN 01-113

Where:

m_r = the value of the mass or mass moment of rotational inertia at mass point r.

$\ddot{y}_r$ = absolute translational or angular acceleration of mass point r.

c_{ri} = damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i, maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity.

$\dot{u}_i$ = translational or angular velocity of mass point i relative to the base.

k_{ri} = stiffness coefficient - the external force (moment) required at mass point r to produce a unit deflection (rotation) at mass point i, maintaining zero displacement (rotation) at all other mass points.

Force (moment) is positive in the direction of the displacement (rotation). RN 01-113

u_i = displacement (rotation) of mass point i relative to the base.

Note that Figure 3.7-12 does not attempt to show all of the springs (and none of the dashpots) which are represented in Equation (3.7-1).

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad (3.7\text{-}2)$$

Where:

$\ddot{y}_s$ = absolute translational (angular) acceleration of the base.

$\ddot{u}_r$ = translational (angular) acceleration of mass point r relative to the base.

Equation (3.7-1) can be written as:

$$m_r\,\ddot{u}_r + \sum^{i} c_{ri}\,\dot{u}_i + \sum^{i} k_{ri}\,\dot{u}_i = m_r\ddot{y}_s \qquad (3.7\text{-}3)$$

RN 01-113

For a single degree of freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad (3.7\text{-}4)$$

3.7.2.1.2.2 Modal Analysis

1. Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which were determined by eigen solution of Equation (3.7-3). The right hand side and the damping term are set equal to zero for this purpose as illustrated in Reference [11] (pp. 83 through 111). Thus, Equation (3.7-3) becomes:

$$m_r\ddot{u}_r + \sum^{i} k_{ri}\, u_{\cdot i} = 0 \qquad (3.7\text{-}5)$$ RN 01-113

The equation given for each mass point r in Equation (3.7-5) can be written as a system of equations in matrix form as:

$$[M]\ \{\ddot{\Delta}\} + [K]\ \{\Delta\} = 0 \qquad (3.7\text{-}6)$$

Where:

[M] = mass and rotational inertia matrix.

$\{\Delta\}$ = column matrix of the general displacement and rotation at each mass point relative to the base.

[K] = square stiffness matrix.

$\{\ddot{\Delta}\}$ = column matrix of general translational and angular accelerations at each mass point relative to the base, $d^2\ \{\Delta\}/dt^2$.

Harmonic motion is assumed and the $\{\Delta\}$ is expressed as:

$$\{\Delta\} = \{\delta\} \sin \omega t \qquad (3.7\text{-}7)$$ RN 01-113

Where:

$\{\delta\}$ = column matrix of the spatial displacement and rotation at each mass point relative to the base.

ω = natural frequency of harmonic motion in radians per second. RN 01-113

The displacement function and its second derivative are substituted into Equation (3.7-6) and yield:

$$[K]\{\delta\} = \omega^2 [M]\{\delta\} \qquad (3.7\text{-}8)$$ RN 01-113

The determinant $[K] - \omega^2 [M]$ is set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation (3.7-8). This yields n natural frequencies and mode shapes where n equals the number of dynamic degrees of freedom of the system. The mode shapes are all orthogonal to each other and are sometimes referred to as normal mode vibrations. For a single degree of freedom system, the stiffness matrix and mass matrix are single terms and the determinant $[K] - \omega^2 [M]$ when set equal to zero yields simply: RN 01-113

$$k - \omega^2 m = 0$$ RN 01-113

or:

$$\omega = \sqrt{\frac{k}{m}} \qquad (3.7\text{-}9)$$ RN 01-113

where ω is the natural angular frequency in radians per second. The natural frequency in cycles per second is therefore RN 01-113

$$f = \frac{1}{2\pi}\sqrt{\frac{k}{m}} \qquad (3.7\text{-}10)$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation (3.7-8), however, only n-1 of these equations are independent. This means that the elements of $\{\delta\}$ can be expressed only as multiples of one another. Normalizing $\{\delta\}$ such that the maximum displacement (rotation) of any element is unity gives: RN 01-113

ϕ_{rn} = displacement (rotation) of mass point r in mode n relative to the base.

2. Modal Equations

The response of a structure or component is always some combination of its normal modes. However, good accuracy can be obtained by using only the first few modes of vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as (Reference [11], pp. 116-125):

$$\ddot{A}_n + 2\omega_n p_n \dot{A}_n + W_n^2 A_n = -\Gamma_n \ddot{y}_s \qquad (3.7\text{-}11)$$ RN 01-113

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u_{rn}, by the equation:

$$u_{rn} = A_n \Phi_{rn} \qquad (3.7\text{-}12)$$ RN 01-113

Where:

ω_n = natural frequency of mode n in radians per second. RN 01-113

p_n = critical damping ratio of mode n.

Γ_n = modal participation factor of mode n given by:

$$\Gamma_n = \frac{\sum^n m_r \phi'_{rn}}{\sum^n m_r \phi_{rn}^2} \qquad (3.7\text{-}13)$$ RN 01-113

and:

ϕ'_{rn} = value of ϕ_{rn} in the direction of the earthquake. RN 01-113

The essence of the modal analysis lies in the fact that Equation (3.7-11) is analogous to the equation of motion for a single degree of freedom system that will be developed from Equation (3.7-4). Dividing Equation (3.7-4) by m gives:

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \qquad (3.7\text{-}14)$$ RN 01-113

The critical damping ratio of a single degree of freedom system, p, is defined by the equation: RN 01-113

$$p = \frac{c}{c_c} \qquad (3.7\text{-}15)$$

where the critical damping coefficient is given by the expression:

$$c_c = 2m\omega \qquad (3.7\text{-}16)$$ RN 01-113

Substituting Equation (3.7-16) into Equation (3.7-15) and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \qquad (3.7\text{-}17)$$ RN 01-113

Substituting this expression and the expression for k/m given by Equation (3.7-9) into Equation (3.7-14) gives:

$$\ddot{u} + 2\omega p\dot{u} + \omega^2 u = -\ddot{y}_s \qquad (3.7\text{-}18)$$ RN 01-113

Note the similarity of Equations (3.7-11) and (3.7-18). Thus each mode may be analyzed as though it were a single degree of freedom system and all modes are independent of each other. By this method a fraction of critical damping, i.e., c/cc, may be assigned to each mode and it is not necessary to identify or evaluate individual damping coefficients, i.e., c. However, assigning only a single damping ratio to each mode has a drawback. Normally, there are two ways used to overcome this limitation when considering a slightly damped structure supported by a massive moderately damped structure.

The first method is to develop and analyze separate mathematical models for both structures using their respective damping values. The massive moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures is used as a forcing function for the subsequent detailed analysis. The second method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion.

3.7.2.1.2.3 Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (Reference [12], pp. 24-51) (displacement, velocity, and acceleration) of a single-degree-of-freedom system versus its natural frequency of vibration when subjected to a time history motion of its base. Examples of response spectra are shown in Figures 3.7-13 and 3.7-14.

The response spectrum concept can best be explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree of freedom system with a given damping and natural frequency be calculated for a given base motion. The variations in response are established and the maximum absolute value of each response is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency in sufficient detail to establish the complete curve. Other curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis, and the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with that base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$S_{a_n} = \omega_n S_{v_n} = \omega_n^2 S_{d_n} \qquad (3.7\text{-}19)$$ RN 01-113

There are two types of response spectra that must be considered. If a given building is shown to be rigid and to have a hard foundation, the ground response spectrum or ground time history is used. It is referred to as a ground response spectrum. If the building is flexible and/or has a soft foundation, the ground response spectra are modified to include these effects. The response spectrum at various support points must be developed. This is called a floor response spectrum. The specific response spectra used are discussed in Sections 3.7.1 and 3.7.2.5.

3.7.2.1.2.4 Integration of Modal Equations

This method can be separated into the following two basic steps:

1. Integration procedure for the uncoupled modal Equation (3.7-11) to obtain the modal displacements and accelerations as a function of time.

2. These modal displacements and accelerations are combined to obtain the total displacements, accelerations, forces, and stresses.

Integration of these uncoupled modal equations is done by step-by-step numerical integration. The step-by-step numerical integration procedure[13] consists of selecting a suitable time interval, Δt, and calculating modal acceleration, $\ddot{A}_n$, modal velocity, $\dot{A}_n$, and modal displacement, A_n, at discrete time stations Δt apart, starting at t = 0 and continuing through the range of interest for a given time history of base acceleration. RN 01-113

From the modal displacements and accelerations, the total displacements, accelerations, forces, and stresses can be determined as follows:

1. Displacement of mass point r in mode n as a function of time is given by Equation (3.7-12) as:

$$u_{rn} = A_n \ \phi_{rn} \tag{3.7-20}$$

with the corresponding acceleration of mass point r in mode n as:

$$\ddot{u}_{rn} = \ddot{A}_n \phi_{rn} \tag{3.7-21}$$

2. The displacement and acceleration values obtained for the various modes are superimposed algebraically to give the total displacement and acceleration at each time interval.

3. The total acceleration at each time interval is multiplied by the mass to give an equivalent static force. Stresses are calculated by applying these forces to the model. Alternatively these stresses may be determined from the deflections at each time interval.

3.7.2.1.2.5 Integration of Coupled Equations of Motion

The dynamic transient analysis is a time history solution of the response of a given structure to known forces and/or displacement forcing functions. The structure may include linear or nonlinear elements, gaps, interfaces, plastic elements, and viscous and Coulomb dampers. Nodal displacements, nodal forces, pressure, and/or temperatures may be considered as forcing functions. Nodal displacements and elemental stresses for the complete structure are calculated as functions of time.

The basic equations for the dynamic analysis are as follows:

$$[M]\{\ddot{x}\}+[C]\{\dot{x}\}+[K]\{x\}=\{F(t)\} \tag{3.7-22}$$ RN 01-113

where the terms are as defined earlier and {F(t)} and may include the effects of applied displacements, forces, pressures, temperatures, or nonlinear effects such as plasticity and dynamic elements with gaps. Options of translational accelerations input to a structural system and the inclusion of static deformation and/or preload may be considered in the nonlinear dynamic transient analysis. The option of translational input such as uniform base motion to a structural system is considered by introducing an inertia force term of -[M] $\{\ddot{z}\}$ to the right hand side of the basic Equation (3.7-22), i.e.,

$$[M]\{\ddot{x}\}+[C]\{\dot{x}\}+[K]\{x\}=\{F\}-[M]\{\ddot{z}\} \tag{3.7-23}$$ RN 01-113

The vector $\{\ddot{z}\}$ is defined by its components $\{\ddot{z}_i\}$ where i refers to each degree of freedom of the system. $\{\ddot{z}_i\}$ is equal to a_1, a_2, or a_3 if the i(th) degree of freedom is aligned with the direction of the system translational acceleration a_1, a_2, or a_3, respectively. $\{\ddot{z}_i\}$ = 0 if the i(th) degree of freedom is not aligned with any direction of the system translational acceleration. Typical application of this option is a structural system subjected to a seismic excitation of a given ground acceleration record. The displacement {x} obtained from the solution of Equation (3.7-23) is the displacement relative to the ground.

The option of the inclusion of initial static deformation or preload in a nonlinear transient dynamic structural analysis is considered by solving the static problem prior to the dynamic analysis. At each stage of integration in transient analysis, the portion of internal forces due to static deformation is always balanced by the portion of the forces which are statically applied. Hence, only the portion of the forces which deviate from the static loads will produce dynamic effects. The output of this analysis is the total result due to static and dynamic applied loads.

One available method for the numerical integration of Equations (3.7-22) and (3.7-23) is the third order (cubic) integration scheme. In the third-order (cubic) integration scheme values of {x} are assumed to be a cubic function of time over a small time increment, i.e.,

$$\{x_t\} = \{a\} + \{b\}t + \{c\}t^2 + \{d\}t^3 \qquad (3.7\text{-}24)$$

The velocity and acceleration vectors are found by differentiating Equation (3.7-24) with respect to time:

$$\{\dot{x}_t\} = \{b\} + 2\{c\}t + 3\{d\}t^2 \qquad (3.7\text{-}25)$$

$$\{\ddot{x}_t\} = 2\{c\} + 6\{d\}t \qquad (3.7\text{-}26)$$

The unknown coefficients {a}, {b}, {c}, and {d} can be obtained from Equation (3.7-24) in terms of the displacements at time t, t-1, t-2, and t- 3 (i.e., present and three previous values of displacements). Thus for each time interval, the velocity and acceleration arrays may be expressed by:

$$\{\dot{x}_t\} = f_1(\{x_t\},\{x_{t-1}\},\{x_{t-2}\},\{x_{t-3}\}) \qquad (3.7\text{-}27)$$

$$\{\ddot{x}_t\} = f_2(\{x_t\},\{x_{t-1}\},\{x_{t-2}\},\{x_{t-3}\}) \qquad (3.7\text{-}28)$$

Substituting Equations (3.7-27) and (3.7-28) into Equation (3.7-22) and solving for the present value of displacement vector gives:

$$\frac{C_1}{(\Delta t)^2}[M]+\frac{C_2}{(\Delta t)}[C]+[K]\{x_t\}=\{F_t\}+$$

$$f([M],[C],\{x_{t-1}\},\{x_{t-2}\},\{x_{t-3}\}) \qquad (3.7\text{-}29)$$

The above set of simultaneous linear equations is solved to obtain the present values of nodal displacements $\{x_t\}$ in terms of the previous (known) values of the nodal displacements. Since {M}, {C}, and {K} are included in the equation, they can also be time or displacement dependent. RN 01-113

3.7.2.2 Natural Frequencies and Response Loads

A summary of significant frequencies and mode shapes for the representative Seismic Category I structures is presented by Figures 3.7-15 through 3.7-20 for the Reactor Building, Control Building, Auxiliary Building, Intermediate Building, Fuel Handling Building, and Diesel Generator Building, respectively.

Floor response spectra at critical Seismic Category I structure elevations and points of support are presented for the OBE by Figures 3.7-21 through 3.7-27 for the Reactor Building, Figures 3.7-28 and 3.7-29 for the Control Building, Figures 3.7-30 through 3.7-32 for the Auxiliary Building, Figures 3.7-33 and 3.7-34 for the Intermediate Building, Figures 3.7-35 and 3.7-36 for the Fuel Handling Building, Figures 3.7-37 through 3.7-39 for the Diesel Generator Building and Figures 3.7-40 and 3.7-41 for the Service Water Pumphouse. These floor response spectra are used for the seismic qualification of Seismic Category I safety class equipment and components as described in Section 3.7.3. Response loads (displacements, accelerations and masses) are presented by Table 3.7-7a for the OBE.

In Table 3.7-7a and Figures 3.7-21 through 3.7-41, the x component corresponds to the plant East-West direction, the y component corresponds to the plant North-South direction and the z component corresponds to the plant vertical direction. One exception occurs in the case of the service water pumphouse where the x component is perpendicular to the shoreline and the y component is parallel to the shoreline. Both the actual, narrow and artificially broadened response spectra are drawn.

Since the artificial time history has frequency content around 20 cps and between 24 and 30 cps, as shown by Figure 3.7-5, and the building has a horizontal natural frequency at 29.2 cps, as shown by Figure 3.7-15, the secondary peak at 29 cps is higher than the spectrum value at 20 cps. However, the spectrum value at 20 cps is still higher than the maximum floor acceleration of 0.18g. For the vertical earthquake, there is a natural frequency at 22.5 cps. This natural frequency and the frequency content of the artificial time history caused the secondary peaks at 21 and 26 cps.

The SSE response spectrum envelope accelerations have been calculated by scaling the OBE response spectrum envelope accelerations. Since the SSE structured damping is conservatively assumed to be the same as the OBE structural damping, the SSE and OBE response ratio is proportional to the input acceleration value. For buildings on rock, the ratio of SSE to OBE acceleration is 1.5. For buildings on soil, the ratio is 1.67. For buildings on caissons the ratio varies from 1.62 to 1.55 as described by Figure 3.7-43. Scale factors for each of the buildings are as follows:

1. Reactor Building, 1.5 OBE.

2. Auxiliary Building, 1.55 OBE.

3. Control Building, 1.55 OBE.

4. Fuel Handling Building, 1.62 OBE.

5. Intermediate Building, 1.55 OBE.

6. Diesel Generator Building, 1.62 OBE.

7. Service Water Pumphouse, 1.67 OBE.

To properly account for amplification due to flexure of floor slabs, a scaling factor, Υ, has been applied to the response spectrum envelopes for the vertical direction. The magnitude of the scaling factor depends upon the location of equipment, components, or systems on a particular building floor and is discussed in Section 3.7.3.

The maximum floor accelerations at the major building floor elevations and equipment supports correspond to the acceleration at the high frequency ends of the associated floor response spectra.

3.7.2.3 Procedure Used for Modeling

In the seismic analysis, Seismic Category I structures which form a soil-structure interaction system with foundation media are defined as "seismic systems." Except for the Service Water Pumphouse, as mentioned in Section 3.7.2.2, these structures are simulated by lumped mass mathematical models. Other Seismic Category I structures, components, and equipment, defined as "seismic subsystems," are decoupled from the "seismic system" in the mathematical models. The general uncoupling criterion is based upon the ratio of mass of the supported subsystem to that of the supporting system. If the total building mass is used as the supporting mass, the mass ratio is always small. However, if the modal mass is used as the supporting mass, the mass ratio can be greater than 0.01. Based upon the theory of random vibration, the effect of a large mass ratio is a decrease in the mean square responses of the supported subsystem and the supporting system (see Figure 2.12 and 2.13 of Reference [25]). In the uncoupled analysis, the subsystem is analyzed using a floor response spectrum which is obtained by assuming that the subsystem is rigidly attached to the supporting system. This assumption of rigid attachment is equivalent to the zero mass ratio case discussed in Reference [25]. Since the zero mass ratio case yields the highest subsystem response, the uncoupled analysis is always conservative.

The effects of such decoupling upon the dynamic response of the Reactor Building have been investigated in detail. Both the decoupled case and the case where the heavy equipment is incorporated into the "seismic system" mathematical model have been examined. It has been found that, for the interior concrete structure considered, decoupling of the heavy equipment from the supporting structure adds only slight additional conservatism to the structural response and is therefore justified. The criteria used for lumping masses is as described in Section 3.7.2.1.

Dynamic lumped mass models are constructed so that three-dimensional responses can be obtained. For the finite element model of the Service Water Pumphouse supporting medium, two separate, othogonal plane strain models are constructed. Three directions of earthquake ground motion are input to the dynamic models, one at a time, for the seismic analysis, including floor response spectra generation. The spatial combination of such responses is discussed in Section 3.7.2.6.

For analysis of NSSS supplied equipment, primary importance is given to the Reactor Coolant System. The analysis of this system can be performed using several different methods depending upon the level of seismic activity at the plant site. The possible methods include:

1. Linear modal analysis of the primary loop piping and components.

2. Coupled building/loop linear modal analysis.

3. Coupled building/loop non-linear time-history analysis.

Methods (2) and (3) are considered to be "seismic systems" in accordance with the guidance noted in Standard Review Plan 3.7.3. Method (1) and all other systems and components within the NSSS Vendor’s scope F, except primary loop piping, are 02-01
analyzed independently and are classified as "seismic subsystems." Examples of these are the primary system components, auxiliary pumps, branch piping, tanks, etc.

Method (3) was used for the seismic analysis of the Virgil C. Summer Nuclear Station 02-01
Reactor Coolant Loop Piping System. All other analyses of NSSS vendor supplied equipment were performed as "seismic subsystem" analyses.

3.7.2.4 Soil-Structure Interaction

The effect of soil-structure interaction on the seismic response of lumped mass models is represented by six equivalent linear foundation springs: three translational; two rocking; and one torsional (see Section 3.7.2.1).

The validity of the foundation spring method used for the foundation-structure configuration has been investigated. This investigation is summarized as follows:

1. The Seismic Category I structures are supported either on rock supported fill concrete or caissons seated in rock. Therefore, the layering effect is insignificant.

2. Since the plant site has sufficiently deep and uniform overburden, frequency independent foundation springs are considered to be adequate for simulation of the soil-structure interaction phenomenon[14].

3. The effect of embedment depth (Section 3.7.1.4) on the foundation spring values has been considered and, for reasons of conservatism, is ignored.

4. For the Seismic Category I structures supported on caissons, the following considerations have been further investigated through modeling:

   a. The interaction between the caisson (and, if any, the underground portion of the structure) and the surrounding soil medium is accounted for by connecting effective lateral soil springs to the lumped mass of the caisson (and, if any, the underground portion of the structure). The soil springs are calculated based upon the work by Penzien[9] on pile foundations in which Midlin's half-space formulas for the force-displacement relationship are utilized.

      The amplified free-field seismic motion at each elevation of interest in the soil medium has been considered. As shown by Figure 3.7-42, the acceleration time histories, A_1 (t), A_2 (t), . . . A_i (t), represent the amplified motions at the elevations of interest. However, in the lumped mass approach considered, a single acceleration time history with equivalent energy input has been conservatively estimated and applied throughout the elevation of the caisson as shown by Figure 3.7-43.

   b. Settlement of the soil underneath the mat is considered.

   c. The stress levels in the soil at the junction of the underground structures and the lateral soil medium are compared with the corresponding bearing capacity to check the stability of the entire soil-structure system.

   d. Caissons are drilled into competent rock for end bearing and frictional load transfer. Rock properties relative to caisson design and construction are described in Section 2.5.4.

e. The pertinent parameters used in the parametric study of soil-structure interaction are as follows:

   (1) Soil shear modulus (G) is 1.0×10^4 to 3.5×10^4 psi. This range is based upon both in-situ dynamic wave tests and laboratory triaxial tests.

   (2) Dynamic concrete Young's modulus ($E_{dynamic}$) is 1.0 E_{static} to 2.0 E_{static}.

   (3) Settlement of soil underneath the mat: soil takes 0 to 50% of the vertical load.

5. The foundation mat of the caisson supported Intermediate Building is rigidly connected to the Control Building foundation mat for purposes of lateral stability. For such a structure-foundation configuration, both foundation soil springs and caisson soil springs are used in the combined dynamic lumped mass model of the two buildings.

As mentioned in Section 3.7.2.1, the Service Water Pumphouse is analyzed for the soil-structure interaction effect by using the finite element program, FLUSH[8]. Approximate strain dependent shear modules of the foundation medium are first estimated for the expected level of ground motion and surface discharge pressure using the one-dimensional wave propagation theory. Final strain and frequency dependent soil properties and damping ratio are obtained through an iterative sequence of dynamic response calculations for the soil-structure system. The soil property curves presented in Section 2.5 are used.

It is considered that motions in the vicinity of the structure are due to vertical propagation of body waves from underlying, stiffer formations. Accordingly, the basic method for the dynamic response calculation can be illustrated as follows. A control motion compatible with the design response spectra, as presented in Section 3.7.2.1, is first specified. This control motion is then used as input to a finite element model of the soil-structure system and the response is computed at elevations and points of interest.

The computer program FLUSH[8] is also equipped with a transmitting boundary, simulating sufficient extent of the half-space. Furthermore, with use of a viscous boundary on the planar side of the plane-strain finite element model, three dimensional wave propagation phenomena of the foundation medium are taken into account.

3.7.2.5 Development of Floor Response Spectra

Floor response spectra are developed by applying the time history earthquake motion to the multi-degree of freedom models of the Seismic Category I structures with the various mass points representing the described elevation in each structure (see Section 3.7.2.1). Such floor response spectra take into consideration the effect of the three components of earthquake motion and reflect predominant response near the dominant frequencies of the structure.

Due to the general asymmetry of the configuration of the Seismic Category I structures, the responses of the structures are normally three-directional in nature even when subjected to only one component of earthquake motion.

To properly estimate floor response spectrum values, including the three-directional effects of both the earthquake (which would not be likely to produce maximum responses in all three directions simultaneously) and the structural behavior, the following equation is used:

$$S_x = (S_{x1}^2 + S_{x2}^2 + S_{x3}^2)^{1/2}$$

Where:

S_x = the final spectrum value at any frequency point in the x direction.

S_{xk} = the spectrum value at the same frequency point in the x direction due to earthquake component in the k(th) direction (k = 1, 2, 3).

In the development of vertical floor response spectra, additional effects of vertical amplification other than those due to the overall structural response are considered. These are the amplification due to floor flexibility and rocking motion of the structure.

With the inclusion of the floor flexibility in the vertical response calculation, additional response spectrum peaks are determined at the characteristic frequencies of the floor. To account for the rocking motion of the Auxiliary, Control, and Intermediate Buildings, 98-01
several nodal points are selected at the corners of the floor with which walls usually intersect. These corner points are connected to the lumped mass of the floor by rigid links as shown in the torsional model in Fig. 3.7-11a. Additional amplifications are RN 01-113
observed in the floor response spectrum. The final floor response spectrum is constructed to envelop the spectrum peaks attributable to the overall vertical structural response and the floor flexibility, as well as to the rocking motion of the structure.

The effect of torsional response on horizontal floor response spectra, due primarily to the eccentricity of the structural configuration, is discussed in Section 3.7.2.11.

3.7.2.6 <u>Three Components of Earthquake Motion</u>

3.7.2.6.1 Balance of Plant Scope

The time history method is employed in the seismic analysis of Seismic Category I structures. The maximum responses (e.g., accelerations) due to each of the three components of earthquake motion are first calculated separately at a particular point of a structure or of the corresponding mathematical model. These component responses are then combined by taking the square root of the sum of the squares of maximum codirectional responses caused by each of the three components of earthquake motion (see Section 3.7.2.5). This procedure is in conformance with Regulatory Guide 1.92, Revision 1.

3.7.2.6.2 Components and Equipment Provided by the NSSS Vendor

The seismic design of the piping and equipment includes the effect of the seismic response of the supports, equipment, structures, and components. The system and equipment response is determined using three earthquake components, two horizontal and one vertical. The design ground response spectra, specified in Section 3.7.1, are the bases for generating these three input components. Floor response spectra are generated for two perpendicular horizontal directions, (i.e., N-S and E-W) and the vertical direction. System and equipment analysis is performed with these input components applied in the N-S, E-W, and vertical directions. The damping values used in the analysis are those given in Table 3.7-3.

In computing the system and equipment response by response spectrum modal analysis the methods of Section 3.7.3.7 are used to combine all significant modal responses to obtain the combined unidirectional responses.

The combined total response is then calculated using the square root of the sum of the squares formula applied to the resultant unidirectional responses. For instance, for each item of interest such as displacement, force, stresses, etc., the total response is obtained by applying the above described method. The mathematical expression for this method (with R as the item of interest) is:

$$R_C = [\sum_{T=1}^{3} R_T^{\ 2}]^{1/2} \qquad (3.7\text{-}30)$$

Where:

$$R_T = [\sum_{i=1}^{N} R_{Ti}^{\ 2}]^{1/2} \qquad (3.7\text{-}31)$$

and:

R_C = total combined response at a point.

R_T = value of combined response of direction T.

R_{Ti} = absolute value of response for direction T, mode i.

N = total number of modes considered.

The subscripts can be reversed without changing the results of the combination.

For the case of closely spaced modes, R_T in Equation (3.7-31) above is replaced with R_T as given by Equation (3.7-32) in Section 3.7.3.7, where the criteria and justification for meeting the intent of Regulatory Guide 1.92, Revision 1 are presented. RN 01-113

3.7.2.7 Combination of Modal Responses

3.7.2.7.1 Balance of Plant Scope

Since only the time history method is employed in the seismic (system) analysis of Seismic Category I structures, modal responses are algebraically combined, in the time domain, in the solution following the principle of superposition. Regulatory Guide 1.92, Revision 1, is, therefore, not applicable.

3.7.2.7.2 Components and Equipment Provided by the NSSS Vendor

Conformance with the recommendations of Regulatory Guide 1.92, Revision 1 for combination of modal responses is presented in Section 3.7.3.7.

3.7.2.8 Interaction of Noncategory I Structures with Seismic Category I Structures

Complete separation of Seismic Category I structures from adjacent Seismic Category I or non-Seismic Category I structures prevents interaction with or impact from adjacent structures. Seismic effects on non-Seismic Category I structures are investigated to prevent the possibility of damage to Seismic Category I structures due to possible collapse of non-Seismic Category I structures.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The peak width and period coordinates of the floor response spectra (see Section 3.7.2.5) are determined by parametric study based upon the variations of the soil springs and/or material properties of the structure and foundation. In any case, the peak width is assigned a minimum range of ±10% of the center frequency. For cases where the variation of calculated periods due to the various assumptions regarding material properties and soil-structure interaction change the spectrum values, the parametric study, coupled with the enveloping process, is used to conservatively estimate the floor response spectrum values.

3.7.2.10 Use of Constant Vertical Static Factors

Dynamic analyses with vertical earthquake motion are performed instead of using constant vertical load factors.

3.7.2.11 Methods Used to Account for Torsional Effects

A mathematical model with a rigid link connecting the center of mass and center of rigidity at each floor elevation is used to calculate the actual torsional responses. For typical asymmetric structures, such as the Intermediate Building, each floor in the model is taken as a rigid diaphragm with three translational and one torsional degree of freedom. Responses, including the amplification effect of the corner nodal points, as presented in Section 3.7.2.5, are obtained separately for the vertical and horizontal excitations. The combined response is obtained in the manner discussed in Section 3.7.2.6.

3.7.2.12 Comparison of Responses

Since the time history method is used throughout for seismic system analyses, no comparison of responses is needed.

3.7.2.13 Methods for Seismic Analysis of Dams

Methods for seismic analysis of dams are presented in Section 2.5.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

Overturning moments for Seismic Category I structures are determined at the base of the Seismic Category I structures. Each of the three components of earthquake excitation is considered separately. The resultant overturning moment is obtained by combining the three earthquake components in accordance with Section 3.7.2.6. The vertical earthquake component is viewed as reducing the dead weight of the structure in counteracting the overturning moment. Soil reaction is calculated by adding to, or subtracting (whichever controls) the vertical earthquake component from, the dead weight and other loads on the structure. Safety factors of 1.1 and 1.5 are provided against overturning of Seismic Category I structures due to the SSE or OBE, respectively, combined with other appropriate design loads.

3.7.2.15 Analysis Procedure for Damping

3.7.2.15.1 Balance of Plant Scope

A classical modal analysis is applied to the lumped mass soil-structure interaction system. In the modal analysis, different modal damping is specified for each mode according to the concept of weighted average damping[15, 16]. In this approach a higher damping value is specified for a mode in which soil deformation is predominant compared to structural deformation. The basic equation for the weighted average damping is as follows:

$$D_n = \frac{\sum_{i=1}^{N} \left(D_{H,i} + \frac{\omega_n}{\omega_i} D_{v,i}\right) E_{i,n}}{\sum_{i=1}^{N} E_{i,n}}$$ RN 01-113

Where:

D_n = Weighted modal damping for the n(th) mode.

i = 1, 2, 3, . . . , S, S+1, ... , N.

S = The degree of freedom of the structure for which damping values are assigned.

N-S = The degree of freedom representing the soil springs.

$D_{H,i}$ = Hysteretic damping in the i(th) degree of freedom of the soil-structure interaction system.

$D_{v,i}$ = Viscous damping in the i(th) degree of freedom of the system.

ω_n = Frequency of n(th) mode. RN 01-113

ω_i = Frequency at which $D_{v,i}$ is defined.

$E_{i,n}$ = Energy stored in i(th) degree of freedom of the system in n(th) mode.

Damping in structural elements (of superstructure and caissons) generally is hysteretic in nature which shows frequency independent energy loss per cycle. The internal damping in soil is also hysteretic. Radiation damping due to wave propagation from a structure into the soil medium is viscous damping which shows frequency dependent energy loss per cycle. When a structure is founded upon caissons, the overall damping in rocking is primarily hysteretic. Theoretically, radiation damping in swaying is generally very large. However, such radiation damping is set to zero for conservatism. Typical choices for the damping ratios (see Section 3.7.1.3) are as follows:

1. For $i \leq s$

   a. $D_{H,i}$ = (see Table 3.7-1)

   b. $D_{v,i} = 0$

2. For Rocking Mode Soil Spring

   a. $D_{H,i} = 5\%$

   b. $D_{v,i} = 0$

3. For Swaying Mode Soil Spring

   a. $D_{H,i} = 10\%$

   b. $D_{v,i} = 0$

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3.7.2.15.2 Components and Equipment Provided by the NSSS Vendor

For components and equipment provided by Westinghouse, either the lowest damping value associated with the elements of the system is used for all modes, or an equivalent modal damping value is determined according to the energy distribution in each mode. Testing programs for damping were done for the reactor coolant loop[3]. 02-01

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

3.7.3.1 Seismic Analysis Methods

3.7.3.1.1 Balance of Plant Scope

Seismic analysis is performed for those subsystems that can be modeled to correctly predict the seismic response. A component is modeled as a multi-degree-of-freedom, lumped mass system with mass free interconnections and sufficient mass points to ensure adequate representation. The resulting system is analyzed using the response spectrum modal analysis technique. An alternative time history method may also be applied. The time history method, when used, conservatively simulates the response spectrum envelope of interest. A stress analysis is then performed using the inertia forces or equivalent static loads obtained from the dynamic analysis. Moments, shears, accelerations, deflections and stresses are calculated on a mode by mode basis. The total seismic response is obtained by combining each modal response using the square root of the sum of the squares method. The absolute sum of the responses is considered for closely spaced, in phase modes as set forth in Section 3.7.3.7. In cases for which some dynamic degrees of freedom do not contribute to the total response, kinematic condensation is employed in the analysis.

3.7.3.1.2 Components and Equipment Provided by the NSSS Vendor

Seismic analysis methods for subsystems within Westinghouse scope of responsibility are given in Section 3.7.2.1.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Balance of Plant Scope

One SSE plus five OBE events are assumed to occur during the life of the plant. Ten (10) maximum stress cycles are assumed to occur during each event.

3.7.3.2.2 Components and Equipment Provided by the NSSS Vendor

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

The OBE is conservatively assumed to occur 20 times over the life of the plant. The number of maximum stress cycles for each occurrence depends on the system and component damping values, complexity of the system and component, duration, and frequency content of the input earthquake. A precise determination of the number of maximum stress cycles can only be made using time history analysis for each item. Instead, a time history study has been conducted to arrive at a realistic number of maximum stress cycles.

To determine the conservative equivalent number of cycles of maximum stress associated with each occurrence, an evaluation was performed considering both equipment and its supporting building structure as single-degree-of-freedom systems. The natural frequencies of the building and the equipment are conservatively chosen to coincide. The damping in the equipment and building are equivalent to the damping values in Table 3.7-3.

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

This study was conservative since it was performed with single-degree-of-freedom models which tend to produce a more uniform and unattenuated response than a complex interacted system. The conclusions indicate that 10 maximum stress cycles for flexible equipment (natural frequencies less than 33 Hz) and 5 maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz) for each of 20 OBE occurrences should be used for fatigue evaluation.

3.7.3.3 Procedure Used for Modeling

Equipment within the balance of plant scope is modeled as a series of discrete mass points, connected by mass free members, having sufficient mass points to ensure adequate representation of dynamic behavior. Detailed modeling of piping systems is described in Section 3.7.3.8.

Procedures used for modeling the equipment and components provided by the NSSS vendor are described in Section 3.7.2.1.2.

3.7.3.4 Basis for Selection of Frequencies

3.7.3.4.1 Balance of Plant Scope

Where applicable for analysis of equipment, modes selected as significant include the following:

1. Natural frequencies less than 30 Hz.

2. Frequencies above 30 Hz with modal participation factors greater than 10% of the fundamental mode participation factor.

For piping systems where a detailed seismic analysis is performed, all modes with natural frequencies of less than 30 Hz are included in the response calculation.

For piping systems where the simplified seismic support spacing method is employed, the limits of seismic support spans are established under the condition that the fundamental frequencies are higher than those associated with the dominating peak of the floor response spectrum.

3.7.3.4.2 Equipment and Components Provided by the NSSS Vendor

The analysis of the equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports based upon the mass and stiffness characteristics of the system, will determine the seismic accelerations which the equipment must withstand.

Three (3) ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment

support. The equipment acceleration value in this case corresponds to the low-period region of the floor response spectra.

2. If the equipment is very flexible relative to the structure, the equipment will show very little response.

3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

In the above cases, equipment under earthquake loadings is designed to be within code allowable stresses.

Also, as noted in Section 3.7.3.2.2, rigid equipment/support systems have natural frequencies greater than 33 Hz.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

3.7.3.5.1 Balance of Plant Scope

If the fundamental frequency of the component is greater than 30 Hz, the component is analyzed statically. The equivalent static forces are obtained by multiplying the lumped mass of each mass point by the appropriate maximum floor acceleration. The maximum floor acceleration is obtained from the response spectra envelope at the high frequencies.

3.7.3.5.2 Equipment and Components Provided by the NSSS Vendor

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 single-degree-of-freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree-of-freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50% to account conservatively for the increased modal participation.

3.7.3.6 Three Components of Earthquake Motion

For the balance of plant scope the responses to the two horizontal and the vertical component seismic inputs are calculated separately for the entire subsystem. The maximum value of a particular response due to simultaneous action of three components of earthquake were obtained by taking the square root of the sum of the squares of corresponding maximum response values to each of the three components calculated separately. This procedure is in conformance with Regulatory Guide 1.92.

For components and equipment provided by the NSSS vendor, methods used to account for three components of earthquake motion are given in Section 3.7.2.6.2.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Balance of Plant Scope

The combination of modal responses is limited to the response spectrum modal analysis technique.

Two consecutive modes are defined as closely spaced if their frequencies differ from each other by 10% or less of the lower frequency. For modes that are not closely spaced, the maximum value of the response of a given element of a system or component, subjected to a single independent spatial component (response spectrum) of a three-component earthquake, is obtained by taking the square root of the sum of the squares of corresponding maximum values of the response of the element attributed to individual significant modes of the system or component.

If some or all of the modes are closely spaced, they are divided into groups that include all modes having frequencies between the lowest frequency in the group and a frequency 10% higher. The representative maximum value of a particular response of a given element of a system or component attributed to each such group of modes is first obtained by taking the sum of the absolute values of the corresponding peak values of the response of the element attributed to individual modes in that group. The representative maximum value of this particular response attributed to all the significant modes of the system or component is then obtained by taking the square root of the sum of the squares of corresponding representative maximum values of the response of the element attributed to each closely spaced group of modes and the remaining modal responses for the modes that are not closely spaced.

This procedure is in conformance with Regulatory Guide 1.92.

3.7.3.7.2 Components and Equipment Supplied by the NSSS Vendor

For response spectra analysis, the total unidirectional seismic response is obtained by 02-01
combining the individual modal responses utilizing the square root of the sum of the squares method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10% of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root 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:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2\sum_{j=1}^{S}\sum_{K=M_j}^{N_j-1}\sum_{\ell=K+1}^{N_j} R_K R_\ell \varepsilon_{K\ell} \quad (3.7\text{-}32)$$ RN 01-113

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:

$$\varepsilon_{K\ell} = \left\{1 + \left[\frac{\omega'_K - \omega'_\ell}{\beta'_K \omega'_K \omega'_\ell}\right]^2\right\}^{-1} \quad (3.7\text{-}33)$$

and:

$$\omega'_j = \omega_j[1-(\beta'_j)^2]^{1/2} \quad (3.7\text{-}34)$$

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$$\beta'_j = \beta_j + \frac{2}{\omega_j t_d} \quad (3.7\text{-}35)$$

Where:

ω_j = frequency of closely spaced mode j.

β_j = fraction of critical damping in closely spaced mode j. RN 01-113

t_d = duration of the earthquake.

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Frequency	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20

There are two groups of closely spaced modes, namely with modes {2, 3, 4} and {6, 7}. Therefore:

S = 2 number of groups of closely spaced modes.

M_1 = 2 lowest modal number associated with group 1.

N_1 = 4 highest modal number associated with group 1.

M_2 = 6 lowest modal number associated with group 2.

N_2 = 7 highest modal number associated with group 2.

N = 8 total number of modes considered.

The total response for this system is, as derived from the expansion of Equation (3.7-32):

$$R_T^2 = [R_1^2 + R_2^2 + R_3^2 + \ldots + R_8^2] + 2R_2R_3\varepsilon_{23} + R_2R_4\varepsilon_{24} + 2R_3R_4\varepsilon_{34} + 2R_6R_7\varepsilon_{67} \qquad (3.7\text{-}36)$$ RN 01-113

For time history analysis, each earthquake component direction is analyzed separately. For each of these analysis, at each time step, the global response of interest is obtained by superposition of the individual modal responses. 02-01

The preceding gives the criteria and justification for meeting the intent of Regulatory Guide 1.92, Revision 1.

3.7.3.8 Analytical Procedures for Piping

3.7.3.8.1 Balance of Plant Scope

The piping system geometry, cross sectional dimension and physical properties of each pipe segment and the restraint conditions are supplied as inputs to the PIPDYN II computer program during construction. During operations, the approved programs are specified in the VCSNS piping design guide. The mass of each piping segment is lumped at the element nodes by the computer. Additional concentrated masses are specified separately for valves, actuators, and other concentrated weights at the centers of gravity for the individual assembly or subassemblies to represent both bending and torsional effects of the assembly. 02-01

The restraint conditions of supports are specified in three translational and three rotational directions in the model, in either global or local coordinates for each support point. The restraints may be free, rigid, or elastic with a specified spring constant for each translational or rotational direction. When coupling effects between any two joint degrees of freedom are significant, a 6 by 6 stiffness matrix is used to describe an elastic foundation. Moment release at nodal points is used for pin connections or flexible joints whenever applicable.

The computer then formulates a discrete system of equations based upon the input data. The resulting homogeneous equations are solved as an eigenvalue problem. The floor response spectrum method is used in calculating the responses of each mode including nodal displacements, end forces, and moments and support loads. These modal responses are combined by the square root of the sum of the squares method for all modes with frequencies less than 30 Hz. In addition, the effects of the modes not included are added to the square root of the sum of the squares response as one term, using the highest frequency from the square root of the squares response under 30 Hz to obtain the total response. The definition and grouping method of combining close modes, described in Section 3.7.3.7 is applied in nodal deflection, element end forces and moments, acceleration, and support loads.

The responses from the two horizontal and the vertical component of an earthquake are calculated separately as described above. These responses are then combined using the square root of the sum of the squares method. The resultant end moments are finally used in the applicable ASME Code, Section III, equations for stress evaluation.

For certain 2 inch and smaller pipes or cold pipes of larger sizes, a simplified floor response spectrum dynamic analysis is performed. An iterative analytical procedure is followed for each pipe size, schedule, and response spectrum input, while using span length as a variable. The dynamic system equations are solved for the various span lengths and associated support reactions and pipe stresses are determined. Maximum allowable span lengths are then selected on the basis of a stress and/or natural frequency criteria for each of the various parameters. Seismic supports/restraints are then spaced to be within the allowable span limits established by this stress/frequency criteria. The pipe frequency is determined using a multi-span, simple supported model with the maximum spans established as described in Section 3.7.3.4. Justification for this approximate analysis method has been demonstrated by comparison of the results of detailed dynamic analyses of various piping systems with the results of the simplified method.

A scaling factor (gamma factor) is applied to the response spectrum envelopes for the vertical direction. This is used to account for amplification due to flexure of floor slabs and, therefore, is a function of the location of the components on a particular floor.

Typical scaling factors used for seismic analysis are presented in Table 3.7-8 and Figures 3.7-44 through 3.7-46.

Where relative structural movement exists within the same structure or where piping spans between adjacent structures, the effect of differential piping support movements are evaluated. The relative movements of pipe supports are considered separately in each of the spatial directions. The results are then combined by the square root of the sum of the squares method.

3.7.3.8.2 Components and Equipment Provided by the NSSS Vendor

The Class 1 piping systems are analyzed to the rules of the ASME Code, Section III, NB 3650. When response spectrum methods are used to evaluate piping systems supported at different elevations, the following procedures are used. The effect of differential seismic movement of piping supports is included in the piping analysis according to the rules of the ASME Code, Section III, Paragraph NB 3653. According to ASME definitions, these displacements cause secondary stresses in the piping system. The response quantity of interest induced by differential seismic motion of the support is computed statically by considering the building response on a mode-by- mode basis.

In the response spectrum dynamic analysis for evaluation of piping systems supported at different elevations, the most severe floor response spectrum corresponding to the support locations is used.

The selection, location, and use of snubbers for these systems is based upon the stress requirements of the ASME Code, Section III, Articles NB/NC/ND-3600 or other controlling design criteria.

3.7.3.9 Multiply Supported Equipment Components with Distinct Inputs

3.7.3.9.1 Balance of Plant Scope

Any equipment supported at different locations (elevations and/or floors) is analyzed by imposing a single conservative response spectrum at each location. This response spectrum is constructed in such a way that it conservatively envelopes the pertinent response spectra of the different locations.

3.7.3.9.2 Components and Equipment Provided by the NSSS Vendor

When response spectrum methods are used to evaluate Reactor Coolant System primary components interconnected between floors, the procedures of the following paragraphs are used. There are no components in the Westinghouse scope of analysis which are connected between buildings. The primary components of the Reactor Coolant System are supported at no more than two floor elevations.

A response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis are the most severe floor response spectra.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis. The results of the building analysis are reviewed on a mode-by-mode basis to determine the differential motion in each mode. Per ASME Code rules, the stress caused by differential seismic motion is clearly secondary for piping (NB 3650) and component supports (NF 3231). For components, the differential motion is evaluated as a free end displacement, since, per NB 3213.19, examples of a free end displacement are motions "that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping". The effect of the differential motion is to impose a rotation on the component by the building. This motion, then, being a free end displacement and being similar to thermal expansion loads, causes stresses which are evaluated with ASME Code methods including the rules of NB 3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7.3.10 <u>Use of Constant Vertical Static Factors</u>

3.7.3.10.1 Balance of Plant Scope

The response spectrum method is used for the vertical seismic subsystem dynamic analysis. However, for the cases where the equipment's lowest frequency in the vertical direction is more than 30 Hz, the maximum floor acceleration is used for equipment design.

3.7.3.10.2 Components and Equipment Provided by the NSSS Vendor

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

3.7.3.11 Torsional Effects of Eccentric Masses

If the torsional effect of a valve operator or other eccentric mass is likely to have a significant effect upon the results of the analysis described in Section 3.7.3.1 for Seismic Category I systems, the eccentric mass and its moment arm are included in the mathematical model described in Sections 3.7.3.3 and 3.7.3.8.

3.7.3.12 Buried Seismic Category I Piping Systems

Seismic analysis of buried safety class piping is performed in three phases, as follows:

1. Calculation of maximum soil strain and curvature resulting from the propagation of seismic waves, considering an estimated relative contribution of each wave type (shear wave, compression wave, and Rayleigh wave).

2. Determination of the extent to which the pipe deforms elastically as a result of soil strain and curvature, or as a result of relative ground movement at building/soil interfaces, considering the influence of friction forces between soil and pipe and treatment of soil as a continuous elastic support.

3. Calculation of the stresses in the pipe which result from such elastic deformation and comparison with allowable stress as described by Table 3.9-2.

References [21] through [24] were used as the basis for the analytical determination of seismic stresses in buried safety class piping. Sources for various parameters used in the analysis are as follows:

1. Maximum acceleration for SSE, Section 2.5.2.10.

2. Fill surface elevations and subsoil conditions, Sections 2.5.4.4 and 2.5.4.8.

3. Properties ($\gamma_t, \bar{c}, \overline{Q}, \mu, \text{and } G_{max}$) of Zone I and Zone II fill soils, Sections 2.5.4.5.2 and 2.5.6.4.

4. Properties ($\gamma_t, \bar{c}, \text{and } \overline{Q}$)of Zone III fill soil are assumed, based upon Section 2.5.4.5.2 and US Navy Design Manual, NAVDOCK DM-7, "Soil Mechanics, Foundations and Earth Structures" Bureaus of Yards and Docks, 1962.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

A Seismic Category I piping system is analyzed by including the piping extending to at least the first restraint in each of the three mutually orthogonal directions beyond the defined Seismic Category I boundaries. Whenever necessary, piping segments and restraints beyond the region described above are included to ensure that both the elastic reaction and the effects of masses of the non-Seismic Category I piping on the Seismic Category I piping are adequately represented. Those portions of piping which form interfaces between Seismic Category I and non-Seismic Category I are designed to satisfy Seismic Category I requirements.

3.7.3.14 Seismic Analyses for Reactor Internals

Fuel assembly component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling. The time history floor response based on a standard seismic time history normalized to SSE levels is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The component seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in References [4] and [17].

Fuel assembly lateral structural damping obtained experimentally is presented in Figure 3-4 of Reference [18]. The data indicate that no damping values less than 10% were obtained for fuel assembly displacements greater than 0.11 inches.

The distribution of fuel assembly amplitudes decreases as one approaches the center of the core. The average amplitude for the minimum displacement fuel assembly is well above 0.11 inches for the SSE.

Fuel assembly displacement time history for the SSE seismic input is illustrated in Figure 2-3 of Reference [18]. RN 01-113

The control rod drive mechanisms (CRDM) are seismically analyzed to confirm that system stresses under the combined loading conditions as described in Section 3.9.1 do not exceed allowable levels as defined by the ASME Code, Section III for Upset and Faulted conditions. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation and the resultant seismic bending moments along the length of the CRDM are calculated. The corresponding stresses are then combined with the stresses from the other loadings required and the combination is shown to meet the ASME Code Section III requirements.

3.7.3.15 Analysis Procedure for Damping

3.7.3.15.1 Balance of Plant Scope

The composite modal damping approach is used to account for element-associated damping. This is based upon the use of the mass as a weighting function in generating the composite modal damping. The formulation leads to:

$$\bar{\beta}_j = \{\phi\}^T [\bar{M}] \{\phi\}$$

Where:

$\bar{\beta}_j$ = Equivalent modal damping factor of the j(th) mode.

$[\bar{M}]$ = The modified mass matrix constructed from element matrix formed by the product of the damping factor as provided in Table 3.7-9 for the element and its mass matrix.

$\{\phi\}$ = The j(th) normalized model vector.

In the cases of equipment which consists of elements of the same damping factor, the above equation is reduced to a single damping factor.

3.7.3.15.2 Components and Equipment Provided by the NSSS Vendor

Analysis procedures for damping for subsystems in Westinghouse's scope of responsibility are given in Section 3.7.2.15.2.

3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison with Regulatory Guide 1.12

The seismic instrumentation provided is in general conformance with Regulatory Guide 1.12 (see Appendix 3A) for a maximum foundation acceleration of less than 0.3g.

Since the soil-structure interaction is negligible, the "Free Field" triaxial time-history accelerograph is omitted as permitted by ANSI N18.5[19].

3.7.4.2 Location and Description of Instrumentation

The types and locations of seismic instrumentation are as follows:

1. Triaxial Time-History Accelerometer

   One triaxial accelerometer is located at each of the following locations

   a. Reactor Building foundation mat outside of the Reactor Building (see Figure 3.7-47).

   b. Reactor Building ring girder outside the Reactor Building (see Figure 3.7-48).

   The output of both triaxial sensor units (accelerometers) is recorded by a solid-state recording system in the Control Building. The accelerometer located on the Reactor Building foundation mat also functions as a seismic trigger for the actuation of the solid-state recording system. An alarm is sounded when the recording system is activated.

2. Triaxial Response Spectrum Recorder

   One triaxial response spectrum recorder, capable of permanently recording peak response as a function of frequency for both horizontal motions and vertical motion, is provided at each of the following locations:

   a. Reactor Building foundation mat outside the Reactor Building (see Figure 3.7-47).

   b. Steam generator support (see Figure 3.7-51).

   c. On the Intermediate Building roof at elevation 463'-0" (see Figure 3.7-52).

   d. On the foundation of the Auxiliary Building at elevation 374'-0" (see Figure 3.7-53).

   Each triaxial response spectrum recorder will record 12 frequencies, 1/3 of an octave apart, beginning with 2 Hz and ending with 25.4 Hz.

3. Triaxial Seismic Switch

   A triaxial seismic switch, located at the Reactor Building foundation mat, is provided to actuate an alarm in the control room (see Figure 3.7-47).

   The seismic monitoring instrumentation shown in Table 3.7-11 should be maintained with a high level of availability. Each of these instruments shall be demonstrated operable by the performance of the channel check, channel calibration, and analog channel operational test as established in the appropriate test procedures.

Each of the above seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01g shall be restored within 24 hours and a channel calibration performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

The availability of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required.

Criteria for the selection of types and locations of seismic instrumentation are in accordance with Regulatory Guide 1.12 (see Appendix 3A). Where multiple locations for a particular instrument are possible, the location selected is based upon analytical results which show that an amplified response is expected at the selected location. If an earthquake occurs, the recorded responses of the previously discussed seismic instrumentation, except for the triaxial seismic switch, are compared to calculated responses as discussed in Section 3.7.4.4.

For instruments using a plate as a mounting adapter, Table 3.7-10 presents the calculated lowest natural frequency (of three dimensions) of each mechanical system monitored (adapter plate, plus instrument). For instruments not using a plate adapter, the instruments are rigidly bolted to the concrete surface as a mounting accordance with the instrument manufacturer's instructions.

Instrument assemblies are specified and designed to be free of spurious resonances within the frequency range of the instrument.

3.7.4.3 Control Room Operator Notification

Control room signals available to the operator are as follows:

1. Indication and an audible alarm are actuated when the triaxial seismic switch at the Reactor Building foundation signals that the OBE peak ground acceleration has been exceeded in either of the horizontal directions or in the vertical direction.

2. Indication and a common alarm are actuated when any of the 12 elements of each triaxial section of the triaxial response spectrum recorder at the Reactor Building foundation mat exceeds the frequency setpoint. Two setpoints are provided for each element. Exceeding the first setpoint illuminates a yellow light at an

acceleration equivalent to 2/3 OBE design response spectra. A red light is illuminated if acceleration exceeds the OBE design response spectra.

3. Indication and an audible alarm are actuated when the accelerometer at the Reactor Building foundation mat detects acceleration greater than 0.01g in either horizontal direction or greater than 0.0067g in the vertical direction signifying that the triaxial time-history accelerometer recording system has started. The recording system is located in the relay room below the control room.

3.7.4.4 Comparison of Measured and Predicted Responses

In the event of an earthquake, the control room supervision determines whether or not there are initial indications that the OBE acceleration level has been exceeded. This is accomplished by inspection of the indications and alarms described in Section 3.7.4.3. In approximately 15 minutes after the occurrence of the earthquake, data from the triaxial time history accelerometer located on the Reactor Building foundation mat will be analyzed and retrieved from the relay room recording/analysis system for use by the control room supervision in assessing whether the OBE design of the plant has been exceeded. This is accomplished as follows:

1. The triaxial time history accelerometer on the Reactor Building foundation mat records acceleration as a function of time. The data is recorded in solid-state and then processed via computer analysis to produce:(a) comparisons of the recorded earthquake to the foundation OBE design for each directional component, (b) a calculation of the Cumulative Absolute Velocity as compared to a threshold criterion for each directional component, and (c) a summary determination of exceedance of the plant OBE design basis. The summary position on exceedance of the plant OBE design can then be used by the control room supervision to determine if shut down of the plant is warranted.

2. Plant operator walkdowns will also commence after the occurrence of an earthquake to evaluate any unusual plant conditions which might exist, such as, pump or valve leakages, excess equipment vibrations or deformations, structural cracking, fallen objects, etc. Reports on the extent of damage, or lack thereof, will be used by the control room supervision in conjunction with the seismic instrumentation results and other control room indications to establish a position on whether the plant OBE design has been exceeded and if shut down is required. This decision should be made within approximately eight (8) hours of the earthquake occurrence

3. If the decision to shut down the plant is made, it should be conducted in a controlled process using existing plant procedures.

4. As part of the evaluation of the effects of the earthquake, the measured responses from the other sensors located throughout the plant will be compared to the design response for their respective locations. These comparisons will be used to

evaluate the overall impact of the earthquake as it relates to the design of the plant and components.

3.7.5 SEISMIC DESIGN CONTROL

3.7.5.1 Balance of Plant Scope

Safety class components and equipment are designed using floor response spectra as input with the exception of certain instruments, generally locally mounted devices, which are procured using fixed values of 1.5g (wall mounted devices) and 3.0g (pipe mounted devices) as conservative values for multiple locations. Vendors are responsible for the design and qualification by analysis or test of components and equipment within their scope. This responsibility includes review of the design, analyses, or tests by one or more qualified engineers other than the engineer(s) who originated the design, performed the analyses, or developed the tests.

Safety class piping systems are also designed using floor response spectra as input. Gilbert is responsible for design and qualification of piping systems by analysis during construction (see Section 3.7.3). This responsibility includes review of the design and analyses by one or more qualified engineers other than the engineer(s) who originated the design or performed the analyses. During operations, VCSNS staff is responsible for design and qualification of BOP piping sysgtems by analyses which may be performed by various contractors per VCSNS piping design guides or their predecessors. 02-01

The snubber vendor maintains acceptance and qualification test reports on file for each type of snubber which has been used as a piping restraint for the Virgil C. Summer Nuclear Station.

The loading conditions and transients analyzed are described for each system in the piping system design specification. The analytical procedure described in Section 3.7.3.8 is followed and the results are compared to the applicable design Section of the ASME Code or other controlling design requirements.

For certain small diameter piping (2 inch and smaller), the seismic design is implemented by the application of simplified seismic support criteria which are subjected to independent review.

3.7.5.2 Components and Equipment Provided by the NSSS Vendor

The following procedure is implemented for Westinghouse supplied safety-related mechanical equipment that falls within one of the many categories which have been analyzed as described in Sections 3.7.2 and 3.7.3 and has been shown to be rigid with all natural frequencies greater than or equal to 33 Hz.

1. Equivalent static acceleration factors for the horizontal and vertical directions are included in the equipment specification. The vendor must certify the adequacy of the equipment to meet the seismic requirements as described in Section 3.7.3.

2. When the floor response spectra are developed the cognizant engineer responsible for the particular component checks to ensure that the acceleration factors are less than those given in the equipment specification. If accelerations exceed those in the equipment specifications, the designs are rechecked to verify equipment adequacy.

All other Westinghouse supplied safety-related equipment is analyzed or tested as described in Sections 3.7.2, 3.7.3, and 3.10.

3.7.6 REFERENCES

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TABLE 3.7-0

FREQUENCY INTERVALS USED FOR CALCULATION OF RESPONSE SPECTRA

Frequency Range (Hz)	Increments (Hz)
0.2 - 3.0	0.0622
3.0 - 3.6	0.1425
3.6 - 5.0	0.1857
5.0 - 8.0	0.20
8.0 - 15.0	0.3532
15.0 - 18.0	0.5414
18.0 - 22.0	0.80
22.0 - 34.0	1.450

TABLE 3.7-1

DAMPING FACTORS
PERCENT OF CRITICAL DAMPING (1)

Component or Structure	Value Used OBE & SSE	Regulatory Guide 1.61 Value OBE	Regulatory Guide 1.61 Value SSE
Reactor Building	2.0	2.0	5.0
Concrete Support Structures Inside the Reactor Building	2.0	4.0	7.0
Assemblies & Structures			
Bolted & Riveted	2.5	4.0	7.0
Welded	2.0	2.0	4.0
Vital Piping Systems			
Larger than 12 Inch Diameter	2.0/3.0 (2)	2.0	3.0
12 Inch Diameter and Smaller	1.0/2.0 (2)	1.0	2.0

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NOTE:

(1) See Reference [1]

(2) Code Class N-411, “Alternative Damping Values for Seismic Analysis of Class 1, 2 and 3 Piping Sections, Section III, Division 1,” is acceptable for piping analyses for systems supported on all building structures per Regulatory Guide 1.84, Rev. 25.

TABLE 3.7-2

DAMPING FACTORS
PERCENT OF CRITICAL DAMPING
FOR STRUCTURES WITH SOIL INTERACTION

	Range of Shear Wave Velocity	Damping Factor
On Rock	$C > 6000$ fps	2-5
On Firm Soil	$C \geq 2000$ fps	5-7
On Soft Soil	$C < 2000$ fps	7-10

NOTE: See reference [1].

TABLE 3.7-3

DAMPING VALUES USED FOR SEISMIC SYSTEMS ANALYSIS FOR WESTINGHOUSE SUPPLIED EQUIPMENT(3)

RN 16-003

Item	Damping (Percent of Critical) Upset Conditions (OBE)	Faulted Condition (SSE, DBA)
Primary Coolant Loop System - Components and Large Piping (1)	2 (2)	4 (2)
Small Piping	1 (2)	2 (2)
Welded Steel; Structures	2	4
Bolted and/or Riveted Steel Structures	4	7

(1) Generally applicable to 12 inch or larger diameter piping.

(2) Code Case N-411, "Alternative Damping Values for Seismic Analysis of classes 1, 2 and 3 Piping Sections, Section III, Division 1," is acceptable for piping analyses for systems supported on all building structures per Regulatory Guide 1.84, Rev. 25.

(3) Damping values noted were applied to the design of the Integrated Head Assembly (IHA) which replaced the service structure on the original reactor vessel head. The IHA is installed on the replacement Reactor Vessel Closure Head. RN 16-003

TABLE 3.7-4

FOUNDATION SEISMIC DESIGN PARAMETERS

Foundation Material	Compressional Wave Velocity (ft/sec)	In-Situ Density (lbs/cu ft)	Poisson's Ratio (μ)	Shear Modulus (G) or Modulus of Rigidity (lbs/sq in)	Modulus of Subgrade Reaction[1] (lbs/cu ft)
Saprolite	1000-3000	110-135	0.35	1×10^4 to 3.5×10^4	$\frac{5 \times 10^6}{\sqrt{BL}}$
Weathered and jointed rock	(12,000-13,000)[2]	(140-160)[2]	0.30	5×10^5	$\frac{2 \times 10^8}{\sqrt{BL}}$
Sound rock	15,000	165	0.20	2×10^6	$\frac{8 \times 10^8}{\sqrt{BL}}$

(1) "BL" is contact area of square foundation. For other contact area shapes, subgrade modulus must be modified (see Barkan, 1962).

(2) Numbers in parentheses are estimated values.

NOTE: Young's Modulus, $E = 2(1 + \mu)G$.

TABLE 3.7-5

FOUNDATION ELEVATION AND FOUNDATION TYPE
SEISMIC CATEGORY I STRUCTURES

Structure	Foundation Elevation (ft)	Foundation Type
Reactor Building		
North	341(1) to 396	Fill Concrete
East	348(1) to 396	Fill Concrete
South	362(1) to 396	Fill Concrete
West	367(1) to 396	Fill Concrete
Entire Structure	396 to 408	Mat
Control Building		
North	366(1) to 407	Fill Concrete
East	371(1) to 407	Fill Concrete
South	366(1) to 407	Fill Concrete
West	371(1) to 407	Fill Concrete
Entire Structure	407 to 411	Mat
Diesel Generator Building		
Entire Structure	Table 3.7-6	Caissons
Service Water Pumphouse	386	Mat
Service Water Intake Structure	367	Mat
Service Water Discharge Structure	408	Mat
Auxiliary Building		
North	354(1) to 384	Fill Concrete
South	368(1) to 370	Fill Concrete
North	384 to 388	Mat
South	370 to 374	Mat
Fuel Handling Building		
Entire Structure	Table 3.7-6	Caissons
Intermediate Building		
Entire Structure	Table 3.7-6	Caissons

(1) Bottom elevation determined in field to suit actual condition.

TABLE 3.7-6

FOUNDATION DATA FOR SAFETY CLASS STRUCTURES SUPPORTED ON CAISSONS

Structure	Elevation of Cap (ft)	Underlying Soil	Elevation in No. 3 Rock for Seismic Input
Diesel Gen. Building	394 to 421	Zone I, II, III fill and silty fine sand, medium dense to dense	369 to 363
Fuel Handling Building	409 to 430	Zone I, II, III fill and silty fine sand, medium dense to dense	345
Intermediate Building	394 to 409	Zone I, II, III fill and silty fine sand, medium dense to dense	375

TABLE 3.7-7

DIMENSIONS OF SEISMIC CATEGORY I STRUCTURES

Structure	Plan Dimensions (ft)	Height (ft)	Embedment Depth (ft)	Top Elevation (ft)
Reactor Building	134, OD	206	39	602
Control Building	84 x 141	98	28	505
Diesel Generator Building	66 x 67	83 and 56	41 and 14	477
Service Water Pumphouse	70 x 79	73	49	459
Service Water Intake Structure	166 x 18	21 and 30	--	388 and 396.5
Service Water Discharge Structure	35 x 33	6 to 15	15	425
Auxiliary Building	120 x 190	--	--	--
North	--	127	51	511
South	--	141	65	511
Fuel Handling Building	75 x 123	95 and 102	26 and 5	511
Intermediate Building	85 x 198	76 and 78	39 and 26	485

TABLE 3.7-7a

RESPONSE LOADS FOR SEISMIC CATEGORY I STRUCTURES

Elevation (Ft)	Mass (Kip-Sec^2/Ft)	Acceleration (g)			Displacement (in)		
		X	Y	Vert.	X	Y	Vert.
Service Water Pumphouse							
390'	184.938	0.200	0.160	0.161	0.765	0.612	0.063
425'	211.736	0.235	0.188	0.173	0.899	0.719	0.068
436'	107.693	0.262	0.209	0.175	1.002	0.799	0.069
441'	55.416	0.276	0.220	0.186	1.055	0.841	0.073
459'	82.328	0.327	0.261	0.190	1.250	0.998	0.074
Control Building							
412'	257.93	0.144	0.168	0.078	0.015	0.027	0.010
425'	141.15	0.183	0.212	0.083	0.028	0.047	0.010
436	120.46	0.208	0.260	0.110	0.035	0.061	0.017
448'	95.91	0.263	0.309	0.124	0.055	0.074	0.019
463'	150.39	0.328	0.365	0.130	0.078	0.087	0.019
482'	191.86	0.429	0.433	0.137	0.100	0.103	0.020
505'	196.97	0.501	0.490	0.142	0.117	0.114	0.020
Auxiliary Building							
374'	490.55	0.117	0.117	0.082	0.011	0.009	0.014
388'	679.64	0.134	0.129	0.098	0.019	0.016	0.012
397'	227.16	0.150	0.203	0.089	0.028	0.038	0.015
412'	1086.12	0.232	0.268	0.106	0.055	0.057	0.015
436'	900.96	0.338	0.363	0.109	0.085	0.083	0.015
463'	658.05	0.501	0.444	0.116	0.137	0.104	0.016
485'	390.81	0.554	0.485	0.118	0.154	0.114	0.016
511'	50.15	0.650	0.498	0.120	0.175	0.120	0.016
Intermediate Building							
412'	398.45	0.180	0.308	0.209	0.031	0.072	0.025
436'	432.54	0.274	0.461	0.222	0.043	0.111	0.026
463'	368.40	0.357	0.574	0.235	0.065	0.136	0.027
485'	106.24	0.372	0.541	0.242	0.065	0.140	0.027
Reactor Building Shell							
435'	203.54	0.161	0.157	0.157	0.040	0.040	0.005
462'	180.06	0.202	0.200	0.195	0.079	0.079	0.008
481'	148.76	0.243	0.242	0.220	0.109	0.109	0.009
500'	173.73	0.296	0.295	0.242	0.139	0.139	0.010

TABLE 3.7-7a (Continued)

RESPONSE LOADS FOR SEISMIC CATEGORY I STRUCTURES

Elevation (Ft)	Mass (Kip-Sec^2/Ft)	Acceleration (g)			Displacement (in)		
		X	Y	Vert.	X	Y	Vert.
Reactor Building Shell (Cont)							
523’	180.06	0.350	0.350	0.267	0.176	0.176	0.012
546’	153.11	0.397	0.399	0.288	0.211	0.212	0.013
582’	517.95	0.476	0.475	0.301	0.257	0.258	0.014
Interior Concrete							
407’	468.51	0.118	0.114	0.112	0.005	0.005	0.002
427’	144.29	0.168	0.162	0.124	0.012	0.011	0.003
431’	59.24	0.180	0.170	0.126	0.014	0.013	0.003
435’	136.40	0.192	0.183	0.129	0.016	0.016	0.003
439’	47.11	0.201	0.195	0.130	0.018	0.017	0.003
445’	109.50	0.219	0.221	0.131	0.021	0.020	0.003
462’	330.75	0.298	0.304	0.134	0.028	0.027	0.004
475’	64.35	0.342	0.347	0.134	0.032	0.030	0.004
Diesel Generator Building							
427’	127.7	0.466	0.470	0.107	0.228	0.216	0.006
436’	117.2	0.474	0.478	0.111	0.264	0.240	0.006
463’	84.1	0.520	0.514	0.111	0.336	0.348	0.006
476’	67.7	0.841	1.420	0.090	0.384	0.384	0.006
Fuel Handling Building							
436’	882.82	0.303	0.320	0.230	0.101	0.130	0.023
463’	6.78	0.304	0.328	0.230	0.209	0.279	0.036
495’	8.85	0.846	0.974	0.225	0.337	0.456	0.051
512’	6.18	0.814	0.884	0.225	0.405	0.550	0.059

TABLE 3.7-7b

RESULTS OF TRANSIENT ANALYSIS - TYPICAL TORSIONAL MODEL

Transient Analysis Sum X-EQ

	System Accelerations (Total)					
	Translation			Bending Response	Torsional Response	Bending Response
Joint	X	Y	Z	X	Y	Z
701	0.174155E 02 0.968478E 01	-0.216468E 00 0.958479E 01	0.194451E 01 0.240496E 01	-0.339760E-02 0.232497E 01	0.317496E-01 0.216497E 01	-0.128979E-01 0.950479E 01
70102	0.151535E 02 0.968478E 01	-0.297819E 00 0.822482E 01	-0.166849E 01 0.682485E 01	-0.339760E-02 0.232497E 01	0.317496E-01 0.216497E 01	-0.128979E-01 0.950479E 01
70103	0.151535E 02 0.968478E 01	-0.135537E 01 0.950479E 01	-0.414053E 01 0.216497E 01	-0.339760E-02 0.232497 E 01	0.317496E-01 0.216497E 01	-0.128979E-01 0.950479E 01
70104	0.192796E 02 0.968478E 01	0.664329E 00 0.822482E 01	-0.386272E 01 0.216497E 01	-0.339760E-02 0.232497E 01	0.317496E-01 0.216497E 01	-0.128979E-01 0.950479E 01
70105	0.192796E 02 0.968478E 01	-0.795434E 00 0.992478E 01	-0.166849E 01 0.682485E 01	-0.339760E-02 0.232497E 01	0.317496E-01 0.216497E 01	-0.128979E-01 0.950479E 01
702	0.143161E 02 0.984478E 01	-0.200542E 00 0.950479E 01	-0.223558E 01 0.232497E 01	-0.339760E-02 0.232497E 01	0.270121E-01 0.216497E 01	-0.128978E-01 0.950479E 01
70202	0.125049E 02 0.984478E 01	-0.299360E 00 0.822482E 01	-0.143435E 01 0.232497E 01	-0.339960E-02 0.232497E 01	0.270121E-01 0.216497E 01	-0.128978E-01 0.950479E 01
70203	0.125049E 02 0.984478E 01	-0.135695E 01 0.950479E 01	0.377615E 01 0.224497E 01	-0.339760E-02 0.232497E 01	0.270121E-01 0.216497E 01	-0.128978E-01 0.950479E 01
70204	0.162128E 02 0.984478E 01	0.662784E 00 0.822482E 01	0.357268E 01 0.224497E 01	-0.339760E-02 0.232497E 01	0.270121E-01 0.216497E 01	-0.128978E-01 0.950479E 01
70205	0.162128E 02 0.984478E 01	-0.792828E 00 0.992478E 01	-0.143435E 01 0.232497E 01	-0.339760E-02 0.232497E 01	0.270121E-01 0.216497E 01	-0.128978E-01 0.950479E 01
703	0.130849E 02 0.984478E 01	-0.137671E 00 0.950479E 01	-0.197403E 01 0.232497E 01	-0.339755E-02 0.232497E 01	0.221927E-01 0.216497E 01	-0.128976E-01 0.950479E 01
70302	0.115773E 02 0.984478E 01	-0.303186E 00 0.822482E 01	-0.131435E 01 0.232497E 01	-0.339755E-02 0.232497E 01	0.221927E-01 0.215497E 01	-0.128976E-01 0.950479E 01
70303	0.115773E 02 0.984478E 01	-0.135311E 01 0.950479E 01	-0.320794E 01 0.216497E 01	-0.339755E-02 0.232497E 01	0.221927E-01 0.216497E 01	-0.128976E-01 0.950479E 01
70304	0.146807E 02 0.984478E 01	0.658934E 00 0.822482E 01	0.303730E 01 0.224497E 01	-0.339755E-02 0.232497E 01	0.221927E-01 0.216497E 01	-0.128976E-01 0.950479E 01
70305	0.146807E 02 0.984478E 01	-0.795415E 00 0.992478E 01	-0.131435E 01 0.232497E 01	-0.339755E-02 0.232497E 01	0.221927E-01 0.216497E 01	-0.128976E-01 0.950479E 01

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TABLE 3.7-7c

REACTOR BUILDING MODAL PARTICIPATION FACTORS

Mode	Participation Factor	Frequency (cps)
*1 (dominant mode)	26.48	4.35
*2	22.68	11.47
*3	12.18	15.68
*4	22.25	29.15
*5	-8.03	32.96
*6	13.69	40.64
*7	-6.36	45.87
8	-0.97	56.10
9	-1.26	56.77
10	-0.11	66.38
11	0.06	76.14
12	-1.01	79.20
13	0.37	111.01
14	-1.79	147.12
15	-0.66	157.47
16	0.69	208.84
17	0.37	213.11
18	0.01	239.89
19	0.35	312.96
20	-0.12	385.35

* Significant modes

TABLE 3.7-8

GAMMA SCALING FACTORS

	Gamma Scaling Factor
Reactor Building Shell and Interior Concrete	1.0
Control Building	See Figure 3.7-44
Diesel Generator Building	1.0
Fuel Handling Building	1.0
Auxiliary Building	See Figure 3.7-45
Intermediate Building	See Figure 3.7-46
Service Water Pumphouse	1.0

TABLE 3.7-9

DAMPING FACTORS
PERCENT OF CRITICAL DAMPING [1]

Component or Structure	Working Stress, No More Than About One-Half Yield Point	At, or Just Below Yield Point	Beyond Yield Point with Permanent Strain Greater Than Yield Point Limit Strain
Reactor Building	2.0	5.0	Not applicable
Concrete Support Structures Inside the Reactor Building	2.0	5.0	
Assemblies L Structures			
a. Bolted & Riveted	2.5	5.0	
b. Welded	2.0	5.0	
Vital Piping Systems [2]			
Other Concrete Structures above Ground	2.0	5.0	

(1) Reference: "Seismic Design Criteria for Nuclear Reactor Facilities" by Nathan M. Newmark and William J. Hall, Proceedings, Fourth World Conference on Earthquake Engineering, January 13 - January 18, 1969, Santiago, Chile.

(2) Refer to Table 3.7-1 for damping factors.

TABLE 3.7-10

SEISMIC INSTRUMENTATION SENSING ELEMENTS [1]

Ident. No.	Sensing Element Description	Location	FSAR Figure Reference	Mounting Type	Lowest Natural Frequency of Mounting [2]
IYM-1780	Accelerometer	Reactor building, foundation mat, outside reactor building	3.7-47	Bolted to foundation	[3]
IYM-1782	Seismic switch	Reactor building, foundation mat, outside reactor building	3.7-47	Bolted to foundation	[3]
IYM-1783	Response spectrum recorder with switches	Reactor building, foundation mat, outside reactor building	3.7-47	Bolted to foundation	[3]
IYM-1784	Accelerometer	Reactor building, top of ring girder, outside reactor building	3.7-48	Bolted to ring girder concrete	[3]

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Amendment 02-01

TABLE 3.7-10 (Continued)

SEISMIC INSTRUMENTATION SENSING ELEMENTS [1]

Ident. No.	Sensing Element Description	Location	FSAR Figure Reference	Mounting Type	Lowest Natural Frequency of Mounting [2]
IYM-1785	Response spectrum recorder	Steam generator C, upper lateral support, inside Reactor Building	3.7-51	Bolted to adapter plate	30.9 Hz
IYM-1786	Response spectrum recorder	Intermediate Building roof, elevation 463’	3.7-52	Bolted to concrete floor	[3]
IYM-1787	Response spectrum recorder	Auxiliary Building foundation elevation 374’	3.7-53	Bolted to foundation	

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NOTES:

1. All listed sensing elements are triaxial sensors.

2. Lowest natural frequency, in three dimensions, of mounting plate.

3. This instrument does not use a plate as a mounting adapter. The instrument is bolted rigidly to the concrete surface that it monitors. The instrument is free from spurious resonances within its frequency range.

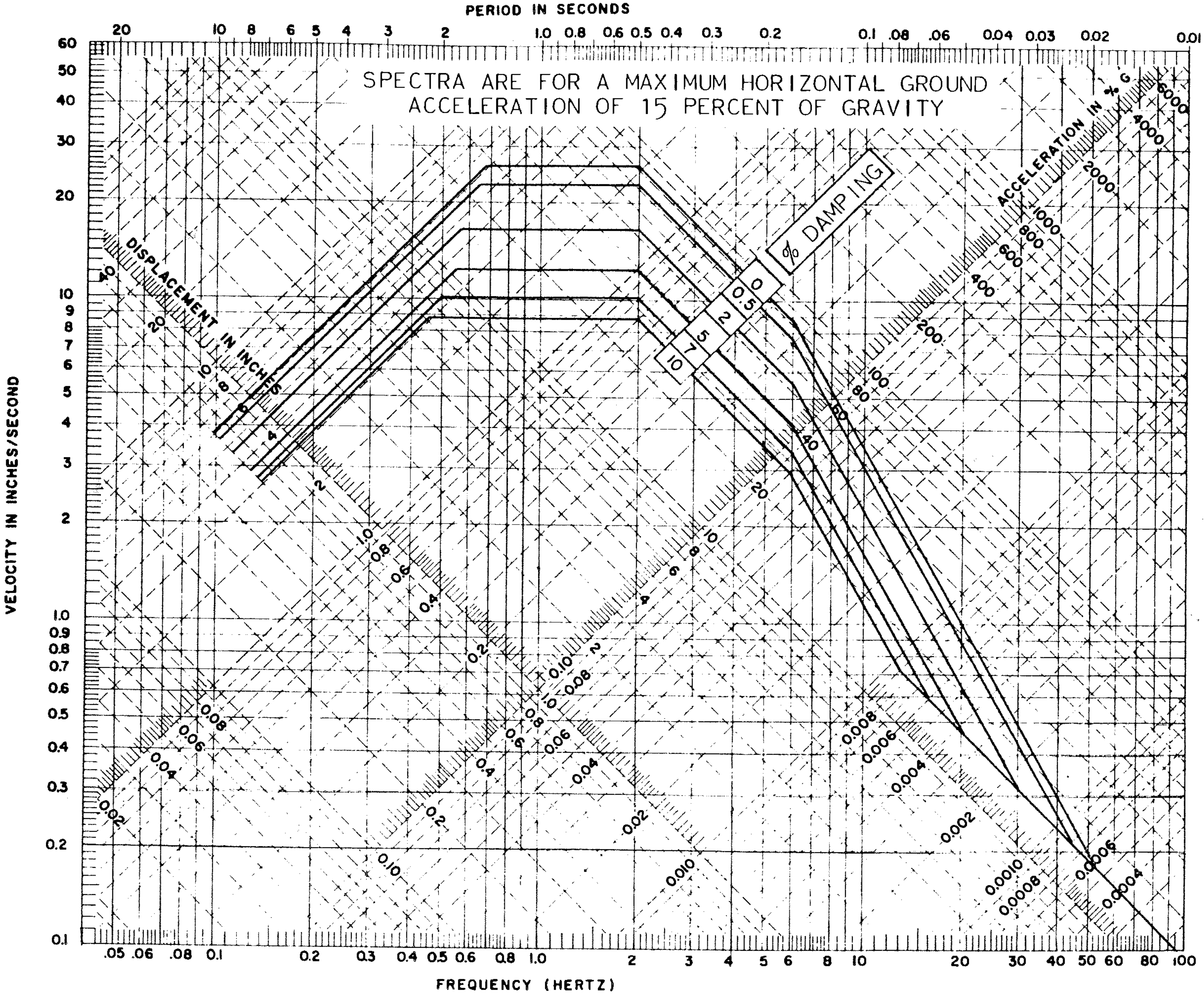
TABLE 3.7-11

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs System, including the following components:		
a. IYM-1780 Reactor Building Foundation Mat Accelerometer/Trigger	0.1 to 40 Hz 0.01 to 1.0g	1*
b. IYM-1784 Reactor Building Ring Girder Accelerometer	0.1 to 40 Hz 0.01 to 1.0g	1
2. Triaxial Seismic Switch		
a. IYM-1782 Reactor Building Foundation Mat	0.1 to 30 Hz 0.01 to 0.25g	1*
3. Triaxial Response-Spectrum Recorders		
a. IYM-1783 - Reactor Building Foundation Mat	(1)	1*
b. IYM-1785 - Steam Generator Support	(1)	1
c. IYM-1786 - Intermediate Bldg., Elev. 463’	(1)	1
d. IYM-1787 - Auxiliary Bldg. Foundation	(1)	1

* With control room indication and/or alarm.

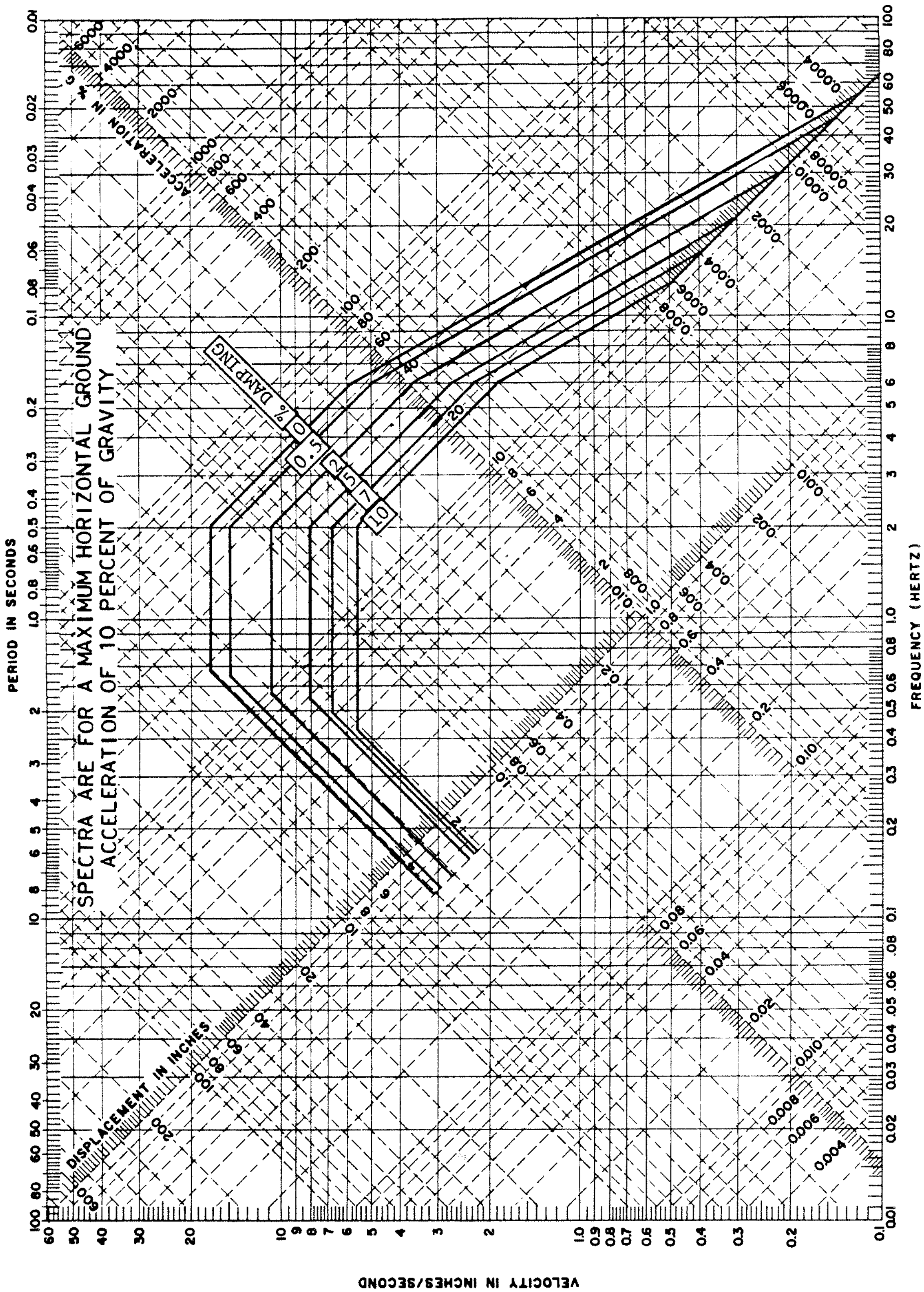
(1) Range varies for the multiple elements of the instrument, i.e., 1.6g at 2 Hz, 10g at 5 Hz, 34g at 10 Hz, 12g at 16 Hz.



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectra Safe Shutdown Earthquake, Structures Founded on Rock

Figure 3.7-1



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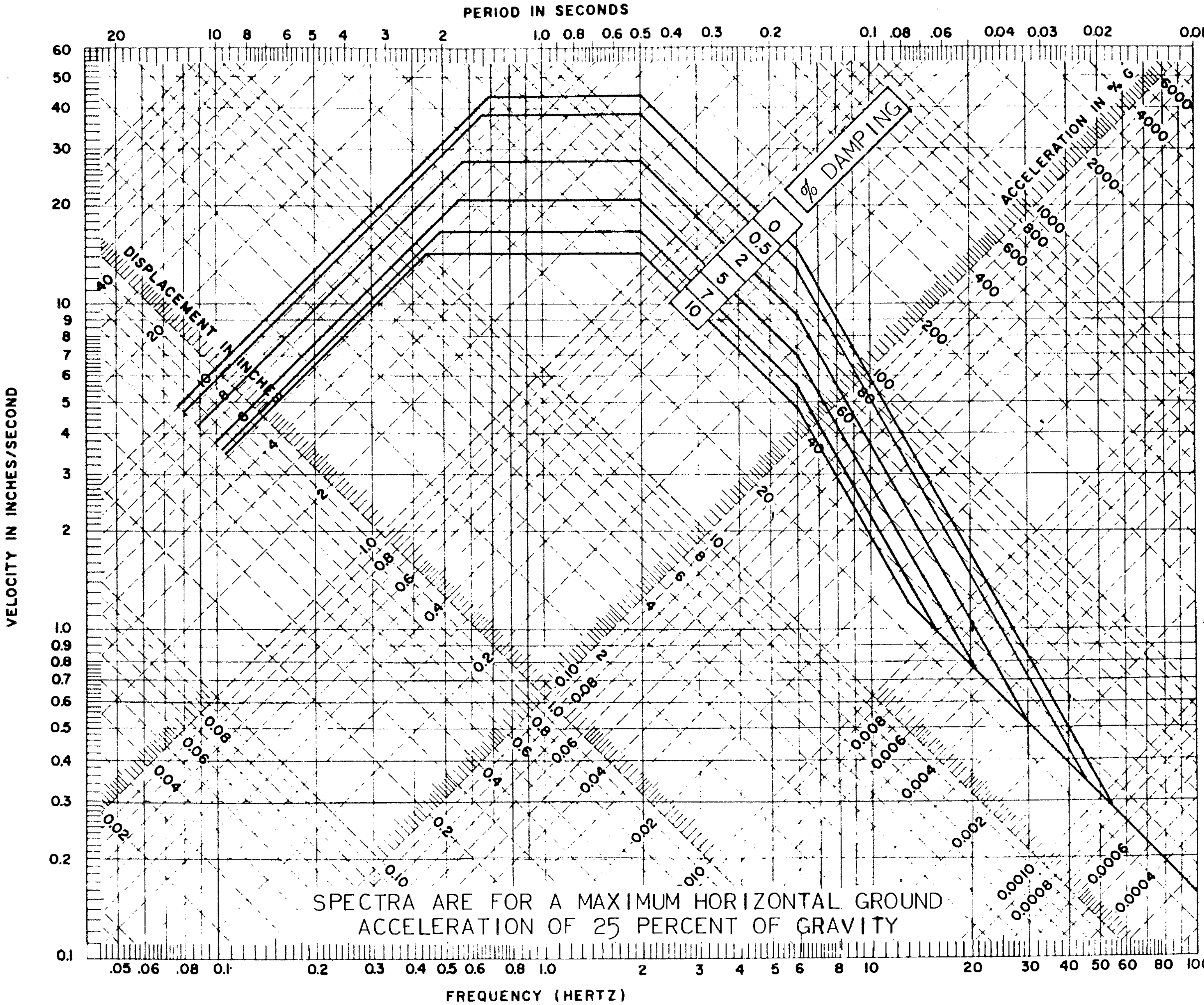
Response Spectra Operating Basis
Earthquake, Structures Founded on Rock

Figure 3.7-2



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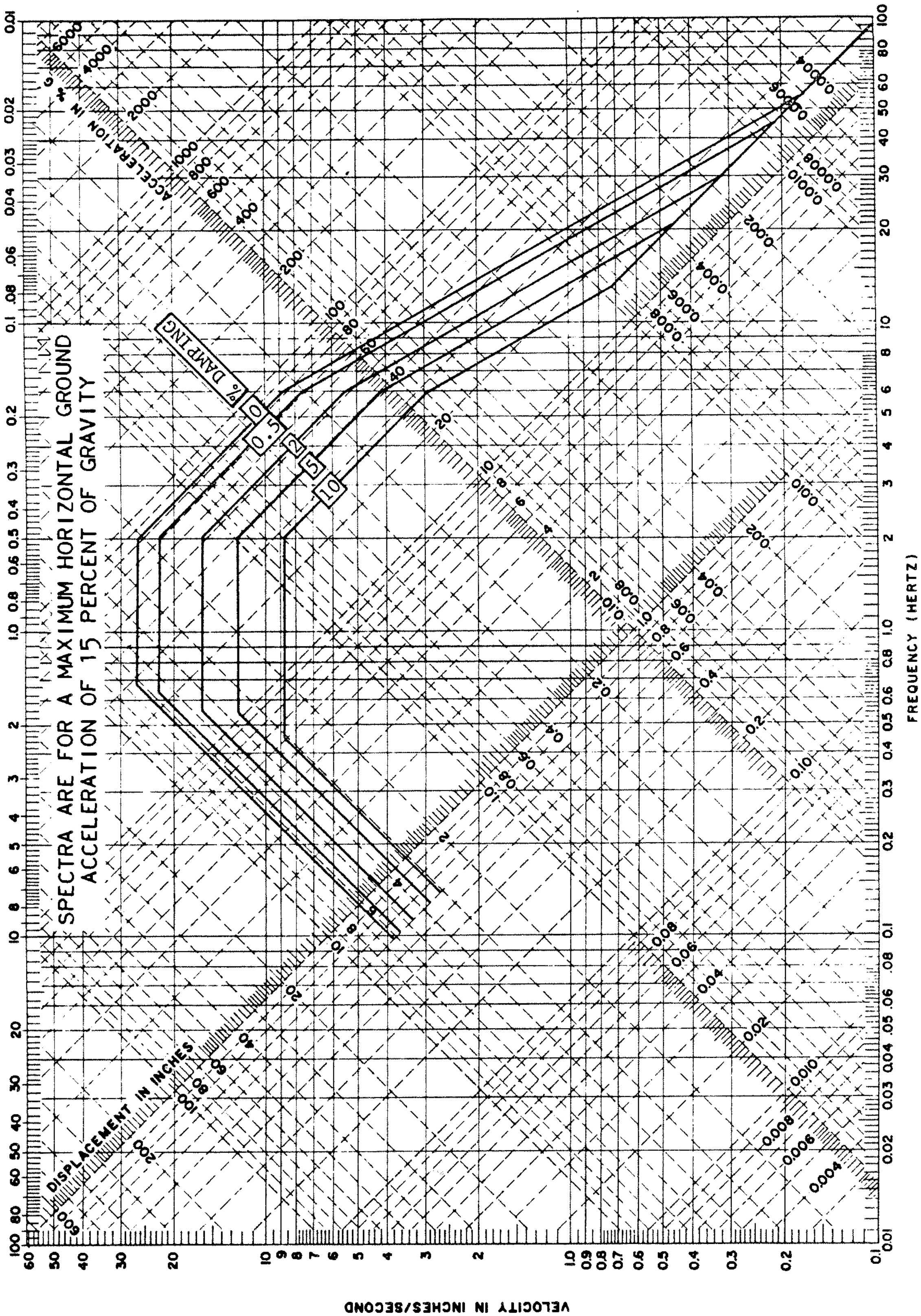
Response Spectra Safe Shutdown Earthquake, Structures Founded on Soil

Figure 3.7-3



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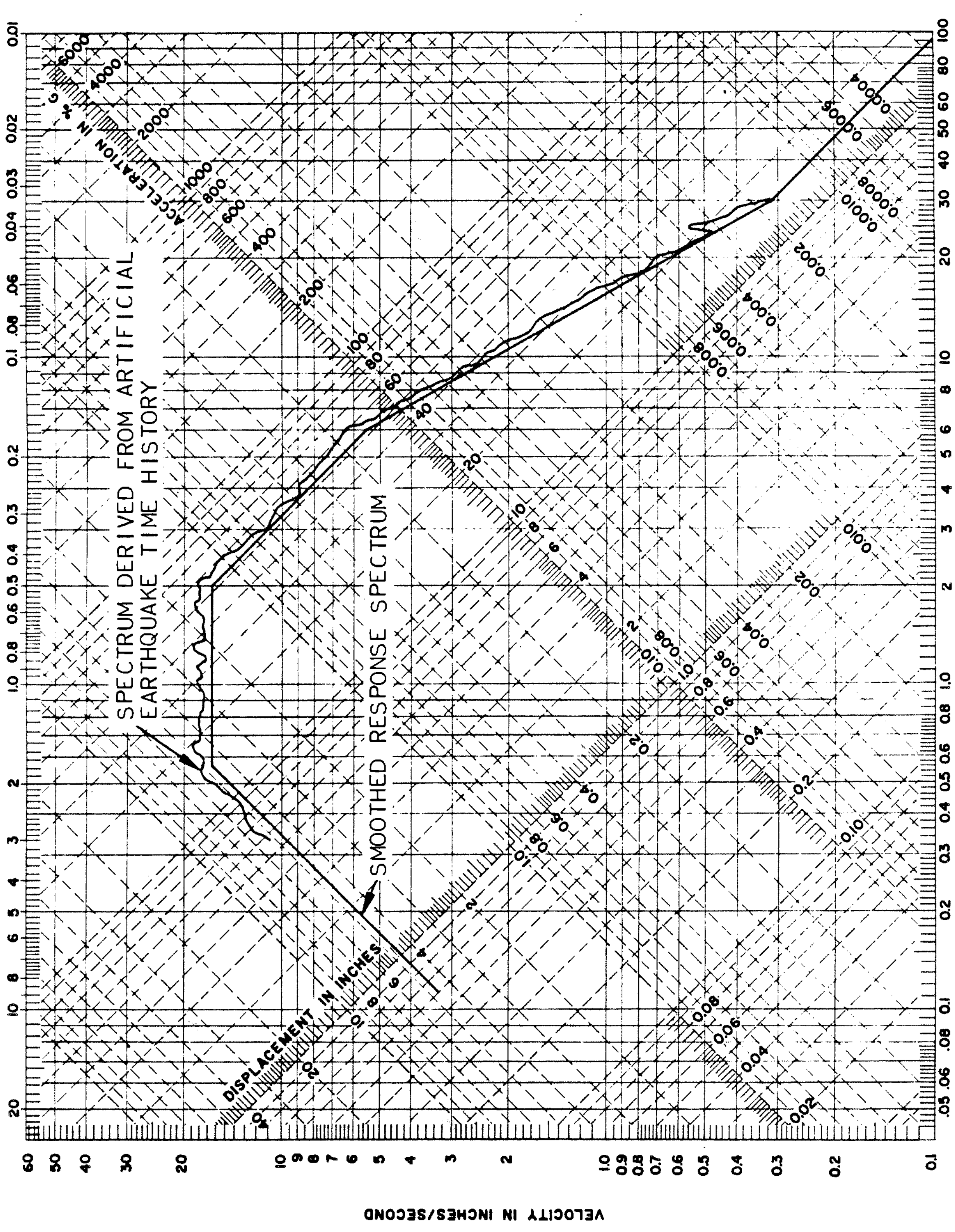
Response Spectra Operating Basis Earthquake, Structures Founded on Soil

Figure 3.7-4



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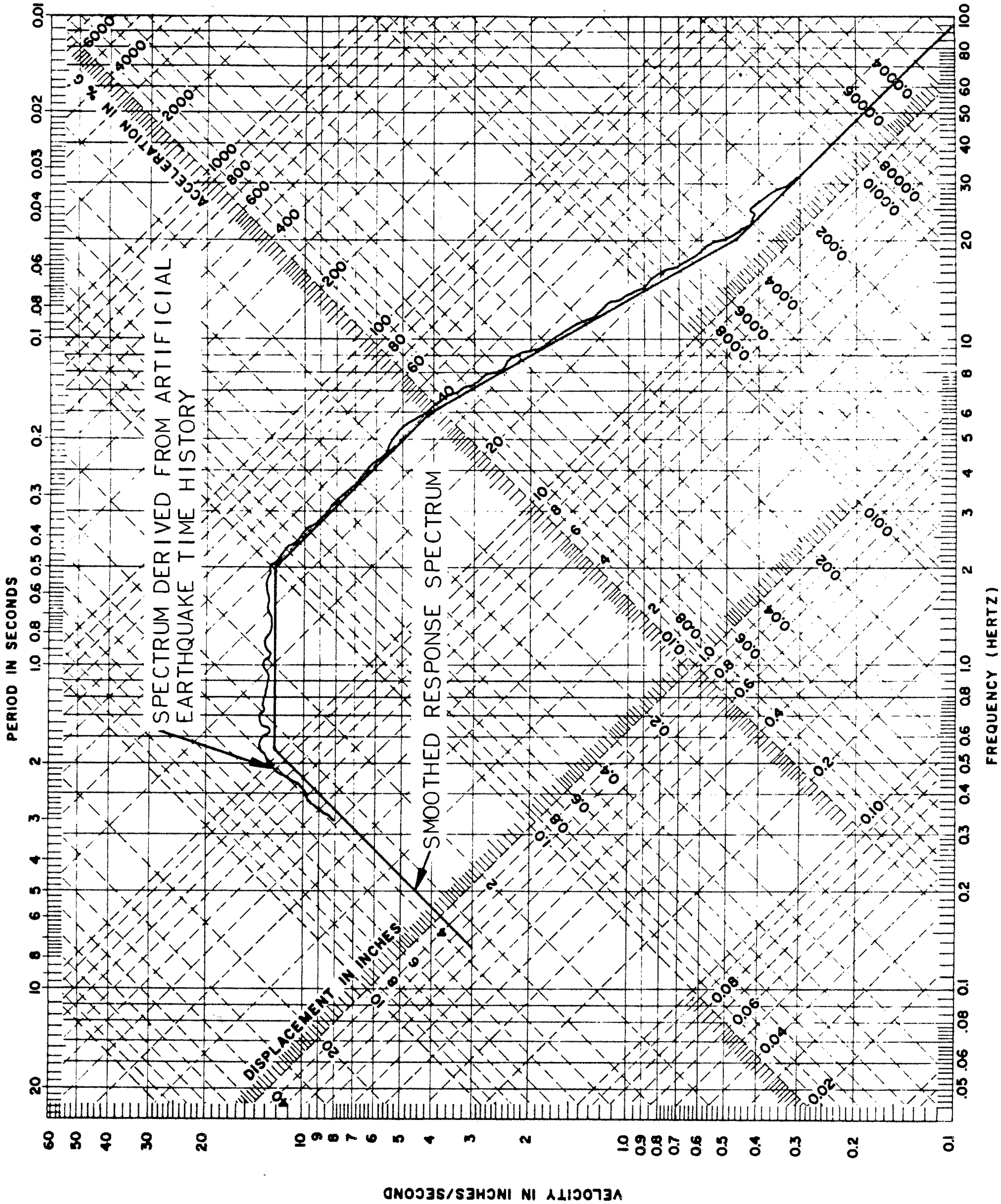
VIRGIL C. SUMMER NUCLEAR STATION

Comparison of Response Spectra at 2 Percent Critical Damping

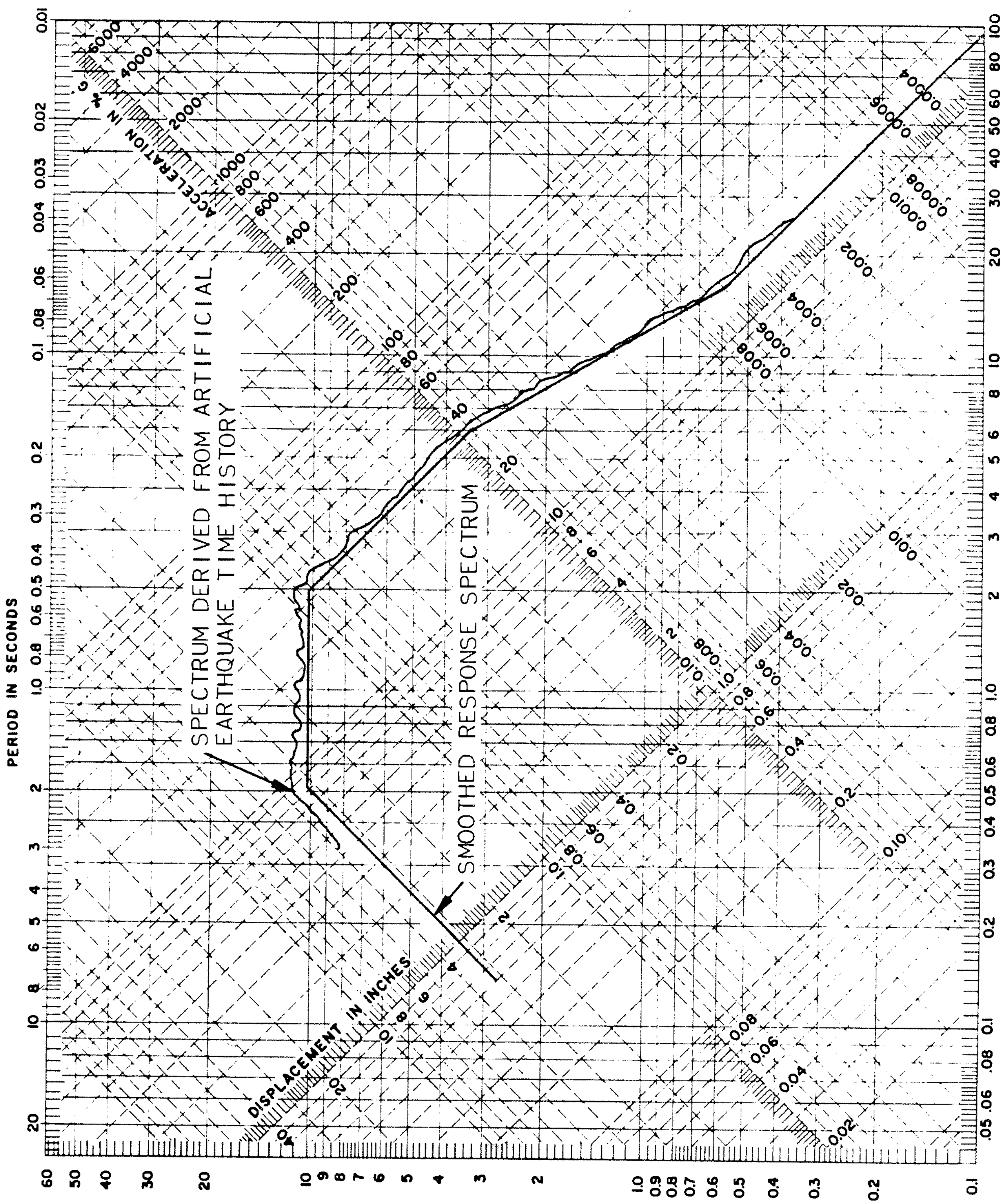
Figure 3.7-5



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Comparison of Response Spectra at 5 Percent Critical Damping

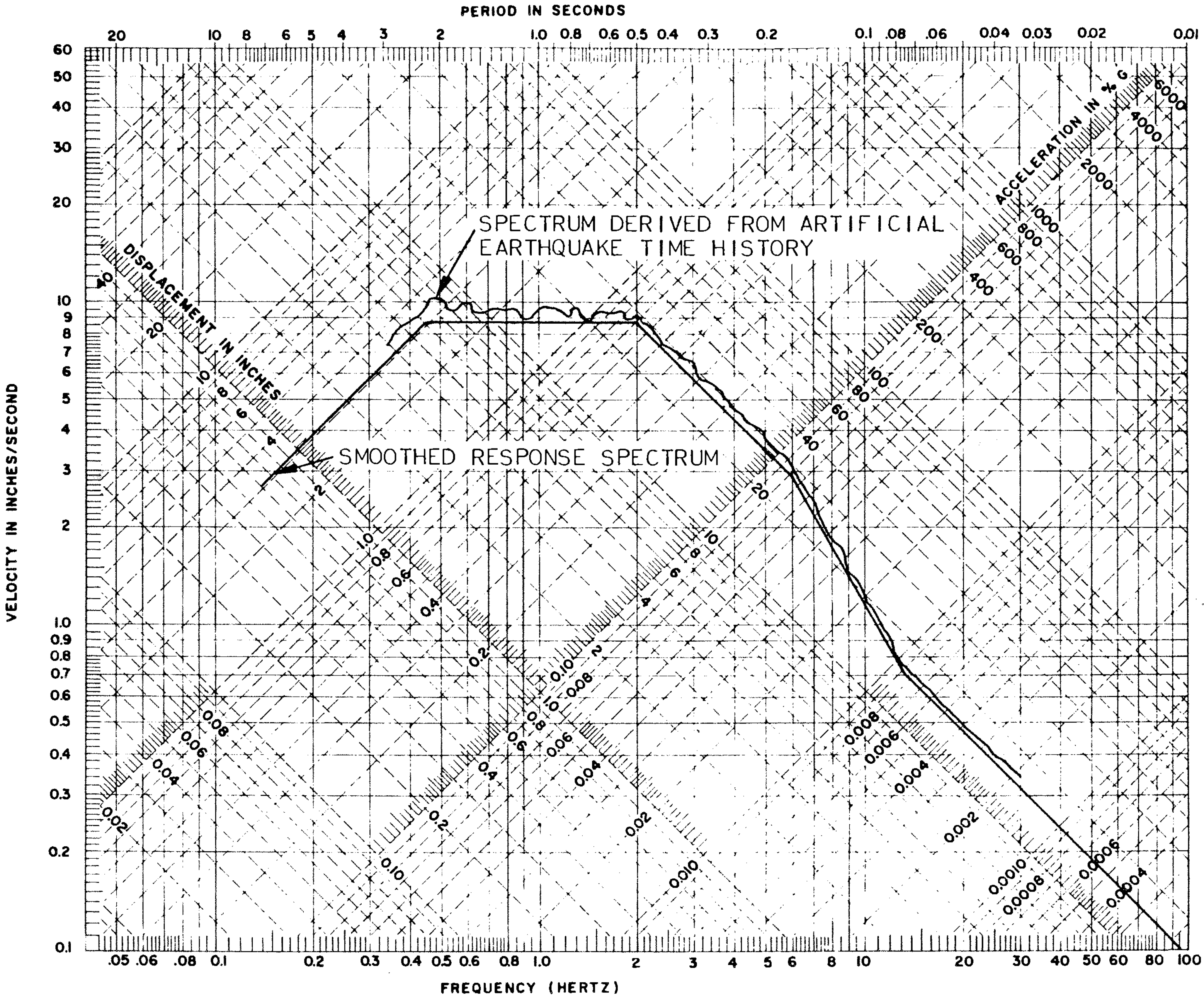
Figure 3.7-6



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Comparison of Response Spectra at 7 Percent Critical Damping

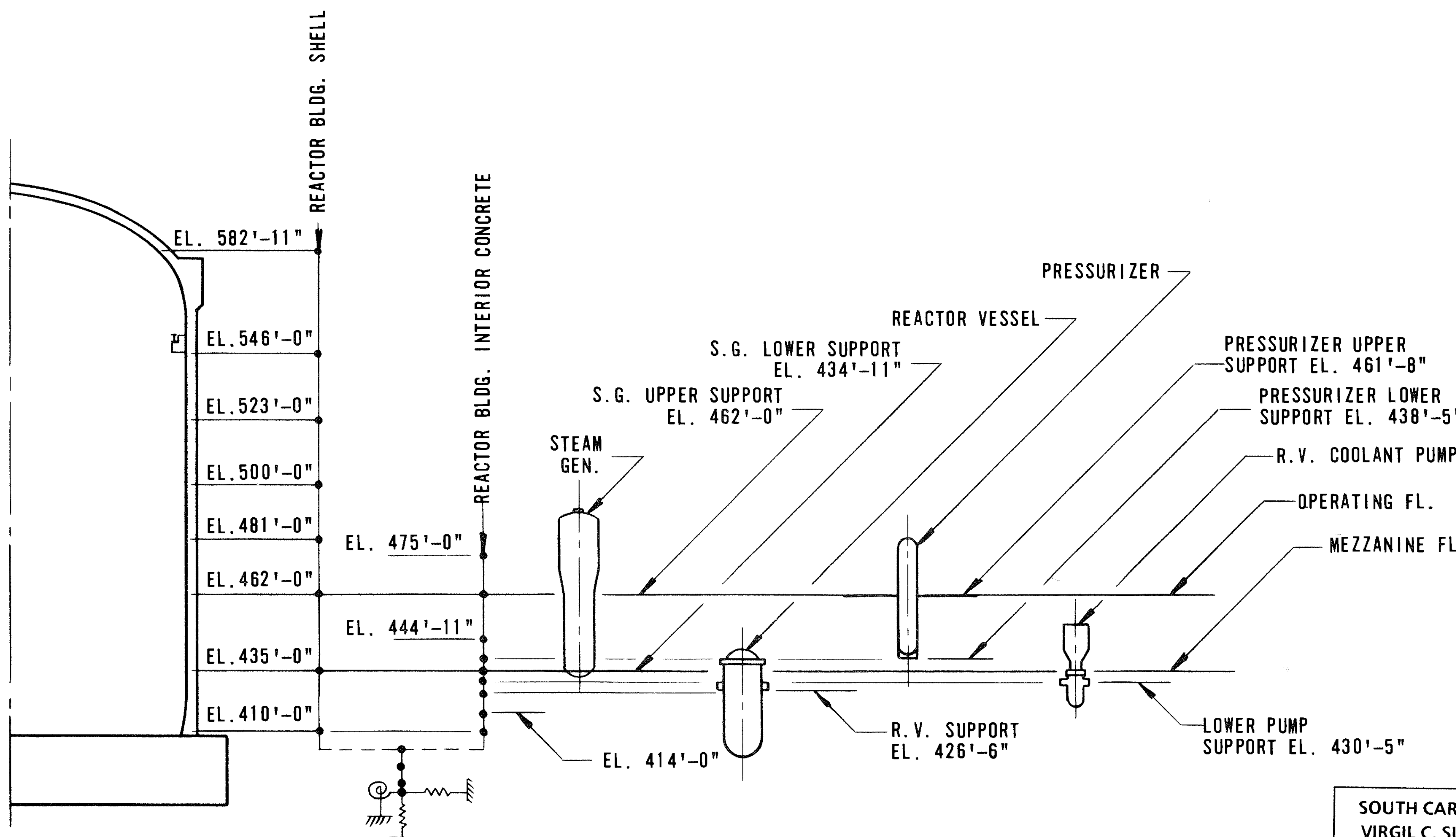
Figure 3.7-7



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Comparison of Response Spectra at 10 Percent Critical Damping

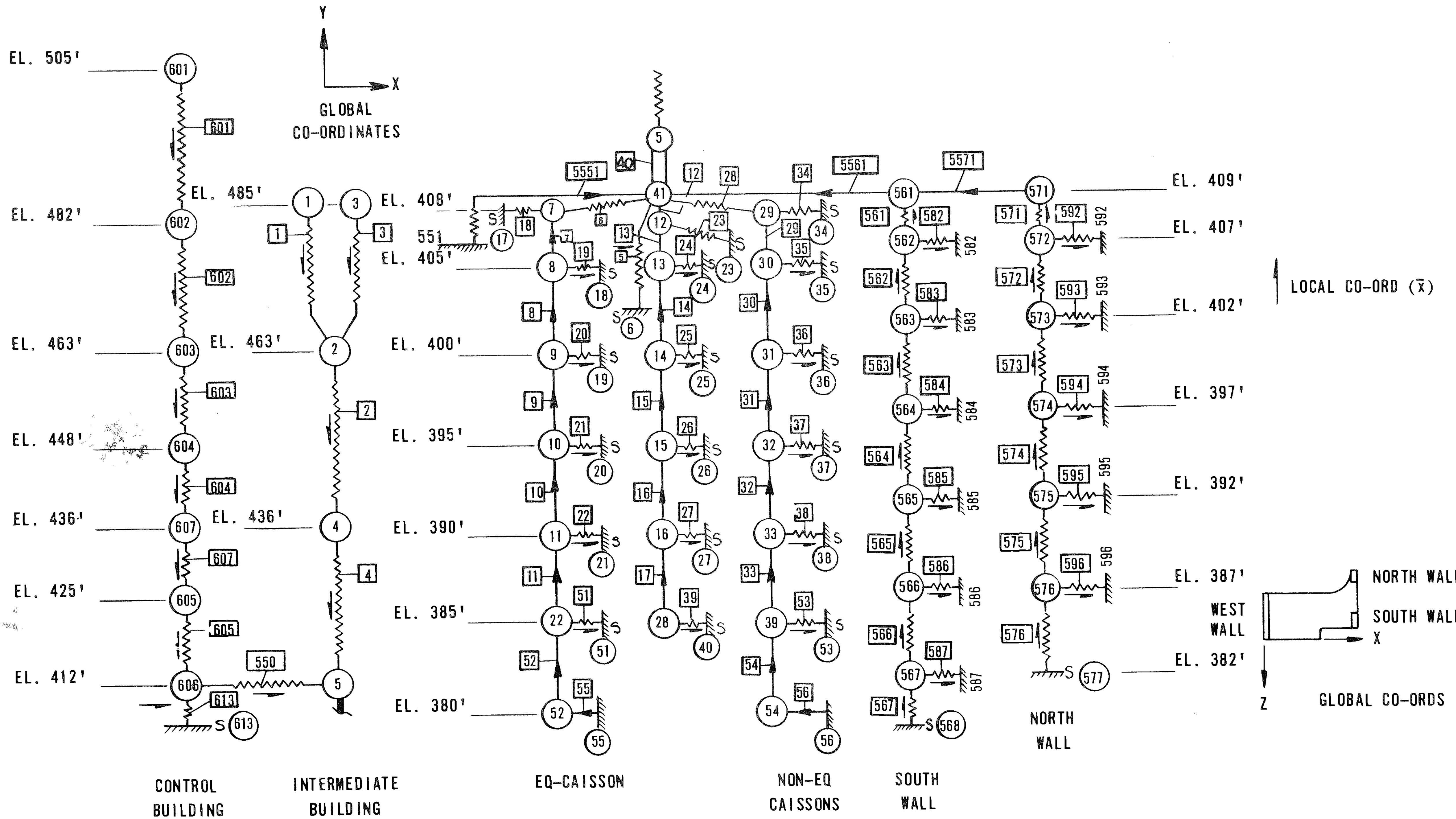
Figure 3.7-8



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Lumped Mass Model Reactor Building Shell and Interior Concrete

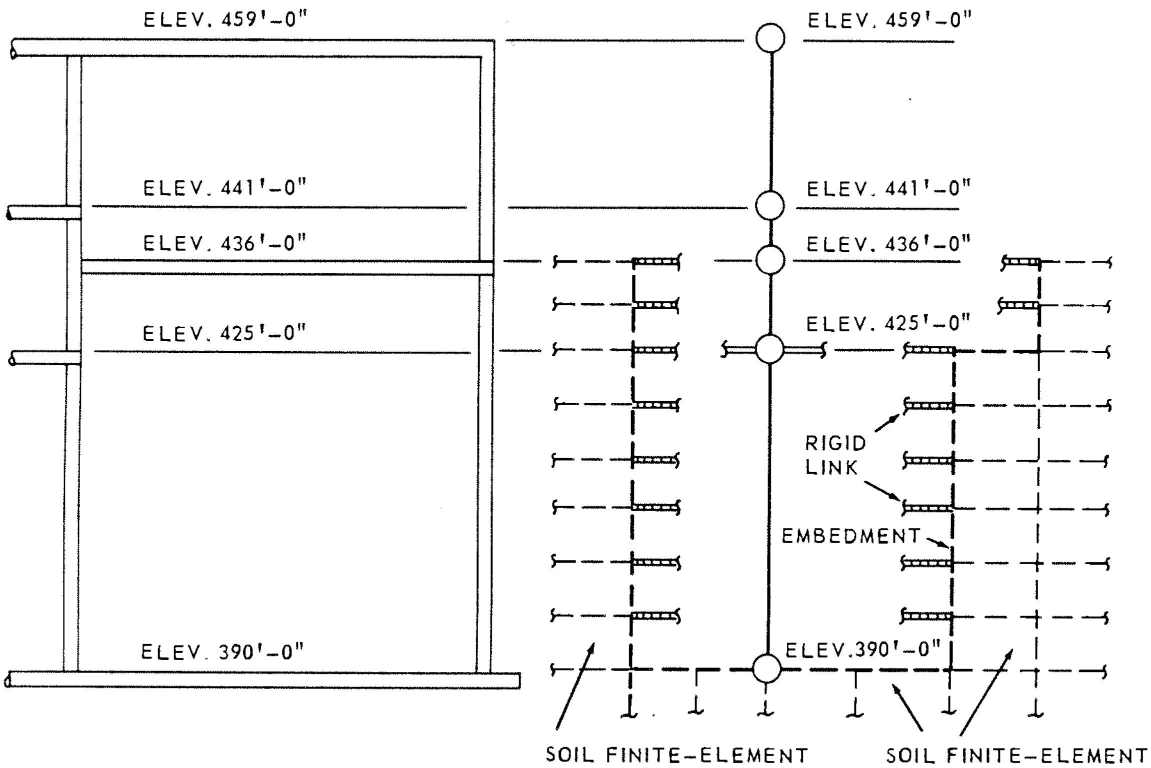
Figure 3.7-9



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Lumped Mass Model of
Intermediate Building

Figure 3.7-10



SOUTH CAROLINA ELECTRIC & GAS CO.
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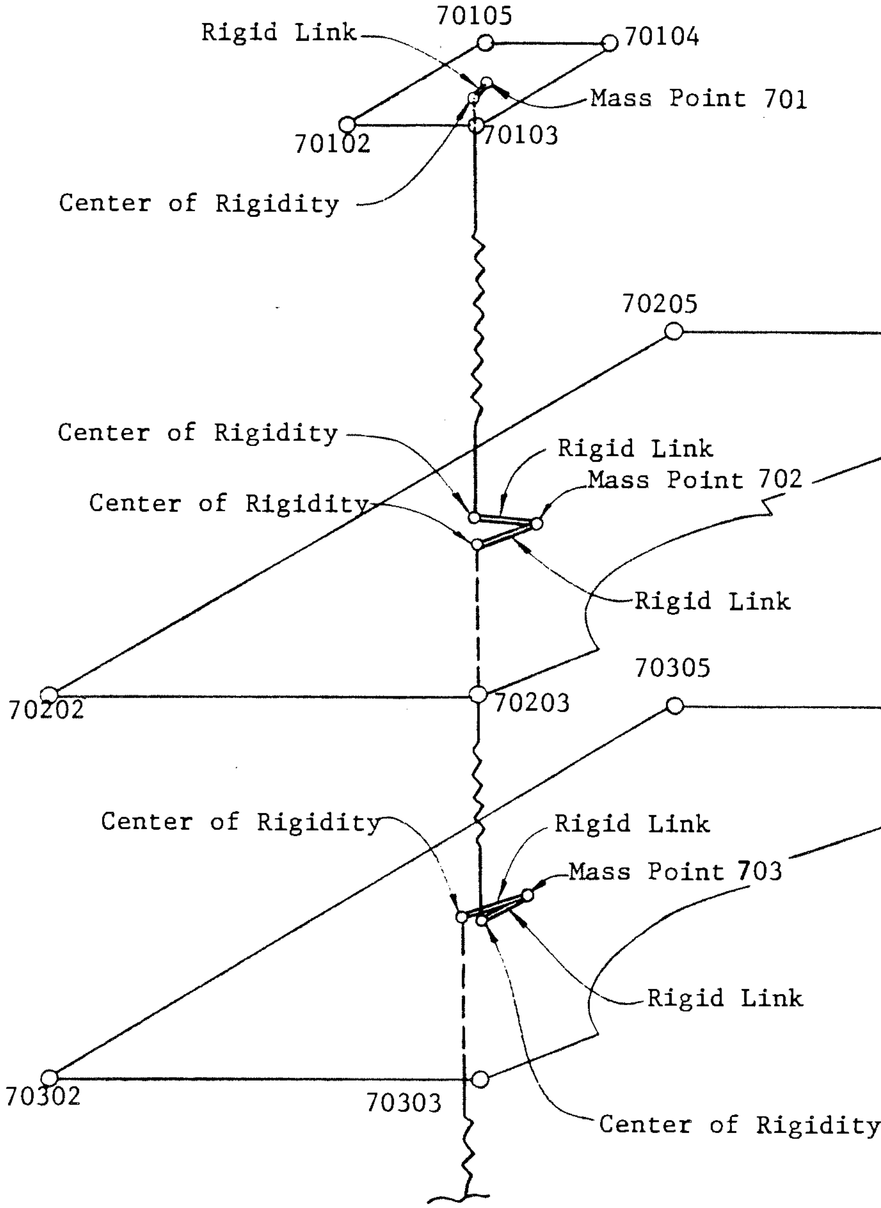
Finite Element Model of the Service Water Pumphouse

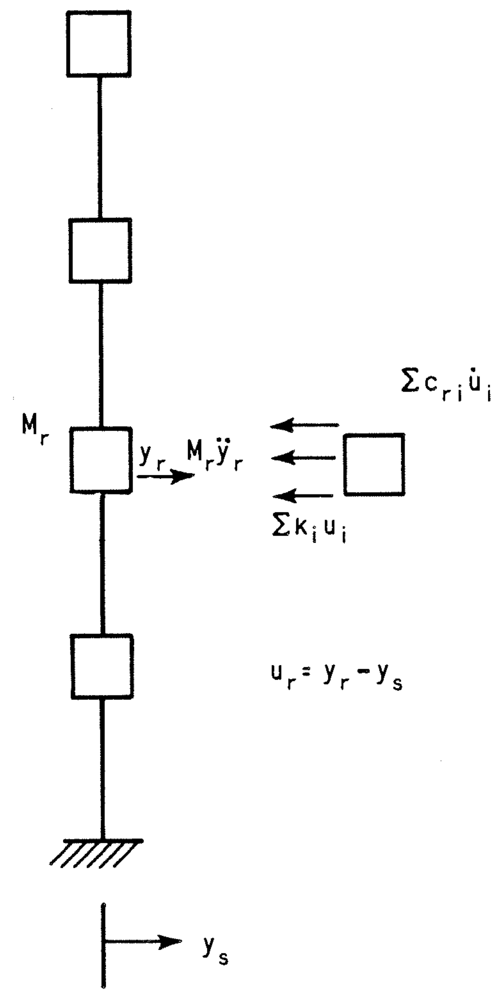
Figure 3.7-11



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Typical Torsional
Model

Figure 3.7-11a



SOUTH CAROLINA ELECTRIC & GAS CO.
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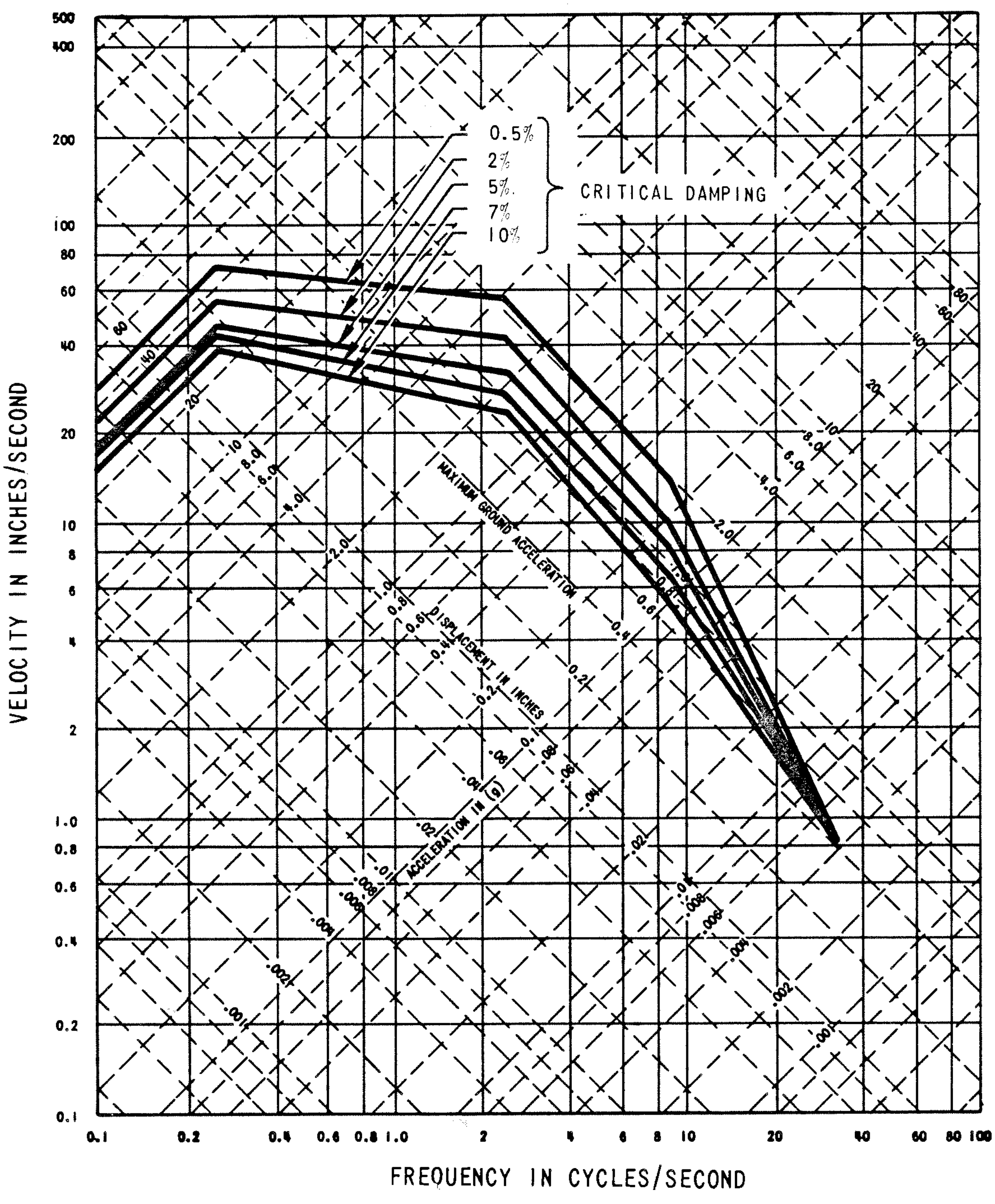
Multi-Degree-of-Freedom System

Figure 3.7-12



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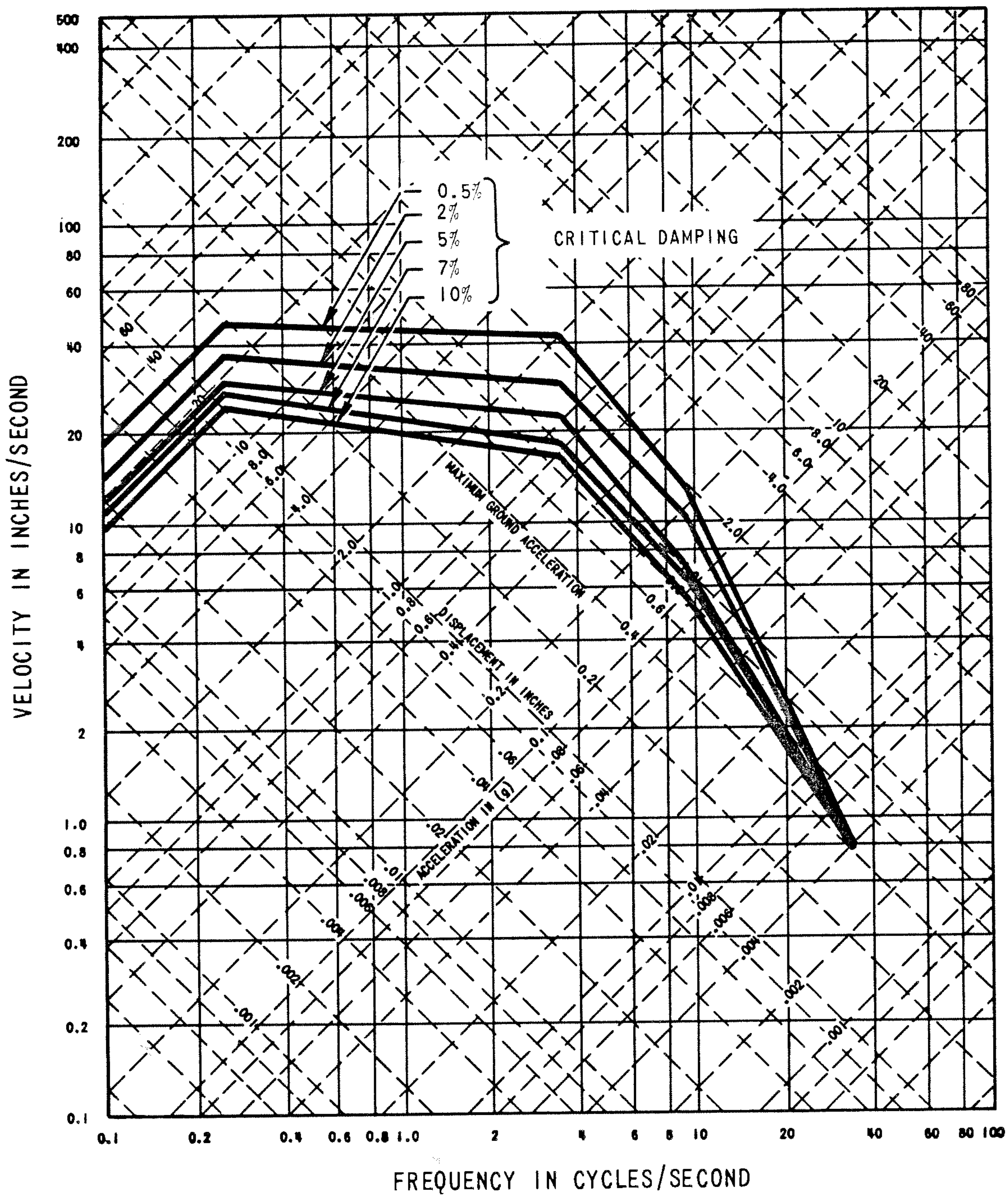
Horizontal Design Response Spectra - Scaled to 0.4g Horizontal Ground Acceleration

Figure 3.7-13



SOUTH CAROLINA ELECTRIC & GAS CO.
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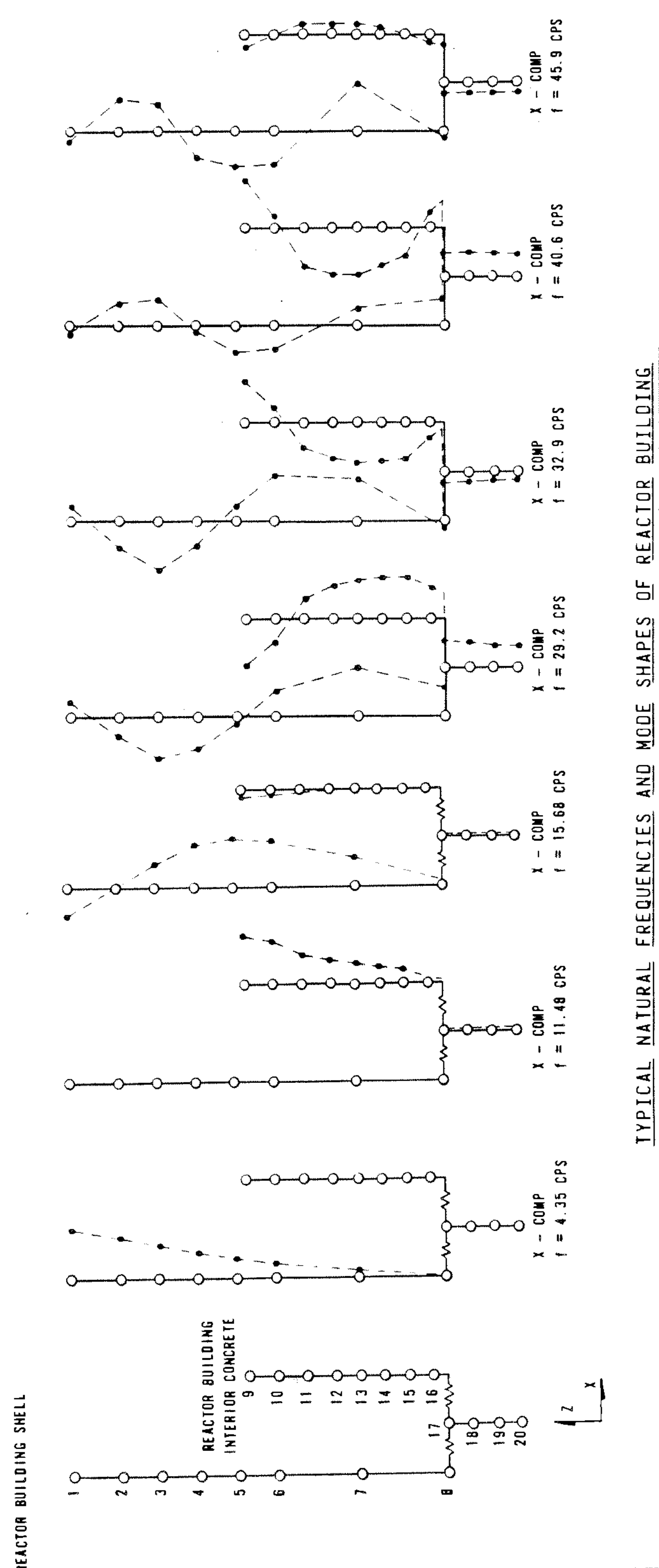
Vertical Design Response Spectra - Scaled to 0.4g Horizontal Ground Acceleration

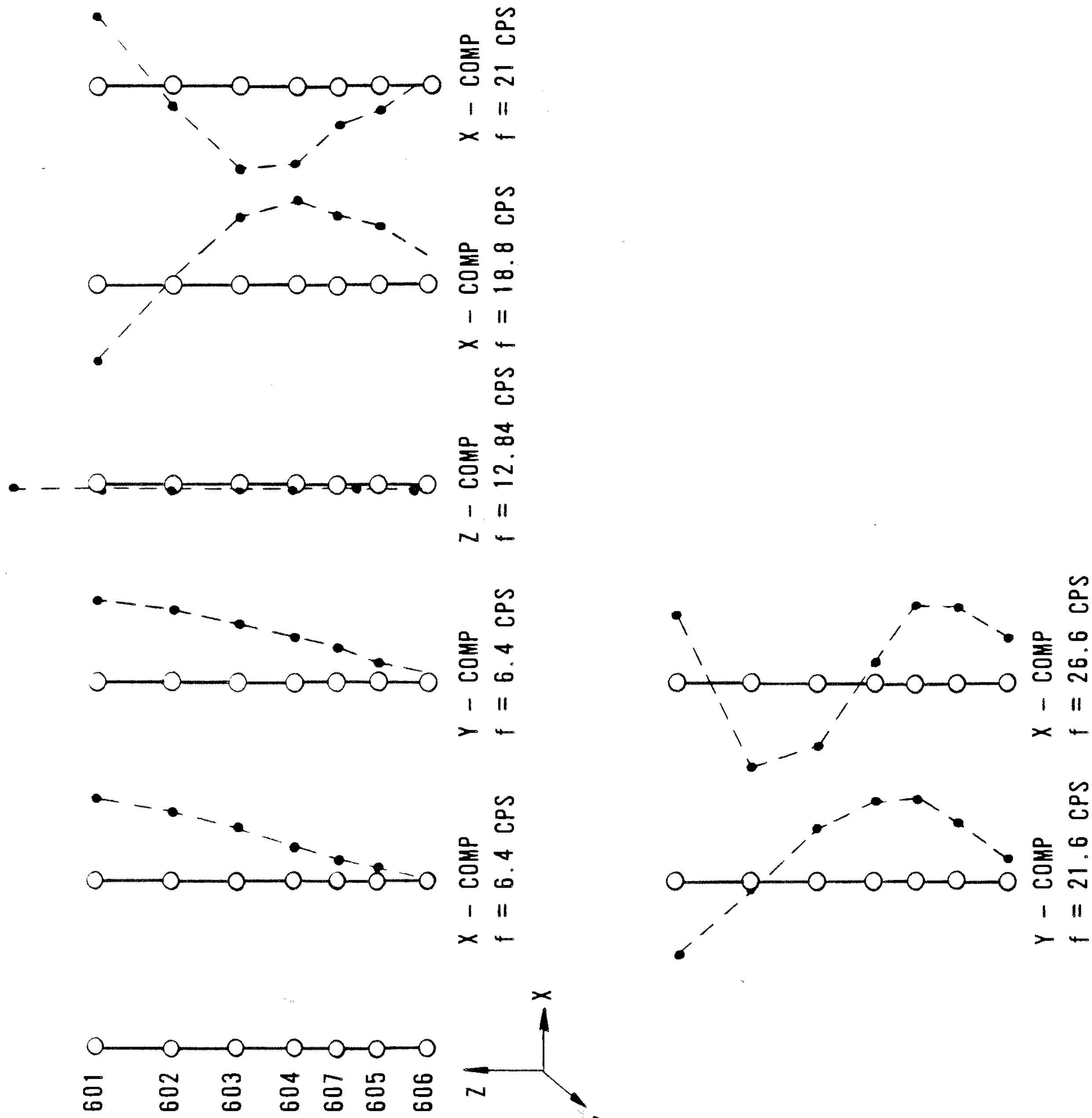
Figure 3.7-14



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Typical Natural Frequencies and Mode Shapes of the Reactor Building

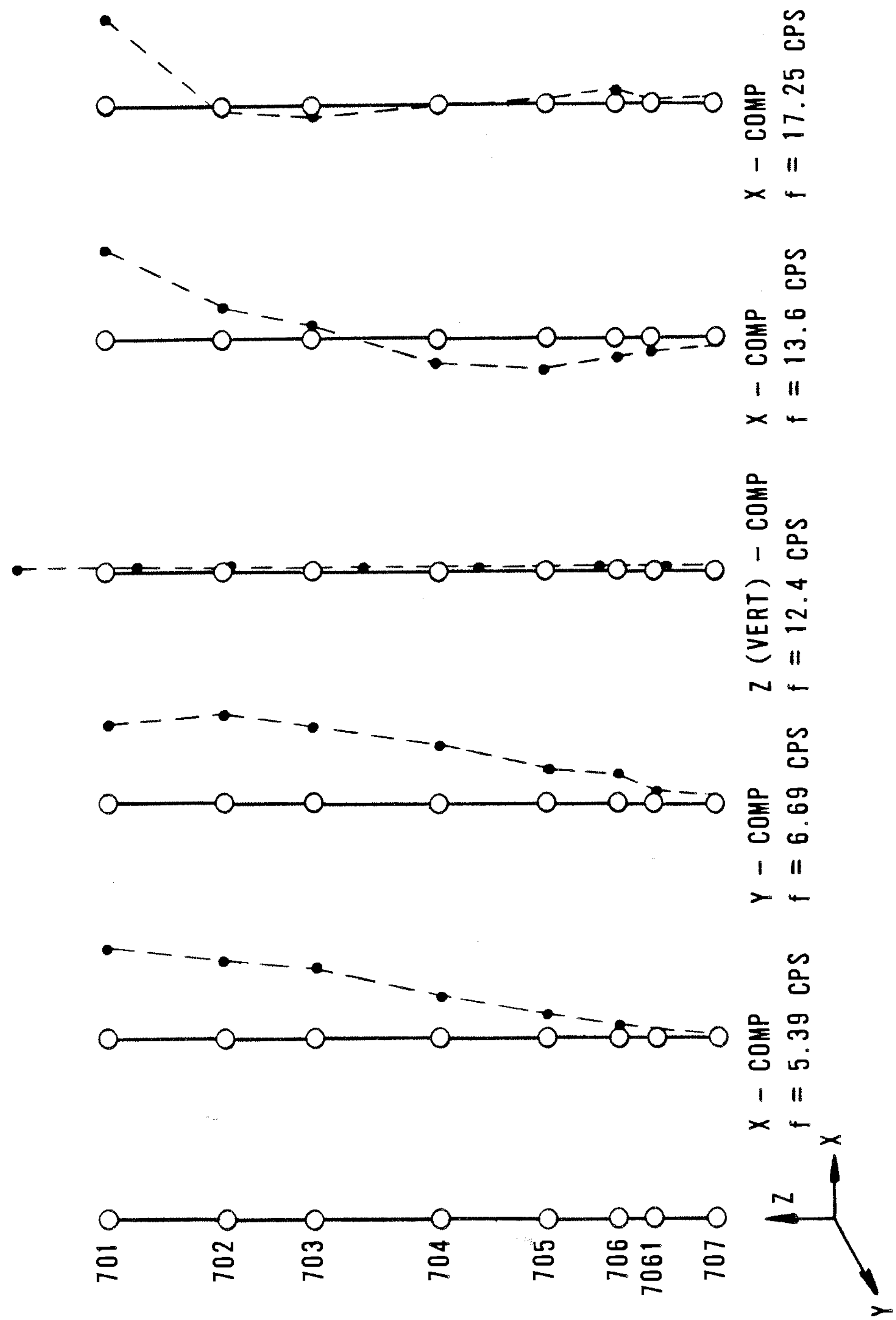
Figure 3.7-15



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Typical Natural Frequencies and Mode Shapes of the Control Building

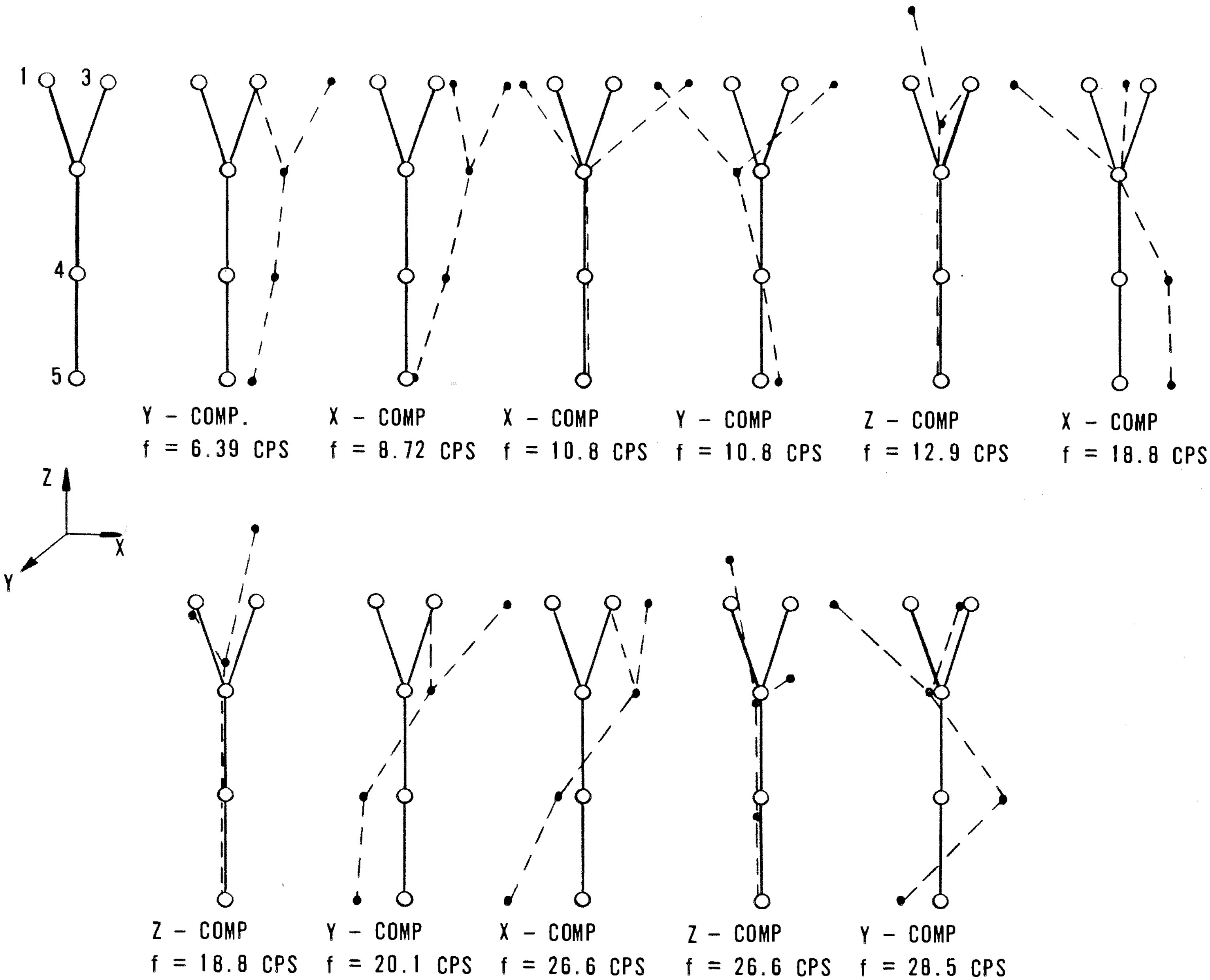
Figure 3.7-16



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Typical Natural Frequencies and Mode Shapes of the Auxiliary Building

Figure 3.7-17



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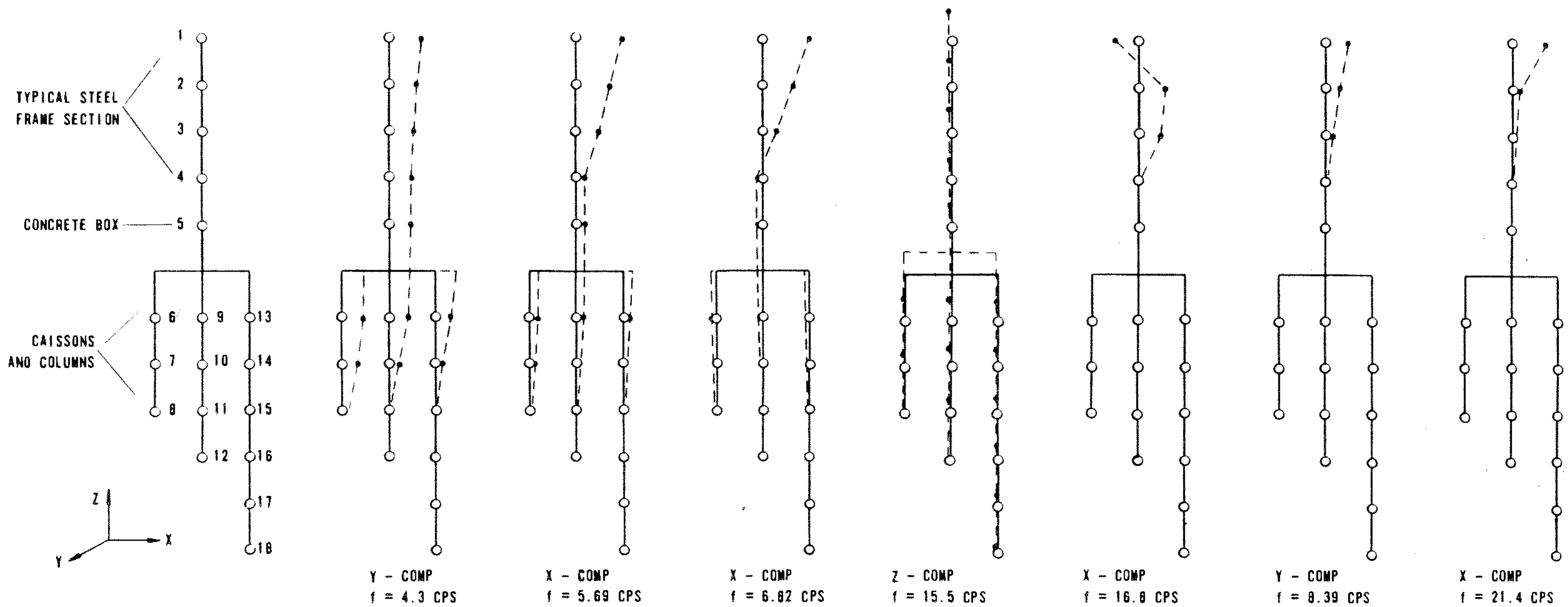
Typical Natural Frequencies and Mode Shapes of the Intermediate Building

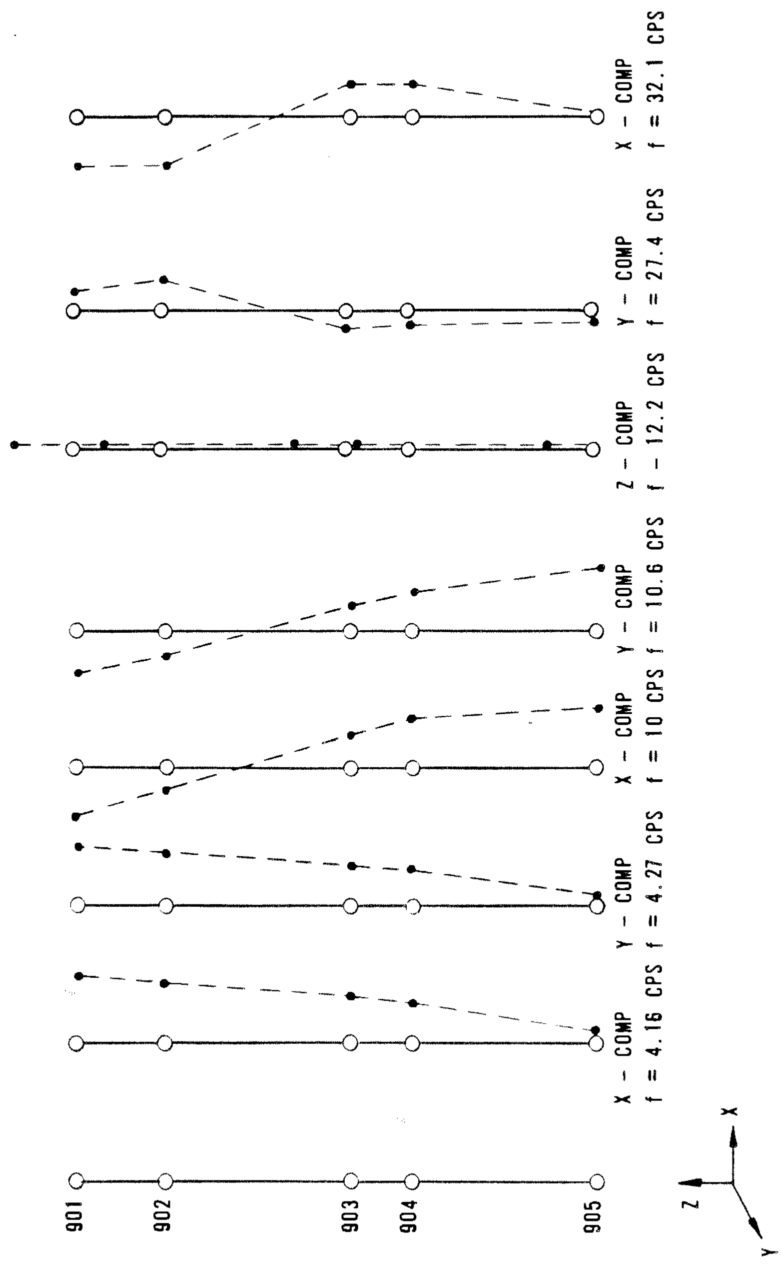
Figure 3.7-18



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Typical Natural Frequencies and Mode Shapes of the Fuel Handling Building

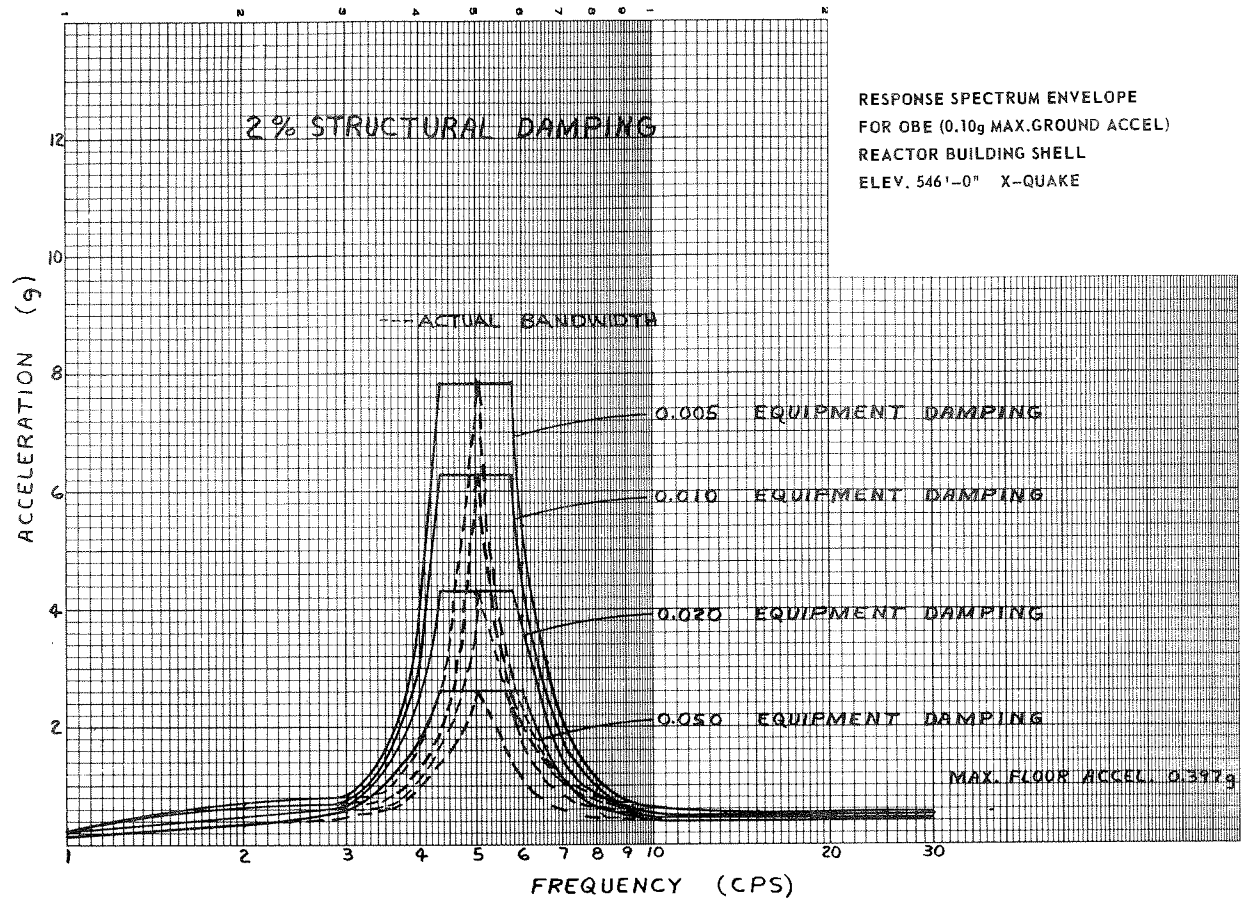
Figure 3.7-19

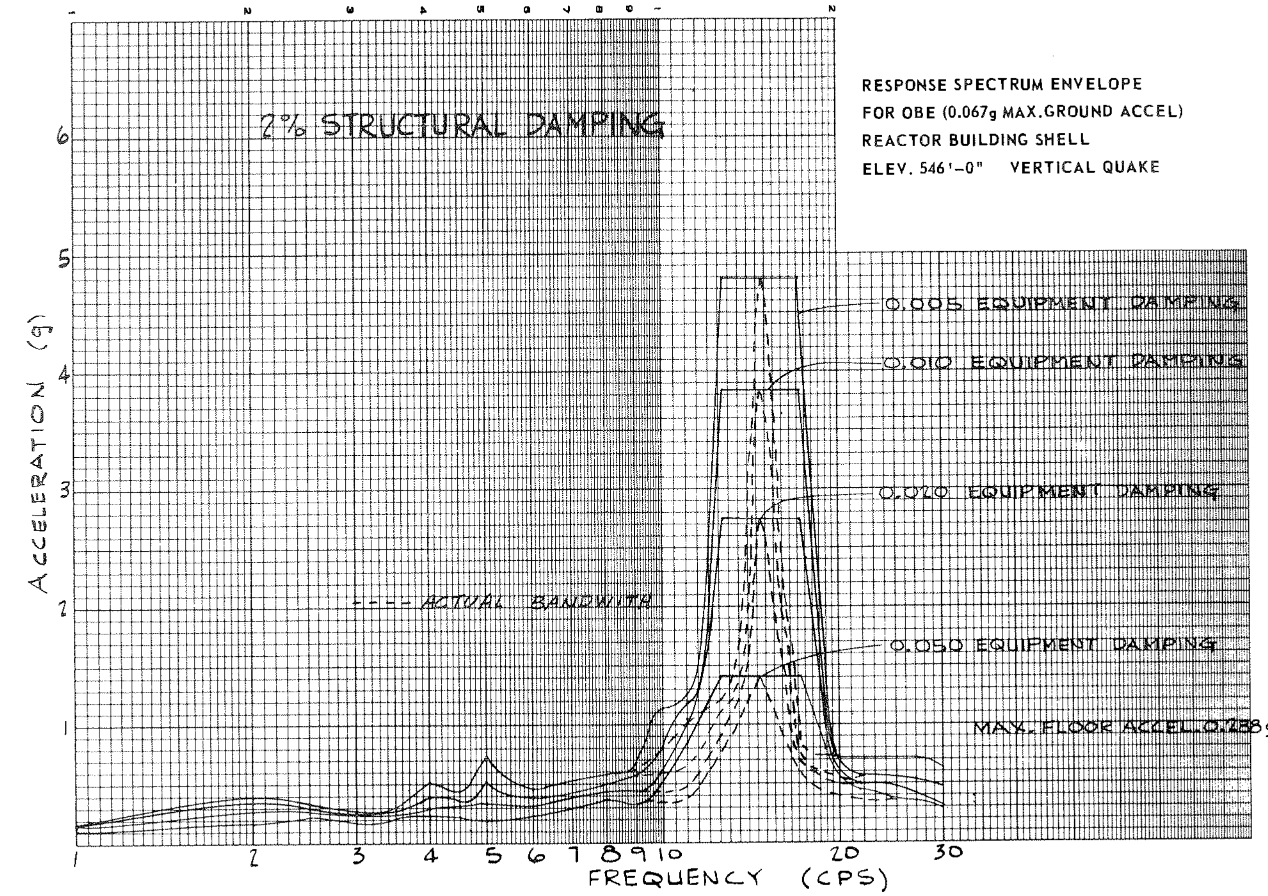


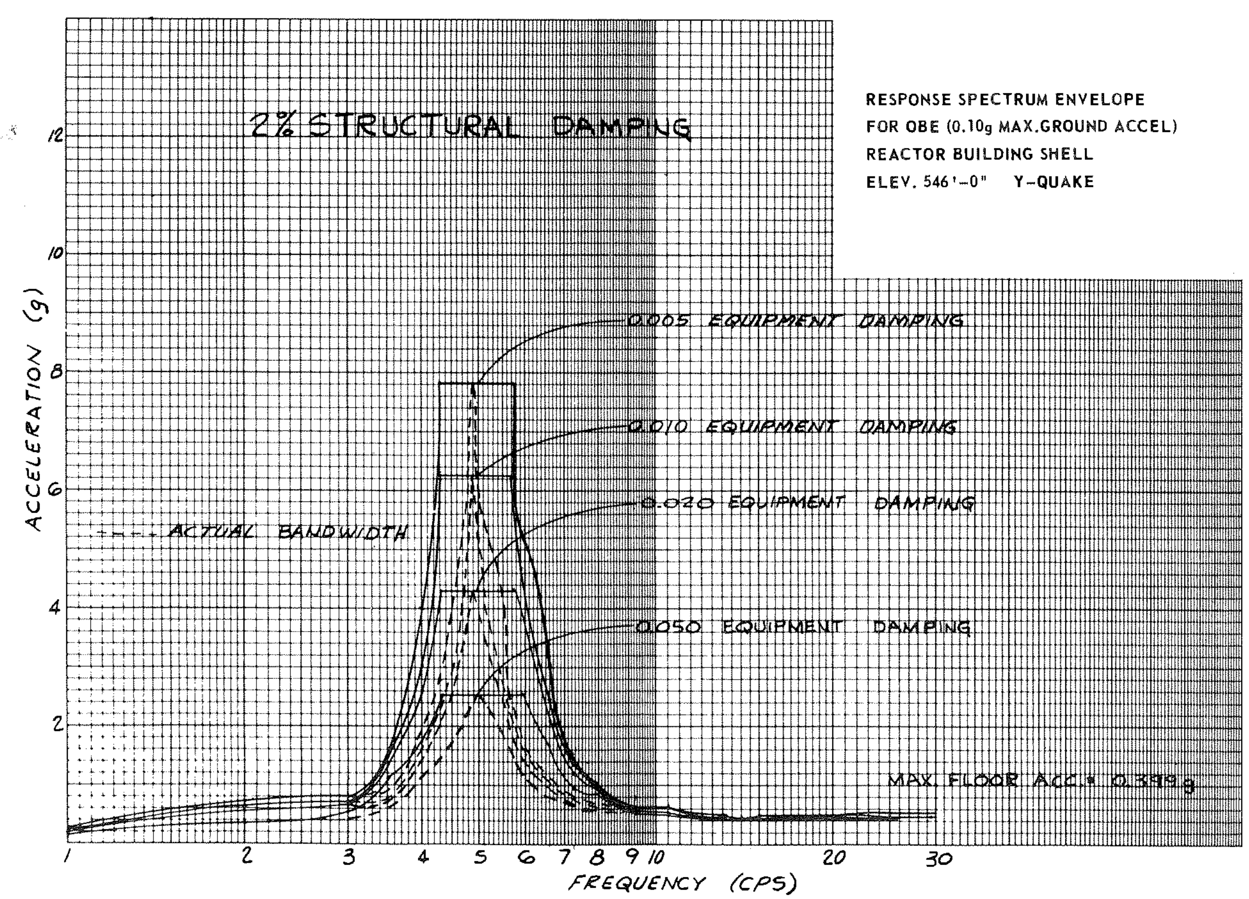
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Typical Natural Frequencies and Mode Shapes of the Diesel Generator Building

Figure 3.7-20



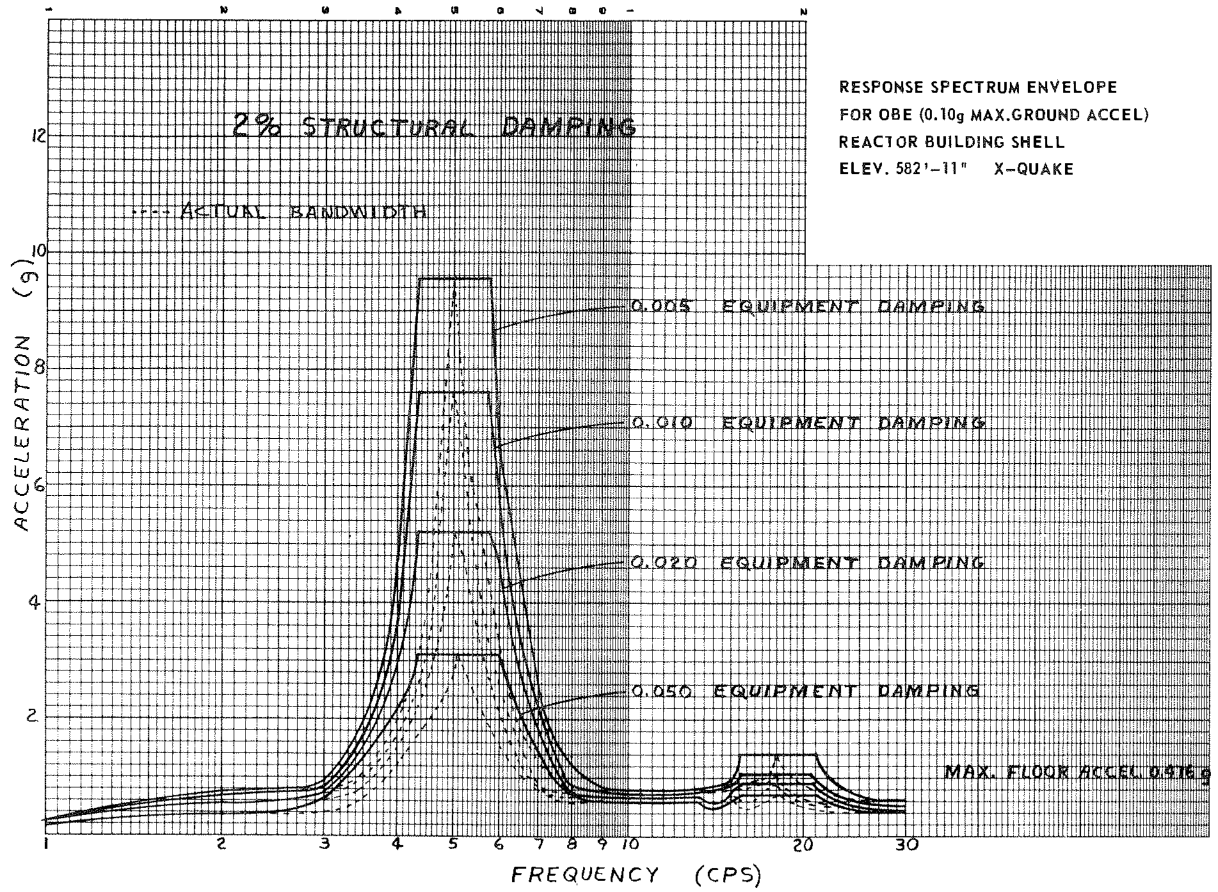


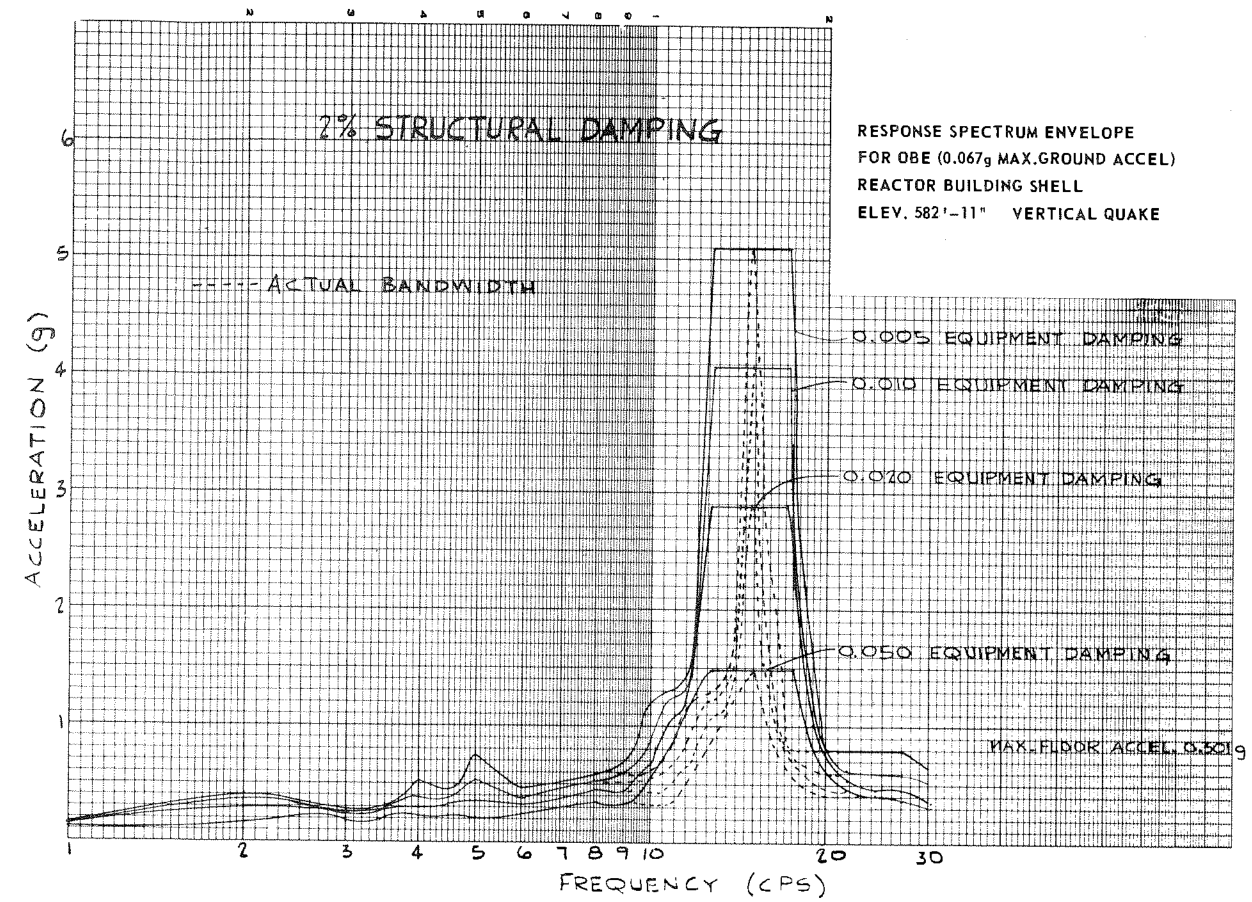


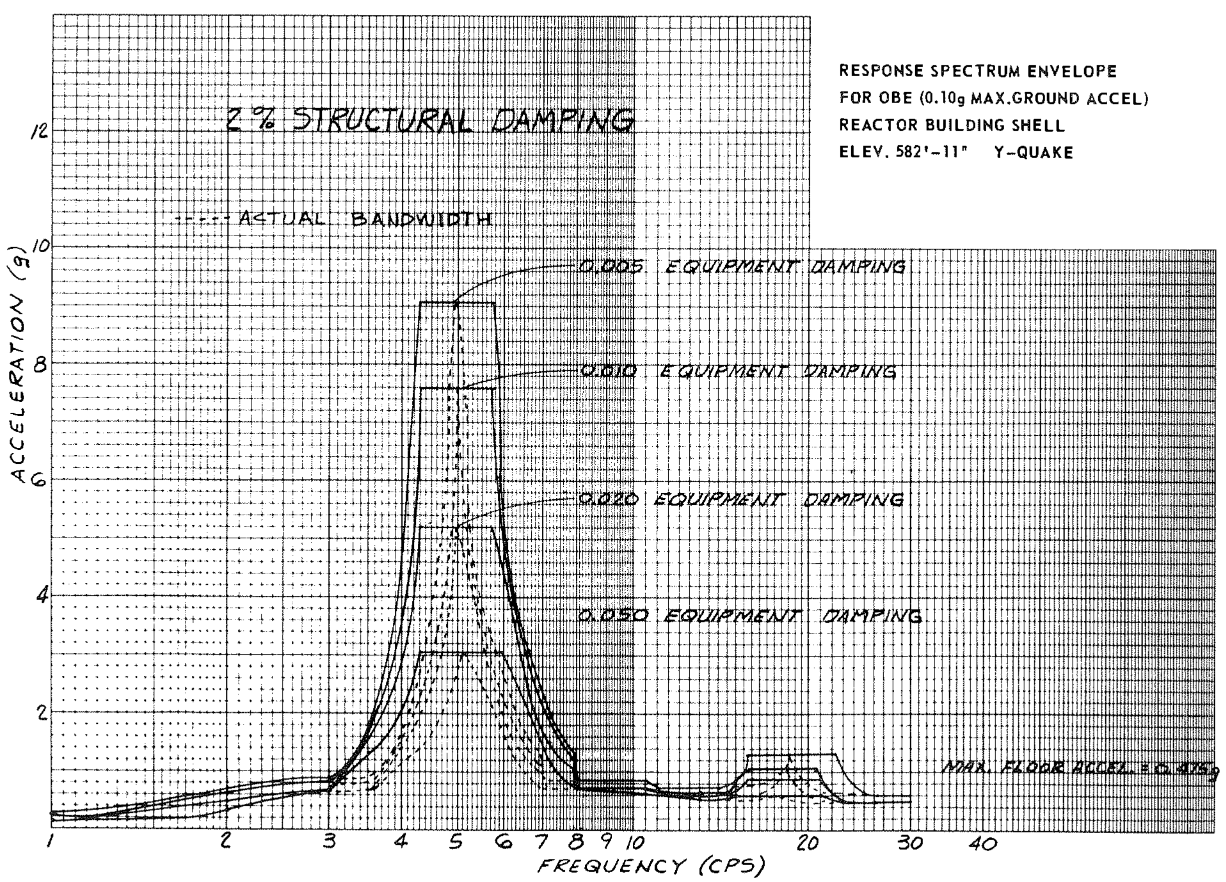
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes Reactor Building Shell Elevation 546' - 0"

Figure 3.7-21



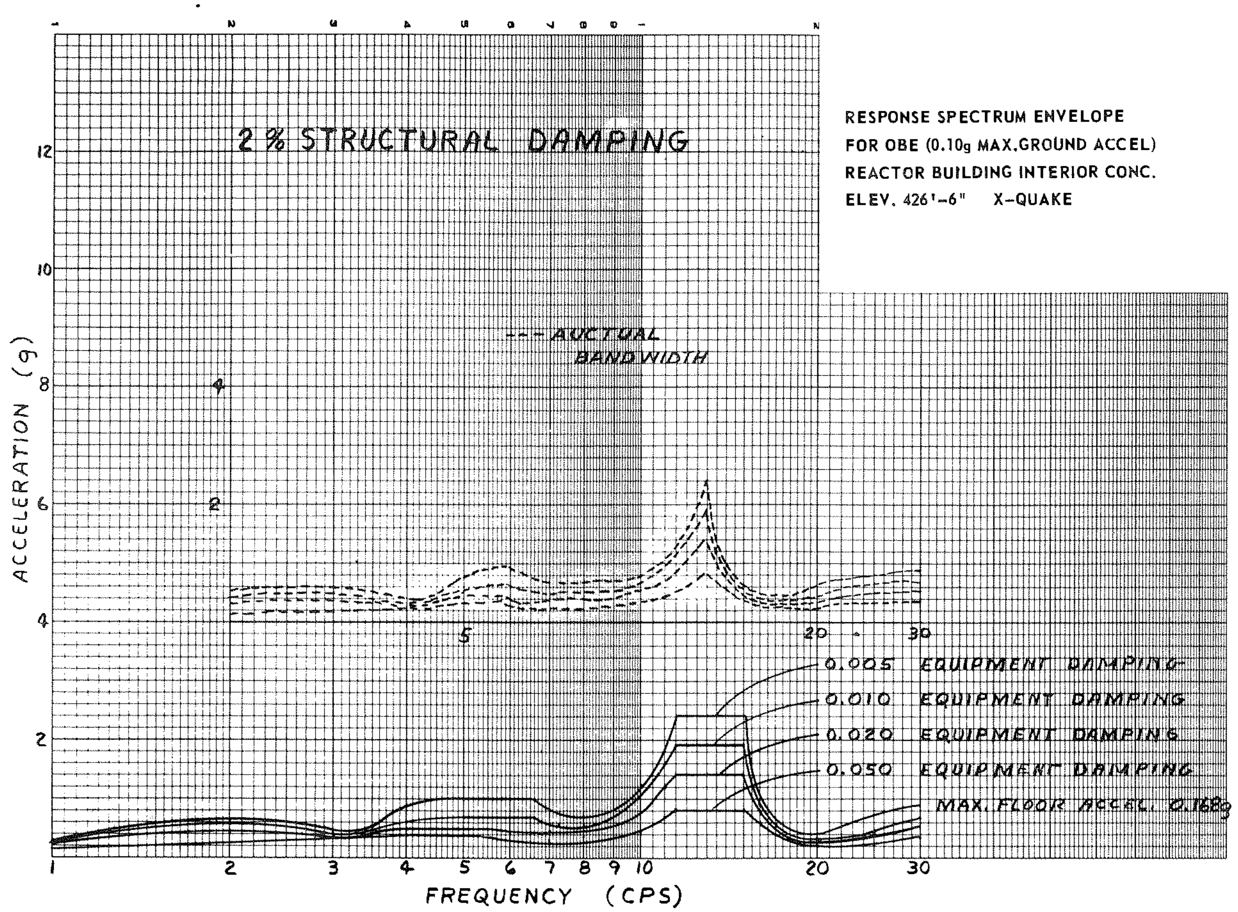


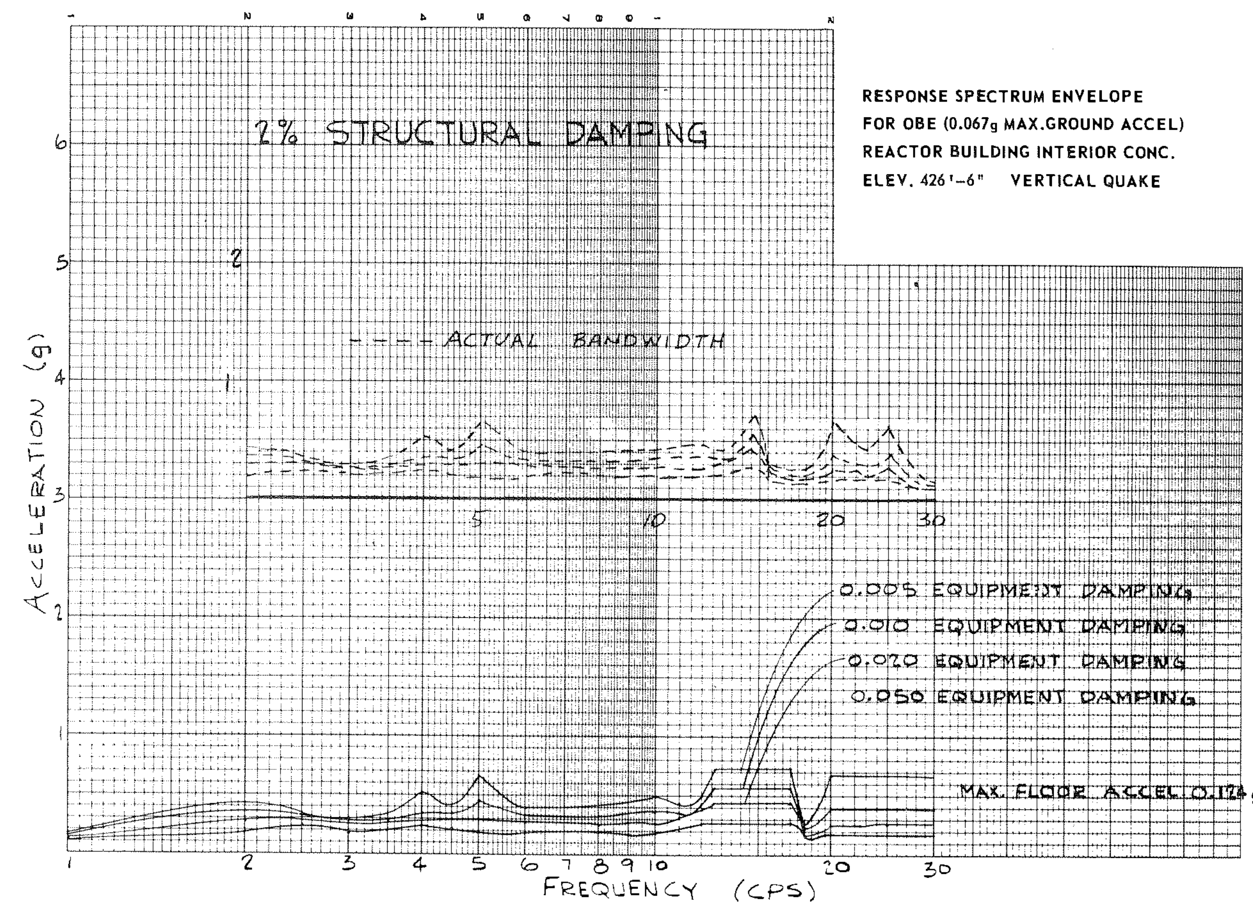


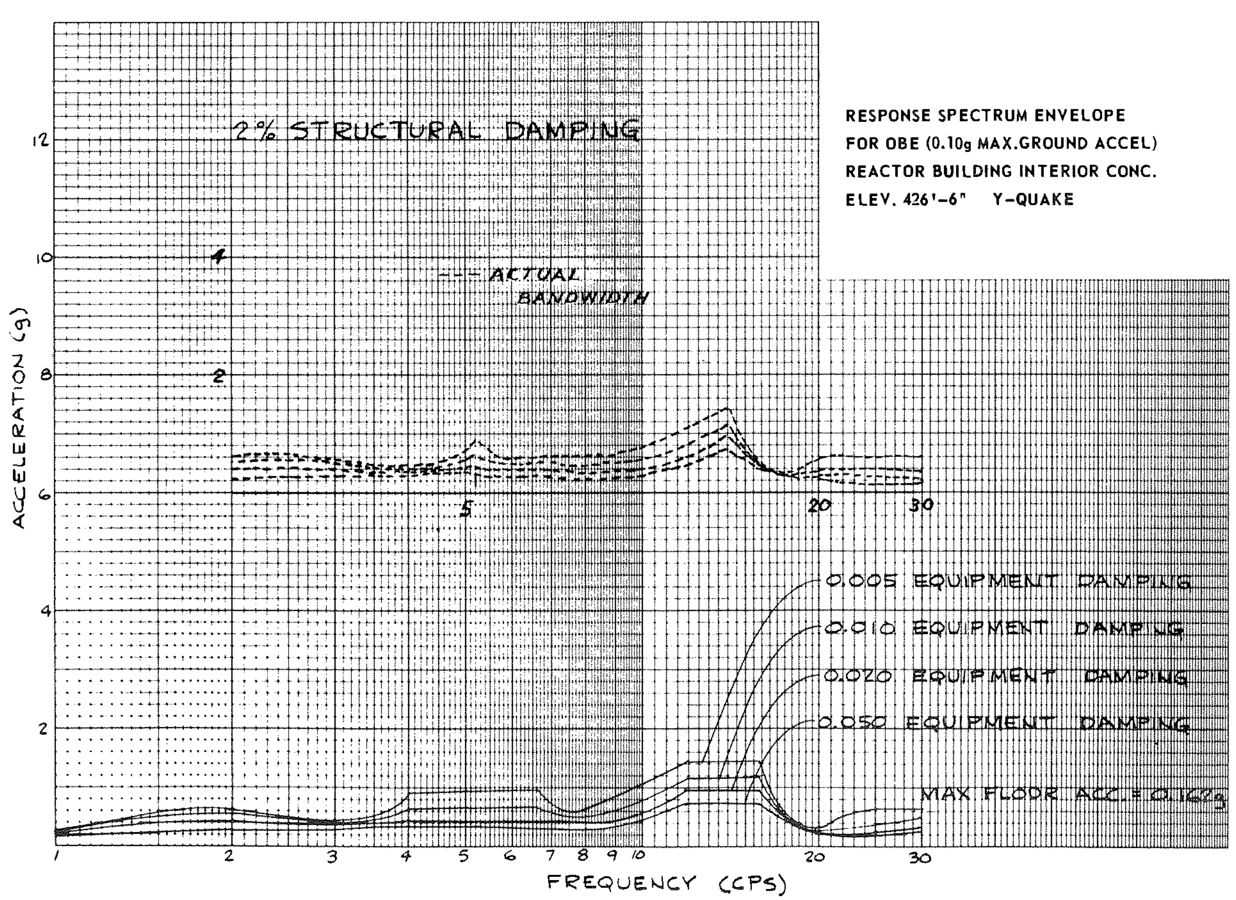
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes Reactor Building Shell Elevation 582' - 11"

Figure 3.7-22



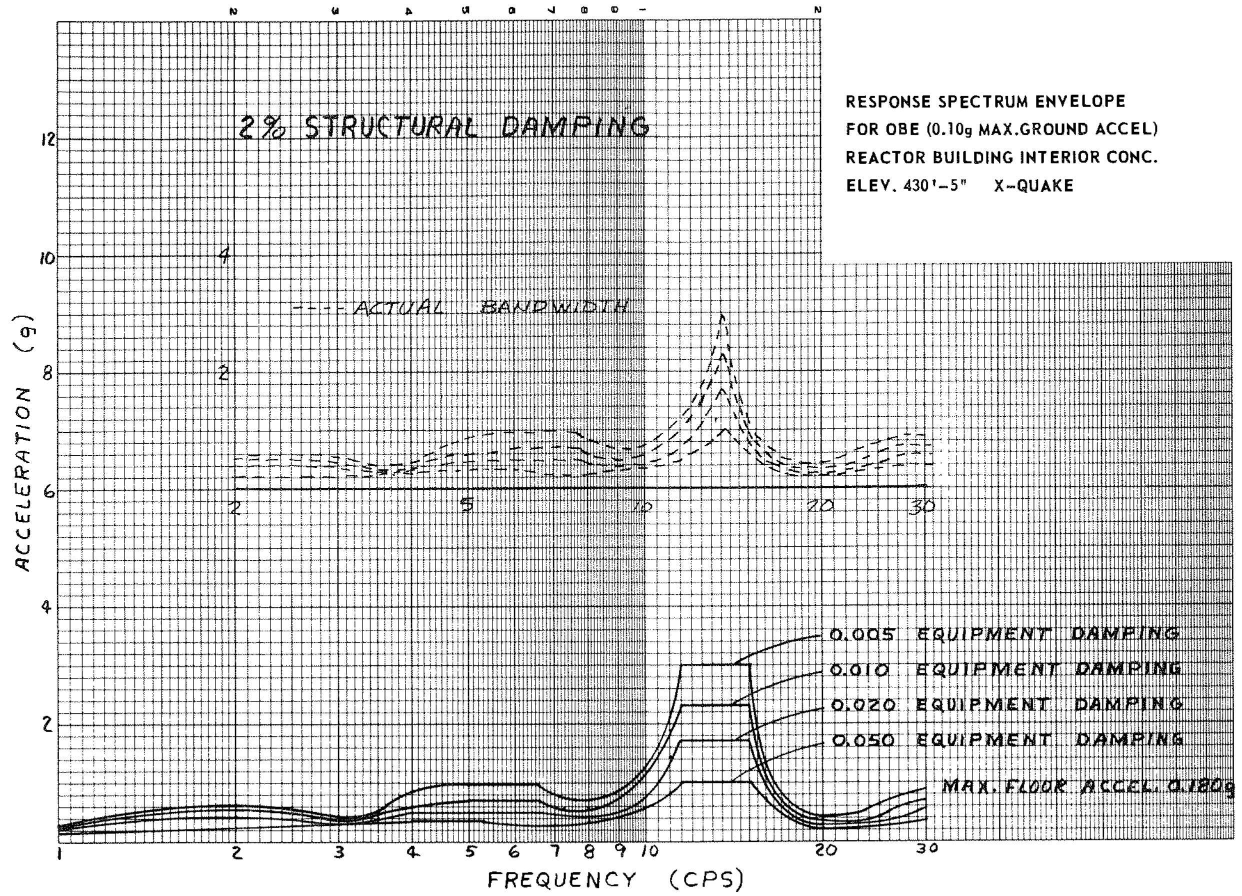


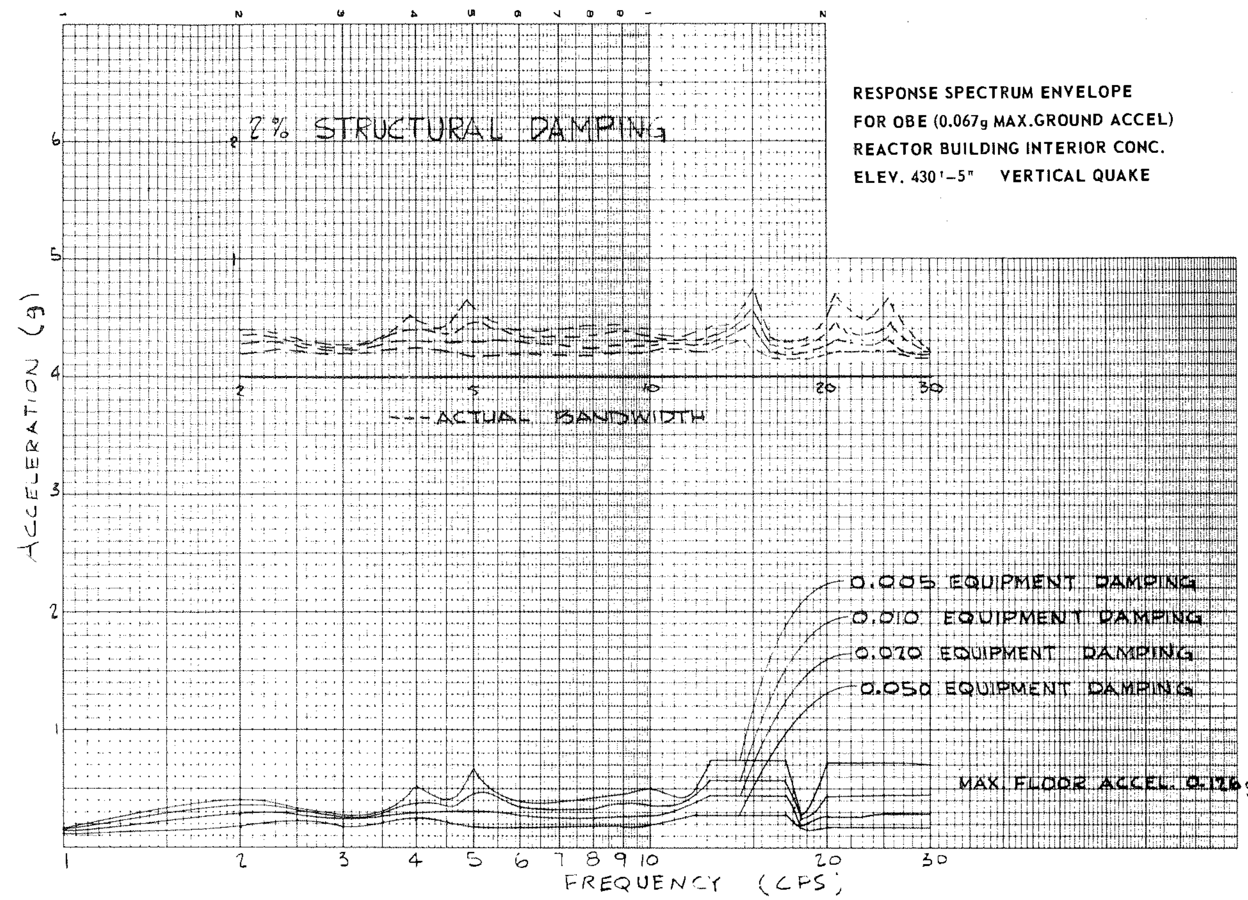


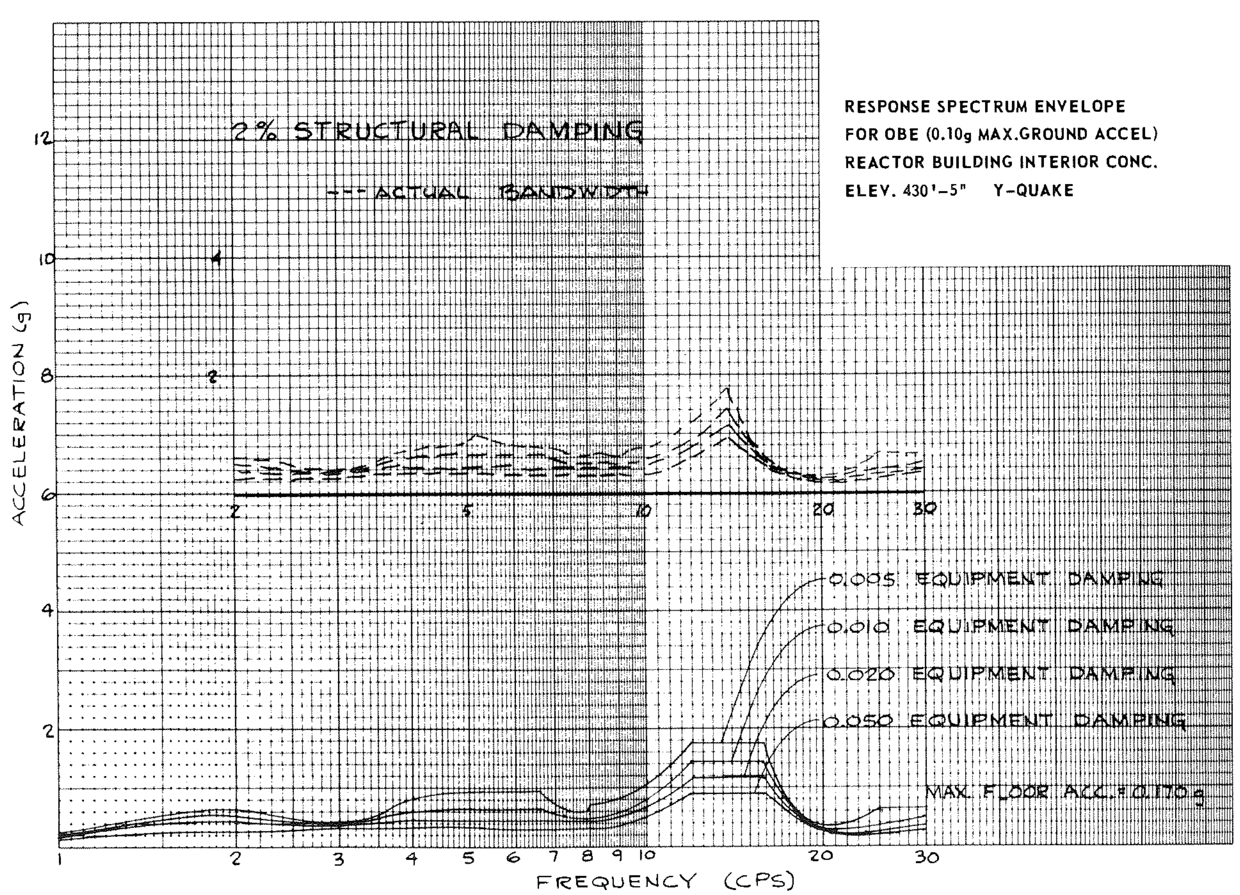
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes Reactor Building Interior Concrete Elevation 426' - 6"

Figure 3.7-23



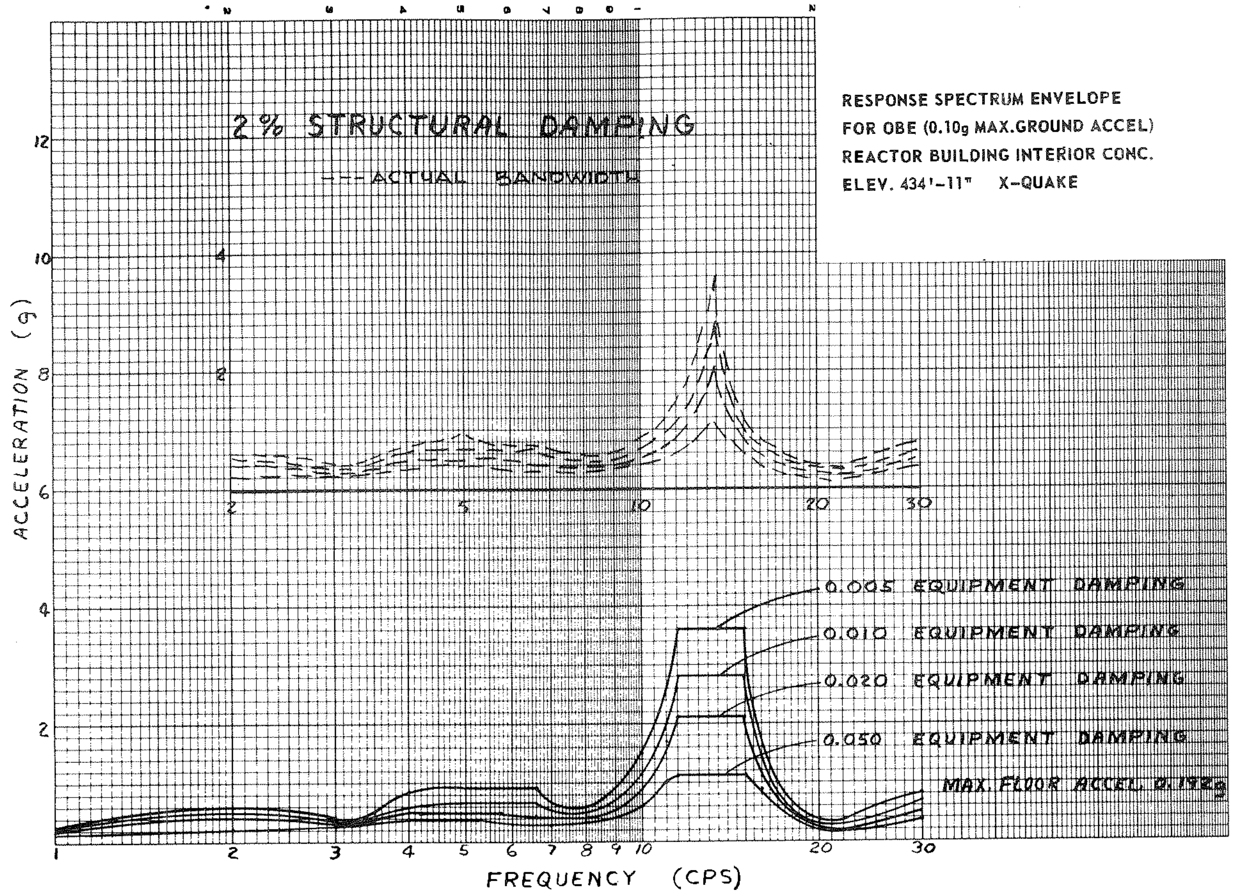


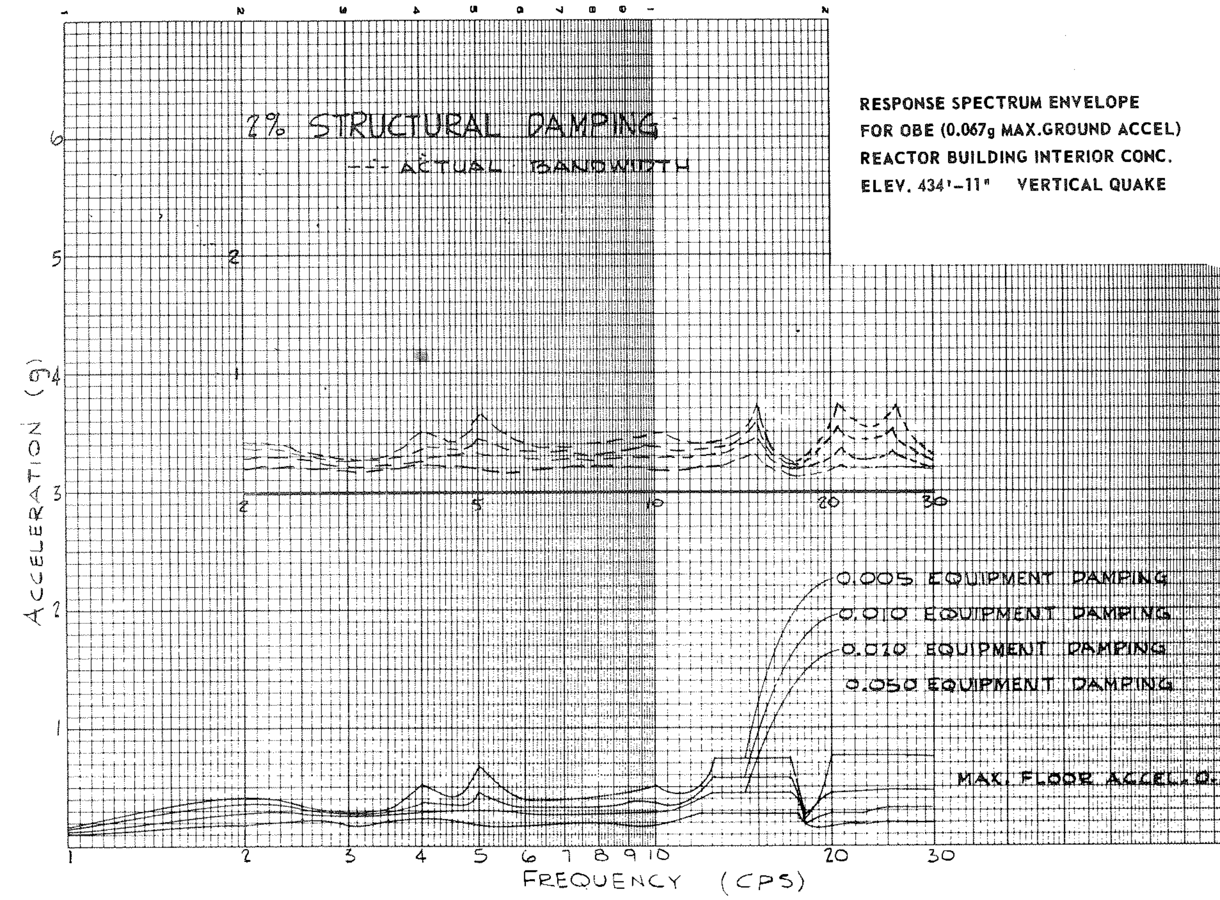


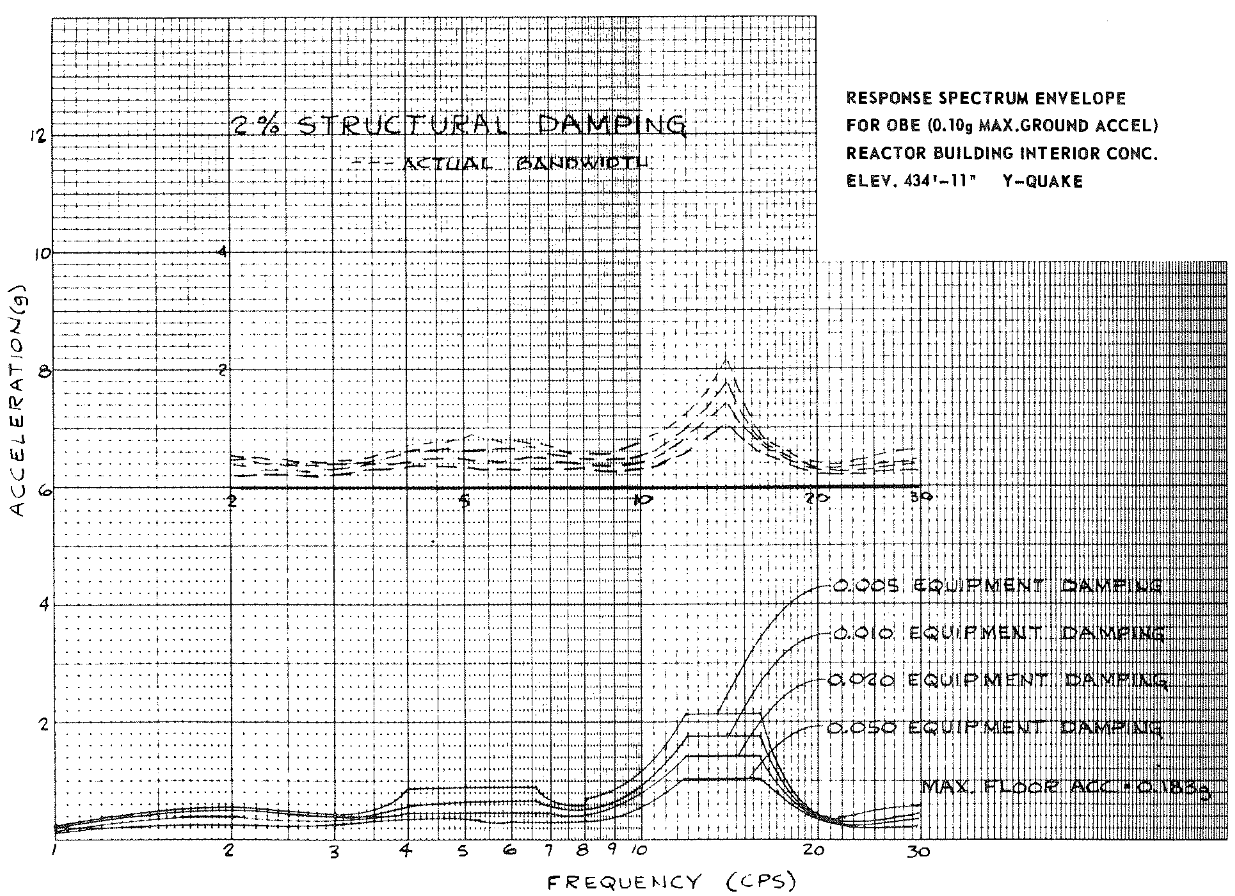
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes Reactor Building Interior Concrete Elevation 430' - 5"

Figure 3.7-24



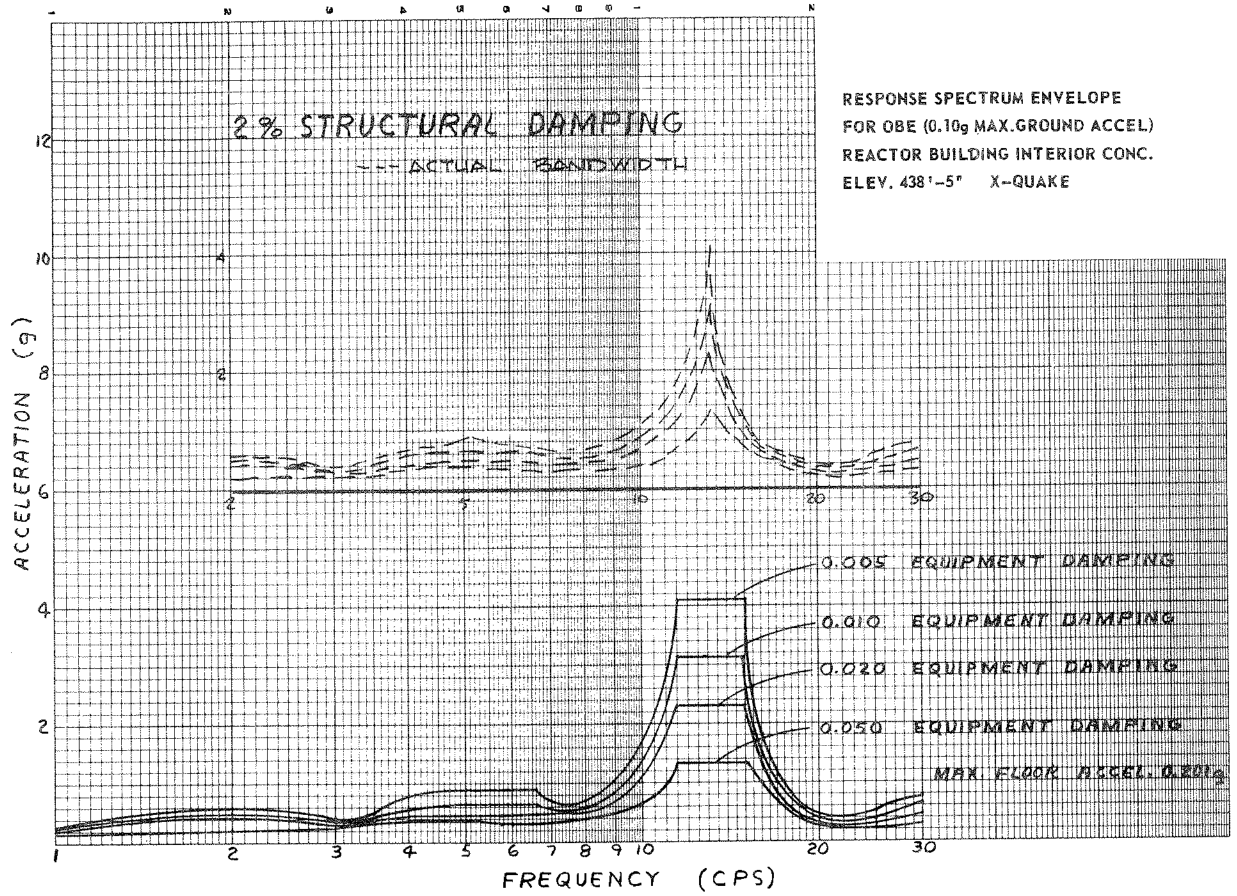


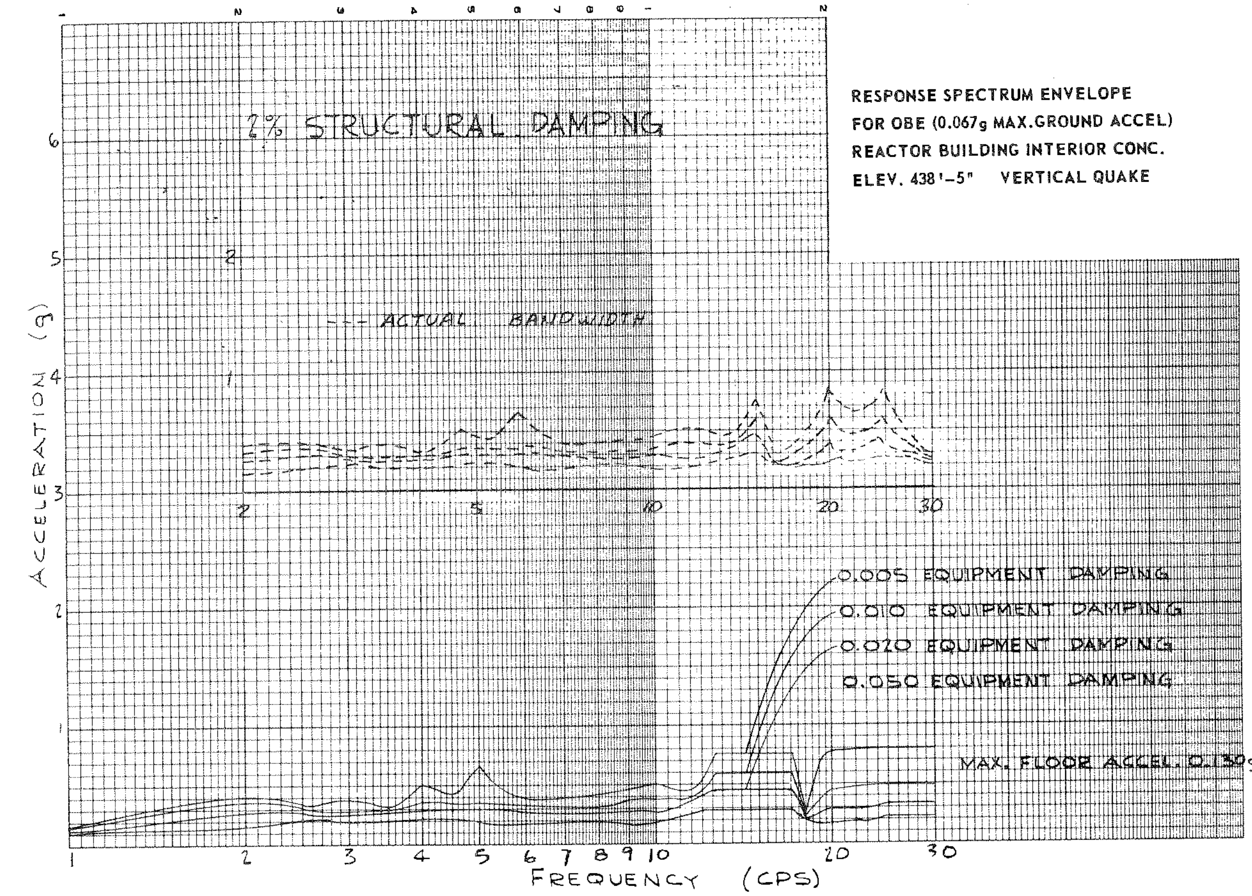


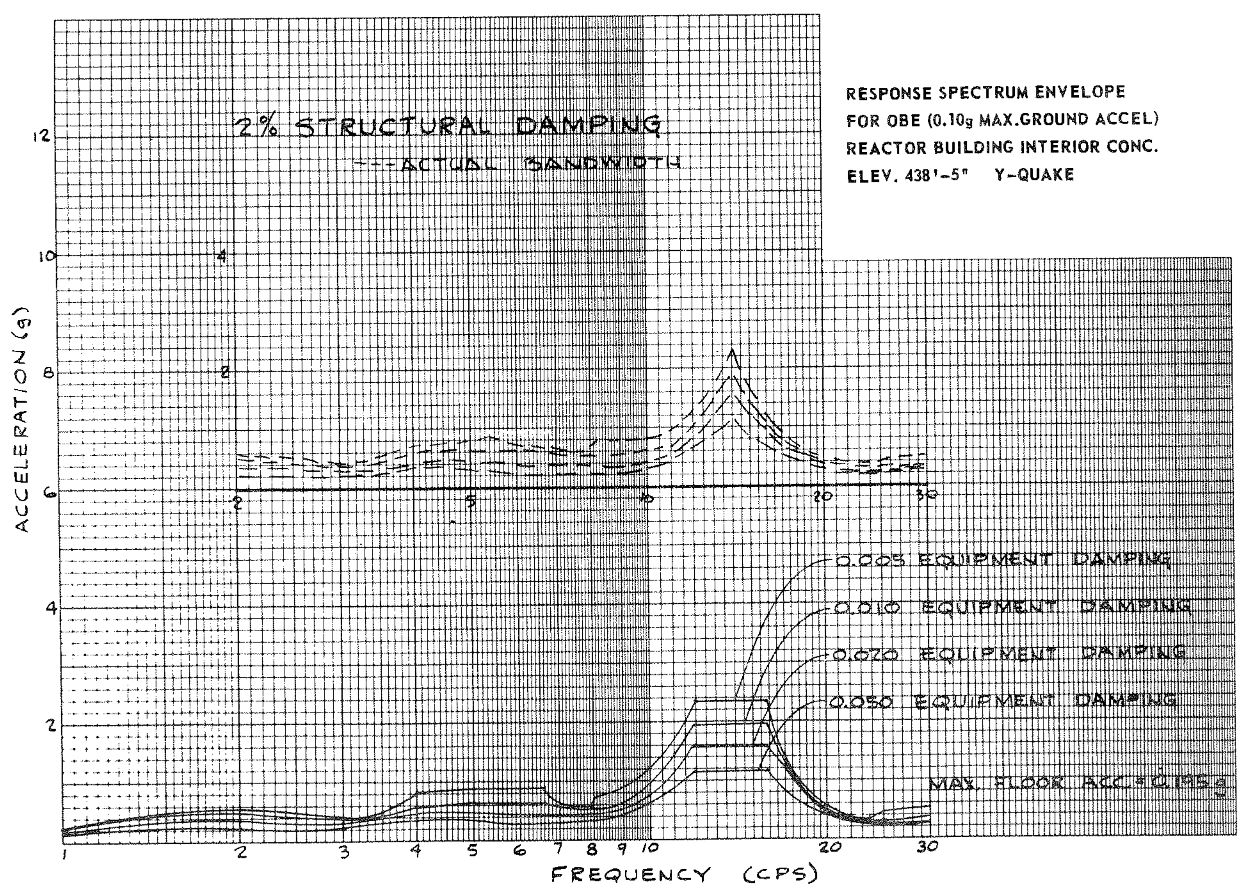
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes Reactor Building Interior Concrete Elevation 434' - 11"

Figure 3.7-25



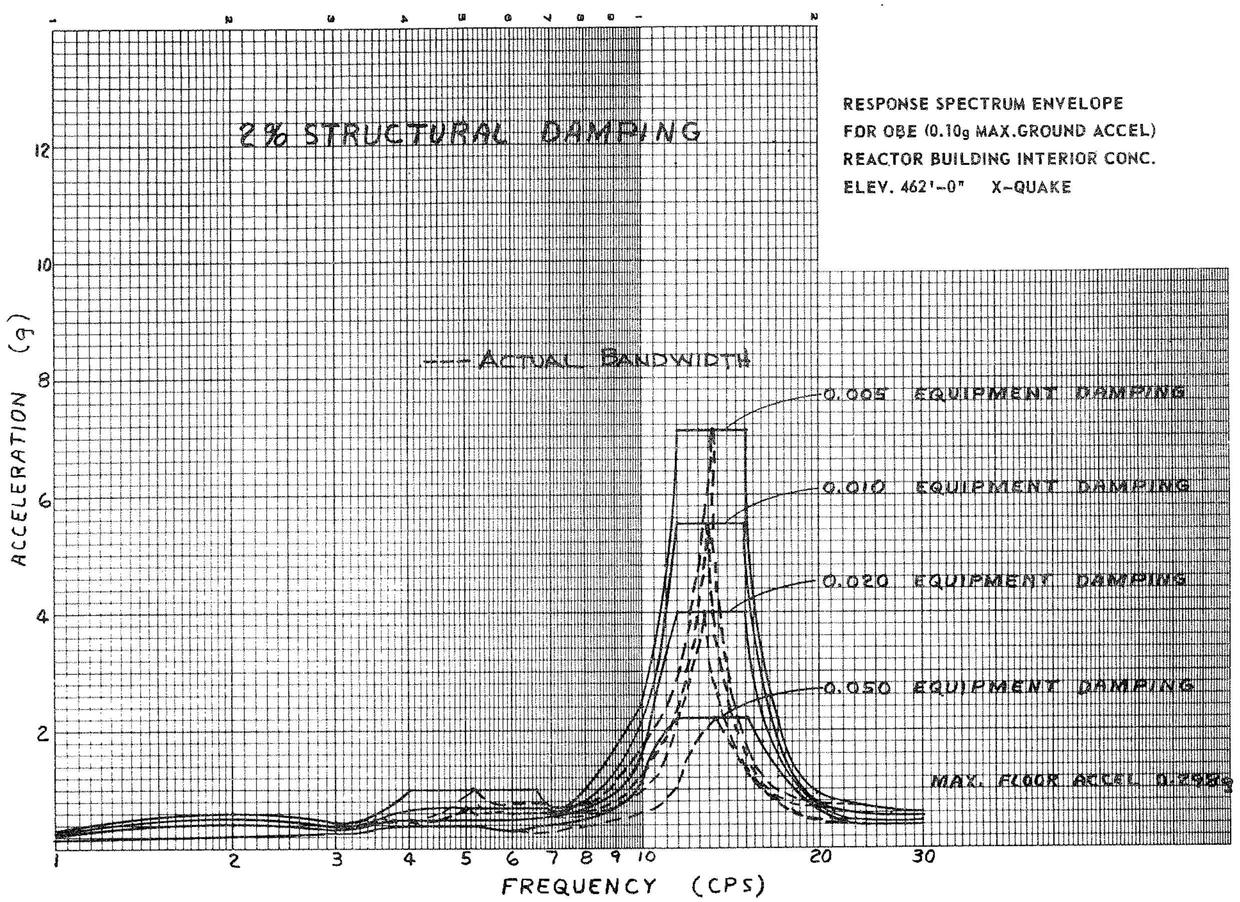


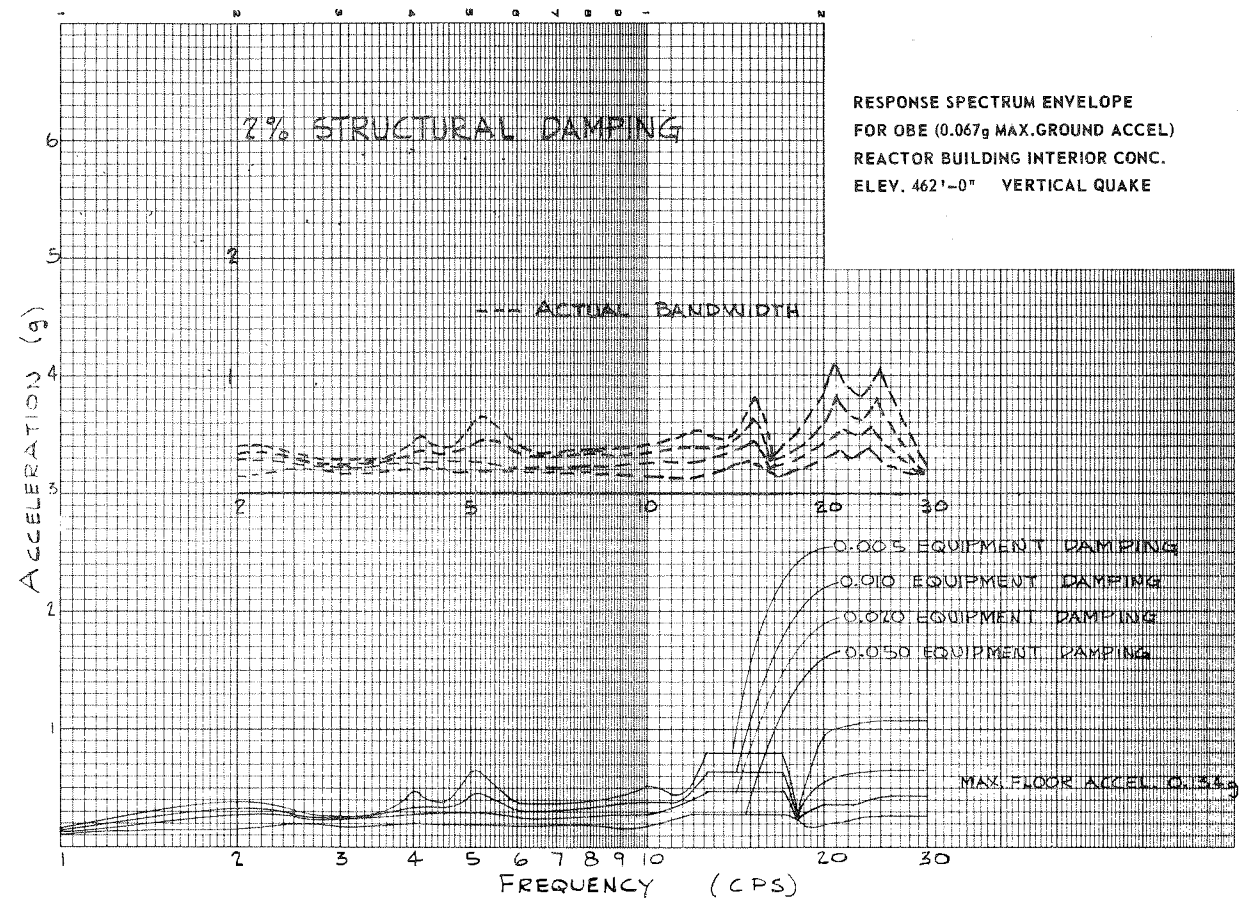


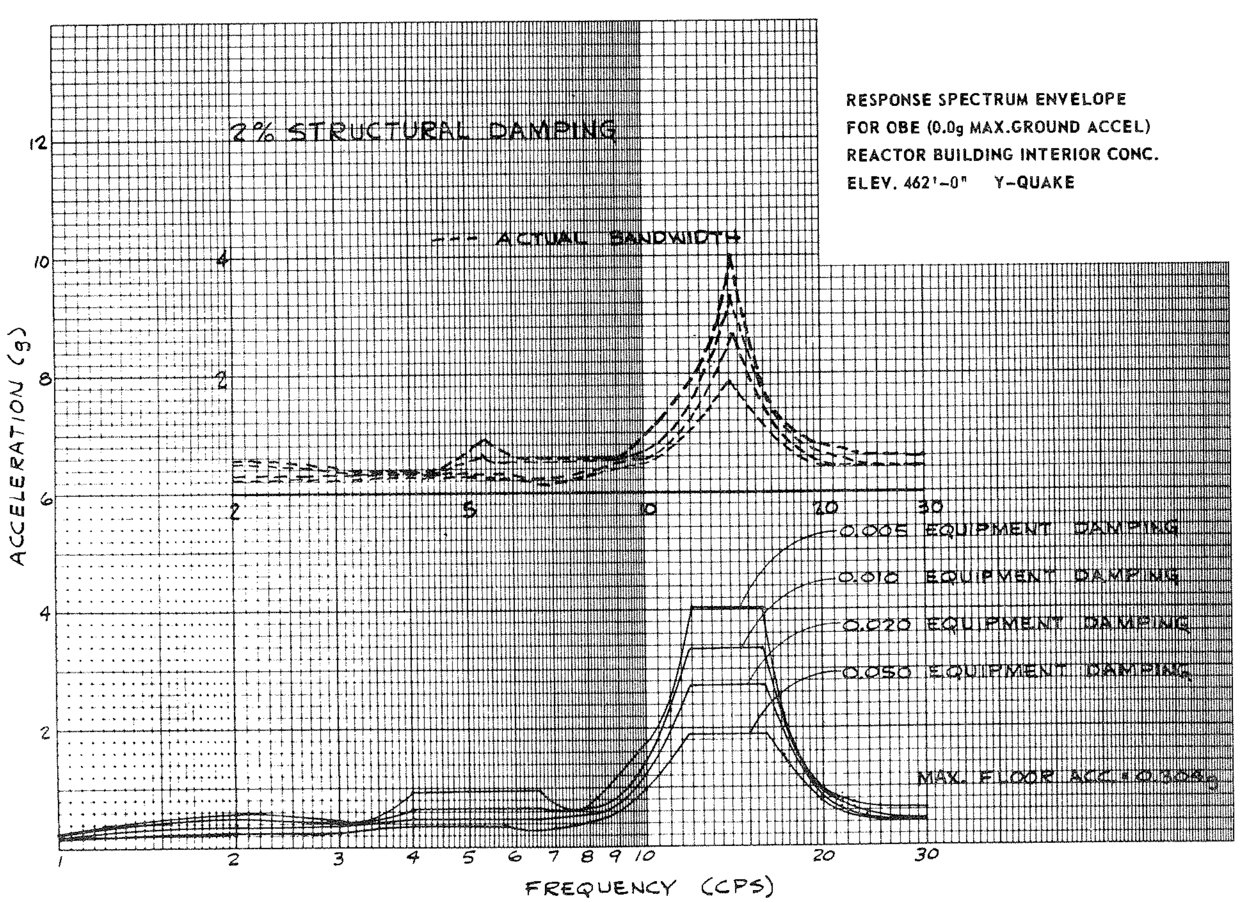
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes Reactor Building Interior Concrete Elevation 438' - 5"

Figure 3.7-26



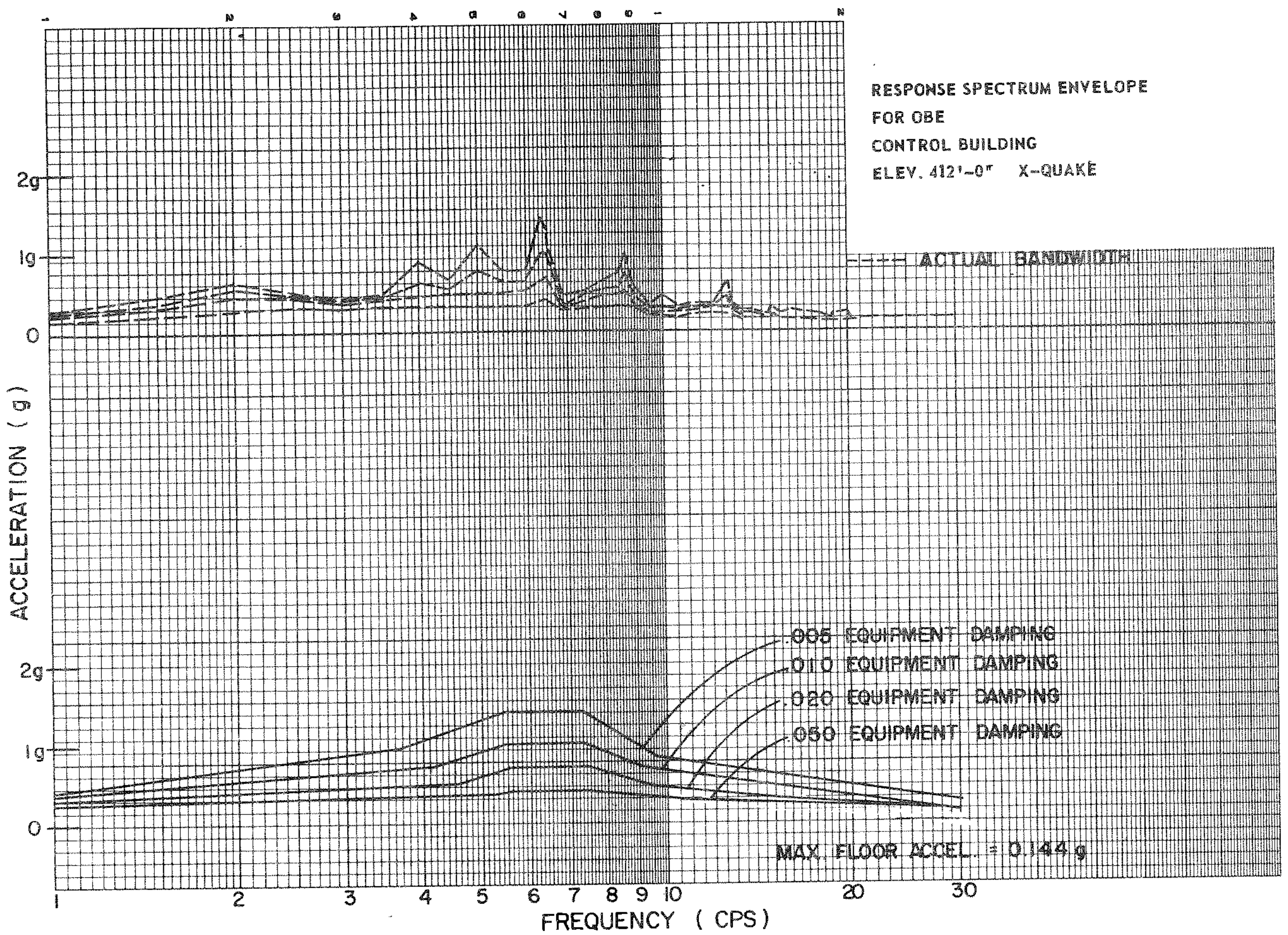


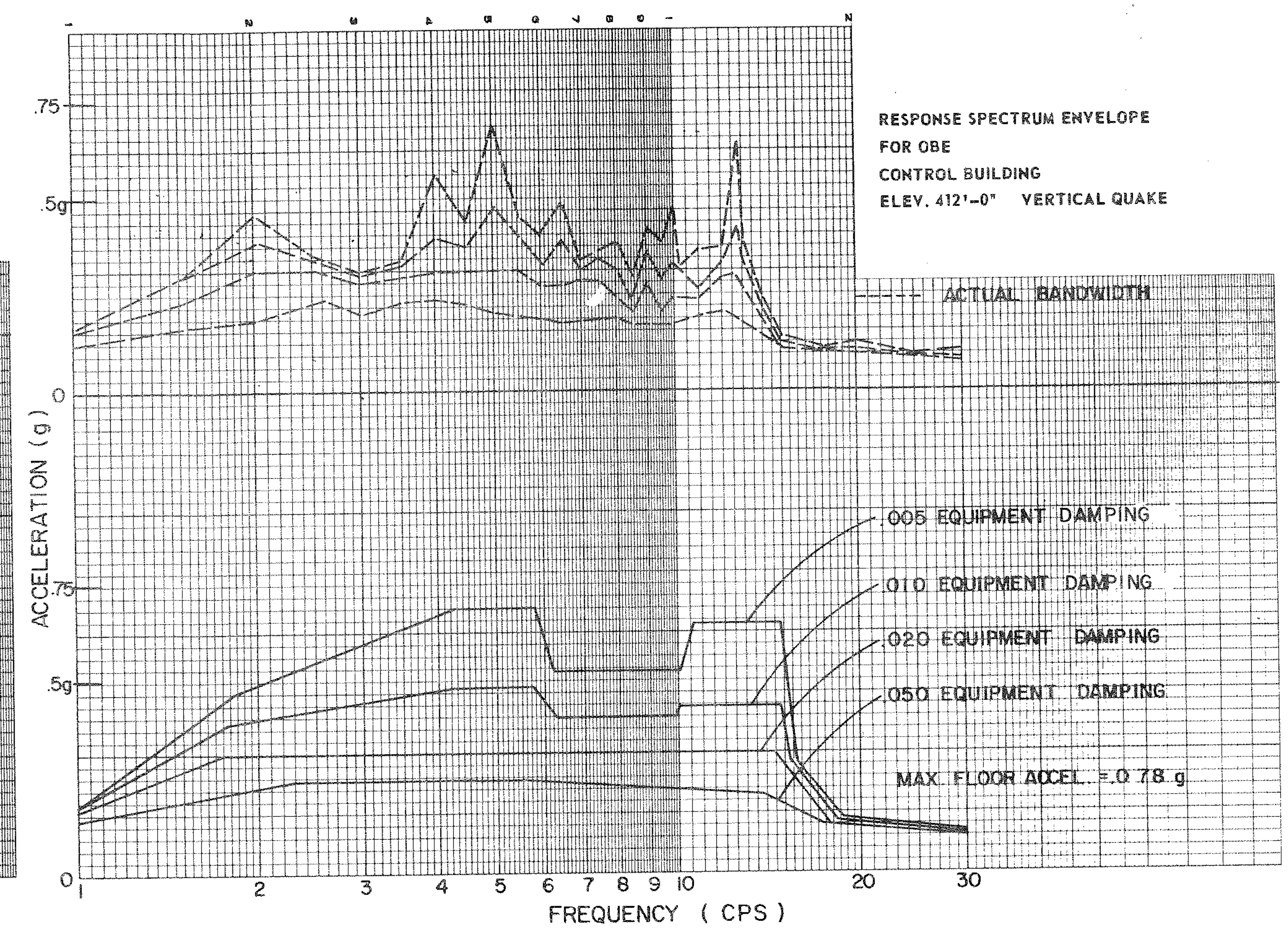


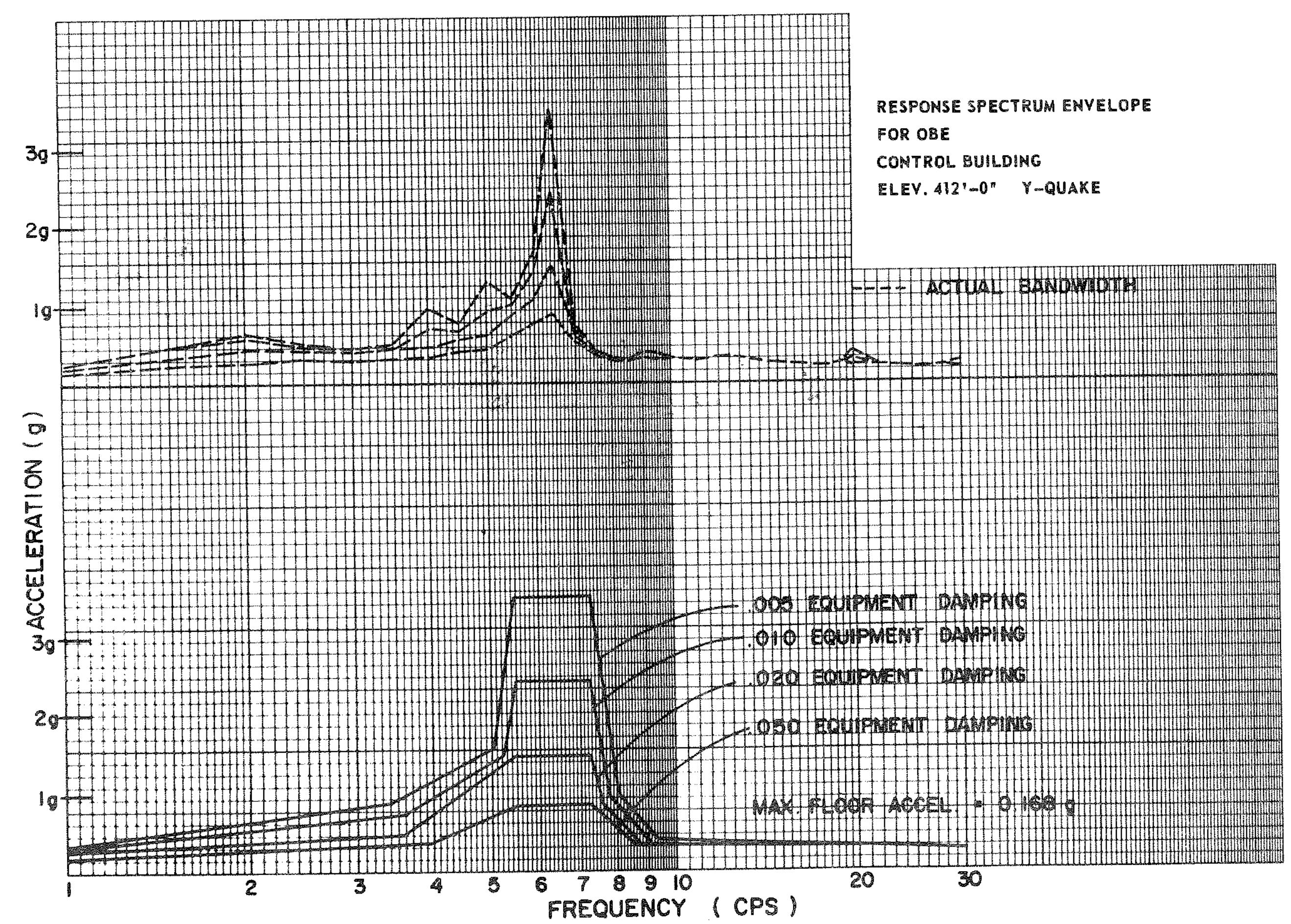
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes Reactor Building Interior Concrete Elevation 462' - 0"

Figure 3.7-27



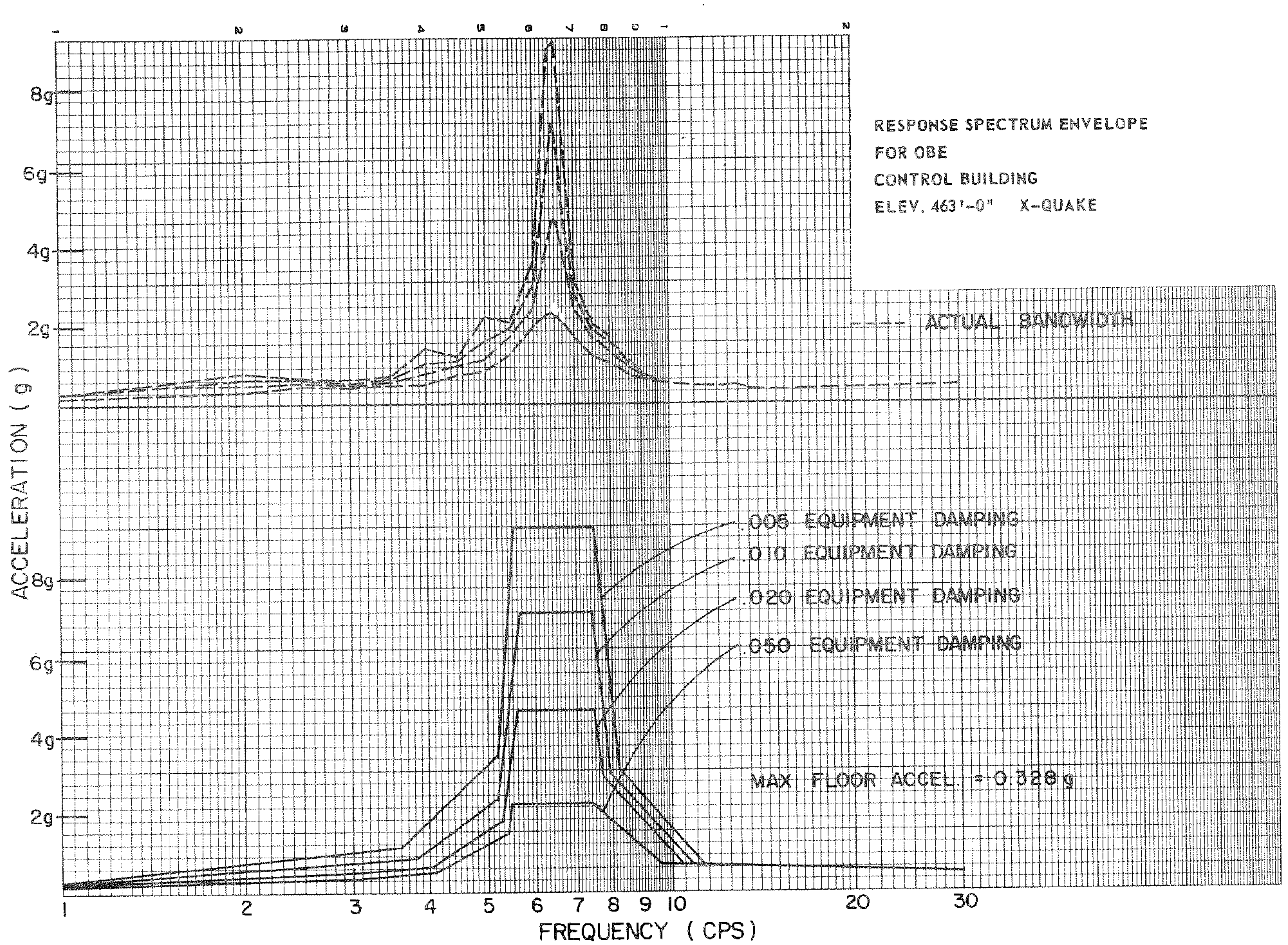


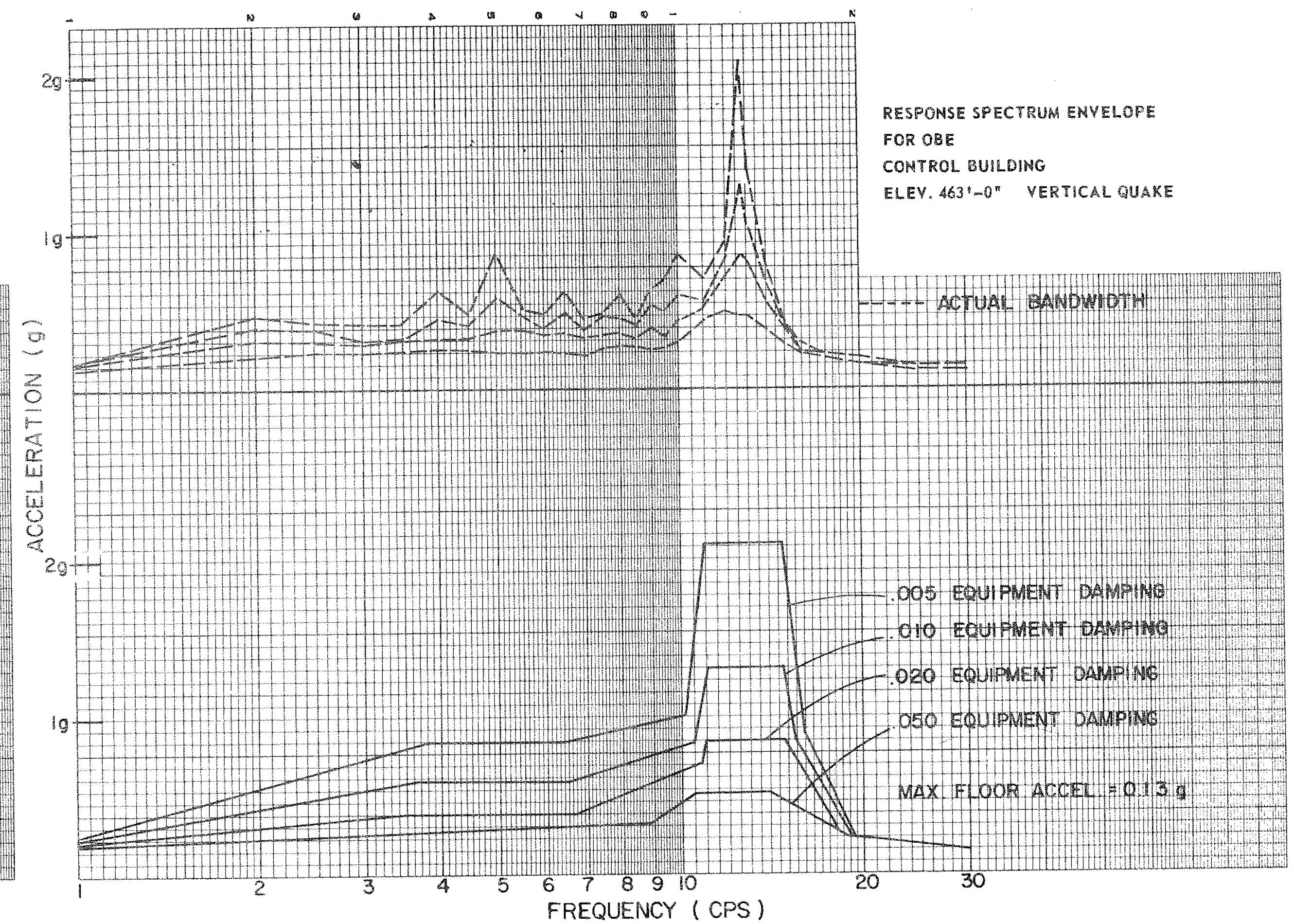


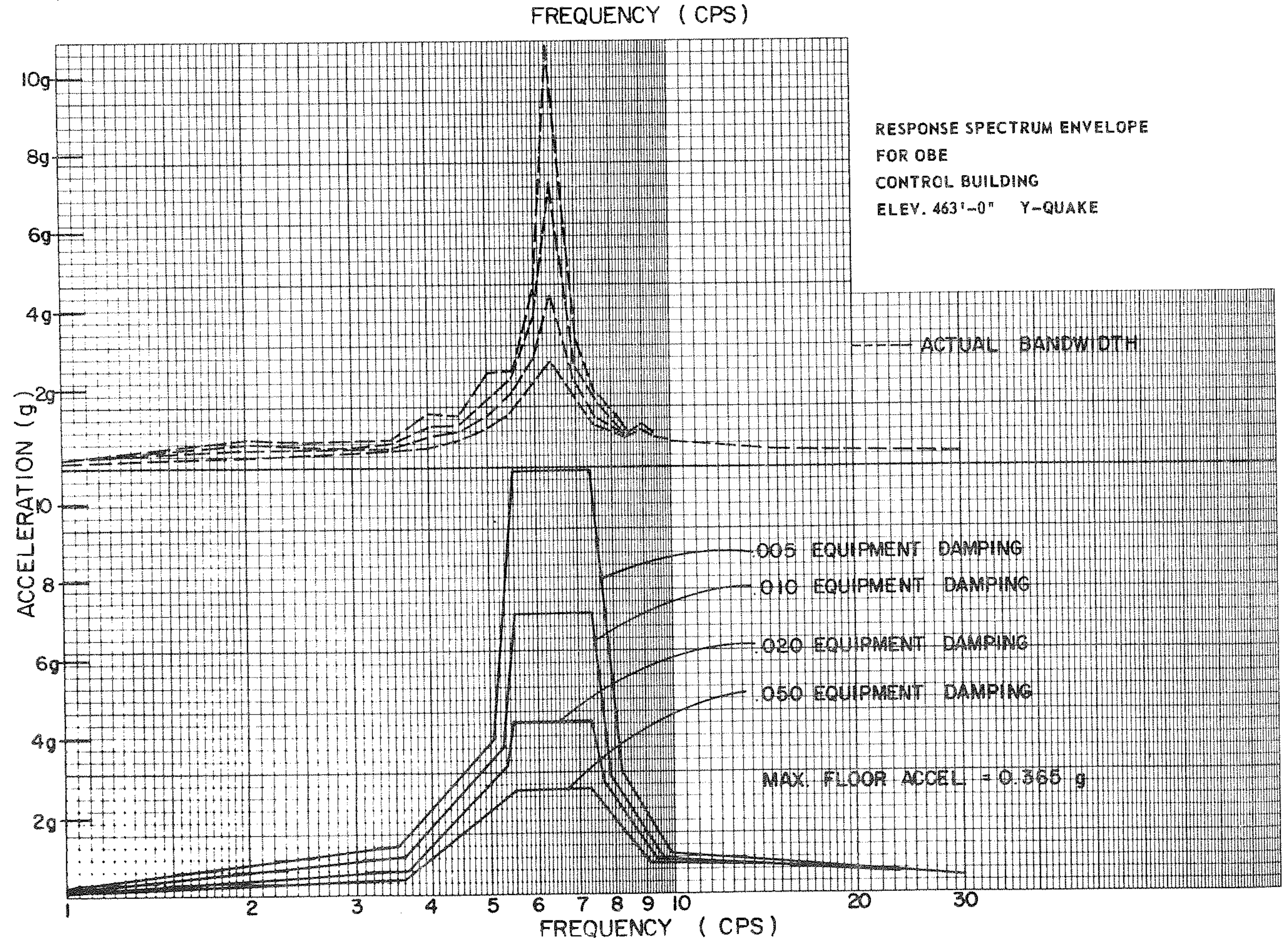
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Control Building
Elevation 412' - 0"

Figure 3.7-28



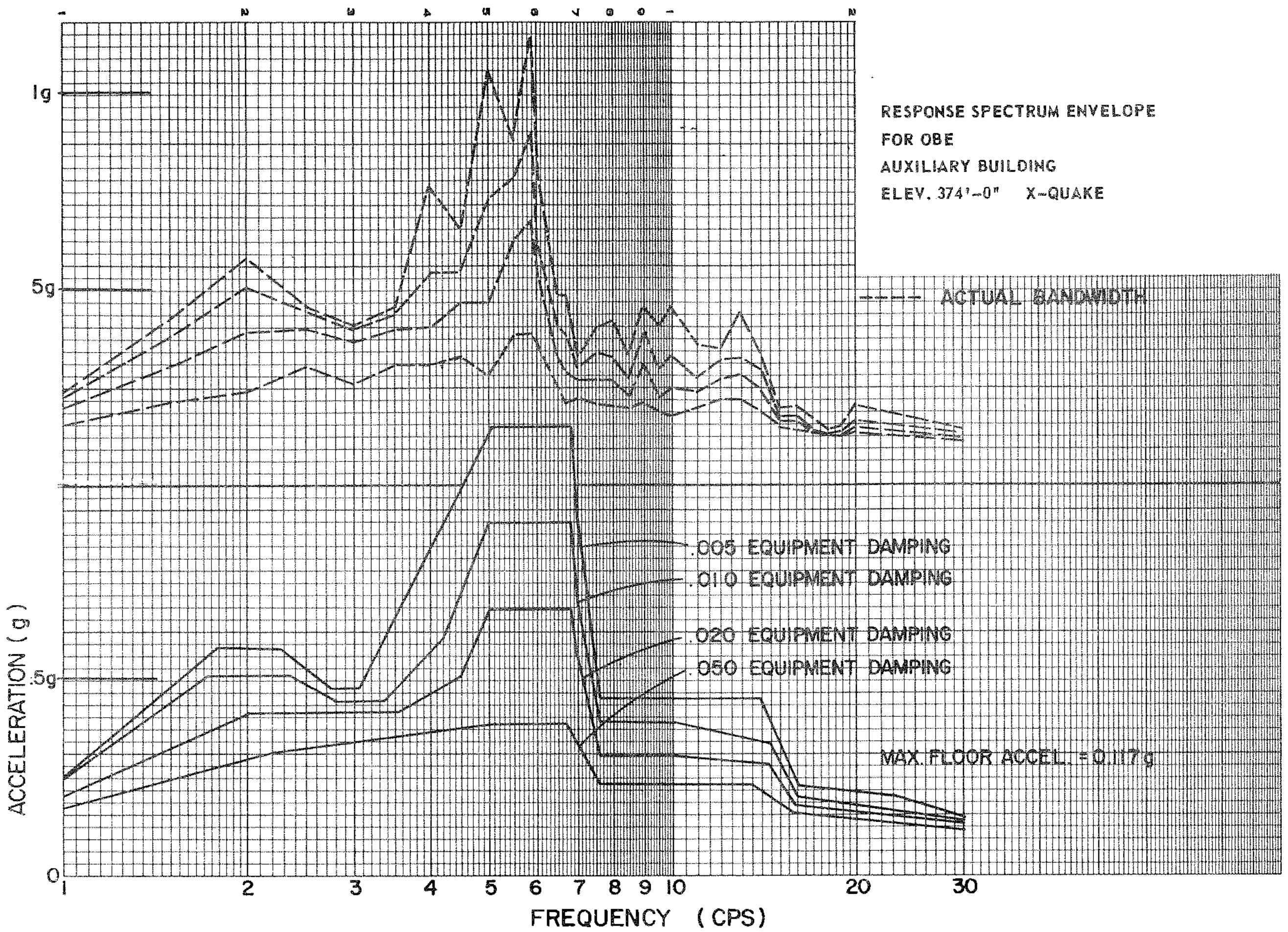


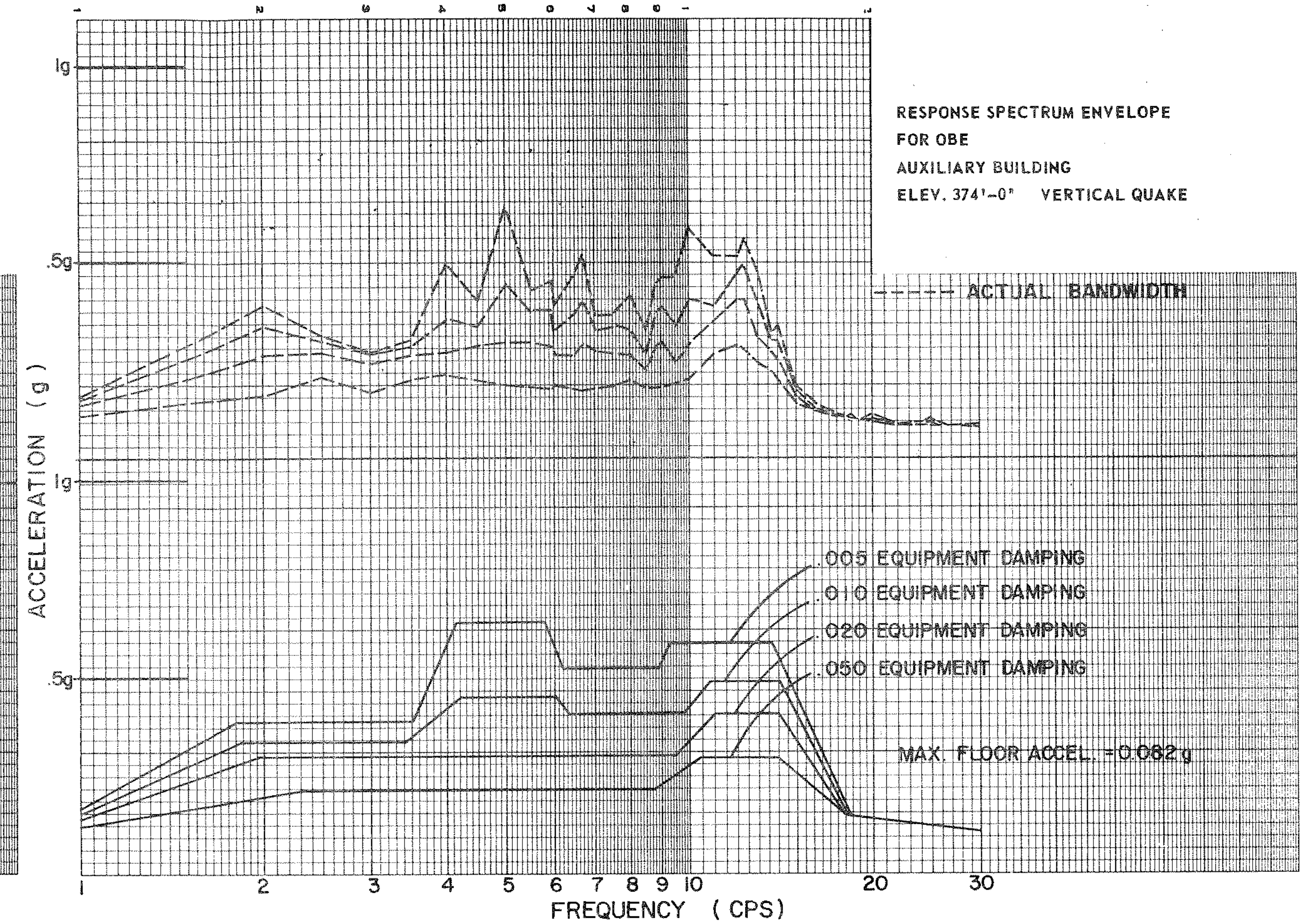


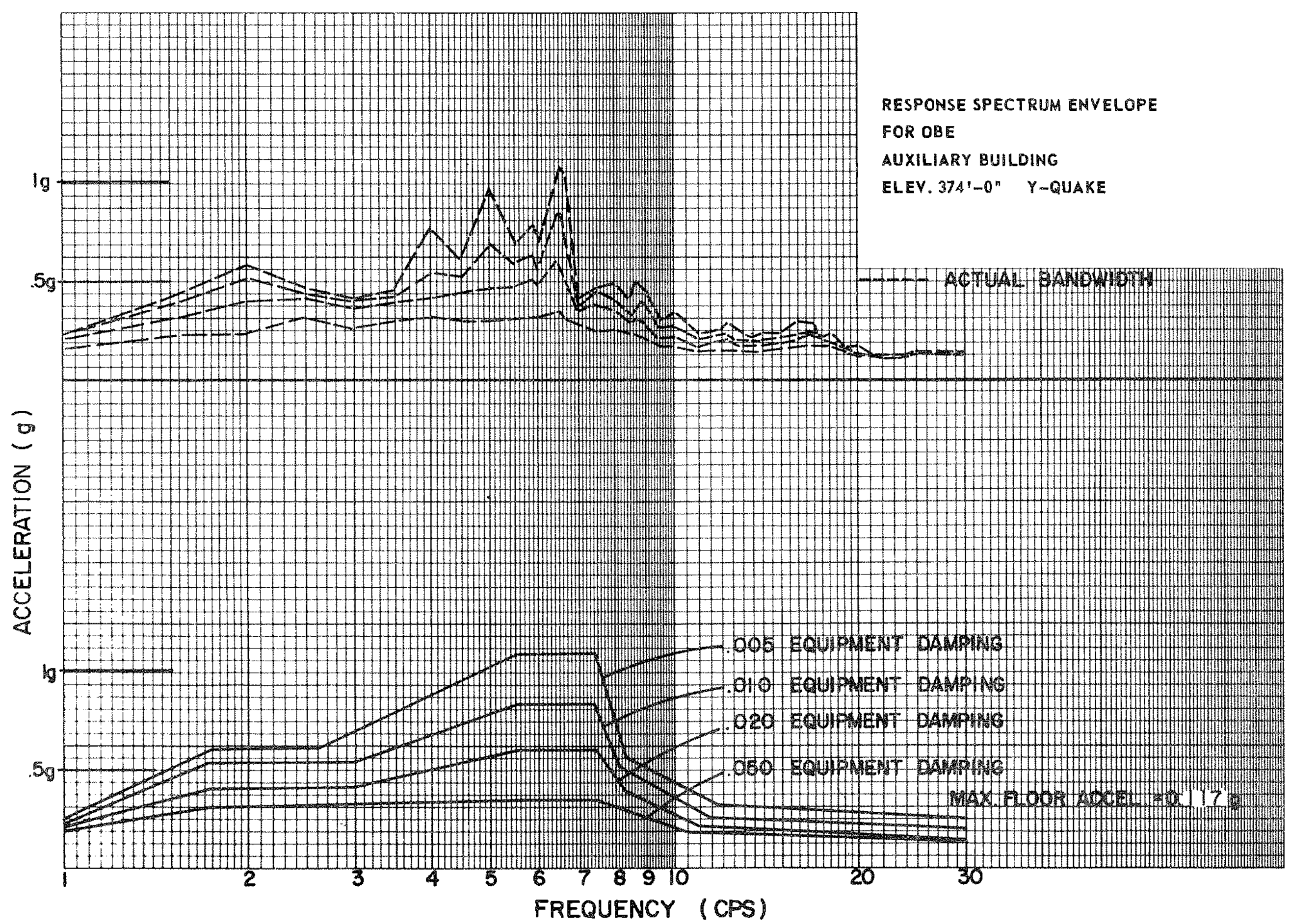
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Control Building
Elevation 463' - 0"

Figure 3.7-29



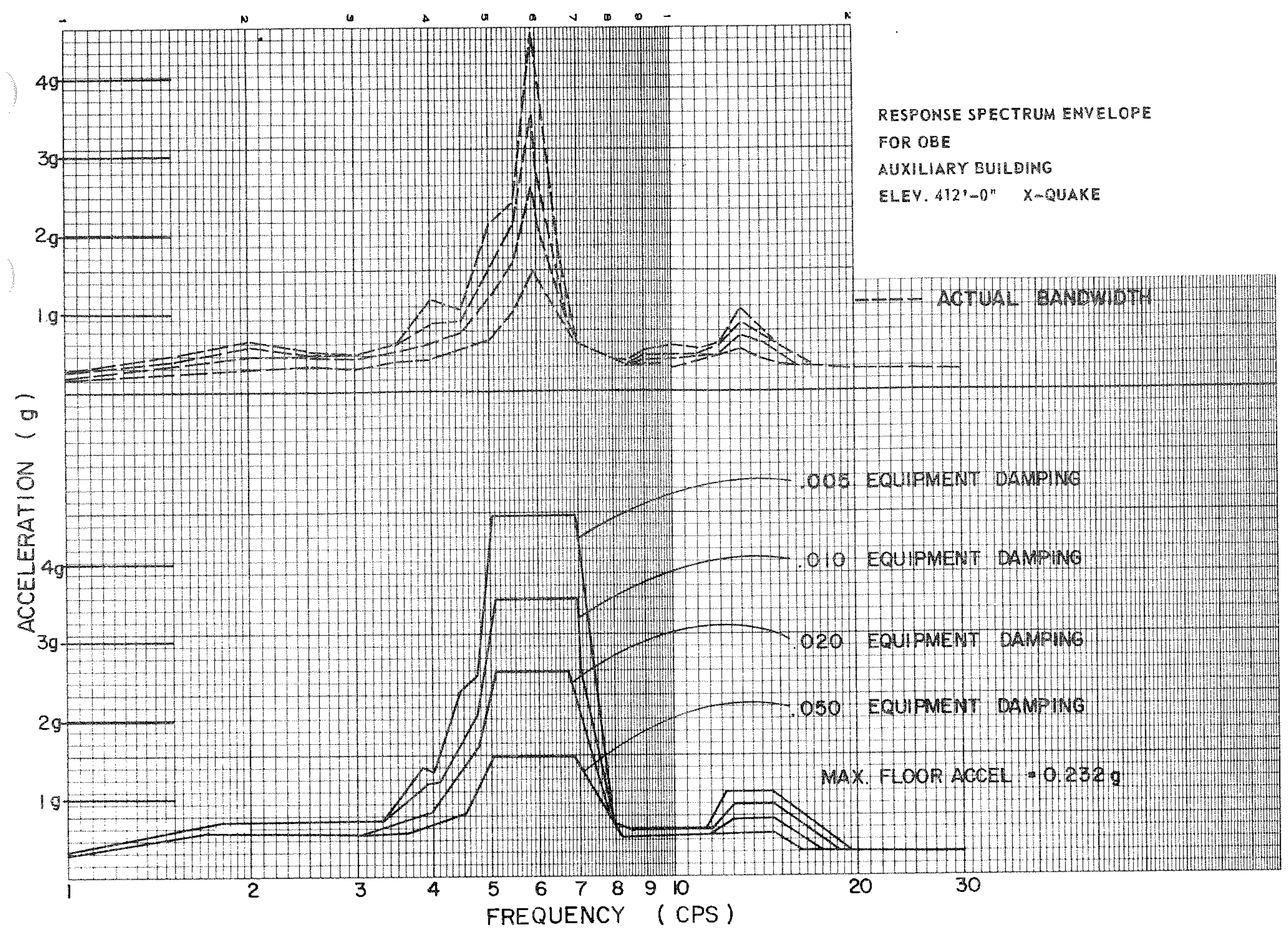


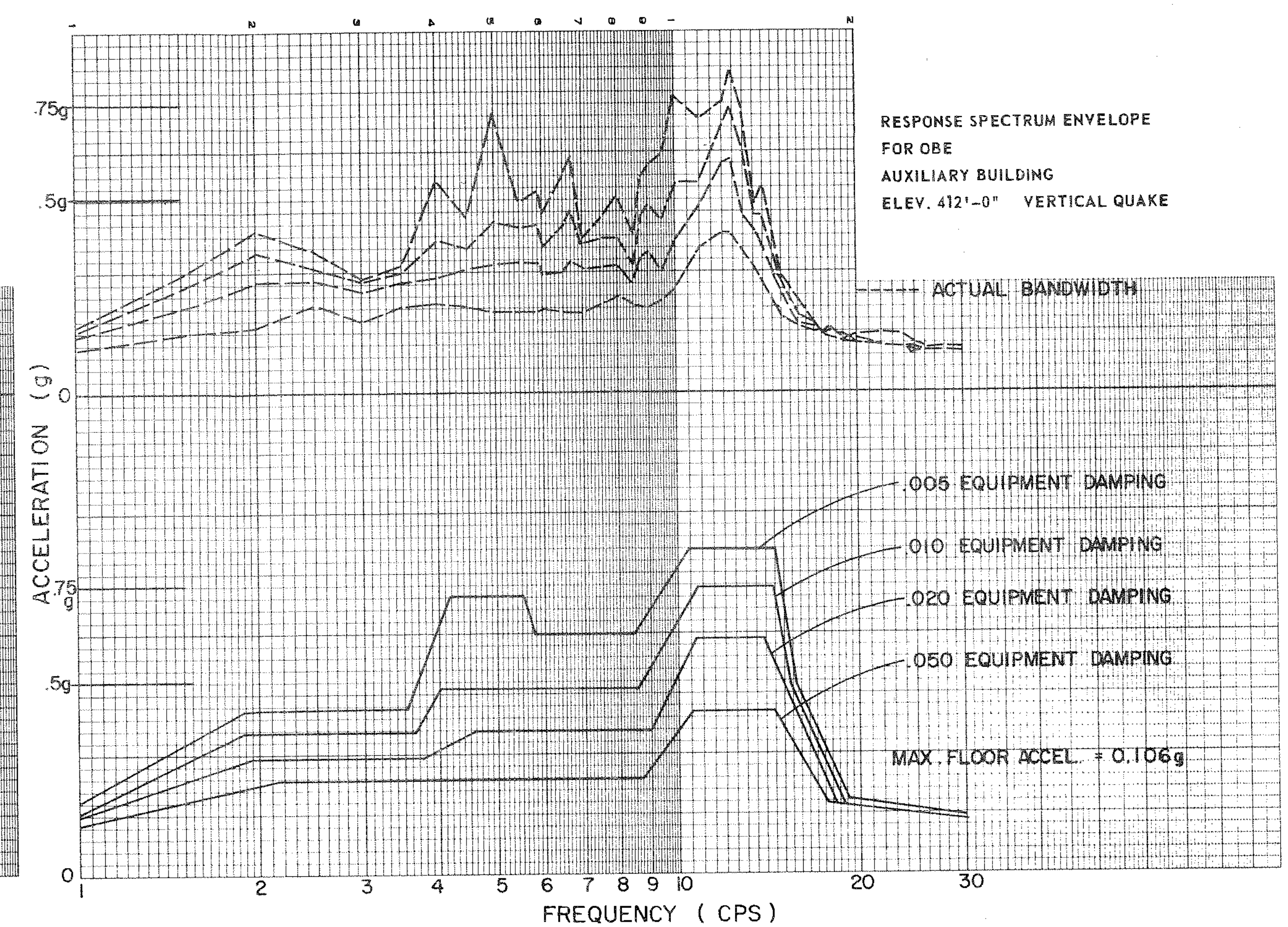


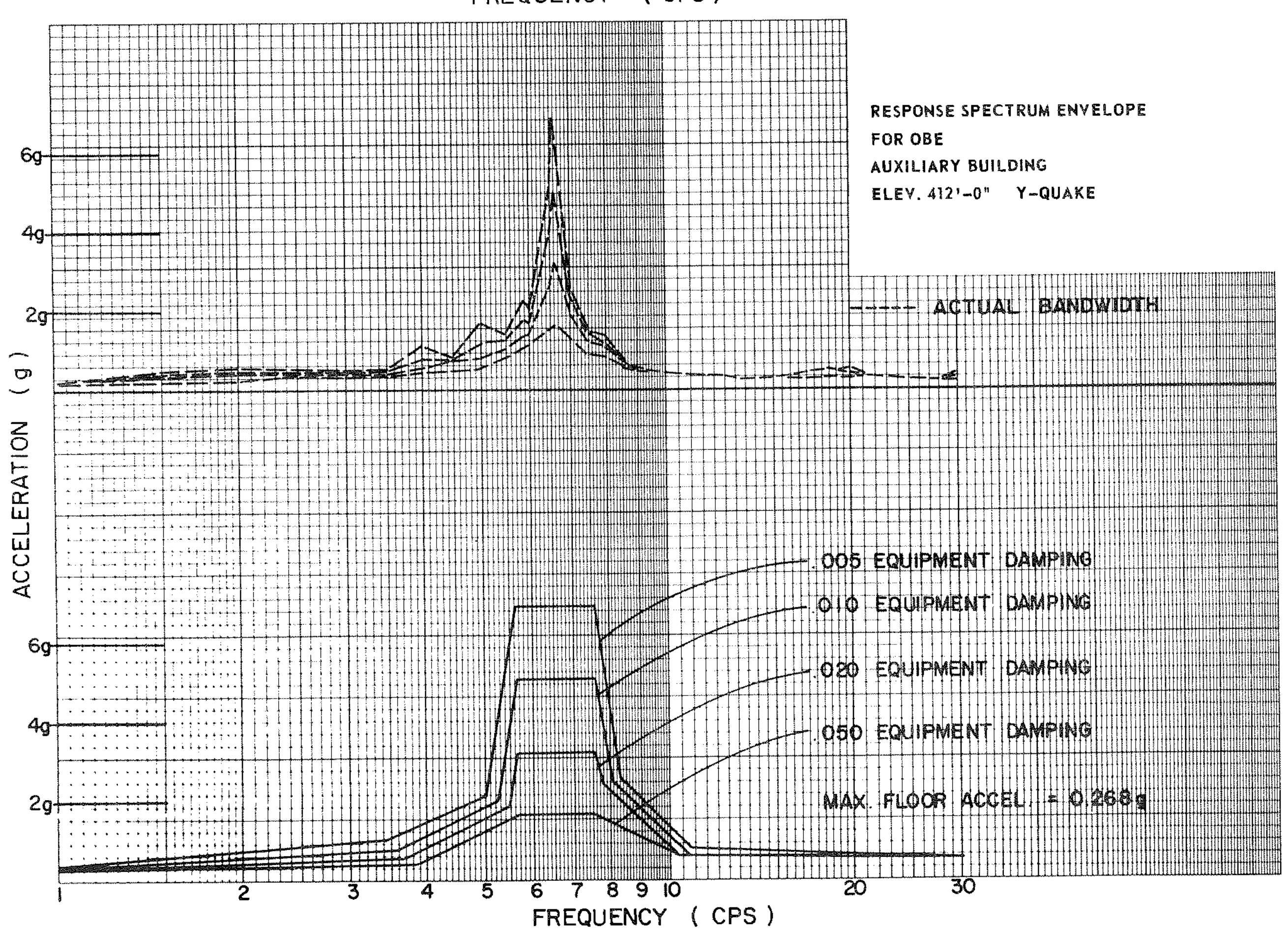
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Auxiliary Building
Elevation 374' - 0"

Figure 3.7-30



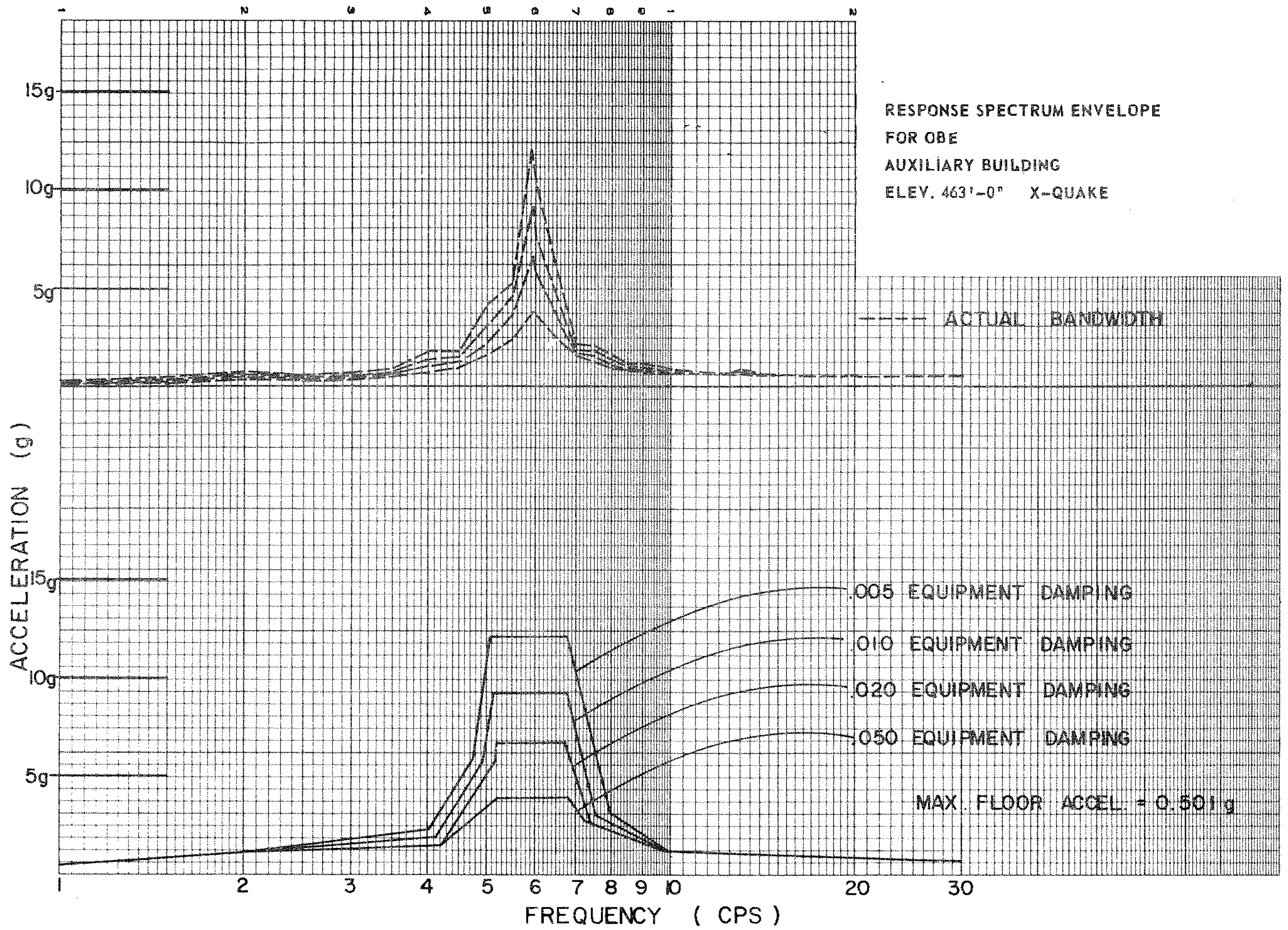


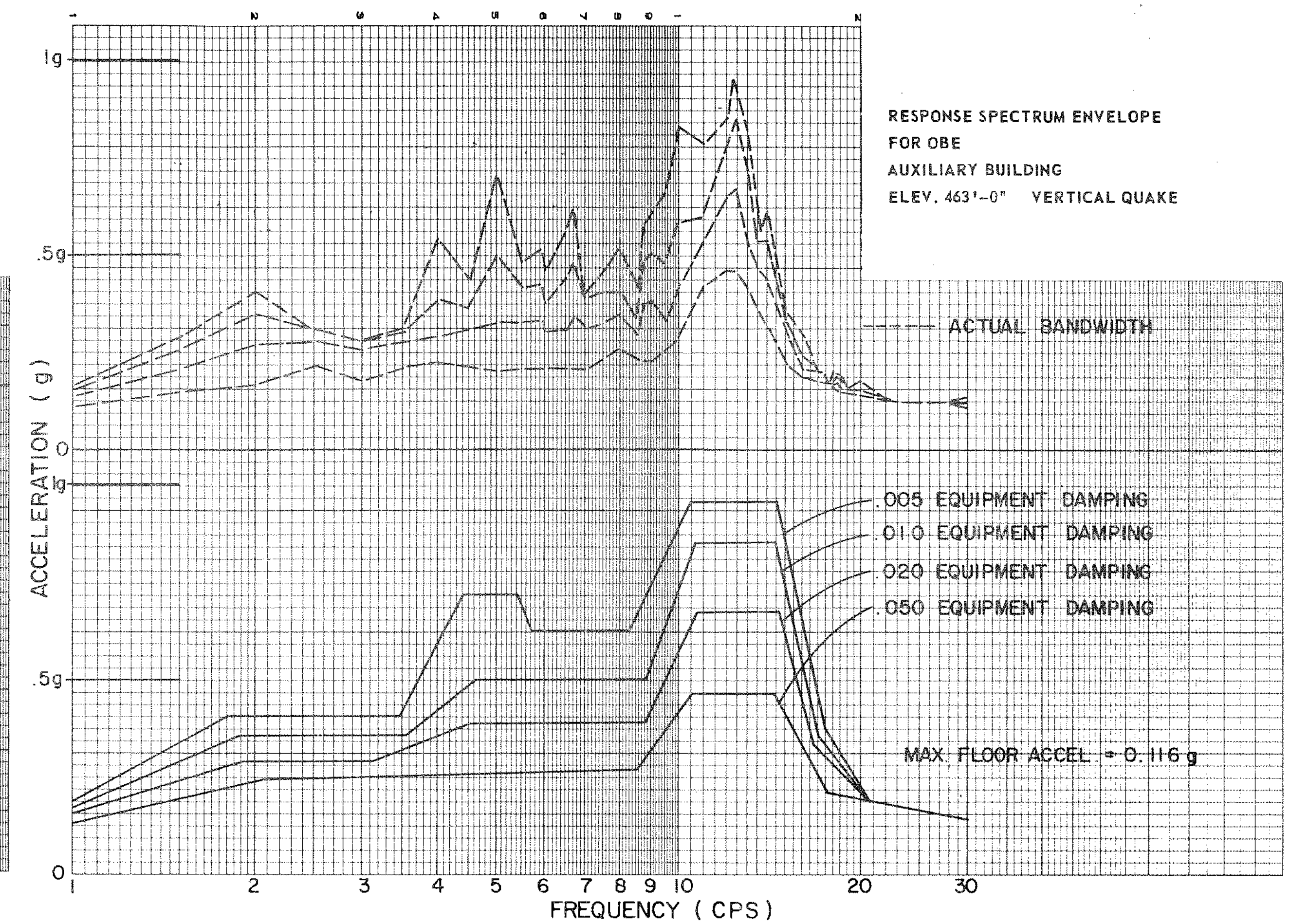


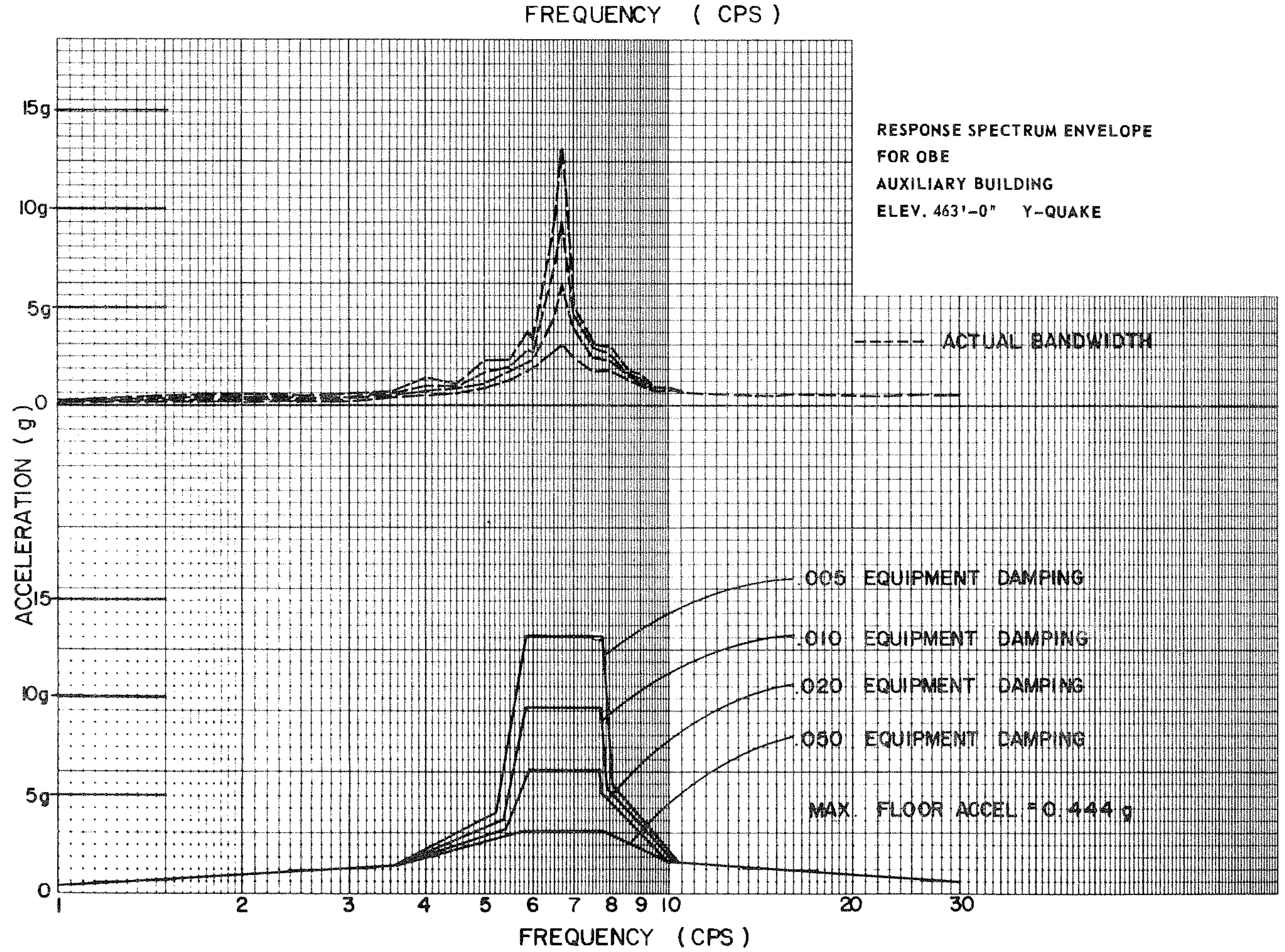
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Auxiliary Building
Elevation 412' - 0"

Figure 3.7-31



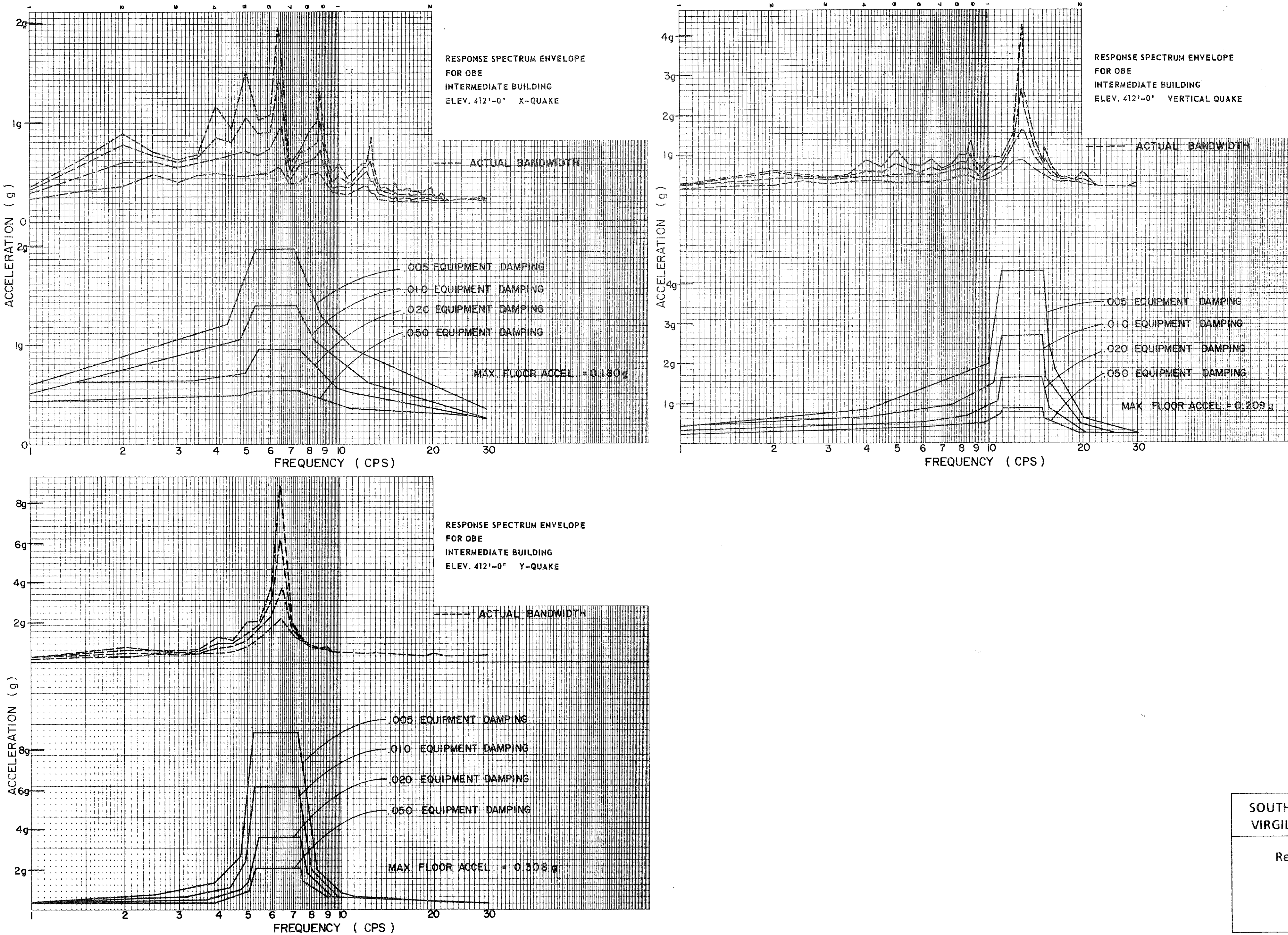




SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Auxiliary Building
Elevation 463' - 0"

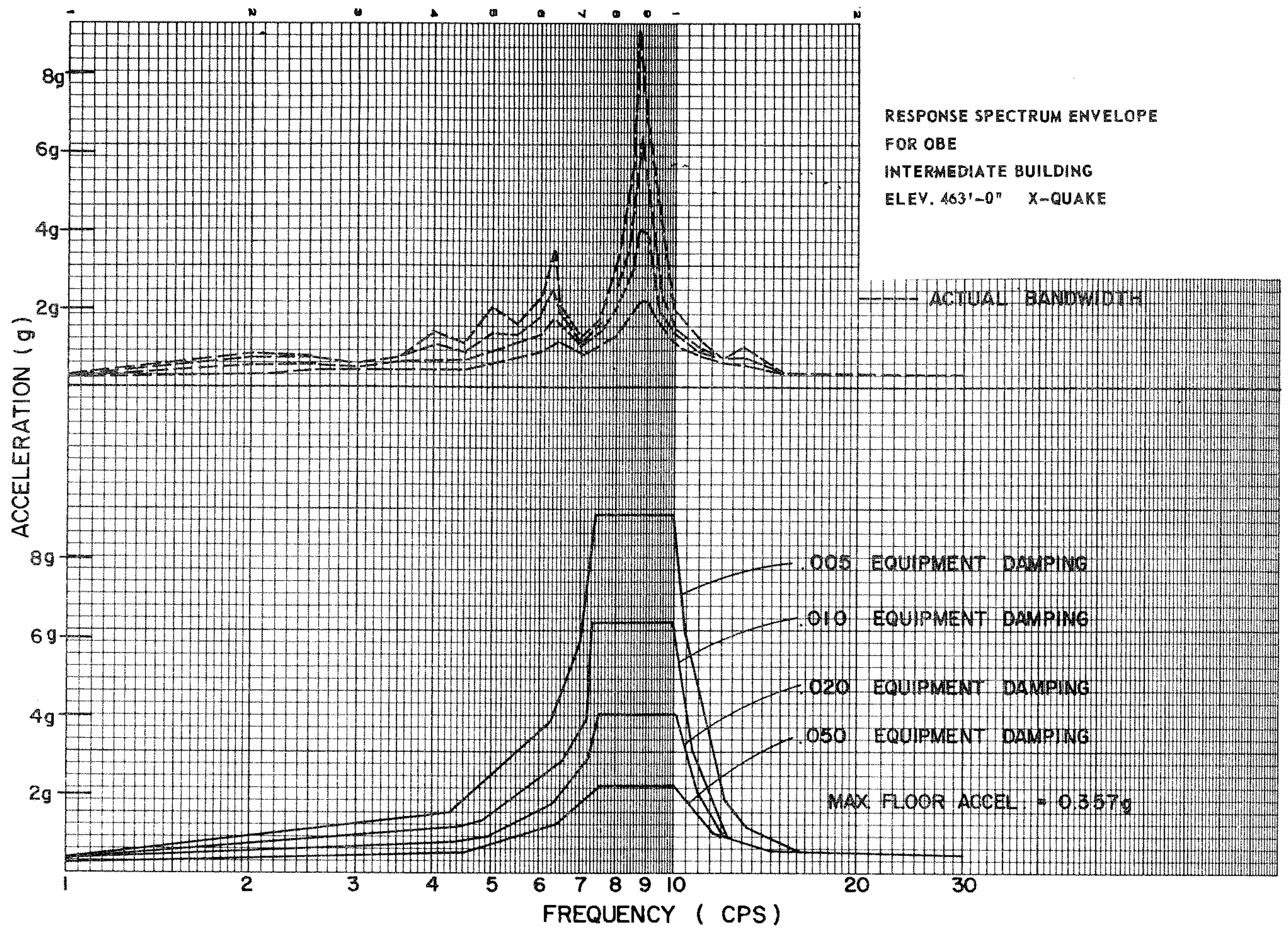
Figure 3.7-32

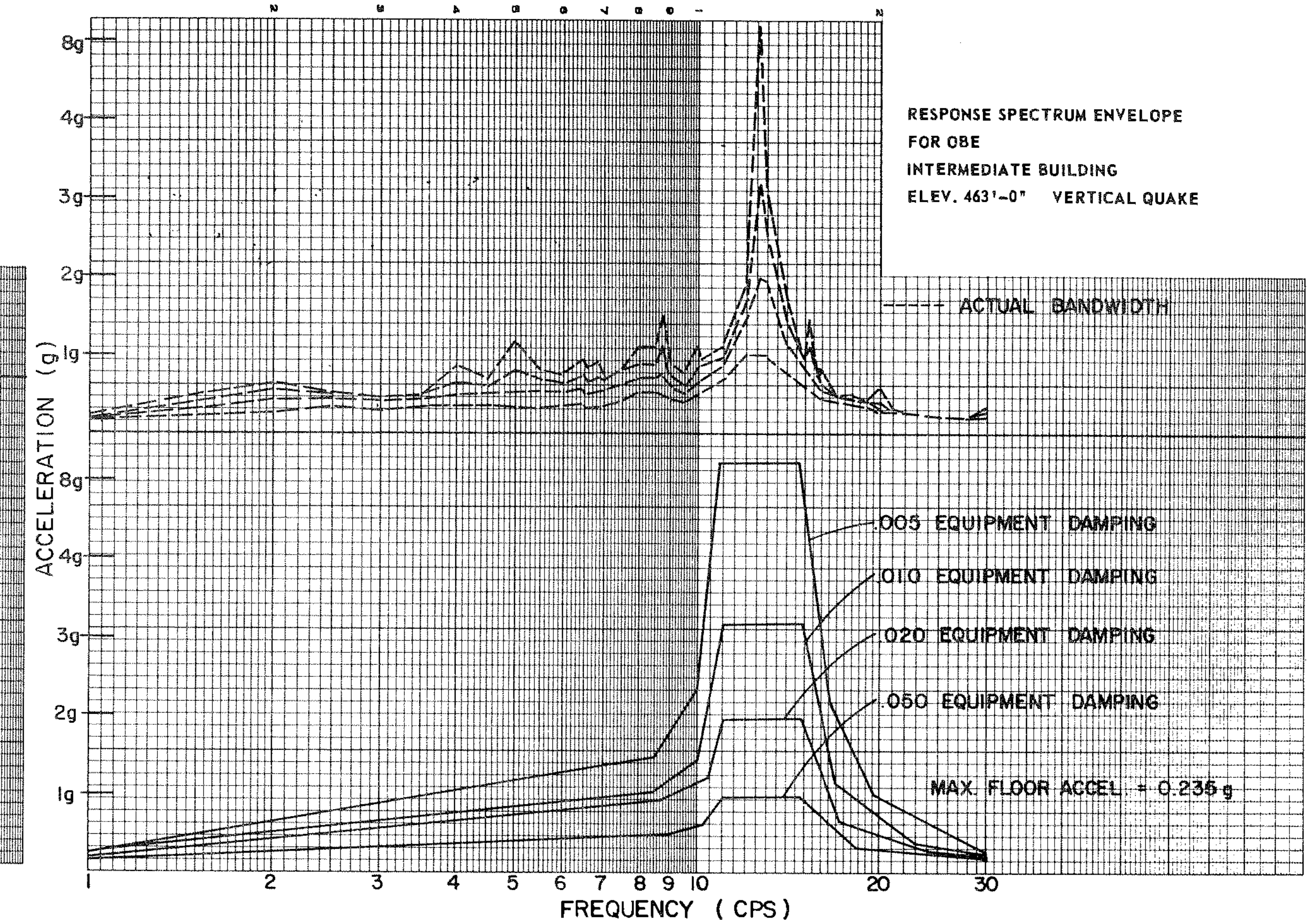


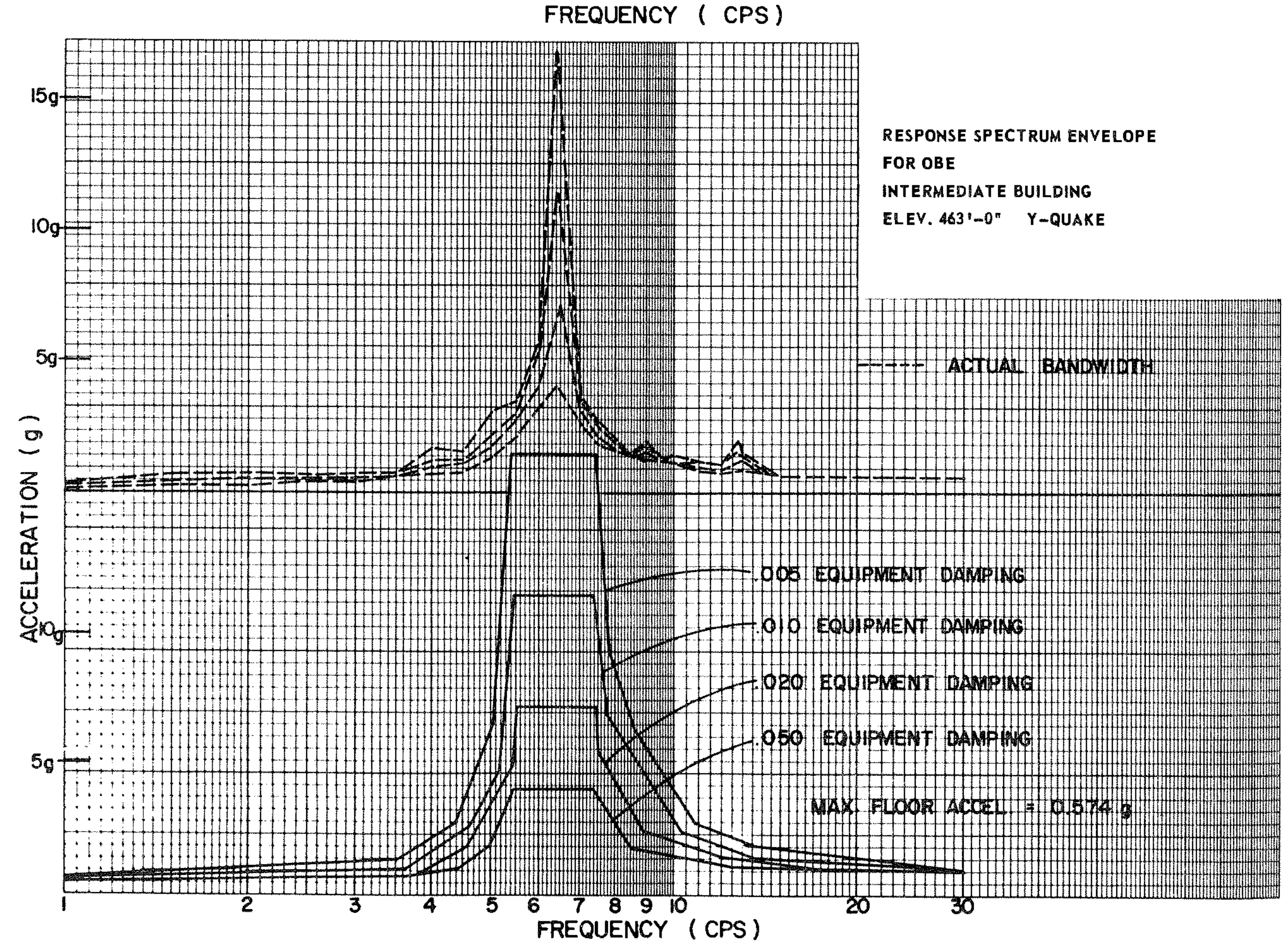
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Intermediate Building
Elevation 412' - 0"

Figure 3.7-33



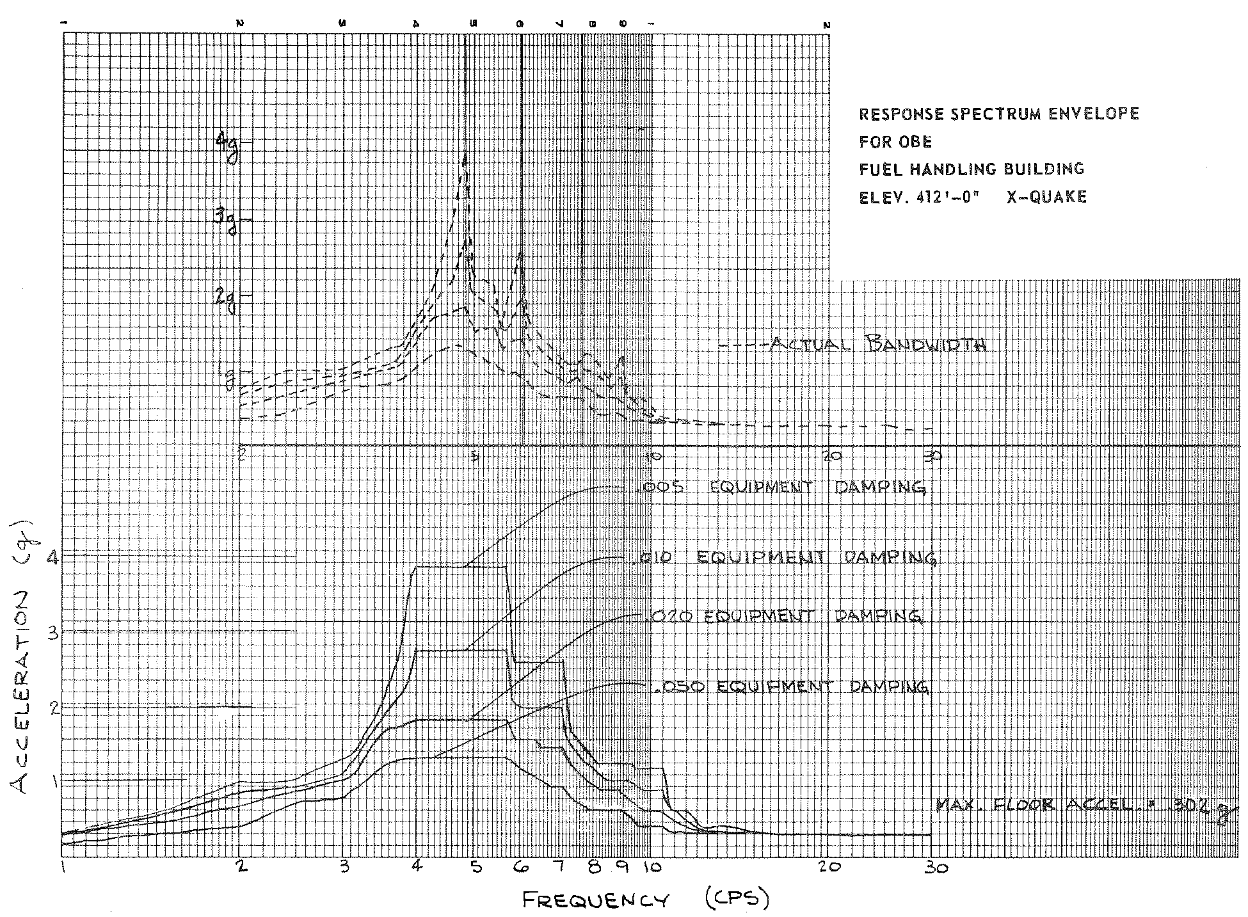


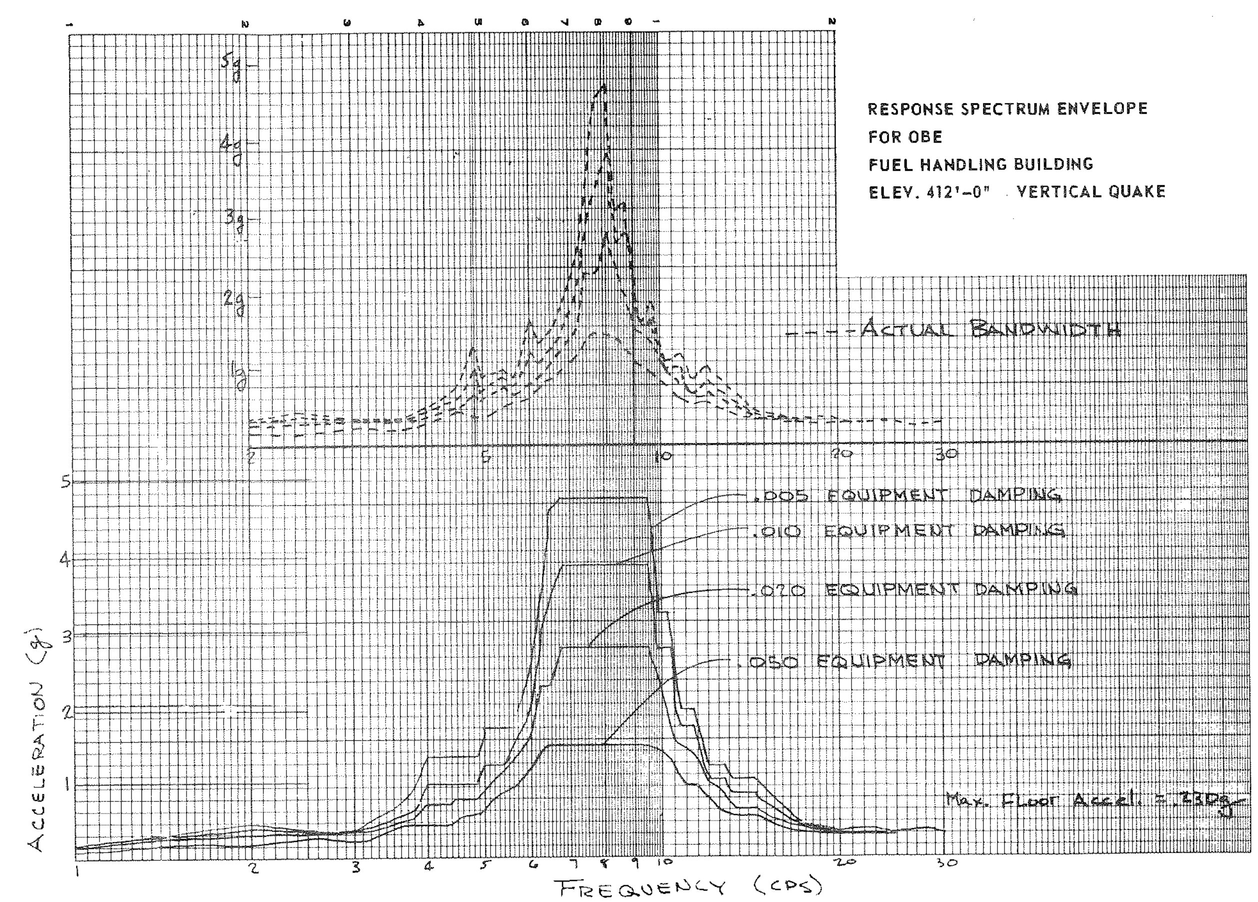


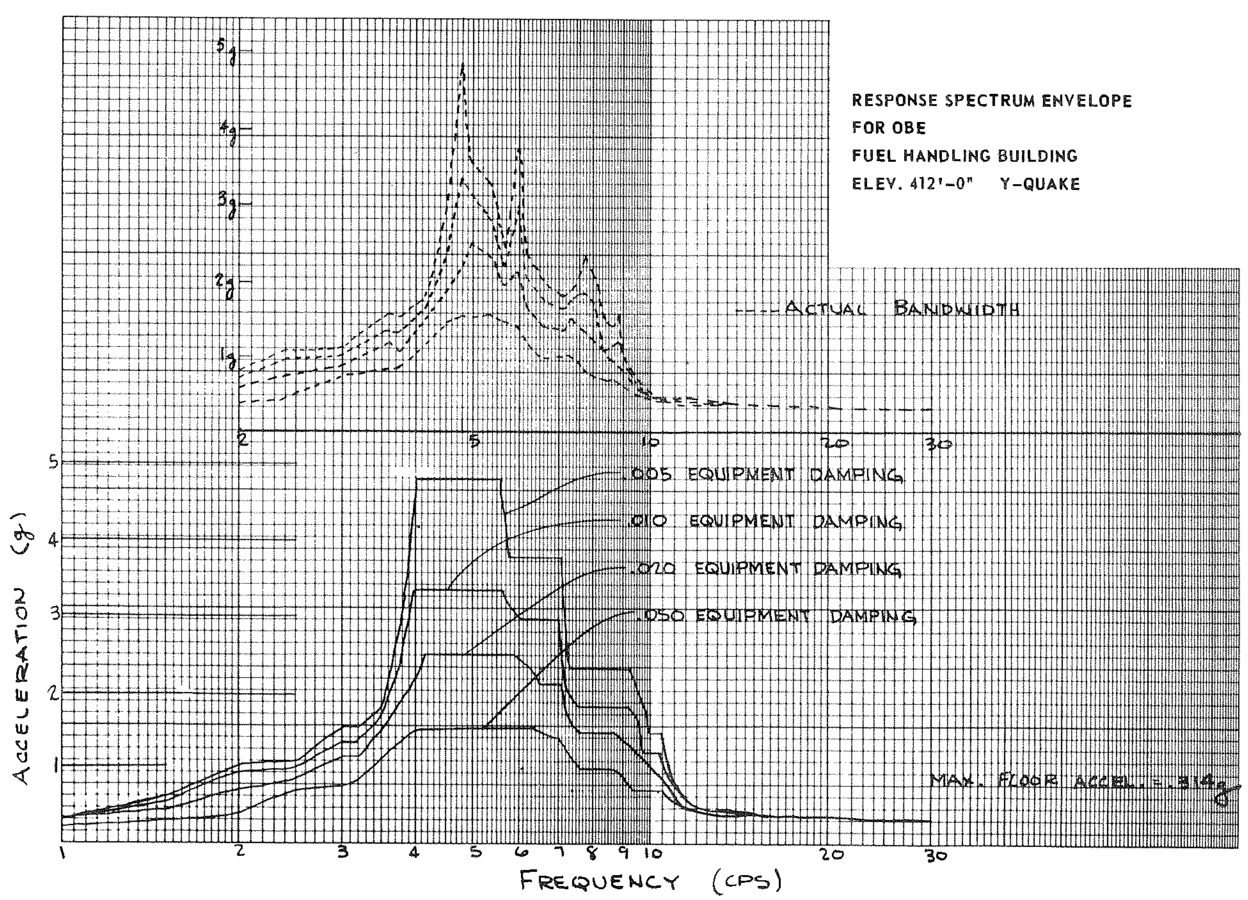
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Intermediate Building
Elevation 463' - 0"

Figure 3.7-34



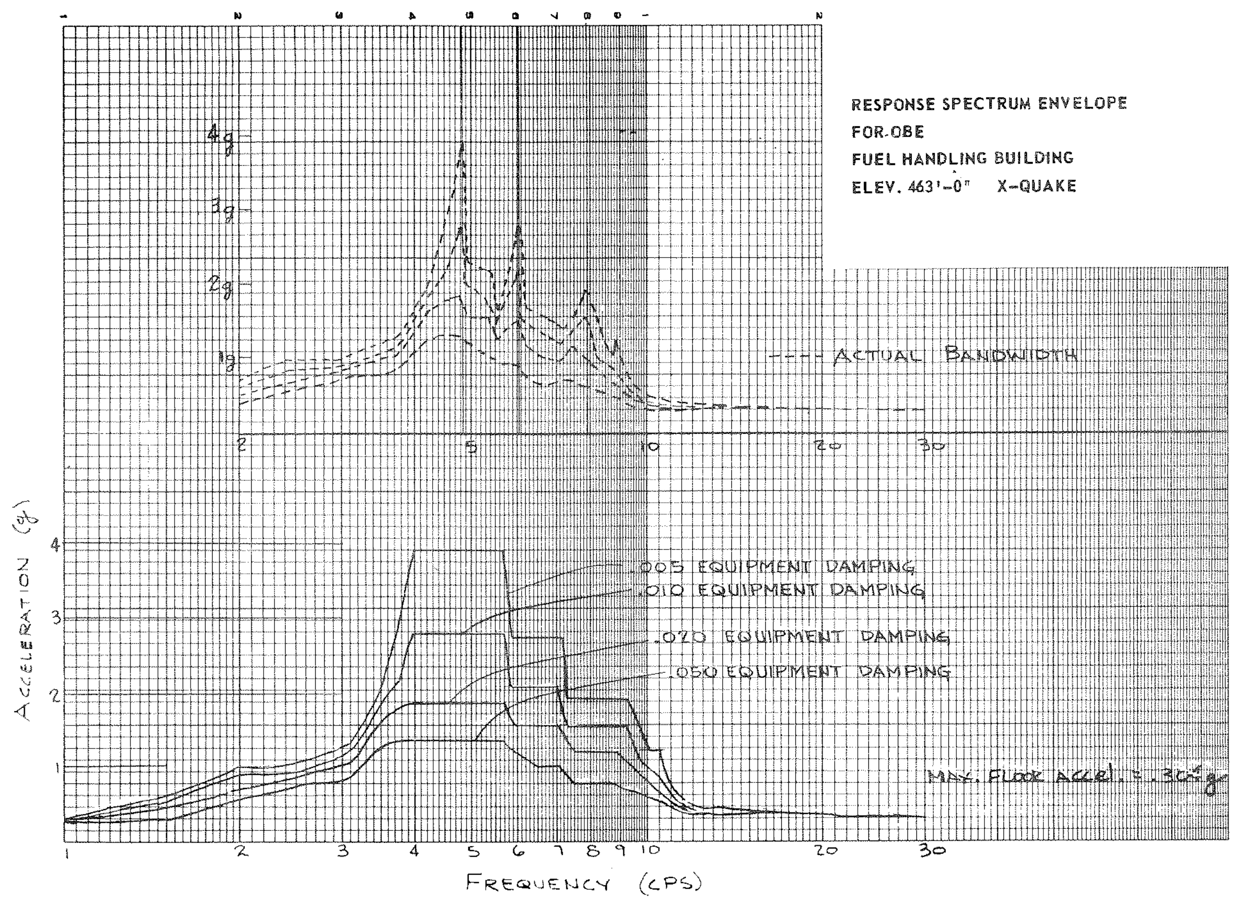


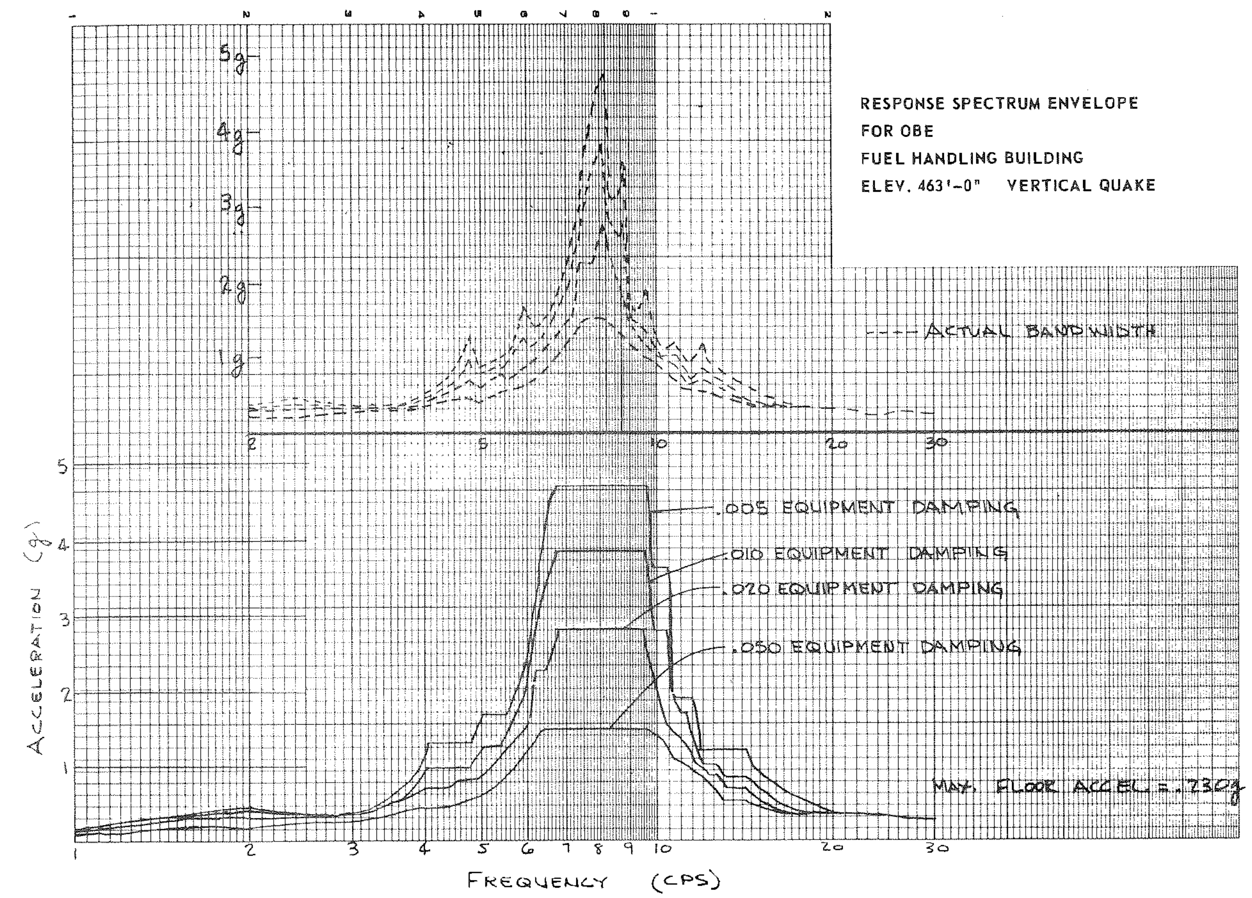


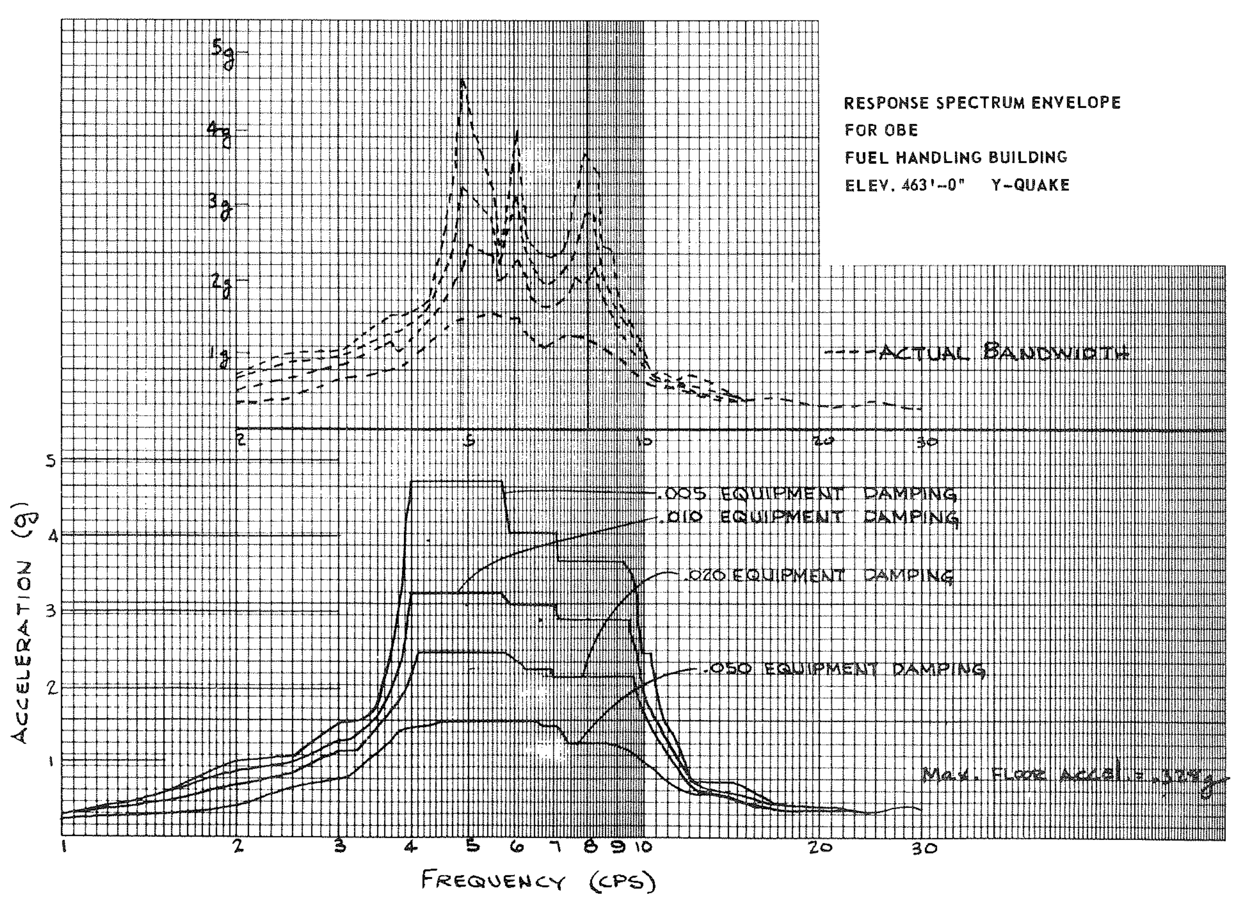
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Fuel Handling Building
Elevation 412' - 0"

Figure 3.7-35



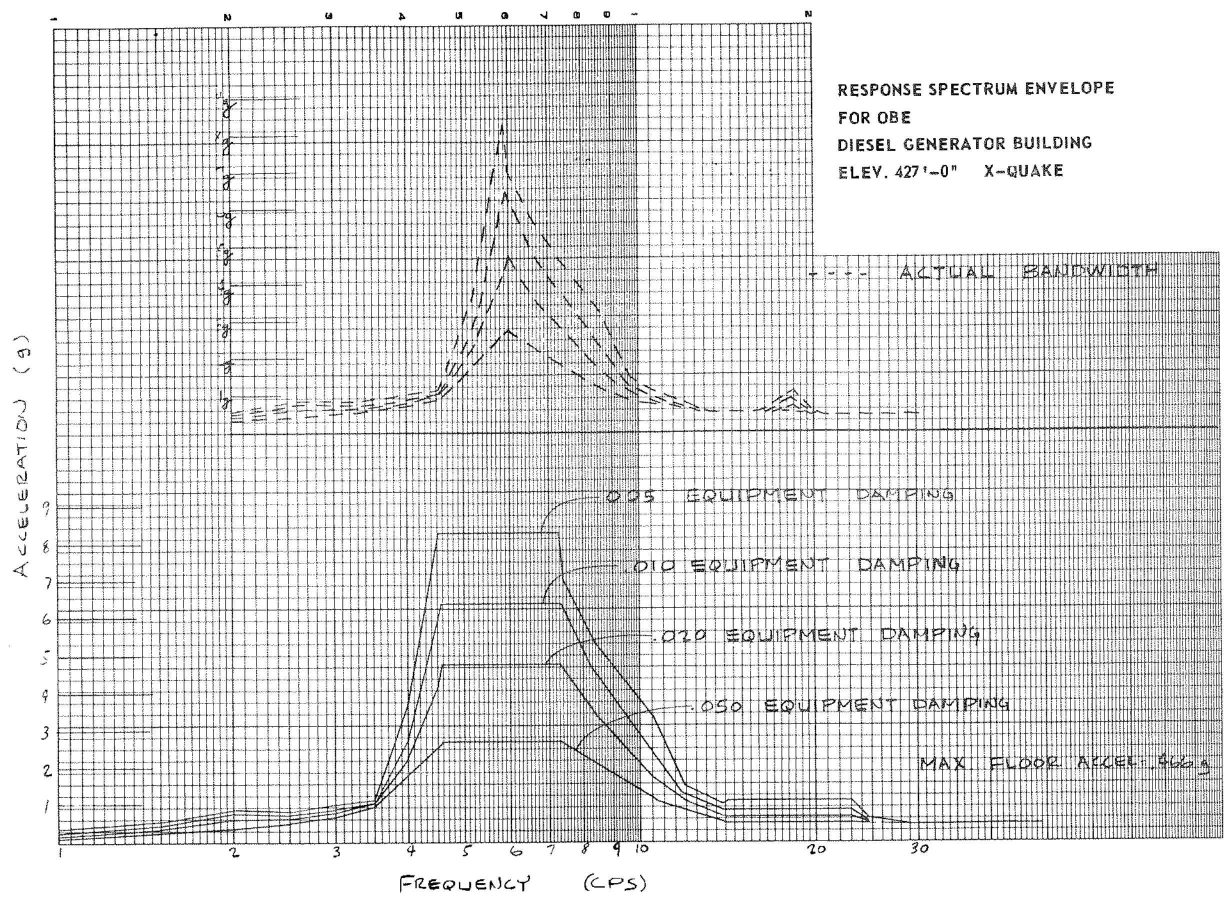


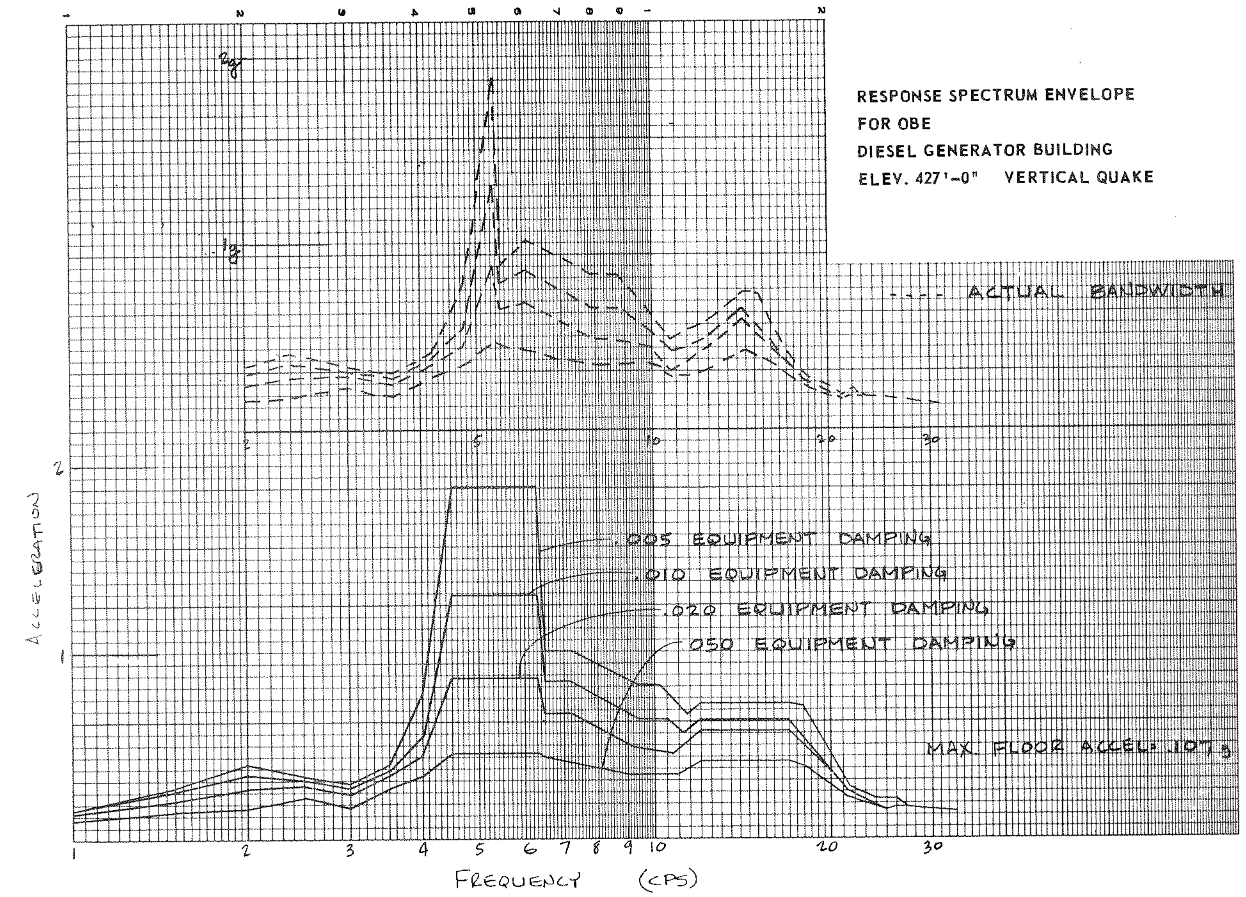


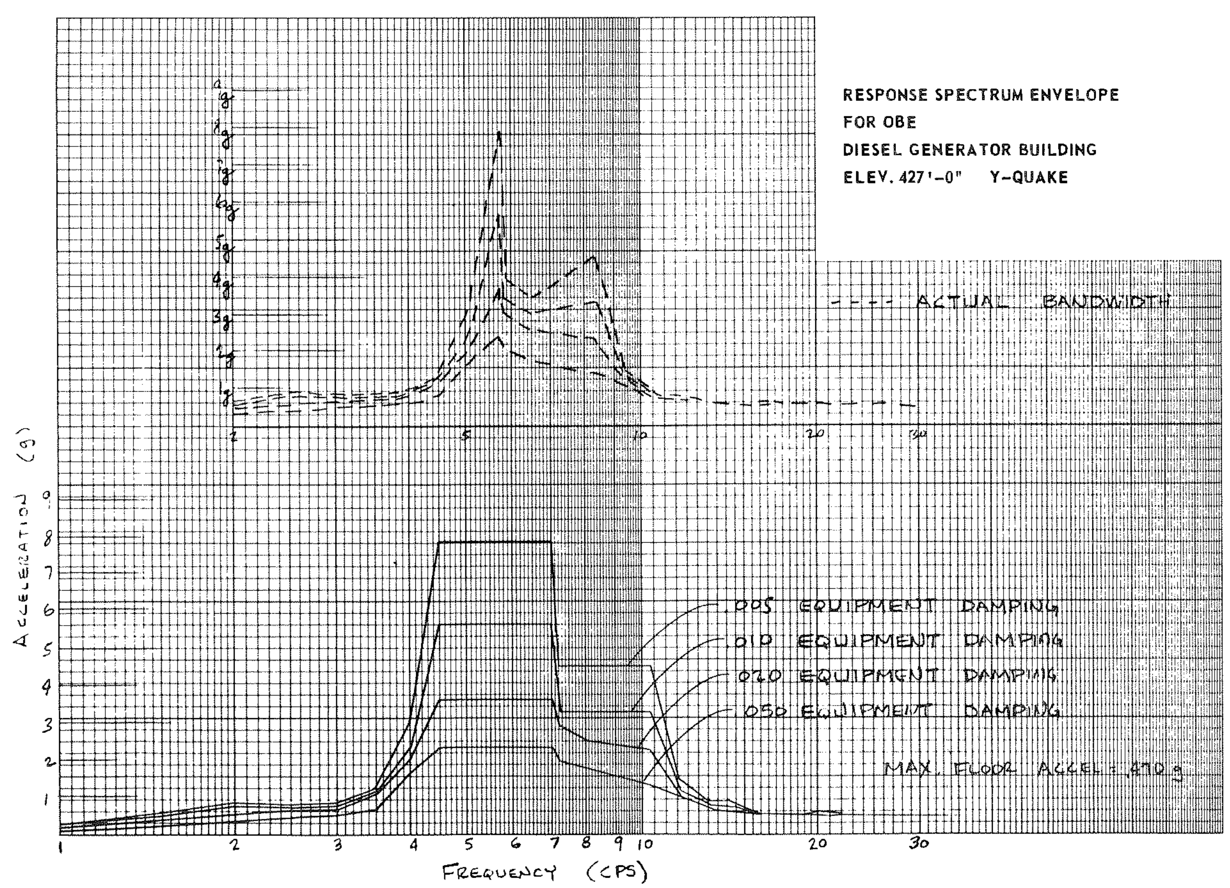
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Fuel Handling Building
Elevation 463' - 0"

Figure 3.7-36



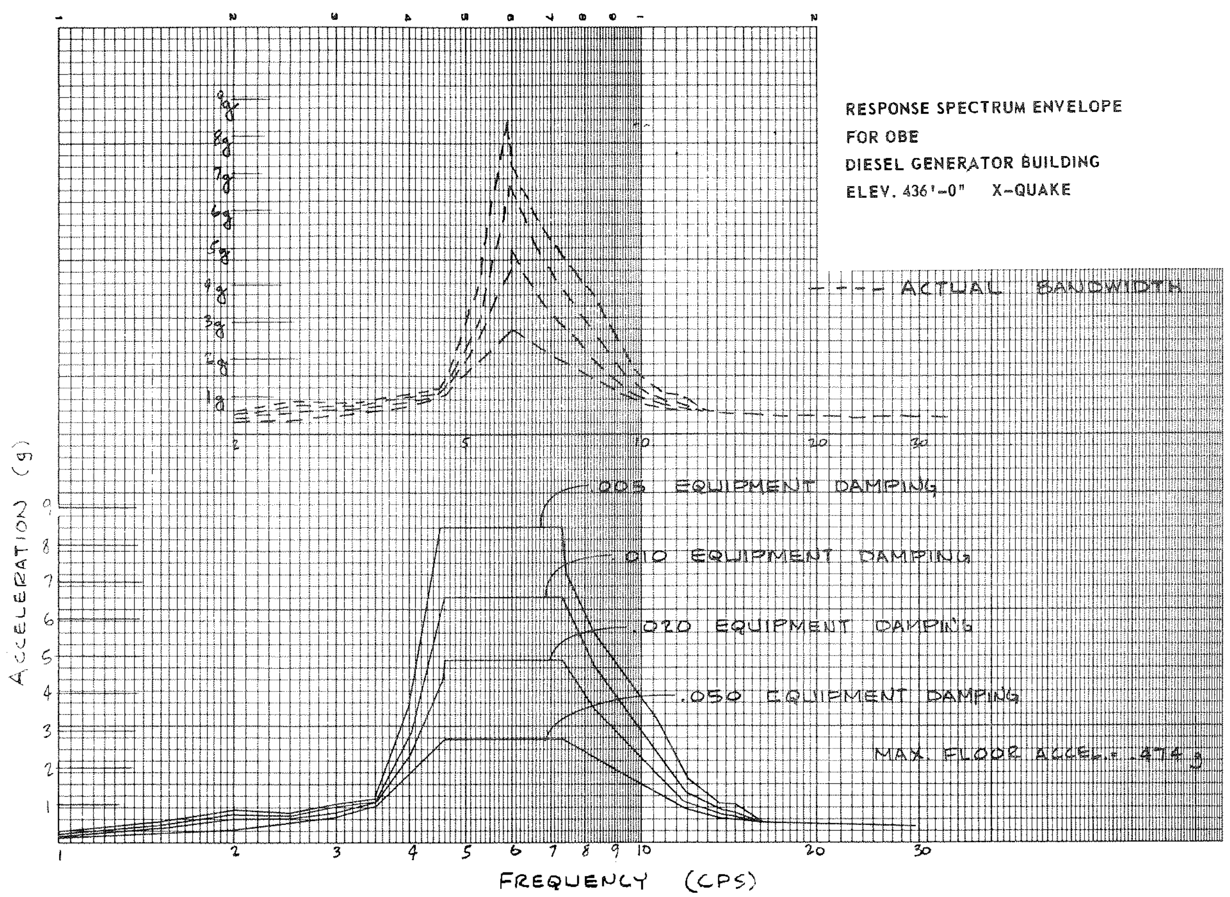


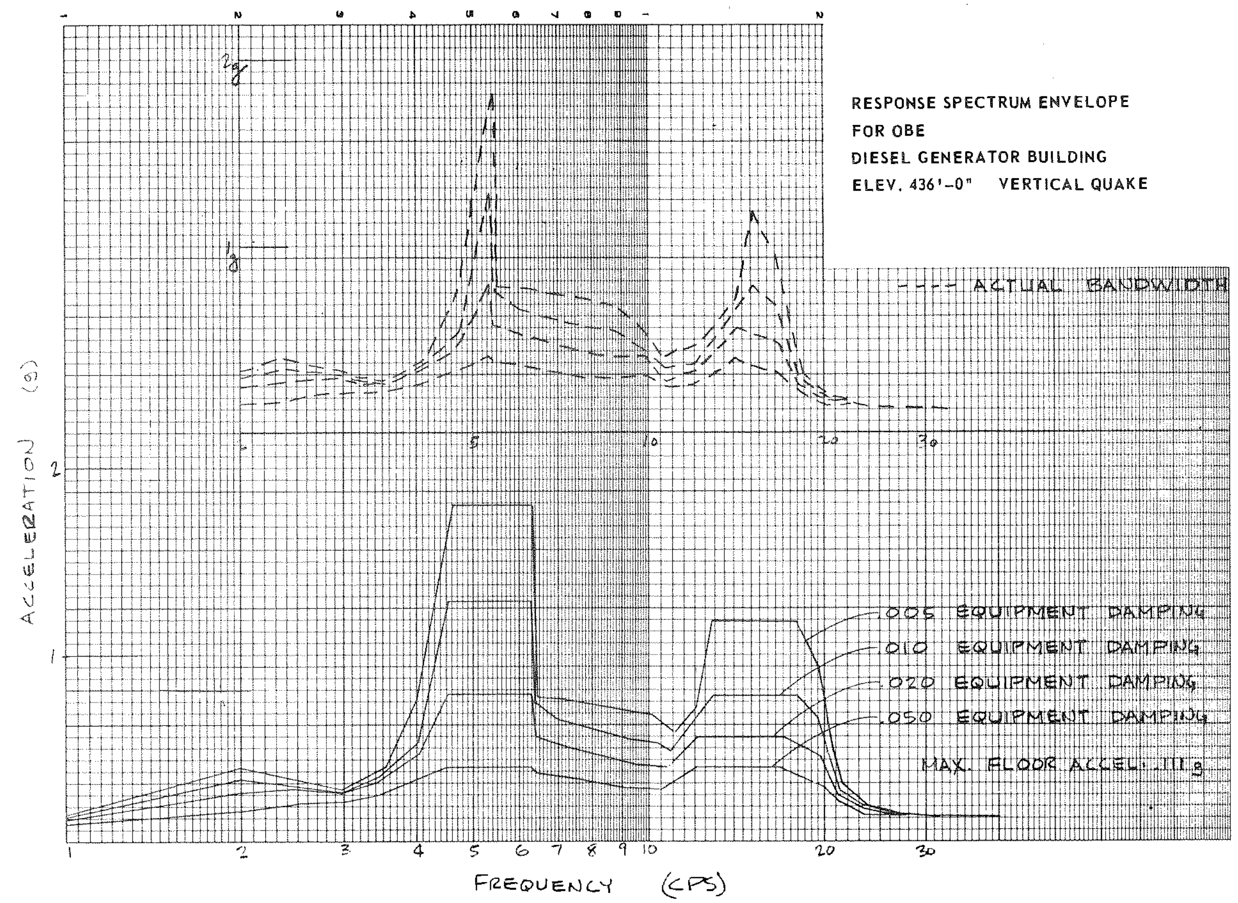


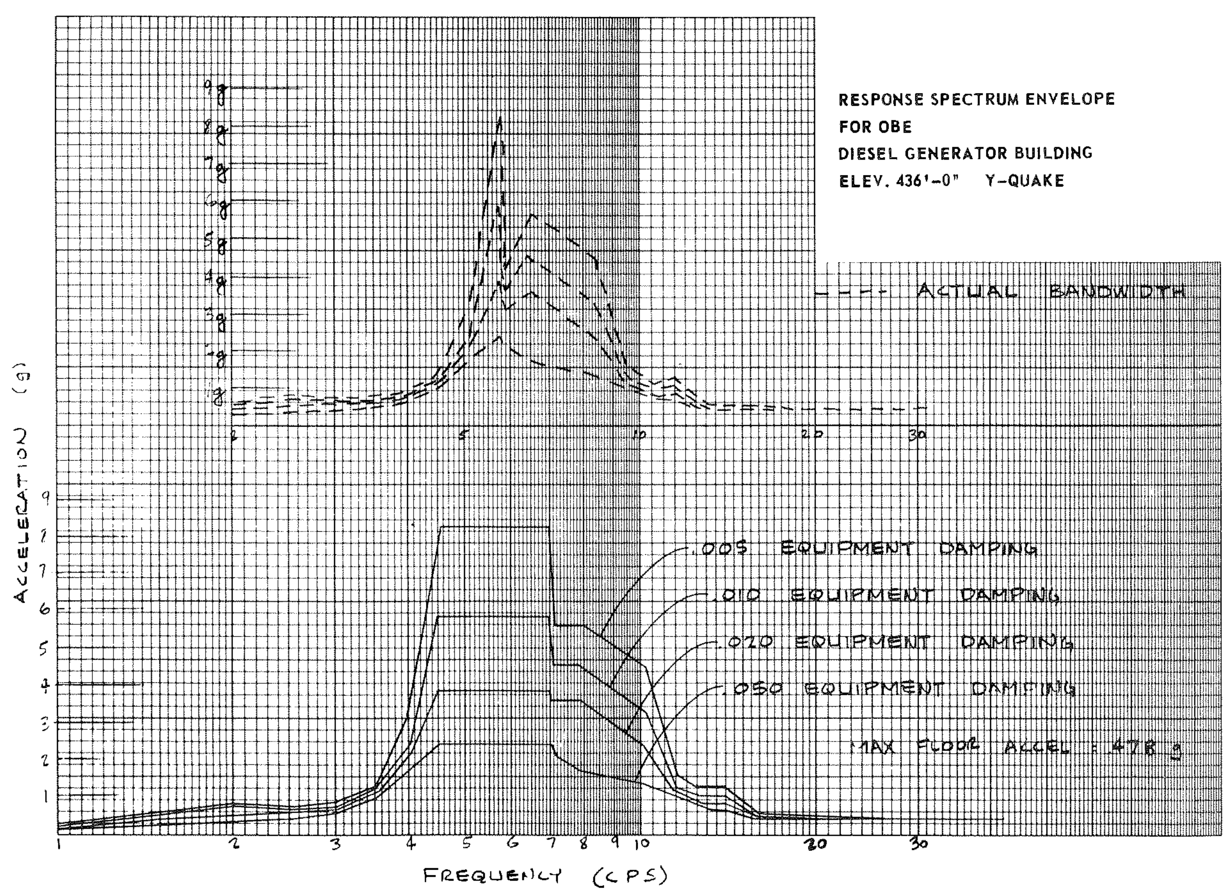
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Diesel Generator Building
Elevation 427' - 0"

Figure 3.7-37



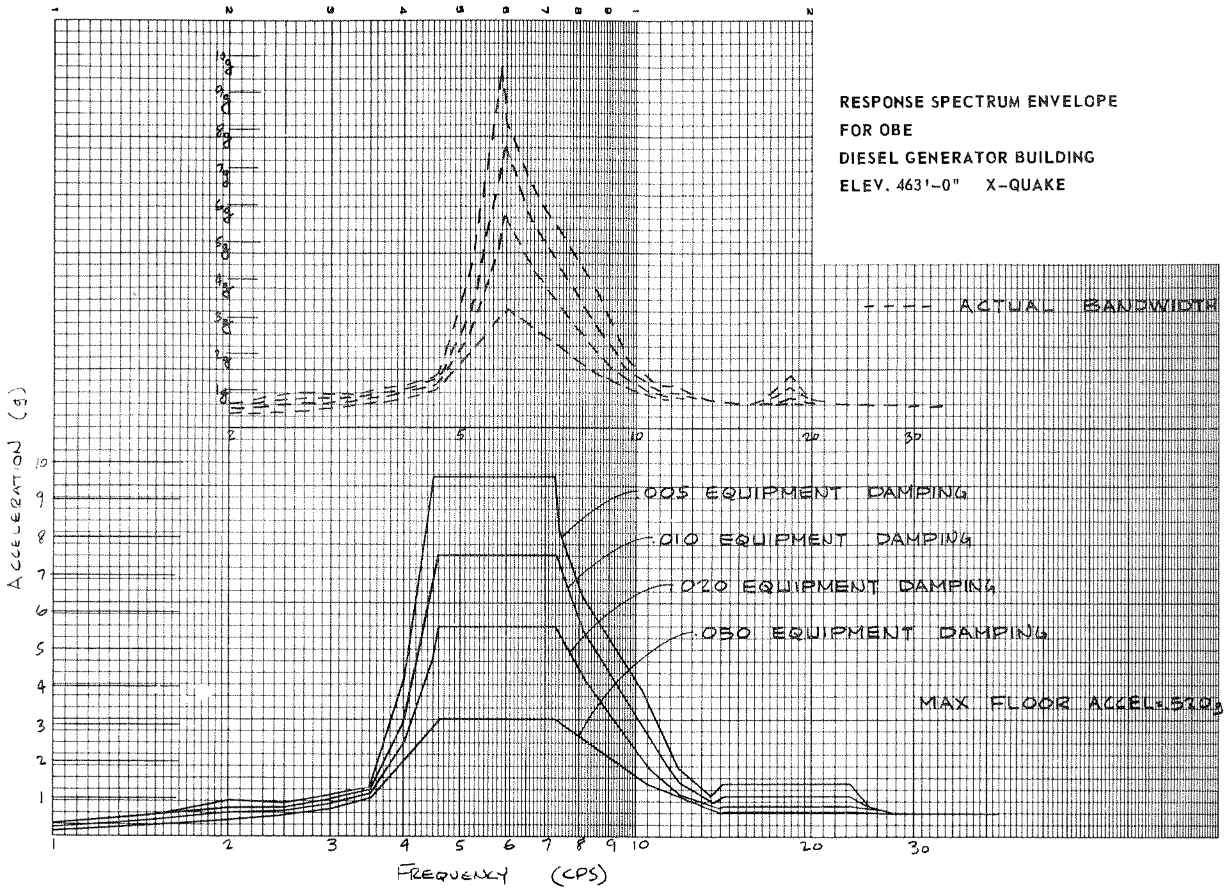


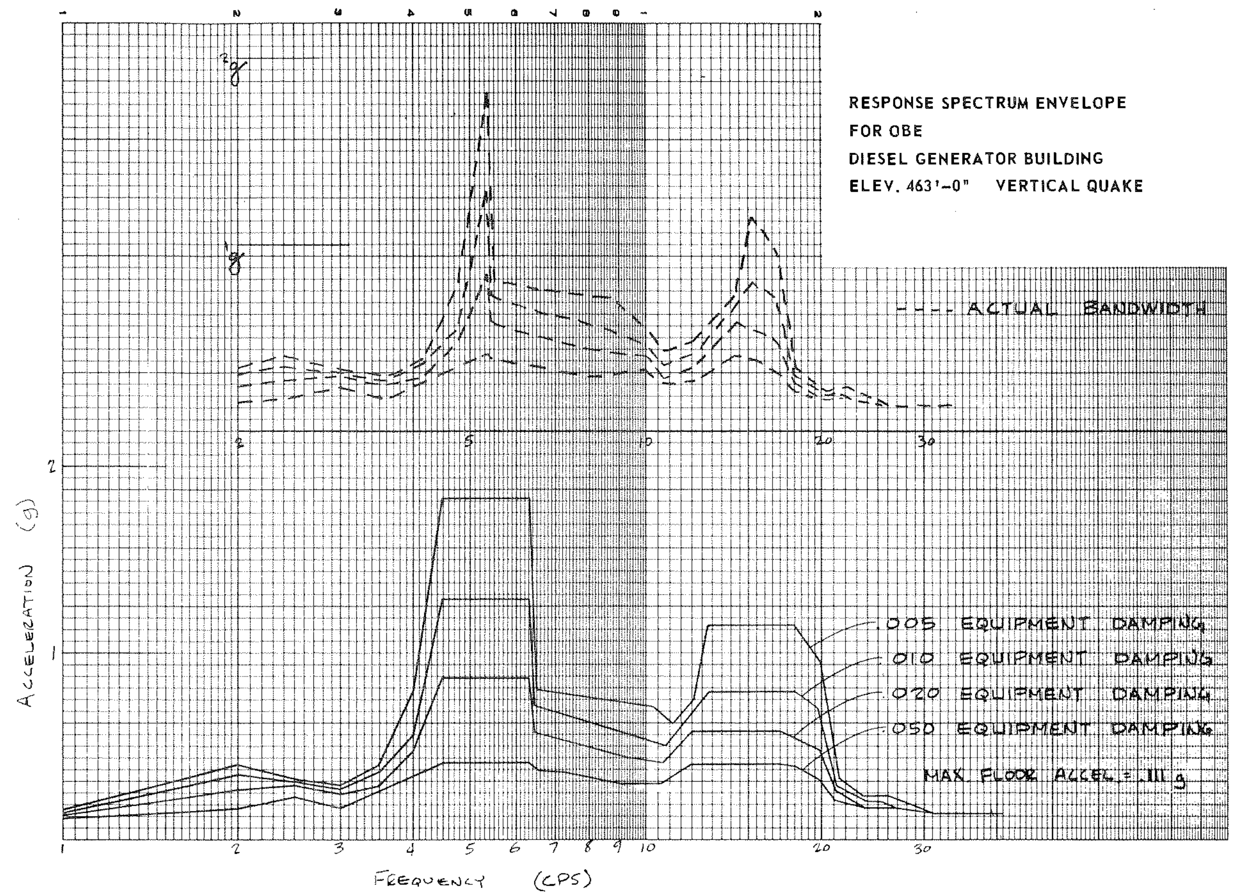


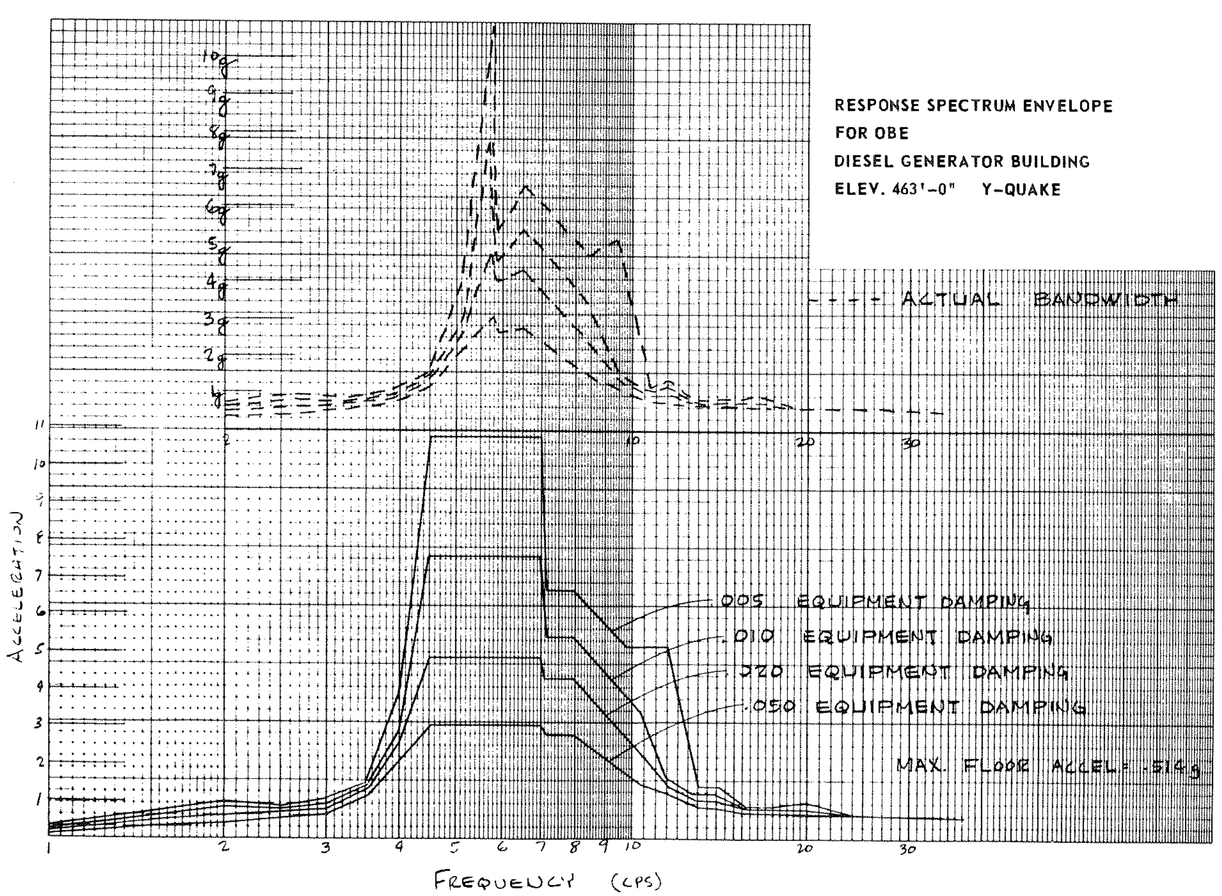
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Diesel Generator Building
Elevation 436' - 0"

Figure 3.7-38



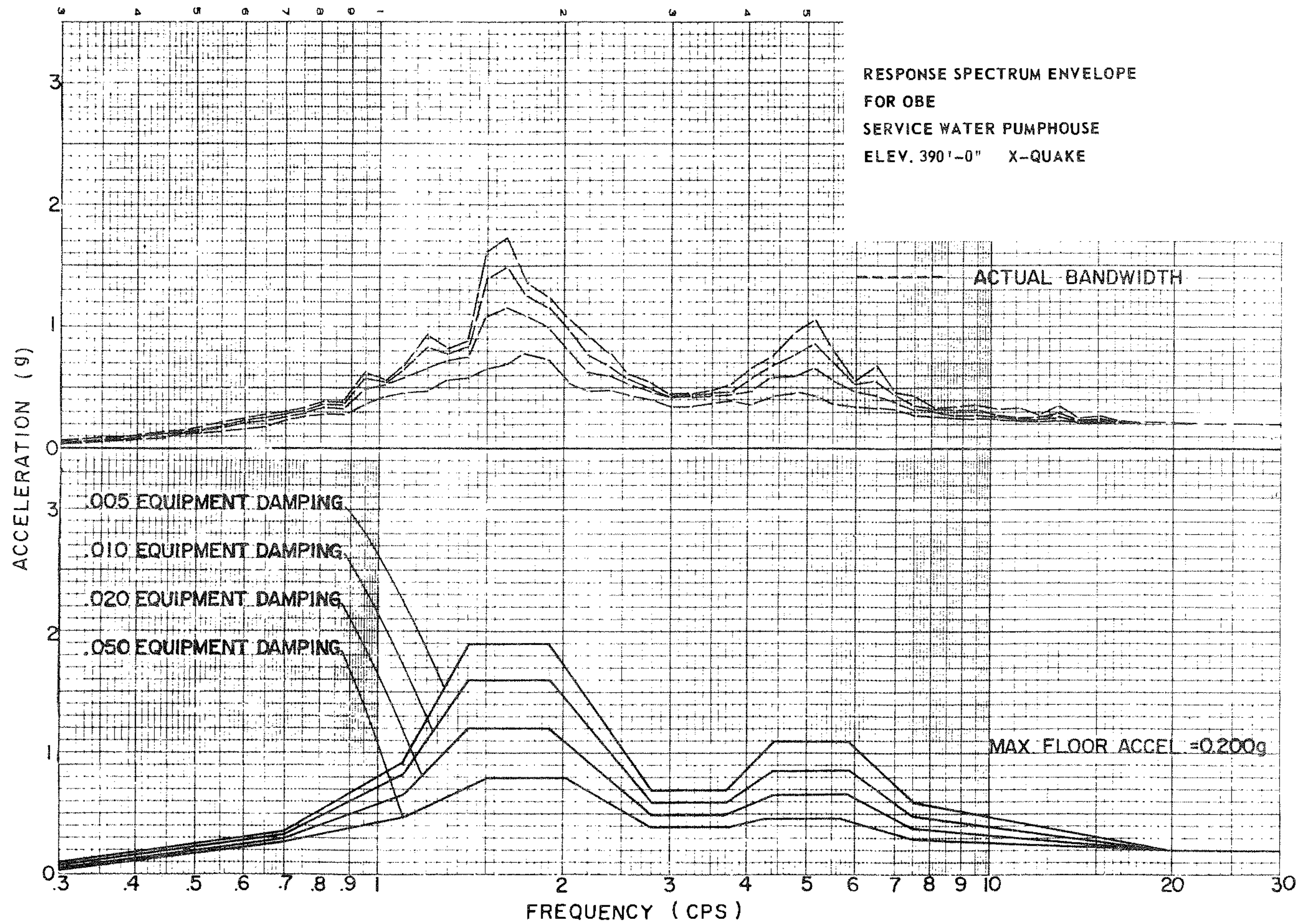


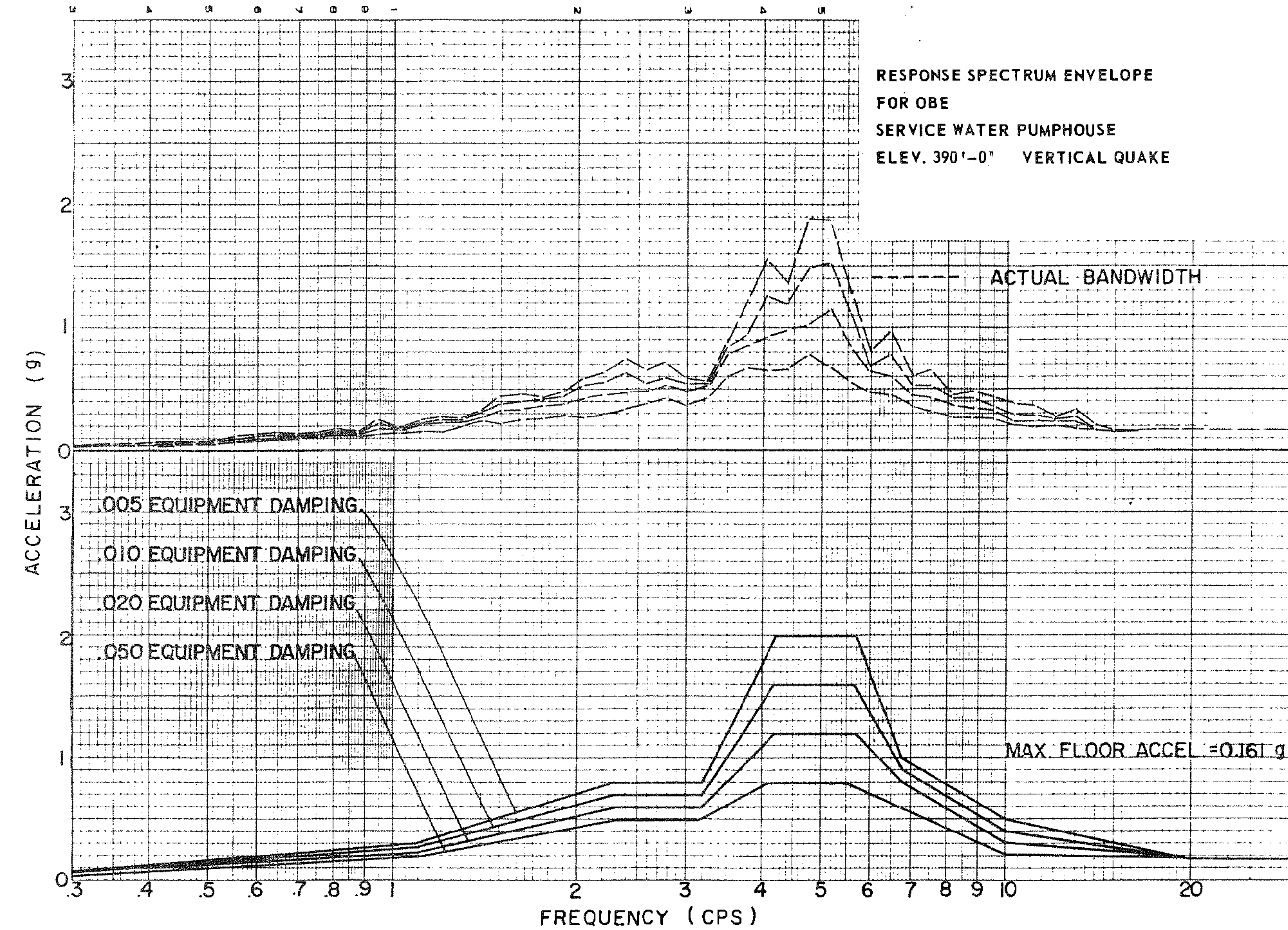


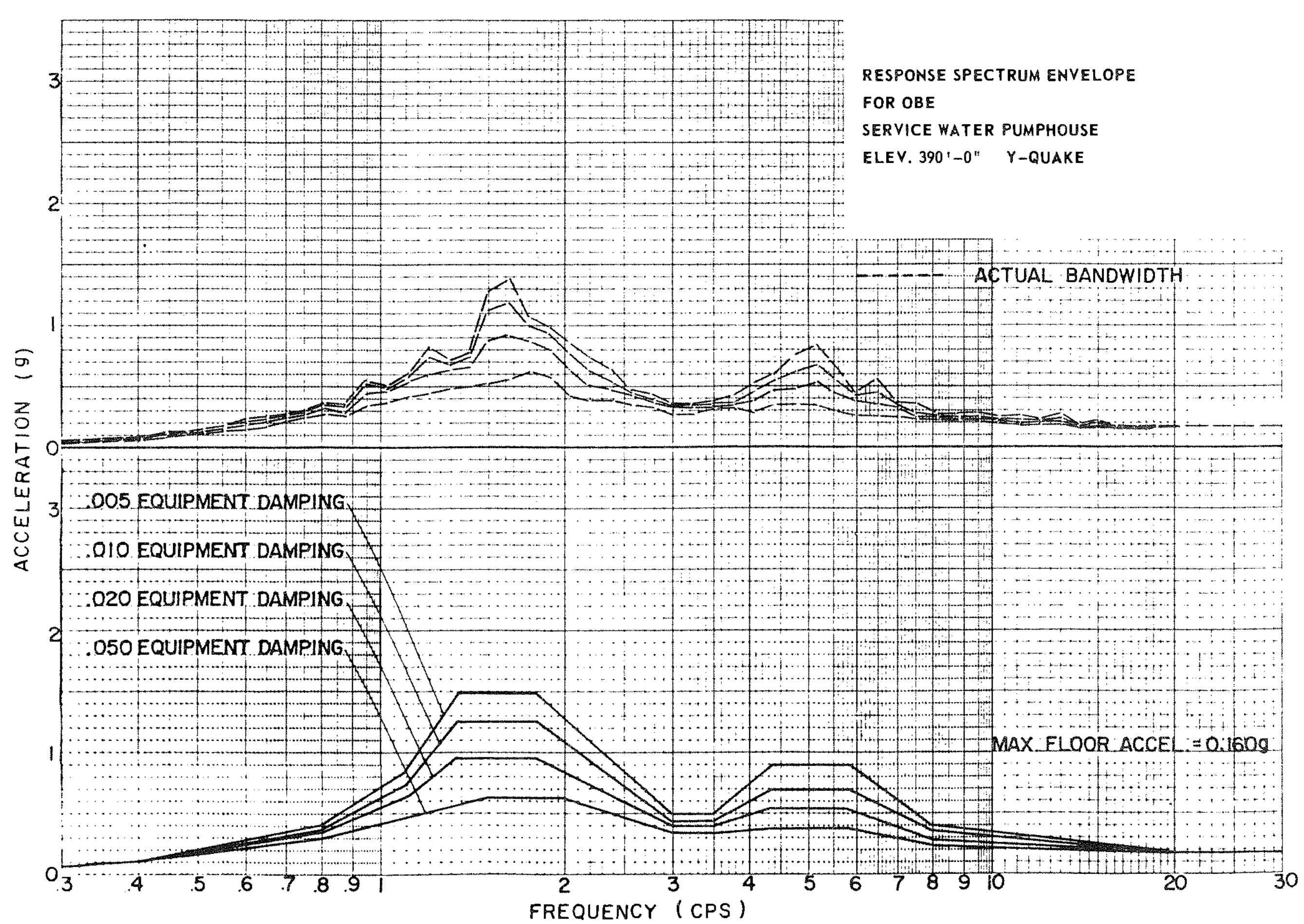
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Diesel Generator Building
Elevation 463' - 0"

Figure 3.7-39



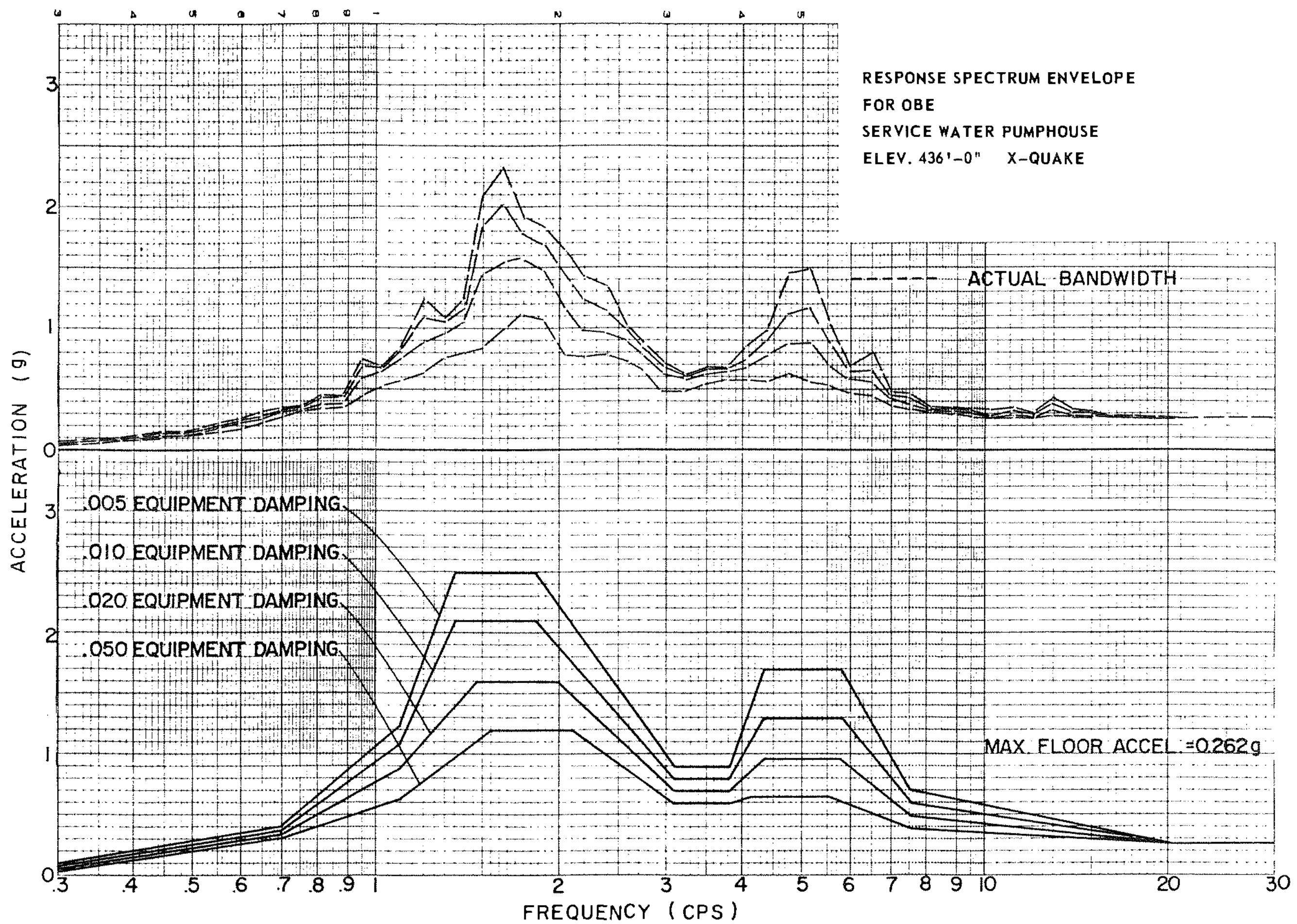


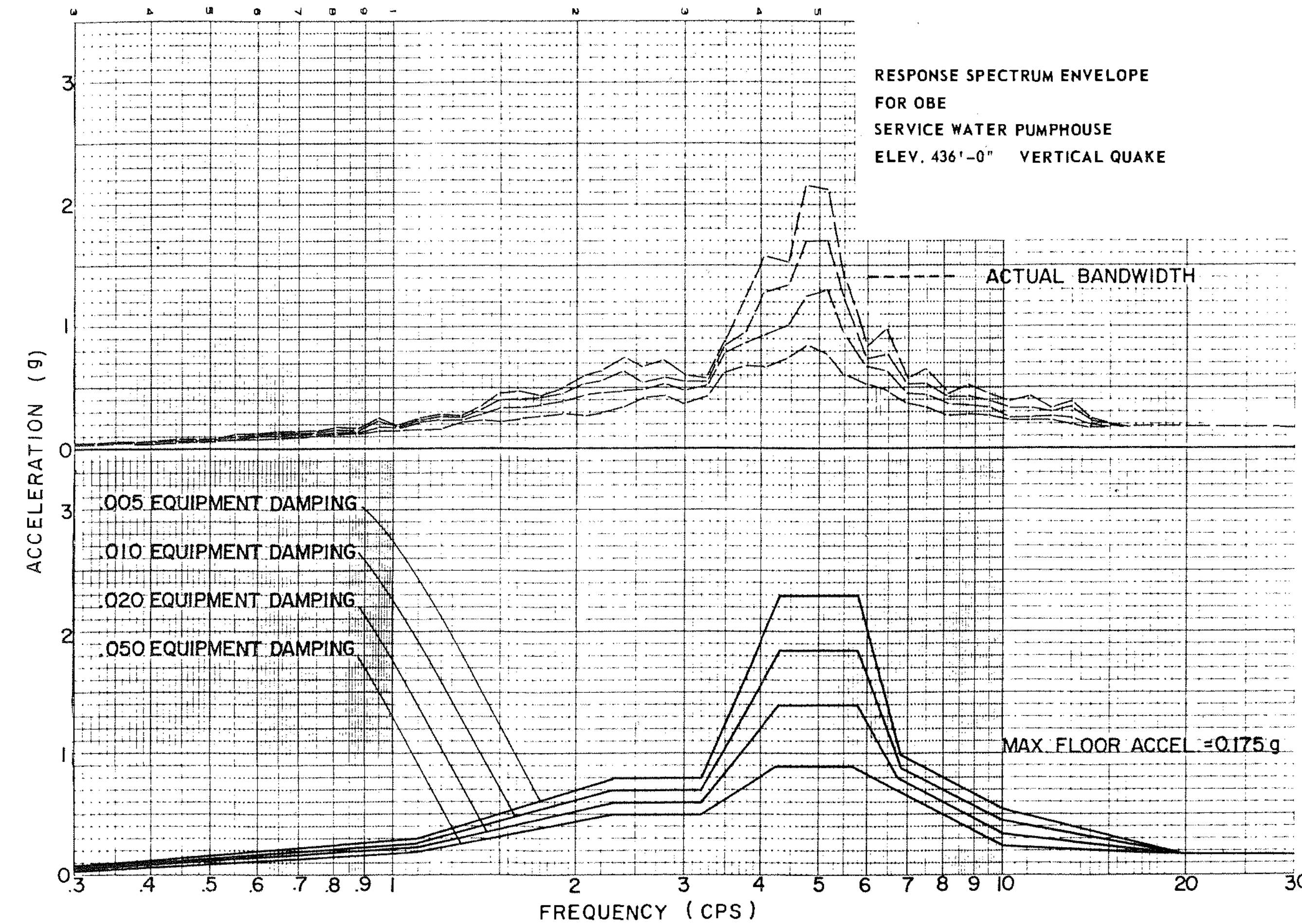


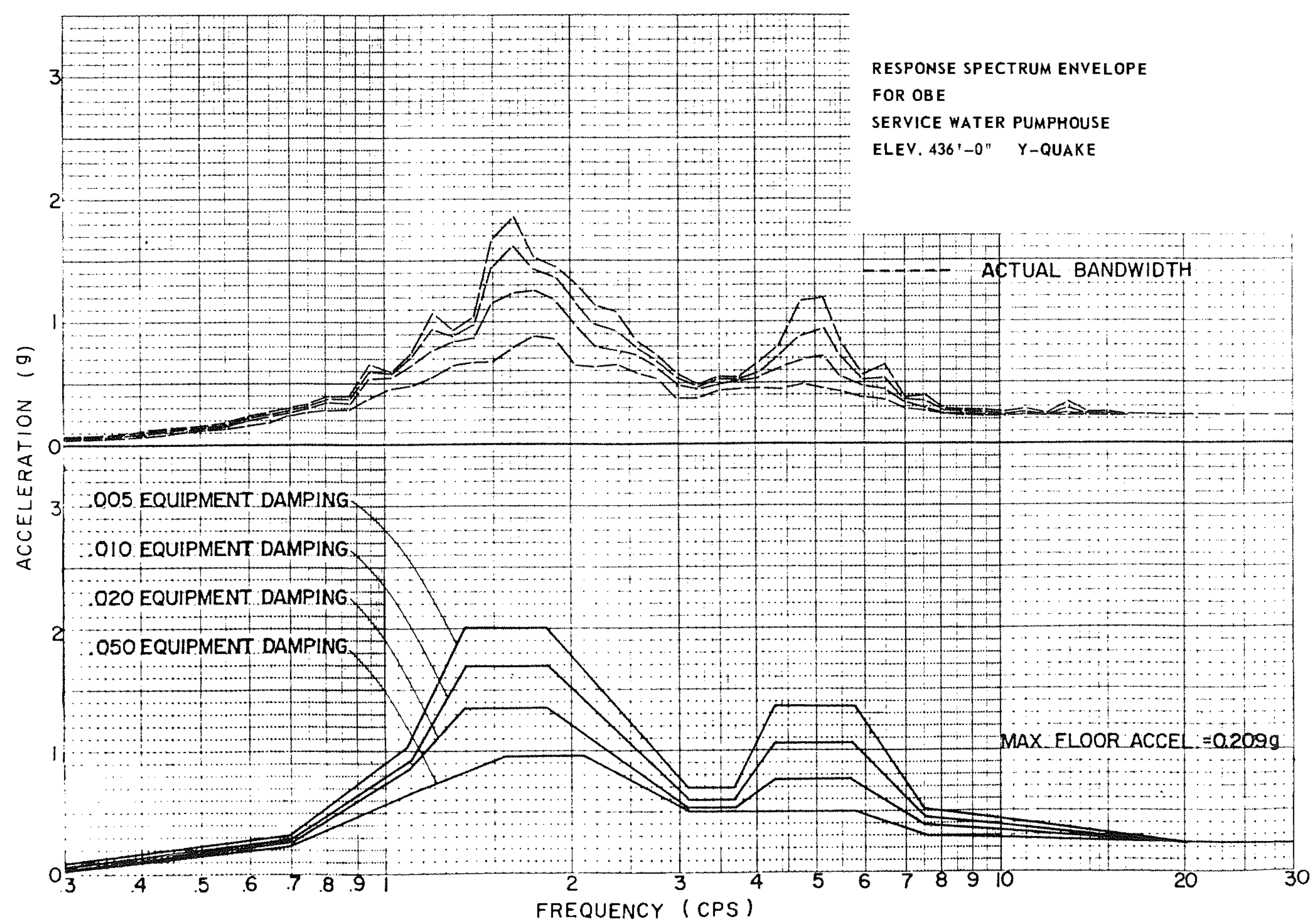
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Service Water Pumphouse
Elevation 390' - 0"

Figure 3.7-40



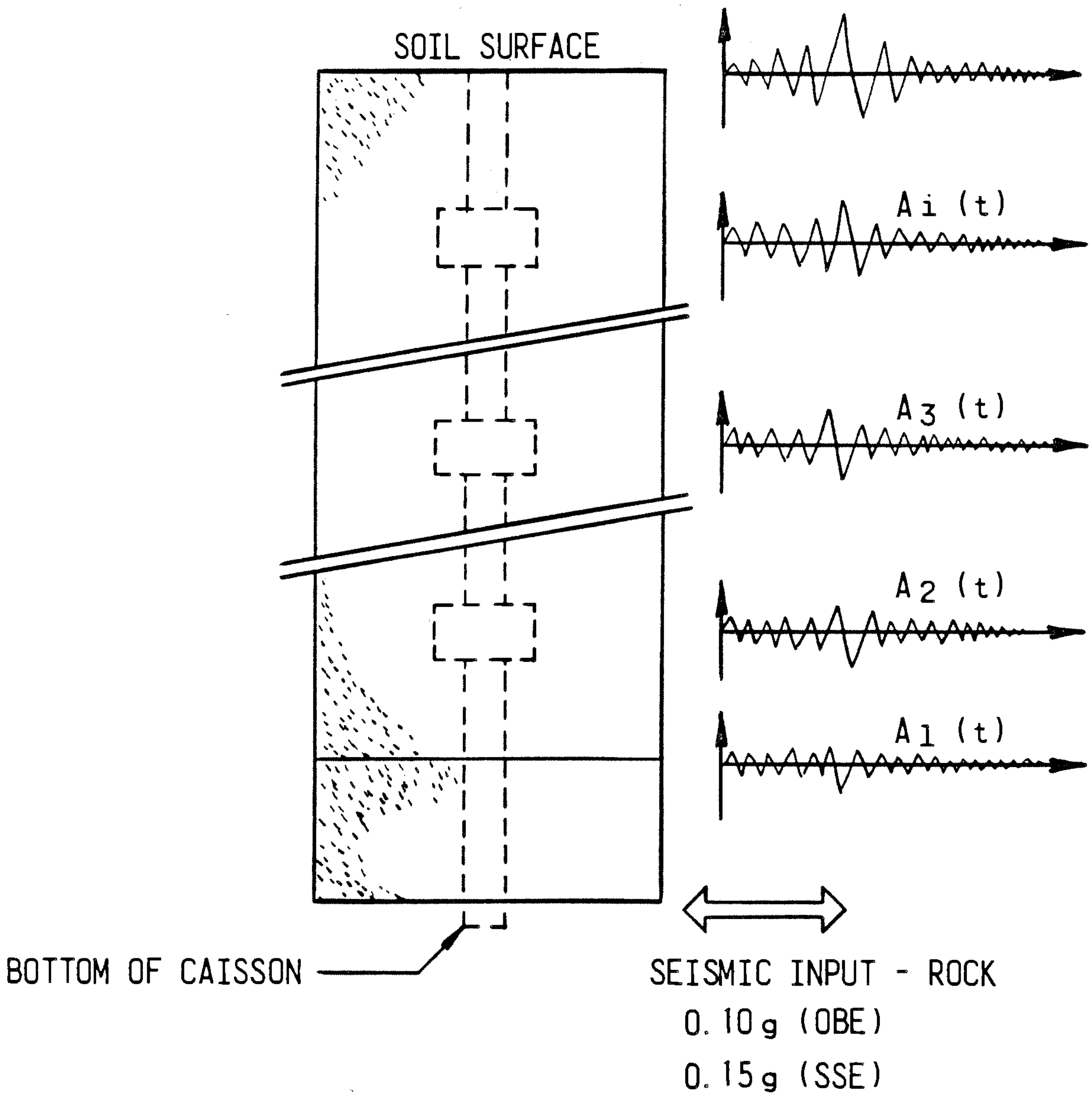




SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Response Spectrum Envelopes
Service Water Pumphouse
Elevation 436' - 0"

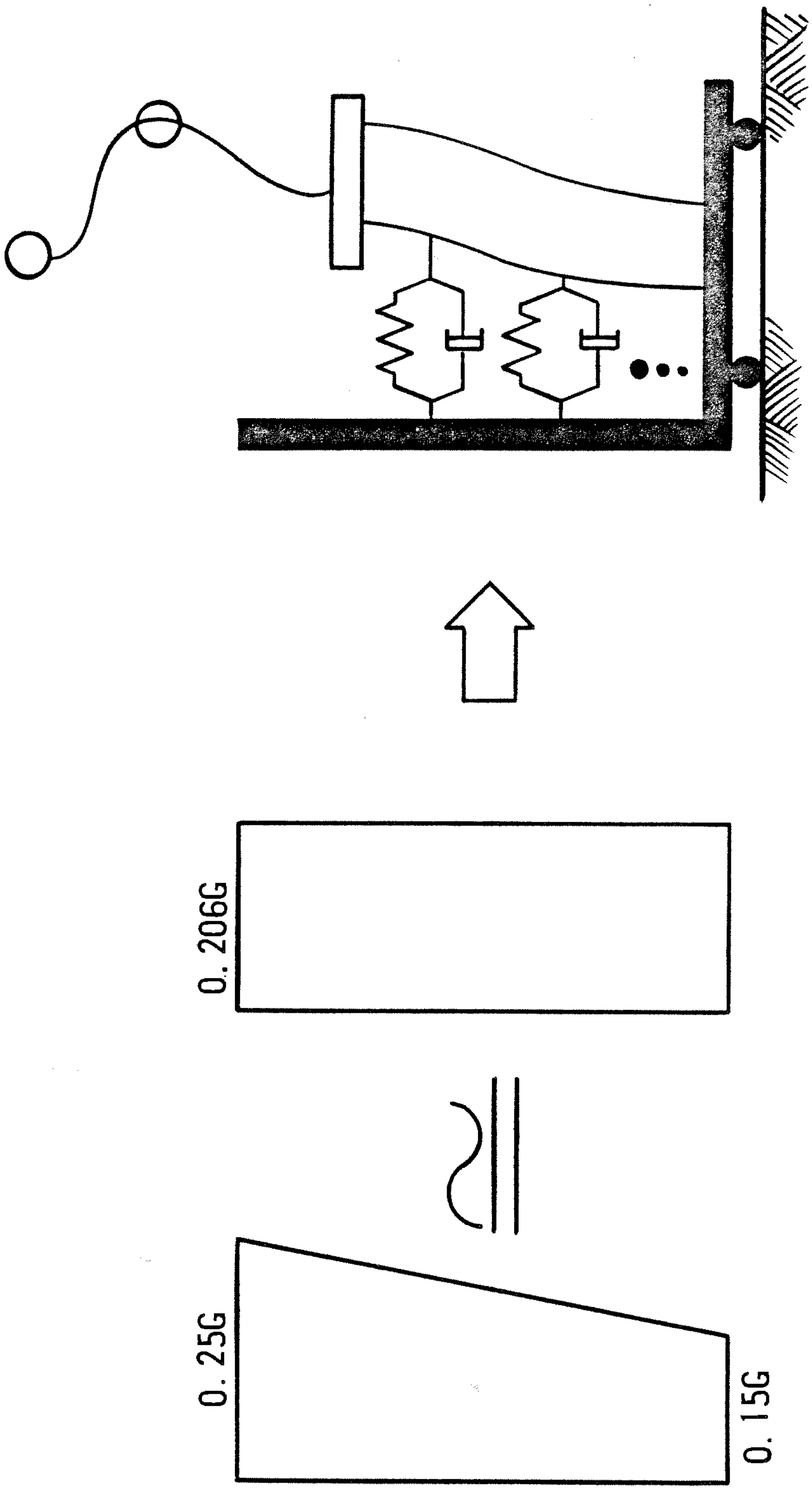
Figure 3.7-41



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Soil Shear Beam Model

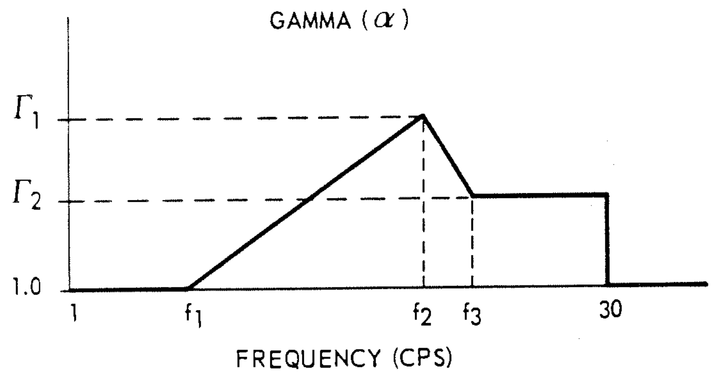
Figure 3.7-42



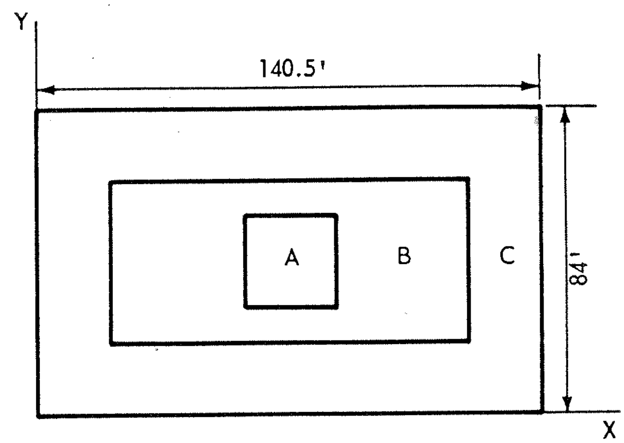
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Equivalent Acceleration Based Upon Equalized Energy

Figure 3.7-43



CONTROL BUILDING PLAN



ZONE	COORDINATES (FT.)	
	X	Y
A	57.5 ~ 82.5	29.5 ~ 54.5
B	20.0 ~ 57.5 82.5 ~ 120.5	20.0 ~ 29.5 54.5 ~ 64.0
C	REMAINING AREA	

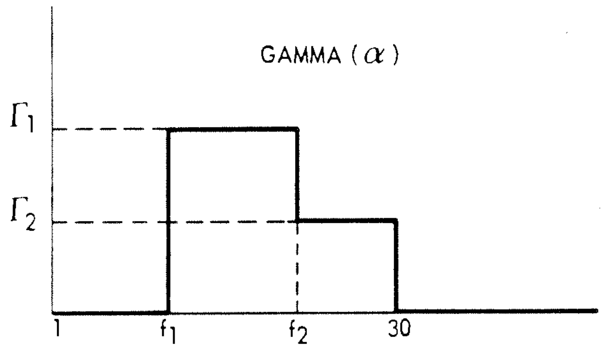
ELEV.	ZONE	Γ_1	Γ_2	f_1	f_2	f_3
505'	A	9.5	3.6	5.0	12.0	16.0
	B	6.9	2.9	5.0	12.0	16.0
	C	1.8	1.8	5.0	12.0	–
482'	A	4.0	1.8	5.5	16.0	19.0
	B	3.0	1.6	5.5	14.0	19.0
	C	1.2	1.2	5.5	14.0	–
463'	A	4.0	1.5	5.5	13.0	19.0
	B	3.7	1.3	5.5	12.5	19.0
	C	1.1	1.1	5.5	12.5	–
452'	A	4.0	1.3	5.5	12.0	17.0
	B	3.0	1.1	5.5	12.5	17.0
	C	1.0	1.0	–	–	–

ELEV.	ZONE	Γ_1	Γ_2	f_1	f_2	f_3
436'	A	2.2	1.0	6.0	12.5	15.0
	B	2.2	1.0	9.0	12.5	15.0
	C	1.0	1.0	–	–	–
412'	A	1.0	1.0	–	–	–
	B	1.0	1.0	–	–	–
	C	1.0	1.0	–	–	–

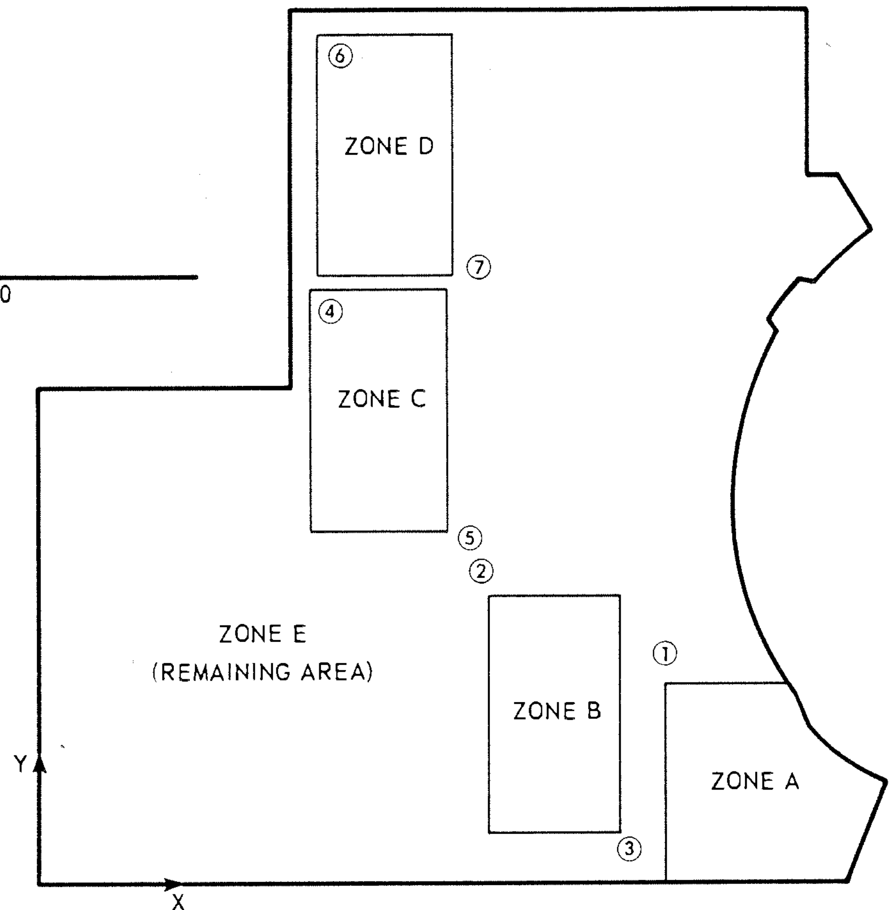
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Gamma Scaling Factor
Control Building

Figure 3.7-44



ZONE	CORNER COORDINATES (FT.)	
	X	Y
1	129.0	43.0
2	88.0	64.0
3	116.0	11.0
4	47.5	124.5
5	76.5	77.5
6	52.5	184.0
7	81.5	131.0



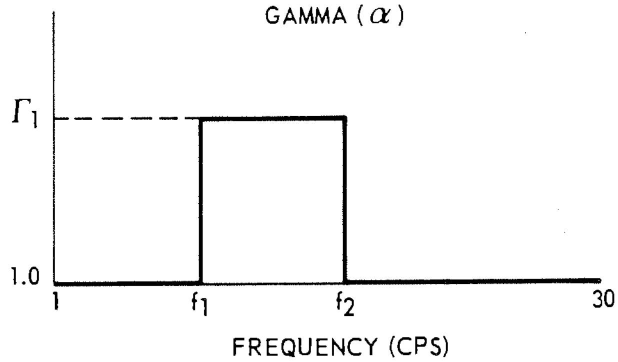
AUXILIARY BUILDING PLAN

ELEV.	ZONE	Γ_1	f_1	f_2	Γ_2
511'	C	2.6	19	30	–
	E	1.0	–	–	1.0
485'	A	8.9	15	26	2.0
	B	4.4	19	30	–
	C	2.6	19	30	–
	D	7.9	12	24	2.7
	E	1.0	–	–	1.0
463'	A	8.9	15	26	2.0
	E	1.0	–	–	1.0
436'	A	8.7	15	26	2.0
	E	1.0	–	–	1.0
412'	A	8.6	15	26	2.0
	E	1.0	–	–	1.0
397' 388' 374'	E	1.0	–	–	1.0

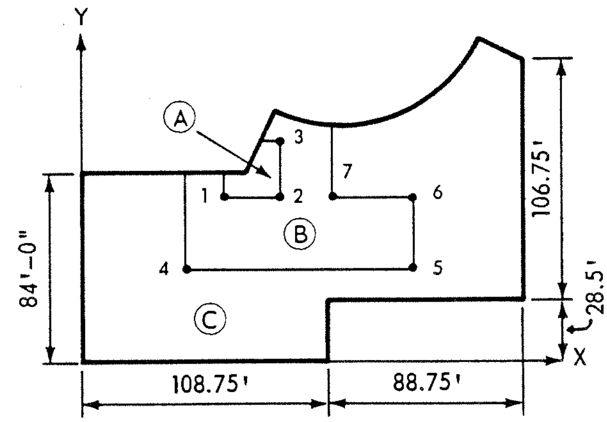
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Gamma Scaling Factor
Auxiliary Building

Figure 3.7-45



INTERMEDIATE BUILDING PLAN



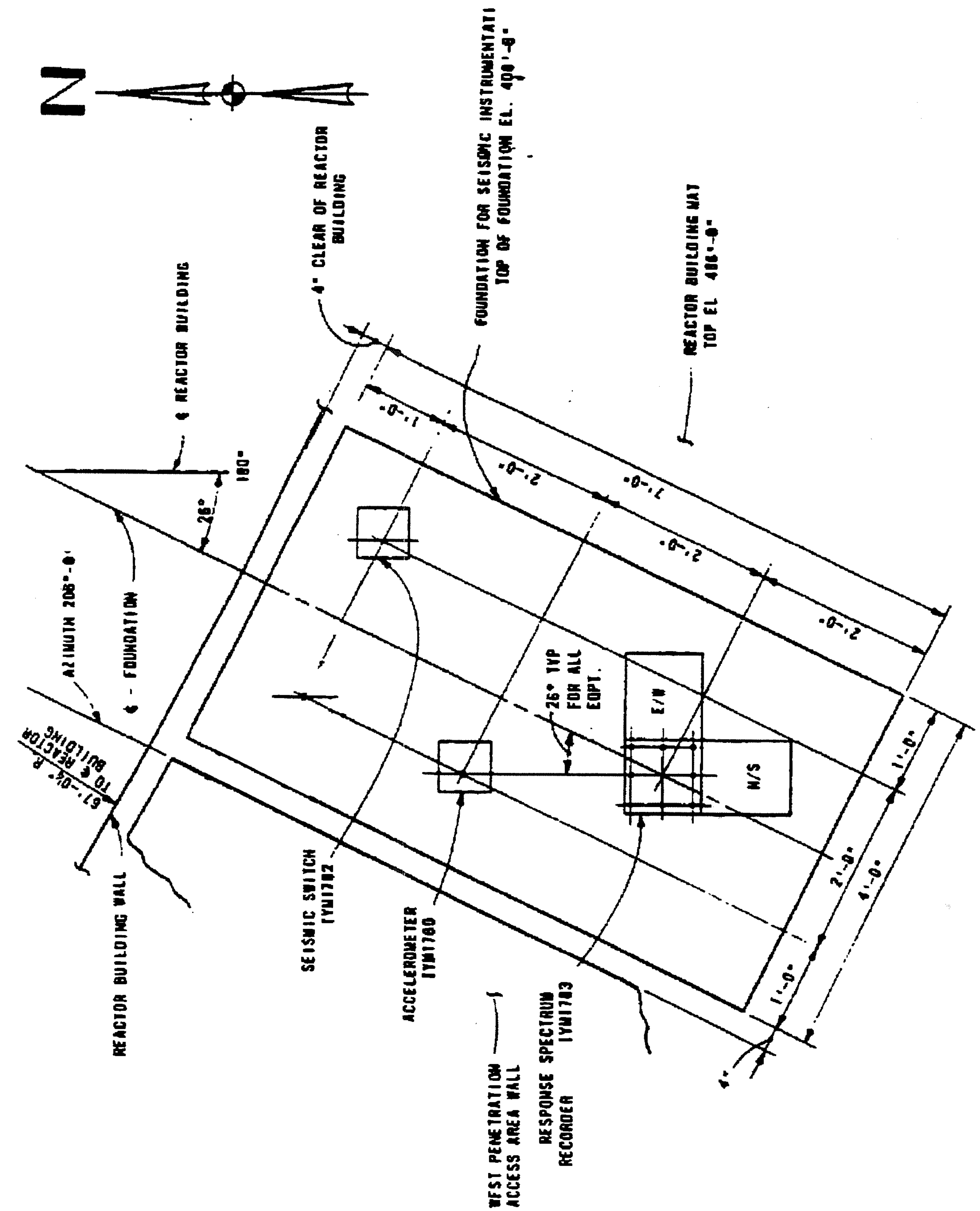
ELEV.	ZONE	Γ_1	f_1	f_2
483	ALL	1.0	–	–
463	A	4.1	8	19
	B	2.2	8	19
	C	1.0	–	–
450	A	3.1	7.5	19
	B	2.1	8	19
	C	1.0	–	–
436	A	2.5	8	19
	B	1.5	8	19
	C	1.0	–	–
412	A	1.4	9	10.5
	B	1.0	–	–
	C	1.0	–	–

ZONE	CORNER COORDINATES (FT.)	
	X	Y
1	64	75
2	89	75
3	89	95
4	41	41.5
5	150	41.5
6	150	70
7	110	70

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Gamma Scaling Factor
Intermediate Building

Figure 3.7-46



AMENDMENT 02-01
MAY 2002

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

SEISMIC INSTRUMENTATION ON
REACTOR BUILDING FOUNDATION MAT
EXCERPT B-809-468
REV. 0
Figure 3.7-47

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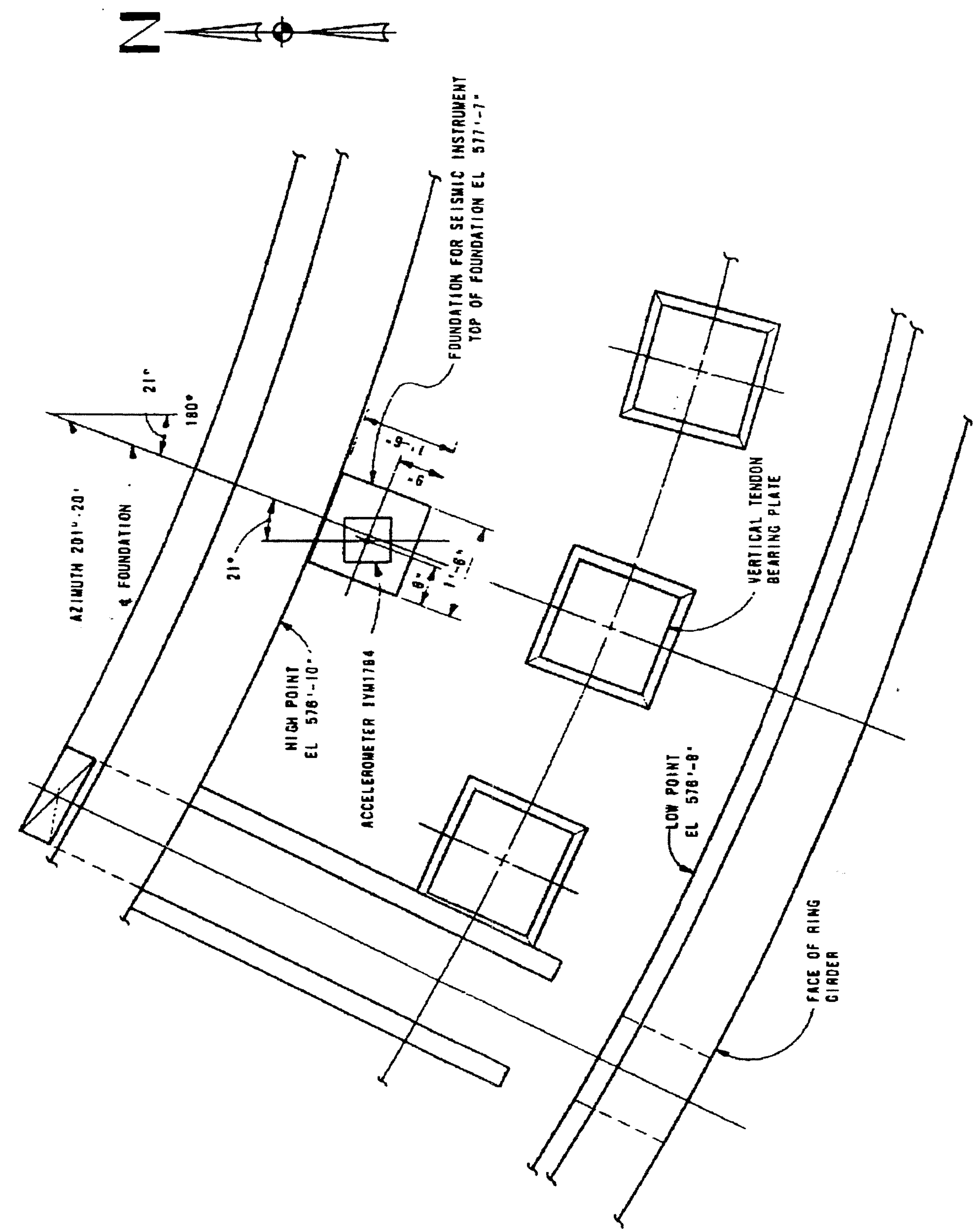
RN 01-068



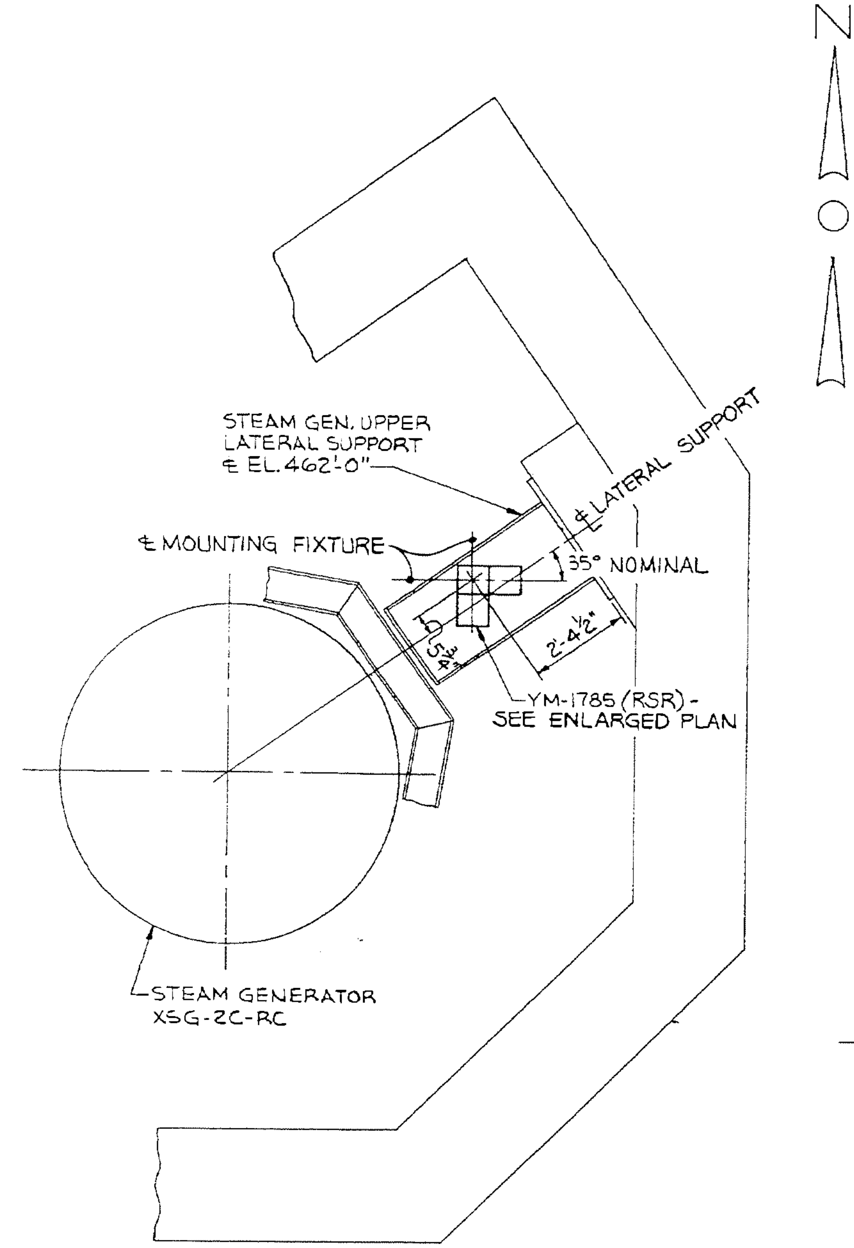
AMENDMENT 02-01
MAY 2002

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

SEISMIC INSTRUMENTATION ON
TOP OF RING GIRDER
EXCERPT B-809-469)
REV. 0
Figure 3.7-48

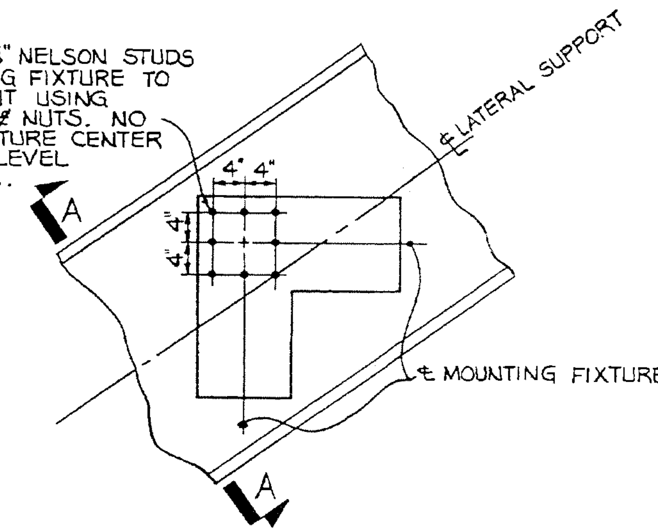
Figure 3.7-49 Deleted by Amendment 96-03

Figure 3.7-50 Deleted by Amendment 96-03



PLAN @ STEAM GENERATOR XSG-2C-RC

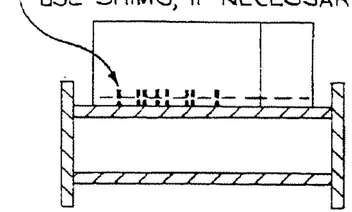
SCALE: $\frac{1}{4}$" = 1'-0"



ENLARGED PLAN

SCALE: $\frac{3}{4}$" = 1'-0"

(8) CS. STUDS - NELSON TYPE $\frac{3}{8}$" x 2" CPL ($\frac{3}{8}$-16 THREAD) TO BE FIELD FURNISHED. USE MOUNTING FIXTURE AS A TEMPLATE TO MARK STUD LOCATIONS ON LATERAL SUPPORT. SURFACE PREPARATION AND STUD INSTALLATION TO BE IN ACCORDANCE W/ STUD MANUFACTURER'S INSTRUCTIONS. USE SHIMS, IF NECESSARY, TO RELIEVE ANY INTERFERENCE ON FIXTURE DUE TO STUD FILLETS.



SECTION A-A

SCALE: $\frac{3}{4}$" = 1'-0"

NOTES:

1. THE 35° MEASUREMENT IS NOMINAL ONLY. ALIGN NORTH/SOUTH AXIS OF FIXTURE TO WITHIN 3° ($\frac{1}{4}$" IN $4\frac{3}{4}$") OF PLANT NORTH/SOUTH LINE. TOLERANCE ON INSTR. ℄ LOCATION IS $\pm\frac{1}{2}$".
2. ORIENT THE YM-1785 MOUNTING FIXTURE TO PERMIT ASSEMBLING COMPONENTS ON IT IN THE POSITION SHOWN. (FOR CLARITY, VERTICAL SENSOR NOT SHOWN) UPPER MOUNTING SURFACE OF FIXTURE TO BE HORIZONTALLY LEVEL WITHIN $\frac{1}{2}$" ($\frac{1}{10}$" IN 12"). REF. ENGDAHL DWGS. 12004 & 120801 (IMS-28-369 & 371)
3. INSTALL PER SP-216-044461-000, CATEGORY I:08.1.3

REFERENCES:

PLANT LAYOUT DWG. E-001-034

Ⓦ DWG. 1457F23, SHT. 10 (IMS-07-153-10) – STM. GEN. SUPPORTS

AMENDMENT 02-01
MAY 2002

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

FSAR Figure 3.7–51

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

I & C DETAIL

STM GEN XSG-2C-RC UPPER LATERAL SUPPORT

EARTHQUAKE INSTRUMENT YM-1785

DESIGN ENGINEERING

V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
1	12/01/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

NO SCALE — DRAWING NUMBER B-809-473 — SHT. NUMBER — REV 1

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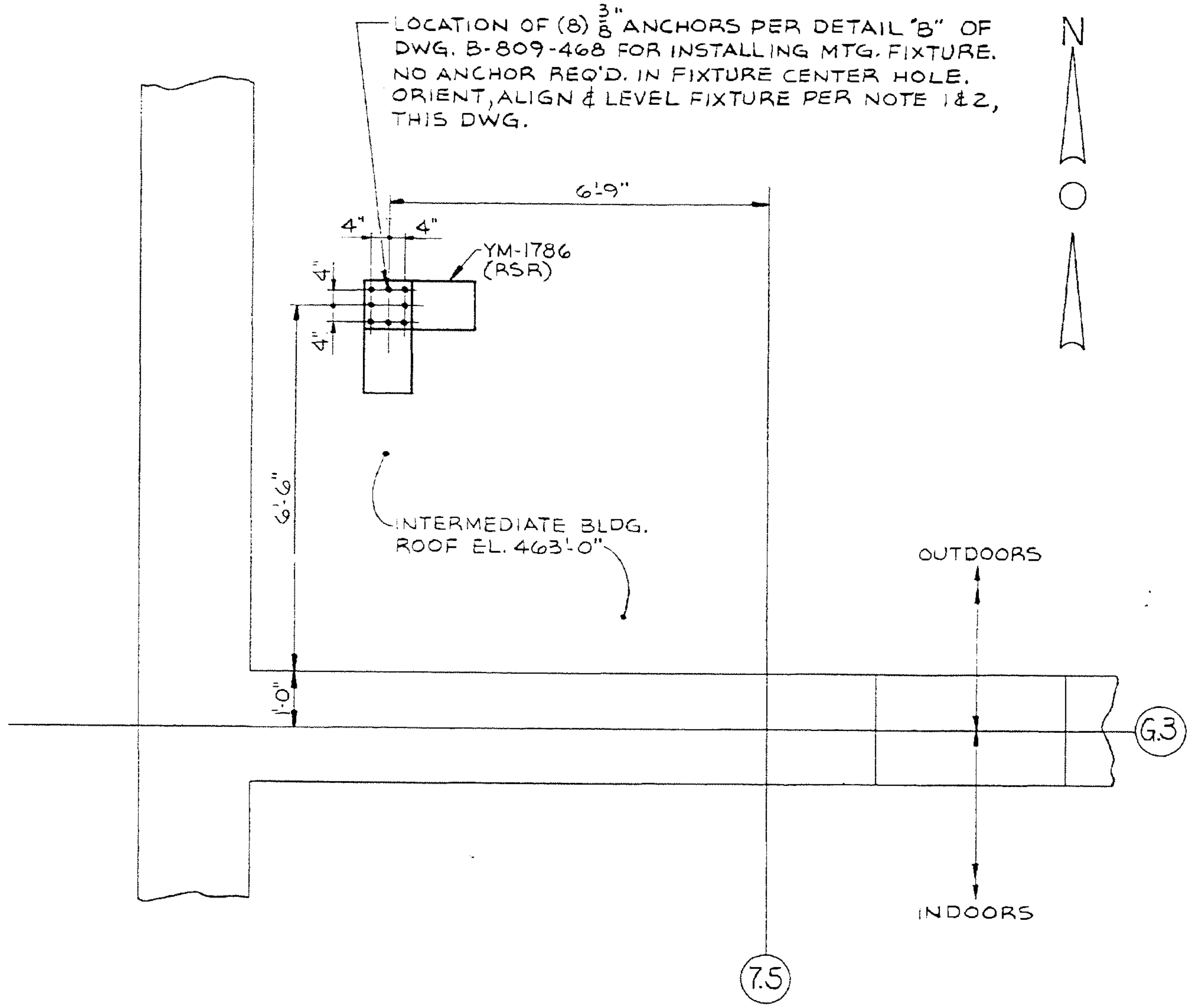
RN 01-055



PLAN

NOTES:

1. TOLERANCE ON INSTR. ℄ LOCATION IS ±2". ALIGN NORTH/SOUTH AXIS OF INSTR. TO WITHIN 3° (1/4" IN 4 3/4") OF PLANT NORTH/SOUTH LINE.
2. ORIENT THE YM-1786 MTG. FIXTURE TO PERMIT ASSEMBLING COMPONENTS ON IT IN THE POSITION SHOWN (FOR CLARITY, VERT. SENSOR NOT SHOWN) UPPER MTG. SURFACE OF FIXTURE TO BE HORIZONTALLY LEVEL WITHIN 1/2° (1/10" IN 12"). REF. ENGDAHL DWGS. 12004 & 120801 (IMS-28-369 & 371)
3. INSTALL PER SP-216-044461-000, CATEGORY 1:08.1.3

REFERENCES:

E-001-032 PLANT LAYOUT.

AMENDMENT 02-01
MAY 2002

FSAR Figure 3.7–52

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
I & C DETAIL
INTERMEDIATE BUILDING EARTHQUAKE INSTR
YM-1786
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

NO SCALE — B-809-470 — DRAWING NUMBER — SHT. NUMBER — REV 1

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
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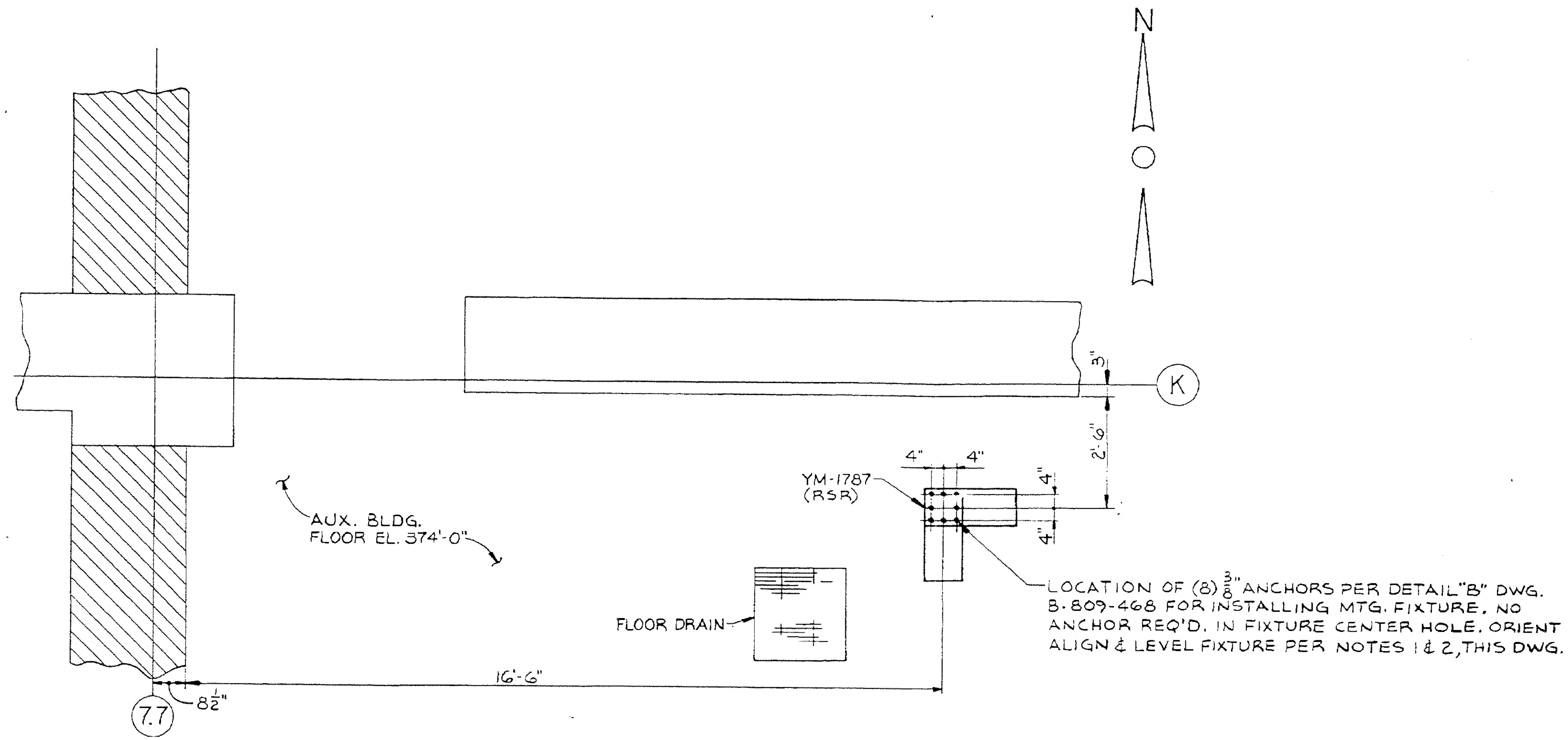
RN 01-095



PLAN

NOTES:

1. TOLERANCE ON INSTR. ℄ LOCATION IS ± 2". ALIGN NORTH/SOUTH AXIS OF INSTR. TO WITHIN 3° (1/4" IN 4 3/4") OF PLANT NORTH/SOUTH LINE.
2. ORIENT THE YM-1787 MTG. FIXTURE TO PERMIT ASSEMBLING COMPONENTS ON IT IN THE POSITION SHOWN (FOR CLARITY VERT. SENSOR NOT SHOWN). UPPER MTG. SURFACE OF FIXTURE TO BE HORIZONTALLY LEVEL WITHIN 1/2° (1/10" IN 12")
   REF. ENGDAHL DWGS. 12004 & 120801 (IMS-28-369 & 371)
3. INSTALL PER SP-216-044461-000, CATEGORY 1:08.1.3

REFERENCES:

E-001-001 PLANT LAYOUT.

AMENDMENT 02-01
MAY 2002

DRAWING LEGIBILITY
CLASS 1
SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
1	10/01/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

FSAR Figure 3.7–53

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION

I & C DETAIL
AUXILIARY BUILDING EARTHQUAKE INSTR
YM-1787

SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE: NO SCALE
DRAWING NUMBER: B-809-471
SHT. NUMBER:
REV: 1

RN 01-095

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3.8 DESIGN OF CATEGORY 1 STRUCTURES

3.8.1 CONCRETE REACTOR BUILDING

3.8.1.1 Description of the Reactor Building

The Reactor Building is a post tensioned, reinforced concrete structure with an integral steel liner. The Reactor Building consists of a cylindrical wall, a shallow dome roof and a foundation mat with a depressed incore instrumentation pit under the reactor vessel. The foundation mat bears on fill concrete which extends to competent rock. At the underside of the Reactor Building foundation mat a tendon access gallery is formed into the top of the fill concrete. A retaining wall, extending approximately 1/4 of the way around the Reactor Building, protects the below grade portions of the Reactor Building wall from the subgrade. Adjacent buildings surround the remaining three quarters of the Reactor Building.

The Reactor Building is lined on the inside face with a steel plate liner which forms a leaktight membrane. The liner is not considered to provide structural reinforcement. Figures 1.2-2 through 1.2-7, 1.2-9, and 1.2-10 show overall plans and sections of the Reactor Building and its relationship to adjacent structures. RN 99-101

3.8.1.1.1 Reactor Building Concrete

3.8.1.1.1.1 Foundation Mat

The Reactor Building foundation consists of a 12 foot thick structural mat, which supports the Reactor Building shell, and the concrete and steel internal structures. The structural mat is supported by fill concrete which extends down to competent rock. Thickness of the fill concrete varies from 27 feet to 52 feet, approximately. See Figure 3.8-1.

The structural mat is circular in plan with a diameter of 154 feet and extends 10 feet beyond the Reactor Building shell cylindrical wall. The mat concrete has a specified minimum compressive strength, f'_c, of 5000 psi and is reinforced with ASTM 615 Grade 60 reinforcing bars. The incore instrumentation pit is located near the center of the mat and is "keyhole" in shape in plan. The base of the pit is 13 feet thick and the walls vary in thickness from 8 feet to 13 feet.

The uppermost fill concrete, extending from elevation 384' to elevation 396', has a diameter of 154 feet. A 6 foot wide by 8 foot high tendon access gallery is formed into the fill concrete directly under the Reactor Building shell cylindrical wall. This gallery provides access to the vertical tendon anchorages. The fill concrete has a specified minimum compressive strength of 3000 psi and is reinforced around the gallery with ASTM 615 Grade 60 reinforcing bars.

The remaining fill concrete, extending from elevation 340' - 365' (varies) to elevation 384', has a diameter of 174 feet. This concrete has a specified minimum compressive strength of 1500 psi and is lightly reinforced on its vertical face with ASTM 615 Grade 60 reinforcing bars.

Refer to Section 3.8.5.1 for additional discussion of the Reactor Building foundation.

3.8.1.1.1.2 Cylindrical Wall and Dome

The Reactor Building shell consists of concrete with a specified minimum compressive strength of 5000 psi and is reinforced with ASTM 615 Grade 60 reinforcing bars. The 4 foot thick cylindrical wall has an inside diameter of 126 feet and a height of 149 feet from the top of the structural mat (elevation 408') to the spring line (elevation 557'). The 3 foot thick shallow dome roof has an inside central radius of 81'-4-1/2" and a transition inside radius of 30 feet. The apex of the outside face of the dome is located at elevation 602'-0-1/4". The shell is post tensioned by ungrouted tendons. Each tendon consists of 170, 1/4 inch diameter, stress relieved wires. The wire is in accordance with ASTM A421. The cylindrical wall contains 115 vertical tendons and 150 hoop tendons. The wall employs the 3 buttress-240 degree hoop tendon concept. The dome contains a total of 99 tendons arranged in a three-way system with 33 tendons per band. Components of the tendon system are identified in Figure 3.8-2.

The Reactor Building shell is provided with a 16 foot ID equipment hatch, a 9 foot OD personnel airlock and a 5 foot OD personnel emergency air-lock. The wall is thickened to 7'-6" in the areas of the equipment hatch and personnel airlock. Reinforcement around the hatch and airlocks is shown in Figures 3.8-3 and 3.8-4. Other major penetrations in the shell include three 56 inch OD sleeves for the main steam lines, three 38 inch OD sleeves for the feedwater lines, and a 24 inch OD sleeve for the fuel transfer tube. Reinforcement around typical main steam line and feedwater line penetrations is shown in Figures 3.8-5 through 3.8-8. The adjacent set of main steam and feedwater line penetrations is reinforced as shown in Figures 3.8-9 and 3.8-10.

The general wall reinforcement layout is shown in Figure 3.8-11. The hoop bars are 1 layer of #14 bars at 12 inch centers on each face for most of the wall height. The vertical bar sizes vary up the wall. On the inside face, the vertical bars (1 layer at 12 inches, center to center) consist of #14 bar Cadwelded to #18 dowels in the lower portion of the wall height. Numbers 11 and 14 bars are used in the upper portion. On the outside face, the vertical bars (1 layer at 12 inches, center to center) consist of #18 bars in the lower portion of the wall and #14 bars in the top portion of the wall. The Cadweld splices are generally arranged in a 3 foot minimum, 4 bar stagger pattern for both hoop and vertical bars. In penetration areas, vertical bars are either deflected around the penetration sleeve; or, in those cases where the deflection would be too large, bars are terminated at each face and a U bar connects the inside face and outside face vertical bars.

Radial shear reinforcement is provided in the lower 20 feet of the wall, in the upper 20 feet of the wall under the spring line, and in the dome above the ring girder. In the bottom 10 feet of the wall, typically #7 ties at 9 inch centers vertically by 12 inch centers horizontally are used. In the next 10 feet, #5 ties varying from 6 inches to 12 inches vertically by 12 inches horizontally are used. Radial shear reinforcement in the upper 20 feet of the wall consists of #7, #6, and #5 ties. The vertical spacing of these bars varies between 8, 12, and 18 inches. Depending upon design requirements, the bars are spaced horizontally so as to enclose either each set of inside-outside face vertical bars or every other set of inside-outside face vertical bars. In the areas surrounding the polar crane brackets, #5 ties at 6 inch centers vertically by 12 inch centers horizontally are used. For those areas of the Reactor Building wall that are not provided with radial shear reinforcement, #5 radial tension ties are provided, having an average spacing horizontally of 3 feet 6 inches and within 1 inch clear, above and below, each hoop tendon. RN 99-101

Reinforcement details for the dome are shown in Figure 3.8-11a. Meridional reinforcement consists of #11 bars on each face. The spacing of these bars varies as they extend radially toward the apex to Cadweld with a square reinforcement grid of #10 bars. The hoop reinforcement consists of #14 and #11 bars spaced as shown. Radial shear reinforcement consisting of #9 and #7 bars is provided in the dome for a distance of 16 feet above the ring girder. Radial tension ties consisting of #5 bars exist over the remainder of the dome. These ties provide an area density of 0.25 square inches per square foot of dome surface.

Three (3) buttresses, spaced 120° apart and extending full height from the top of the foundation mat to the underside of the ring girder, serve as anchorage points for the hoop tendons. The buttresses are nominally 12'-10" wide and 6'-6" thick. The inside face of each buttress is flush with the inside face of the cylindrical wall.

Buttresses are reinforced in both the meridional (vertical) and hoop directions. Inside face vertical reinforcement is #14 bar at 12 inches and extends full height. Outside face vertical reinforcement consists of #18 bar at 12 inches for the lower half and #14 bar at 12 inches for the upper half. In the hoop direction, inside face reinforcement is identical to that for the cylindrical wall. Outside face hoop reinforcement consists of #11 bar at 11-1/2 inches. In addition to the reinforcement described, shear and bursting reinforcement is also provided around the tendon anchorage. Number 9 U-shaped bars are used for shear reinforcement and spirals (2 sets behind each tendon base plate) for bursting reinforcement. The spirals have an outside diameter of 15 inches and 21 inches each and are made from #5 bar at 3 inch pitch. Typical buttress details are shown in Figure 3.8-12.

3.8.1.1.1.3 Ring Girder

The junction of the dome and cylinder wall is thickened to provide space for the anchorage of the dome and vertical prestressing tendons. This thickened junction is referred to as the ring girder and serves as a transfer block for the prestressing forces. The ring girder is 9 feet wide at its top and approximately 19 feet in overall depth. Its outside vertical face is flush with the outside face of the buttresses on the cylindrical wall and is notched at various angles to accept seating of the dome tendon bearing plates. To minimize the depth of these side notches, a fully rotated system of dome bearing plates is employed.

The ring girder is reinforced in both meridional and hoop directions as shown by Figure 3.8-13. Inside face meridional reinforcement is #18 bar at 12 inches center to center. Shear reinforcement consists of #9 bars, shaped, oriented, and spaced as shown by Figure 3.8-13. A set of bursting reinforcement consisting of rings cut from 20 inch OD ASTM A-106 pipe with 1 inch nominal thickness is provided behind each tendon anchorage.

3.8.1.1.2 Steel Liner and Penetrations

3.8.1.1.2.1 Liner Plate

The liner plate is designed to function as a leaktight membrane sealing the entire Reactor Building for any postulated conditions to be encountered throughout the operating life of the plant. In general the liner plate is made from ASME SA516, Grade 60, carbon steel (see Figures 3.8-1 and 3.8-14).

The cylindrical wall liner plate is made from carbon steel. At its base, in the haunch area, a truncated conical transition section tapers inward to accommodate the thickened concrete of the cylindrical shell. This truncated conical section extends approximately 10 feet above the floor plate. The cylindrical portion of the liner is 139 feet high and has an inside diameter of 126 feet.

The top of the cylindrical portion of the liner is closed by a dome with a minor radius of 30 feet and a major radius of 81'-4-1/2". The ratio of Reactor Building ID to dome height is 3 to 1.

The interior vertical height from the top of the floor liner plate to the interior apex of the dome is 190'-11-3/4". The floor, cylindrical, and dome portions of the liner plate are 1/4 inch thick.

The bottom of the liner consists of flat, floor liner plates, welded to anchors which are embedded in the mat concrete (see Figure 3.8-14).

The liner plate extends downward into the foundation mat to line the incore instrumentation pit, 2 residual heat removal sumps, 2 Reactor Building spray sumps, 1 Reactor Building sump, and 1 incore instrumentation pit sump. The incore instrumentation pit is lined with stainless steel plates and carbon steel plates. Carbon steel plate, 1/4 inch thick, is used to line the incore instrumentation pit walls. Stainless steel, 1/2 inch thick, is used to line the bottom and 1/4 inch thick stainless steel is used to line the walls of the incore instrumentation tunnel. The residual heat removal sump, Reactor Building spray sumps, and Reactor Building sump floors and sidewalls are lined with 1/4 inch thick stainless steel plate (see Figure 3.8-1). The carbon steel is ASME SA516, Grade 60, material. the stainless steel is ASME SA240, type 304 material.

Weld seams which are inaccessible for inspection following construction of the plant are covered with test channels. These test channels are welded continuously along the edge of their flanges to the liner to provide leaktight enclosures for pneumatic leak testing of the liner welds (see Figure 3.8-14). The test channels are zoned into test areas by plates welded to the ends of sections of the channels. Small diameter tubing is connected to the test channels and extends to accessible areas for leak testing after concrete placement.

3.8.1.1.2.2 Penetrations

All Reactor Building penetrations are anchored to the concrete Reactor Building wall or foundation mat so that loads are transferred from the penetrations to the concrete. All penetrations satisfy the requirements of 10 CFR 50, Appendix J. Penetrations are classified into the following groups:

1. Piping Penetrations

   All piping penetrations consist of a sleeve around the outside of the piping. The piping is joined to the sleeve inside the Reactor Building by an attachment plate (see Figure 3.8-15). Outside the Reactor Building, piping is attached to the sleeve by an attachment plate or by a bellows assembly. The attachments at both the inside and outside ends of the sleeve create an interspace between the sleeve and piping. This interspace is tested to ensure that leak rate requirements are satisfied. The bellows and sleeve are designed to withstand the containment design pressure and temperature. RN 01-113

   Each sleeve is integrally welded to a surrounding, thickened liner reinforcing plate. The weld between the sleeve and reinforcing plate is covered by a circular test channel which is used for testing the leak tightness of the weld.

   Anchorage attachments, provided on the outside of the embedded portion of each sleeve, transfer the piping loads into the Reactor Building wall (see Figure 3.8-15).

All hot piping penetrations are insulated as required to prevent the temperature in the concrete adjacent to the sleeve from exceeding 200°F during normal operation. Insulation is provided around the attachment plate and portions of the exposed sleeve, as well as around the process pipe, resulting in a smooth transition in temperature gradients from the pipe through the attachment plate and into the sleeve. This prevents unacceptable thermal stress concentrations in the sleeve, as well as in the hot process pipe.

The containment boundary and weld locations, typical for all piping penetrations, in the main steam line penetration are shown by Figure 3.8-15. RN 01-113

The main steam penetration materials and inspection requirements are listed in Table 3.8-0.

Containment boundary welds to process pipe are accessible for inservice inspection per the ASME Code, Section XI. Welds are examined to the requirements of ASME Section XI as required by 10CFR50.55a. 00-01

The welds attaching the stiffeners and cooling fins to the penetration assembly and the weld between the bellows assembly and the process pipe are examined either by the liquid penetrant or magnetic particle method.

2. Mechanical System Penetrations

A fuel transfer tube penetrates the Reactor Building connecting the refueling canal in the Reactor Building and the fuel transfer canal in the Fuel Handling Building. This penetration consists of a stainless steel pipe installed inside a sleeve (see Figure 3.8-16). This pipe functions as the transfer tube connecting the refueling canals in the Reactor and Fuel Handling Buildings.

An attachment plate is welded to the transfer tube and the end of the sleeve. A protective cover is placed over the attachment plate and sleeve. The cover is welded to the transfer tube and Reactor Building liner to provide an interspace for local leak rate testing of the welds which are part of the containment system.

Details of the residual heat removal sump and penetration are shown in Figure 3.8-17.

3. Electrical System Penetrations

Sleeves through the Reactor Building wall are provided to accommodate electrical and instrumentation cables which pass through the wall (see Figure 3.8-15). The sleeves are welded to the Reactor Building inner reinforcing plates. A leak test channel is placed over the sleeve to liner weld to provide leak test capability.

The electrical leads are installed in the penetration assemblies which are bolted to the electrical penetration sleeve. Each assembly has provisions for leak testing to comply with 10 CFR 50, Appendix J.

4. Spare Penetrations

Spare penetrations consist of sleeves passing through the reactor wall with the liner reinforced around the sleeve (see Figure 3.8-15). Both ends of the sleeve are sealed with butt welded pipe caps to provide an interspace within the sleeve for leak testing. A test channel assembly over the sleeve to liner plate weld is provided to allow leak testing.

Spare penetrations #602-18, #600-12, and #505-18 are fitted with bolted removable blank flanges and Flexitallic type or equal gaskets in place of butt welded end caps. This penetration is for temporary services into Reactor Building as required during shut-down. Leakage rate testing is performed on the penetration as discussed in Chapter 6.2.6, Containment Leakage Testing. 98-01

3.8.1.1.2.3 Access Openings

1. Equipment Hatch

The equipment hatch has an inside diameter of 16 feet as shown in Figure 3.8-18. The hatch is equipped with a hatch cover, mounted on the inside of the Reactor Building.

The hatch cover is double O-ring gasketed and has a leakage test tap between the O-rings. The enclosed space between the O-rings is capable of being leak tested as discussed in Chapter 6.2.6, Containment Leakage Testing. A monorail and hoist are provided inside the Reactor Building for removal of the hatch cover. The hatch cover is suspended from 2 points. 98-01

Two (2) legs with rollers are provided on the hatch cover. The rollers move within a channel anchored to the floor slab to prevent pivoting as the door is moved into the open position.

A concrete shield located outside the Reactor Building acts as a missile and biological shield.

2. Personnel Airlocks

Two (2) personnel airlocks are provided for access to the Reactor Building (see Figures 3.8-19 through 3.8-20b). Each airlock has 2 doors which are sealed with double O-rings. Independent connections are provided between each double O-ring door seal and the handwheel shaft seals for leak testing in accordance with the requirements of Technical Specifications Section 3/4.6.1.3. For leak testing, the personnel airlocks can be independently pressurized to the Reactor Building design pressure.

During normal operation the outer and inner doors are interlocked so that neither door can be opened unless the other door is closed and latched. The inner door can be operated manually from inside the Reactor Building and from inside the airlock. The outer door can be operated manually from inside the airlock and from the outer side of the outer bulkhead.

In addition to the capabilities described above, the emergency personnel airlock has a remote operational capability for both doors from the opposite side.

The mechanical interlocking system can be deactivated to allow both doors to be left open when the plant is shut down. Doors are mechanically latched and are swung by manual means. Each door is furnished with a pressure equalizing valve which is operated by the latching mechanism.

Balanced magnetic switches provide input on airlock position to the Integrated Fire and Security System. Each airlock has position indication on a CRT in the Control Room as follows:

a. Alarm, if either of the 2 airlock doors is not shut.

b. Trouble, if the magnetic switch circuit has a ground or a break.

Additionally, an audible alarm is given upon a change in status.

3.8.1.1.2.4 Structural Attachments

1. Polar Crane Brackets

The polar crane brackets are located on an angular spacing of 11°-15' around the circumference of the Reactor Building with a top elevation of 548'-4-3/4". These brackets provide vertical and lateral support for the polar crane steel runway girders which are described in Section 3.8.3.

The brackets extend through the Reactor Building liner and are anchored into the concrete wall. The liner plate in the area of the brackets is 1/2 inch thick.

Forces and moments created by the crane wheel loads are carried into the brackets and thence to the concrete. See Figure 3.8-21 for details of the crane runway girders and brackets.

The crane and trolley bridge are prevented from being dislodged during an earthquake by lugs on the trolley and on the bridge trucks which extend below the supporting girders. These details are shown in Figure 3.8-22.

2. Pad Overlay Plates

   Small diameter circular overlay plates are fillet welded to the liner plate to support piping, ducts, conduit, and electric cable trays. Studs or angle anchors are provided on the liner behind the attachment plates to transfer loads on the pads into the concrete shell.

3. Equipment Supports

   The Reactor Building cooling units, charcoal filter units, and associated platforms are supported on the inside of the cylindrical shell of the Reactor Building by steel framing members which terminate at brackets located on the liner. The brackets consist of plate sections welded to the liner or to the thicker liner insert plates. The load is transferred into the concrete shell by the embedded portion of the bracket located on the outside of the liner plate. The embedded portions of the bracket consist of welded plate sections and headed concrete anchors.

3.8.1.2 Applicable Codes, Standards, and Specifications

3.8.1.2.1 General

Structural design, fabrication, construction, testing, and inservice inspection of the Reactor Building conform to the following documents unless noted otherwise herein:

1. Southern Standard Building Code, 1969 Edition.

2. American Concrete Institute, "Building Code Requirements for Reinforced Concrete," ACI 318-71.

3. American Concrete Institute, "Criteria for Reinforced Concrete Nuclear Power Containment Structures," ACI Committee 349 report, ACI Journal, January, 1972.

4. American Concrete Institute, "Specification for Structural Concrete for Buildings," ACI 301-72, revised 1975.

5. American Institute of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," February 12, 1969.

6. American Institute of Steel Construction, "Code of Standard Practice for Steel Buildings and Bridges," July 1, 1970.

7. American Welding Society, "Structural Welding Code," AWS D1.1-72.

8. Nuclear Regulatory Commission, Regulatory Guides, as discussed in Appendix 3A and listed below:

    a. Regulatory Guide 1.12, "Instrumentation for Earthquakes."

    b. Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments."

    c. Regulatory Guide 1.29, "Seismic Design Classification."

    d. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

    e. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants."

9. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria 2, 4, 16, 50, 51, 52, and 53.

3.8.1.2.2 Concrete

The following codes, standards, and specifications are used to establish the properties of concrete and concrete coatings and to control concrete placement and concrete coating applications:

1. American Concrete Institute

    a. "Specification for Structural Concrete for Buildings," ACI 301-72, revised 1975.

    b. "Building Code Requirements for Reinforced Concrete," ACI 318-71.

    c. "Mass Concrete for Dams and Other Massive Structures," ACI Title No. 67-17, a report by ACI Committee 207.

2. American Society for Testing and Materials

    a. "Standard Method of Test for surface Moisture in Fine Aggregate," C70-73.

    b. "Standard Method of Test for Specific Gravity and Absorption of Coarse Aggregate," C127-73.

    c. "Standard Method of Test for Specific Gravity and Absorption of Fine Aggregate," C128-73.

d. "Standard Method of Test for Length Change of Hardened Cement Mortar and Concrete," C157-75.

e. "Standard Method of Test for Thermal Conductivity of Materials by Means of the Guarded Hot Plate," C177-71.

f. "Manual of Coating Work for Nuclear Power Plant Primary Containment Facilities," D01.43.

g. "Standard Method of Test for Total Moisture Content of Aggregately by Drying," C566-67.

h. "Standard Method of Test for Chloride Ion in Water and Waste Water," D512-67.

i. "Standard Method of Test for Sulfate Ion in Water and Waste Water," D516-68.

j. ASTM C109-08, Standard Test Method for Compressive Strength of Hydraulic Cement Mortars (Using 2-in Cube Specimens).

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k. ASTM C1107-11, Standard Specification for Packaged Dry Hydraulic-Cement Grout (Nonshrink).

3. U.S. Army Corps of Engineers

a. "Method of Test for Coefficient of Linear Expansion of Concrete," CRD-C39-55.

b. "Method of Test for Flow of Grout Mixtures (Flow Cone Method)," CRD-C79-58.

c. "Method of Test for Flat and Elongated Particles in Coarse Aggregate," CRD-C119-53.

d. "Methods of Sampling and Testing Expansive Grouts," CRD-C589-70.

e. CRD C621-93, Standard Specification for Packaged Dry Hydraulic-Cement Grout (Nonshrink). RN 14-016

4. National Association of Corrosion Engineers, "Coatings and Linings for Immersion Service," TPC No. 2.

5. Concrete Plant Manufacturers Bureau, "Concrete Plant Standard," Revision 5, March, 1973.

6. Truck Mixer Manufacturers Bureau, "Truck Mixer and Agitator Standards," Revision 9, November, 1971.

7. American National Standards Institute

   a. "Protective Coatings (Paints) for the Nuclear Industry," ANSI N5.12-74.

   b. "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," (Proposed) ANSI N101.4-72.

8. Nuclear Regulatory Commission, Regulatory Guide 1.55, "Concrete placement in Category 1 Structures" as discussed in Appendix 3A.

3.8.1.2.3 Reinforcing Steel

The following codes and specifications are used to establish the properties of and to control the fabrication and placement of reinforcing steel:

1. American Society for Testing and Materials

   a. "Standard Specification for Deformed Billet Steel Bars for concrete Reinforcement," ASTM A615-72.

   b. "Standard Methods and Definitions for Mechanical Testing of Steel Products," ASTM A370-72.

2. American Concrete Institute

   a. "Specification for Structural Concrete for Buildings," ACI 301-72, Revision 1975. Bars of Category 1 Concrete Structures."

   b. "Building Code Requirements for Reinforced Concrete," ACI 318-71.

3. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Division 2, 1975.

4. Nuclear Regulatory Commission, Regulatory Guides, as discussed in Appendix 3A and listed below:

   a. Regulatory Guide 1.10, "Mechanical (Cadweld) Splices in Reinforcing Bars of Category 1 Concrete Structures."

   b. Regulatory Guide 1.15, "Testing of Reinforcing Bars for Category 1 Concrete Structures."

3.8.1.2.4 Post Tensioning System

The following codes, standards and specifications are used to establish the properties of and control the placement of the post tensioning system.

1. American Concrete Institute, "Criteria for Reinforced Concrete Nuclear Power Containment Structures," ACI Committee 349 report, ACI Journal, January, 1972.

2. Prestressed Concrete Institute, "Tentative Specification for Post Tensioning Materials," PCI Journal, January-February, 1971.

3. American Society for Testing and Materials

    a. "Specification for Uncoated Stress Relieved Wire for Prestressed Concrete," ASTM A421-65.

    b. "Specification for Conducting Drop Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," ASTM E208-69.

    c. "Specification for Welded and Seamless Steel Pipe," ASTM A53-72a.

    d. "Specification for Zinc (Hot-Galvanized) Coatings on Products Fabricated from Rolled, Pressed and Forged Steel Shapes, Plates, Bars and Strip," ASTM A123-73.

    e. "General Requirements, Specification for Steel Sheet, Zinc-Coated Galvanized by the Hot-Dip Process," ASTM A525-71.

4. American Welding Society, "Structural Welding Code," AWS D1. 1-72.

5. American Society of Mechanical Engineers, <u>Boiler and Pressure Vessel Code</u>,

6. Nuclear Regulatory Commission, Regulatory Guides, as discussed in Appendix 3A and listed below:

    a. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures."

    b. Regulatory Guide 1.103, "Post-Tensioned Prestressed Systems for Concrete Reactor Vessels and Containments."

7. American Welding Society, "Code for Welding and Building Construction," AWS D1.0-69.

3.8.1.2.5 Steel Liner, Penetrations, and Attachments

The following codes, standards, and specifications are used to establish the properties of and control the placement of the steel liner, penetrations, and attachments:

1. American Society of Mechanical Engineers, <u>Boiler and Pressure Vessel Code</u>.

    a. Section II, "Material Specification Parts A and C"

    b. Section V, "Non-Destructive Examination"

    c. Section IX, "Welding Qualifications"

2. American Society of Mechanical Engineers, <u>Boiler and Pressure Vessel Code</u>, Section III, "Nuclear Power Plant Components," Division 1.

    a. Including Winter 1972 Addenda for personnel airlocks and equipment hatch.

    b. Including Summer 1972 Addenda for liner, penetration sleeves, overlay pads, brackets, and insert plates.

    c. Including Winter 1975 Addenda for piping penetration assemblies.

    d. Including 1974 Edition for electrical penetration assemblies.

3. American Society of Mechanical Engineers, <u>Boiler and Pressure Vessel Code</u>, Section III, "Nuclear Power Plant Components," Division 2, 1975.

4. American Welding Society, "Structural Welding Code," AWS D1.1-72, Section 4, Part VI, "Stud Welding."

5. American National Standards Institute

    a. "Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors," ANSI N6.2-1965; approved by American Nuclear Society and American Society of Mechanical Engineers.

    b. "Protective Coatings (Paints) for the Nuclear Industry," ANSI N5.12-74.

    c. "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment facilities," (Proposed) ANSI N101.2-74.

    d. "Quality Assurance for Protective Coatings to Nuclear facilities," (Proposed) ANSI N101.4-72.

6. American Nuclear Society, "Proposed Standard for Leak Rate Testing of Containment Structures for Nuclear Reactors," ANS 7.60, Appendix A.

7. Institute of Electrical and Electronics Engineers, "Standards for Electrical Penetration Assemblies in Containment Structures for Nuclear Generating Stations," IEEE-317-1972.

8. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors."

9. Steel Structures Painting Council, "Near White Metal," SSPC-SP10.

10. Nuclear Regulatory Commission, Regulatory Guides, as discussed in Appendix 3A and listed below:

    a. Regulatory Guide 1.19, "Non-Destructive Examination of Primary Containment Liner Welds."

    b. Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants."

3.8.1.3 Loads and Load Combinations

3.8.1.3.1 Concrete Reactor Building

3.8.1.3.1.1 Load Definitions

Fundamental design loads for the Reactor Building are defined as follows:

1. D - Dead load of structure, hydrostatic loads, and permanent equipment loads.

2. L - Conventional floor and roof live loads and movable equipment loads, including piping, cables, etc. Other loads which vary intensity, such as soil pressures and polar crane loads are all included.

3. F - Post tensioning forces for any given loading condition.

4. P - Design accident pressure load. This pressure is based upon peak calculated pressure with appropriate margin provided for calculational uncertainties as described in Section 6.2.1.

5. R - Local force or pressure on structure or penetration caused by rupture of any one pipe.

6. T_o - Thermal effects under normal operating conditions, including liner expansion, equipment and pipe reactions, and temperature gradients.

7. T_a - Added thermal effects (over and above operating thermal effects) which may occur during a design basis accident and which correspond to a factored design accident pressure (i.e., 1.25P or 1.5P).

8. Earthquake

    a. E - Operating basis earthquake (OBE).

    b. E' - Safe shutdown earthquake (SSE). Vertical and horizontal earthquake accelerations are assumed to act simultaneously if appropriate for determination of maximum stress. Dynamic response characteristics of the structure, supporting rock, and rock pressures are considered in determination of forces due to earthquakes (E and E').

9. Z - External pressure resulting from internal pressure drop due to spray cooling.

10. Z' - Internal pressure resulting from atmospheric pressure drop due to a tornado.

11. W - Design wind load.

12. W' - Tornado wind load and tornado missiles.

3.8.1.3.1.2 Design Loads

1. Dead Load, Hydrostatic Loads, and Permanent Equipment Loads - D

    a. Dead load consists of the weight of the complete structure. A reinforced concrete density of 150 lb/ft^3 is used.

    b. Hydrostatic loads due to groundwater at elevation 423' are used in the design of the structural foundation mat. This results in a height of 27 feet of water acting at the bottom of the mat.

    c. Permanent equipment and piping loads on the Reactor Building shell under service loads (S) and factored loads (U) as follows:

        (1) Pipe penetration - normal reaction (D and T_o) - pipe break and earthquakes (R, E, and E').

        (2) Pipe supports - normal reactions (D and T_o) - pipe break and earthquakes (R, E, and E').

        (3) Polar crane - normal reactions (D and T_o) - earthquakes (E and E').

        (4) Air handling units - normal reactions (D and T_o) - earthquakes (E and E').

2. Snow, Ice, and Polar Crane Loads - L

The Reactor Building is designed to withstand snow or ice loads of 20 lb/ft^2. Polar crane lifted loads consist of 360 tons during construction and 157.5 tons during normal plant shutdown. RN 02-024

3. Post Tensioning Forces - F

Forces on the Reactor Building at 40 years, after all post tensioning losses have occurred, are as follows:

a. The 115 vertical tendons produce a vertical stress resultant of 335 kips/ft in the cylindrical wall.

b. The 150 hoop tendons produce:

(1) An external pressure of 5768 lb/ft^2 corresponding to 375 kip/ft hoop stress resultant in the 10 foot haunch

(2) An external pressure of 11,535 lb/ft^2 corresponding to 750 kip/ft hoop stress resultant in the remaining wall height.

c. The 99 dome tendons (33 per layer) produce an external pressure of approximately 13,835 lb/ft^2 (varies over the dome).

Initial post tensioning forces for the vertical, hoop, and dome tendons are 1.14, 1.18, and 1.15 times the above values, respectively.

The foregoing minimum required level of applied external prestress on the containment was calculated based upon the effects of prestress alone. Subsequent evaluations showed this external prestress could be reduced by including the effects of increased tendon forces due to expansion of the containment shell under accident pressure conditions.

4. Design Accident Pressure Load - P

The Reactor Building design pressure, P, is 57 psig.

5. Operating Temperature - T_o

Temperature profiles, shown by Figure 3.8-23, are based upon the following boundary conditions:

a. Reactor Building Internal Temperature

(1) Maximum, 120°F

(2) Minimum, 50°F

b. Average Monthly Ambient Temperature Outside the Reactor Building

(1) Highest, 83.6°F

(2) Lowest, 35.3°F

These values are the highest and lowest average monthly ambient temperatures ever recorded in the vicinity of the site by the U.S. Weather Bureau.

In addition to the above temperature conditions, the design includes a maximum thermal gradient based upon a 20°F outside temperature. The winter design dry-bulb temperature is expected to be 20°F or higher for 99% of the hours for the months of December, January, and February. The use of 20°F as an equilibrium outside air temperature is a conservative approach in that the Reactor Building wall thermal gradient will be maximized. The effects of the various temperature gradients and the thermal force exerted on the concrete by the Reactor Building liner are considered in the Reactor Building design. The effect of adjacent building temperatures (104°F, maximum; 65°F, minimum) is also considered in the design. 00-01 00-01

6. Added Thermal Effects - T_a

The accident temperature gradients throughout the concrete wall of the Reactor Building are shown by Figure 3.8-24. The effects from these gradients are considered in the design of the Reactor Building. Thermal forces exerted on the concrete by the Reactor Building liner are also considered.

7. Earthquake Loads - E and E'

Site seismology and response spectra are described in Section 2.5. Seismic design of the Reactor Building is based upon the response to ground accelerations described in Section 3.7.

8. External Pressure - Z

The Reactor Building is designed for an internal vacuum, Z, of 3.5 psig which would result from inadvertent actuation of the Reactor Building Spray System.

9. Internal Pressure - Z'

The Reactor Building is designed to withstand an internal pressure, Z' , equal to 3 psig occurring at a rate of 2 psi per second as a result of an atmospheric pressure drop due to a tornado.

10. Wind Load - W

The Reactor Building is designed to withstand wind pressures corresponding to the design wind velocity discussed in Section 3.3.1.

11. Tornado Wind Load - W'

The Reactor Building is designed to withstand tornado wind pressures and tornado generated missiles as discussed in Section 3.3.2.

12. Structural Acceptance Test

Following completion of construction, the Reactor Building is subjected to an internal pressure test at 65.6 psig to verify structural integrity.

3.8.1.3.1.3 Load Combinations

The Reactor Building is designed to withstand various combinations of the loads defined in Section 3.8.1.3.1.1 and discussed in Section 3.8.1.3.1.2. These load combinations are presented in Table 3.8-1.

3.8.1.3.2 Reactor Building Liner and Attachments

3.8.1.3.2.1 Reactor Building Liner and Anchor Design Loads

Load definitions for loads used to design the Reactor Building liner and anchors are in accordance with the ASME Code, Section III, Division 2, Article CC-3220.

Specific loadings applicable to design of the Reactor Building liner and anchors are delineated by the load combinations presented in Table 3.8-2. The following loads apply to the design of the Reactor Building liner and anchors in addition to those discussed in Section 3.8.1.3.1.2.

1. Dead Load - D

   The Reactor Building liner is designed for the dead load of the liner, all bracing, scaffolds, and supports attached to the liner which are used during liner erection and construction.

2. Live Load - L

   The Reactor Building liner is designed for a 100 lb/ft^2 live load on the dome liner plates to account for loads on the dome liner due to reinforcing steel, tendon conduit, personnel, and equipment during construction of the concrete shell. A reduced live load is used during erection of the polar crane.

   The cylindrical portion of the liner also serves as the inside formwork for fresh concrete loads which occur during concrete placement. Live loads due to fresh concrete are calculated in accordance with ACI Publication SP-4, "Formwork for Concrete." A pour rate of 2 feet per hour and a concrete temperature of 60°F are conservatively assumed.

   To support the fresh concrete, the dome liner is supported at close intervals by tying to the rigid system of dome tendon conduits as described in Section 3.8.1.6.1.3. The design of the dome liner to withstand fresh concrete loads takes this support system into account.

3. Wind Load - W

   During construction the Reactor Building liner is designed for wind loads resulting from a wind velocity of 70 mph at 30 feet above grade (based upon a 10 year recurrence interval) in accordance with ASCE Paper 3269 [1].

4. Operating Temperatures - T_o

   An operating temperature range of 50°F to 120°F inside the Reactor Building is considered in the design of the Reactor Building liner. An operating temperature range of 12°F to 92°F outside the Reactor Building is considered in the specification of materials and material properties. Local thermal effects at hot penetrations are also considered. However, these effects do not contribute to the controlling Reactor Building liner design loads since concrete temperature is limited to 200°F, maximum, at such local spots.

   Thermal effects at the time of the structural acceptance test are considered. These temperatures inside the Reactor Building are controlled during such testing.

Transient thermal effects resulting from temperature changes within the Reactor Building during normal operation as well as during startup and shutdown are also considered.

5. Normal Pipe Reactions - R_o

   Load combinations applicable to Reactor Building liner and anchor design, as well as design of insert plates are presented in Table 3.8-2. These load combinations comply with the applicable portions of the ASME Code, Section III, Division 2, Table CC-3230-1.

   Fatigue effects are considered based upon the criteria stated in the ASME Code, Section III, Division 1, Class MC. Operating conditions involving cyclic application of loads and thermal conditions are considered.

6. Design Accident Pressure Load - P_a

   The Reactor Building design pressure, P_a, is 57 psig.

7. Structural Acceptance Test Pressure Load - P_t

   The Reactor Building structural acceptance test pressure load, P_t, is 65.6 psig.

8. Differential Pressure Load - P_v

   The Reactor Building differential pressure load, P_v, is 3.5 psig below atmospheric pressure and 3 psig above atmospheric pressure.

9. Earthquake Loads - E_o and E_{ss}

   Site seismology and response spectra are described in Section 2.5. Seismic design of the Reactor Building is based upon the response to ground accelerations described in Section 3.7.

10. Tornado Wind Load - W_t

   The Reactor Building is designed to withstand tornado wind pressures and tornado missiles as discussed in Section 3.3.2.

3.8.1.3.2.2 Bracket, Attachment, and Pad Overlay Plate Design Loads

Load Definitions for loads used to design brackets, attachments pad overlay plates, and assorted embedments are in accordance with the ASME Code, Section III, Division 2, Article CC-3220.

Specific loadings applicable to the design of these components are delineated by the load combinations presented in Table 3.8-3. The following loads apply to the design of brackets, attachments, and overlay plates in addition to those discussed in Section 3.8.1.3.2.1:

1. Dead Load - D

   Dead load consists of the weight of specific equipment and its associated support structure.

2. Live Load - L

   Polar crane bracket live load consists of a lifted load of 360 tons during construction or 157.5 tons during normal plant shutdown. Impact loads, as a percentage of wheel load, are also considered as live loads. Access platform live load is 150 lb/ft^2. RN 02-024

3. Earthquake Loads - E_o and E_{ss}

   Earthquake loads are obtained from system analyses of ducts and piping or from the seismic analysis performed by the vendors of equipment such as the polar crane.

These load combinations are in compliance with the ASME Code, Section III, Division 2, Article CC-3750.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 Concrete Reactor Building

For structural analysis purposes, the Reactor Building geometry is idealized in various structural models depending upon the particular structural component to be analyzed. These components are: the structural foundation mat, the general shell, the thickened openings, the ring girder, and the buttresses.

3.8.1.4.1.1 Structural Foundation Mat

Two (2) finite element models are used depending upon whether the load combination creates symmetrical or unsymmetrical uplift of the mat. The models are shown by Figures 3.8-25 and 3.8-26, respectively. The analytical method used is the ELAD (S014) computer program described in Section 3.8.4.4.

3.8.1.4.1.1.1 Symmetrical Uplift

Load combination 4a in Table 3.8-1 causes a symmetrical uplift condition. However, the mat reinforcement is generally controlled by the unsymmetrical uplift condition described in Section 3.8.1.4.1.1.2. The forces and tractions resulting from load combination 4a are applied over the top of the mat and on the portion of the Reactor Building wall included in the model. The forces for the wall at elevation 438' are obtained from an analysis of the general shell which is described in Section 3.8.4.1.2. Hydrostatic pressures due to ground water at elevation 423' are applied along the bottom of the mat at the mat-fill concrete interface. A thin row of elements exists at this interface. These elements are assigned a zero shear modulus, G_{Rz}, to conservatively calculate the radial stress resultant, N_R, which is produced by the mat radial expansion, caused by the 1.5P term of the load combination.

Accommodation of the symmetrical uplift of the mat from the fill concrete is accomplished by removing interface elements which exhibit tensile vertical stresses, σ_z, under the total forces resulting from the load combination 4a. Removal of these elements is an iterative process. The results of the first iteration indicate that tensile σ_z stresses occur in the outermost rings of interface elements and compressive σ_z stresses exist in the remaining elements which are located radially inside these rings. For the second iteration, the tensile stressed elements are removed. The results of the first iteration indicate that the outermost rings of elements which were previously compressive are now tensile. These tensile elements are removed for the next iteration. This process is continued until all the interface elements exhibit compressive σ_z stresses. In this condition, the computed symmetrical uplift occurs from radius 38 feet to 77 feet and the mat is supported by the fill concrete in bearing from radius 0 to 38 feet.

The displacement boundary conditions for the model are applied to the boundaries of the fill concrete as shown by Figure 3.8-25. These boundaries are of sufficient extent so that the specified displacement boundary conditions do not affect the mat stresses.

3.8.1.4.1.1.2 Unsymmetrical Uplift

Unsymmetrical forces on the Reactor Building can result from load combinations 3c, 4b, 5a, and 5b of Table 3.8-1. However, only those load combinations including the seismic forces E or E' are sufficient to produce uplift of the mat. These load combinations produce the largest reinforcement requirements. The extent of uplift is calculated from static overturning analyses which include the hydrostatic uplift force due to ground water at elevation 423'. Structural analyses of the mat are performed for load combinations 4b and 5a.

For analytical purposes each load combination is separated into symmetrical and unsymmetrical parts:

Load Combination		Symmetrical	Unsymmetrical
1.	4b	$(D + F + T_o + T_a + 1.25P)$	$+ (D + 1.25E)$
2.	5a	$(D + F + T_o + T_a + P)$	$+ (D + E')$

These terms are defined in Section 3.8.1.3, except that D represents the reduced effective dead load of the structure that results from the vertical acceleration corresponding to 1.25E or E' in addition to hydrostatic uplift pressures. The terms L and R are not applicable for mat design and are, therefore, not included.

The structural analysis of the symmetrical parts is performed using the ELAD model presented by Figure 3.8-25 in accordance with the iteration procedure previously described. The ELAD Model presented by Figure 3.8-26 is used to analyze the mat for the unsymmetrical parts. The internal stress resultants in the mat obtained from the 2 ELAD models are combined, except where larger rebar requirements result from exclusion of the LOCA. When this occurs, rebar requirements are controlled by D + E'. The stress resultants are obtained by integrating the output stresses across the mat section thickness.

The loads on the mat for the unsymmetrical parts of the load combinations consist of applied loads and reactor loads. The applied loads consist of the hydrostatic pressures plus the dead weight and seismic forces from the internal concrete structures, major equipment, and the Reactor Building shell. The hydrostatic pressures are applied to the mat at elevation 396' and elevation 374'. The seismic forces are determined from the lumped-mass model, time-history analysis of the Reactor Building using the dynamic analysis procedure described in Section 3.7.2.1. From this analysis, the total vertical and horizontal forces and overturning moment are obtained separately for the internal concrete and the shell. Their combined effect is largest at a time when the response accelerations of the shell are largest. The seismic forces at this time for the internal concrete are distributed to the primary and secondary shield walls and steel column ring according to their respective stiffnesses. Tractions and nodal point forces are applied to the top of the mat at their respective radii such that the vertical and horizontal forces and overturning moment of each are represented. The forces and overturning moment for the shell are obtained from the seismic analysis at elevation 438'. These are represented as tractions and nodal point forces on the shell at the corresponding elevation in the ELAD model. The seismic acceleration of the shell from elevation 438' to elevation 408' and of the 12 foot mat are described through the use of body forces on the model. All applied loads can be represented by the 0th and 1st Fourier harmonics for input into ELAD.

The applied loads and the reaction loads are simultaneously applied to the model shown by Figure 3.8-26. The model is in static equilibrium under these loads and displacement boundary conditions are not required. For the ELAD analysis, however, zero displacement boundary conditions must be specified to remove rigid body displacements. For the load equilibrated mat, these zero displacements must be specified at nodes such that internal stresses are not effected. This is accomplished by incorporating a low stiffness column, located at the mat centerline and extending 30 feet below the incore base, into the model. The zero displacement boundary conditions are specified at lower nodes in the column. A review of the results at the interface of the column and incore base indicates that the stresses are practically zero. Hence, incorporation of the column in the model allows for removal of rigid body displacements without influencing mat stiffness or stresses.

The reaction loads are normal and shear tractions which occur at the mat-fill concrete interface to equilibrate the applied loads. These tractions are linearly distributed over the nonuplifted part of the mat. They are zero at the start of uplift and maximum at the mat perimeter. These tractions are accurately described by a 10 term Fourier Series for input to the ELAD analysis.

3.8.1.4.1.1.3 Mat Design

The mat is designed to resist the stress resultants obtained from the analysis previously described. The main reinforcement (radial and hoop) is designed based upon the linear stress and strain distributions discussed in Section 3.8.1.5.1.1. The resulting reinforced concrete section satisfies the criteria of Section 3.8.1.5.1.

The temperature distributions shown by Figures 3.8-23 and 3.8-24 were considered in the mat design. This information was used to establish linear thermal gradients in the incore base, walls, and mat. The design procedure for these gradients is described in Section 3.8.1.4.1.2. The linear gradients were included in the design if a larger reinforcement requirement resulted. This occurred for the bottom reinforcement in the incore base and mat and for the outside reinforcement in the incore walls. In areas of the mat where the liner is exposed (incore base and part of incore walls) the active force of the liner on the concrete due to T_o and T_a was included. The allowance for a cracking reduction of the thermal stresses is described in Section 3.8.1.4.1.2.
The design for transverse shear is in accordance with Chapter 11 of ACI 318-71 [2], except that Section 11.2.2 of ACI 318-71 is not applicable. The calculated nominal shear stress, V_u, is obtained from the transverse shear stress resultant, V_u, using Equation (11-3) of ACI 318-71. Wherever V_u exceeds the permissible shear stress, v_c, of ACI 318-71, Section 11.4, shear reinforcement in the form of closed ties is sized using Equation (11-13) of ACI 318-71. Wherever V_u is less than v_c but greater than 1/2 v_c, minimum closed tie shear reinforcement is provided in accordance with Equation (11-1) of ACI 318-71.

Transverse shear transfer across construction joints is provided by the keys described in Section 3.8.1.6.1.3.

The portion of the mat spanning over the incore pit was considered as a deep beam using References [3] and [4] and the shear provisions of Section 11.9 of ACI 318-71. Because the seismic response of the internal concrete produced a torsion on this portion of the mat, the provisions of Sections 11.7 and 11.8 of ACI 318-71 were also applied.

To ensure the ability of the mat to develop its design capacity, diagonal bars extending from the incore walls through the mat were provided based upon information reported in Reference [5].

Concrete cover of reinforcement is not less than that specified in Table 3.8-4.

The effects of shrinkage, creep and material property variation are discussed in Section 3.8.1.4.1.2.

3.8.1.4.1.2 General Shell

Design of the Reactor Building wall and dome is based upon stress resultants obtained from 2 computer programs:

1. KALNINS Static Program (S043)

2. KALNINS Dynamic Program (S032)

These programs are discussed in Section 3.8.4.4.

The model for the analyses is shown by Figure 3.8-27. The design of additional reinforcement around the unthickened penetrations, including main steam and feedwater, is based upon stress concentration factors applied to the stresses from the KALNINS analyses. The additional effects of pipe rupture are included, as discussed in subsequent paragraphs of this section.

KALNINS static computer program is used to obtain stresses and stress resultants for the individual fundamental loads D, F, P, T_o, T_a, Z, Z', W, and W', defined in Section 3.8.1.3. The unsymmetrical fundamental loads W and W' are discussed in Section 3.3. Their effective pressures, shown by Figures 3.3-2 and 3.3-3, are represented in the computer program using a 10 term Fourier Series. Design procedures for tornado missile loads are described in Section 3.5.3.1.

KALNINS dynamic computer program is used to obtain the stresses and stress resultants in the shell due to E and E'. In these analyses, the 2% structural damping curves of Figures 3.7-1 and 3.7-2 are used.

Stress results from this analysis are increased by 20% based upon a comparison of shell overturning moments from KALNINS dynamic computer program and the time history seismic analysis discussed in Section 3.7.2.1.

The stresses and stress resultants for the load combinations were obtained by applying the appropriate load factors to the fundamental load results combined according to the load combination listed in Table 3.8-1. Since the shell model is fixed at the base, the results near the base were modified to account for the effect of the mat uplift under load combinations 4a, 4b, and 5a. For the remainder of the load combinations, the fixed base model yields conservative results. In cases where the combined stresses are entirely compressive through the shell thickness, those stresses are compared with the allowable criteria in Section 3.8.1.5. Where tensile concrete stresses exist, uncracked stress resultants are used in a cracked section design of reinforcement according to the criteria specified in Section 3.8.1.5.1.1.

The resulting concrete section satisfies the criteria stated in Section 3.8.1.5.1. Such reinforcement satisfies the crack control requirements of Paragraph 2.5.4 of Reference [6]. The splice requirements of Paragraph 2.5.3 of Reference [6] are also satisfied through use of Cadweld splices for all main vertical and hoop reinforcement in the wall. Cadweld splices are also used for dome reinforcement which is required to resist tensile forces while subjected to perpendicular tensile stresses and for the #14 hoop bars.

Concrete cover of reinforcement is not less than that specified in Table 3.8-4.

3.8.1.4.1.2.1 Prestress

The level of prestress at 40 years, accounting for all losses, is established using the minimum requirements of load combination 6, Table 3.8-1. These losses are identified as follows:

1. Friction

   The friction losses used in the design are based upon the following wobble, K, and curvature, μ, coefficients:

   a. For hoop and dome tendons

      (1) K = 0.0003

      (2) μ = 0.16

   b. For vertical tendons:

      (1) K = 0.0003

      (2) μ = 0.0

2. Elastic Shortening

   The prestress loss due to elastic shortening is based upon 1/2 of the elastic shortening due to initial prestress forces.

3. Creep of Concrete

   Prestress loss due to creep is based upon the following calculated 40 year concrete creep strains:

   a. Vertical, 210 micro in/in.

   b. Hoop, 390 micro in/in.

   c. Dome, 330 micro in/in.

   These values are used in an interaction prestress loss analysis [7].

4. Shrinkage

   Prestress loss due to shrinkage is based upon a conservative 40 year value of 100 micro in/in.

5. Steel Relaxation

   Prestress loss due to tendon wire relaxation is based upon a 40 year stress relaxation value of 8.5%. This is used in an interaction prestress loss analysis [7].

   The results of the third tendon surveillance indicated that loss in prestress force was occurring at a faster rate than originally predicted . Wire stress relaxation testing was performed at Lehigh University Fritz Engineering Laboratory to verify the 8.5% stress relaxation value assumed for the original design. Testing was performed on samples of wire from actual tendons in the containment. Samples were tested at several temperatures including the temperature representative of the average temperature at the location of the tendons within the containment shell. Based on the results, a wire stress relaxation value of 12.8% for 40 years is used to predict prestress losses for the fourth and subsequent surveillances.

The level of initial prestress, which is necessary to satisfy the 40 year requirements, does not exceed the criteria stated in Section 3.8.1.5.1.2.

The design procedures for tendon anchorages are presented in Section 3.8.1.4.1.4.

3.8.1.4.1.2.2 Temperatures T_o and T_a

The T_o temperature distributions, shown in Figure 3.8-23 and the T_a temperature distributions shown in Figure 3.8-24 for winter and summer startup and shutdown conditions, are included in the design. At sections in the shell where stresses due to these actual distributions do not cause concrete tension when combined with stresses for the remaining part of the load combination, the uncracked stresses are compared with the criteria stated in Section 3.8.1.5.1.

In cases where the combination of these stresses produces a concrete tensile stress, a cracking reduction of the thermal stress only is permitted. If the temperature distribution under consideration is nonlinear, an effective linear distribution is obtained. The effective linear gradient is such that it has the same average temperature and produces the same moment about the center of gravity of the uncracked section as does the actual nonlinear temperature distribution. The steady-state temperature distributions are linear. The linear gradients are then used in the design as follows:

1. The uncracked thermal stress resultants produced by the average temperature across the section plus the liner expansion are combined with the uncracked stress resultants for the remaining part of the load combination. The combined stress resultants are N and M.

2. The balance of the temperature distribution is a linear gradient, ΔT, which has a zero average temperature. The additional effects of this gradient are determined using the following procedure:

    a. Analyze the section for N plus M based upon a cracked section and a linear strain.

    b. Determine the location of the neutral axis, the slope, S_L, of the strain diagram, and the resulting stresses as determined by N and M above.

    $$S_L = \frac{f_c}{kd\,E_c}$$

    where:

    f_c = Maximum concrete stress

    kd = The neutral axis

    E_c = Modulus of elasticity of concrete

c. Determine the slope, S_T, of the strain diagram due to the applied temperature gradient based upon an uncracked section:

$$S_T = \frac{\alpha \Delta T}{t}$$

where:

α = Coefficient of thermal expansion

ΔT = Temperature differential across the wall

t = Thickness of the wall

d. Equate the slope of the final strain diagram, S_c, to S_L plus S_T. Strain due to the applied load plus thermal gradient is then:

$$S_c = \frac{f_c}{kd\,E_c} + \frac{\alpha \Delta T}{t}$$

e. Knowing the slope of the final strain diagram, S_c, and equating the sum of the forces of the final stress diagram to the applied axial force, the location of the neutral axis for the combined loading of N plus M plus temperature gradient can be determined as follows:

(1) Net axial load, N, excluding thermal gradient:

$N = A_s f_s - 1/2\ f_c\ kd$

(2) Net axial load, N, plus thermal gradient:

$N = A_s f_{st} - 1/2\ f_{ct} kd_t$ RN 01-113

(3) By equating the above 2 expressions for N and substituting:

$f_{ct} = S_c kd_t\ E_c$, and

$f_{st} = S_c\ (d - kd_t)\ n\ E_c$, and solve for kd_t

where:

f_s = Tensile stress in reinforcing steel due to N and M

f_{st} = Tensile stress in reinforcing steel due to N and M plus thermal gradient

f_{ct} = Compressive stress in concrete due to N and M plus thermal load.

kd_t = Distance from compressive face of concrete to the neutral axis.

f. Calculate the resulting stresses by back substitution into the equations for f_{ct} and f_{st}, above.

The design includes the effects of the following:

1. Adjacent building temperatures.

2. Average temperature differences between the shell and ring girder during startup and shutdown.

The active force of the liner on the concrete under T_a produces a tensile axial stress resultant which is limited by the compressive yield capacity of the liner plate material. For purposes of reinforcement design, a liner compressive yield strength of 45 ksi is used based upon the average strength from mill test reports. The tensile axial stress resultant in the concrete is determined from the difference between 45 ksi and the liner stress under normal operating conditions. This tensile force is combined directly with stress resultants on the section without regard to time differences between P and T_a. These time differences are considered, however, in determining the effects of the nonlinear accident temperature distribution on concrete stress.

3.8.1.4.1.2.3 Radial Shear

Radial shear acts perpendicular to the wall of the Reactor Building.

Reinforcement for radial shear is in accordance with Chapter 11 of ACI 318-71, except that Section 11.2.2 of ACI 318-71 is not applicable.

The calculated nominal shear stress v_u, is obtained from the radial shear stress resultant, v_u, using Equation (11-3) of ACI 318-71. Wherever v_u exceeds the permissible shear stress, v_c, shear reinforcement is provided using Equation (11-13) of ACI 318-71. The permissible shear stress, v_c, is described in Section 3.8.1.5.1. RN 01-113

Radial shear transfer across construction joints is provided by the keys described in Section 3.8.1.6.1.3.

3.8.1.4.1.2.4 Tangential Shear

Tangential (membrane) shear acts tangential to the wall of the Reactor Building. The provisions of Section 11.16 of ACI 318-71, are not applicable to design for this shear.

Load combinations 4a and 5b of Table 3.8-1 produce tangential shear along with membrane tensile stress resultants which are resisted by vertical and hoop reinforcement. The design criteria for the reinforcement is that the steel ratio, p_v, provided for tangential shear, both vertically and horizontally, is at least RN 01-113

$$p_v = \frac{A_v}{ta} = \frac{v_u}{\phi f_y},$$ RN 01-113

and that this steel ratio is in addition to that required to resist membrane tensile forces where:

A_v = Area of reinforcement within distance a

t = Thickness of the wall

a = spacing measured perpendicular to bars

v_u = Nominal tangential shear stress RN 01-113

ϕ = 0.85

f_y = Minimum specified yield strength of reinforcement

The value of the nominal shear stress, v_u, does not exceed $8\sqrt{f'_c}$. RN 01-113

Tangential shear transfer across construction joints is provided by the roughening of these joints as described in Section 3.8.1.6.1.3.

3.8.1.4.1.2.5 Radial Tension Under Post Tensioning

To prevent lamination under prestress, radial ties are provided to resist the forces caused by the curvature of the hoop and dome tendons.

In the dome, the ties enclose both the top surface reinforcing and extend past the lower layer of prestressing tendons, and conform to the larger of Criterion (a) or Criterion (b). For the cylindrical shell, the ties enclose the reinforcing of both surfaces and conform to Criterion (a).

1. Criterion (a)

   The area of the ties shall be sufficient to resist, at 0.5 f_y, the radial components of all prestress forces multiplied by the distance from the outside surface to the centroid of the tendon system and divided by the total shell thickness.

2. Criterion (b)

The area of the ties shall be sufficient to resist, at 0.5 f_y, the radial components of the prestress forces in the innermost layer of tendons.

The maximum tie spacing does not exceed the thickness of the shell.

3.8.1.4.1.2.6 Penetrations

The following discussion pertains to the reinforced concrete design around all penetrations except the equipment hatch and personnel air locks which are discussed in Section 3.8.1.4.1.3.

The stress concentration increase of shell stresses and the resulting reinforcement requirements are determined for all penetrations using the information in References [8] and [9]. The additional concrete stress resultants due to pipe rupture effects from the main steam and feedwater lines are determined using References [10], [11], and [12] in conjunction with results of an ELAD analysis of the main steam penetration.

The main steam and feedwater pipe rupture loads, R, on the Reactor Building include reactions from the penetration sleeve and jet impingement. Impact of the ruptured pipe is prevented by the restraints described in Section 3.6. The forces on the sleeve are calculated axial loads, shears, and moments from the pipe. The moment on the sleeve is the ultimate moment the pipe is capable of developing. The axial load and shear on the sleeve are based upon thrust forces which have been increased to account for dynamic effects.

The reinforcement around penetrations consists of the general shell vertical and hoop bars supplemented by additional vertical, horizontal, and diagonal bars. This system of reinforcement is designed to resist the forces in the concrete as discussed above. The general shell bars are deflected around the penetration sleeves except where prohibited by the penetration size. In such cases, the shell bars terminate in a 90 degree bend that is continuous with the opposite face bar. The deflected shell bars are provided with tie-back bars which are designed to resist calculated forces caused by the change of direction of the deflected bars. In the areas of the main steam and feedwater penetrations it was necessary to increase the size of the general shell reinforcement in addition to providing the supplemental bars.

Pipe rupture forces from the main steam line are large enough to produce a punching shear requirement for radial shear ties. These ties are based on Equation (11-13) of ACI 318-71 with $v_c = 2\sqrt{f'_c}$. The nominal shear stress, v_u, is less than the $6\sqrt{f'_c}$ allowable stress appearing in Section 3.8.1.5.1. RN 01-113

Pipe rupture forces for the feedwater line are not large enough to cause v_u to exceed the $4\sqrt{f'_c}$ allowable stress given in Section 3.8.1.5.1. However, shear ties are provided to resist the axial thrust on the penetration sleeve.

3.8.1.4.1.2.7 Shrinkage and Creep

The effect of concrete shrinkage on the design stress resultants is insignificant due to the large volume-to-surface ratios of the Reactor Building elements. Rapid drying of the Reactor Building exterior was minimized by use of the 14 day water curing process described in Section 3.8.1.6.1.3. However, the main reinforcement is located sufficiently close to the surface to adequately control shrinkage cracking. Any stresses which may have been produced in the reinforcement due to surface drying tend to be compressive and thus, do not reduce the capacity of the reinforcement since the reinforcement is provided to resist tensile forces.

For purposes of calculating prestress losses, a conservative value for 40 year shrinkage strain of 100 micro in/in is used as discussed previously. This is approximately 10 times greater than would be calculated using the information in Reference [13].

The effect of concrete creep under dead and prestress loads is included in the prestress loss calculations and in the structural analysis. Concrete creep with time sheds compressive stress from the concrete to the liner. This results in less concrete compressive stress available to resist the design accident and extreme environmental load combinations and increases reinforcement requirements. The increased liner stress is accompanied by increased strains.

Both of these effects are considered in the design through the use of an effective Young's Modulus, E'_c. This modulus is obtained from the creep curves in Reference [13]. From these curves, specific creep strains, sc, are obtained, considering both concrete age at loading and duration of loading. The effective modulus is expressed in terms of specific creep as:

$$E_c^{'} = \frac{E_c}{1 + sc\,E_c}$$

where:

E_c = Instantaneous Young's Modulus = 4 x 10^6 psi

sc = Specific creep (micro in/in/psi)

3.8.1.4.1.2.8 Material Property Variations

Material quality control, described in Section 3.8.1.6, assures that minimum strength material requirements are achieved. Overstrength of the liner and mechanical property variation of the concrete materials are considered in the design.

As discussed previously, the liner, during a temperature increase, exerts a tensile force on the concrete. For design purposes, the magnitude of this force is assumed to be limited by the compressive yield strength of the liner. The liner material identified in Section 3.8.1.6 has a minimum specified yield strength of 32 ksi. However, for the design of reinforcement, a liner compressive yield strength of 45 ksi was used. This is based upon the average yield strength of the same grade of liner material as obtained from mill tests.

The effect of variations of Young's Modulus between concrete pours was considered. Testing laboratory data on the design mix and the field compression test data were reviewed. Based upon this review, it was concluded that the variations of the mechanical properties were sufficiently close to the design values to have an inconsequential effect on structural analysis results.

3.8.1.4.1.3 Access Openings

The design of the thickened Reactor Building wall in the vicinity of the equipment hatch and personnel airlock is based upon the stress resultants obtained from a finite element analysis of the local areas. A doubly curved, quadrilateral shell element is used. The boundary conditions for the local model, shown by Figure 3.8-28, are obtained from the general shell analysis discussed in Section 3.8.1.4.1.2. The local model is arranged so that the appropriate horizontal and vertical symmetries are used with the applied load and displacement boundary conditions. The local model extends into the surrounding normal 4 foot wall to the extent that effects of the thickened portions have dissipated and the general shell boundary conditions are valid. In the preliminary analysis phase, a course mesh finite element model of the overall cylindrical shell, including buttresses and thickened openings, is used to provide the appropriate dimensional extent of the finer mesh localized model.

Analyses are performed for the individual fundamental loads. Stress resultants are then combined according to the load combinations determined to be critical in the general shell analysis. Internal pressure load on the access hatch door is applied to the perimeter of the opening as a distributed shear load. Hoop prestressing forces are applied as an equivalent external pressure. The added local effects of the tendon draping pattern are considered. The treatment of thermal loads is the same as that descried for the general shell.

In the design, the tangential shear stresses due to earthquake loading are based upon the course mesh model results. The model consists of the entire cylindrical shell with the thickened access areas, fixed at the base. Displacements at the top, obtained from the general shell analysis, are introduced as boundary conditions. The thickened access opening is oriented 90° to the earthquake direction.

Other aspects of the design procedure are the same as described for the general shell.

3.8.1.4.1.4 Buttresses

The buttresses are designed to resist local tendon anchorage forces, as well as the meridional and hoop stress resultants induced by the load combinations in Table 3.8-1.

The concrete bearing stresses behind the anchorage plates are based upon initial prestress at lockoff.

Local bursting stresses due to tendon anchorage forces are conservatively calculated assuming an isolated equivalent block and using the three independent analysis procedures described in References [14], [15], and[16]. The results obtained from the 3 methods are evaluated to arrive at the amount of bursting reinforcement required.

Spalling stresses are considered in accordance with Guyon's recommendation and further investigated on the basis of a "bending away" mechanism as described in Reference [17].

Requirements for ties connecting the buttress to the shell wall are obtained by hypothesizing a vertical failure plane at the Interface of the wall and buttress. An unbalanced hoop tendon force is assumed to exist which is resisted by the ties according to the shear friction provisions of Section 11.15 of ACI 318-71.

The design of the buttress reinforcement is based upon the results of the course mesh finite element model described in Section 3.8.1.4.1.3 and the results of a plane stress finite element analysis using Wilson's Program (SO66), described in Section 3.8.4.4. The discontinuity effect of the mat in the vertical direction is considered using classical shell theory and accounting for the buttress thickness.

The model for the plane stress analysis is shown by Figure 3.8-29. The boundaries of the model, at the buttress centerline, and midway between buttresses, are lines of symmetry with respect to Displacements. Roller Displacement boundary conditions are applied at these boundaries. The internal accident pressure and hoop tendon forces are applied as normal tractions. The action of the liner under a temperature increase is represented as an equivalent internal pressure.

Other aspects of the design procedures are the same as described for the general shell.

3.8.1.4.1.5 Ring Girder

The design of the ring girder is based upon the stress results obtained from the computer programs ELAD, KALNINS Static, and KALNINS Dynamic. All 3 programs are used specifically for the analysis of axisymmetric structures subjected to both axisymmetric and nonaxisymmetric load cases. These computer programs are described in Section 3.8.4.4.

The ELAD model, shown by Figure 3.8-30, represents the entire Reactor Building above elevation 497'. This boundary is far enough below the ring girder (60 feet) so that only membrane stresses occur. Hence, the specification of "roller" Displacement boundary conditions is applicable for the fundamental loads used on the model. The ELAD model includes the 2 layers of anchorage pockets for the dome prestress tendons. The model is used to obtain stress results for the fundamental loads: D, L, F, P, T_o, T_a, Z, and Z' .

These stresses are integrated across the concrete section to obtain the stress resultants used in the reinforcement design. They are combined with the general shell stress resultants for W, W', E, and E' to construct the load combinations in Table 3.8-1.

The prestress fundamental load, F, is sub-divided into 3 separate load cases:

1. Vertical Prestress, F_v

2. Dome prestress, F_d

3. Hoop prestress, F_h

The dome prestress load, F_d, is a nonaxisymmetric load in that, within 60 degree dome segments, the anchorage forces may vary in magnitude depending upon whether 1 or 2 tendon layers are anchored. Because of this it was necessary to consider stress results at 2 azimuths in the ring girder. These are the locations where the anchorage forces are maximum and minimum.

The remaining part of the dome prestress load is represented by meridionally varying normal and meridional shear tractions applied to the appropriate dome elements. These occur simultaneously with nonuniform anchorage forces at the lower and upper anchorage pockets which are represented by a 2 term Fourier Series.

The treatment of operating and accident temperature loads T_o and T_a is discussed in Section 3.8.1.4.1.2.2. Differential heating and cooling between the ring girder and the wall and dome, as measured by the average temperatures in these sections, are examined. The resulting stresses are included in reinforcement design for the ring girder.

At normal operating conditions, the effective linear thermal gradients in the ring girder, the calculation basis of which has been previously discussed, are less than the steady-state gradients. Hence, for such conditions the steady-state values are used in the reinforcement design. At accident temperature conditions, higher equivalent linear thermal gradients are obtained. These values are used in the design.

The design for tendon anchorage forces is discussed in Section 3.8.1.4.1.4.

Other aspects of the design procedures are the same as described for the general shell.

3.8.1.4.2 Liner Plate

Reactor Building liner plate design and analytical procedures are in accordance with the ASME Code, Section III, Division 2, Article CC-3600.

3.8.1.4.2.1 Cylinder and Dome Liner Plate

Since the concrete Reactor Building shell is much stiffer than the liner plate, strains in the liner are compatible with the strains in the Adjacent shell concrete to which the liner is attached at close intervals. The strains in the liner are obtained from the analyses of the Reactor Building shell described in Section 3.8.1.4.1.

Strains in the liner due to localized conditions and transients associated with design basis accident temperature are computed separately based upon the equation

$\varepsilon = \alpha\Delta T$

where:

ε = Strain

α = Coefficient of thermal expansion for the liner

ΔT = Change in temperature of the liner

These strains are combined with the strains due to the other loadings in the load combinations, including design basis accident or localized thermal loadings due to postulated pipe rupture.

3.8.1.4.2.2 Floor Liner Plate

The thickened knuckle plate at the floor to sidewall transition of the liner is analyzed using KALNINS Static Program (see Section 3.8.4.4). Boundary conditions are taken from the analyses of the cylindrical wall and the mat analyses described in Sections 3.8.1.4.1 and 3.8.5.4.

The floor liner and the portions of the liner extending into the incore instrumentation pit are analyzed with the surrounding mat and interior concrete structures using the ELAD finite element computer program (see Section 3.8.4.4). The mat analyses are discussed in Sections 3.8.1.4.1 and 3.8.5.4. Analysis of interior concrete structures is described in Section 3.8.3.4.

3.8.1.4.2.3 Brackets, Attachments, Pad Overlay Plates, and Inserts

Manual calculations are employed in the analyses of pad overlay plates and inserts. Brackets and attachments together with their anchorages into the Reactor Building wall are designed in accordance with the provisions of AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings" [18].

3.8.1.4.2.4 Anchors

The analysis and design of the liner anchors considers the effects of the following:

1. Unbalanced loads resulting from the variation of liner curvature.
2. Variation in anchor Spacing.
3. Misalignment of liner plate seams.
4. Liner thickness greater than nominal due to tolerances specified by ASME SA20.
5. Variation in liner plate yield stress higher than the specified minimum.
6. Local concrete crushing in the anchor zone.
7. Variation in anchor stiffness due to variation of concrete modulus.
8. Voids in the concrete behind the liner.

The liner anchorage system consists of meridional angles (3 by 3 by 1/4 inch) Spaced at 15 inch centers. The effects of Adjacent panel buckling are considered in the design of the anchor to liner welds and the anchors.

The design and analysis of the steel liner and anchor system was based upon the considerations and approaches described in Reference [19]. The referenced approach considers a liner anchorage system utilizing the same basic parameters as were used for this design.

The analysis considers a 1 inch wide strip of liner plate and angle anchor. The mathematical model consists of a simplified spring-node system as shown by Figure 3.8-30a. Anchorage Displacements have been determined based upon the solution of the equilibrium equations for Displacement at each node.

The following simplifying and conservative assumptions have been considered in the analysis. Internal pressure accompanying the high temperature for accident conditions has been disregarded. Buckling inward, in a direction opposite to initial outward curvature, has been disregarded. The model considers initially straight panels with the exception of the buckled panel.

The stable panels are assumed to exert an in-plane shear force corresponding to their yield stress although the critical load, considering an initially straight member, may be less than the load corresponding to yield stress.

Certain material, fabrication, and construction variables affect the liner-anchor system design. Initial inward curvature of the assumed buckled panel and variation in anchor Spacing decrease the post buckling strength of the buckled panel. Liner thicknesses in the unbuckled panels of 16% greater than nominal due to rolling tolerances result in increased anchor forces and deformations. However, this effect is considerably less than an increase in yield stress above the minimum yield stress [19].

For this design, an increase in the yield stress of 35% over the nominal yield stress (43 ksi used versus 32 ksi nominal yield SA516, Grade 60 material) is conservatively adequate to account for material yield, as well as liner thickness variation.

ASME Code Class MC tolerances are adhered to for alignment of liner plate seams. Nominal weld reinforcement at the weld seams assure no decrease in panel strength due to misalignment within specified tolerances.

Local concrete crushing in the anchor zone may occur. Any crushing is expected to occur at the corners where the anchor is welded to the liner. This local crushing is accounted for in the design by using a load-Displacement relation for the anchor based upon tests of the liner-anchor system to the ultimate load [19]. Concrete modulus is relatively uniform for the Reactor Building shell concrete and, therefore, does not affect the liner anchor design. Small local voids in the vicinity of an anchor will not affect the overall stability of the liner-anchor system. The anchor load carrying ability will decrease only in the local area with resulting redistribution of liner forces to the Adjacent area of anchorage, assuring against gross instability of the inner-anchor system in the area.

In summary, certain material and fabrication variables will tend to decrease the load carrying capability of the buckled panel while, on the other hand, certain variables will increase the strength of the unbuckled panels, resulting in greater in-plane forces across the liner anchors. For the limiting case, the post buckling Capacity of the buckled panel was assumed to be zero while the yield stress in the unbuckled panel was considered to be 35% greater than the minimum specified yield, as shown in Table 4 of Reference [19].

For this limiting case, the factor of safety for the maximum computed deflection versus the ultimate deflection is 8. Ultimate deflections are obtained through tests described in Reference [19] for a liner-anchor system utilizing the same variables.

Anchors around insert plates are designed using manual calculations. Anchors at brackets and penetration sleeves serve as anchorages for equipment of piping loads transmitted into the concrete shell. In addition, these embedments anchor the insert plates, assuring liner stability in these local areas.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Concrete Structure

To keep the Reactor Building basically elastic under Service Load Combinations, S, and below the range of general yield under Factored Load Combinations, U, the Allowable (S) and Permissible (U) stresses and strains specified in this section are used. These allowables apply to the load combinations given in Table 3.8-1.

3.8.1.5.1.1 Permissibles for Factored Load Combinations, U

The concrete, nonprestressed reinforcement and tendon stresses do not exceed the permissible values specified in this section. The permissible values for concrete stresses and nonprestressed reinforcement stresses and strains are based upon the Capacity reduction factors, ϕ of ACI 318-71, except that $\phi = 0.9$ is used for axial compression occurring either with or without bending.

1. Axial Forces and Moments

   The concrete maximum compressive stress is limited to $\phi\ (0.85\ f'_c) = 0.765\ f'_c$. The nonprestressed reinforcement is subject to a $\phi\ f_y = 0.90\ f_y$ stress limit and to a strain limit of 1.5 times its yield strain. The section has a Capacity to resist the controlling design forces and moments without exceeding these permissible values. In evaluation of this Capacity, the section is kept below a general yield state by taking the concrete stress distribution to be linear from the cracked section neutral axis to the extreme compression fiber. The distribution of the strain across the section is taken as linear.

   The tensile strength of the concrete is not relied upon to resist the design forces and moments.

2. Radial and Transverse Shear

   The shear forces which are oriented in the thickness direction of the Reactor Building are termed "transverse shear" for the structural foundation mat and "radial shear" for the wall, ring girder, and dome.

   For the structural foundation mat, if the calculated transverse shears exceed the permissible values given in ACI 318-71, Sections 11.4 and 11.9, shear reinforcement is provided. Design of the reinforcement is in accordance with the requirements noted in Section 3.8.1.4.

   For the wall, ring girder, and dome, if the calculated radial shears exceed the permissible values given in ACI 318-71, Section 11.5, as modified below, shear reinforcement is provided. Design of the reinforcement is in accordance with the requirements noted in Section 3.8.1.4.

The exceptions to ACI 318-71, Section 11.5, are as follows:

a. Equation (11-10) of ACI 318-71 is not applicable.

b. Equation (11-11) of ACI 318-71 is replaced by

$$v_{ci} = K\sqrt{f'_c} + \left[\frac{M_{cr}\dfrac{V}{M'} + V_i}{b'd} \right]$$

Where:

$$K = 1.75 - \frac{0.036}{np} + 4.0np$$

but is not less than 0.6 for $p \geq 0.003$. For $p < 0.003$, the value of K is zero.

$$M_{cr} = \frac{I_g}{y_t}\left[6\sqrt{f'_c} + f_{pe}\right]$$

but is not less than zero. However, in case the section has been cracked earlier, the term $6\sqrt{f'_c}$ is replaced by zero.

c. The value of v_{ci} is calculated using alternate Methods A and B, as specified below. The smaller value of v_{ci} thus obtained is used in accordance with ACI 318-71, Section 11.6. The lower limit of $1.7\sqrt{f'_c}$ placed on v_{ci} by ACI 318-71 does not apply.

(1) Method A

Each live load is applied independently, in turn. All other live loads comprising the loading condition are treated as existing dead loads. Each such individually considered live load is designated "Q".

The following definitions are used:

(a) f_{pe} - Compressive stress in concrete (due to all loads except "Q") at the extreme fiber of a section at which tension stresses are caused by "Q"; f_{pe} is calculated at a distance d/2 from the section being investigated for shear, measured in the direction of decreasing moment. Relaxation and creep losses are considered if this leads to more conservative design.

(b) V - Shear caused by "Q" only at the section under consideration.

(c) M' - Maximum moment caused by "Q" only at a distance d/2 from the section under consideration, measured in the direction of decreasing moment.

(d) V_i - Shear caused by prestress and load at the section under consideration.

(2) Method B

All live loads comprising a given loading condition are applied simultaneously.

The following definitions are used:

(a) f_{pe} - Compressive concrete stress caused by post tensioning and dead load at the extreme fiber of a section at which tensile stresses are caused by the live loads; f_{pe} is calculated at a distance d/2 from the section being investigated for shear, in the direction of decreasing moment. Relaxation and creep stresses are considered if this leads to a more conservative design.

(b) v - Shear caused by the live loads at the section under consideration.

(c) M' - Maximum moment caused by the live loads at a distance d/2 from the section under consideration, measured in the direction of decreasing moment.

(d) v_i - Shear caused by prestress and dead load at the section under consideration.

d. Equation (11-12) of ACI 318-71 is replaced by

$$v_{cw} = 3.5\sqrt{f_c'}\left[1 + \frac{f_{pc}}{3.5\sqrt{f_c'}}\right]^{1/2}$$

Where:

f_{pc} is positive for compression.

3. Radial Tension under Post Tensioning

The radial forces caused by the curvature of the hoop and dome tendons are resisted entirely by ties. Design of the ties is discussed in Section 3.8.1.4.

4. Tangential Shear

This shear occurs in the Reactor Building wall during earthquake motion. It is oriented in the plane of the wall. The provisions of ACI 318-71, Section 11.16, are not applicable. Reinforcement is provided to resist the entire value of tangential shear. Design of the reinforcement is in accordance with the requirements noted in Section 3.8.1.4. The value of the nominal shear stress, v_u, is not allowed to exceed 8 $\sqrt{f'_c}$. RN 01-113

5. Punching Shear

This shear occurs in the Reactor Building wall around piping penetrations and the large access openings under specific load combinations. The local shears produced by the polar crane bracket supports also fall under this classification. The nominal shear stress, v_u, is not allowed to exceed 4 $\sqrt{f'_c}$.unless shear reinforcement is provided. If shear reinforcement is designed in accordance with the requirements noted in Section 3.8.1.4, v_u is not allowed to exceed 6 $\sqrt{f'_c}$. RN 01-113

6. Torsion

Under a specific load combination, the primary shield wall produces a torsional force upon the portion of the structural foundation mat which spans over the incore instrumentation pit. For this condition, the provisions of ACI 318-71, Section 11.7, are applicable.

7. Bearing

The provisions of ACI 318-71, Section 10.14, are applicable. The permissible bearing stress is 0.60 f'_c.

8. Tendon Stresses

The axial tensile capacity of the tendons is not allowed to exceed 0.9 f_{py}, where f_{py} is the minimum specified tensile yield strength of the prestressing steel.

9. Thermal Limitations

   Internal concrete surface temperatures are limited to 350°F, except in local areas exposed to steam jets, where 575°F is the limitation.

3.8.1.5.1.2 Allowables for Service Load Combinations, S

The concrete, nonprestressed reinforcement and tendon stresses do not exceed the allowable values specified herein. These allowables apply to all S Load Combinations in Table 3.8-1, including the initial prestress condition. For this condition, the effects of the prestressing sequence are considered. This includes consideration of stresses and deformations before prestressing, as well as at intermediate stages of the prestressing operation.

The concrete stresses are such that:

1. Membrane compressive stresses do not exceed $0.45f'_c$.

2. Extreme fiber compressive stresses due to membrane plus bending do not exceed $0.60f'_c$.

3. The tensile strength of the concrete is not relied upon to resist the design forces and moments. It is a serviceability requirement, however, that the membrane tensile stresses at transfer of prestress do not exceed $\sqrt{f'_c}$.

4. Compressive stress under the tendon anchor bearing plates does not exceed either $0.60f'_{ci}\sqrt[3]{A_2 / A_1}$ or f'_{ci}, where f'_{ci} is the compressive strength of the concrete at the time of initial prestress. The bearing plate area is A_1 and A_2 is the maximum area of the portion of the anchorage surface that is geometrically similar to, and concentric with, the area of the bearing plate.

5. Bearing stresses, excluding those for the tendon anchor bearing plates, do not exceed 35% of those permitted for factored loads.

6. The allowable shear and torsion stresses are taken as 55% of those permitted for factored loads.

7. The nonprestressed reinforcement stresses are such that tensile and compressive stresses do not exceed 24,000 psi, except for shear reinforcement, in which case these stresses do not exceed 30,000 psi.

8. The tendon stresses are such that:

   a. The tension stress during stressing is limited to 0.80 f_{pu}, where f_{pu} is the minimum specified ultimate tensile strength of the wire.

   b. The tension stress immediately after anchoring does not exceed 0.70 f_{pu}.

9. The temperature in the concrete is limited to 150°F, except in local areas such as pipe penetration locations where 200°F is the limitation.

3.8.1.5.2 Reactor Building Liner and Attachments

3.8.1.5.2.1 Liner and Anchors

Structural acceptance criteria for the Reactor Building liner are in accordance with the allowable stresses and strains stated in the ASME Code, Section III, Division 2, Table CC-3720-1. These allowables apply to the corresponding load combinations presented in Table 3.8-2.

Acceptance criteria for the Reactor Building liner anchors are in accordance with the allowable displacements stated in the ASME Code, Section III, Division 2, Table CC-3730-1. These allowables apply to the corresponding load combinations presented in Table 3.8-2.

3.8.1.5.2.2 Brackets, Attachments, and Pad Overlay Plates

Structural acceptance criteria for brackets, attachments, and pad overlay plates are presented in Table 3.8-3.

3.8.1.6 <u>Materials, Quality Control, and Special Construction Techniques</u>

3.8.1.6.1 Concrete

3.8.1.6.1.1 Material

1. Cement

   Cement used is Type II, moderate heat of hydration, with the following additional requirements:

   a. Maximum 58% sum of tricalcium aluminate and tricalcium silicate for 7 and 28 day cube strengths.

   b. Minimum compressive cube strength of 4800 psi at 28 days or 3000 psi at 7 days.

c. Maximum fineness of 4000 cm^2/gm for 7 and 28 day cube strengths.

d. Air content of cement mortar: 7% minimum; 12% maximum.

e. Loss on ignition of the cement: 2% maximum.

f. Minimum fineness of 2800 cm^2/gm for 7 day cube strengths.

g. For 7 day cube strength, the minimum limit of tricalcium aluminate is 42%.

Cement complies with the requirements of the ASME Code, Section III, Division 2, Article CC-2221.

2. Aggregates

a. Fine Aggregate

Fine aggregate complies with the requirements of ACI 301-72 [20] and ASTM C33. The fine aggregate complies with the ASME Code, Section III, Division 2, Article CC-2222.1(a), (e), and (g).

b. Course Aggregate

Course aggregate complies with the requirements of ACI 301-72 and ASTM C33. Based upon compressive strength tests, tolerance of ±5% on each sieve size of standard No. 67 gradation is allowed. South Carolina State Highway Department (SCSHD) Course Aggregate gradation 8M size and the gradation of ASTM C33 No. 8 is used. Course aggregate complies with the ASME Code, Section III, Division 2, Article CC-2222, with respect to potential reactivity, petrographic examination, and aggregate shapes and sizes. Course aggregate shapes are tested in accordance with CRD-C119, "Method of Test for Flat and Elongated Particles in Course Aggregate" using the "count" method which is more conservative than the "weight" method.

3. Water

Mixing water complies with the requirements of the ASME Code, Section III, Division 2, Article CC-2223 and the following additional requirements:

a. Sulfate ion, 250 ppm

b. Total solids content, 1000 ppm

4. Admixtures

The following admixtures are used:

a. Air entraining admixture complying with ASTM C260.

b. Water reducing admixture complying with ASTM C494, Types A and D.

Admixtures comply with the requirements of the ASME Code, Section III, Division 2, Article CC-2224.

3.8.1.6.1.2 Concrete Mixes

1. Specified Properties

The specified properties are as follows:

a. Concrete compressive strength of 5000 psi at 90 days.

Fly ash is not used in the concrete mix. The schedule of testing to verify that concrete reaches a compressive strength of 5000 psi at 90 days is as follows: one cylinder is tested at 7 days for information; 1 at 28 days for information; and 2 at 90 days for verification of 90 day strength.

b. Maximum slump of 4 inches

c. Air content, 4 to 8%

d. Unit weight, 142.1 lb/ft^3

Confirmatory tests for modulus of elasticity, Poisson's ratio, coefficient of thermal expansion, shrinkage, uniaxial creep, tensile strength, and thermal conductivity were performed to provide actual property values for comparison with assumed design values. These values compare favorably with those assumed for the design and, as such, are confirmatory.

2. Concrete Mix Proportioning

Concrete mix proportioning complies with the requirements of the ASME Code, Section III, Division 2, Article CC-2232.

3.8.1.6.1.3 Concrete Construction

1. Formwork

   For the wall the steel liner is used as an inside form with conventional formwork on the outside face. Special precautions are taken to limit the stress in the steel liner. These precautions consist of ties between the formwork and a concrete maximum pour rate of 24 inches per hour. Also, the liner is locally stiffened by temporary bracing to hold it within tolerance.

   The dome liner serves as an inside form. It is supported at intervals by means of ties attached to a system of rigid dome tendon conduit.

   Formwork design complies with the requirements of the ASME Code, Section III, Division 2, Article CC-4251.

2. Construction Joints

   a. Structural Foundation Mat

      The vertical joints for the 8 pie shaped concrete placements consist of nominal 3 inch by 12 inch keys extending through the entire mat thickness and are sufficient to resist the calculated transverse shears at the joints. These keys are performed with expanded metal (1/4 by No. 18) which is cast in place to serve as a bulkhead for the concrete placement. The remaining mat joints are provided with removable, wood formed keys. Keys are sized to resist calculated transverse shears where applicable.

   b. Wall and Dome

      Joints are designed to resist calculated radial (transverse) and tangential shears. Horizontal joints in the wall are provided with nominal 3 inch by 12 inch keys in addition to being roughened to 1/4 inch amplitude using water under pressure. Horizontal joints are provided with a 1/2 inch layer of mortar just prior to placement of the next lift. In the upper and lower part of the wall, key sizes are provided based upon calculated shear requirements. Vertical joints in the wall are provided with nominal 3 inch by 12 inch keys and are roughened to 1/4 inch amplitude. Vertical joints in the wall are formed using a combination of a 12 inch wide section of expanded metal (cast-in-place) and removable wood bulkheads. Both extend the height of the pour. The expanded metal is centered at the horizontal conduit radius. Construction joints in the ring girder are roughened to 1/4 inch amplitude and are sufficiently reinforced to transfer shears by shear friction. Construction joints in the dome are formed using cast-in-place expanded metal.

Construction joints comply with the requirements of the ASME Code, Section III, Division 2, Article CC-4252.

3. Batching of Concrete

   Batching of concrete complies with the requirements of the ASME Code, Section III, Division 2, Article CC-4222. The batch plant is calibrated to ASTM C94 on a 90 day frequency by the testing laboratory.

4. Mixing of Concrete

   Mixing of concrete complies with the requirements of the ASME Code, Section III, Division 2, Article CC-4223.

5. Conveying of Concrete

   Conveying of concrete complies with the requirements of the ASME Code, Section III, Division 2, Article CC-4224.

6. Placing of Concrete

   Placing of concrete complies with the requirements of the ASME Code, Section III, Division 2, Article CC-4225.

   Placement of concrete under cold and hot weather conditions complies with the requirements of the ASME Code, Section III, Division 2, Article CC-4260.

7. Consolidation

   Consolidation complies with the requirements of the ASME Code, Section III, Division 2, Article CC-4226.1. Designed mixes using 3/8 inch maximum size aggregate are provided for use in areas of congestion.

8. Curing

   Elements of the concrete Reactor Building, including structural foundation mat, wall, ring girder, and dome are cured in accordance with Chapter 14, "Massive Concrete," of ACI 301-72, except that only continuous water curing is permitted for the foundation mat. In addition the requirements of ACI 306-66 [21] are followed instead of ACI 301-72, Section 14.5.4.

   Curing of concrete complies with the requirements of the ASME Code, Section III, Division 2, Article 4240.

9. Mass Concrete

Sections 3'-9" or larger in the smallest dimension are considered as mass concrete. Such sections are placed and cured in accordance with the mass concrete provisions of ACI 301-72, with the following additional requirements:

a. Section 14.5.4 of ACI 301-72, which discusses the drop of temperature at the end of the curing period is replaced by the more complete requirements of ACI 306-66.

b. The time between placement of abutting concrete segments was a minimum of 3 days for the shell and a minimum of 7 days for the foundation mat.

c. The temperature in the Reactor Building structural foundation mat was monitored throughout the curing period until a downward trend in temperature was observed. In addition, at no time during the curing period did the concrete exceed the predetermined design value of 175°F.

d. Continuous wet curing was the method of curing for the Reactor Building foundation mat.

3.8.1.6.1.4 Quality Control

1. Cement

Quality controls for cement comply with the requirements of the ASME Code, Section III, Division 2, Article CC-5221, with respect to ASTM C150 as modified in Section 3.8.1.6.1.1 for physical and chemical properties. In addition the temperature of the cement was limited to a maximum of 170°F.

2. Aggregates

Quality control for aggregates complies with the requirements of the ASME Code, Section III, Division 2, Article CC-5224, for flat and elongated particles, friable particles, lightweight particles, Los Angeles abrasion, potential reactivity, and soundness. Absorption is tested using ASTM C127 or ASTM C128 procedures prior to approval for use. Gradation is tested using ASTM C136 procedures within 1 day of use. Moisture content is determined at least daily in accordance with ASTM C566 or ASTM C70 prior to production with a minimum frequency of 500 tons. The quantity of material finer than No. 200 sieve per ASTM C117 is determined within 1 day of use. Organic impurities are determined in accordance with ASTM C40 weekly during production.

3. Water and Ice

   The following tests are performed every 6 months:

   a. Effect on compressive strength, in accordance with ASTM C109

   b. Setting time, in accordance with ASTM C191

   c. Soundness, in accordance with ASTM C151

   d. Chloride content, in accordance with ASTM D512

   e. Sulfates, in accordance with ASTM D516

4. Admixtures

   The admixture supplier is required to submit a certificate of compliance with ASTM C260 or ASTM C494 prior to or with each delivery of admixture.

5. Concrete

   Quality control requirements for concrete comply with the requirements of the ASME Code, Section III, Division 2, Article CC-5230 for the following:

   a. Mixer uniformity

   b. Sampling method

   c. Compression cylinder

   d. Compressive strength

   e. Slump

   f. Air content

   g. Temperature

   h. Unit weight/yield

Test to confirm the validity of the engineering design for the following parameters were initiated upon approval of the concrete mix designs:

a. Modulus of elasticity, in accordance with ASTM C469

b. Poisson’s ratio, in accordance with ASTM C469

c. Coefficient of thermal expansion, in accordance with CRD-C39

d. Shrinkage, in accordance with ASTM C157

e. Uniaxial creep, in accordance with ASTM C512

f. Splitting tensile strength, in accordance with ASTM C496

g. Thermal conductivity, in accordance with ASTM C177

Regulatory Guide 1.55 is addressed in Appendix 3A.

3.8.1.6.2 Reinforcing Steel and Cadweld Splices

3.8.1.6.2.1 Material

Reinforcing steel is Grade 60 billet steel which conforms to the requirements of ASTM A615. Materials for reinforcing steel and for Cadweld splices (mechanical splice sleeves) comply with the requirements of the ASME Code, Section III, Division 2, Article CC-2310. Regulatory Guide 1.15 is addressed in Appendix 3A.

3.8.1.6.2.2 Cadweld Splices

Reinforcing bars, sizes #11, #14, and #18, are spliced using Cadweld splices in specified locations. The Cadweld splices develop the tensile strength of the reinforcement. Cadwelding is performed in accordance with the ASME Code, Section III, Division 2, Article CC-4333. Regulatory Guide 1.10 is discussed in Appendix 3A.

3.8.1.6.2.3 Quality Control

1. Reinforcing Steel

   a. Tensile Tests

      Testing frequency complies with the requirements of the ASME Code, Section III, Division 2, Article CC-2331.1. Where the producing mill had not determined tensile properties at the frequency required by Article CC-2331.1 of the Code, user tests were performed by a testing laboratory at a frequency that satisfies Article CC-2331.1 requirements.

b. Acceptance Standards for Tensile Tests

(1) Where the producing mill determined tensile properties, acceptance standards for such tests complied with the requirements of Article CC-2331.2 of the Code.

(2) Where user tests were performed by a testing laboratory, acceptance standards were as follows:

(a) If a test specimen failed to satisfy the tensile requirements of ASTM A615-72, Table 2, 4 random specimens were selected and tested.

(b) If all 4 specimens satisfied the ASTM A615-72, Table 2, requirements, the reinforcement was accepted. If one of the 4 specimens failed to satisfy these requirements, a second set of 4 random specimens was selected and tested. If 2 or more of the original 4 specimens failed to satisfy these requirements, the material represented by the specimens was rejected.

(c) If one or more of the second set of 4 specimens fail to satisfy the ASTM A615-72, Table 2, requirements, the material represented by the specimens is rejected.

c. Bend Tests

Bend tests comply with the requirements of the ASME Code, Section III, Division 2, Article CC-2332.

d. Chemical Analyses

Chemical analyses comply with the requirements of the ASME Code, Section III, Division 2, Article CC-2333.

e. Traceability

Reinforcing steel traceability complied with Article CC-2320 of the ASME Code, Section III, Division 2.

2. Cadweld Splices

Cadweld splice testing and acceptance criteria are in accordance with Article CC-4333 of the ASME Code, Section III, Division 2.

3.8.1.6.3 Post Tensioning System

The Post Tensioning System generally conforms to the intent of the requirements of the ASME Code, Section III, Division 2. Minor deviations from the Code exist because procurement and part of the fabrication of components predate issuance of this portion of the Code. However, these deviations either exceed Code requirements or satisfy the intent of the Code. Where deviations exist, they are explained. Applicable cross references to articles of the Code are provided at ends of paragraphs in Sections 3.8.1.6.3.1 through 3.8.1.6.3.4.

3.8.1.6.3.1 Material

Materials for the components of the Post Tensioning System are as follows:

1. Semi-Rigid Conduit and Screw Couplings

   Semi-rigid conduit and screw couplings, for wall tendons, are fabricated from sheet steel, galvanized on both surfaces and manufactured in accordance with ASTM A 525-71. This galvanized sheet steel is spirally formed into circular tube (Article CC-2441(a)).

2. Rigid Conduit and Sleeve Couplings

   Schedule 40 conduit and Schedule 60 couplings, for dome tendons, are fabricated from piping conforming to ASTM A53-72a, Type S, Grade B, requirements. Both surfaces of Schedule 40 pipe are galvanized in accordance with the requirements of ASTM A123-73 (Article CC-2441(a)).

3. Trumpets

   Trumpets are manufactured from hot rolled electric welded (HREW) steel tubing (Article CC-2441(a)).

4. Transition Funnels

   Transition funnels are spun from cold rolled sheet steel (Article CC-2441(a)).

5. Permanent End Caps, Shims, and Gaskets

   Materials for these components are as follows:

   a. Permanent end caps are fabricated from hot rolled steel conforming to the requirements of ASTM A36-70a and HREW tubing.

   b. Permanent end cap shims are fabricated from hot rolled steel conforming to ASTM A36-70a.

c. Permanent end cap gaskets are manufactured from Neoprene durometer 60.

6. Wire

Wire is manufactured in accordance with ASTM A421-65 using type BA anchorage (Article CC-2421).

7. Anchorage Components

a. Bearing plates are fabricated from steel plate conforming to ASTM A36-70a with S2 supplementary requirements (Article CC-2432).

b. Anchor heads are fabricated from AISI 4142 steel, heat treated to Rockwell hardness 42 (Article CC-2433).

c. Bushings are fabricated from AISI 4142 steel, heat treated to Rockwell hardness 42 (Article CC-2433).

d. Shims are fabricated from steel plate conforming to ASTM A36-70a with S2 supplementary requirements.

8. Corrosion Protection Grease

a. Temporary corrosion protection grease applied to tendon wires is Visconorust 1702 with an overcoating of Visconorust 1601 or their equivalents (Article CC-2442.2.2).

b. Permanent corrosion protection grease pumped into the conduit is Visconorust 2090P-4 or its equivalent (Article CC-2442.3.2(a) and (b)).

9. Tendon Performance Test

Tendon qualification by performance tests is reported in the Three Mile Island Nuclear Station Unit 1 Final Safety Analysis Report.

The Post Tensioning Tendon System is qualified in accordance with Regulatory Guide 1.103 (see Appendix 3A).

The Post Tensioning Tendon System also satisfies the requirements of the ASME Code, Section III, Division 2, Article CC-2460.

The tendon system has been tested without failure at -73°F for 500 cycles of load range of 60% to 80% of guaranteed ultimate tensile strength. This test is above and beyond the requirements of the ASME Code.

3.8.1.6.3.2 Fabrication

1. Conduit

   a. Semi-Rigid

      Semi-rigid conduit is fabricated from sheet steel formed into tubing by the "Spiro" process. This tubing is then cut into lengths suitable for shipment to the construction site.

   b. Rigid

      Rigid conduit is fabricated from Schedule 40 pipe. The conduit is cut and bent in the shop to dimensions indicated by the shop drawings.

2. Anchorage Components

   Anchorage components are fabricated in accordance with shop drawings and fabrication procedures of the tendon manufacturer.

3. Permanent End Caps and Shims

   Permanent end caps and shims are fabricated in accordance with the shop drawings and fabrication procedures of the tendon manufacturer.

4. Tendons

   a. Buttonheads

      Buttonheads are of the cold upset type. The tendon manufacturer is required to develop dimensional and split (crack) criteria to ensure that buttonhead strength exceeds that of the wire (Article CC-4432.3).

   b. Tendons

      The individual wires of a tendon are cut to "as measured" length, then threaded through the anchor heads and buttonheaded on one end of the wire in the shop. When all wires for a particular tendon are cut, threaded through the anchor head and buttonheaded, the tendon is coated with the corrosion protection greases, twisted, banded, coiled, and protected for shipment to the construction site (Article CC-4432).

3.8.1.6.3.3 Installation

1. Conduit

   a. Semi-Rigid

      Semi-rigid conduit is cut to length onsite and placed in the Reactor Building structure in the positions shown by the construction drawings. Any required bending is accomplished by hand. Joints are formed by threading an oversize coupling from one conduit section onto the next. Joints are taped to prevent intrusion of concrete.

   b. Rigid

      Each section of shop bent, Schedule 40 conduit for the Reactor Building dome carries an identification mark placed on it by the conduit supplier. This mark corresponds to the conduit section mark on the approved construction drawings. The constructor places each dome tendon conduit section in the location designated by the construction drawings.

2. Bearing Plate/Trumpet Embedded Anchorages

   Bearing plate/trumpet embedded anchorages are cast into the Reactor Building structure. The embedded anchorages required a steel frame for temporary support before being cast in concrete. No form of welding or heating of the bearing plate portion of embedded anchorages is allowed. When an embedded anchorage is erected, the constructor attaches a temporary end cover to it. This temporary end cover remains in position until tendon placement to prevent entrance of debris and water.

3. Conduit Cleaning

   The internal surface of the conduit is first cleaned by pulling cloths through to remove debris. The internal surface is then coated with Visconorust 209OP-4 corrosion protection grease or its equivalent.

4. Tendon Installation

   Tendons are removed from onsite storage, unwrapped, and, placed on an uncoiler. After attaching a "Kellums Grip," the tendon is pulled through the conduit by means of a winch connected to the "Kellums Grip" attachment.

5. Buttonheading

   Field formed buttonheads are formed on the wires of each tendon in accordance with the same criteria used for shop formed buttonheads.

6. Tendon Stressing

Tendons are not stressed until all concrete for the complete Reactor Building shell has been placed and has reached a minimum strength of 5000 psi. The tendon manufacturer is required to calculate, for each tendon, the anticipated elongation at each stressed end. These elongation calculations are based upon established friction and wobble coefficients.

a. Force and Strain Measurements

(1) Each tendon is initially stressed to a maximum of 80% of the minimum guaranteed ultimate capacity of the tendon. The jacking force is then reduced to 70% of ultimate capacity and locked off. Stress-strain curves prepared by the wire manufacturer for production lots of the material are used to establish tendon elongation. These curves are then used by the tendon manufacturer to establish the final gage reading and elongation of each stressed tendon. If loss of prestress force in any tendon due to broken wires and/or unacceptable buttonheads exceeds either 3% for 1 tendon or 2% for 2 adjacent tendons, the tendon is rejected and either repaired or replaced.

(2) After taking up initial slack by jacking to a maximum jack pressure of 2000 psig, force strain measurements are taken. These measurements are obtained by measuring the tendon elongation and comparing it with the force indicated by the jack pressure gage. Prior to being sent to the site, all pressure gages are calibrated using a deadweight tester, traceable to the National Bureau of Standards. Once at the site, one of the gages is set aside for purposes of calibration verification. Calibration of each gage is verified after every 24 hours operation for that gage. Prior to being sent to the site, stressing rams are calibrated for ram area by use of a load cell, traceable to the National Bureau of Standards. The pressure gage-stressing ram system indicates tensioning force within an accuracy of ±2%. During stressing, records are made of elongation, as well as pressure obtained for each tendon. At the pressure gage reading equivalent to 80% of the guaranteed ultimate capacity of the tendon, the tendon elongation is measured simultaneously at each stressing end. The measured elongation is compared to that calculated by the contractor (using average load-elongation curves). If the maximum differential between measured and calculated elongation exceeds -5%, or +10% the contractor is required to immediately report the discrepancy in writing (Article CC-4464.1).

b. Stressing Sequence

The tendon stressing sequence is as follows (Article CC-4462):

(1) Vertical tendons

(2) Dome tendons

(3) Horizontal tendons

The contractor is required to develop a complete, detailed stressing sequence for each type of tendon. This detailed sequence is required to comply with the requirements given below. Tendons are stressed simultaneously, as described below:

(1) Vertical Tendons

Vertical tendons are stressed from the top end only, using 6 sets of stressing equipment equally spaced around the ring girder, stressing 6 vertical tendons simultaneously.

As an alternate to stressing a group of 6 equally spaced vertical tendons simultaneously, the group of 6 tendons is divided into 2 subgroups with 3 tendons in each subgroup. The 3 tendons within each subgroup are equally spaced and simultaneously stressed. The 2 subgroups may be stressed independently but stressing of both subgroups must be completed prior to stressing the next group of 6 equally spaced tendons.

(2) Dome Tendons

Except where prevented by jacking equipment interferences, dome tendons are stressed from both ends using 6 sets of stressing equipment to stress these tendons simultaneously. The set of 3 tendons for which simultaneous stressing is specified is defined as a sequence. Where the 3 tendons in a sequence cannot be stressed simultaneously because of an interference between the stressing equipment for one tendon and that of another, consecutive stressing is permitted. In this event, the 3 tendons are stressed within 8 hours; that is, the attainment of overstress pressure on the third tendon of a sequence occurs within 8 hours of the attainment of a jack pressure of 2000 psi, the preliminary tensioning level, for the first tendon in that sequence.

(3) Horizontal Tendons

Horizontal tendons are stressed from both ends using 6 sets of stressing equipment. Three (3) horizontal tendons are stressed simultaneously.

7. Corrosion Protection

After the tendons are installed, the permanent corrosion protection grease, Visconorusts 2090P-4 or its equivalent is pumped into the tendon conduit to completely fill the remaining air space. When all air is displaced, signified by a solid stream of grease at the discharge end, pumping is stopped and the conduits are sealed at 0 psig. To facilitate pumping the grease, it is heated so that minimum grease temperature at discharge end is 115°F.

8. Vertical Tendon Retensioning

Because of increased wire relaxation losses as described in Section 3.8.1.4.1.2.1, retensioning of the vertical tendons was implemented in early 1990. The vertical tendons were restored close to their initial lock-off force of 1402 kips without detensioning in order to maintain the minimum average required vertical prestress force for the plant 40 year life. The loading on the containment structure from retensioning vertical tendons does not impose any additional or combination of loads not previously considered in the original analysis and design. The sequence for retensioning vertical tendons was selected to provide even application of the incremental vertical prestress around the containment walls during the retensioning process.

3.8.1.6.3.4 Quality Control

1. Wire

Certified physical and chemical mill test reports for each reel of wire are obtained by the tendon manufacturer (Article CC-2423). Only full diameter test pieces are used in the physical tests (Article CC-2422). Load strain curves certifying physical properties of each mill heat of material are obtained by the tendon manufacturer.

In addition to the requirements of ASTM A421-65, stress relaxation property information is required by the construction specification. In the early stage of the contract, stress relaxation properties of the wire were obtained by the tendon manufacturer from the steel producer (Article CC-2424). These properties were based upon tests performed on material previously manufactured under the same ASTM or other specification and produced in the same plant, utilizing the same procedures that were employed to produce material for the production tendons. Data furnished with the test results includes (Article CC-2424):

a. Detailed test method

b. Initial stress

c. Final stress

d. Test time

e. Temperature limits

f. Mathematical tools used to interpret test results

g. Percentage stress relaxation properties for design life.

In view of the fact that the tendon manufacturer had wire of differing stress relaxation properties in storage, stress relaxation property tests are required to ensure that the correct material is used to fabricate tendons. The frequency of testing is, as a minimum, one test per 50 tons of steel or per heat produced, whichever is smaller. Test specimens are selected at random.

Details of this test, are given in the quality control program of the tendon manufacturer. The tendon manufacturer is required to have the test results available for inspection and to forward applicable test results with each shipment of tendons to the job site.

The wire is inspected for rust, splits, cuts, and bends in accordance with the requirements set forth in the tendon manufacturer's quality assurance manual.

2. Anchorage Components

The tendon manufacturer obtains certified physical and chemical mill test reports to verify compliance with the material specification. The frequency of testing is, as a minimum, one test per mill heat or heat treatment batch, whichever is applicable. The tendon manufacturer is required to obtain copies of the certified test reports and to forward applicable copies with each shipment of anchorage components to the job site.

Heat treated components, such as anchor heads, are hardness tested after heat treating (Article CC-2433.2).

The tendon manufacturer fabricates the load carrying anchorage components in accordance with the following requirements:

a. Components, excluding shims, are low stress stamped with unique identifying numbers which give complete tractability from manufacture of the steel to delivery to the job site.

b. Documentation sheets are developed to record the manufacturing record of each individual component. Copies of the applicable, completed sheets accompany each shipment of anchorage components to the job site.

c. Certified copies of physical and chemical mill test results are obtained to verify compliance with the tendon manufacturer's specification requirements. As a minimum, certified test results are provided for each heat of material or heat treatment batch, whichever is applicable.

d. At least 10% of all anchorage components are inspected for hole and thread tolerances, alignment, dimensional control, weld porosity, notches, etc. The tolerances, methods of inspection, frequency of inspection, and acceptance criteria are stated on the tendon manufacturer's shop drawings and in the quality control manual. Inspection records are forwarded with each shipment to the job site.

e. Weld and welder qualification are accomplished prior to commencement of the work. Both qualifications are in accordance with the requirements of AWS D1.0-69 [23].

3. Conduit

Certified copies of the mill certificates for the chemical and physical properties of each heat of steel used in the manufacture of each type of conduit are obtained.

The semi-rigid conduit, when continuously supported on a flat surface, is sufficiently strong to withstand the weight of a 200 pound man without permanent deformation (Article CC-2441(b)).

To determine that the fabricated conduit has minimum required internal dimension, a "pig" of diameter 1/4 inch less than the conduit inside diameter is pulled through the conduit.

After cutting and bending (where applicable), a minimum of 10% of the sections of conduit in each shipment, selected at random, are inspected to verify compliance with the above requirements.

Dimensional tolerances for cutting and bending conduit are checked against established tolerances.

Upon receipt at the job site, the conduit and couplings are visually inspected for damage. If the conduit is damaged or there is evidence of conduit damage, the conduit is inspected for detrimental dents and other defects. To determine if a dent is detrimental, a cylindrical unpainted steel pig, free of corrosion and sharp edges, and having a diameter of either 4 3/4 inches (Schedule 40 conduit) or 4 1/2 inches (semi-rigid conduit) by approximately 20 inches long with tapered ends and lead and tail lanyards, is pulled through the conduit. If a dent or dents prevent passage of the pig, the conduit is rejected and replaced.

After the conduit is placed in the Reactor Building structure, location is checked against established tolerances.

Before and after placement of concrete around conduit, the conduit is checked for dents and restrictions by pulling through 24 inch long, untapered pig. For rigid (Schedule 40) conduit, a cylindrical steel pig of 4 1/2 inch diameter is used. For the semi-rigid conduit, a 4 1/4 inch diameter pig is used (Article CC-5423).

Coupling welds joining sections of rigid conduit are made in accordance with the following:

a. Welders are qualified in accordance with the requirements of AWS D1.1-72 [24].

b. Welds are made in accordance with approved welding procedures.

c. Ten percent (10%) of the welds, selected at random, are liquid penetrant inspected using the solvent removable process.

d. The weld is subjected to pneumatic pressure testing to ensure leak tightness.

4. Bearing Plate/Trumpet Assemblies

Bearing plate/trumpet assemblies are examined for dimensional accuracy in accordance with the criteria given in the tendon manufacturer’s fabrication procedures and shop drawings (Article CC-5422.1).

5. Permanent End Caps

Permanent end caps are inspected in accordance with the criteria given in the tendon manufacturer’s fabrication and quality assurance procedures.

6. Tendon Fabrication

a. Buttonheads

The tendon manufacturer is required to develop buttonhead criteria by means of tests. The wire used in the tests is required to be of the same size and manufacture as that used in the production of tendons. These tests are required to demonstrate that buttonheads conforming to the buttonhead criteria develop the full strength of the wire and that failure takes place in the wire, not the buttonhead. As a minimum, buttonhead criteria include:

(1) Wire - manufacturer, type, quality, etc.

(2) Shape - spherical, flat topped, etc.

(3) Diameter with tolerances

(4) Eccentricity of head and seat surfaces

(5) A bearing surface for 360° of the buttonhead

(6) Hardness range of anchor on which the buttonhead bears

(7) Allowable number of splits in any one buttonhead

(8) Size, inclination, and location of splits (cracks), including maximum values for a single split (crack) and a total value for the maximum number of allowable splits (cracks) in any one buttonhead.

The tendon manufacturer inspects buttonheads in accordance with the following requirements utilizing the buttonhead criteria:

(1) All buttonheads are visually inspected for malformation. Any buttonheads which appear to be malformed due to size or eccentricity are included in the random check noted in items (3) and (4), below. All double buttonheads are inspected for compliance with acceptance criteria for size, eccentricity, slips, and splits.

(2) The frequency of visual inspection for splits is 100%. Any buttonhead found to have a split or splits is checked to determine if it complies with the split criteria.

(3) A random check of buttonhead size using "Go" and "No-Go" gages is made for a minimum of 10% of the buttonheads of each tendon.

(4) Buttonhead eccentricity is checked for a minimum of 5% of the buttonheads per tendon, selected at random, to ensure compliance with the criteria.

(5) All wires in a tendon are cut to the same length by cutting the wires under the same conditions.

(6) As the wire for each tendon is cut, the tendon manufacturer inspects it for rust. Rust grade inspection criteria are noted in the construction specification.

b. Tendons

After fabrication and before twisting and banding, the tendons are inspected to ensure that each wire is coated with corrosion protection grease.

Each tendon is inspected prior to shipping to ensure that it consists of 170 wires, is twisted and banded and is wrapped in protective material (Article CC-5424).

7. Corrosion Protection Grease

Both temporary and permanent corrosion protection greases are checked to ensure that water soluble chloride, nitrate, and sulfide content is less than 10 ppm each (Article CC-2442.3.2).

The following requirements apply to bulk filling of conduit with permanent corrosion protection grease:

a. For vertical tendons, bulk filling is completed within 8 1/2 months of tendon installation and within 7 months for hoop and dome tendons.

b. Bulk filling sealing pressure is 0 psig.

c. Minimum grease discharge temperature is 115°F.

d. Determination that conduit is completely filled is based upon observation of a solid stream of grease at the discharge point.

3.8.1.6.4 Steel Liner, Penetrations, and Attachments

3.8.1.6.4.1 Material

1. Liner Plate

   Material for the liner shell, dome, and floor conforms to ASME SA516, Grade 60, impact tested at minus 25°F in accordance with the ASME Code, Section III, Division 1, Article NE-2300, and Division 2, Article CC-2511. In addition, all liner materials thicker than 3/8 inch are ultrasonically examined in accordance with the requirements of ASME SA578, including supplementary requirements S1, S3, and S4.

   Material for the incore instrumentation pit floor and walls and for sumps in the structural foundation mat conforms to ASME SA240, Grade 304. The finish on the plates is hot rolled, annealed, and pickled as specified in ASME SA480.

2. Liner Attachments

   Liner attachments include angle anchors, studs, brackets, and overlay plates. Material for these items conforms to the following requirements:

   a. Plate

      (1) Carbon steel material conforms to ASME SA516, Grade 60 or Grade 70, impact tested at minus 25°F in accordance with the ASME Code, Section III, Division 1, Article NE-2300, and Division 2, Article CC-2522. In addition, material thicker than 3/8 inch which is stressed in the through thickness direction satisfies the ultrasonic examination requirements of ASME SA578, including supplementary requirements S1, S3, and S4.

      (2) Stainless steel material conforms to ASME SA240, Grade 304. The finish on plates is hot rolled, annealed, and pickled as specified in ASME SA480.

   b. Rolled Structural Sections

      Rolled structural sections conform to ASME SA36 or ASTM A36.

   c. Studs

      Material for studs conforms to ASTM A108, Grade 1015 or Grade 1018 of ASTM A276, Type 304.

3. Penetrations

Materials for different components of penetrations are as follows:

a. Carbon Steel Pipe Sleeves

Material for penetration sleeves of less than 30 inch diameter conforms to ASME SA333, Grade 6 (seamless), impact tested at minus 30°F or below.

Material for penetration sleeves of 30 inch diameter and larger conforms to ASME SA155, Grade KCF 70, Class 1 (welded) impact tested at minus 25°F in accordance with the ASME Code, Section III, Division 1, Article NE-2300, and Division 2, Article CC-2523.

b. Stainless Steel Pipe Sleeves

Material for stainless steel pipe sleeves conform to ASME SA312, Grade 304.

c. Carbon Steel Plates

(1) Liner insert plates conform to ASME SA516, Grade 60, normalized, ultrasonically examined and impact tested at minus 25°F in accordance with the ASME Code, Section III, Division 1, Article NE-2300, and Division 2, Article CC-2523.

(2) Plate material attached to penetration sleeves for concrete anchorage conforms to ASME SA516, Grade 70, normalized, ultrasonically examined and impact tested at minus 25°F in accordance with the ASME Code, Section III, Division 1, Article NE-2300.

d. Penetration Pipe Fittings

Penetration pipe fitting material conforms to ASME SA420, Grade WPL-6 (seamless), impact tested at minus 50°F.

e. Spare Penetrations with Removable End Caps

Spare electrical penetrations #505-18, #600-12, and #602-18 have removable end caps of the following materials:

(1) Securing Studs: ASME SA320 GR L43.

(2) Slip-on Flanges:

#505-18, #600-12 (Outside and Inside) ASME SA350 GR LF2.

#602-18 (Outside) ASME SA350 GR LF2.

#602-18 (Inside) ASME SA 105 C.S.

(3) Blind Flanges: ASME SA350 GR LF2.

(4) Gaskets: Flexitallic type or equal.

4. Access Openings

Materials listed below are used for the equipment hatch, personnel access airlock, and personnel emergency airlock:

a. Carbon Steel

(1) Plate

Carbon steel plates, except liner insert plates, are ASME SA516, Grade 70, normalized and impact tested at minus 25°F in accordance with the ASME Code, Section III, Division 1, Article NE-2300.

Liner insert plates are ASME SA516, Grade 60, normalized and impact tested at minus 25°F in accordance with ASME Code, Section III, Division 1, Article NE-2300.

(2) Structural Shapes

Structural shapes are ASME SA36.

(3) Pipe

Pipe is ASME SA333, Grade 6, seamless.

(4) Pipe Fittings

Pipe fittings are ASME SA420, Grade WPL-6 (seamless) and impact tested at minus 50°F.

(5) Bars

Bars are ASTM A108, Grade 1015.

b. Stainless Steel

(1) Plate

Plate is ASME SA240, Type 304.

(2) Bar and Structural Shapes

Bars and structural shapes are ASME SA479, Type 304.

(3) Pipe

Pipe is ASME SA312, Type 304 (seamless).

(4) Pipe Fittings

Pipe fittings are ASME SA182, Grade F, Type 304 or ASME SA403, Type 304.

3.8.1.6.4.2 Fabrication

The steel liner, penetration sleeves, reinforced insert plates and attachments are fabricated and installed in accordance with the ASME Code, Section III, Division 1, Article NE-4000, and Division 2, Articles CC-4122, 4521 (except item e), 4523 (except 4523.2), 4532, and 4533 (except 4533.2).

3.8.1.6.4.3 Construction

1. Liner Plate

After the concrete structural foundation mat is placed to the base floor elevation, the 1/4 inch thick floor liner plates are installed by welding to embedments in the mat. During this period, the transition section at the lower portion of the 1/4 inch thick cylindrical liner wall plate is also erected. The liner serves as the internal form for placement of the shell concrete. Liner seams are double butt welds, except for the first 2 horizontal welds. On these welds, backing plates are used. The liner plate is continuously anchored to the shell by vertical angle anchors spaced at the maximum center to center distance of 15 inches and horizontal angles spaced at approximately 5 foot centers.

After completion of the cylindrical liner wall, the 1/4 thick dome liner is erected on and supported by a steel space frame consisting of interconnected trusses. This space frame is located within the Reactor Building and is supported from the previously erected dome transition segment of the liner. When erection is complete the space frame is removed.

Careful attention is given to the erection of the shell to ensure that all rings match properly. Fitting and alignment are in accordance with the ASME Code, Section III, Division 1, Article NE-4230. Dimensional tolerances for welded joints are as follows:

a. Root face (land), ±1/16 inch

b. Groove angle, ±10°

c. Groove radius, ±1/32 inch

d. Root opening (without backup), +1/8 inch, -1/16 inch

e. Root opening (with backup), +1/4 inch, 0 inch

The following tolerances apply to the as-built liner:

a. Cylindrical Liner

   Tolerance is ±2 inches with respect to the horizontal radius from the center of the Reactor Building and ±2 inches with respect to elevation. Plumbness must be within 1/2 inch in 10 feet. Maximum out of plumb condition is 2 inches at any elevation. This complies with the ASME Code, Section III, Division 2, Article CC-4522.1.

b. Dome Liner

   Tolerance is ±3 inches with respect to the design radius in compliance with the ASME Code, Section III, Division 2, Article CC-4522.2.

c. Local Bulges

   Local bulges, flat spots, or discontinuities in the cylindrical liner are within the tolerances presented in Table 3.8-5 at the center of a stringline chord.

2. Penetrations

   Penetration assemblies are shop fabricated and field installed by welding to larger diameter penetration sleeves. The penetration sleeves, with reinforcing plates, are installed in the liner prior to concrete placement.

   Tolerances for location of penetration sleeves and attachments are ± 1/2 inch, measured at the inside face to the liner, with respect to azimuth and elevation. The location tolerance for the outside end of the penetration sleeve is ± 1/2 inch with respect to the actual location at the inner face of the liner.

3.8.1.6.4.4 Quality Control

1. Codes and Standards

   Welding operator qualification and weld procedure qualification are in accordance with the ASME Code, Section IX. Stud welding procedure qualification and operator qualification are in accordance with AWS D1.1-72 [24], Part VI, Section 4.

   Fabrication and installation comply with the requirements of the ASME Code, Section III, Division 1, Article NE-4000 and with portions of Division 2.

   Non-Destructive examination of welds complies with the ASME Code, Section III, Division 1, as well as with Regulatory Guide 1.19 (see Appendix 3A).

2. Non-Destructive Examination and Testing

   a. Complete Penetration Welds

      Liner welds are examined by 2% spot radiography in accordance with Regulatory Guide 1.19 (see Appendix 3A) and the ASME Code, Section III, Division 2, Article CC-5531.2. Liner welds not capable of being radiographed are examined by 100% magnetic particle detection.

      Liner welds accessible to vacuum box testing are 100% tested. Where vacuum box testing is not possible, welds are examined by 100% magnetic particle, liquid dye penetrant, or ultrasonic methods in addition to required radiography.

      All welds are 100% visually examined.

      Welds provided with test channels are pressure tested for leak tightness to the design pressure in accordance with the ASME Code, Section V, Article T-1020.

   b. Attachment Welds

      Full penetration attachment welds are examined by 100% magnetic particle, ultrasonic, or liquid dye penetrant methods.

      Fillet welds are examined by magnetic particle or liquid dye penetrant methods. Twenty percent (20%) of these welds, selected at random, are examined.

   c. Stud Welds

      Stud welds are examined using stud welding test method specified by AWS D1.1-72, Paragraph 4.29.

3. Acceptance Standards

   a. Radiography is performed in accordance with the ASME Code, Section III, Division 1, Article NE-5320, and Division 2, Article CC-5542. In addition, surface pinholes are removed by grinding.

   b. Magnetic particle examination is performed in accordance with the ASME Code, Section III, Division 1, Article NE-5340, and Division 2, Article CC-5545.

   c. Liquid dye penetrant examination is performed in accordance with the ASME Code, Section III, Division 1, Article NE-5350, and Division 2, Article CC-5544.

   d. Ultrasonic testing is performed in accordance with the ASME Code, Section III, Division 1, Article NE-5330.

   e. Vacuum box testing reveals through thickness discontinuities by formation of a continuous stream of bubbles in the bubble solution. Formation of single, small bubbles is not considered relevant. Defects revealed by a continuous stream of bubbles are unacceptable. Refer to the ASME Code, Section III, Division 2, Article CC-5546.1.1.

   f. Stud weld acceptance standards are in accordance with AWS D1.1-71, Paragraphs 4.29 and 4.30, and the ASME Code, Section III, Division 2, Article CC-5547.

4. Preliminary Tests

   a. Test channels are shop tested for structural integrity at 115% of design pressure. In addition, such test channels are leak tested at 100% of design pressure.

   b. Test channels located over butt welds in the liner are leak tested at 100% of design pressure before and after concrete placement.

3.8.1.7 Testing and Inservice Inspection Requirements

3.8.1.7.1 Structural Acceptance Test

The structural acceptance test complies with the requirements of Regulatory Guide 1.18 (see Appendix 3A) for prototype containments and satisfies the criteria of the ASME Code, Section III, Division 2, Article CC-6000, except as noted in subsequent paragraphs of this section.

3.8.1.7.1.1 General

During structural acceptance testing, the Reactor Building is subjected to internal air pressures ranging from 0 psig to 65.6 psig (115% of the design pressure of 57 psig) and back to 0 psig. Pressurization and depressurization of the Reactor Building is accomplished in 10 increments: 0 psig to 12 psig, 12 psig to 30 psig, 30 psig to 45 psig, 45 psig to 55 psig, 55 psig to 65.6 psig, 65.6 psig to 55 psig, 55 psig to 45 psig, 45 psig to 30 psig, 30 psig to 12 psig, and 12 psig to 0 psig. Incremental pressurization and depressurization during testing is necessary to allow strain gage readings, deflection measurements, and other observations to be made and also to allow for evaluation and comparison of data with the predicted results as a function of internal pressure as the test proceeds.

When pressurizing and depressurizing the Reactor Building, the pressure change is stopped within -0, +0.10 psig of the desired pressure level increment, except for 65.6 psig and 0 psig.

After the desired pressure has been reached, a wait of at least 60 minutes is required. This permits Reactor Building strains and stresses to adjust before strains, deflection, and other observed data are recorded.

The locations at which displacement measurements are obtained are listed in Table 3.8-6. Four (4) radial and 1 vertical displacement measurements are made at 6 meridians on the cylindrical wall, plus a vertical displacement measurement at the apex of the Reactor Building dome. Radial, vertical, and tangential displacements are measured at 6 equally spaced and symmetrically aligned points on the horizontal and vertical center axes of the equipment hatch.

The design locations of strain gages for strain measurements are listed in Table 3.8-7. Minor variations from these locations occurred in field placement of the gages and are documented. Strain measurements are made at the base, at 4 intermediate points, and at the spring line of the cylindrical wall on the meridian. Strain measurements are also made around the equipment hatch at 4 points symmetrically aligned on the horizontal and vertical axes. The strain measurements are made at 3 positions within the cylindrical wall: inside face, middle, and outside face. Horizontal, vertical, and shear strain in the concrete is measured under the bearing plate of one vertical tendon post tensioning anchor (see Figures 3.8-31 through 34).

The test program includes visual examination of the accessible exterior concrete surface of the Reactor Building. The locations of areas for crack pattern charts are listed in Table 3.8-8 and shown by Figures 3.8-35 through 3.8-37. Concrete cracking is charted in these designated exterior areas of the Reactor Building. These areas include the structural foundation mat and cylindrical wall intersection, mid-height of the cylindrical wall, one quadrant around the equipment hatch, the buttress and cylindrical wall intersection and the intersection cylindrical wall, ring girder, top shelf of ring girder, and dome of Reactor Building. The remaining exterior concrete surface which is accessible from existing floors and platforms is visually examined for cracks.

3.8.1.7.1.2 Preparation of Testing

The installation of all optical instrument targets, direct current displacement transducers (DCDT's), white wash for crack pattern charting, tapes, strain gages, junction boxes, conduit, wires, readout instruments, instrument support structures, and platforms is completed, checked, and tested prior to initiating pressurization of the Reactor Building.

During construction, strain gages are attached to reinforcing bars and waterproofed; also, the lead wires are installed. The waterproofed strain gages and the lead wires are protected against damage during construction. It is recognized that prior to the test, when each strain gage is being checked out, malfunction of gages may occur. Therefore, to avoid having to remove concrete to regain access to the gage, 2 gages are installed at each specified location. The primary and redundant strain gages are attached to a 4'-10" long #6 reinforcing bar with 180 degree hook on each end. This bar is called a sister bar. The gages are placed as close together as practical on the reinforcing bar.

The cable leads from the primary and redundant strain gages are kept separate to prevent a single accident from causing malfunction of both strain gages.

Prior to and during the structural acceptance test, the inside air temperature of the Reactor Building is controlled within limits.

Outside air temperature varies, but due to the relatively short period of time involved in the performance of the structural acceptance test, normal fluctuations of the outside air temperature do not appreciably affect the temperature in the concrete shell.

Both inside and outside air temperature readings are taken and recorded at least twice daily for 2 weeks prior to the start of the structural acceptance test and at each pressure level during the test. The locations of thermocouples for temperature measurement are listed in Table 3.8-9.

3.8.1.7.1.3 Interpretation of Test Data

Test data obtained during the structural acceptance test is interpreted and compared with the analytical values obtained during the analysis and design of the Reactor Building (see Figure 3.8-38). If the test data does not show good agreement with the analytical values, the discrepancies are resolved.

If the test data include any displacements which exceed the predicted extremes, such discrepancies are resolved by means which could include a review of the design, evaluation of measurement errors or material variability, and, possibly, an exploration of the structure.

3.8.1.7.1.4 Calculated Responses

Displacements of the Reactor Building under pressure are computed at several points on the structure. A graph for each of these points shows the limiting displacement for a pressure of 65.6 psig. Pressure versus displacements is plotted at the 0 psig, 30 psig, 45 psig, 55 psig, and 65.6 psig levels during both pressurization and depressurization of the Reactor Building.

Stress cracking from the applied pressure loading is not anticipated due to the residual compressive stress remaining in the shell at the maximum pressure of 65.6 psig. However, existing shrinkage cracks will propagate in length and enlarge slightly in width when the Reactor Building is pressurized. Also, some new hairline cracks may occur that were not initially observed. The crack pattern will be random and the crack width enlargement should be less than 0.010 inch.

3.8.1.7.2 Post Tensioning System Inservice Inspection

The inservice inspection program for the Post Tensioning System tendons is performed in accordance with the requirements of Regulatory Guide 1.35 (see Appendix 3A). Details of this program are presented in the Technical Specifications.

3.8.1.8 Assessment of Design Criteria

A review of Sections 3.8.1.1 through 3.8.1.7 and of the ASME Code [28], Article CC-3000, indicates that most likely areas for significant differences in design criteria would be in Section 3.8.1.3.1.3, "Load Combinations," and in Section 3.8.1.5, "Structural Acceptance Criteria," as it relates to membrane and bending stress conditions in the Reactor Building. These 2 areas are addressed by Items 1 and 2, below. Finally, specific differences are investigated at 4 locations in the Reactor Building. RN 99-101

1. Load Combinations

   a. Comparison

      The load combinations appearing in ASME Code Table CC-3230-1 are compared to the load combinations presented by Table 3.8-1.

      (1) The Test load combination of the ASME Code is represented by load combination 2 of Table 3.8-1.

      (2) The Construction load combination of the ASME Code is represented by load combination 1 of Table 3.8-1, except that the design wind, W, is not included. However, the design wind, W, does appear in the Construction load combination 1 of Table 3.8-2, which is applicable to the free standing liner. For the concrete Reactor Building structure, inclusion of the design wind, W, in the Construction load combination,

with the requirement of a Service condition, is not expected to control design. Consequently, the difference between Section 3.8.1 and the ASME Code is insignificant.

(3) The Normal load combination of the ASME Code is represented by load combination 3.a of Table 3.8-1.

(4) The Service Severe Environmental load combination of the ASME Code which includes E_o is not represented. This is investigated later. The ASME Code Service Severe Environmental load combination including W is not represented. However, the load combination including W is expected to be less severe than that including E_o.

(5) The Factored Severe Environmental load combination of the ASME Code which includes $1.5E_o$ is effectively represented by load combination 5.a of Table 3.8-1 because E' equals 1.5E equals $1.5E_o$. However, P_v (denoted as Z in Table 3.8-1) is not included in load combination 5.a of Table 3.8-1. This is investigated later.

The Factored Severe Environmental load combination of the ASME Code which includes 1.5W is not represented. However, this load combination is expected to be less significant than either the ASME load combination including $1.5E_o$ or the tornado wind load combination (load combination 5.b) of Table 3.8-1.

(6) The Factored Extreme Environmental load combination of the ASME Code which includes E_{ss} is represented by load combination 5.a of Table 3.8-1. However, P_v is not included in load combinations 5.a of Table 3.8-1. The addition of P_v tends to increase concrete compressive stresses and liner strains. However, since E' equals 1.5E equals $1.5E_o$, investigation of this ASME Code load combination is addressed by Item (5), above.

The Factored Extreme Environmental load combination of the ASME Code which includes W_t is represented by load combination 5.b of Table 3.8-1.

(7) The Factored Abnormal load combination of the ASME Code which includes 1.5P is represented by load combination 4.a of Table 3.8-1.

The Factored Abnormal load combination of the ASME Code which includes $1.25R_a$ is not represented due to the absence of a 1.25 factor on R in the load combinations of Table 3.8-1. Consideration of R_a would be of potential significance only in the evaluation of localized areas at high energy piping penetrations rather than the overall design of the Reactor Building shell.

(8) The first 2 Factored Abnormal/Severe Environmental load combinations of the ASME Code are represented by load combination 4.b of Table 3.8-1. The third ASME Code load combination is not represented in the load combinations of Table 3.8-1. This is investigated later.

(9) The Factored Abnormal/Extreme Environmental load combination of the ASME Code is represented by load combination 5.a of Table 3.8-1.

b. Conclusions

Based upon the above comparisons of the ASME Code and Section 3.8.1 load combinations, the following ASME Code load combinations would be candidates for investigation of compliance with the ASME Code flexural and membrane stress/strain acceptance criteria:

(1) Service - Severe Environmental

$D + L + F + T_o + E_o + R_o + P_v$

(2) Factored - Severe Environmental

$D + 1.3L + F + T_o + 1.5E_o + R_o + P_v$

(3) Factored - Extreme Environmental

$D + L + F + T_o + E_{ss} + R_o + P_v$

(4) Factored - Abnormal/Extreme Environmental

$D + L + F + T_o + E_o + 0.5E_{ss} + W + H_a$

Since E_{ss} equals $1.5E_o$ and since the live load, L, is inconsequentially small relative to other loads appearing in the above load combinations, it is necessary only to investigate either one of the Factored load combinations, under items (2) and (3), above. The Factored Extreme load combination, Item (3), above, is selected.

In the load combination of Item (4), above, H_a has a value of zero.

2. Structural Acceptance Criteria

   a. Comparison

      The criteria which would limit Reactor Building membrane and bending stress conditions are compared.

      (1) Concrete

          The most significant differences between the ASME Code and Section 3.8.1.5 appear in the allowable compressive stresses as shown by Table 3.8-16.

      (2) Reinforcing Steel

          The allowable stresses and strains of Section 3.8.1.5 are more conservative than those of the ASME Code as shown by Table 3.8-17.

      (3) Liner

          As noted in Section 3.8.1.5.2.1, the ASME Code allowables of ASME Code Table CC-3720-1 are used.

   b. Conclusions

      From the above comparisons it is determined that the Section 3.8.1.5 acceptance criteria for some compressive stresses are not as stringent as the ASME Code criteria. These include Service primary membrane stresses, Service primary membrane plus bending stresses, and Factored primary membrane stresses. Consequently, it is necessary to investigate the FSAR load combinations which produce large compressive stresses in addition to the ASME Code load combinations previously identified (see 1.b, above).

      The total load combinations to be investigated, along with the allowable concrete compressive stresses, are presented by Table 3.8-18. These allowable stresses depend upon the conditions appearing in ASME Code Table CC-3136.5-1.

3. Investigation of Reactor Building at Four Locations

   Tables 3.8-19 through 3.8-22 show the predicted Reactor Building concrete stresses and linear strains at 4 locations. Also shown are the corresponding ASME Code allowables. It should be noted that all predicted values are less than the allowables.

Concrete compressive stresses for those load combinations including T_o are conservatively large because a cracking reduction of the thermal stresses was not used.

4. Summary and Conclusion

The assessment of the Virgil C. Summer Nuclear Station Reactor Building has consisted of:

a. Comparing the ASME Code load combinations with those appearing in Table 3.8-1 of the FSAR. From this comparison the ASME Code load combinations which differ from those in Table 3.8-1 were identified.

b. Comparing the ASME Code structural acceptance criteria governing membrane and flexural stress conditions in the Reactor Building shell with those appearing in Section 3.8.1.5. From this comparison the ASME Code allowable concrete compressive stresses were identified as being generally more stringent than those in Section 3.8.1.5.

c. Identifying those load combinations in Table 3.8-1 which would be impacted by the more stringent ASME Code allowables.

d. For the load combinations identified as described in items a and c, above, determining at 4 locations on the Reactor Building shell, the flexural and membrane stresses in the concrete and the stresses and strains in the liner.

e. Comparing these stresses and strains with the ASME Code allowables, in Tables 3.8-19 through 3.8-22.

A review of Tables 3.8-19 through 3.8-22 indicates that the Virgil C. Summer Nuclear Station Reactor Building design is conservative with respect to a design based upon the use of the ASME Code and the Standard Review Plan. It is noted that minimum specified values for material properties, rather than actual material properties, were used in this evaluation.

3.8.2 STEEL CONTAINMENT

The Virgil C. Summer Nuclear Station design does not employ a steel containment structure.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

3.8.3.1 Description of the Internal Structures

The internal structures of the Reactor Building consist of the following:

1. Primary shield wall surrounding and supporting the reactor vessel.

2. Secondary shield walls, surrounding and laterally supporting the steam generator.

3. Refueling cavity and fuel transfer canal.

4. Mezzanine floor, at elevation 436' and an operating floor, at elevation 463', both consisting of 2 foot thick concrete slabs supported by structural steel framing.

5. Polar crane supports.

6. Concrete basement slab (4 feet thick) at elevation 412' is supported by the structural foundation mat.

The general arrangement of these structures is shown by Figures 3.8-39 through 3.8-42 and the structures are described below.

3.8.3.1.1 Reactor Support System

The reactor vessel supports are described in Section 5.5.14.

3.8.3.1.2 Steam Generator Support System

The steam generator supports are described in Section 5.5.14.

3.8.3.1.3 Reactor Coolant Pump Support System

The reactor coolant pump supports are described in Section 5.5.14.

3.8.3.1.4 Pressurizer Support System

The pressurizer supports are described in Section 5.5.14.

3.8.3.1.5 Primary Shield Wall

3.8.3.1.5.1 Description

The primary shield wall is a reinforced concrete wall surrounding and supporting the reactor vessel. The wall varies in thickness as shown by Figure 3.8-43. The inside diameter is approximately 16 feet. The lower portion of the wall, below the base slab (elevation 412') is a 4 foot thick cylindrical section surrounded by structural foundation mat concrete. Vertical support of the primary shield wall is provided by the Reactor Building foundation mat.

The inside of the primary shield wall is basically circular throughout its height. The outside is circular from the structural foundation mat to the top of the basement slab where it becomes polygonal to accommodate surrounding structures, such as secondary shield walls, fuel transfer canal slab, and fuel canal walls.

Between the base slab (elevation 412') and the fuel transfer canal (elevation 437'-2-1/2") steel assemblies are embedded into the primary shield wall. Their functions are described in Section 3.8.3.1.5.2. The general arrangement of these steel assemblies is shown in Figure 3.8-43.

3.8.3.1.5.2 Primary Shield Wall and Embedded Steel Assemblies

Embedded steel assemblies provide support for the reactor vessel support system, provide pipe rupture restraint for the reactor coolant piping, and restrict the buildup of pressure and temperature on the primary shield wall and on the reactor vessel, should a loss of coolant accident (LOCA) occur.

The functions performed by the steel assemblies are described in detail below:

1. Sleeves around Reactor Coolant Piping

   Each reactor coolant pipe between the reactor vessel and steam generator passes through a double walled sleeve (displacement restrictor) embedded in the primary shield wall (see Figure 3.8-44). The double walled sleeves consist of 2 cylindrical sleeves interconnected with radial ribs. Reactor coolant pipe break displacement limiters are attached to the inside of the inner sleeve. The annular space between the 2 sleeves functions as an air flow duct during normal operation, limiting the surrounding concrete temperature. See Figures 3.8-45 and 3.8-46.

2. Baffle Assemblies

Baffle assemblies are installed around the primary coolant pipes at the inside face of the primary shield wall. These assemblies limit effluent flow into the reactor cavity from a postulated pipe break. Each baffle directs the flow into the double walled sleeve and into the inservice inspection hatch. The baffles are internally ribbed steel box structures embedded in the primary shield wall. See Figures 3.8-47 and 3.8-48.

The baffle assemblies are interconnected circumferentially. This circumferentially shaped assembly transmits postulated pipe break effluent loads to the primary shield wall.

3. Anchor Assemblies under Reactor Vessel Supports

Anchorage assembly embedments are provided under each reactor vessel support to transfer loads from the reactor vessel support to the primary shield wall. See Figures 3.8-43, 3.8-49, and 3.8-50.

Each assembly functions as a bearing shear lug assembly supported by wide flange sections cast into the primary shield wall. The anchorage assembly attachments transfer vertical, radial, and tangential loads from the reactor vessel support to the primary shield wall.

4. Neutron Detector Block Outs

Eight (8) blockouts are provided at the inside periphery of the primary shield wall to accommodate installation and inspection of neutron detector equipment around the reactor vessel. See Figures 3.8-43, 3.8-44, and 3.8-51.

3.8.3.1.6 Secondary Shield, Refueling Cavity, and Fuel Transfer Canal Walls

The secondary shield wall forms 3 compartments which are located adjacent to and are connected with the primary shield wall. These compartments are of polygonal shape in plan and form enclosures for the reactor coolant system equipment.

The function of these enclosures is to protect the Reactor Building from the effects of a postulated pipe break, provide biological shielding, and provide lateral support for the reactor coolant system equipment. The pressurizer is enclosed in a separate compartment.

The refueling cavity/fuel transfer canal is located above and adjacent to the reactor vessel and is a stainless steel lined reinforced concrete structure. The walls of the refueling cavity/fuel transfer canal form part of the secondary shield wall system. For details of the secondary shield wall and the refueling cavity/fuel transfer canal, see Figures 3.8-39 through 3.8-42.

3.8.3.1.7 Operating and Mezzanine Floors

The operating floor slab at elevation 463' and the mezzanine floor slab at elevation 436' are supported by a structural steel framing system and the secondary shield walls. The inner edge of these reinforced concrete slabs is keyed and doweled into the secondary shield walls. The outer edge stops short of the Reactor Building liner to provide separation from the Reactor Building.

The structural steel framing system consists of radially oriented girders supported on the inboard ends by concrete corbels or steel brackets attached to the secondary shield wall. The outboard ends are supported by steel columns. The columns are spaced along a 59 foot radius, with base plates supported on piers cast in the basement floor slab at elevation 412'. Perimeter girders between the columns support the outboard portion of the operating and mezzanine floor slabs. Steel beams frame between the girders to support the floor slabs. Plans and sections showing the structural steel framing system are shown by Figures 3.8-39 through 3.8-42.

3.8.3.1.8 Polar Crane Supports

The polar crane is supported by perimeter runway girders around the Reactor Building at elevation 552' approximately (top of rail). The circular crane rail is attached to the top flange of the perimeter runway girders. The runway girders include a weldment to engage the polar crane seismic uplift lugs. The runway girders are shown by Figures 3.8-21 and 3.8-22.

3.8.3.1.9 Concrete Basement Slab

A 4 foot thick concrete basement slab, located at elevation 412', is supported by the Reactor Building foundation mat. This slab supports and anchors all the internal structures and equipment. The Reactor Building base mat liner is not penetrated for anchorage of any internal structure or equipment.

3.8.3.2 Applicable Codes, Standards, and Specifications

3.8.3.2.1 General

Structural design and materials for the internal concrete and steel structures conform to the following documents unless noted otherwise herein:

1. "Southern Standard Building Code," 1969 Edition.

2. American Concrete Institute, "Building Code Requirements for Reinforced Concrete," ACI 318-71.

3. American Concrete Institute, "Specification for Structural Concrete for Buildings," ACI 301-72, revised 1975.

4. American Institute of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," February 12, 1969.

5. American Institute of Steel Construction, "Code of Standard Practice for Steel Buildings and Bridges," July 1, 1970.

6. American Welding Society, "Structural Welding Code," D1.1-72.

7. Regulatory Guides, as discussed in Appendix 3A and listed below:

    a. Regulatory Guide 1.12, "Instrumentation for Earthquakes."

    b. Regulatory Guide 1.29, "Seismic Design Classification."

    c. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

8. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria 2 and 4.

3.8.3.2.2 Concrete

The following codes, standards, and specifications are used to establish the properties of concrete and to control concrete placement:

1. American Concrete Institute

    a. "Specification for Structural Concrete for Buildings," ACI 301-72, revised 1975.

    b. "Building Code Requirements for Reinforced Concrete," ACI 318-71.

    c. "Mass Concrete for Dams and Other Massive Structures," ACI Title No. 67-17, a report by ACI Committee 207.

2. American Society for Testing and Materials

   a. "Standard Method of Test for surface Moisture in Fine Aggregate," C70-73.

   b. "Standard Method of Test for Specific Gravity and Absorption of Coarse Aggregate," C127-73.

   c. "Standard Method of Test for Specific Gravity and Absorption of Fine Aggregate," C128-73.

   d. "Standard Method of Test for Length Change of Hardened Cement Mortar and Concrete," C157-75.

   e. "Standard Method of Test for Thermal Conductivity of Materials by Means of the Guarded Hot Plate," C177-71.

   f. "Standard Method of Test for Total Moisture Content of Aggregate by Drying," C566-67.

   g. "Standard Method of Test for Chloride Ion in Water and Waste Water," D512-67.

   h. "Standard Method of Test for Sulfate Ion in Water and Waste Water," D516-68.

   i. ASTM C109-08, Standard Test Method for Compressive Strength of Hydraulic Cement Mortars (Using 2-in Cube Specimens).

   j. ASTM C1107-11, Standard Specification for Packaged Dry Hydraulic-Cement Grout (Nonshrink). RN 14-016

3. U.S. Army Corps of Engineers

   a. "Method of Test for Coefficient of Linear Expansion of Concrete," CRD-C39-55.

   b. "Method of Test for Flow of Grout Mixtures (Flow Cone Method)," CRD-C79-58.

   c. "Method of Test for Flat and Elongated Particles in Coarse Aggregate," CRD-C119-53.

   d. "Methods of Sampling and Testing Expansive Grouts," CRD-C589-70.

   e. CRD C621-93, Standard Specification for Packaged Dry Hydraulic-Cement Grout (Nonshrink). RN 14-016

4. Concrete Plant Manufacturers Bureau, "Concrete Plant Standards," Revision 5, March, 1973.

5. Truck Mixer Manufacturers Bureau, "Truck Mixer and Agitator Standards," Revision 9, November, 1971.

6. Regulatory Guide 1.55, "Concrete placement in Category 1 Structures," (see Appendix 3A).

3.8.3.2.3 Reinforcing Steel

The following codes and specifications are used to establish the properties of and to control the fabrication and placement of reinforcing steel:

1. American Society for Testing and Materials

    a. "Standard Specification for Deformed Billet Steel Bars for Concrete Reinforcement," ASTM A615-72.

    b. "Standard Methods and Definitions for Mechanical Testing of Steel Products," ASTM A370-72.

2. American Concrete Institute

    a. "Specification for Structural Concrete for Buildings," ACI 301-72, revised 1975.

    b. "Building Code Requirements for Reinforced Concrete," ACI 318-71.

3. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Division 2, 1975.

3.8.3.2.4 Structural Steel

Codes and specifications for design, fabrication, and erection of structural steel are as stated in Section 3.8.3.2.1, items 5 and 6.

3.8.3.3 Loads and Load Combinations

3.8.3.3.1 Load Definitions

Loads used in the design of the interior concrete and steel structures are defined as follows:

1. Normal Loads

    a. D - Dead load of the structure, including any permanent equipment loads.

    b. L - Live loads, including any movable equipment loads which may vary in intensity.

    c. T_o - Thermal effects and loads under normal operating or shutdown conditions.

    d. R_o - Pipe reactions under normal operation or shutdown conditions.

    e. F - Hydrostatic loads due to pressure of liquid under shutdown conditions.

2. Severe Environmental Loads

    E - Loads generated by an OBE

3. Extreme Environmental Loads

    E' - Loads generated by an SSE

4. Abnormal Loads

    a. P_a - Pressure equivalent static load within or across a compartment due to a pipe break, including dynamic factors.

    b. T_a - Thermal loads due to thermal conditions occurring during a pipe break.

    c. R_a - Pipe reactions due to thermal conditions during a pipe break.

    d. Y_r - Equivalent static load due to the reaction on a broken pipe during a postulated break, including dynamic load factors, except where time history analysis is performed.

    e. Y_j - Jet impingement equivalent static load due to the jet from a broken pipe during a postulated break, including dynamic load factors.

f. Y_m - Missile equivalent static load due to impact of any postulated missile, including dynamic load factors.

3.8.3.3.2 Design Loads

1. Normal Loads

   a. Dead Load, D

   A reinforced concrete density of 150 lb/ft^3 is used. Density of structural steel is 490 lb/ft^3. Permanent equipment loads and cable tray loads are used in the design.

   b. Live Loads, L

   The live loads considered in the design are the weight of any equipment or systems not permanently attached, plus an allowance for transient loads during construction, operation, or maintenance.

   c. Thermal Effects, T_o

   Temperature induced forces are evaluated for the various sections of the interior structures based upon the most adverse temperature gradients predicated upon normal plant operation temperature ranges (Refer to Section 6.2.1).

   d. Pipe Reactions, R_o

   Pipe Load reactions are considered, as appropriate, for the design of the internal structures. Thus reactions are considered as live loads.

   e. Hydrostatic Loads, F

   Hydrostatic loads are based upon a water density of 62.4 lb/ft^3.

2. Severe Environmental Loads, E

   Seismic design of the internal structures is based upon the response to OBE ground accelerations as described in Section 3.7.

3. Extreme Environmental Loads, E'

   Seismic design of the internal structures is based upon the response to SSE ground accelerations as described in Section 3.7.

4. Abnormal Loads

a. Design Accident Pressure, P_a, and Temperature, T_a

The various components of interior structures are designed for the worst applicable postulated accident conditions. These conditions vary with different components and are dependent upon the postulated accident; for a discussion of the accident types investigated, see Section 6.2.1.

b. Pipe Break Effects, R_a, Y_r, Y_j

The reactions from pipe break are considered in the design of the interior structures where applicable. See Section 3.6.

c. Missile Impact, Y_m

The effect of interior missile impact on the internal structures is considered (see Section 3.5).

3.8.3.3.3 Load Combinations

3.8.3.3.3.1 Interior Concrete Structures

The internal concrete structures, including primary shield walls, secondary shield walls, and floor slabs are designed for the load combinations presented in Table 3.8-10.

3.8.3.3.3.2 Structural Steel Structures

The floor framing system and polar crane runway girders are designed for the load combinations presented in Table 3.8-11.

3.8.3.4 <u>Design and Analysis Procedure</u>

3.8.3.4.1 Reactor Coolant System Supports

Description of the models and analytical methods, loads and loading combinations, and allowable limits for the Reactor Coolant System supports are given in Section 5.2.1.10.

3.8.3.4.2 Primary Shield Wall

3.8.3.4.2.1 Structural Model

The primary shield was analyzed in 2 phases:

1. The entire wall from the base (elevation 387'-8") to the top (elevation 437'-2-1/2") was analyzed as an axisymmetric thick cylinder using the finite element computer program ELAD for axisymmetric solids of revolution (see Section 3.8.4.4).

2. Due to major penetration discontinuities and the restraint effects of the secondary shield walls above the basement slab, that portion of the primary shield wall was modeled using the finite element computer program MRI/STARDYNE (see Section 3.8.4.4) with thick element criteria.

The boundary condition assumed for the first phase analysis at the basement floor slab and the structural base mat was one of radial restraint. Design loads were applied at their point of application to the primary shield wall and results were combined in accordance with the load combinations given in Table 3.8-10.

The structural boundary at the top of the basement floor slab (elevation 412') was assumed to be fixed for the second phase analysis. The effects of the secondary shield walls, fuel transfer canal walls, and fuel transfer canal slab were modeled as displacement boundaries. Actual penetrations were modeled. Loads were applied individually at their point of application and results were combined in accordance with the load combinations given in Table 3.8-10.

The structural behavior predicted by both models at the assumed boundary of the second phase model (top of basement slab) was found to be compatible. The resultant stresses from both models were used to determine the concrete and reinforcing steel requirements in accordance with the strength design method given in ACI 318-71.

3.8.3.4.2.2 Application of Loads and Analytical Procedure

The elastic analysis was performed by applying the basic loads listed in Section 3.8.3.3. These loads were applied separately and the resultant forces obtained from each analysis are combined with appropriate load factors according to the load combinations listed in Table 3.8-10.

From the results of these combinations, the maximum stress resultants to which an element under investigation is subjected were determined.

The reinforcement area was then calculated according to ACI 318-71 using the strength design method. Earthquake, temperatures, and accident loads were treated as follows:

1. Earthquake

   The accelerations of the primary shield wall in horizontal and vertical directions at different elevations of the wall were calculated as described in Section 3.7.

   The absolute value of the seismic forces was added to the applicable load combinations.

2. Thermal Loads

   Thermal gradients were generated from the time history studies of the accident conditions (see Section 6.2.1). The thermal gradients occurring at various hours after startup, shutdown, and accident were used as input in the analyses. The forces obtained from these analyses were superimposed separately on load combinations to determine the maximum effect.

3. Loss of Coolant Accidents

   a. Pressure Loads

      A postulated guillotine break at the weld connecting the reactor coolant pipe with the reactor vessel nozzle for the cold leg (see Figures 3.8-43 and 3.8-44) was considered as described in Chapter 6. The reactor coolant released by this postulated break is directed toward the steam generator compartment through the penetration and toward the refueling cavity through the inservice inspection hatch due to the baffle assembly flow restriction as described in Section 3.8.3.1.5.1. The flow back into the primary cavity is limited by the narrow gap between the nozzle and the steel baffle assembly surrounding the nozzle.

      The pressure in the annulus of the primary shield wall caused by the limited back flow of reactor coolant is maximum at the postulated break location and decreases away from it, creating a spatial distribution of accident pressure which is applied to the inside of the primary shield wall structure model. The pressure buildup in the penetration at the location of the postulated break is also applied simultaneously with the reactor cavity pressures. The transient pressure includes a dynamic factor for the static input on the structural model.

   b. Pipe Reactions and Jet Forces

      The forces associated with postulated reactor coolant pipe ruptures, such as pipe reactions and jet forces are also considered. The reactor coolant pipes pass through double walled steel sleeves embedded in the primary shield wall. Lateral movement of the pipe in the event of an accident is restricted by shims between the pipe periphery and the sleeve inner wall along the passage of the pipe through the primary wall thickness. Due to this restriction of lateral movement of the pipe in the event of a rupture near reactor vessel nozzle, the loss of reactor coolant and the dynamic reaction of the pipe are limited. The restricted flow also limits the jet impingement force. Axial movement of the pipe is limited by the reactor coolant loop geometry. The vertical reaction of the hot leg pipe in case of a break at the steam generator elbow is taken by a vertical support column located outside the primary shield wall. Lateral movement of the hot leg pipe in case of a break is resisted by shims between the pipe periphery and the sleeve in the wall. Lateral and vertical movement of the cold leg pipe is resisted by shims as described above. The loads transmitted to the primary shield wall from the reactor vessel supports due to accident, thermal, and seismic effects are applied to the model separately and are combined in accordance with Table 3.8-10.

3.8.3.4.3 Secondary Shield Walls, Refueling Cavity Walls, Fuel Transfer Canal Walls and Slab, and Mezzanine and Operating Floor Slabs

The interior structures were modeled using the finite element technique with the computer program MRI/STARDYNE (see Section 3.8.4.4). A composite finite element model was generated which included all of the significant interior structures, including the primary shield wall, secondary shield wall, fuel transfer canal walls and slab, the adjacent mezzanine and operating floor slabs, the 4 foot thick basement floor slab, and steel columns. Due to the completeness of the model, the only boundary condition was the contact surface between the basement floor slab and the Reactor Building structural foundation mat which was simulated by springs. The spring constants were determined in accordance with the half space theory.

The element size and number of nodes were determined by parametric studies and the ASCE publication, "Guidelines for Finite Element Idealization." Loads were applied to the model individually at their point of application and the results were combined in accordance with Table 3.8-10. Associated pipe break loads were applied to the model using a dynamic load factor. Based upon the results of the combined stresses, the concrete was checked against allowable stresses and the area of reinforcing required was determined utilizing the strength design method given in ACI 318-71. Horizontal shears were reacted at the basement slab through friction and by bearing of the embedded portion of the primary shield wall below the basement slab. Local effects of jet impingement forces were investigated using yield line theory.

3.8.3.4.4 Interior Structural Steel (Operating and Mezzanine Floor Framing)

The mezzanine and operating floors are supported by beam and girder steel framing. The girders are supported by the secondary shield walls and outboard perimeter columns as described in Section 3.8.3.1.7.

The concrete slabs are designed to transfer horizontal forces to the secondary shield walls.

The structural steel is designed for the temperature increase caused by the postulated accidents. The thermal gradient between the structural steel frame and the concrete slab was incorporated in this design. Structural steel is designed for the pipe break pressure, jet forces, and LOCA differential pressures by application of a dynamic load factor to the resultant loads.

The pipe reactions associated with pipe breaks were determined by a dynamic analysis of the affected structural steel. Hand calculations were used for the design, except for column design where computer program S027 was used. (See Section 3.8.4.4).

3.8.3.4.5 Polar Crane Supports

The polar crane runway girders are designed as simply supported beams spanning between the brackets described in Section 3.8.1.

Crane wheel loads, determined for the various loads and load combinations listed in Table 3.8-3, are applied to the polar crane runway girders at different locations to determine maximum shears, moments, and reactions.

Girders are provided with slotted holes in one end to allow thermal growth under operating and accident conditions. Rail splices are also provided with gaps and slotted holes. Polar crane wheels are designed to allow for expansion of polar crane runway girders.

Vent openings are provided in the girders to prevent collapse during pressurization of the Reactor Building.

3.8.3.4.6 Basement Floor Slab

The 4 foot thick basement floor slab at elevation 412' was analyzed in conjunction with the analysis of the secondary shield wall. The model used is discussed in Section 3.8.3.4.3. Loads were applied independently at their point of application in the entire model and results were combined in accordance with the load combinations given in Table 3.8-10. Associated pipe break loads were applied to the model using a dynamic load factor. The total horizontal shear was reacted by the embedded portion of the primary shield wall in bearing and friction between the basement slab and the Reactor Building liner and base mat.

The entire interior structure, including the basement slab, was analyzed to determine adequate stability. Consideration of stiffness of the primary shield wall, secondary shield walls, and basement slab was included in the overturning analysis.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reactor Coolant System Supports

The loads and loading combinations considered in the analysis of the Reactor Coolant System supports and allowable stress criteria are given in Section 5.2.1.10.

3.8.3.5.2 Interior Concrete Structures

Allowable stresses were maintained within the limits shown in Table 3.8-10.

3.8.3.5.3 Steel Structures and Supports

Allowable stresses were maintained within the limits shown in Table 3.8-11.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Concrete

Information regarding interior structure concrete materials, quality control, and special construction techniques is identical to that presented in Section 3.8.1.6.1. The compressive strength of the concrete is 5000 psi at 90 days.

3.8.3.6.2 Reinforcement

Information regarding interior concrete structure reinforcement materials, quality control, and special construction techniques is the same as that presented in Section 3.8.1.6.2.

3.8.3.6.3 Structural Steel

Materials for interior steel structures conform to the following requirements:

1. Structural Steel Shapes and Plates

    a. ASTM A36-70a

    b. ASTM A572-70a, Grade 50

2. High Strength Bolts

    ASTM A325-71

3. Ordinary Bolts

    ASTM A307-68, Grade A

4. Welding Rods

   AWS D1.1-72, Type E7015, E7016, and E7018

5. Concrete Anchor Studs

   ASTM A108, Grades 1010, 1015, 1017, or 1020

6. Anchor Bolts

   ASTM A36-70a

7. Nuts for Anchor Bolts

   ASTM A307-68, Grade A

8. Polar Crane Rails

   ASTM A1-68a

9. Embedded Steel Assemblies

   a. Reactor Vessel Support Assemblies

      (1) ASTM A302-B plate steel with supplemental requirements S1, S3, and S4, and ultrasonic inspection for internal discontinuities in accordance with ASTM A578, acceptance level 1.

      (2) ASTM A588 wide flanges and plates

      (3) ASTM A441 bar

   b. Neutron Detector Liner Boxes

      (1) ASTM A302-B plate steel with supplemental requirements S1, S3, and S4, and ultrasonic inspection for internal discontinuities in accordance with ASTM A578, acceptance level 1.

      (2) SAE C-1012 threaded coupling, 9-1/2 inch OD (non-nuclear safety class).

      (3) ASTM A53 or ASTM A106 pipe (non-nuclear safety class).

c. Reactor Cavity Liners

   (1) The liner serves as a concrete form and provides close tolerances for optimum air passage around the reactor vessel ASTM A302-B plate steel with supplemental requirements S1, S3, and S4, and ultrasonic inspection for internal discontinuities in accordance with ASTM A578, acceptance level 1. Alternatively, ASTM A709, Grade 36, as required.

   (2) ASTM A588 angles

d. Pipe Sleeve Assemblies

   (1) ASTM A302-B plate steel with supplemental requirements S1, S3, and S4, and ultrasonic inspection for internal discontinuities in accordance with ASTM A578, acceptance level 1.

   (2) Headed concrete anchor studs, 3/4 inch by 3-3/8 inch.

   (3) Headed concrete anchor studs, 7/8 inch by 7-3/16 inch.

   (4) ASTM A36 threaded rod, 3/8 inch (non-nuclear safety class).

   (5) ASTM A213, type 316 stainless steel tubing, 1/2 inch OD (non-nuclear safety class).

   (6) ASTM A36 angle, 1-1/4 inch by 1-1/4 inch by 1/8 inch (non-nuclear safety class).

e. Cover Plates

   (1) ASTM A302-B plate steel with supplemental requirements S1, S3, and S4, and ultrasonic inspection for internal discontinuities in accordance with ASTM A578, acceptance level 1.

   (2) ASTM A307 bolts, 3/4 inch diameter (non-nuclear safety class).

   (3) ASTM A588 angle, 4 inch by 4 inch by 1/2 inch

   (4) Threaded studs, 3/4 inch diameter

   (5) Headed concrete anchor studs, 3/4 inch by 3-7/8 inch

Material and quality control requirements conform to AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1969 Edition.

Fabrication and erection of structural steel, installation and inspection of high strength bolts, and welding and welding inspection are in accordance with ANSI N45.2.5 [25].

3.8.3.7 Testing and Inservice Inspection Requirements

Except for testing required to comply with material specifications, no testing or inservice inspection of internal structures is planned.

3.8.4 OTHER SEISMIC CATEGORY 1 STRUCTURES

3.8.4.1 Description of the Structures

3.8.4.1.1 General

With the exception of the Service Water Pumphouse, intake and discharge structure, the Seismic Category 1 structures surround the Reactor Building. These structures are seismically independent of the Reactor Building and of each other. However, the Control and Intermediate Building foundations are tied together, while the superstructures are separated (see Figure 3.8-52).

The separation of the structures, typical joints at foundation level, and typical floor and roof levels are indicated by Figure 3.8-52. The relative motions of adjacent Seismic Category 1 structures at critical elevations were determined to establish the required seismic separation.

The Non-Seismic Category 1 Hot Machine Shop and Turbine Building are designed to withstand earthquake loads and tornado wind loads to the extent required for prevention of damage to Seismic Category 1 structures. General materials of construction are concrete with a compressive strength of 3000 psi; ASTM A615, grade 60, reinforcing steel; and ASTM A36 structural steel.

Both rubber (styrene-butadiene) and polyvinyl chloride waterstops are used. Polyvinyl chloride waterstop locations and expected radiation levels are listed in Table 3.8-11a.

3.8.4.1.2 Auxiliary Building

The Auxiliary Building is a reinforced concrete shear wall (box type) structure with 6 main floor levels. It is approximately 110 feet wide by 190 feet long. The lowest level, a sub-basement, is located 61 feet below grade and the roof level is 50 feet above grade. The roof and exterior walls are of 2 foot thick reinforced concrete designed to prevent damage to safety-related equipment areas from tornado missiles and their effects. The floors are reinforced concrete supported on concrete walls. Areas immediately adjacent to the Reactor Building utilize steel framing and concrete slabs tied into the adjacent Auxiliary Building concrete walls.

A Non-Seismic Category 1 structure of steel frame, metal siding, and metal roof deck is located on the roof of the Auxiliary Building. The basic details of this steel structure are provided by Figures 3.8-72 and 3.8-73. This structure is classified as Non-Seismic Category 1 because the equipment contained in it is nonsafety related, primarily HVAC components, such as plenums and fans, as well as a number of electrical panels. Generally, this equipment is bolted to the slab at Elevation 485'-0". This structure was designed to withstand earthquake forces, based upon the "Uniform Building Code," 1973 Edition, and 100 mph design wind loads. The elastic method of analysis was used for the design. The loads are applied to the braced frame structure, using conventional frame analysis methods. The steel roof deck is used as a diaphragm to transfer the lateral loads due to earthquake or wind to the braced frame and supports. Load combinations and acceptance criteria are as stated in Table 3.8-14, Section 1.a, with a 1/3 increase in allowable stresses for seismic and wind loads in accordance with Reference [18], where applicable. In addition, this Non-Seismic Category 1 structure is designed to withstand the SSE and tornado wind. The elastic method of analysis was used and loading combinations are as stated in Table 3.8-14, Section 2.a. The acceptance criterion used is that forces in the structure do not exceed the ultimate capacity of the structure. This method is considered conservative in comparison with a full plastic analysis.

The steel roof structure has been checked for its response to missile impact. Specifically, calculations indicate that impact by the design missile, a steel rod, with a diameter of 1 inch, a length of 3 feet, and weight of 8 pounds, cannot render the structure, or any portion of the structure, unstable. Impact by this missile cannot cause a structural steel component to become detached from the overall structure such that a secondary missile is generated. Potential impact by a utility pole was not considered for this particular structure since the base of the steel, at Elevation 485'-0", is more than 30 feet above plant grade, at Elevation 435'-0".

Demonstration of the capability of the structure to withstand impact of the indicated missile satisfies the provisions of Standard Review Plan, Section 3.5.1.4, Paragraph III-4.

The Seismic Category 1 portion of the Auxiliary Building has a roof and external walls of 2 foot thick reinforced concrete designed to prevent damage to safety related equipment areas from tornado missiles and tornado missile effects. This design was developed in accordance with ACI 318-71 and ACI 349, using the loads and load combinations of Table 3.8-12. Therefore, the Seismic Category 1 portion of the Auxiliary Building will not be damaged by either an earthquake or a tornado.

The Auxiliary Building is separated from other buildings by a space to prevent load transfer during an OBE or SSE. See Figures 3.8-53 and 3.8-54 for details.

3.8.4.1.3 Intermediate Building

The Intermediate Building is an L-shaped, box type structure with 2 main floors and a partial third floor. It is approximately 90 feet wide by 200 feet long. The lower level of the structure is 23 feet below grade. The low roof is 28 feet above grade and the high roof is 70 feet above grade. The 2 story area has steel columns and beams supporting concrete slabs. The 3 story area, approximately 84 feet by 30 feet in plan, on the west side of the structure consists of concrete walls, roof, and floor slabs. The 2 story structure houses the main steam and feedwater lines. The roof structure is designed in specific areas to provide pressure relief by means of steel sacrificial panels in case of a pipe rupture. Equipment or systems essential for safe shutdown are not located on the floor under these sacrificial panels. The exterior walls and roofs are reinforced concrete, designed to prevent tornado missile damage to Seismic Category 1 equipment. The Intermediate Building is separated from other buildings above the foundation to prevent load transfer during an OBE or SSE. See Figures 3.8-54 and 3.8-55 for details.

3.8.4.1.4 Diesel Generator Building

The Diesel Generator Building is a reinforced concrete structure approximately 67 feet long, 65-1/2 feet wide and 42 feet above grade, as shown by Figure 3.8-56. The general basement level is 8 feet below grade. The cable and piping pit area extends down to 35 feet below grade. The structure is founded in a caisson system, as described in Section 3.8.5.1.6.

The walls and roof of the Diesel Generator Building are designed to prevent damage to safety-related equipment from tornado missiles and their effects. Removable missile shields are provided in front of external equipment access openings. Labyrinth shields are provided at external access doors. The Diesel Generator Building is separated from other buildings to prevent load transfer during an OBE or SSE. See Figure 3.8-56.

3.8.4.1.5 Control Building

The Control Building is a steel framed structure with concrete exterior shear walls and a concrete roof. This building has 4 main levels and is approximately 84 feet wide by 140 feet long. The lower 3 levels are further divided into upper cable spreading and plenum areas and lower rooms. The basement floor is 23 feet below grade. The concrete roof is approximately 70 feet above grade. The exterior walls and roof are reinforced concrete designed to prevent tornado missiles from damaging safety-related equipment within the building. The main interior framing consists of steel columns, steel girders in the north-south direction, and steel beams in the east-west direction. Horizontal forces are transferred to the exterior walls by diaphragm action of the concrete floor slabs. The Control Building is separated from other buildings above the foundation by a space to prevent load transfer during an OBE or SSE. See Figures 3.8-55 and 3.8-57 for details. 02-01

3.8.4.1.6 Fuel Handling Building

The Fuel Handling Building is a steel frame superstructure founded on a reinforced concrete substructure. This building has 2 main floor levels and a roof. It is approximately 120 feet long by 68 feet wide. The lowest level of the structure is approximately 23 feet below grade. The roof is approximately 76 feet above grade.

The principle elements of the building are the stainless steel lined spent fuel pool, the cask loading pit, the decontamination area, excess liquid waste processing areas, the fuel transfer canal which connects the cask loading pit and spent fuel pool, the railroad access area (floor elevation 436'), and the new fuel storage area. The upper floor level consists of a concrete slab supported by concrete walls and structural steel framing. A 125 ton overhead traveling fuel handling crane which spans 62'-8" is provided. The main hook of the crane lifts and handles the spent fuel cask. The 125 ton main hoist incorporates a single-failure-proof design as discussed in Section 3.12.4.3. The auxiliary hook (15 ton capacity) lifts and handles new fuel and new fuel containers. The fuel handling crane is physically restrained from traveling over the spent fuel pool by bumper stops mounted on the crane rails (see Section 9.1) and by posts which are welded to the ends of the fuel handling crane runway beams. The Fuel Handling Building is separated from other buildings by a space to prevent load transfer during an OBE or SSE. See Figures 3.8-58 through 3.8-60 for details. RN 13-018

3.8.4.1.7 Service Water Pumphouse

The Service Water Pumphouse is a reinforced concrete building adjacent to the service water pond. The structure consists of a forebay and pump chamber which is approximately 54 feet by 51 feet. The pump chamber extends from elevation 390' to the operating floor at elevation 436'. A reinforced concrete superstructure and roof approximately 79 feet by 70 feet extends from operating floor to elevation 459'. The roof covers the forebay/pump chamber, the control area, and the discharge pipe pits. The main intake structure enters the forebay at elevation 390'. Monorails are provided on the underside of the roof structure. The concrete walls and roof are designed to protect Seismic Category 1 equipment from damage due to tornadoes and their effects. See Figures 3.8-61 and 3.8-62.

3.8.4.1.8 Service Water Intake Structure

The Service Water Intake Structure is a reinforced concrete rectangular box culvert with 2 reinforced concrete wing walls at the intake end. The structure is mostly buried within the West Embankment as shown by Figures 1.2-1 and 3.8-63. The portion which is not covered with soil is submerged within the service water pond. The function of the Service Water Intake Structure is to draw water from the service water pond into the Service Water Pumphouse. The connection between the Service Water Intake Structure and the Service Water Pumphouse is described in Section 3.8.5.1. The Service Water Intake Structure is designed to withstand loads applied under normal operating conditions, as well as under the extreme environmental conditions.

The Service Water Intake Structure (SWIS) is designed as a 2 dimensional structure in the transverse direction. The loads are primarily sustained by sections normal to the longitudinal axis. All load combinations and loads specified in Standard Review Plan (SRP) 3.8.4 for reinforced concrete strength design were satisfied.

The controlling load combination included dead, hydrostatic, and soil pressure loads for the unique conditions of the SWIS being dewatered, via stop logs, and the Service Water Pond being completely filled. Provisions for stop logs have been made at the intake end of the SWIS and at the entrance to the pump chamber.

A comparison of factored internal forces and moments to the acceptance limits of SRP 3.8.4 results in the most critical section having a strength which is 20% in excess of that required. The load combinations which included OBE and SSE result in the most critical section having a strength of 150% and 230% respectively in excess of those required.

As far as the longitudinal direction of the SWIS is concerned, the original design of the SWIS included longitudinal reinforcing steel for distribution of cracks associated with thermal and shrinkage effects. The original design did not include consideration of the differential settlement that occurred. Since the unanticipated settlement has occurred, and the cracks have been grouted, the longitudinal reinforcing steel now serves exactly the same purpose as originally intended.

The longitudinal direction could crack due to a seismic event. It is conservatively calculated using Newmark's theory that the actual maximum crack width that could occur due to an SSE is 0.16 inches. The safety margin in the longitudinal direction can only be based upon the degree of cracking and its effect on safe function of the SWIS. It was determined that the SWIS will function satisfactorily with a crack width of 1/2 inch. The 1/2 inch is a lower bound number, but nevertheless could be used to determine a safety margin by comparing it with the actual calculated crack width of 0.16 inches. On this basis, the safety margin of the structure in the longitudinal direction can be said to be at least 3.0.

3.8.4.1.9 Service Water Discharge Structure

The Service Water Discharge Structure is located on the southeast edge of the West Embankment of the service water pond. The discharge structure, as shown by Figure 3.8-64, is a reinforced concrete structure which consists of an expansion chamber and a sill with crest at elevation 414'. The crowns of two 30 inch diameter discharge pipes which terminate at the Service Water Discharge Structure are at elevation 412'-6". The pipes are connected to the Service Water Discharge Structure by flexible connections.

3.8.4.2 Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, and testing of Seismic Category 1 structures are covered by codes, standards, and guides which are applicable in whole or in part. A list of such documents is as follows:

3.8.4.2.1 General

1. Southern Standard Building Code, 1969 Edition.

2. American Concrete Institute, "Building Code Requirements for Reinforced Concrete," ACI 318-71.

3. American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures," ACI 349-76.

4. American Concrete Institute, "Specification for Structural Concrete for Buildings," ACI 301-72, revised 1975.

5. American Institute of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," February 12, 1969.

6. American Institute of Steel Construction, "Code of Standard Practice for Steel Buildings and Bridges," July 1, 1970.

7. American Welding Society, "Structural Welding Code," AWS D1.1-72.

8. American National Standards Institute, "Supplementary Quality Assurance Requirements for Installation and Testing of Concrete Structural Steel, Soils and Foundations during the Construction Phase of Nuclear Power Plants," ANSI N45.2.5.

9. Nuclear Regulatory Commission, Regulatory Guides, as discussed in Appendix 3A listed below:

    a. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

    b. Regulatory Guide 1.29, "Seismic Design Classification."

    c. Regulatory Guide 1.55, "Concrete placement in Category 1 Structures."

    d. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

    e. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants."

10. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria 2 and 4.

3.8.4.2.2 Concrete

The following codes, standards, and specifications are used to establish the properties of concrete and to control concrete placement:

1. American Concrete Institute

    a. "Specification for Structural Concrete for Buildings," ACI 301-72, Revised 1975.

    b. "Building Code Requirements for Reinforced Concrete," ACI 318-71.

    c. "Mass Concrete for Dams and Other Massive Structures," ACI Title Number 67-17, a report by ACI Committee 207.

2. American Society for Testing and Materials

    a. "Standard Method of Test for surface Moisture in Fine Aggregate," C70-73.

    b. "Standard Method of Test for Specific Gravity and Absorption of Coarse Aggregate," C127-73.

    c. "Standard Method of Test for Specific Gravity and Absorption of Fine Aggregate," C128-73.

    d. "Standard Method of Test for Length Change of Hardened Cement Mortar and Concrete," C157-75.

    e. "Standard Method of Test for Thermal Conductivity of Materials by Means of the Guarded Hot Plate," C177-71.

    f. "Standard Method of Test for Total Moisture Content of Aggregate by Drying," C566-67.

    g. "Standard Method of Test for Chloride Ion in Water and Waste Water," D512-67.

    h. "Standard Method of Test for Sulfate Ion in Water and Waste Water," D516-68.

    i. ASTM C109-08, Standard Test Method for Compressive Strength of Hydraulic Cement Mortars (Using 2-in Cube Specimens).

    j. ASTM C1107-11, Standard Specification for Packaged Dry Hydraulic-Cement Grout (Nonshrink).

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14-016

3. U. S. Army Corps of Engineers

   a. "Method of Test for Coefficient of Linear Expansion of Concrete," CRD-C39-55.

   b. "Method of Test for Flow of Grout Mixtures (Flow Cone Method)," CRD-C79-58.

   c. "Method of Test for Flat and Elongated Particles in Coarse Aggregate," CRD-C119-53.

   d. "Methods of Sampling and Testing Expansive Grouts," CRD-C589-70.

   e. CRD C621-93, Standard Specification for Packaged Dry Hydraulic-Cement Grout (Nonshrink). RN 14-016

4. Concrete Plant Manufacturers Bureau, "Concrete Plant Standard," Revision 5, March, 1973.

5. Truck Mixer Manufacturers Bureau, "Truck Mixer and Agitator Standard," Revision 9, November, 1971.

3.8.4.2.3 Reinforcing Steel

The following codes and specifications are used to establish the properties of and to control the fabrication and placement of reinforcing steel:

1. American Society for Testing and Materials

   a. "Standard Specification for Deformed Billet Steel Bars for Concrete Reinforcement," ASTM A615-72.

   b. "Standard Methods and Definitions for Mechanical Testing of Steel Products," ASTM A370-72.

2. American Concrete Institute

   a. "Specification for Structural Concrete for Buildings," ACI 301-72, Revision, 1975.

   b. "Building Code Requirements for Reinforced Concrete," ACI 318-71.

3.8.4.3 Loads and Load Combinations

3.8.4.3.1 Load Definitions

All the major loads encountered or postulated in the plant are listed below. The loads listed are not necessarily applicable to all the structures and their elements. Loads and the applicable load combinations for which each structure is designed depend upon the conditions to which that particular structure may be subjected.

1. Normal Loads

   Normal loads, which are those loads encountered during normal plant operation and shutdown include:

   a. D - Dead loads or their related internal moments and forces, including any permanent equipment loads and hydrostatic loads.

   b. L - Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure.

   c. T_o - Thermal effects and loads under normal operating or shutdown conditions, based upon the most critical transient or steady-state condition.

   d. R_o - Pipe reactions under normal operating or shutdown conditions, based upon the most critical transient or steady-state condition.

2. Severe Environmental Loads

   Severe environmental loads include:

   a. E - Loads generated by the OBE

   b. W - Loads generated by the design wind specified for the plant.

3. Extreme Environmental Loads

   Extreme environmental loads include:

   a. E' - Loads generated by the SSE

   b. W_t - Loads generated by the design tornado specified for the plant. Tornado loads include loads due to the tornado wind pressure, the tornado created differential pressure, and tornado generated missiles.

4. Abnormal Loads

Abnormal loads are those loads generated by a postulated high energy pipe break accident, including:

a. P_a - Pressure equivalent static load within or across a compartment generated by the postulated break, including an appropriate dynamic load factor to account for the dynamic nature of the load.

b. T_a - Thermal loads under thermal conditions generated by the postulated break, including T_o.

c. R_a - Pipe reactions under thermal conditions generated by the postulated break, including R_o.

d. Y_r - Equivalent static load on the structure generated by the reaction on the broken high energy pipe during the postulated break, including an appropriate dynamic load factor to account for the dynamic nature of the load.

e. Y_j - Jet impingement equivalent static load on a structure generated by the postulated break, including an appropriate dynamic load factor to account for the dynamic nature of the load.

f. Y_m - Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elastoplastic behavior is assumed where applicable with appropriate ductility ratios, provided excessive deflections do not result in loss of function of any safety-related system.

3.8.4.3.2 Load Combinations for Concrete Structures

Load combinations for concrete structures are listed in Table 3.8-12.

For service load conditions, either the working stress design method or the strength design method is used. Where soil and hydrostatic pressures are present, in addition to all service load condition load combinations listed in Table 3.8-12, where they are included in L and D, respectively, the requirements of Sections 9.3.4 and 9.3.5 of ACI 318-71 [2] are also satisfied (i.e., dead load and liquid levels are varied).

For factored load conditions, which represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the strength design method is used.

In load combinations c, d, and e of Table 3.8-12, the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, are used unless a time-history analysis is performed to justify otherwise. Load combinations b, d, and e of Table 3.8-12 and the corresponding structural acceptance criteria, presented in Table 3.8-12, are first satisfied without the tornado missile load in load combination b and without Y_j, Y_r, and Y_m in load combinations d and e. When considering these concentrated loads, local strength capacities are occasionally permitted to be exceeded where it is demonstrated that there is no loss of function of any safety-related system.

The criteria for loads and load combinations used in the designs of the Service Water Intake and Service Water Discharge Structures are summarized in Table 3.8-13. Since the structures are buried and/or submerged, no live, wind, or tornado loads were considered. The loads considered included the dead weight of the structures, lateral, and vertical earth pressure, lateral fluid pressure, and dynamic forces imparted by the OBE and the SSE. Different loading combinations were investigated so that the structures could resist the most critical load combination without distress.

3.8.4.3.3 Load Combinations for Steel Structures

Load combinations for steel structures are listed in Table 3.8-14.

For service load conditions, either the elastic working stress design methods of Part 1 of the AISC Specification [18] or the plastic design methods of Part 2 of the AISC Specification are used.

In the load combinations listed in Table 3.8-14 for factored load conditions, thermal loads are neglected when it is shown that they are secondary and self-limiting in nature.

In load combinations (3), (4), and (5) of Table 3.8-14 for factored load combinations for either elastic working stress design methods or plastic design methods, the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, are used unless a time-history analysis is performed. Load combinations (2), (4), and (5) of Table 3.8-14 for factored load combinations for either elastic working stress design methods or plastic design methods and the corresponding structural acceptance criteria, presented in Table 3.8-14, are first satisfied without the tornado missile load in load combination (2) and without Y_j, Y_r, and Y_m in load combinations (4) and (5). When considering these concentrated loads, local section strengths are occasionally permitted to be exceeded where it is demonstrated that there is no loss of function of any safety related system.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 General

The design and analytical procedures utilized for Seismic Category 1 structures are in accordance with the following codes:

1. For concrete structures, ACI 318-71 and ACI 349.

2. For steel structures, AISC Specification.

The computer programs utilized are described in Section 3.8.4.4.10. Methods used for seismic analysis are discussed in Section 3.7.

3.8.4.4.2 Auxiliary Building

The concrete floor slabs and roof were designed as one or two way slabs, or a combination of both, using geometry aspect ratio as a criterion. The floors and roof were analyzed using plate theory or computer program MASS 01. Where the plate theory was used, parametric studies were performed on the boundary conditions to obtain upper bound solutions. The computer program used modeled the slabs as a series of discrete nodes. The number of nodes used was determined from previous studies. The boundary conditions included in the model were based upon stiffness considerations.

Lateral slab loads were carried in diaphragm action to the walls. The vertical slab loads were transmitted to columns and walls and the lateral loads, including torsional slab loads were transmitted to the shear walls. The shear walls were analyzed with consideration for shear and flexural requirements, including the effects of penetrations.

The loads were applied independently to the walls and slabs and the results were combined in accordance with the load combinations given in Table 3.8-11 and 3.8-12. Pipe break forces were applied using dynamic load factors. Based upon the results of the analysis, the concrete and reinforcement requirements were obtained using the strength design method of ACI 318-71. Those portions of the Auxiliary Building where steel framing was used were designed in accordance with the AISC Specification. The roof slab and exterior walls were designed for local load conditions, i.e., missile penetration, utilizing the techniques described in Section 3.5.3.

3.8.4.4.3 Intermediate Building

With the exception of the 3 story tower, the roof and floors of the Intermediate Building were analyzed and designed as described in Section 3.8.4.4.2 for the Auxiliary Building. The application of the MASS 01 computer program, as described in Section 3.8.4.4.2, considered the stiffness contribution of the concrete slabs and the structural steel members and their mutual participation.

The 3 story concrete tower was analyzed and designed using plate theory. Parametric studies were performed on the boundary conditions to obtain upper bound solutions.

Lateral loads on both portions of the structure were carried by floors and roofs acting as diaphragms. The 3 story tower portion is tied to the balance of the structure by diaphragm action which carries all horizontal loads, including torsion.

The roof slab and exterior walls were designed for local load conditions, i.e., missile penetration, utilizing the techniques described in Section 3.5.3.

The loads were applied independently to the walls and slabs and the results were combined in accordance with the load combinations given in Tables 3.8-12 and 3.8-14. Pipe break forces were applied using dynamic load factors. Based upon the results of the analysis, concrete and reinforcement requirements were obtained using the strength design method of ACI 318-71. Where steel framing was used, it was designed in accordance with the AISC Specification.

3.8.4.4.4 Diesel Generator Building

The concrete walls and slabs were analyzed by moment distribution or moment coefficient methods. The floor and roof slabs were analyzed and designed as diaphragms to transmit lateral loads to the vertical shear walls and subsequently to the foundations. The loads were applied independently and the results were combined in accordance with the load combinations given in Table 3.8-12. Based upon the results of the analysis, concrete and reinforcement requirements were obtained using the strength design method of ACI 318-71.

The roof slab and exterior walls were designed for local load conditions i.e., missile penetration, utilizing the techniques described in Section 3.5.3.

3.8.4.4.5 Control Building

The interior structural steel frame was analyzed and designed in accordance with the provisions of the AISC Specification using the working stress method. The concrete floor slabs were analyzed utilizing moment coefficients and were designed in accordance with ACI 318-71 using the strength design method.

The floor and roof slabs were designed as diaphragms to transmit lateral loads to the exterior shear walls. The exterior walls were analyzed and designed as shear walls. Reinforcing steel was provided in accordance with the provisions of ACI 318-71. Loads were individually applied at their point of application and were combined in accordance with the load combination of Tables 3.8-12 and 3.8-14.

The roof slab and exterior walls were designed for local load conditions, i.e., missile penetration, utilizing the techniques described in Section 3.5.3.

3.8.4.4.6 Fuel Handling Building

The concrete spent fuel pool of the Fuel Handling Building was analyzed utilizing the computer programs MASS 01 and Slabs and Mats (S092) (see Section 3.8.4.4.10). The structure was idealized as a series of contiguous structural models utilizing boundary conditions which are based upon stiffness considerations employing parametric studies. The analytical results from these models were checked at the assumed boundaries to assure compatible structural behavior. Additional evaluations were performed to assess the impact of the new high-density fuel storage racks on the floor, walls and substructure of the Spent Fuel Pool. These evaluations were performed using the public domain, general-purpose finite element analysis program ANSYS (version 5.4). The structure was idealized using finite element models constructed to ensure realistic boundary conditions that do not perturb the internal forces and bending moments local to the spent fuel pool. Thermal finite element analysis assessed the stresses generated in the pool reinforced concrete structure by the normal operating pool water temperature loads and accident boiling temperature loads. RN 03-017

The balance of the concrete portion of the Fuel Handling Building was analyzed using moment coefficients and plate theory.

The structural steel portions of the Fuel Handling Building were analyzed using classic hand calculational approaches and were designed in accordance with the AISC Specification. Diagonal bracing in the roof and wall was designed to transfer lateral load to the foundations.

Loads were independently applied to the structures at their point of application and the results were combined in accordance with the load combinations given in Tables 3.8-12 and 3.8-14.

The concrete portions of the structure were designed in accordance with the strength design method of ACI 318-71.

3.8.4.4.7 Service Water Pumphouse

The analysis and design procedures used for the Service Water Pumphouse are similar to those used for the Diesel Generator Building, with the exception that the seismic soil loads on the embedded portion of the structure were determined by use of the FLUSH computer program, as described in Section 3.7.

The roof slab and exterior walls were designed for local load conditions, i.e., missile penetration, utilizing the techniques described in Section 3.5.3.

3.8.4.4.8 Service Water Intake Structure

The Service Water Intake Structure was assumed to be an infinitely long rectangular tube-like structure subjected to different load combinations as described in Section 3.8.4.3. Thus, the effects of loads act mainly in the plane of the cross section of the structure, which is shown by Figure 3.8-63. The structure was further assumed to be elastic and the moment distribution method was used for analysis. The foundation of the structure is the bottom slab. Dynamic lateral earth loadings were determined using the solution of Seed and Whitman [26].

The wing walls at the intake end are free standing walls with no earth behind them. Under operating conditions they are submerged and do not experience unbalanced forces. The loading which would be imparted to the walls during seismic events due to the inertia of the water was also considered using the Westergaard [27] solution. The static loading case was found to control. The wing walls are tied together with a bottom slab which serves as a foundation as shown by Figure 3.8-63. This slab also provides the lateral stability of the walls.

The ultimate strength design method was used to proportion and determine the amount of reinforcing steel required for the structure. All design procedures and allowable stresses are in compliance with ACI 318-71.

3.8.4.4.9 Service Water Discharge Structure

The different loads and load combinations used for the Service Water Discharge Structure are similar to those used for the Service Water Intake Structure and are shown by Table 3.8-13. The dynamic lateral earth pressure loading on abutment and wing walls was determined using the solution of Seed and Whitman [26]. The design computes the dynamic pressure considering saturated unit weight of soil below water level.

The abutment wall is designed as a slab spanning between 2 wing walls. The wing walls are designed as slabs spanning between counterfort walls. The heel slab between the counterforts is designed for the weight of soil downward. The base slab (the basin slab) is designed for the maximum uplift pressure due to differential water pressure. The base slab is also designed for soil pressure from overturning moment.

The stability against sliding is provided by the weight of the structure, weight of the soil on the heel slab, shearing resistance of the soil medium, and by shear keys under the abutment wall and the sill wall. The passive pressure at the toe of the structure provides added safety against sliding. The stability against overturning is adequately provided by the weight of the structure and the soil. The stability factors for overturning, sliding, and flotation are in accordance with the minimum factors of safety specified in Table 3.8-15.

The strength design method of ACI 318-71 was adopted for the design of the Service Water Discharge Structure.

3.8.4.4.10 Description of Computer Programs

Listed below are descriptions of the computer programs used in the analysis and design of plant structures.

1. STRUDL (S084)

   STRUDL is a widely used, well known, analytical program developed by Massachusetts Institute of Technology that was released to the public domain in November, 1968. This program has a wide range of usage for static and dynamic analysis of frame members and reinforced concrete structures. STRUDL includes the capability for linear and nonlinear, static and dynamic analysis.

   The program was run on the Gilbert Associates, Inc., Reading, PA IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2 and the McDonnell Douglas, St. Louis, Mo., IBM 360/195 dual processor computer system under operating system OS/MVT/ASP release 21.8.

2. MRI/STARDYNE

   The MRI/STARDYNE analysis system consists of a series of compatible computer programs designed to analyze linear elastic structural models. The system can be used to evaluate a wide variety of static and dynamic problems. Only the static version was used for Virgil C. Summer Nuclear Station. The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal conditions, nodal applied loads, and/or prescribed displacements. MRI/STARDYNE was developed by the Mechanics Research Institute and used by Control Data Corporation since 1971. It is a large capacity finite element program which is designed for the analysis of truss, frame, and plate structures. Based upon the direct stiffness method, the program assembles the individual element stiffnesses into a global structure stiffness matrix, using appropriate matrix transformation and combination techniques. This system of equations is then solved for the generalized coordinates by Cholesk decomposition. A complete discussion of the theory used in STARDYNE and the theoretical and analytical verification is provided in the "STARDYNE Theoretical Manual," Control Data Corporation (Publication No. 86616300).

   This program has been widely used for analysis of complex structures since its release for commercial use in 1971. The program was run under revision E, dated May, 1973, on the Control Data Corporation CYBERNET system in New York on a CDC 6600 computer under version 3.3 of the SCOPE operating system.

3. ELAD (S014)

ELAD is a computer program designed to determine elastic deformations, stresses, strains, and principal values of stress within axisymmetric solid structures of arbitrary shape subjected to axisymmetric or nonaxisymmetric pressure, concentrated loads and temperatures. All boundary conditions consistent with the theory of elasticity are permitted. A linear, elastic stress-strain relationship is assumed throughout the model. ELAD was developed by the Service Bureau Corporation, Inglewood, CA, under contract to the Air Force Weapons Laboratory, Kirtland AFB and is published as Technical Report No. AFWL-TR-69-70, dated October, 1969, available from the U. S. Government Document Clearing House.

This program is widely available in the public domain and was run on the McDonnell Douglas, St. Louis, MO, IBM 360/195 dual processor computer system under OS/MVT/ASP release 21.8 and also on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

4. DYNAL (S085)

DYNAL was developed by the Computer Science Department of McDonnell Douglas Automation Company and was operated under release 3.2, dated February 2, 1973, updated to September 10, 1973. The structural dynamic analyses available in DYNAL are based upon the modal superposition method using time history analysis. A simplified set of equations is formed in terms of "normal coordinates" and then solved. These "normal coordinates" are obtained by forming the stiffness and mass matrices of the structural system and solving for the normal modes and frequencies by the HOW method . The program capabilities for analysis using shock spectrum excitation or response spectrum were not utilized. Output obtained includes structural response in terms of displacement, velocity and accelerations at selected nodal points, maximum accelerations, and floor response curves.

DYNAL is available in the public domain and has been widely used since its commercial release in 1970. The program is written in the same language as STRUDL and was run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2 and on the McDonnell Douglas, St. Louis, MO. IBM 360/195, dual processor computer system under operating system OS/MVT/ASP release 21.8.

5. KALNINS Static Program (S043)

KALNINS uses a multisegment method of direct numerical integration of boundary value problems and was developed by Arturs Kalnins and published in the Journal of Applied Mechanics, Volume 31, September, 1964, pages 467-476, and in the Journal of the Acoustical Society of America, Volume 36, July, 1964, pages 1355-1365. The program calculates elastic deflections and stresses in a thin walled, axisymmetric shell when subjected to any arbitrary surface, edge, and/or ring loads. The solution is based upon the linear theory of elasticity and takes into consideration bending, as well as membrane action of the shell in response to applied load. Results are in terms of resultant forces and couples with stresses calculated by assuming a linear distribution through the thickness.

This program has been widely used for thin shell analysis since its release to the public domain in 1968. The program was run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

6. Column Design Program (SO27)

This Column Design Program selects steel columns for axial load plus end moments about major and/or minor axes per the allowable stress provisions in Part 1 of the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Building," 1969 Edition.

The program accepts and combines axial loads and end moments for gravity load and lateral (wind and seismic) loads causing major and/or minor axis bending. Column height and support conditions about both axes complete the input for each column. Support condition options include either assumed effective length (K) factors or standard support conditions of beam and column stiffness factors that are used to calculate K factors.

This program has been verified by hand calculations and comparison to other previously verified programs.

The program was run on Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

7. KALNINS Dynamic Program (S032)

The program calculates the natural frequencies and mode shapes (stresses as well as displacements) of symmetric or nonsymmetric free vibration of rotationally symmetric elastic shells using the method of analysis published by Arturs Kalnins in the Journal of the Acoustical Society of America, Volume 36, July, 1964, pages 1355-1365. The program is applicable to axially symmetric shells to which any number of axisymmetric branches are attached.

The program determines either all natural frequencies within a prescribed frequency interval or a specified number of consecutive natural frequencies above a given frequency. The mode shapes of all displacements and stresses (or stress-resultants) are calculated and printed out for any desired number of points.

KALNINS Dynamic Program is available in the public domain and has been widely used since its commercial release in 1969.

The program was run on the Gilbert Associates, Inc., Reading, PA IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

8. Center of Gravity and Mass Moment (S042)

This program computes the center of mass, the total mass, and the mass moment of inertia about 3 axes for floor and wall systems. These values are computed for use in the dynamic analyses of buildings.

This program has been verified by hand calculations.

The program was run on the Gilbert Associates, Inc., Reading, PA IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

9. Spring Constants of Piles (S046)

This program calculates the Winkler continuous spring constants for soil-caisson interaction. The Mindline equation, which defines the displacement components produced by a concentrated force within an isotropic half-space, is employed to evaluate the weighted average displacements of all the caissons in the group, to account for group action of the caissons, and to evaluate the Winkler continuous spring constants for the caisson group at specific depths.

This program has been verified by hand calculations.

The program was run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operation system O/S 21.8 MVT with HASP 3.2.

10. DYREC (S061)

    The DYREC program calculates dynamic responses of 2 dimensional lumped mass systems for translational and/or rotational motion. The program allows translation in one direction and inplace rotation. The response is calculated by direct numerical integration of the equations of motion. The program allows nonlinear material properties and gaps between elements. DYREC has been used primarily to model and analyze pipe whip problems.

    This program has been verified by hand calculations and comparison to another widely used, generally accepted program (ANSYS).

    The program was being run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

11. Wilson's Program (S066)

    Wilson's Program is a finite element program for stress analysis of 2 dimensional elastic solids and axisymmetric bodies of revolution. The program calculates stresses and strains for inplane pressure, displacement and concentrated type loading. Thermoelastic effects are included.

    Wilson's Program was obtained from the University of Pennsylvania. This program has been widely used since its release to the public domain in 1961.

    The program was run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

12. Wall Stiffness (SO73)

    The program calculates individual wall stiffnesses in 2 perpendicular directions, sums the wall stiffnesses in the 2 directions, locates the center of rotation (rigidity) for the entire wall system, and calculates the total rotational stiffness, Ip, about the center of rigidity.

    This program has been verified by hand calculations.

    The program was run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

13. Slabs and Mats (S092)

This program was developed to solve slab and foundation mat problems. The program generates and solves a set of finite difference equations for mat deflections. Moments and shears are calculated from the deflections at user selected grid points.

The method of solution is the Finite Difference Technique applied to the Lagrange-Germain Biharmonic Equation.

This program has been verified by comparison to a program in the public domain (NASTRAN) and published information.

The program was run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

14. Fourier/Wind (G019)

This program calculates coefficients to represent the variation of wind pressure with respect to circumferential coordinates by a Fourier Series. The program is also applicable to any even function with a period of 2π as long as the approximation of the integrals for the Fourier coefficients is sufficiently accurate for the function.

This program has been verified by hand calculations.

The program is being run on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operating system O/S 21.8 MVT with HASP 3.2.

15. MASS-01 (S106)

The computer program, MASS-01, "MAT AND SLAB SOLVER," is utilized to analyze mat, slab, and wall bending problems. The program is based upon finite difference theory, to form a symmetrical matrix, to solve the uncoupled Lagrange-Germain-Huber variable thickness form of the biharmonic partial differential equation for static transverse slab loading. The slab is divided into a discrete number of node points so that the resulting network forms the basis of the matrix.

The boundary conditions which MASS-01 can analyze are not limited to the classical boundaries (i.e., clamp, simple, and free). A slope deflection approach to the boundary conditions also enables the user to obtain solutions to elastic boundaries. Thus, the user can specify, for both interior and exterior boundaries, a geometry consisting of walls and columns perpendicular to the slab and composite or noncomposite beam framing. The effect of interaction, both bending and torsion, is also included so that the computer model effectively represents the structure.

This program has been verified by comparison to a program in the public domain (STARDYNE) and to published information.

The program was executed on the Gilbert Associates, Inc., Reading, PA, IBM 370/155 computer under IBM operation system O/S 21.8 MVT with HASP 3.2.

16. FLUSH (S115)

The computer program FLUSH is a further development of the complex response finite element program LUSH (released to public domain in 1974). FLUSH includes new features such as transmitting boundaries, beam elements, an approximate 3 dimension ability, deconvolution within the program, and out-of-code equation solver.

The program is based upon a recognized public domain program (LUSH) and has been compared to published sample problems.

The program was executed at United Computing Systems, Inc., Kansas City, MO, on multiple CDC mainframes (i.e., - 6600, CYBER 7418, CYBER 175) under APEX/SL.

17. ANSYS

ANSYS is a general purpose finite element computer program for the solution of diversified analysis problems. Analysis capabilities include static and dynamic; plastic, creep and swelling; small and large deflections; and steady-state and transient heat transfer. ANSYS structural analysis capabilities include static, elastic, plastic, creep, dynamic, seismic, large deflection, and stability analysis. ANSYS has an extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), and direct interfaces (compression only), with many of the elements containing complete plastic, creep and swelling capabilities. The ANSYS program is used in both the linear and nonlinear analysis of special two and three dimensional components and structures subject to loads such as: shear, axial, bending, torsion, pressure, and temperature. Output consists of nodal displacements, stresses, temperatures, etc., for use in evaluating the component or structure against specified acceptance criteria. Versions 5.4, 10.0A1, 13.0 and 14.0 of the ANSYS computer program have been used to analyze components and structures at VCSNS Unit 1.

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3.8.4.5 Structural Acceptance Criteria

Structural acceptance criteria for each of the loading combinations considered in the design of other Category 1 structures are presented in Tables 3.8-12, for concrete structures other than the Service Water Intake Structure, and 3.8-14, for steel structures. For the Service Water Intake and Discharge Structures, all allowable stresses are in accordance with ACI 318-71.

3.8.4.6 Materials, Quality Control and Special Construction Techniques

3.8.4.6.1 Materials

3.8.4.6.1.1 Concrete

1. Cement

   Cement for other Seismic Category 1 structures may be ASTM C150, Type II cement, described in Section 3.8.1.6.1.1 or it may be ASTM C150, Type I cement. Type I cement conforms to ASTM C150 and also to the optional ASTM C150 requirements for having 0.60% maximum alkalais and 50% minimum final penetration for false set. Type I cement can also be accepted having a compressive cube strength of 3000 psi minimum, at 7 days, but must also satisfy the above mentioned optional ASTM C150 requirements.

   Cement complies with ACI 349, Section 3.2.

2. Aggregates

   Aggregates comply with ACI 349, Section 3.3.

3. Water

   Water complies with ACI 349, Section 3.4.

4. Admixtures

   Admixtures used comply with ACI 349, Section 3.6.

5. Grout

   Grout for base plates is either a factory premixed, nonshrink, nonmetallic or metallic grout or a laboratory designed, field mixed, nonshrink grout. Grout has no linear contraction and a maximum of 0.30% linear expansion in accordance with Corps of Engineers Standard CRD C-621 or ASTM C1107. The compressive strength of the grout is 6,000 psi minimum cube strength at 28 days when made and tested in accordance with ASTM C109. The grout specimens for strength and linear expansion testing are mixed to a flowable consistency measured by the time of the efflux from a standardized flow cone between 20 and 30 seconds. The flow cone shall conform to the requirements of ASTM C 939. RN 14-016

3.8.4.6.1.2 Concrete and Grout Mixes

1. Concrete Mix Proportioning

   Concrete mix proportioning complies with ACI 349, Section 4.2.

   Fly ash is not used in the concrete mix. Concrete compressive strength is at least 3000 psi at 28 days, except in the following locations where concrete compressive strength is at least 5000 psi at 90 days:

   a. Intermediate Building

      Concrete compressive strength is 5000 psi at 90 days in the column pads, the east penetration access area, and in concrete surrounding pipe restraint embedments.

   b. Fuel Handling Building

      Concrete compressive strength is 5000 psi at 90 days in all walls and slabs above elevation 463' and in the spent fuel pool walls.

   The schedule of testing to verify that concrete reaches a compressive strength of 5000 psi at 90 days is as follows: One (1) cylinder is tested at 7 days for information; 1 at 28 days for information; and two at 90 days for verification of 90 day strength.

2. Grout Mix Proportioning

   Premixed grout is combined with water to provide a minimum 6000 psi strength at 28 days.

   Laboratory designed, field mixed, nonshrink grouts are proportioned and tested in accordance with ASTM C109 to provide a minimum 6000 psi at 28 days in a workable, flowable consistency.

3.8.4.6.1.3 Reinforcing Steel and Cadweld Splices

1. Material

   Reinforcing steel is Grade 60 billet steel which conforms to the requirements of ASTM A615. Materials for reinforcing steel and for Cadweld splices (mechanical splice sleeves) comply with the requirements of Section 3.5 of ACI 349 for reinforcing and Section 7.5 of ACI 349 for Cadweld splices.

2. Cadweld Splices

   Reinforcing bars, sizes S14 and S18 are spliced using Cadweld splices. The sleeves develop the tensile strength of the reinforcement. Cadwelding is performed in accordance with Section 7.5 of ACI 349.

3.8.4.6.1.4 Structural Steel

Structural steel conforms to ASTM A36-70a.

3.8.4.6.1.5 Non-Bearing Plates

Non-Bearing column splice fill plate steel complies with ASTM A569 and ASTM A659 and has a maximum carbon content of 0.16 to 0.25%. These fill plates are classified as Non-Seismic Category 1.

3.8.4.6.1.6 Stainless Steel

Stainless steel plate material conforms to ASTM A240, Grade 304, is cold rolled and subsequently annealed and pickled to produce a surface finish of 75 RMS, maximum.

3.8.4.6.2 Quality Control

3.8.4.6.2.1 Concrete

Quality control requirements for concrete comply with ANSI N45.2.5 [25]. Specific requirements for components are as follows:

1. Cement

   Quality control requirements for cement comply with ANSI N45.2.5.

2. Aggregates

   Quality control requirements for aggregates comply with ANSI N45.2.5 for moisture content (ASTM C566), potential reactivity (ASTM C289), and soundness (ASTM C88). In addition the following tests are required at the frequency indicated:

   a. Flat and elongated particles (CRD C-119), monthly.

   b. Gradation (ASTM C136), within 1 day of use.

   c. Material finer than No. 200 sieve (ASTM C117), within 1 day of use.

   d. Organic impurities (ASTM C40), weekly during production.

e. Friable particles (ASTM C142), monthly during production.

f. Lightweight pieces (ASTM C123), monthly during production.

g. Los Angeles abrasion (ASTM C131 or C535), every 6 months.

3. Water and Ice

Quality control requirements for water and ice comply with ANSI N45.2.5.

4. Admixtures

The admixture supplier is required to submit a certificate of compliance with ASTM C260 or C494 prior to or with each delivery of admixture.

5. Grout

Quality control requirements for grout comply with ANSI N45.2.5, except that, as a minimum, during production grout is tested weekly. In addition, grout is tested at a minimum rate of 4 specimens for every 2000 pounds of grout material used.

3.8.4.6.2.2 Steel

The quality control requirements for steel structures comply with ANSI N45.2.5.

3.8.4.6.2.3 Reinforcing Steel and Cadweld Splices

Quality control provisions for reinforcing steel and Cadweld splices are discussed in Section 3.8.1.6.2.3.

3.8.4.6.3 Construction

There are no special construction techniques employed. Concrete construction complies with ACI 301-72 and steel construction complies with Reference [18].

3.8.4.7 Testing and Inservice Surveillance Requirements

Minimum testing requirements for the Service Water Pumphouse and the Service Water intake structure are provided in Section 2.5.4.10.6.2. 02-01

There is no testing or inservice surveillance planned for other Seismic Category 1 structures.

3.8.5 FOUNDATIONS

Foundation types and arrangements of Seismic Category 1 structures surrounding the Reactor Building are shown by Figure 3.8-65. These Seismic Category 1 structures are separated by a seismic gap which extends through the foundations except between the Control Building and the Intermediate Building foundations. The relationship of adjacent, separated structures and details of typical seismic separation are shown by Figure 3.8-52.

With the exception of the Reactor Building, the structural concrete has a compressive strength of 3000 psi. Reinforcing steel is ASTM A615, Grade 60.

Structural fill concrete, caissons, mats on existing soil, or compacted fill are used as foundation media for the Seismic Category 1 structure to transmit loads to competent rock.

Structural fill concrete is under the Reactor Building, Auxiliary Building, and Control Building as shown in Figure 3.8-65. The fill concrete extends from approximately elevation 340' to the undersides of the Seismic Category 1 structural mats. The fill concrete was placed in 5 foot thick layers, utilizing mass concrete construction techniques. The compressive strength of the fill concrete was 1500 psi, with the exception of the 12 foot thick layer immediately under the Reactor Building base mat and the uppermost 2 foot layer under the Auxiliary Building which had a compressive strength of 3000 psi. Fill concrete segments were keyed to each other on their vertical faces and the horizontal interfaces were roughened to assure shear friction resistance.

The Intermediate, Fuel Handling, and Diesel Generator Buildings are supported on caissons as shown in Figure 3.8-65.

Existing soil or compacted fill supported structures are the Service Water Intake Structure, the Service Water Discharge Structure, and the Service Water Pumphouse as shown in Figures 3.8-62, 3.8-63, and 3.8-64.

3.8.5.1 <u>Description of Foundations</u>

3.8.5.1.1 Reactor Building Foundation

A description of the physical characteristics of the structural foundation mat and fill concrete is presented in Section 3.8.1.1.1.1. Structural aspects are discussed in Section 3.8.1.4.1 and the following sections.

3.8.5.1.1.1 Structural Foundation Mat

The 12 foot thick structural foundation mat is reinforced with ASTM A615, Grade 60 reinforcement. Pertinent details of the reinforcement are as follows:

1. A radial and hoop grid of bars located at the top and bottom of the mat.

2. A rectangular grid of bars located at the top and bottom of the incore base.

3. A grid of vertical and horizontal bars at each face in the incore wall which is dowelled into the incore base and the mat.

4. Shear ties in the mat and incore walls.

5. Inclined bars spaced along the perimeter of the incore wall which are dowelled into the mat.

Figure 3.8-66 shows the bar sizes used and the spacing of this reinforcement. This reinforcement is designed to resist the load combinations presented in Table 3.8-1. All reinforcement except shear ties consists of No. 14 and No. 18 bars. Splicing of bars is accomplished by the use of Cadweld splices. The splices are staggered so that splices on adjacent bars are not less than 2 feet apart.

The structural foundation mat is constructed in the following sequence:

1. The incore base is placed in a single 13 foot pour.

2. The incore wall is placed in a single 8 foot pour.

3. The mat is placed in 12 foot thick, pie shaped sections.

Keys are used to transmit shear between these elements.

Horizontal shear transfer between the structural foundation mat and fill concrete is provided by both of the following:

1. The lateral bearing of the incore pit on the fill concrete.

2. The friction of the structural mat against the fill concrete at elevation 396' which is roughened to 1/4 inch amplitude.

Either of the above conditions acting alone is adequate to transfer the horizontal shear force.

3.8.5.1.1.2 Foundation Details

The joint at elevation 396' is continuously sealed along the perimeter of the structural foundation mat from external groundwater. This seal consists of a continuous 8 foot wide mortar joint external to the tendon gallery and continuous water stops, one external and one internal to the tendon gallery. The main function of these seals is to prevent entrance of groundwater into the tendon gallery. The structural mat and liner are designed to resist the full hydrostatic force.

Horizontal shear transfer between the 5 foot high sections of fill concrete and competent rock is provided by friction between the lists. Prior to placement of a lift, a layer of mortar is applied to the top of the previous lift which has been roughened to approximately 1/4 inch amplitude.

3.8.5.1.2 Auxiliary Building Foundation

The Auxiliary Building is founded on a 4 foot thick structural reinforced concrete mat which is supported on fill concrete extending down to competent rock. A waterproofing membrane is provided between the fill concrete and structural mat. Horizontal shears are transferred to the fill concrete through a series of shear keys. The shear keys are 6 feet square in plan and extend 1 foot deep into the fill concrete.

The Auxiliary Building mat is stepped from elevation 374' to elevation 388'. The plan and details of the Auxiliary Building foundation mat are shown by Figures 3.8-67 through 3.8-70.

3.8.5.1.3 Intermediate Building Foundation

The Intermediate Building foundation consists of a 3 foot thick basement floor slab which acts in conjunction with a series of grade beams to transfer vertical loads to the reinforced concrete caissons and shear/bearing walls and to concrete piers. The shear/bearing wall foundations and the reinforced concrete caissons are founded on competent rock. The piers are founded on fill concrete which extends beyond the Reactor Building and Auxiliary Building. Horizontal shears are transferred through the basement floor slab to the shear/bearing walls and to the Control Building base mat.

The Control Building base mat and the Intermediate Building basement floor and grade beam system are structurally tied together so that they act in concert through diaphragm action to transfer horizontal shears, including torsion, from the Intermediate Building.

Plans and details are shown by Figures 3.8-67, 3.8-70, and 3.8-71. RN 01-068

3.8.5.1.4 Control Building Foundation

The Control Building foundation consists of a 4 foot thick reinforced concrete mat, the function of which is the transfer of vertical load from the superstructure columns to the fill concrete supporting the structural mat. The exterior shear walls are supported by the fill concrete through the 4 foot thick base mat. The fill concrete extends approximately 45 feet down to competent rock. Vertical reinforcing steel extends from the exterior shear walls approximately 25 feet into the layers of fill concrete. These vertical reinforcing bar anchorages are designed to resist the Control Building seismic uplift overturning forces, as well as the horizontal shears through shear friction.

Plans and details are shown by Figures 3.8-67 and 3.8-71. RN 01-068

3.8.5.1.5 Fuel Handling Building Foundation

The foundation system for the Fuel Handling Building consists of a concrete mat formed by the bottom of the spent fuel pool and fuel cask pit and is stepped up at the railroad bay. This mat is supported by reinforced concrete piers which extend to the fill concrete adjacent to the Reactor and Auxiliary Buildings and by reinforced concrete caissons which extend to competent rock on the north and east sides of the Fuel Handling Building mat. These caissons are outboard of the Reactor and Auxiliary Buildings. The reinforced concrete piers are attached to the fill concrete by reinforcing steel dowels and shear key sockets into the fill concrete. The caissons are socketed into the competent rock. Perimeter grade beams are provided at the outboard edges of the railroad bay mat.

Horizontal shears are transmitted from the base mat to the piers and caissons. The piers and caissons share the horizontal load with the piers resisting the major portion of the load.

Plans and details are shown by Figures 3.8-58, 3.8-59, 3.8-60, and 3.8-67. 02-01

3.8.5.1.6 Diesel Generator Building

The Diesel Generator Building is founded on a basement slab and grade beam system which is supported by reinforced concrete caissons extending to competent rock. Horizontal forces are resisted by caissons shear and soil-structure interaction. The reinforced concrete caissons also resist seismic overturning moments. Foundation plans and details are shown by Figures 3.8-56 and 3.8-67. 02-01

3.8.5.1.7 Service Water Pumphouse

The Service Water Pumphouse is founded on a structural foundation mat at elevation 390'. The valve pit and control areas at elevation 425' are supported by columns which extend to the supporting foundation mat at elevation 390'. The mat is founded on compacted fill which extends from the underside of the base mat to elevation 354'. The compacted fill is supported on in-situ soils (saprolite) from elevation 354' to elevation 350' where decomposed rock exists. Competent rock exists at approximately elevation 300'.

The Service Water Pumphouse structure is separated from the Service Water Intake Structure and from the connecting pipes and conduits by flexible joints, which accommodate relative settlement and seismic movement.

Foundation plan and details are shown on Figures 3.8-61 and 3.8-62.

3.8.5.1.8 Service Water Intake Structure

The base of the Service Water Intake Structure is at elevation 367'. The structure bears on fill material except for a portion of the inlet which rests on in-situ soils.

The structural features of the base slab are shown by Figure 3.8-63.

3.8.5.1.9 Service Water Discharge Structure

The base slab of the Service Water Discharge Structure bears at elevation 408' partly on the decomposed rock and partly on the fill concrete that extends to the decomposed rock.

The structural features of the base slab are shown by Figure 3.8-64.

3.8.5.2 Applicable Codes, Standards, and Specifications

Codes, standards, and specifications applicable to the Reactor Building foundation are discussed in Section 3.8.1.2. Those applicable to the foundations of other Seismic Category 1 structures are discussed in Section 3.8.4.2.

3.8.5.3 Loads and Load Combinations

3.8.5.3.1 Load Definitions

The loads considered for the Reactor Building foundation include those defined in Section 3.8.1.3.1. Loads considered for other Seismic Category 1 structure foundations include those defined in Section 3.8.4.3.1 plus those defined below.

Additional loads considered are as follows:

1. H - lateral earth pressure

2. F' - buoyant force resulting from the probable maximum flood.

3.8.5.3.2 Load Combinations

Load combinations considered for the Reactor Building foundation are listed in Table 3.8-1. Those considered for other Seismic Category 1 structure foundations are listed in Tables 3.8-12 and 3.8-13, for reinforced concrete structures, and 3.8-14, for steel structures. In addition to the load combinations presented in Tables 3.8-1, 3.8-12, 3.8-13, and 3.8-14, the load combinations listed in Table 3.8-15 are used to assure that sliding and overturning due to earthquake, wind, and tornado, as well as flotation do not exceed allowable factors of safety.

3.8.5.4 Design and Analysis Procedures

3.8.5.4.1 Reactor Building

The Reactor Building foundation design and analytical procedures are described in Section 3.8.1.4.

3.8.5.4.2 Auxiliary Building

The Auxiliary Building mat foundation was analyzed using plate coefficient methods and by use of the computer program STRUDL (see Section 3.8.4.4). The base mat was divided into representative equivalent frames and was analyzed using the STRUDL computer program. In both methods the supporting foundation medium was idealized as an elastic support.

The lateral loads were transmitted by diaphragm action of the base mat and were resisted by the shear keys which are integral with the base mat and supporting fill concrete. The shear keys and the local effects to the base mat were analyzed and designed using hand calculations.

The loads were independently applied at their point of application and the results were combined in accordance with the load combinations given in Table 3.8-14. Overall stability was checked to assure conformance with the load combinations given in Table 3.8-15.

The concrete stresses were checked and reinforcing steel requirements were determined in accordance with the strength design method of ACI 318-71 [2].

3.8.5.4.3 Intermediate Building

The base mat of the Intermediate Building was analyzed using the MASS 01 computer program (see Section 3.8.4.4) for vertical loads. The base mat was modeled as a structurally supported slab system with the structural stiffness of the supports included in the model.

Boundary conditions were the foundation medium (competent rock) and the first superstructure level where appropriate stiffnesses were determined.

The base mat was analyzed for horizontal shear using Wilson's finite element computer program (S066) (see Section 3.8.4.4). The boundary conditions were modeled at the shear walls, caissons, and the Control Building base mat. Structural stiffnesses of the boundary conditions were determined and included in the model.

The loads were applied independently at their point of application and the results were combined in accordance with the load combinations given in Table 3.8-12. Overall foundation stability was checked to assure conformance with the load combinations given in Table 3.8-15.

The concrete stresses were checked and the reinforcing steel requirements were determined in accordance with the strength design method of ACI 318-71.

3.8.5.4.4 Control Building

The base mat of the Control Building was analyzed using the finite difference computer program, Slabs and Mats (S092) (see Section 3.8.4.4). Boundary conditions included in the model were the foundation medium (fill concrete) and the first level of superstructure above the base mat. The vertical reinforcing steel, which extends from the perimeter concrete walls and is embedded into the fill concrete, was modeled as tension springs.

Loads were applied independently at their point of application and the results were combined in accordance with the load combinations given in Table 3.8-12. Overall foundation stability was checked to assure conformance with the load combinations given in Table 3.8-15.

The concrete stresses were checked and reinforcing steel requirements were determined in accordance with the strength design method of ACI 318-71.

3.8.5.4.5 Fuel Handling Building

The Fuel Handling Building base mat and grade beam system was analyzed using the Slabs and Mats (S092) and MASS 01 (S106) computer programs (see Section 3.8.4.4). The base mat system was modeled as a structurally supported slab with the supporting piers and caissons as the foundation boundaries. The superstructure boundaries were the structural steel interfaces and the top elevations of the fuel pool walls. The structural stiffnesses of these boundary conditions were included in the model.

Portions of the base mat were analyzed using hand calculations to confirm the structural behavior predicted by the computer programs and/or to study local effects.

Horizontal shears were distributed through the base mat to the piers and caissons based upon stiffness considerations.

Additionally, the spent fuel pool was analyzed for thermal gradients associated with operating and accident conditions. Computer programs MASS 01 (S106) and Wilson's finite element program (S066) (see Section 3.8.4.4) were used in the analysis. The spent fuel pool was initially modeled utilizing substructure techniques. Wilson's finite element program (S066) was employed for the analysis of axial deformation. The MASS 01 (S106) computer program was used to analyze the flexural deformation. The substructures were joined by applying restoring forces to the individual substructures to ensure the compatibility of displacements. The resulting stresses were combined with other stresses. The spent fuel pool was re-analyzed to assess the impact of the new high-density fuel storage racks on the floor, walls and substructure. These evaluations were performed using the general-purpose finite element analysis program ANSYS. The structure was idealized using finite element models constructed to ensure realistic boundary conditions that do not perturb the internal forces and bending moments local to the spent fuel pool. Thermal finite element analysis assessed the stresses generated in the pool reinforced concrete structure by the normal operating pool water temperature loads and accident boiling temperature loads. RN 03-017

The loads were applied individually at their point of application and the results were combined in accordance with the load combinations given in Table 3.8-12. Overall foundation stability was checked to ensure conformance with load combinations given in Table 3.8-15.

Concrete stresses were checked and reinforcing steel requirements were determined in accordance with the strength design method of ACI 318-71.

3.8.5.4.6 Diesel Generator Building

The basement slab and grade beam system was analyzed as a rigid mat supported by caissons. The axial loads induced by seismic rocking moments were distributed to the caissons by the moment-area method. Grade beams are provided to span between walls where required, to transfer caisson reactions into the superstructure. The STRUDL computer program (see Section 3.8.4.4) was used for grade beam analysis wall and caisson connections to the beam were assumed to be pinned for the model.

The caisson were analyzed for lateral loads utilizing the STRUDL computer program. The caissons were fixed at the base and were restrained against rotation at the top. The surrounding soil was idealized as soil springs. Caisson moments and shears were determined by applying top displacements consistent with the dynamic analysis.

The loads were applied independently and the results were combined in accordance with the load combinations given in Table 3.8-12. Overall foundation stability was checked to ensure conformance with the load combinations given in Table 3.8-15.

The concrete stresses were checked and the reinforcing steel requirements were determined in accordance with the strength design method of ACI 318-71.

3.8.5.4.7 Service Water Pumphouse

The Service Water Pumphouse base mat was assumed to be rigid and was analyzed using plate theory and moment coefficients.

Loads were independently applied at their point of application and the results were combined in accordance with the load combinations given in Table 3.8-12. Overall foundation stability was checked to assure conformance with the load combinations and acceptance criteria given in Table 3.8-15. Horizontal shears are transferred from the structure to the soil through friction.

The concrete stresses were checked and the reinforcing requirements were determined in accordance with the strength design method of ACI 318-71.

Based on the settlement analysis described in Section 2.5.4.10.6.2, differential settlement across the structure is found to be within acceptable limits for the structure and within the working tolerances of the equipment for satisfactory operation of the plant.

Flexible joints are designed for the relative settlement between the structure and the soil embedded services, as well as the Service Water Intake Structure.

3.8.5.4.8 Service Water Intake Structure

The structural design of the Service Water Intake Structure foundation is described in Section 3.8.4.4.8.

Settlement of the Service Water Intake Structure is described in Section 2.5.4.10.6.2. The post construction relative settlement between the intake structure and the Service Water Pumphouse is accommodated by a flexible connection, as discussed in Section 3.8.5.4.7.

3.8.5.4.9 Service Water Discharge Structure

The design and analysis procedure for the Service Water Discharge Structure base slab is described in Section 3.8.4.4.9.

The settlement of the Service Water Discharge Structure, bearing on decomposed rock is negligible, as noted in Section 2.5.4.

3.8.5.5 <u>Structural Acceptance Criteria</u>

For the Reactor Building the allowable limits which constitute the acceptance criteria are discussed in Section 3.8.1.5. Structural acceptance criteria for all other foundations are listed in Tables 3.8-12, for concrete Seismic Category 1 structures other than the Service Water Intake Structure and 3.8-14 for steel Seismic Category 1 structures. For the Service Water Intake Structure foundation, allowable stresses used in the design are in accordance with ACI 318-71. In addition, for the additional load combinations discussed in Section 3.8.5.3, the factors of safety against overturning, sliding, and flotation are presented in Table 3.8-15.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Materials, quality control, and construction of the Reactor Building foundation are discussed in Section 3.8.1.6. Materials and quality control for foundations of other Seismic Category 1 structures are discussed in Section 3.8.4.6. No special construction techniques are employed in the construction of other Seismic Category 1 foundations.

3.8.5.7 Testing and Inservice Surveillance Requirements

There are no special testing or inservice surveillance requirements for Seismic Category 1 foundations.

3.8.6 REFERENCES

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TABLE 3.8-0

MAIN STEAM PENETRATION MATERIALS AND INSPECTION REQUIREMENTS

Item	Material	Inspection Requirements
Process Pipe	33" OD, 1.70" wall, SA 106, Grade C	Ultrasonic examination, ASTM 213; impact tested
Sleeve	56" OD, 1.25" wall, SA 155 KCF 70, Class 1	Longitudinal butt welds radio-graphed; impact tested
Spacer Rings	56" OD, 1.25" wall, SA 516, Grade 70	Longitudinal butt welds radio-graphed; impact tested
Attachment Plates	2" thick, SA 516, Grade 70	Impact tested
Spacer Plate	1" thick, SA 516, Grade 70	Impact tested
Bellows Assembly		
Bellows	0.037" thick, SA 240, T304	Longitudinal butt weld radiographed
Ring	46" ID, 0.50" wall, SA 516, Grade 70	
Plate	1" thick, SA 516, Grade 70	
Stiffeners	1" thick, SA 516, Grade 70	
Cooling Fins	1/4" thick, SA 516, Grade 70	

TABLE 3.8-1

REACTOR BUILDING CONCRETE LOAD COMBINATIONS

1. Preoperation

   $S = D + L + F + T_o$

2. Structural Acceptance Test

   $S = D + L + F + 1.15P$ + thermal effects anticipated at time of test.

3. Normal Operation

   a. $S = D + L + F + T_o + Z$
   b. $U = 1.4D + 1.7L + F + T_o + Z$
   c. $U = D + 1.25L + F + T_o + Z + 1.256$ (E or W)

4. Design Accident

   a. $U = D + L + F + T_o + T_a + 1.5P + R$
   b. $U = D + L + F + T_o + T_a + 1.25P + R + 1.25$ (E or W)

5 Extreme Environmental

   a. $U = D + L + F + T_o + T_a + P + E' + R$
   b. $U = D + L + F + T_o + W' + Z'$

6. Post Tensioning Level

   The post tensioning level is such that the membrane stress resultants remain compressive for two load combinations:

   a. $D + F + P + T_o + T_a$
   b. $D + F + 1.2P + T_o$

TABLE 3.8-2

REACTOR BUILDING LINER AND ANCHOR LOAD COMBINATIONS

1. Construction

   $D + L + T_o + W$

2. Test

   $D + L + F + P_t$ + thermal effects anticipated at time of test.

3. Normal

   $D + L + F + T_o + R_o + P_v$

4. Environmental

   a. Severe, $D + L + F + T_o + R_o + (E_o \text{ or } W) + P_v$ RN 01-113
   b. Extreme, $D + L + F + T_o + R_o + (E_{ss} \text{ or } W_t) + P_v$

5 Abnormal

   $D + L + F + P_a + T_a + R_a$

6. Abnormal/Severe Environmental

   $D + L + F + P_a + T_a + E_o + R_a$

7. Abnormal/Extreme Environmental

   $D + L + F + P_a + T_a + E_{ss} + R_a + R_r$

NOTE:

Wind load was considered during erection of reactor building liner. For definition of terms, refer to the ASME Code, Section III, Division 2. RN 01-113

TABLE 3.8-3

ATTACHMENT, BRACKET AND PAD OVERLAY PLATE LOAD COMBINATIONS

1. Construction

   $S = D + L$

2. Normal

   $S = D + L + R_o$

3. Severe Environmental

   $S = D + L + R_o + E_o$

4. Abnormal/Extreme Environmental

   $1.5S = D + L + R_a + E_{ss}$ RN 99-101

NOTE:

"Specification for Design, Fabrication, and Erection of Structural Steel for Buildings," American Institute of Steel Construction.

TABLE 3.8-4

MINIMUM CONCRETE COVER FOR REINFORCEMENT

Location	Minimum Cover	
Dome		18 and 14S - 2-1/4 inches
		Others - 2 inches
Cylinder		18 and 14S - 2-1/4 inches
		Others - 2 inches
Foundation Mat	Bottom Reinforcing	18 and 14S - 3 inches
		Others - 3 inches
	Top Reinforcing	18 and 14S - 2-1/4 inches
		Others - 2-1/4 inches

TABLE 3.8-5

TOLERANCES FOR LOCAL BULGES, FLATSPOTS OR DISCONTINUITIES IN CYLINDRICAL PORTION OF REACTOR BUILDING LINER

Chord Length	Centerline Theoretical Offset (in)	Tolerance Measured in One Plate More Than 12 Inches from Weld (in)	Tolerance Measured across Weld (in)
10' - 0"	2-3/8	± 7/16	± 5/8
15' - 0"	5-1/2	± 1	± 1-1/2
20' - 0"	9-5/8	± 1-3/4	± 2-5/8

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TABLE 3.8-6

DISPLACEMENT MEASUREMENT LOCATIONS FOR REACTOR BUILDING STRUCTURAL ACCEPTANCE TEST

Instrument No. & Gage	Elevation	Azimuth	Notes	
Cylinder Base and Cylinder Wall Radial Displacement - Direct Current Displacement Transducer (DCDT) Locations				
DCDT 1, 2, 3	420'-0"	59°, 101°, 162°-30'	Radial displacement	02-01
DCDT 4, 5, 6	420'-0"	243°-20', 308°, 347°	Radial displacement	
DCDT 7, 8, 9	483'-0"	243°-20', 308°, 347°	Radial displacement	
Cylinder Wall and Dome Radial Displacement - Scale and Jig Transit Locations				
Scale 10, 11	483'-0"	59°, 162°-30'	Radial displacement	
Scale 12, 13, 14	557'-0"	59°, 101°, 162°-30'	Radial displacement	
Scale 15, 16, 17	557'-0"	243°-20', 308°, 347°	Radial displacement	
Scale 18, 19, 20	576'-5"	59°, 101°, 162°-30'	Radial displacement	
Scale 21, 22, 23	576'-5"	243°-20', 308°, 347°	Radial displacement	
Cylinder Ledge Vertical Displacement - Invar Tape and DCDT Locations				
DCDT 30, 31, 32	576'-5"	59°, 101°, 162°-30'	Vertical displacement	
DCDT 33, 34, 35	576'-5"	243°-20', 308°, 347°	Vertical displacement	
Dome Apex Vertical Displacement Invar Tape and DCDT Locations				
DCDT 36	599'-0"	-	Vertical displacement	
DCDT 37, 40, 43, 46, 49, 52	472'-6"	-	Vertical displacement	
DCDT 38, 41, 44, 47, 50, 53	472'-6"	-	Radial displacement	02-01
DCDT 39, 42, 45, 48, 51, 54	472'-6"	-	Tangential displacement	

TABLE 3.8-6 (Continued)

DISPLACEMENT MEASUREMENT LOCATIONS FOR REACTOR BUILDING STRUCTURAL ACCEPTANCE TEST

Equipment Hatch Displacement - (DCDT) Locations

Instrument No. & Gage	Elevation	Azimuth	Notes
DCDT 55	492'-6"	101°	Vertical displacement
DCDT 58	486'-6"	101°	Vertical displacement
DCDT 61	480'-6"	101°	Vertical displacement
DCDT 64	464'-6"	101°	Vertical displacement
DCDT 67	458'-6"	101°	Vertical displacement
DCDT 70	452'-6"	101°	Vertical displacement
DCDT 56	492'-6"	101°	Radial displacement
DCDT 59	486'-6"	101°	Radial displacement 02-01
DCDT 62	480'-6"	101°	Radial displacement
DCDT 65	464'-6"	101°	Radial displacement
DCDT 68	458'-6"	101°	Radial displacement
DCDT 71	452'-6"	101°	Radial displacement
DCDT 57	492'-6"	101°	Tangential displacement
DCDT 60	486'-6"	101°	Tangential displacement
DCDT 63	480'-6"	101°	Tangential displacement
DCDT 66	464'-6"	101°	Tangential displacement
DCDT 69	458'-6"	101°	Tangential displacement
DCDT 72	452'-6"	101°	Tangential displacement

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TABLE 3.8-7

STRAIN GAGE LOCATIONS FOR REACTOR BUILDING
STRUCTURAL ACCEPTANCE TEST

Cylinder Wall and Base Junction - Strain Gage and Rebar Locations

Instrument No. & Gage	Elevation	Azimuth	Notes	
Primary Strain Gage 73, 75, 77	410'-6"	8°	Meridional strain	02-01
Redund. Strain Gage 1073, 1075, 1077	410'-6"	8°	Meridional strain	
Primary Strain Gage 74, 76, 78	410'-6"	8°	Hoop strain	
Redund. Strain Gage 1074, 1076, 1078	410'-6"	8°	Hoop strain	
Primary Strain Gage 79, 81, 83	412'-6"	8°	Meridional strain	
Redund. Strain Gage 1079, 1081, 1083	412'-6"	8°	Meridional strain	
Primary Strain Gage 80, 82, 84	412'-6"	8°	Hoop strain	
Redund. Strain Gage 1080, 1082, 1084	412'-6"	8°	Hoop strain	

Cylinder Wall and Dome Junction - Strain Gage and Rebar Locations

Instrument No. & Gage	Elevation	Azimuth	Notes
Primary Strain Gage 85, 87, 89	553'-6"	8°	Meridional strain
Redund. Strain Gage 1085, 1087, 1089	553'-6"	8°	Meridional strain
Primary Strain Gage 86, 88, 90	553'-6"	8°	Hoop strain
Redund. Strain Gage 1086, 1088, 1090	553'-6"	8°	Hoop strain
Primary Strain Gage 91, 93, 95	557'-0"	8°	Meridional strain

TABLE 3.8-7 (Continued)

STRAIN GAGE LOCATIONS FOR REACTOR BUILDING
STRUCTURAL ACCEPTANCE TEST

Cylinder Wall and Base Junction - Strain Gage and Rebar Locations

Instrument No. & Gage	Elevation	Azimuth	Notes
Redund. Strain Gage 1091, 1093, 1095	557'-0"	8°	Meridional strain
Primary Strain Gage 92, 94, 96	557'-0"	8°	Hoop strain
Redund. Strain Gage 1092, 1094, 1096	557'-0"	8°	Hoop strain

Equipment Access Opening - Strain Gage and Rebar Locations

Instrument No. & Gage	Elevation	Azimuth	Notes
Primary Strain Gage 97, 99, 101, 103, 105, 107, 109, 111, 113, 115, 117, 119	472'-6"	-	Meridional strain
Redund. Strain Gage 1097, 1099, 1101, 1103, 1105, 1107, 1109, 1111, 1113, 1115, 1117, 1119	472'-6"	-	Meridional strain
Primary Strain Gage 98, 100, 102, 104, 106, 108, 110, 112, 114, 116, 118, 120	472'-6"	-	Hoop strain
Redund. Strain Gage 1098, 1100, 1102, 1104, 1106, 1108, 1110, 1112, 1114, 1116, 1118, 1120	472'-6"	-	Hoop strain
Primary Strain Gage 121, 123, 125	488'-6"	101°	Meridional strain
Redund. Strain Gage 1121, 1123, 1125	488'-6"	101°	Meridional strain

TABLE 3.8-7 (Continued)

STRAIN GAGE LOCATIONS FOR REACTOR BUILDING
STRUCTURAL ACCEPTANCE TEST

Instrument No. & Gage	Elevation	Azimuth	Notes
Main Strain Gage 122, 124, 126	488'-6"	101°	Hoop strain
Redund. Strain Gage, 1122, 1124, 1126	488'-6"	101°	Hoop strain
Main Strain Gage, 127, 129, 131	484'-10"	101°	Meridional strain
Redund. Strain Gage 1127, 1129, 1131	484'-10"	101°	Meridional strain
Main Strain Gage 128, 130, 132	484'-10"	101°	Hoop strain
Redund. Strain Gage 1128, 1130, 1132	484'-10"	101°	Hoop strain
Main Strain Gage 133, 135, 137	460'-9"	101°	Meridional strain
Redund. Strain Gage 1133, 1135, 1137	460'-9"	101°	Meridional strain
Main Strain Gage 134, 136, 138	460'-9"	101°	Hoop strain
Redund. Strain Gage 1134, 1136, 1138	460'-9"	101°	Hoop strain
Main Strain Gage 139, 141, 143	454'-9"	101°	Meridional strain
Redund. Strain Gage 1139, 1141, 1143	454'-9"	101°	Meridional strain
Main Strain Gage 140, 142, 144	454'-9"	101°	Hoop strain
Redund. Strain Gage 1140, 1142, 1144	454'-9"	101°	Hoop strain

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TABLE 3.8-7 (Continued)

STRAIN GAGE LOCATIONS FOR REACTOR BUILDING
STRUCTURAL ACCEPTANCE TEST

Cylinder Wall - Strain Gage and Rebar Locations

Instrument No. & Gage	Elevation	Azimuth	Notes
Main Strain Gage 165, 167, 169	420'-6"	8°	Meridional strain
Redund. Strain Gage 1165, 1167, 1169	420'-6"	8°	Meridional strain
Main Strain Gage 166, 168, 170	420'-6"	8°	Hoop strain
Redund. Strain Gage 1166, 1168, 1170	420'-6"	8°	Hoop strain
Main Strain Gage 171, 173, 175	450'-6"	8°	Meridional strain
Redund. Strain Gage 1171, 1173, 1175	450'-6"	8°	Meridional strain
Main Strain Gage 172, 174, 176	450'-6"	8°	Hoop strain
Redund. Strain Gage 1172, 1174, 1176	450'-6"	8°	Hoop strain
Main Strain Gage 177, 179, 181	483'-0"	8°	Meridional strain
Redund. Strain Gage 1177, 1179, 1181	483'-0"	8°	Meridional strain
Main Strain Gage 178, 180, 182	483'-0"	8°	Hoop strain
Redund. Strain Gage 1178, 1180, 1182	483'-0"	8°	Hoop strain
Main Strain Gage 183, 185, 187	520'-6"	8°	Meridional strain

Reformatted Per
Amendment 02-01

TABLE 3.8-7 (Continued)

STRAIN GAGE LOCATIONS FOR REACTOR BUILDING
STRUCTURAL ACCEPTANCE TEST

Instrument No. & Gage	Elevation	Azimuth	Notes
Redund. Strain Gage 1183, 1185, 1187	520'-6"	8°	Meridional strain 02-01
Main Strain Gage 184, 186, 188	520'-6"	8°	Hoop strain
Redund. Strain Gage 1184, 1186, 1188	520'-6"	8°	Hoop strain

Under Vertical Tendon Anchorage at 7° 32' 20"

Primary Strain Gage 189	Refer to Figure 3.8-33	Vertical
Redund. Strain Gage 1189	Refer to Figure 3.8-33	Vertical
Primary Strain Gage 190	Refer to Figure 3.8-33	Horizontal
Redund. Strain Gage 1190	Refer to Figure 3.8-33	Horizontal
Primary Strain Gage 191	Refer to Figure 3.8-33	45°
Redund. Strain Gage 1191	Refer to Figure 3.8-33	45°

Adjustment in location of instruments and gages may be made in the field to clear interferences.

TABLE 3.8-8

CRACK PATTERN AREA LOCATIONS FOR REACTOR BUILDING STRUCTURAL ACCEPTANCE TEST

Area No.	Center Line Elevation	Center Line Azimuth	Dimensions
145	~563'-9-1/2"	266°	7'-0" x ~41'-0"
146	488'-6"	266°	7'-0" x 7'-0"
147	415'-6"	266°	7'-0" x 7'-0"
148	483'-0"	~302°-27'	7'-0" x ~12'-2"
149	485'-0"	~111°-42'	25'-10" x 27'-2"

NOTE:

Adjustment in location of whitewash areas may be made in the field to clear interferences.

TABLE 3.8-9

THERMOCOUPLE LOCATIONS FOR REACTOR BUILDING STRUCTURAL ACCEPTANCE TEST

Thermocouple No.	Elevation	Azimuth	
150, 153	412'-0"	59°, 308°	
151, 152, 154	416'-0"	162°-30', 243°-20', 347°	
155, 156, 157, 158, 159	~483'-0"	59°, 162°-30', 243°-20', 308°, 347°	02-01
160, 161, 162, 163, 164	576'-0"	59°, 162°-30', 243°-20', 308°, 347°	

02-01

NOTE:

Adjustment in the location of thermocouples may be made in the field to clear interferences.

TABLE 3.8-10

LOAD COMBINATIONS FOR INTERNAL CONCRETE STRUCTURES

1. Load Combinations for Service Load Conditions

   a. $U = 1.4D + 1.7L$

   b. $U = 1.4D + 1.7L + 1.9E$

   c. $U = (0.75)\ (1.4D + 1.7L + 1.7T_o + 1.7R_o)$

   d. $U = (0.75)\ (1.4D + 1.7L + 1.9E + 1.7T_o + 1.7R_o)$

   e. $U = 1.2D + 1.9E$

   f. $U = 1.4D + 1.7L + 1.4F$

2. Load Combinations for Factored Load Conditions

   a. $U = D + L + T_o + R_o + E'$

   b. $U = D + L + T_o + R_o + 1.5P_o$

   c. $U = D + L + T_o + R_o + 1.25P_o + 1.0\ (Y_r + Y_j + Y_m) + 1.25E$

   d. $U = D + L + T_o + R_o + 1.0P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0E'$

“U” is the section strength required to resist the loads based upon the strength design methods of ACI 318-71.

TABLE 3.8-11

LOAD COMBINATIONS FOR INTERNAL STRUCTURAL STEEL STRUCTURES

1. Load Combinations for Service Load Conditions

   a. $S = D + L$

   b. $S = D + L + E$

   c. $1.5S = D + L + R_o + T_o + E$

2. Load Combinations for Factored Load Conditions

   a. $1.6S = D + L + T_o + R_o + E'$

   b. $1.6S = D + L + T_a + R_a + P_a$

   c. $1.6S = D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E$

   d. $1.7S = D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$

“S” is the required section strength based upon the elastic stress design method and allowable stresses defined in AISC, “Specification for the Design, Fabrication and Erection of Structural Steel for Buildings,” 1969 Edition.

TABLE 3.8-11a

POLYVINYL CHLORIDE WATERSTOPS

Building	Elevation	Expected Normal Operation Radiation Level	
Auxiliary Building (1)	386'-0"	$< 10^4$ to 10^6 Rad, TID (2), 40 yr	02-01
	410'-0"	$< 10^6$ Rad, TID, 40 yr	
	412'-0"	$< 10^4$ to 10^6 Rad, TID, 40 yr	
Control Building (1)	411'-0"	Negligible	
Intermediate Building (1)	407'-0"	$< 10^4$ Rad, TID, 40 yr	
	411'-0"	$< 10^4$ Rad, TID, 40 yr	
	412'-0"	$< 10^6$ Rad, TID, 40 yr	02-01
Fuel Handling Building (1)	412'-9"	$< 10^4$ to 10^6 Rad, TID, 40 yr	
	416'-8"	$< 10^4$ to 10^6 Rad, TID, 40 yr	
Diesel Generator Building (1)	400'-0"	Negligible	
	421'-0"	Negligible	
Reactor Building (3)	387'-0" to 387'-8"	$< 6.5 \times 10^3$ Rad, TID, 40 yr	02-01
	388'-0"	$< 6.5 \times 10^3$ Rad, TID, 40 yr	
	394'-10"	$< 6.5 \times 10^3$ Rad, TID, 40 yr	
	396'-0"	$< 6.5 \times 10^3$ Rad, TID, 40 yr	

NOTES:

1. Radiation levels are from Table 3.11-3. Shielding provided by concrete cover on waterstops is neglected.

2. TID - Total integrated dose.

3. Shielding provided by concrete cover on waterstops is considered.

TABLE 3.8-12

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR CONCRETE STRUCTURES

1. Load Combinations for Service Load Conditions

a. Load Combinations Considered when Working Stress Design Method is Used

		Acceptance Criteria
(1)	D + L	S [(1)]
(2)	D + L + E	S
(3)	D + L + W	S

When thermal stresses due to T_o and R_o are present, the following load combinations are also considered: 02-01

		Acceptance Criteria
(4)	$D + L + T_o + R_o$	1.3S
(5)	$D + L + T_o + R_o + E$	1.3S
(6)	$D + L + T_o + R_o + W$	1.3S

Both cases of L having its full value or being completely absent are checked. 02-01

b. Load Combinations Considered when Strength Design Method is Used

		Acceptance Criteria
(1)	1.4D + 1.7L	U [(2)]
(2)	1.4D + 1.7L + 1.9E	U
(3)	1.4D + 1.7L + 1.7W	U

TABLE 3.8-12 (Continued)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR CONCRETE STRUCTURES

When thermal stresses due to T_o and R_o are present, the following load combinations are also considered: 02-01

		Acceptance Criteria
(4)	$(0.75)\ (1.4D + 1.7L + 1.7T_o + 1.7R_o)$	U
(5)	$(0.75)\ (1.4D + 1.7L + 1.9E + 1.7T_o + 1.7R_o)$	U
(6)	$(0.75)\ (1.4D + 1.7L + 1.7W + 1.7T_o + 1.7R_o)$	U

Both cases of L having its full value or being completely absent area checked. In addition, the following load combinations are considered: 02-01

		Acceptance Criteria
(7)	$1.2D + 1.9E$	U
(8)	$1.2D + 1.7W$	U

2. Load Combinations for Factored Load Conditions

		Acceptance Criteria
a.	$D + L + T_o + R_o + E'$	U
b.	$D + L + T_o + R_o + Wt$	U
c.	$D + L + T_a + R_a + 1.5P_a$	U
d.	$D + L + T_a + R_a + 1.25P_a + 1.0(Y_j + Y_r + Y_m) + 1.25E$	U
e.	$D + L + T_a + R_a + 1.0P_a + 1.0(Y_j + Y_r + Y_m) + 1.0E'$	U

Both cases of L having its full value or being completely absent are checked.

TABLE 3.8-12 (Continued)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR CONCRETE STRUCTURES

NOTES:

(1) S - For concrete structures, S is the required section strength based upon the working stress design method and the allowable stresses defined in Section 8.10 of ACI 318-71. Increases in allowable stresses for concrete due to seismic or wind loadings are not used.

(2) U - For concrete structures, U is the section strength required to resist design loads based upon the strength design methods described in ACI 318-71.

TABLE 3.8-13

LOADS AND LOAD COMBINATIONS FOR THE SERVICE WATER INTAKE AND DISCHARGE STRUCTURES

1. Definitions and Nomenclature

D = dead weight of concrete and weight of soil above the Intake Structure

E = is the Operating Basis Earthquake

E' = is the Safe Shutdown Earthquake

H = is the lateral earth pressure

F = hydrostatic pressure

2. Load Combinations

U is defined as required strength to resist design loads.

a. End of Construction
U = 1.4D + 1.7H
U = 0.9D + 1.7H, where D reduces the effect of H 02-01

b. Operating Condition
U = 1.4D + 1.7H
U = 1.4D + 1.7H + 1.9E
U = 0.9D + 1.7H, where D reduces the effect of H 02-01
U = 1.4D + 1.4F + 1.7H
U = 1.4D + 1.4F + 1.7H + 1.87E (Critical Loading Combination)
U = 0.9D + 1.87E

c. Extreme Environmental Conditions
U = D + H + E'
U = D + H
U = D + H + 1.25E + $F^{(1)}$
U = 0.9D + H + E' + $F^{(1)}$, where D reduces effect of H 02-01

NOTE:
(1) F for discharge structure only.

TABLE 3.8-14

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR STEEL STRUCTURES

1. Load Combinations for Service Load Conditions

a. Load Combinations Considered when Elastic Working Stress Design Methods are Used

		Acceptance Criteria
(1)	D + L	S [(1)]
(2)	D + L + E	S
(3)	D + L + W	S

When thermal stresses due to To and Ro are present, the following load combinations are also considered: 02-01

		Acceptance Criteria
(4)	$D + L + T_o + R_o$	1.5S
(5)	$D + L + T_o + R_o + E$	1.5S
(6)	$D + L + T_o + R_o + W$	1.5S

Both cases of L having its full value or being completely absent are checked. 02-01

b. Load Combinations Considered when Plastic Design Methods are Used

		Acceptance Criteria
(1)	1.7D + 1.7L	Y [(2)]
(2)	1.7D + 1.7L + 1.7E	Y
(3)	1.7D + 1.7L + 1.7W	Y

When thermal stresses due to To and Ro are present, the following load combinations are also considered: 02-01

		Acceptance Criteria
(4)	$1.3 (D + L + T_o + R_o)$	Y
(5)	$1.3 (D + L + E +T_o + R_o)$	Y
(6)	$1.3 (D + L + W +T_o + R_o)$	Y

Both cases of L having its full value or being completely absent are checked. 02-01

TABLE 3.8-14 (Continued)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR STEEL STRUCTURES

2. Load Combinations for Factored Load Combinations 02-01

a. Load Combinations Considered when Elastic Working Stress Design Methods are Used. (3)

	Acceptance Criteria
(1) $D + L + T_o + R_o + E'$	1.6S
(2) $D + L + T_o + R_o + W_t$	1.6S
(3) D + L + Ta + Ra + Pa	1.6S
(4) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E$	1.6S
(5) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E'$	1.7S

b. Load Combinations Considered when Plastic Design Methods are Used

	Acceptance Criteria
(1) $D + L + T_o + R_o + E'$	0.9Y
(2) $D + L + T_o + R_o + W_t$	0.9Y
(3) $D + L + T_a + R_a + 1.5P_a$	0.9Y
(4) $D + L + T_a + R_a + 1.25P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.25E$	0.9Y
(5) $D + L + T_a + R_a + 1.0P_a + 1.0\ (Y_j + Y_r + Y_m) + E'$	0.9Y

TABLE 3.8-14 (Continued)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA
FOR STEEL STRUCTURES

NOTES:

(1) S - For structural steel, S is the required section strength based upon the elastic design methods and the allowable stresses defined in Part I of the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1969 Edition. Increases in allowable stresses for steel due to seismic or wind loadings are not used.

(2) Y - For structural steel, Y is the required section strength required to resist design loads and is based upon plastic design methods described in Part 2 of the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1969 Edition.

(3) For these combinations, the plastic modulus of steel shapes is used in computing the required section strength, S.

TABLE 3.8-15

LOAD COMBINATIONS USED TO CHECK AGAINST SLIDING, OVERTURNING AND FLOTATION

Load Combination	Minimum Factors of Safety Overturning	Sliding	Flotation
1. D + H + E	1.5	1.5	-
2. D + H + W	1.5	1.5	-
3. D + H + E'	1.1	1.1	-
4. D + H + W_t	1.1	1.1	-
5. D + F	-	-	1.5

TABLE 3.8-16

CONCRETE
COMPARISON OF ALLOWABLE COMPRESSIVE STRESSES

	Service		Factored	
	Membrane	Membrane plus Bending	Membrane	Membrane plus Bending
Section 3.8.1.5	$0.4f_{c'}$ [(1)]	$0.60f_{c'}$ [(1)]	$0.765f_{c'}$ [(2)]	$0.765f_{c'}$ [(2)]
ASME Code	Table CC-3431-1		Table CC-3421-1	

NOTES:

1. Section 3.8.1.5.1.2, Items 1 and 2.

2. Section 3.8.1.5.1.1, Item 1.

TABLE 3.8-17

REINFORCING STEEL
COMPARISON OF ALLOWABLE STRESSES AND STRAINS

	Service		Factored	
	Stress	Strain	Stress	Strain
Section 3.8.1.5	24 ksi (1)	-	$0.9f_y$ (2)	$0.9\varepsilon_y$ (2)
ASME Code	30 ksi	-	$0.9f_y$	$\geq \varepsilon_y$ (3)

NOTES:

1. Section 3.8.1.5.1.2, Item 7.

2. Section 3.8.1.5.1.1, Item 1. Note that the FSAR strain limit of $1.5\varepsilon_y$ results from Reference 3 of Section 3.8.1.2.1. However, due to use of a linear concrete stress distribution in conjunction with $0.9f_y$, the reinforcing steel strain is actually limited to $0.9\varepsilon_y$.

3. ASME Code CC-3422.1.

TABLE 3.8-18

LOAD COMBINATION ALLOWABLE STRESSES
FROM ASME CODE TABLES CC 3421-1 and CC 3431-1

		CODE ALLOWABLE CONCRETE COMPRESSION STRESSES (PSI)				
		AT DISCONTINUITIES - BASE AND SPRING LINE OF WALL		AWAY FROM DISCONTINUITIES - WALL MID-HEIGHT AND DOME APEX		02-01
LOAD COMBINATION	TYPE	MEMBRANE	MEMBRANE + BENDING	MEMBRANE	MEMBRANE + BENDING	
INITIAL PRESTRESS	S	0.35 fc' = 1750	0.60 fc' = 3000	0.35 fc' = 1750	0.45 fc' = 2250	
$D + F_I$						02-01
NORMAL WINTER OPERATION						
D + F	S	0.30 fc' = 1500	0.60 fc' = 3000	0.30 fc' = 1500	0.45 fc' = 2250	
$D + F + P_v$	S	0.30 fc' = 1500	0.60 fc' = 3000	0.30 fc' = 1500	0.45 fc' = 2250	02-01
$D + F + T_o$	S	0.45 fc' = 2250	0.60 fc' = 3000	0.45 fc' = 2250	0.60 fc' = 3000	
$D + F + T_o + P_v$	S	0.45 fc' = 2250	0.60 fc' = 3000	0.45 fc' = 2250	0.60 fc' = 3000	
SEVERE ENVIRONMENTAL						
$D + F + E_o$	S	0.40 fc' = 2000	0.60 fc' = 3000	0.40 fc' = 2000	0.45 fc' = 2250	
$D + F + E_o + P_v$	S	0.40 fc' = 2000	0.60 fc' = 3000	0.40 fc' = 2000	0.45 fc' = 2250	02-01
$D + F + E_o + T_o$	S	0.45 fc' = 2250	0.60 fc' = 3000	0.45 fc' = 2250	0.60 fc' = 3000	
$D + F + E_o + T_o + P_v$	S	0.45 fc' = 2250	0.60 fc' = 3000	0.45 fc' = 2250	0.60 fc' = 3000	
EXTREME ENVIRONMENTAL						
$D + F + E_{ss}$	U	0.60 fc' = 3000	0.85 fc' = 4250	0.60 fc' = 3000	0.75 fc' = 3750	
$D + F + E_{ss} + P_v$	U	0.60 fc' = 3000	0.85 fc' = 4250	0.60 fc' = 3000	0.75 fc' = 3750	02-01
$D + F + E_{ss} + T_o$	U	0.75 fc' = 3750	0.85 fc' = 4250	0.75 fc' = 3750	0.85 fc' = 4250	
$D + F + E_{ss} + T_o + P_v$	U	0.75 fc' = 3750	0.85 fc' = 4250	0.75 fc' = 3750	0.85 fc' = 4250	
ABNORMAL/EXTREME ENVIRONMENTAL						
$D + F + E_o + 0.5E_{ss} + W$	U	0.60 fc' = 3000	0.85 fc' = 4250	0.60 fc' = 3000	0.75 fc' = 3750	02-01
$D + F + E_o + 0.5E_{ss} + W + T_o$		0.75 fc’ = 3750	0.85 fc' = 4250	0.75 fc' = 3750	0.85 fc' = 4250	

F_I IS INITIAL PRESTRESS; F IS PRESTRESS AT STARTUP

TABLE 3.8-19

COMPARISON OF PREDICTED STRESSES AND STRAINS WITH ASME CODE ALLOWABLES AT BASE

		CONCRETE STRESSES (PSI)								MEMBRANE LINER STRESSES (PSI)		LINER STRAINS (MICRO IN/IN)			
		CODE ALLOWABLE		PREDICTED								CODE ALLOW.	PREDICTED MEMBRANE		
				MERIDIONAL			HOOP								
LOAD COMBINATION	TYPE	MEM-BRANE	MEMB. +BEND	I.F.	O.F.	MEMB.	I.F.	O.F.	MEMB.	MERID-IONAL	HOOP	MEM-BRANE	MERID-IONAL	HOOP	
INITIAL PRESTRESS															
D+F	S	-1750	-3000	-1477	391	-543	-252	67	-93	-17982	-4855	-2000	-572	12	02-01
NORMAL WINTER OPERATION															
D+F	S	-1500	-3000	-1368	330	-519	-234	56	-94	-16658	-4499	-2000	-529	11	02-01
$D+F+P_v$	S	-1500	-3000	-1430	360	-539	-245	61	-92	-17124	-4624	-2000	-544	12	
$D+F+T_o$	S	-2250	-3000	-2153	1177	-488	-1492	944	-274	-25558	-16516	-2000	-716	-314	
$D+F+T_o+P_v$	S	-2250	-3000	-2215	1207	-504	-1503	949	-277	-26024	-16641	-2000	-731	-314	
SEVERE ENVIRONMENTAL															
$D+F+E_o$	S	-2000	-3000	-1344	-78	-711	-230	-13	-122	-16475	-4449	-2000	-524	11	02-01
$D+F+E_o+P_v$	S	-2000	-3000	-1406	-48	-727	-241	-8	-125	-16941	-4574	-2000	-538	12	
$D+F+E_o+T_o$	S	-2250	-3000	-2129	739	-695	-1388	875	-257	-25375	-16466	-2000	-710	-314	
$D+F+E_o+T_o+P_v$	S	-2250	-3000	-2191	799	-696	-1499	880	-310	-25841	-16591	-2000	-725	-314	
EXTREME ENVIRONMENTAL															
$D+F+E_{ss}$	U	-3000	-4250	-1332	-282	-807	-228	-48	-138	-16485	-4424	-5000	-524	12	02-01
$D+F+E_{ss}+P_v$	U	-3000	-4250	-1394	-252	-823	-239	-48	-141	-16849	-4549	-5000	-536	12	
$D+F+_{ss}+T_o$	U	-3750	-4250	-2117	565	-776	-1486	840	-323	-25283	-16441	-5000	-707	-314	
$D+F+E_{ss}+T_o+P_v$	U	-3750	-4250	-2179	595	-792	-1497	845	-326	-25749	-16566	-5000	-713	-314	
ABNORMAL/EXTREME ENVIRONMENTAL															
$D+F+E_o+ 0.5E_{ss}+W$	U	-3000	-4250	-1326	-386	-856	-227	-66	-147	-16336	-4411	-5000	-519	11	02-01
$D+F+E_o+ 0.5E_{ss}+W+T_o$	U	-3750	-4250	-2111	-461	-825	-1485	822	-332	-25236	-16428	-5000	-706	-334	

TABLE 3.8-20

COMPARISON OF PREDICTED STRESSES AND STRAINS WITH ASME CODE ALLOWABLES AT WALL MID-HEIGHT

		CONCRETE STRESSES (PSI)								MEMBRANE LINER STRESSES (PSI)		LINER STRAINS (MICRO IN/IN)			
		CODE ALLOWABLE		PREDICTED								CODE ALLOW.	PREDICTED MEMBRANE		
				MERIDIONAL			HOOP								
LOAD COMBINATION	TYPE	MEM-BRANE	MEMB. +BEND	I.F.	O.F.	MEMB.	I.F.	O.F.	MEMB.	MERID-IONAL	HOOP	MEM-BRANE	MERID-IONAL	HOOP	
INITIAL PRESTRESS															
D+F	S	-1750	-2250	-791	-757	-774	-1445	-1439	-1442	-11244	-18237	-2000	-205	-516	02-01
NORMAL WINTER OPERATION															
D+F	S	-1500	-2250	-751	-720	-736	-1335	-1329	-1332	-10625	-16871	-2000	-198	-476	02-01
$D+F+P_v$	S	-1500	-2250	-777	-744	-761	-1387	-1370	-1379	-10853	-17270	-2000	-202	-487	
$D+F+T_o$	S	-2250	-3000	-1821	462	-680	-2407	-148	-1278	-21417	-27675	-2000	-462	-740	
$D+F+T_o+P_v$	S	-2250	-3000	-1847	438	-705	-2459	-199	-1329	-21648	-28074	-2000	-466	-752	
SEVERE ENVIRONMENTAL															
$D+F+E_o$	S	-2000	-2250	-890	-865	-878	-1319	-1311	-1315	-11637	-16940	-2000	-232	-468	02-01
$D+F+E_o+P_v$	S	-2000	-2250	-916	-889	-903	-1371	-1352	-1362	-11865	-17339	-2000	-235	-479	
$D+F+E_o+T_o$	S	-2250	-3000	-1960	317	-822	-2391	-130	-1261	-22429	-27744	-2000	-496	-732	
$D+F+E_o+T_o+P_v$	S	-2250	-3000	-1962	293	-834	-2443	-181	-1312	-22657	-28143	-2000	-500	-744	
EXTREME ENVIRONMENTAL															
$D+F+E_{ss}$	U	-3000	-3750	-960	-938	-949	-1311	-1302	-1309	-12143	-16975	-5000	-249	-464	02-01
$D+F+E_{ss}+P_v$	U	-3000	-3750	-986	-962	-974	-1363	-1353	-1358	-12371	-17374	-5000	-253	-475	
$D+F+_{ss}+T_o$	U	-3750	-4250	-2030	244	-893	-2383	-121	-1252	-22995	-27779	-5000	-515	-728	
$D+F+E_{ss}+T_o+P_v$	U	-3750	-4250	-2056	220	-918	-2435	-172	-1304	-23163	-28178	-5000	-517	-740	
ABNORMAL/EXTREME ENVIRONMENTAL															
$D+F+E_o+ 0.5E_{ss}+W$	U	-3000	-3750	-996	-975	-986	-1307	-1297	-1302	-12402	-16993	-5000	-258	-462	02-01
$D+F+E_o+ 0.5E_{ss}+W+T_o$	U	-3750	-4250	-2066	207	-930	-2379	-116	-1248	-23193	-27797	-5000	-522	-726	

TABLE 3.8-21

COMPARISON OF PREDICTED STRESSES AND STRAINS WITH ASME CODE ALLOWABLES AT WALL SPRING LINE

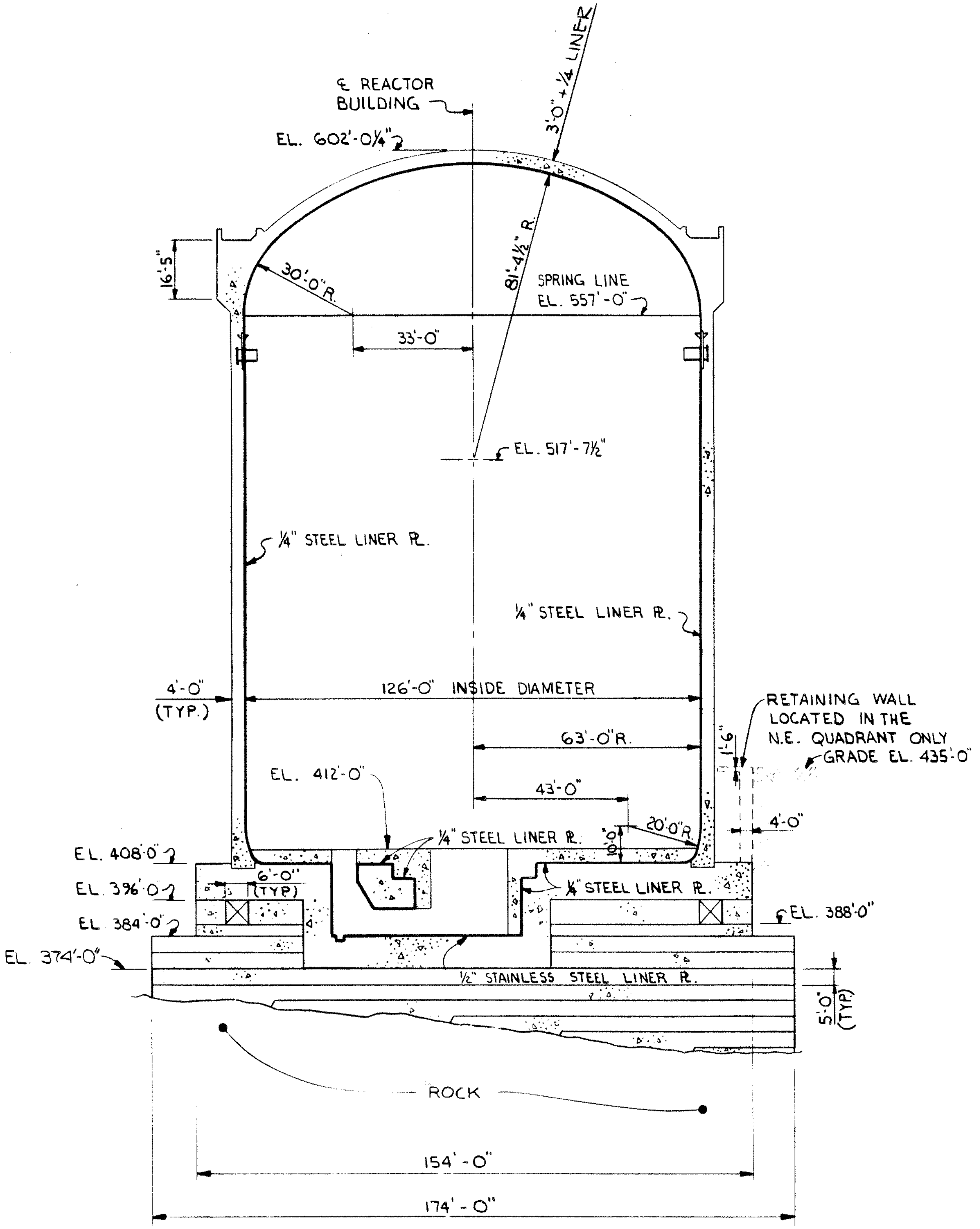
		CONCRETE STRESSES (PSI)								MEMBRANE LINER STRESSES (PSI)		LINER STRAINS (MICRO IN/IN)			
		CODE ALLOWABLE		PREDICTED								CODE ALLOW. MEMBRANE	PREDICTED		02-01
				MERIDIONAL			HOOP								
LOAD COMBINATION	TYPE	MEMBRANE	MEMB. +BEND	I.F.	O.F.	MEMB.	I.F.	O.F.	MEMB.	MERIDIONAL	HOOP	MEMBRANE	MERIDIONAL	HOOP	
INITIAL PRESTRESS															
D+F	S	-1750	-3000	-1101	-264	-683	-670	-606	-638	-14023	-9373	-2000	-390	-183	02-01
NORMAL WINTER OPERATION															
D+F	S	-1500	-3000	-1034	-255	-645	-617	-557	-587	-13149	-8656	-2000	-367	-167	02-01
D+F+P_v	S	-1500	-3000	-1071	-268	-670	-639	-575	-607	-13456	-8846	-2000	-376	-170	
D+F+T_o	S	-2250	-3000	-2107	932	-588	-1702	612	-545	-24043	-19625	-2000	-633	-436	
D+F+T_o+P_v	S	-2250	-3000	-2144	919	-613	-1724	594	-565	-24330	-19815	-2000	-641	-440	
SEVERE ENVIRONMENTAL															
D+F+E_o	S	-2000	-3000	-1073	-271	-672	-586	-521	-554	-13410	-8459	-2000	-378	-158	02-01
D+F+E_o+P_v	S	-2000	-3000	-1110	-284	-697	-608	-539	-574	-13637	-8649	-2000	-384	-162	
D+F+E_o+T_o	S	-2250	-3000	-2146	916	-615	-1671	648	-512	-24304	-19428	-2000	-644	-427	
D+F+E_o+T_o+P_v	S	-2250	-3000	-2177	903	-637	-1693	630	-532	-24591	-19618	-2000	-652	-431	
EXTREME ENVIRONMENTAL															
D+F+E_{ss}	U	-3000	-4250	-1093	-279	-686	-570	-503	-537	-13541	-8360	-5000	-383	-153	02-01
D+F+E_{ss}+P_v	U	-3000	-4250	-1130	-292	-711	-592	-521	-557	-13828	-8380	-5000	-391	-158	
D+F+$_{ss}$+T_o	U	-3750	-4250	-2166	908	-629	-1656	666	-495	-24435	-19329	-5000	-649	-422	
D+F+E_{ss}+T_o+P_v	U	-3750	-4250	-2203	895	-634	-1677	648	-515	-24712	-19519	-5000	-657	-426	
ABNORMAL/EXTREME ENVIRONMENTAL															
D+F+E_o+ 0.5E_{ss}+W	U	-3000	-4250	-1103	-283	-693	-562	-494	-528	-13608	-8310	-5000	-386	-150	02-01
D+F+E_o+ 0.5E_{ss}+W+T_o	U	-3750	-4250	-2176	904	-636	-1647	675	-486	-24502	-19279	-5000	-652	-420	

TABLE 3.8-22

COMPARISON OF PREDICTED STRESSES AND STRAINS WITH ASME CODE ALLOWABLES AT DOME APEX

		CONCRETE STRESSES (PSI)								MEMBRANE LINER STRESSES (PSI)		LINER STRAINS (MICRO IN/IN)			02-01
		CODE ALLOWABLE		PREDICTED								CODE ALLOW.	PREDICTED MEMBRANE		
				MERIDIONAL			HOOP								
LOAD COMBINATION	TYPE	MEM-BRANE	MEMB. +BEND	I.F.	O.F.	MEMB.	I.F.	O.F.	MEMB.	MERID-IONAL	HOOP	MEM-BRANE	MERID-IONAL	HOOP	
INITIAL PRESTRESS															
D+F	S	-1750	-2250	-1380	-1508	-1444	-1470	-1494	-1482	-18295	-19272	-2000	-438	-482	02-01
NORMAL WINTER OPERATION															
D+F	S	-1500	-2250	-1293	-1412	-1353	-1377	-1399	-1388	-17132	-18046	-2000	-410	-451	02-01
$D+F+P_v$	S	-1500	-2250	-1339	-1452	-1396	-1422	-1443	-1433	17503	-18410	-2000	-419	-459	
$D+F+T_o$	S	-2250	-3000	-2345	-212	-1279	-2432	-198	-1315	-27781	-28713	-2000	-671	-712	
$D+F+T_o+P_v$	S	-2250	-3000	-2391	-252	-1322	-2447	-242	-1344	-28152	-29077	-2000	-680	-721	
SEVERE ENVIRONMENTAL															
$D+F+E_o$	S	-2000	-2250	-1293	-1413	-1353	-1356	-1372	-1364	-17119	-17895	-2000	-411	-446	02-01
$D+F+E_o+P_v$	S	-2000	-2250	-1339	-1453	-1396	-1401	-1416	-1409	-17496	-18259	-2000	-421	-455	
$D+F+E_o+T_o$	S	-2250	-3000	-2345	-213	-1279	-2401	-171	-1286	-27768	-28562	-2000	-672	-707	
$D+F+E_o+T_o+P_v$	S	-2250	-3000	-2391	-253	-1322	-2456	-215	-1336	-28079	-28926	-2000	-679	-717	
EXTREME ENVIRONMENTAL															
$D+F+E_{ss}$	U	-3000	-3750	-1293	-1414	-1354	-1345	-1358	-1352	-17112	-17819	-5000	-412	-443	02-01
$D+F+E_{ss}+P_v$	U	-3000	-3750	-1339	-1454	-1397	-1390	-1402	-1396	-17423	-18123	-5000	-419	-453	
$D+F+_{ss}+T_o$	U	-3750	-4250	-2345	-214	-1280	-2400	-159	-1279	-27761	-28486	-5000	-672	-704	
$D+F+E_{ss}+T_o+P_v$	U	-3750	-4250	-2391	-254	-1323	-2445	-201	-1323	-28132	-28850	-5000	-682	-713	
ABNORMAL/EXTREME ENVIRONMENTAL															
$D+F+E_o+ 0.5E_{ss}+W$	U	-3000	-3750	-1293	-1414	-1354	-1340	-1351	-1346	-17109	-17780	-5000	-412	-442	02-01
$D+F+E_o+ 0.5E_{ss}+W+T_o$	U	-3750	-4250	-2345	-214	-1280	-2395	-150	-1273	-27758	-28447	-5000	-672	-703	

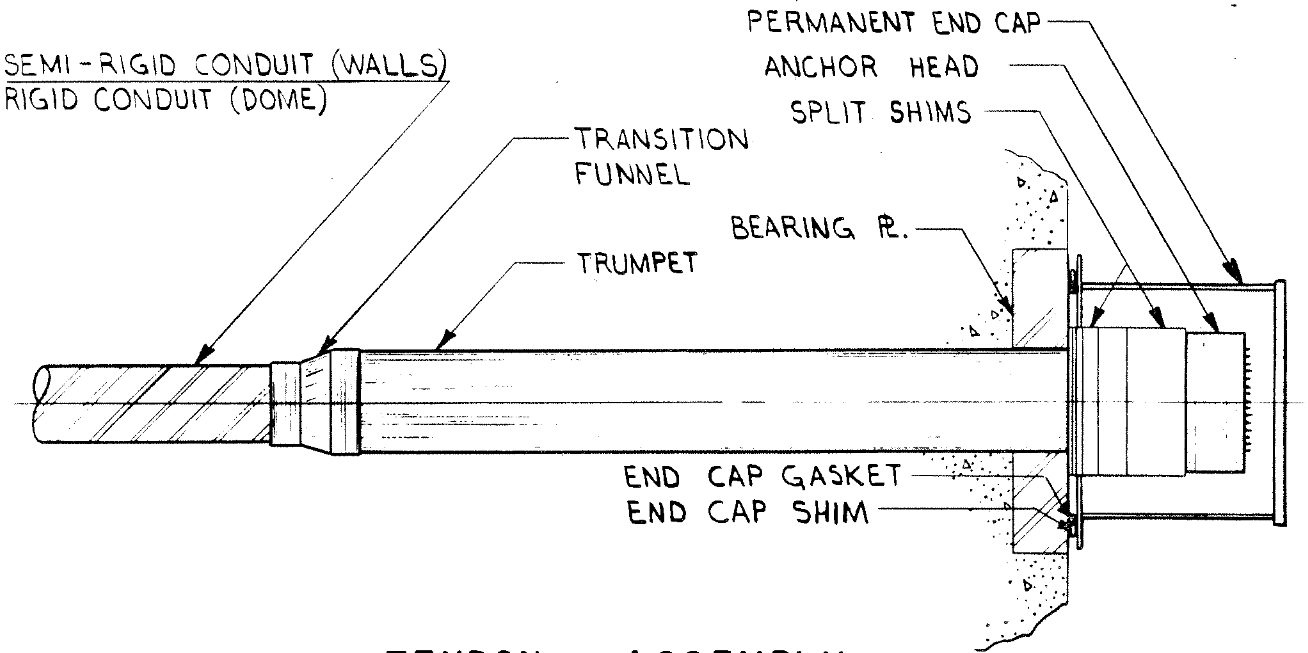
Reformatted Per
Amendment 02-01



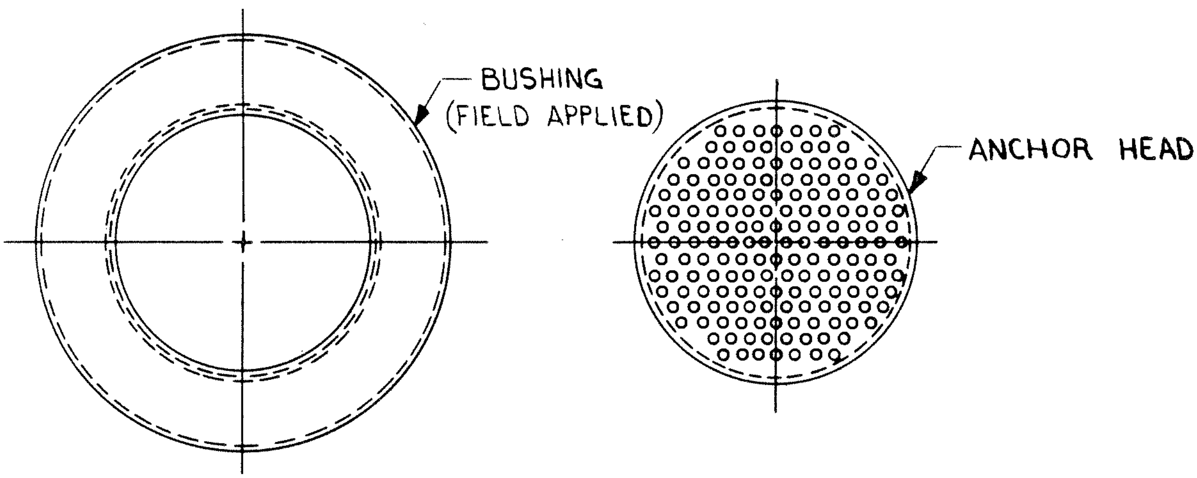
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building
and Foundation

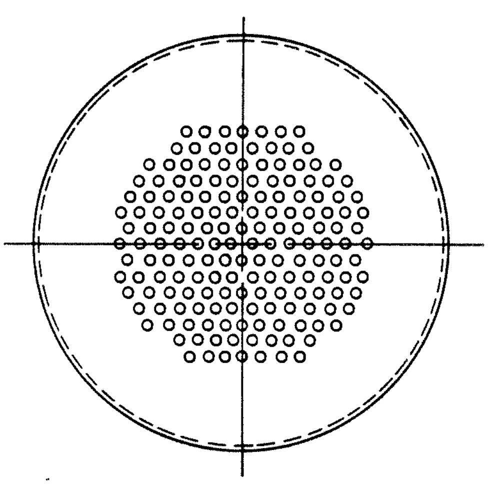
Figure 3.8-1



TENDON ASSEMBLY



SHOP ANCHOR HEAD ASSEMBLY

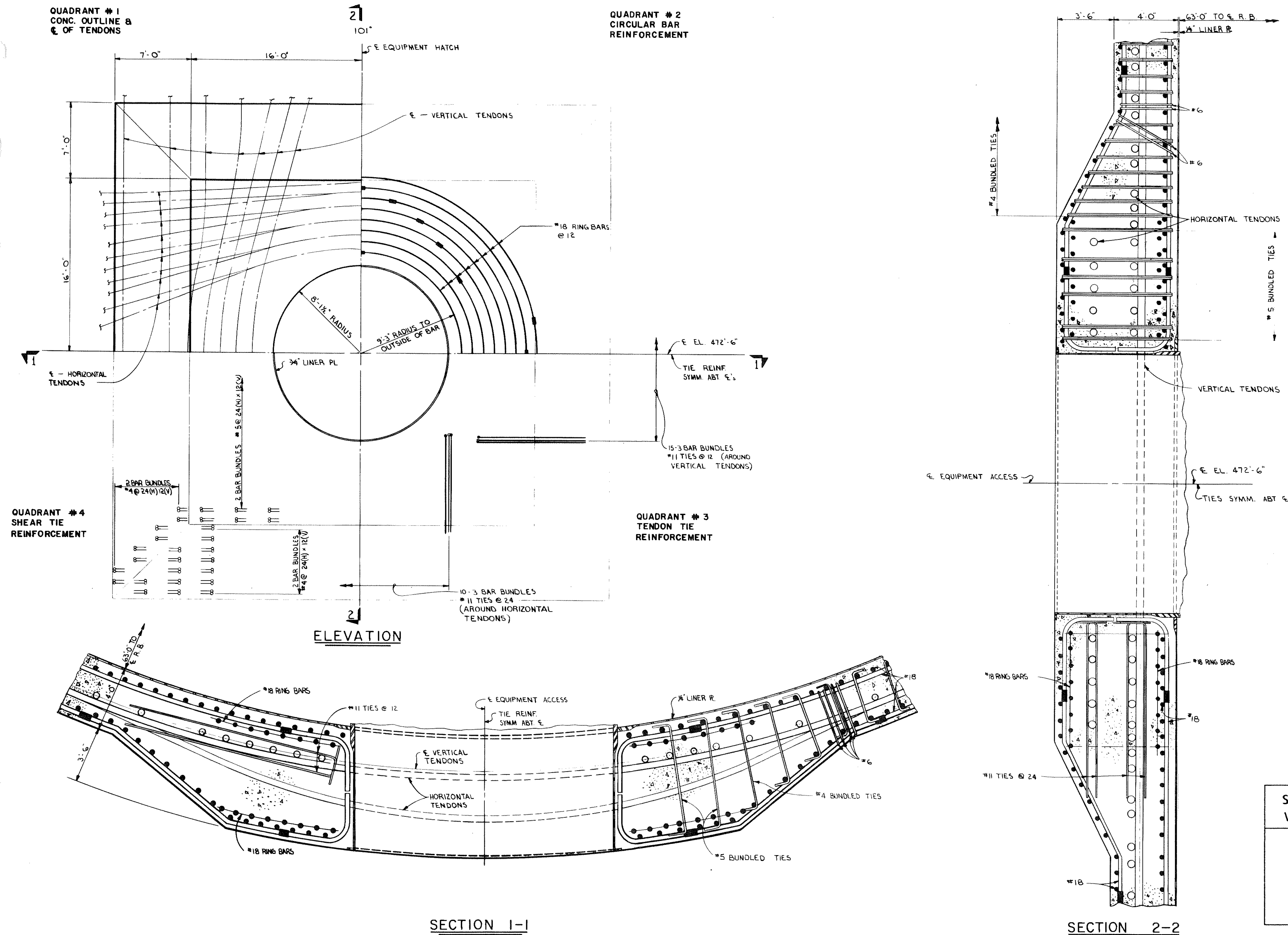


FIELD ANCHOR HEAD

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Tendon Components

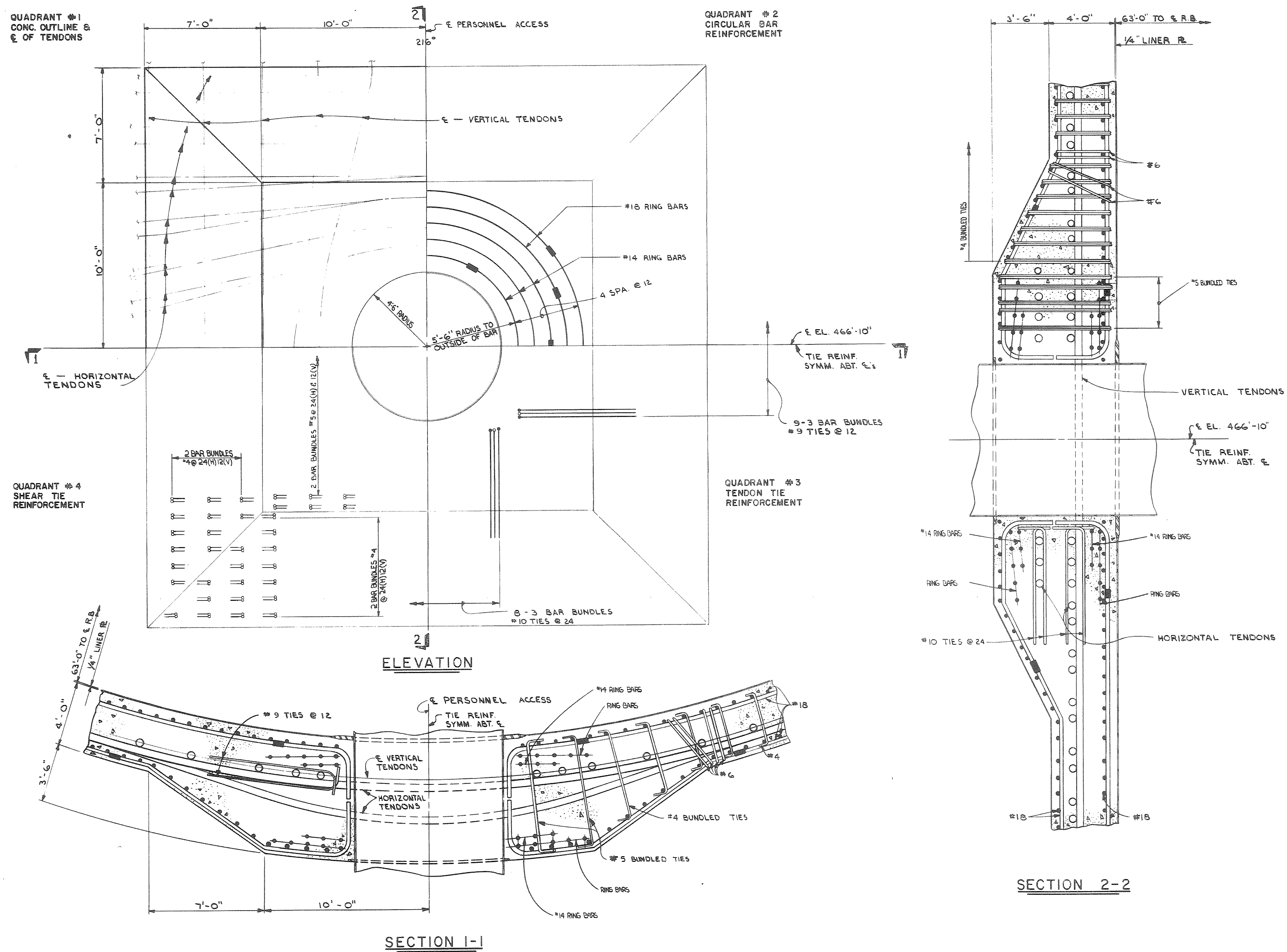
Figure 3.8-2



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Equipment Opening

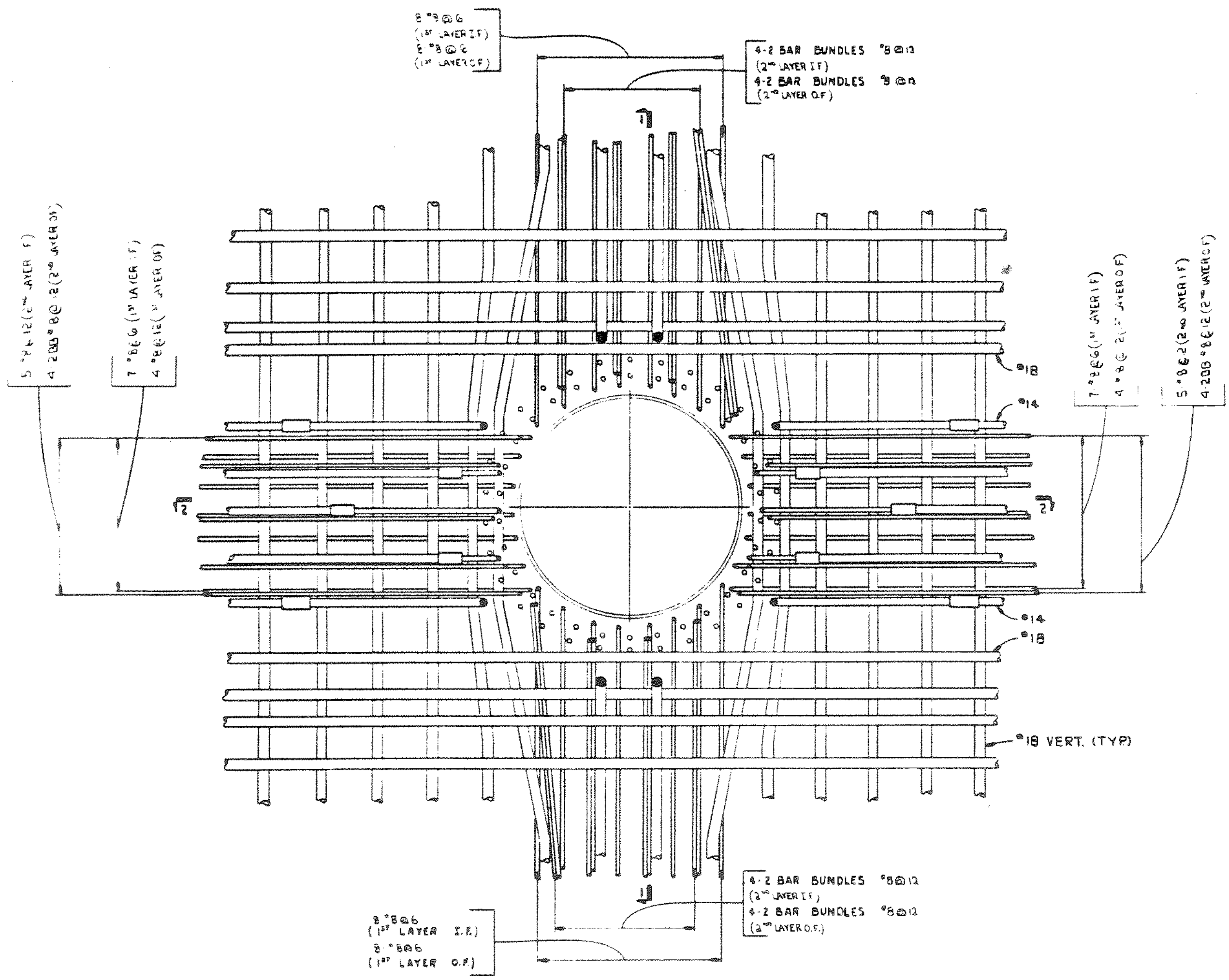
Figure 3.8-3



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Personnel Opening

Figure 3.8-4

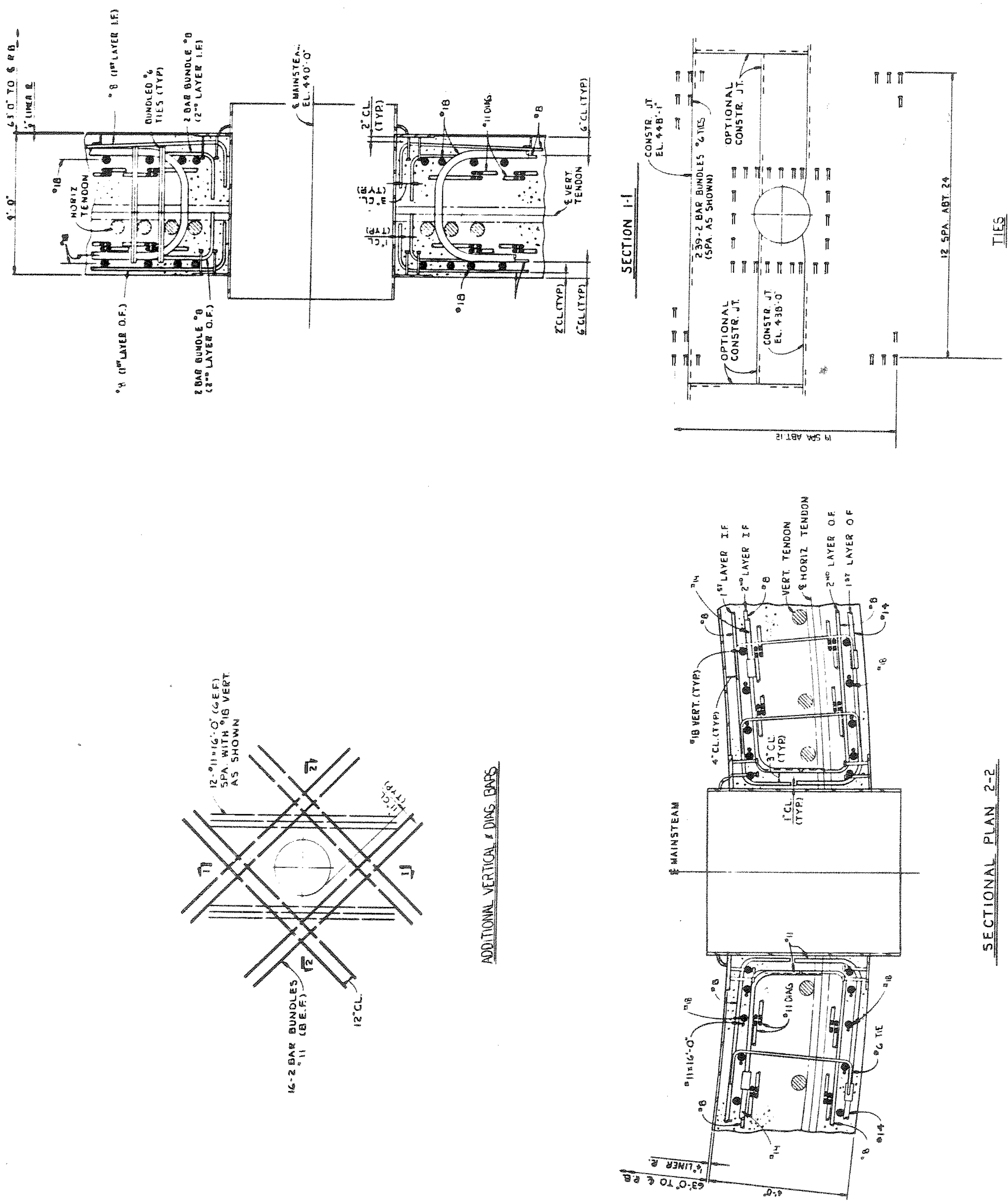


INSIDE FACE ELEVATION

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam
Penetration

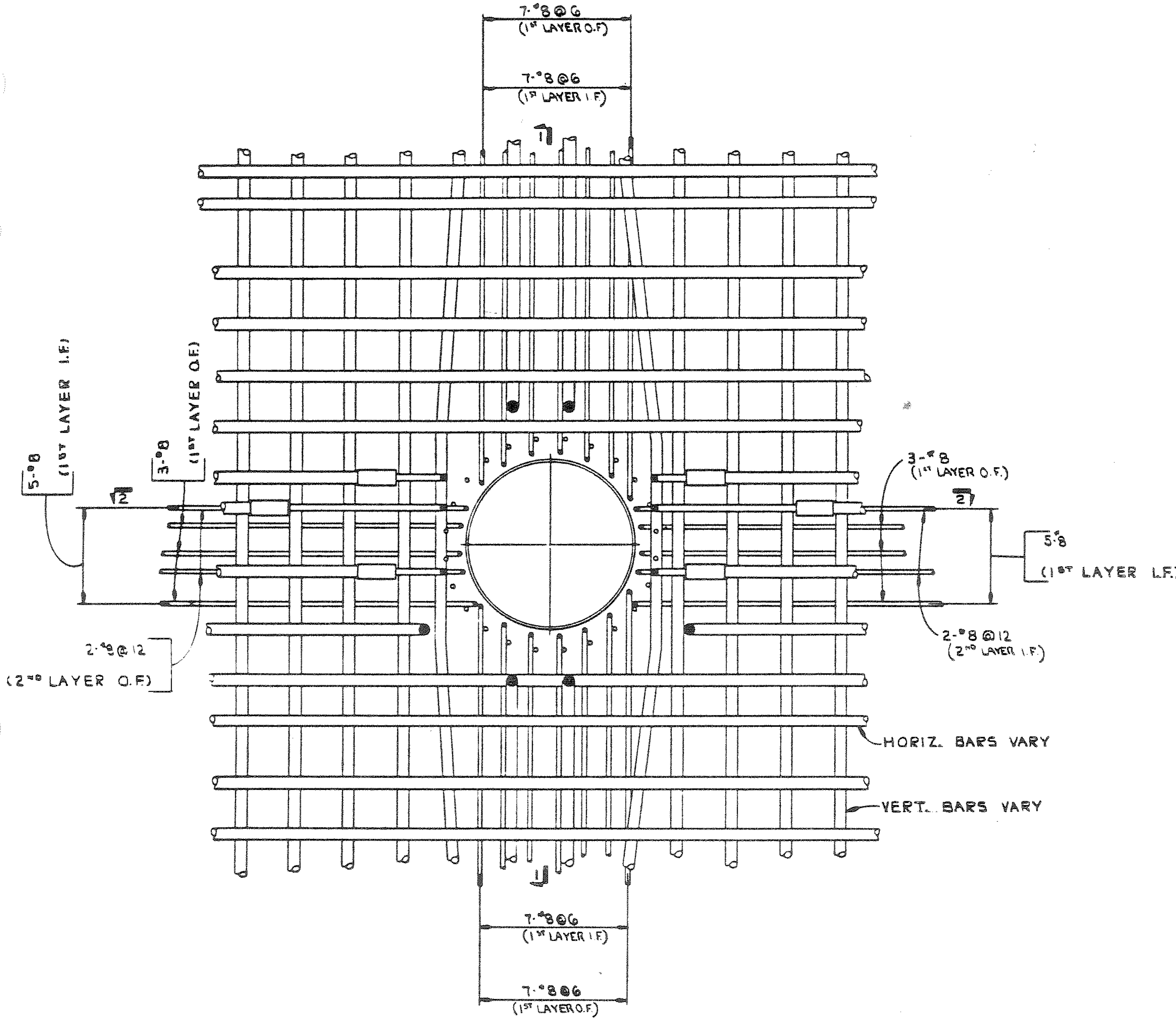
Figure 3.8-5



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam
Penetration

Figure 3.8-6

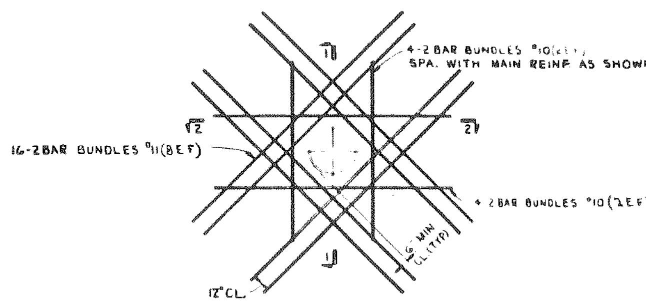


INSIDE FACE ELEVATION

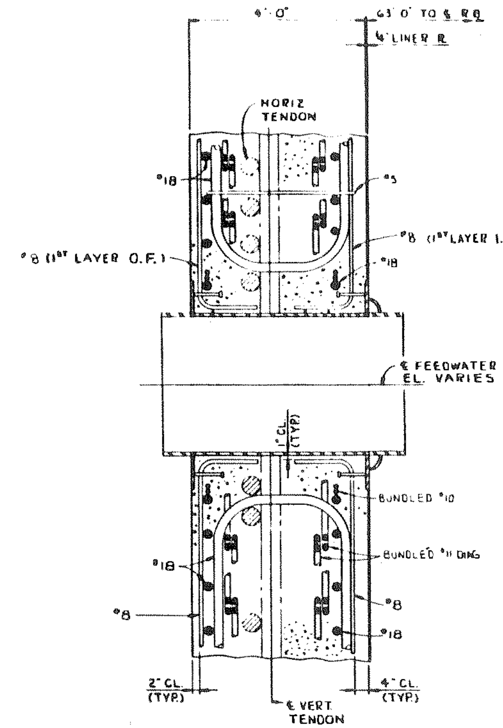
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Feedwater
Penetration

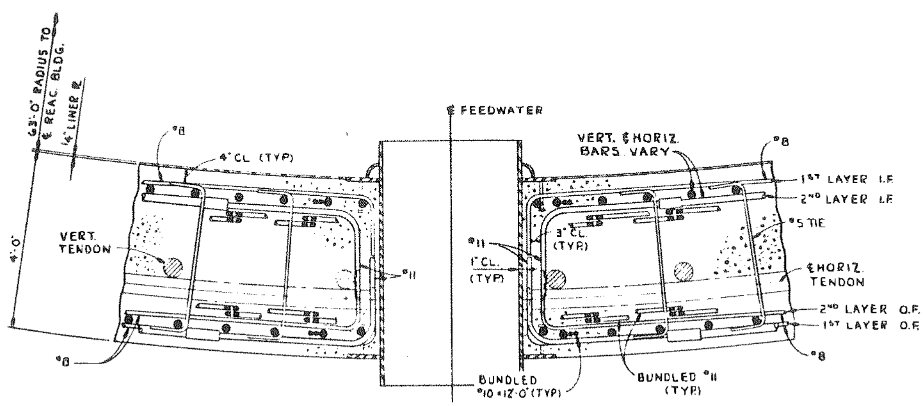
Figure 3.8-7



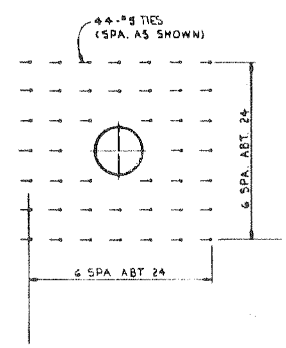
ADDITIONAL VERTICAL & DIAG BARS



SECTION 1-1



SECTIONAL PLAN 2-2

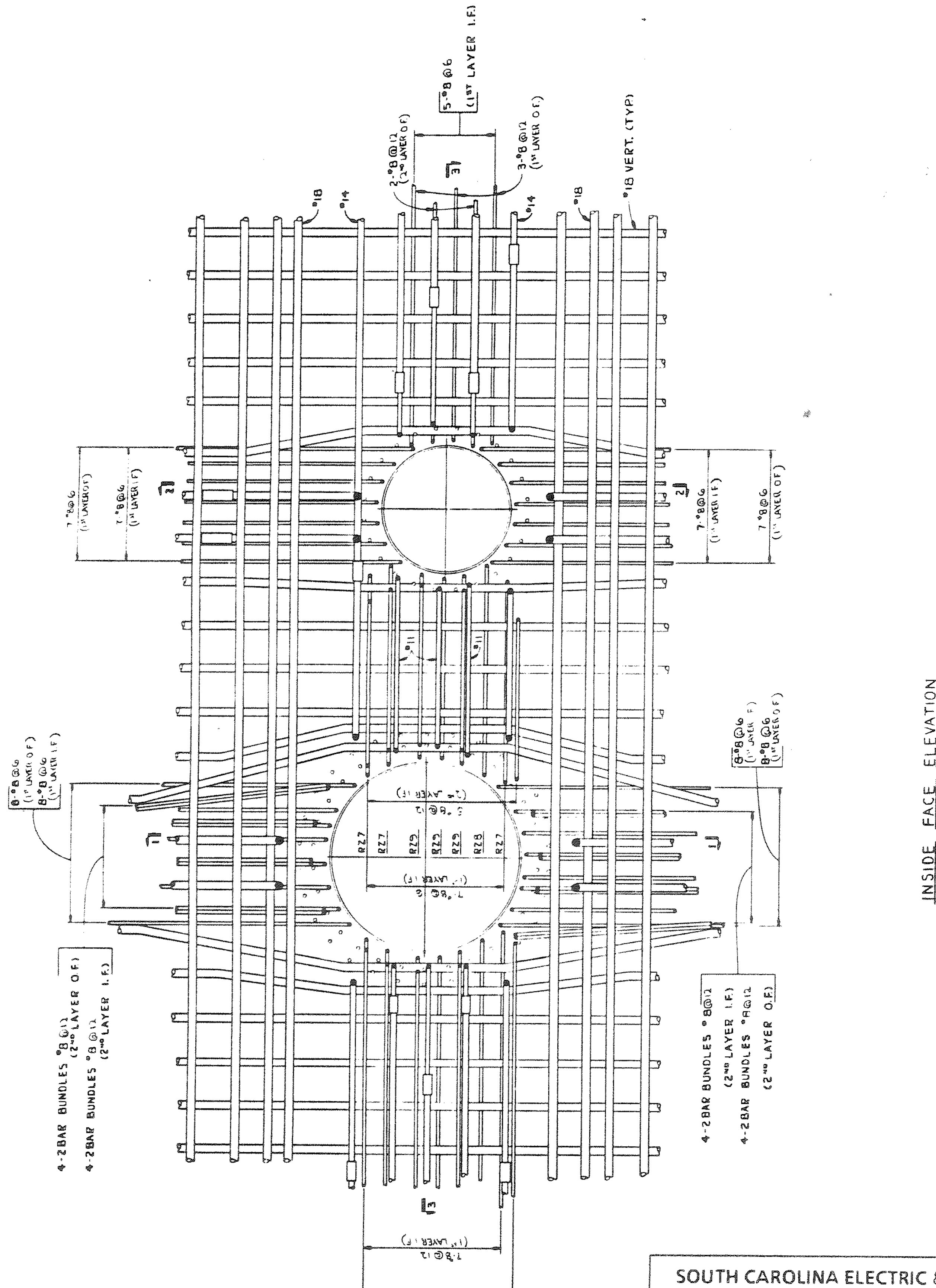


TIES

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Feedwater
Penetration

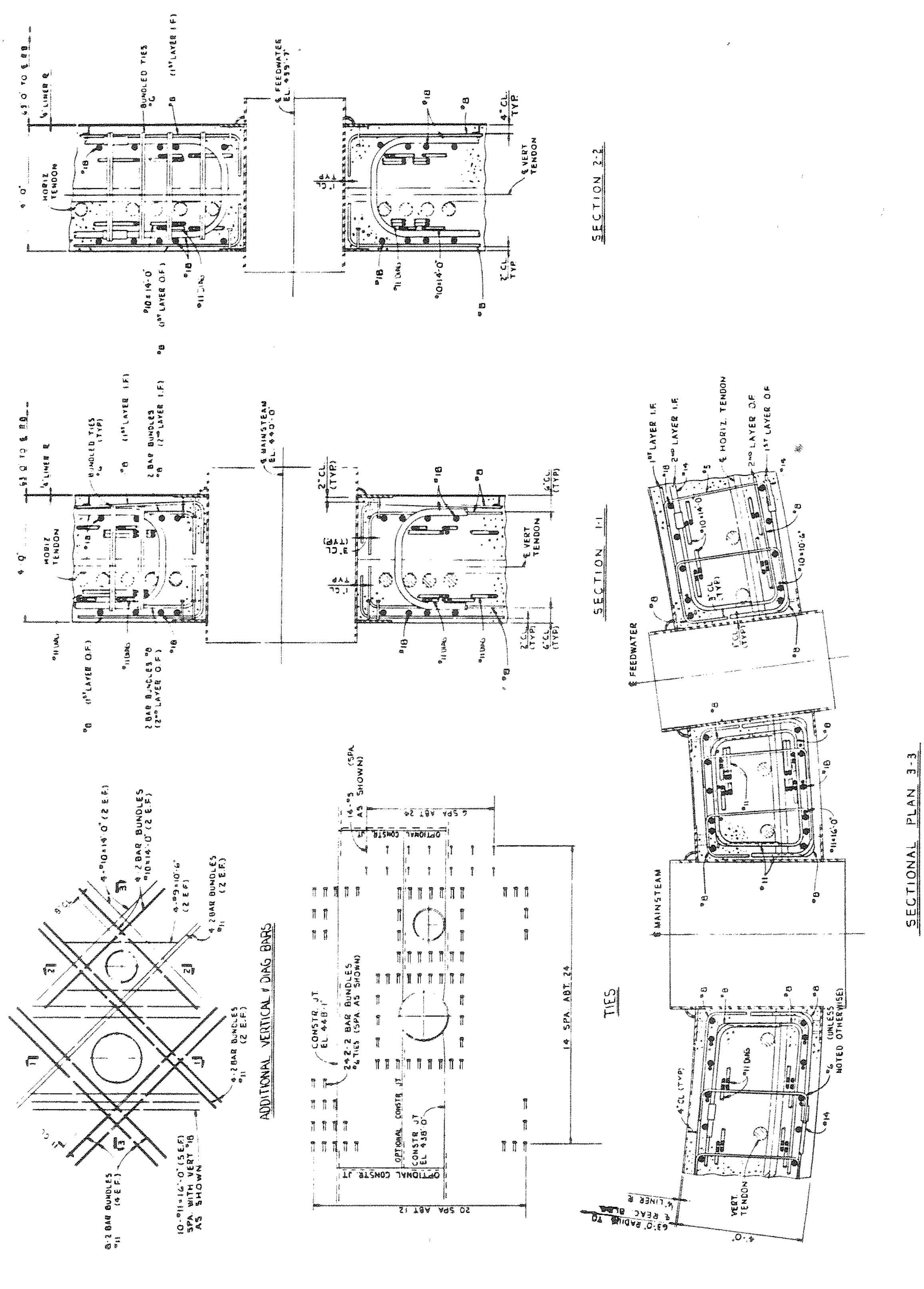
Figure 3.8-8



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam and Feedwater Penetration

Figure 3.8-9



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam and Feedwater Penetration

Figure 3.8-10

GENERAL STAGGER ON HOOP AND VERTICAL BAR CADWELDS IS 3' MIN., 4 BAR STAGGER PATTERN, EXCEPT FOR DOWELS.

INSIDE FACE

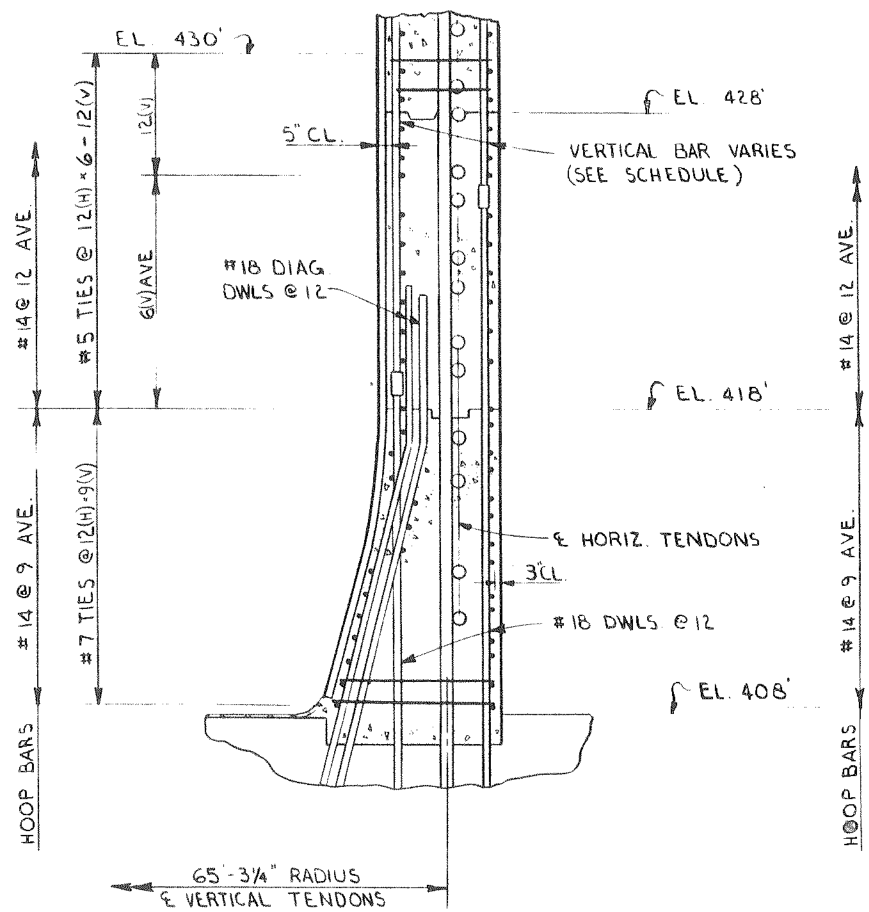
VERTICAL BARS @ 12

#18 DWLS TO 419, 423'-4" ALT. STAGGER
#14 TO 475'-6" AVE.
#11 TO 535'-6" AVE.
#14 INTO RING GIRDER

OUTSIDE FACE

VERTICAL BARS @ 12

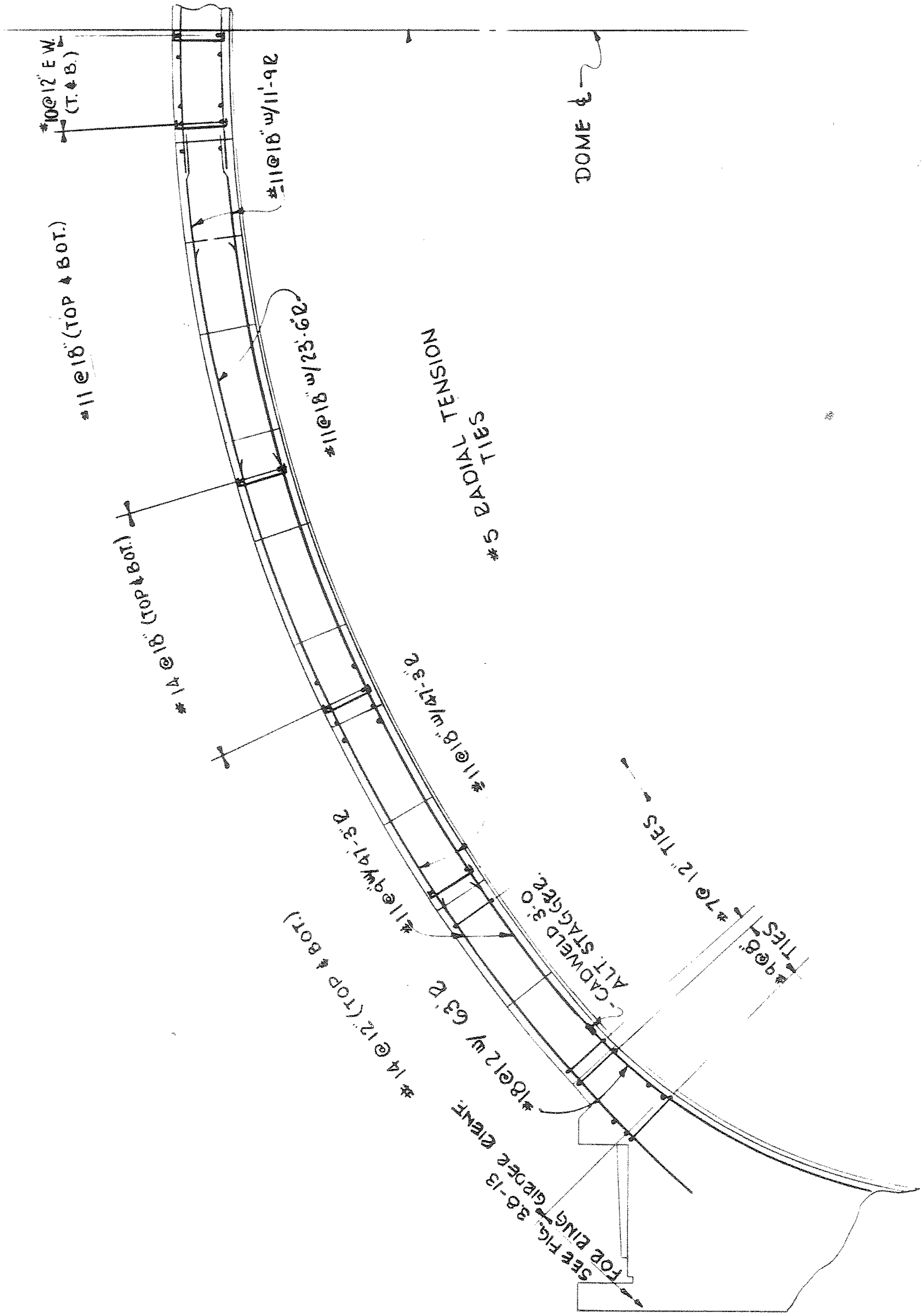
#18 DWLS. TO 430', 434' ALT. STAGGER
#18 TO 455'-6" AVE.
#14 INTO RING GIRDER



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Wall Reinforcement

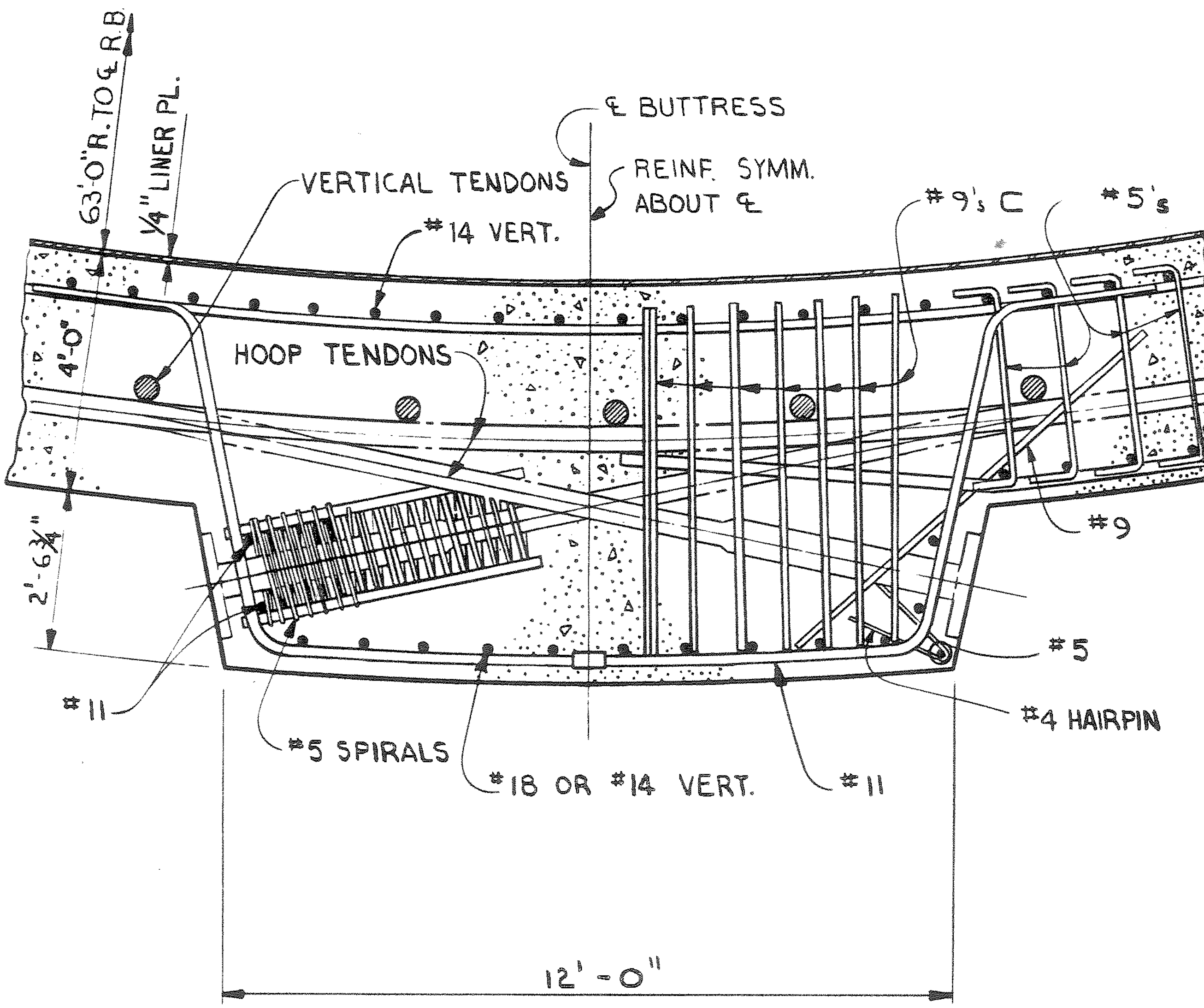
Figure 3.8-11



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Dome Reinforcement

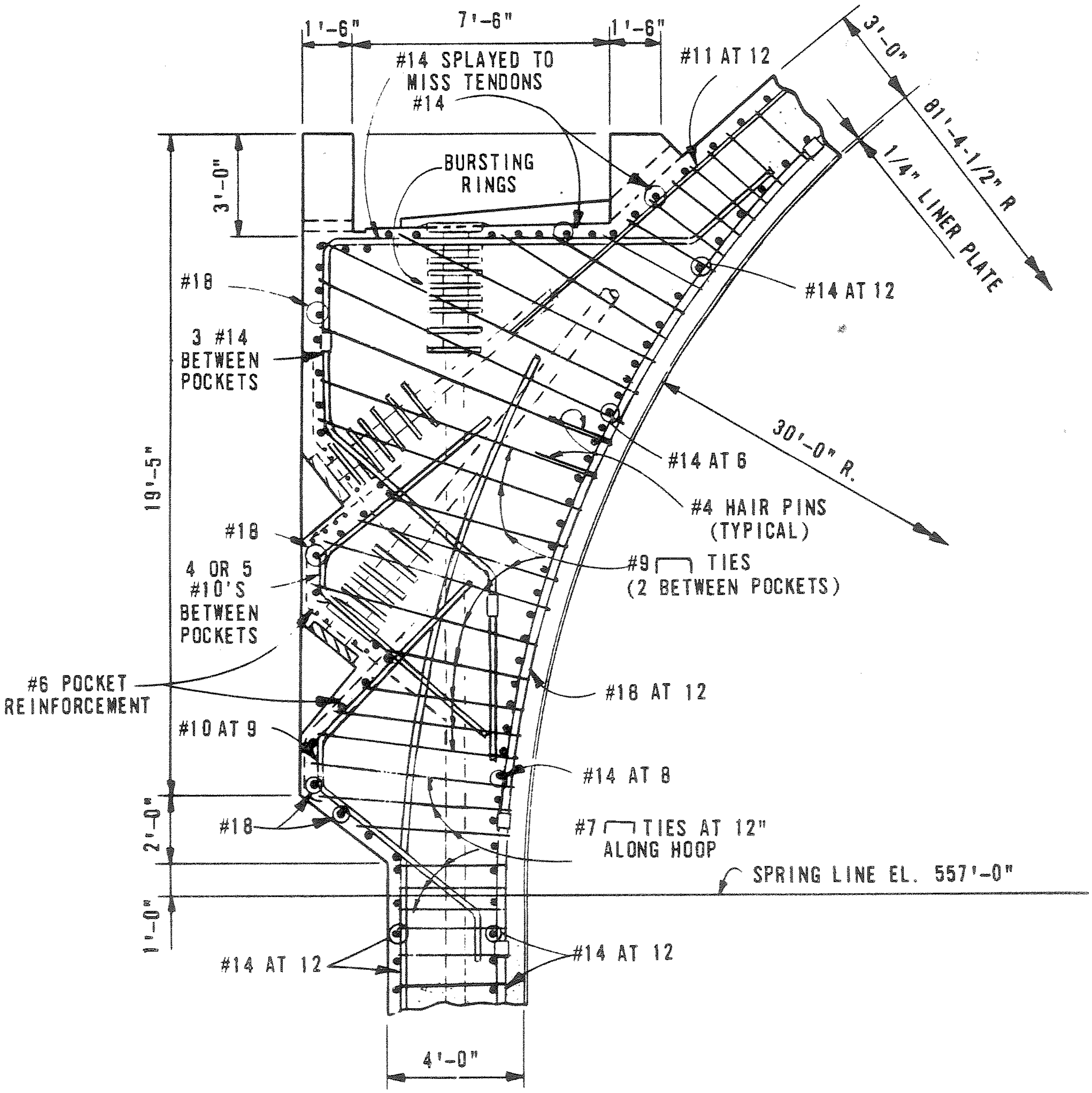
Figure 3.8-11a



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Buttress

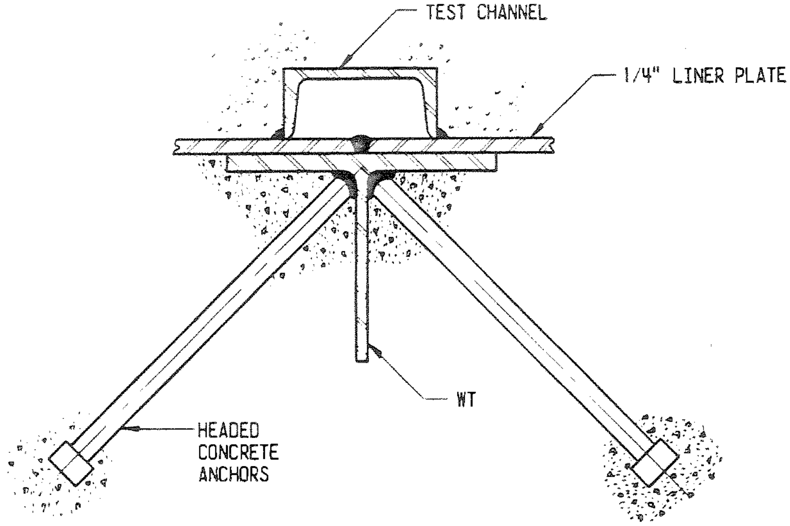
Figure 3.8-12



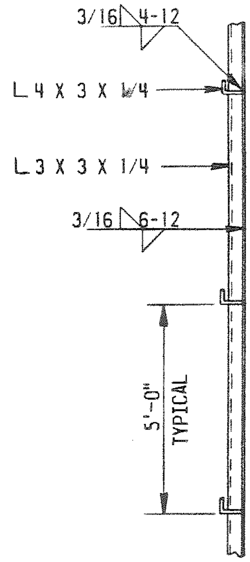
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Ring Girder

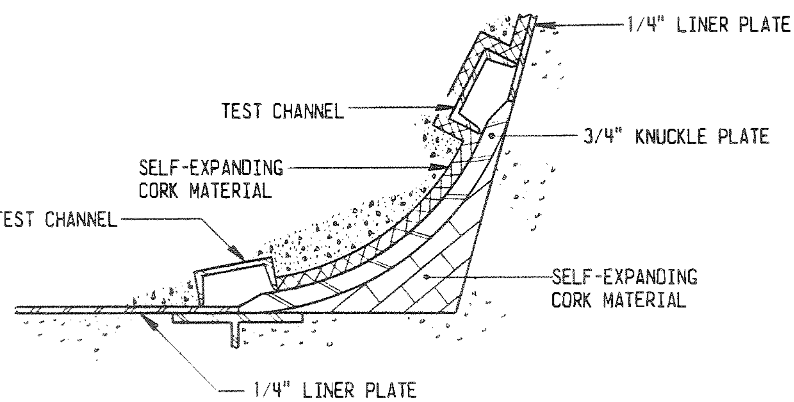
Figure 3.8-13



LINER FLOOR PLATE JOINT DETAIL



TYPICAL SECTION
WALL ANCHORS

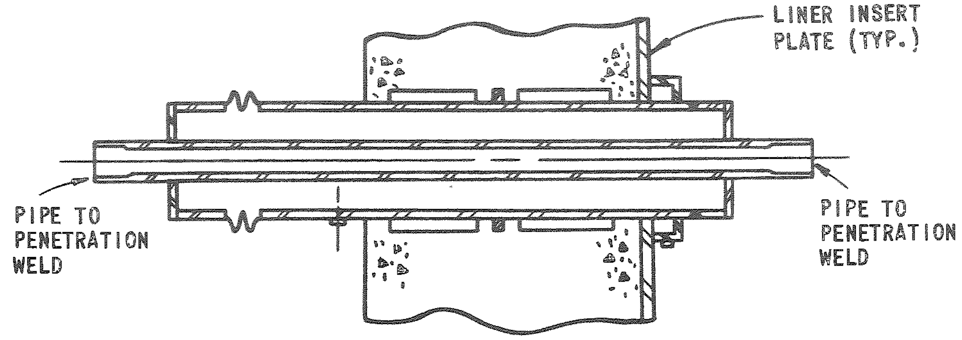


LINER CYLINDER TO BASE PLATE TRANSITION DETAIL

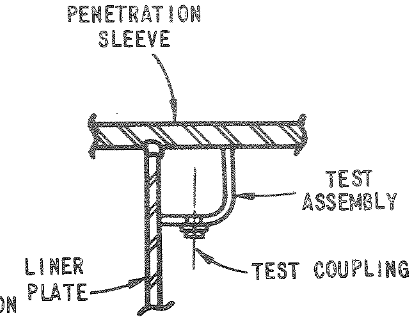
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Liner Cylinder to Base Plate
Transition Details

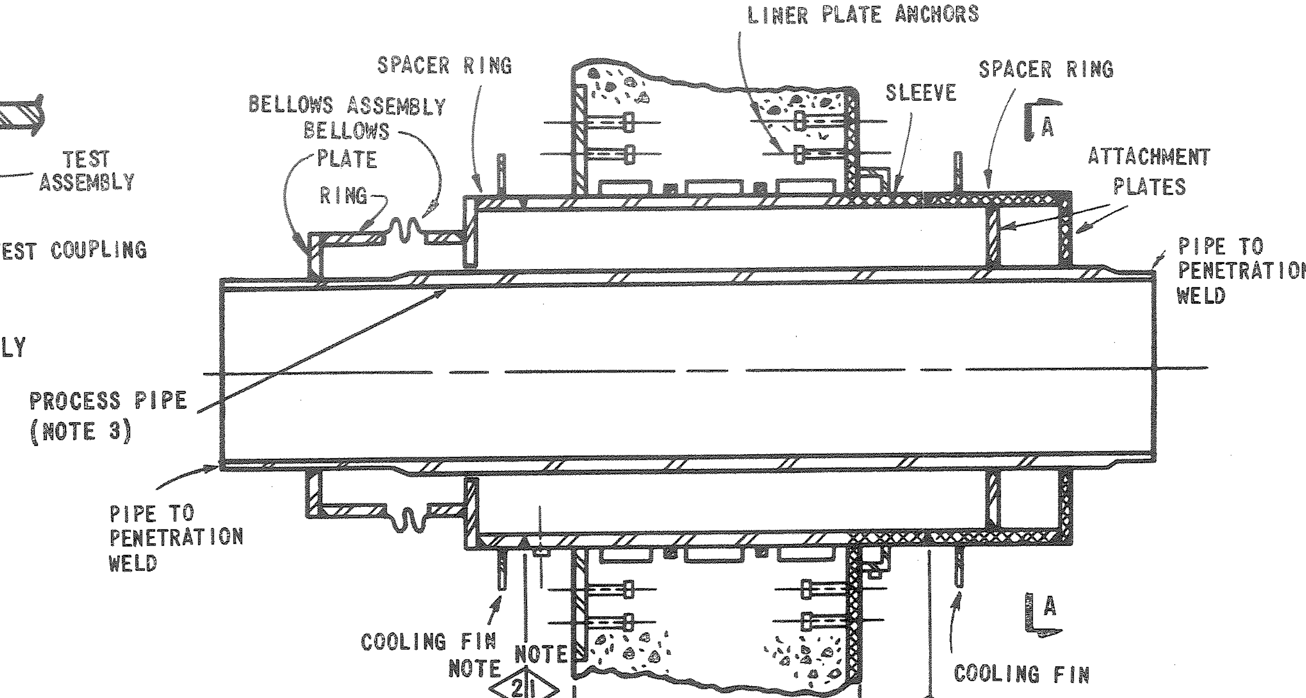
Figure 3.8-14



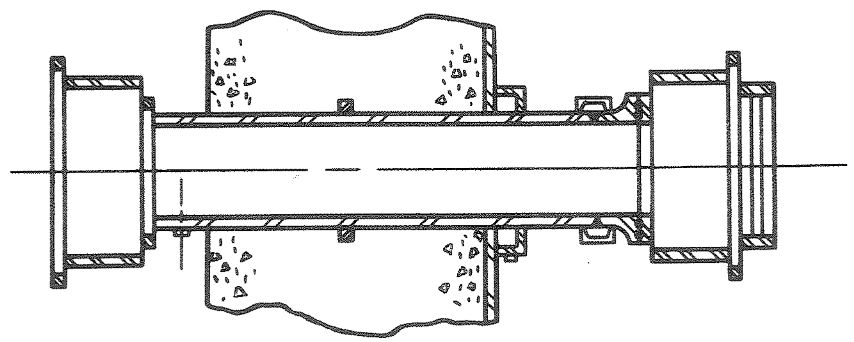
PIPING PENETRATION



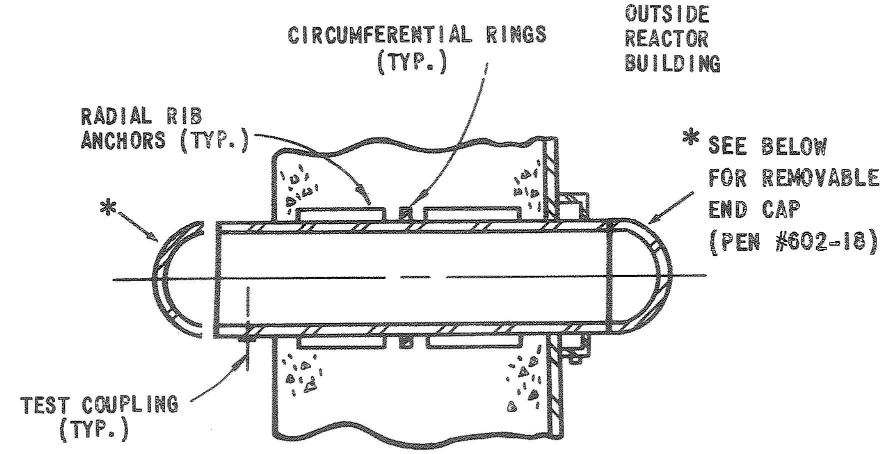
TEST ASSEMBLY DETAIL (TYPICAL)



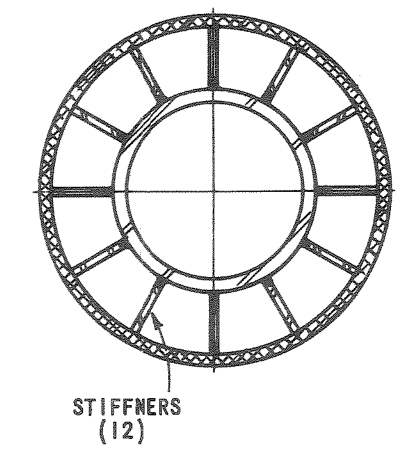
MAIN STEAM PENETRATION



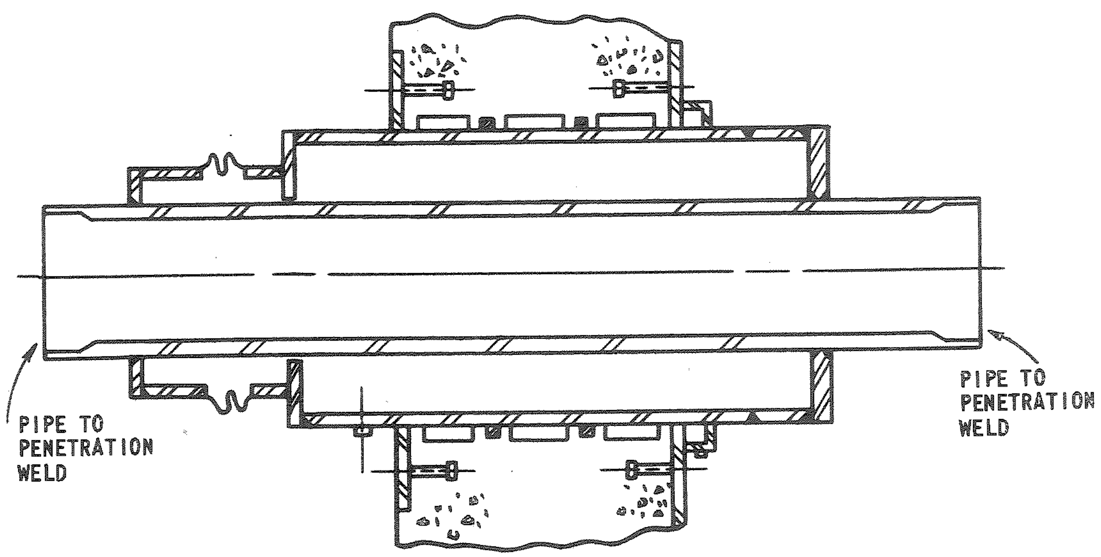
ELECTRICAL PENETRATION



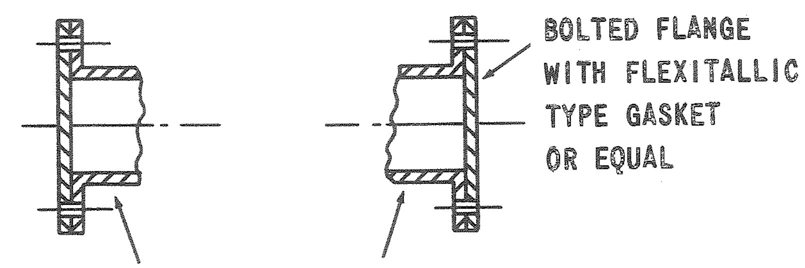
SPARE PENETRATION



SECTION A-A



FEEDWATER PENETRATION



SPARE PENETRATIONS #602-18, #600-12, #505-18 ONLY, WITH REMOVABLE END CAPS

NOTES:

1. CODE CLASS MC.
2. CODE CLASS MC - DESIGNED AND FABRICATED MEETING THE REQUIREMENTS OF SUBSECTION NC OF THE 1974 ASME CODE III, WITH ADDENDA UP TO AND INCLUDING WINTER 1975 AS ALLOWED BY ARTICLE NE-1120.
3. CODE CLASS 2.

xxxxxxx CONTAINMENT BOUNDARY

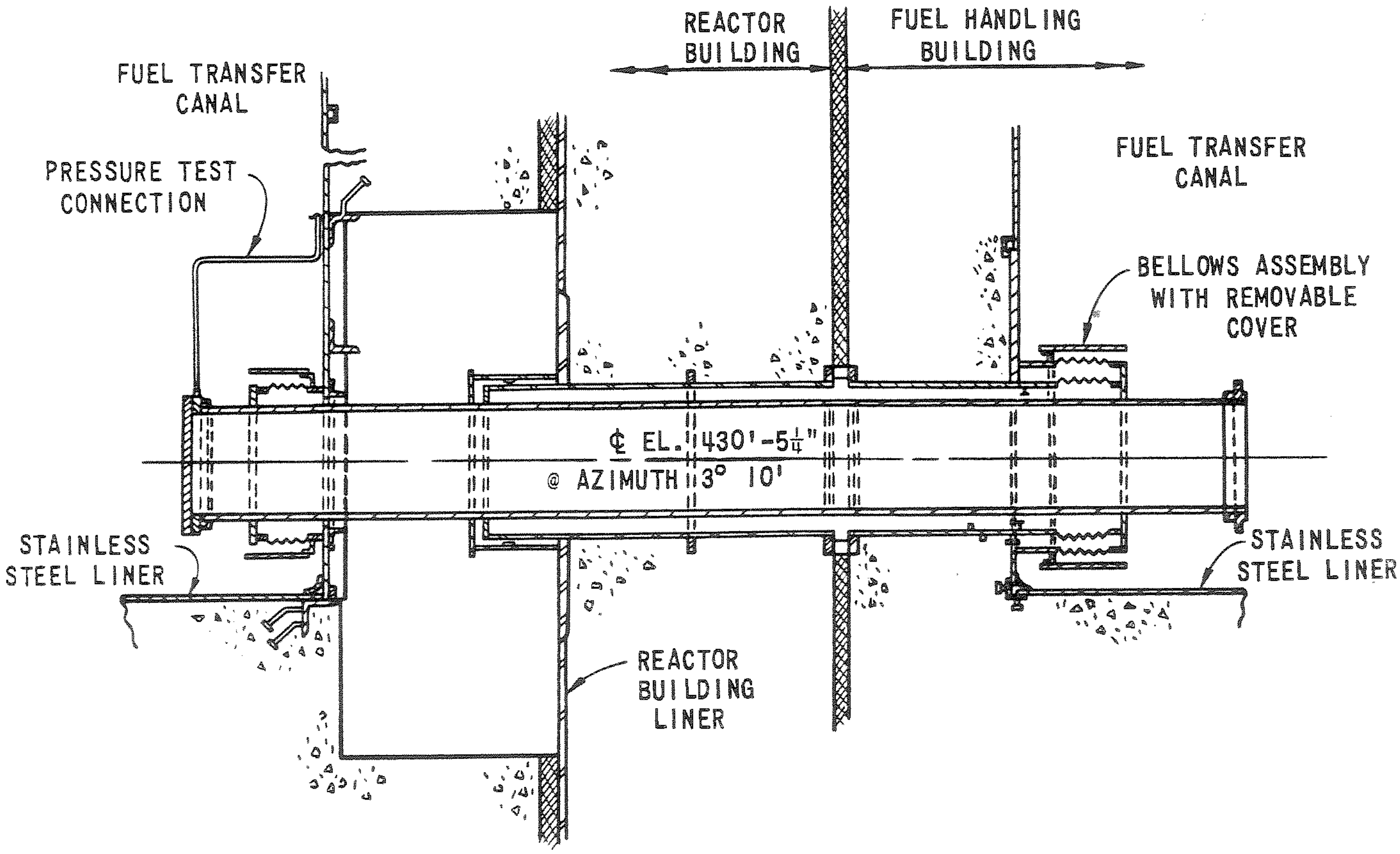
▼ WELD

AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Penetration Details

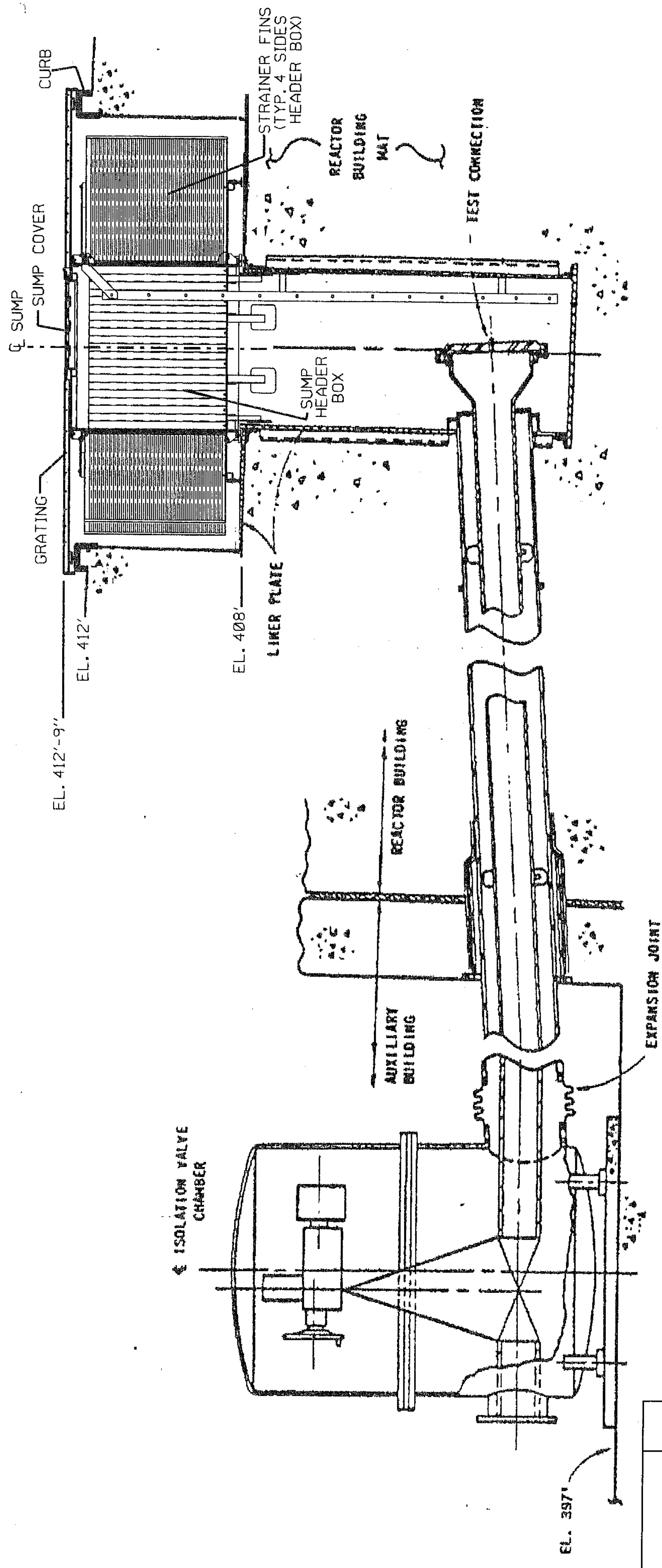
3.8-15



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Fuel Transfer Tube
Penetration

Figure 3.8-16

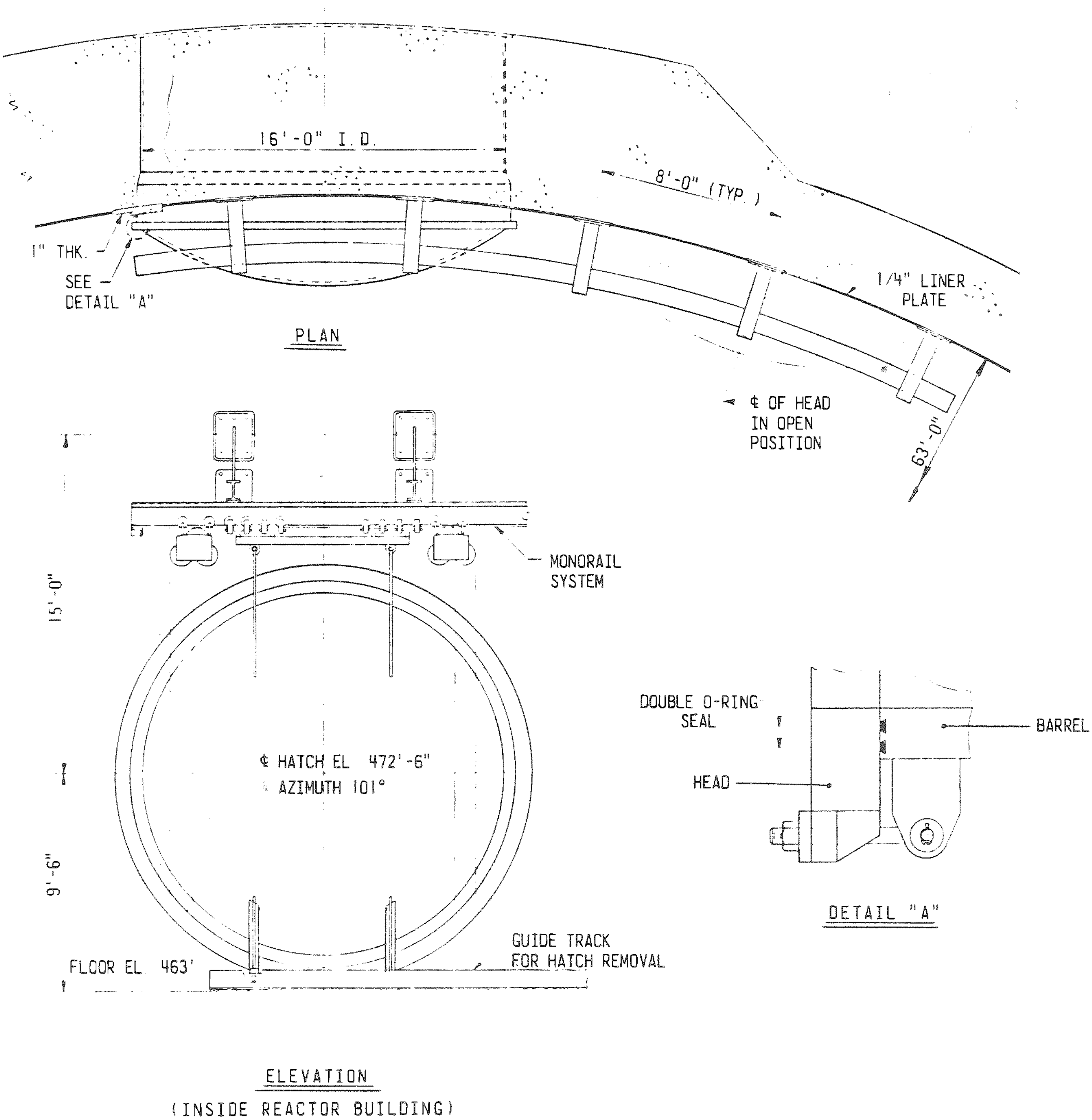


RN 13-022

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RB Recirculation Sumps for Residual Heat Removal and RB Spray Pumps

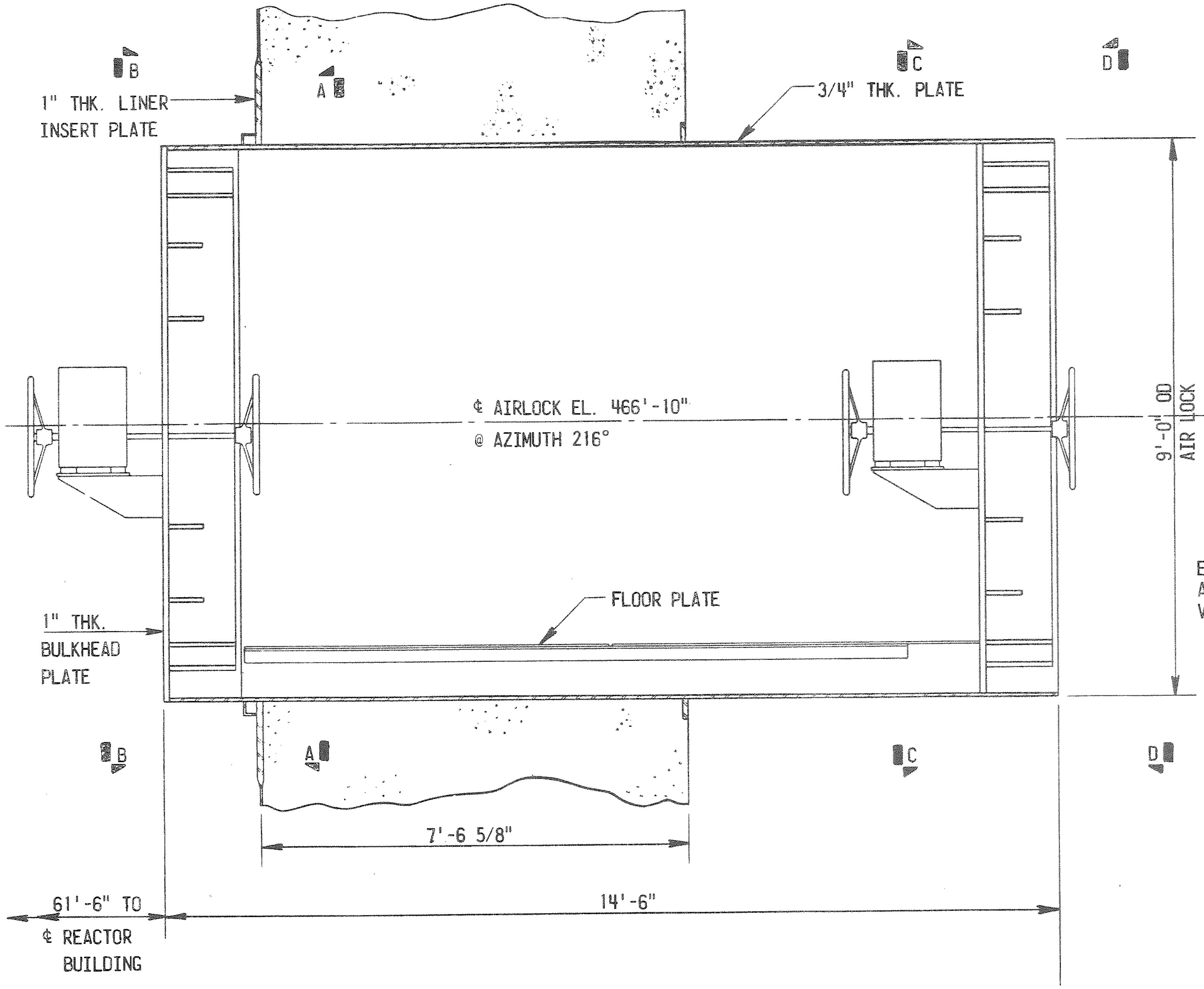
Figure 3.8-17



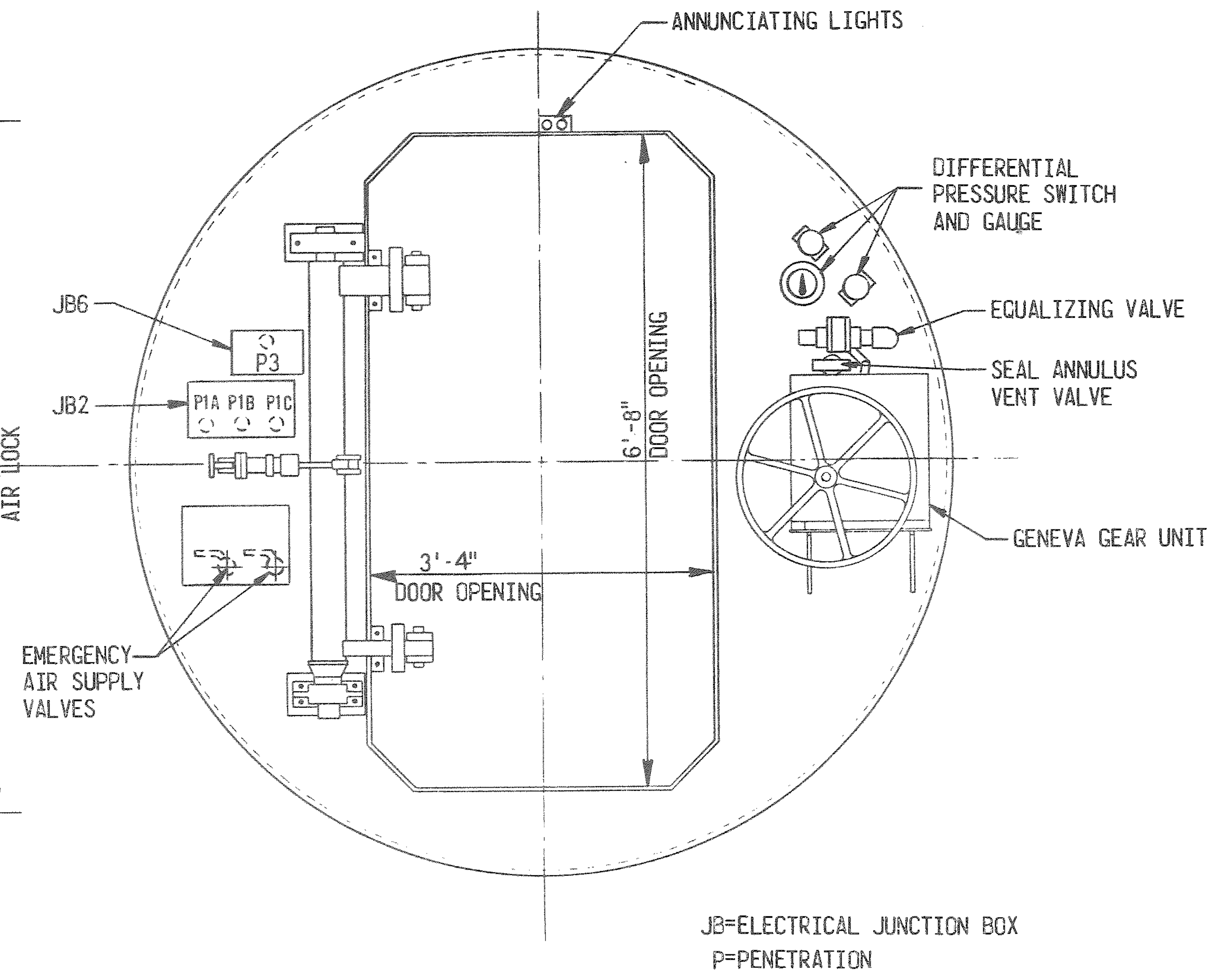
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Equipment Hatch

Figure 3.8-18



ELEVATION

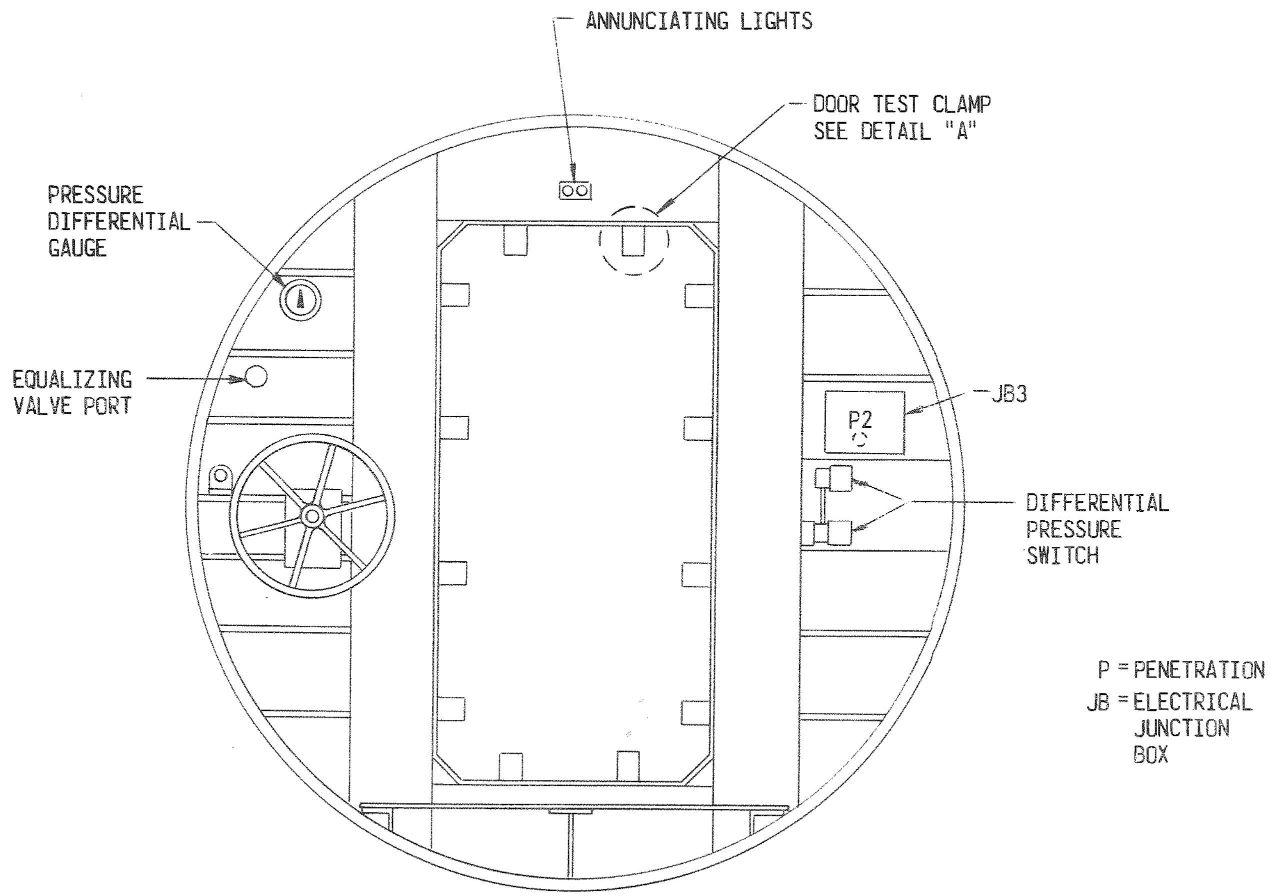


SECTION C-C
ATMOSPHERE BULKHEAD
FROM INSIDE AIRLOCK

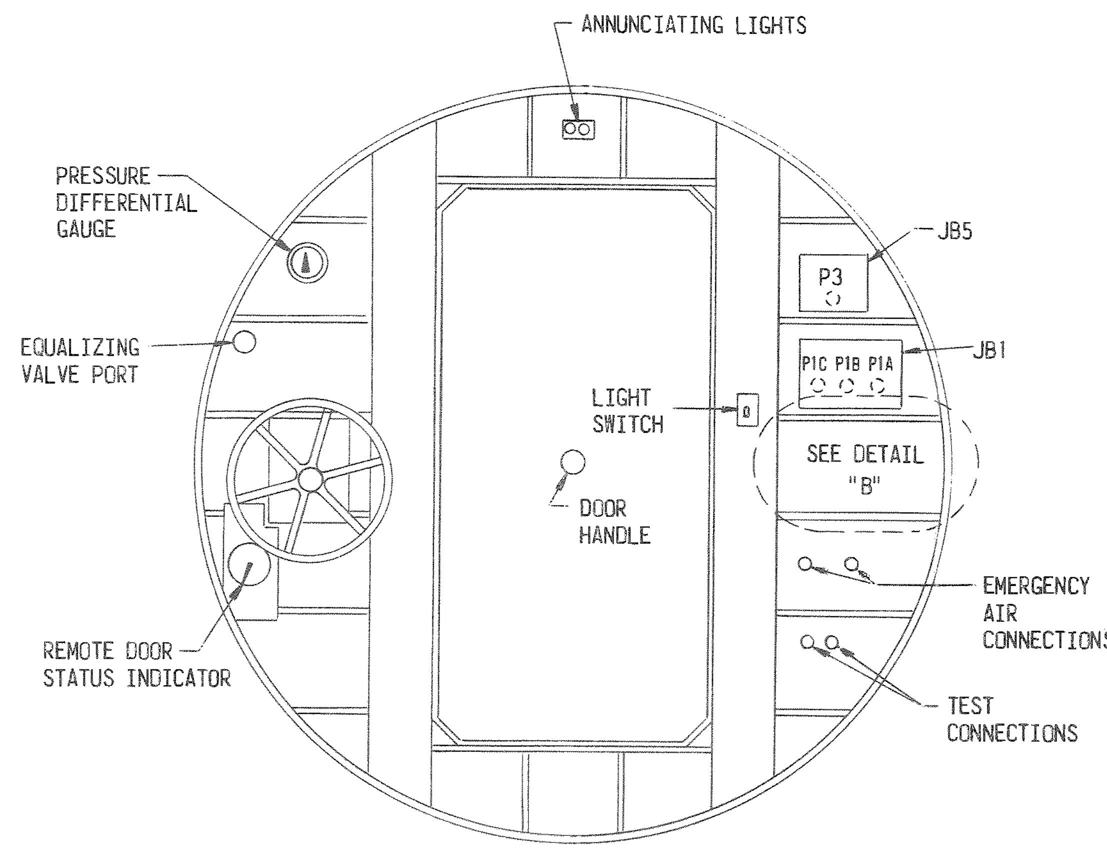
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Personnel Airlock

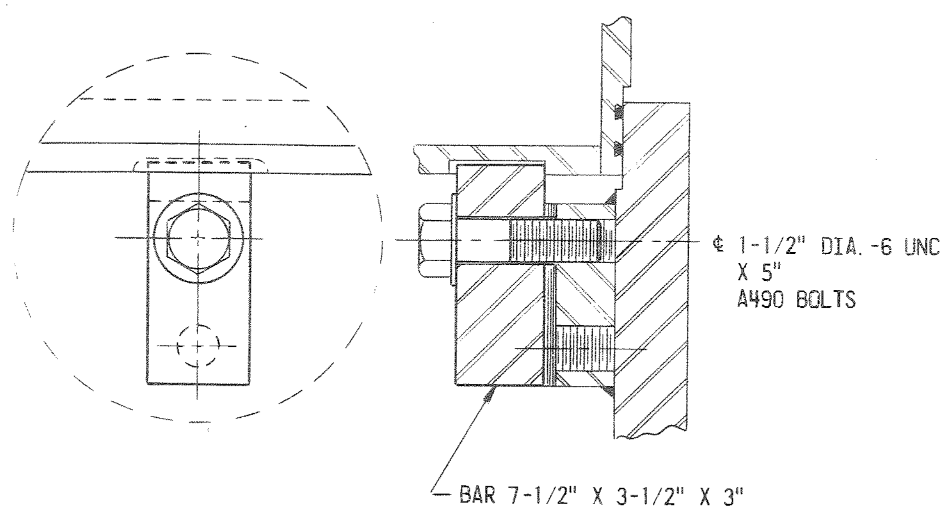
Figure 3.8-19



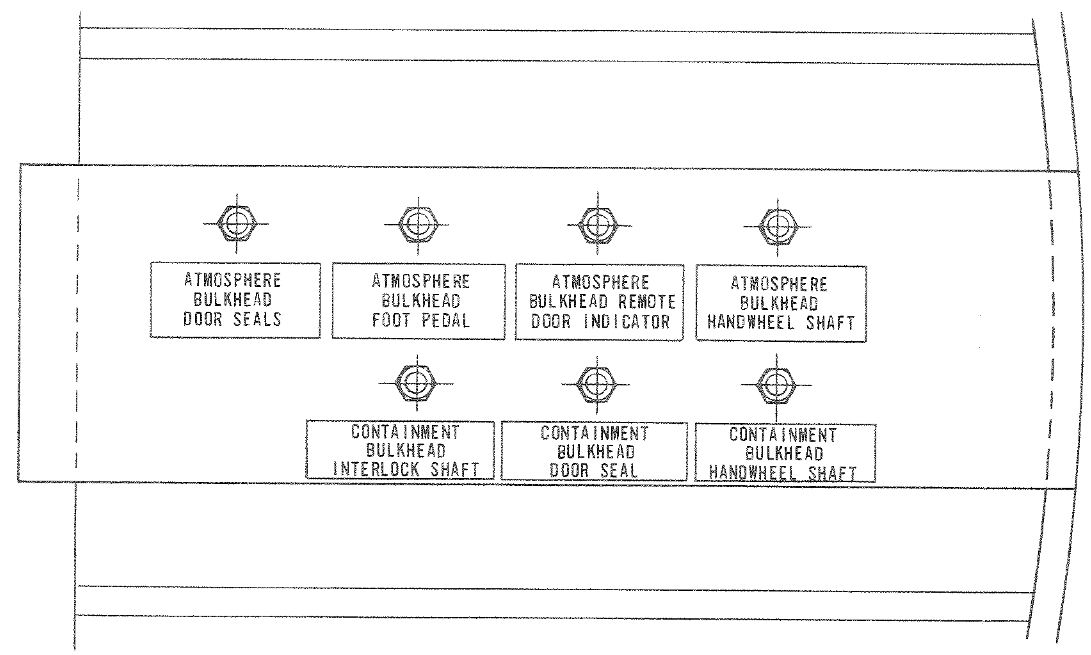
SECTION A-A
CONTAINMENT BULKHEAD
FROM INSIDE AIRLOCK



SECTION D-D
ATMOSPHERE BULKHEAD
FROM ATMOSPHERE



DETAIL "A"
DOOR TEST CLAMPS



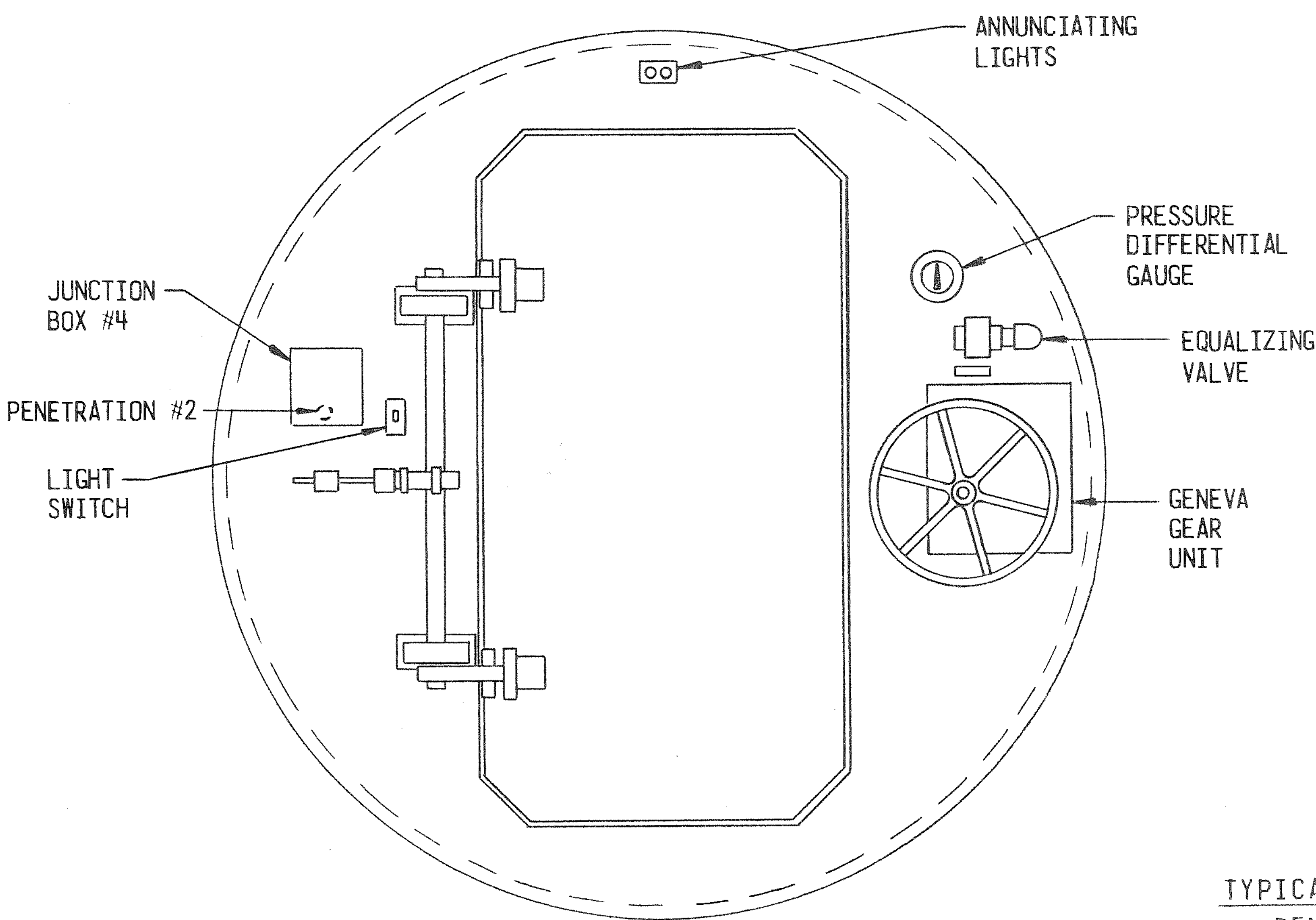
DETAIL "B"

REMOTE TEST PANEL

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

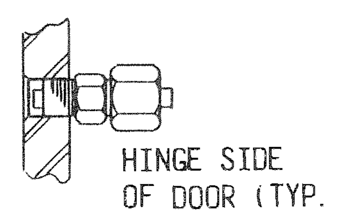
Personnel Airlock Sections
and Details

Figure 3.8-19a



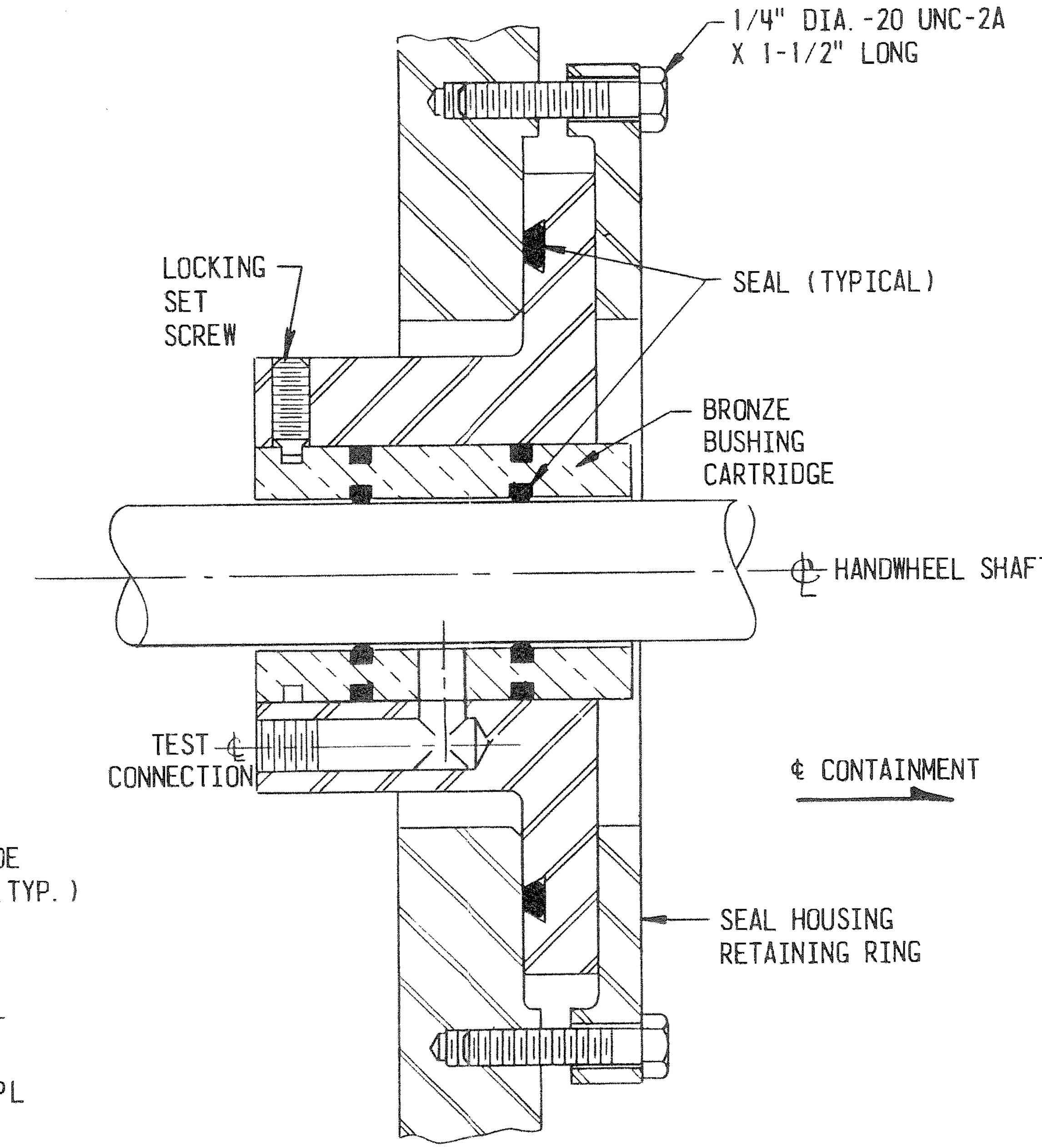
SECTION B-B

CONTAINMENT BULKHEAD
FROM INSIDE CONTAINMENT



TYPICAL ELECTRICAL
PENETRATION

CONAX FITTING-TYPE PL

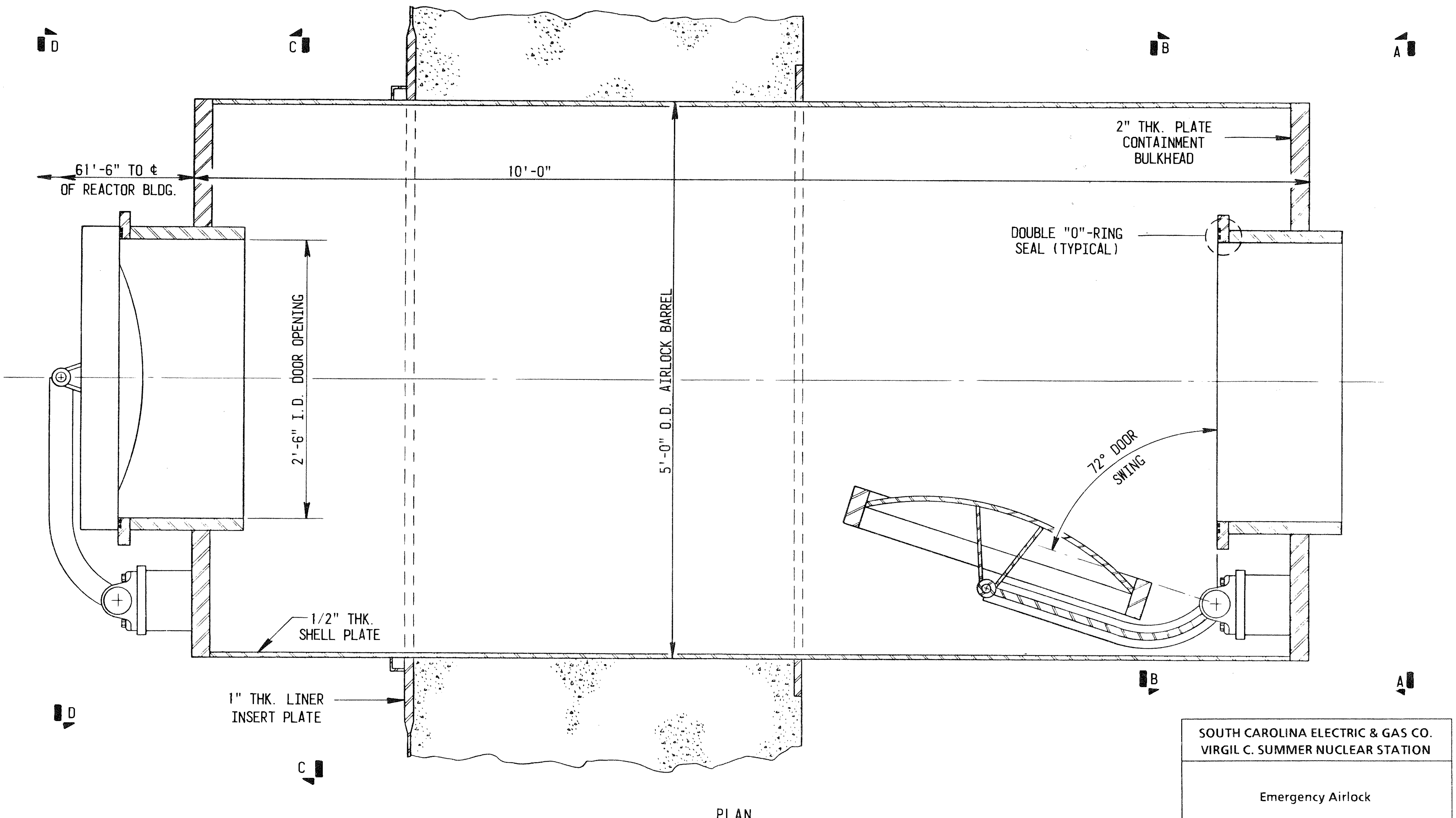


HANDWHEEL SHAFT SEAL

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Personnel Airlock Sections
and Details

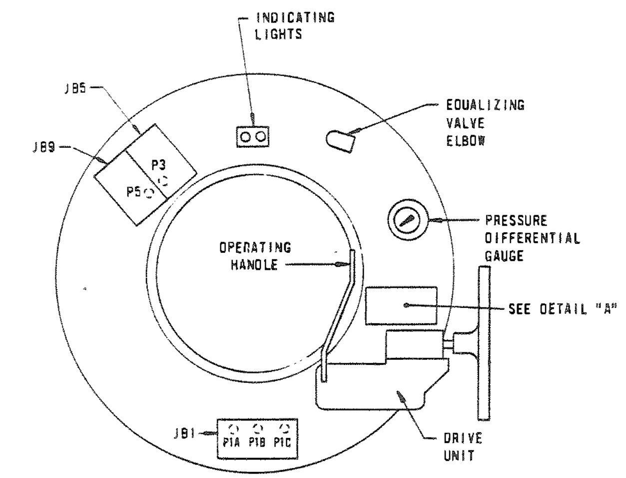
Figure 3.8-19b



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

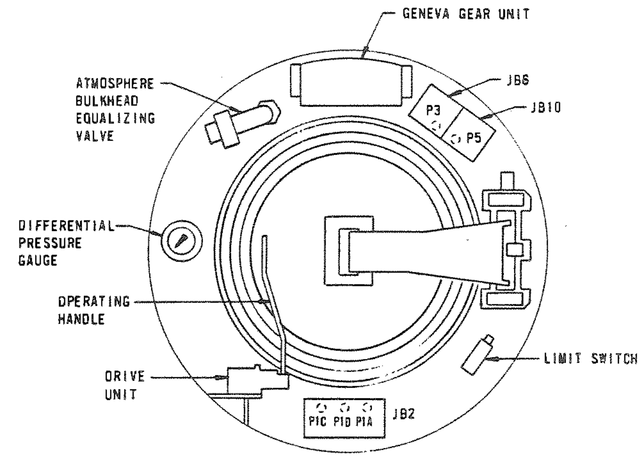
Emergency Airlock

Figure 3.8-20

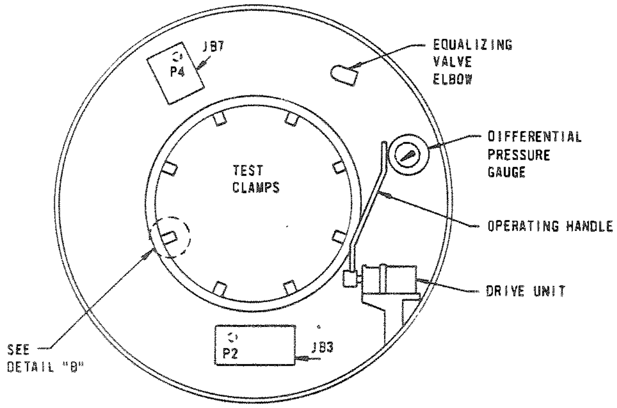


SECTION A-A
ATMOSPHERE BULKHEAD
FROM ATMOSPHERE

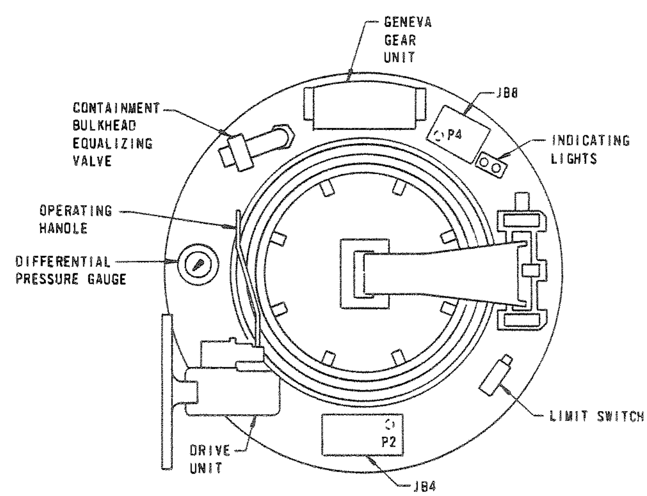
P = PENETRATION
JB = ELECTRICAL JUNCTION BOX



SECTION B-B
ATMOSPHERE BULKHEAD
FROM INSIDE AIRLOCK



SECTION C-C
CONTAINMENT BULKHEAD
FROM INSIDE AIRLOCK

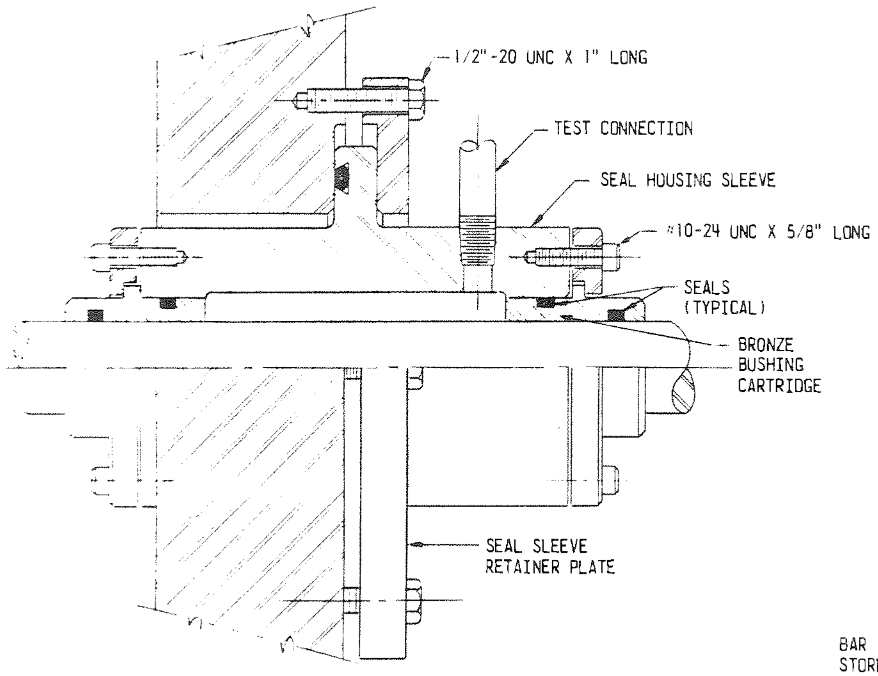


SECTION A-A
CONTAINMENT BULKHEAD
FROM INSIDE CONTAINMENT

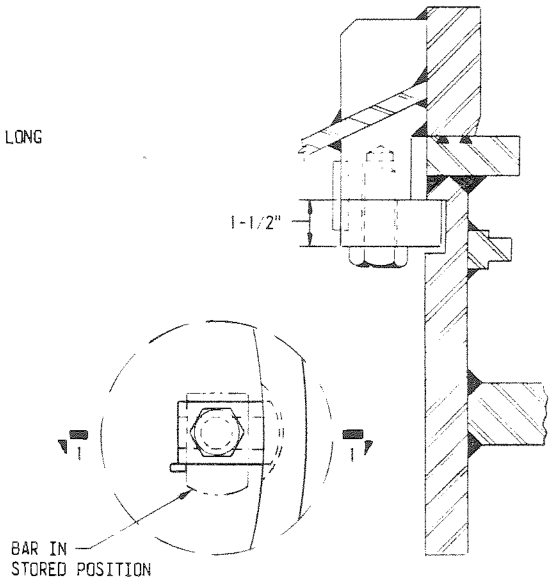
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Emergency Airlock Sections

Figure 3.8-20a

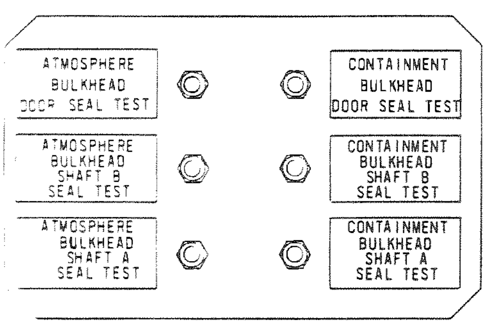


HANDWHEEL SHAFT SEAL



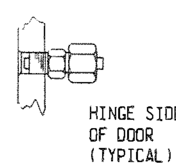
DETAIL "B" SECTION 1-1

DOOR TEST CLAMP



DETAIL "A"

REMOTE TEST PANEL



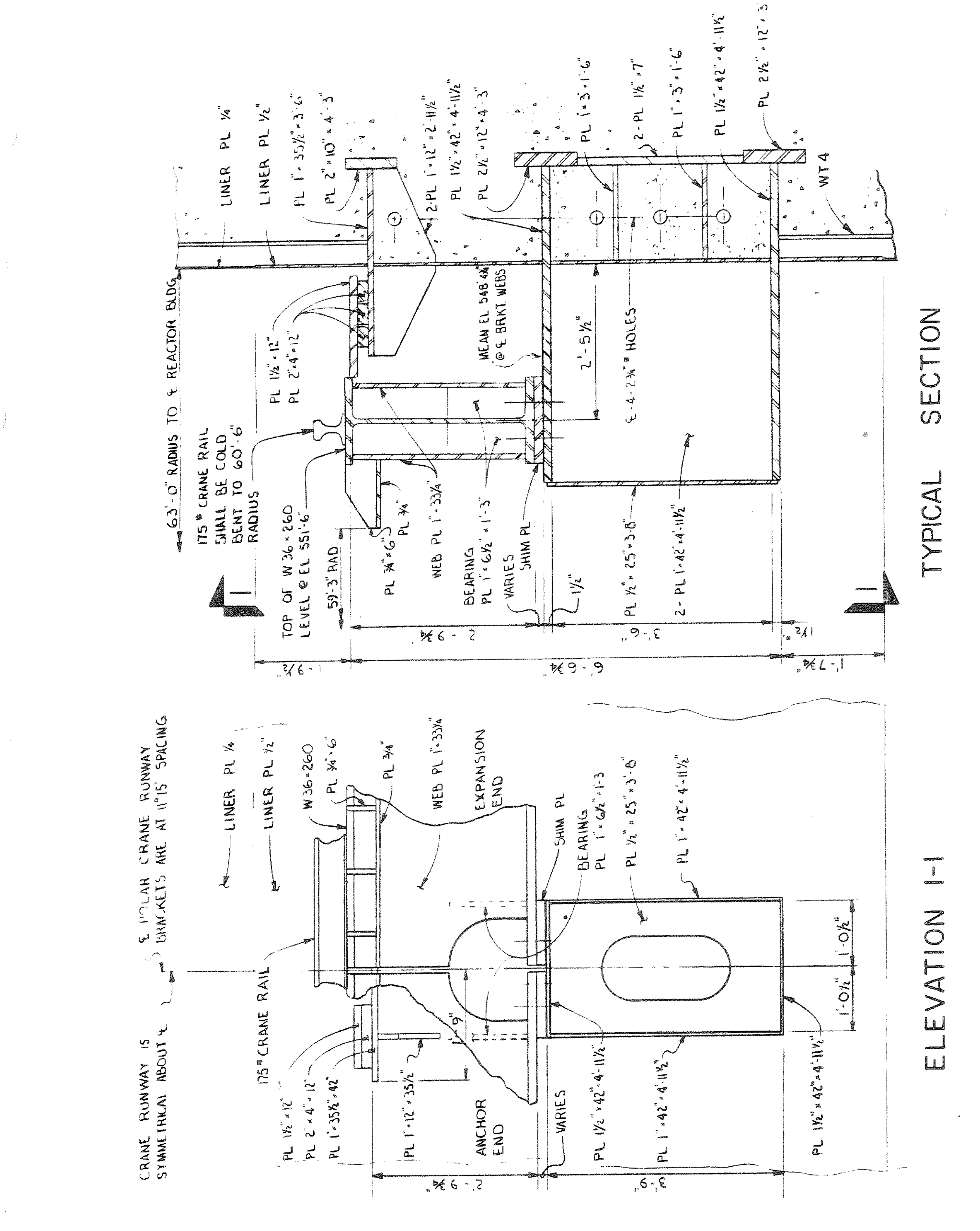
TYPICAL ELECTRICAL PENETRATION

CONAX FITTING TYPE PL

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Emergency Airlock Sections and Details

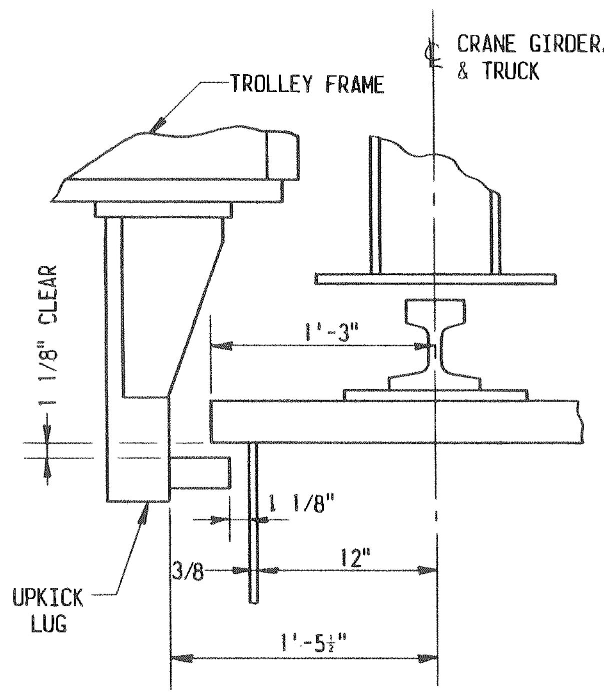
Figure 3.8-20b



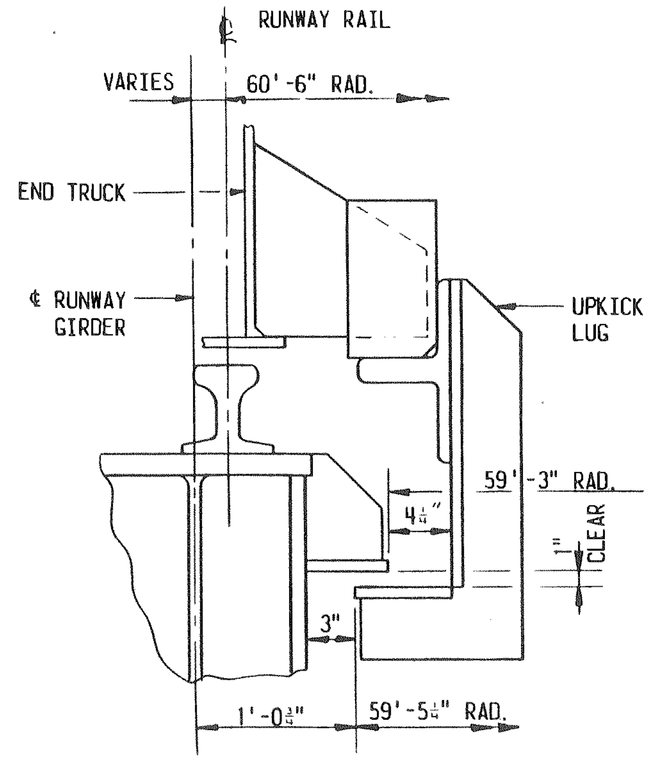
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Polar Crane Brackets
and Runway Beams

Figure 3.8-21



TROLLEY UPKICK LUG DETAIL REACTOR BLDG. CRANE

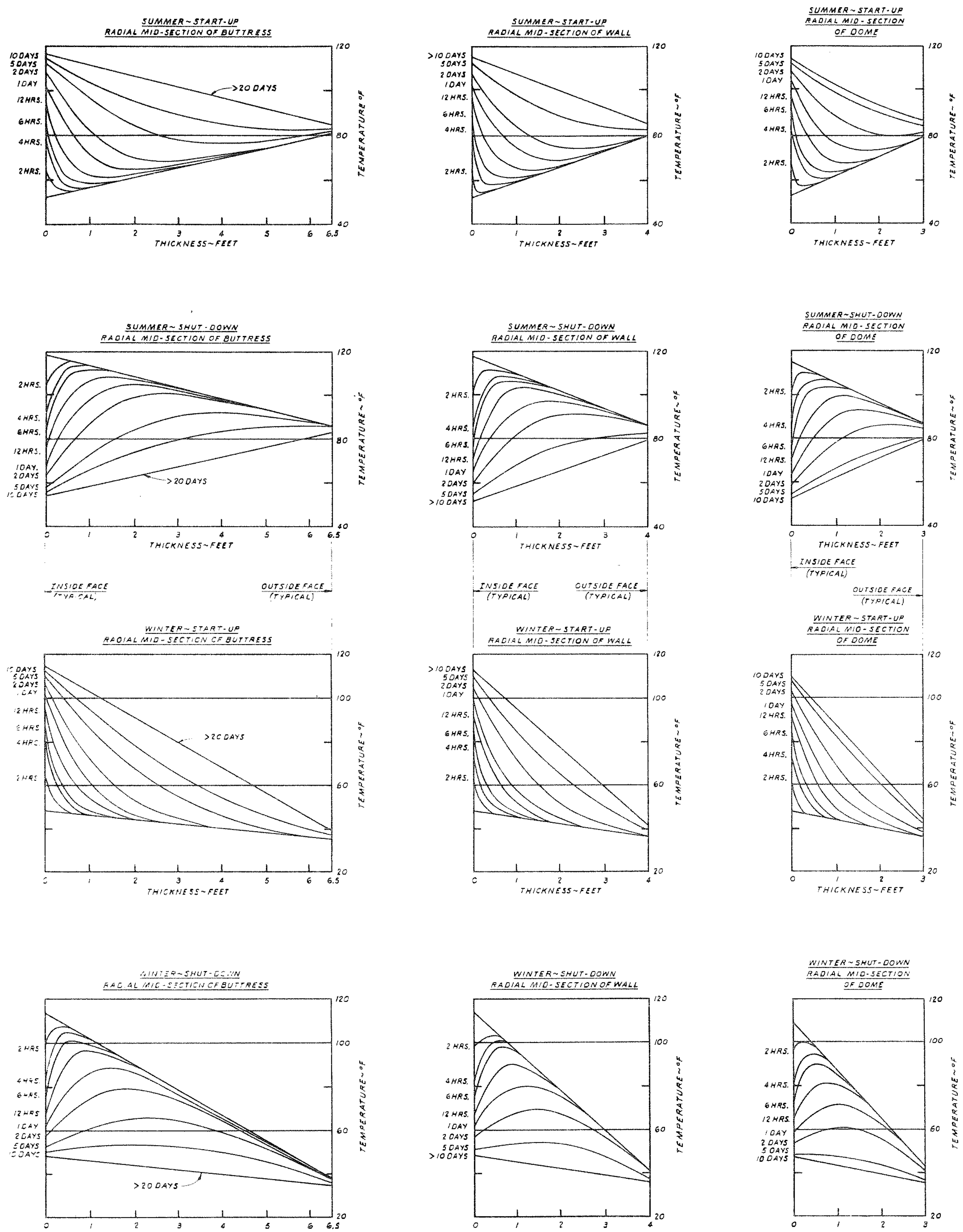


CRANE UPKICK LUG DETAIL
REACTOR BLDG. CRANE

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Trolley and Crane
Uplift Lugs

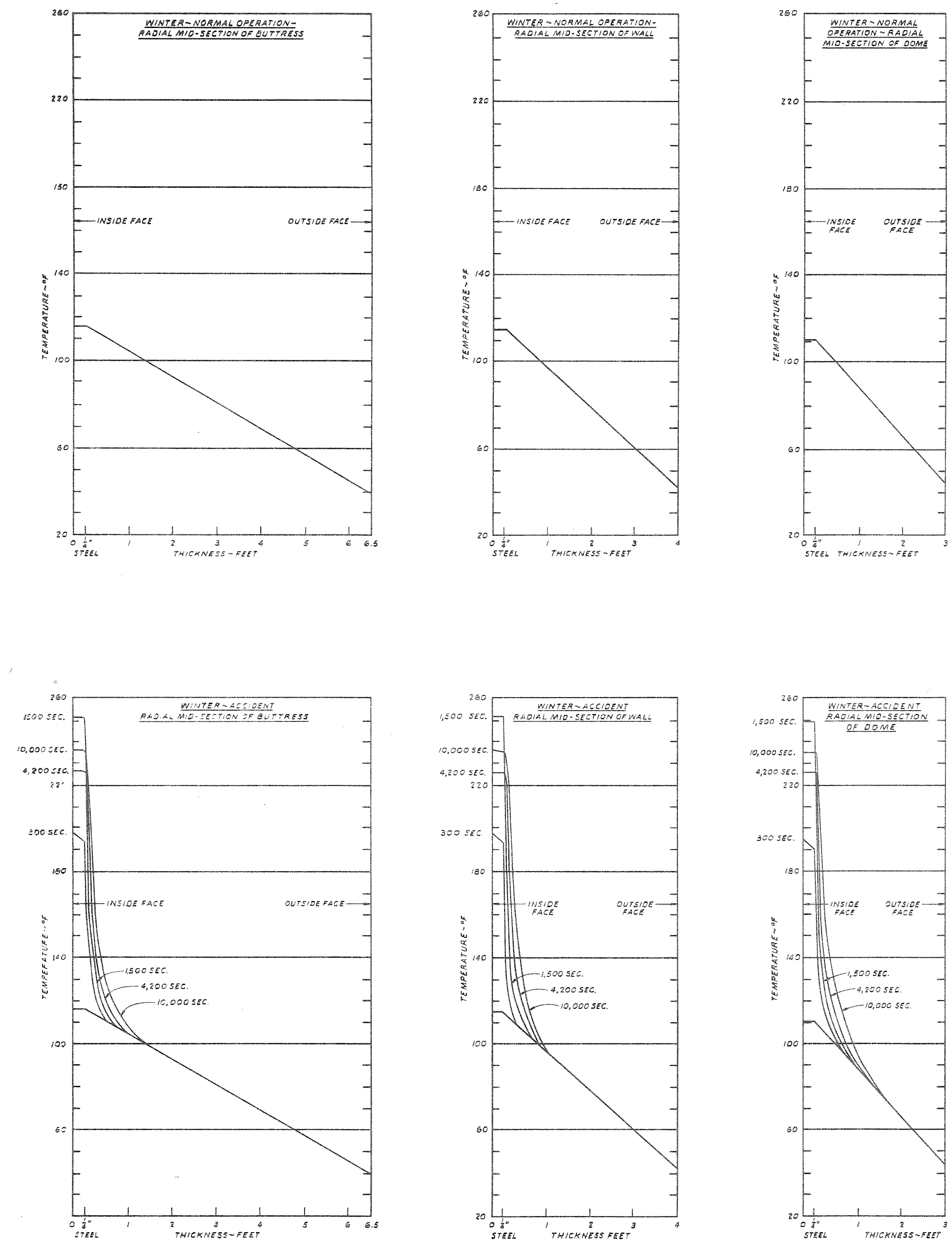
Figure 3.8-22



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Temperature Profiles Through Buttress, Wall and Dome Summer and Winter Start-up and Shut-down

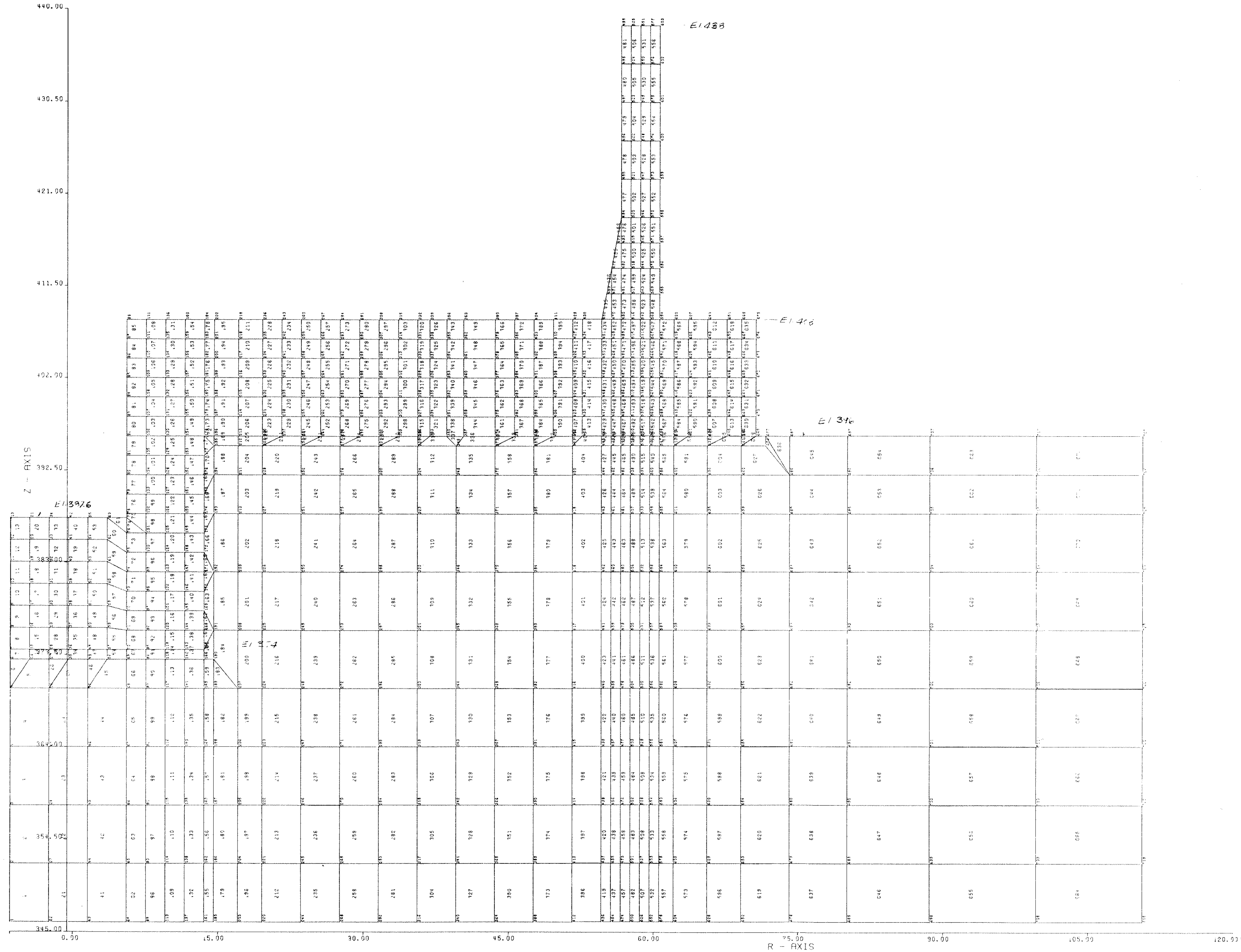
Figure 3.8-23



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Temperature Profiles Through Buttress, Wall and Dome Winter Normal and Accident

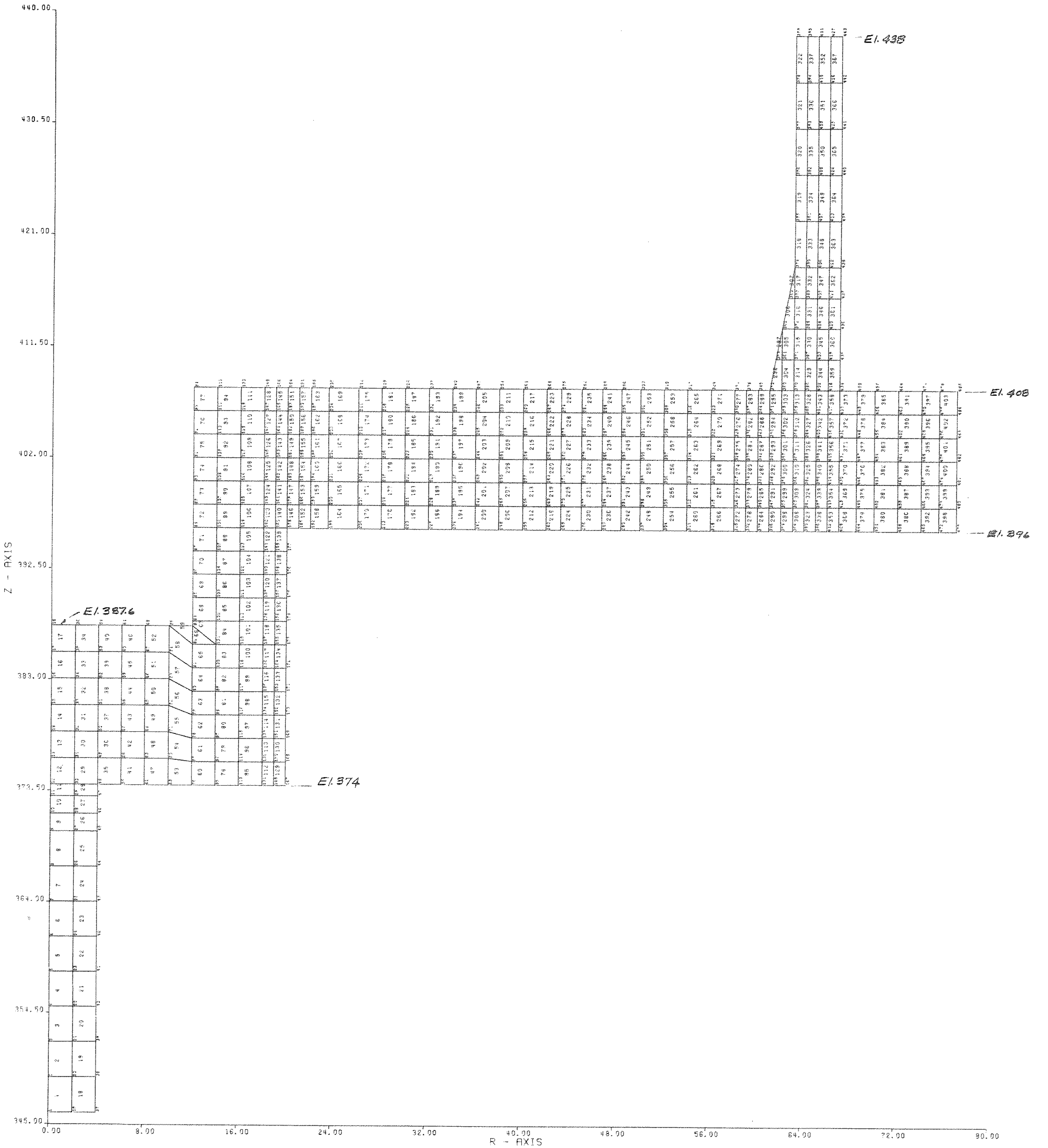
Figure 3.8-24



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

ELAD Model For Symmetrical Loads

Figure 3.8-25

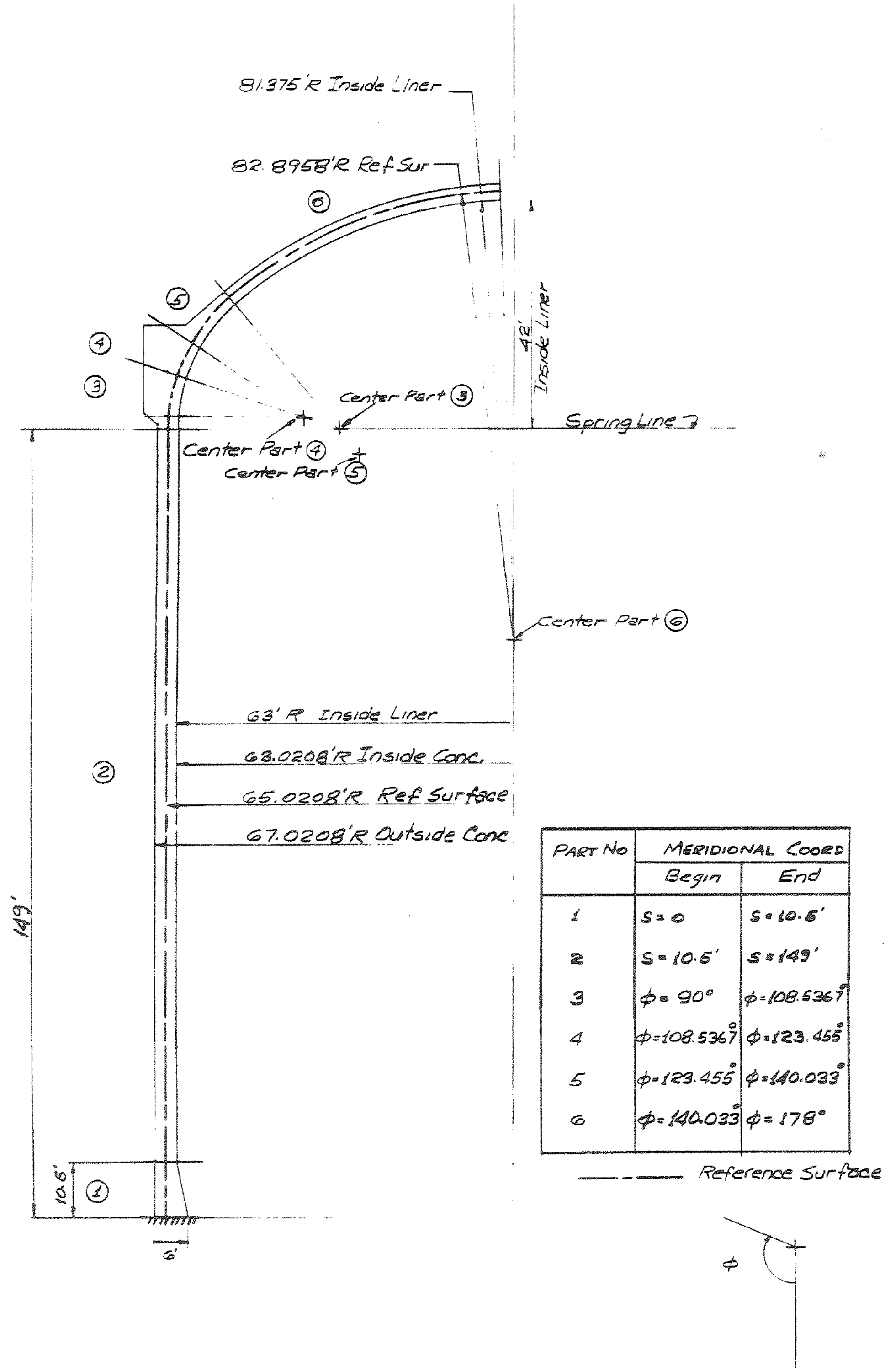


V C SUMMER UNIT 1 D + DBE REACTOR BLDG MAT REVISED GRID

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

ELAD Model For
Unsymmetrical Loads

Figure 3.8-26

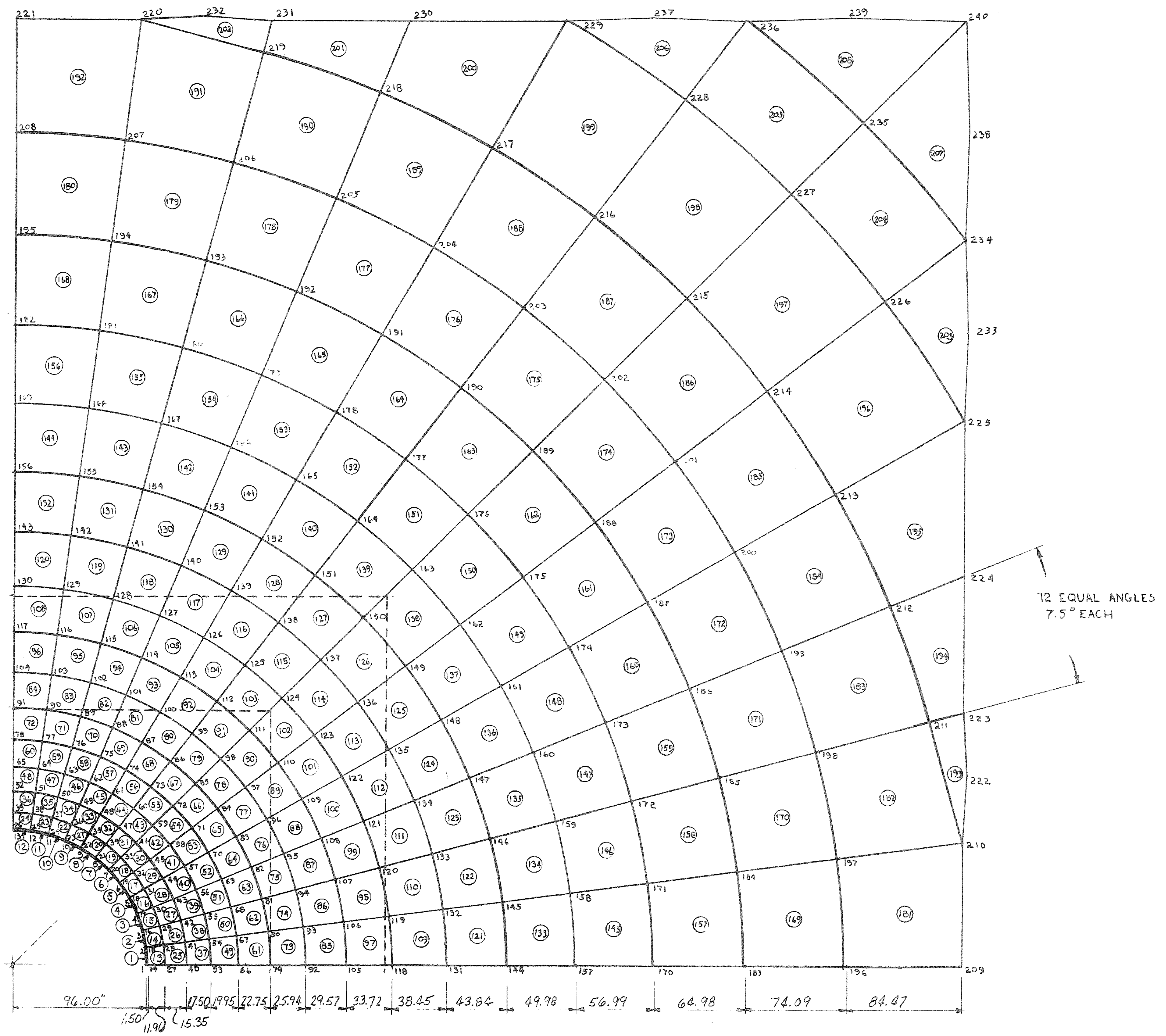


PART No	MERIDIONAL COORD	
	Begin	End
1	S = 0	S = 10.5'
2	S = 10.5'	S = 149'
3	φ = 90°	φ = 108.5367°
4	φ = 108.5367°	φ = 123.455°
5	φ = 123.455°	φ = 140.033°
6	φ = 140.033°	φ = 178°

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Model For Use With KALNINS Program

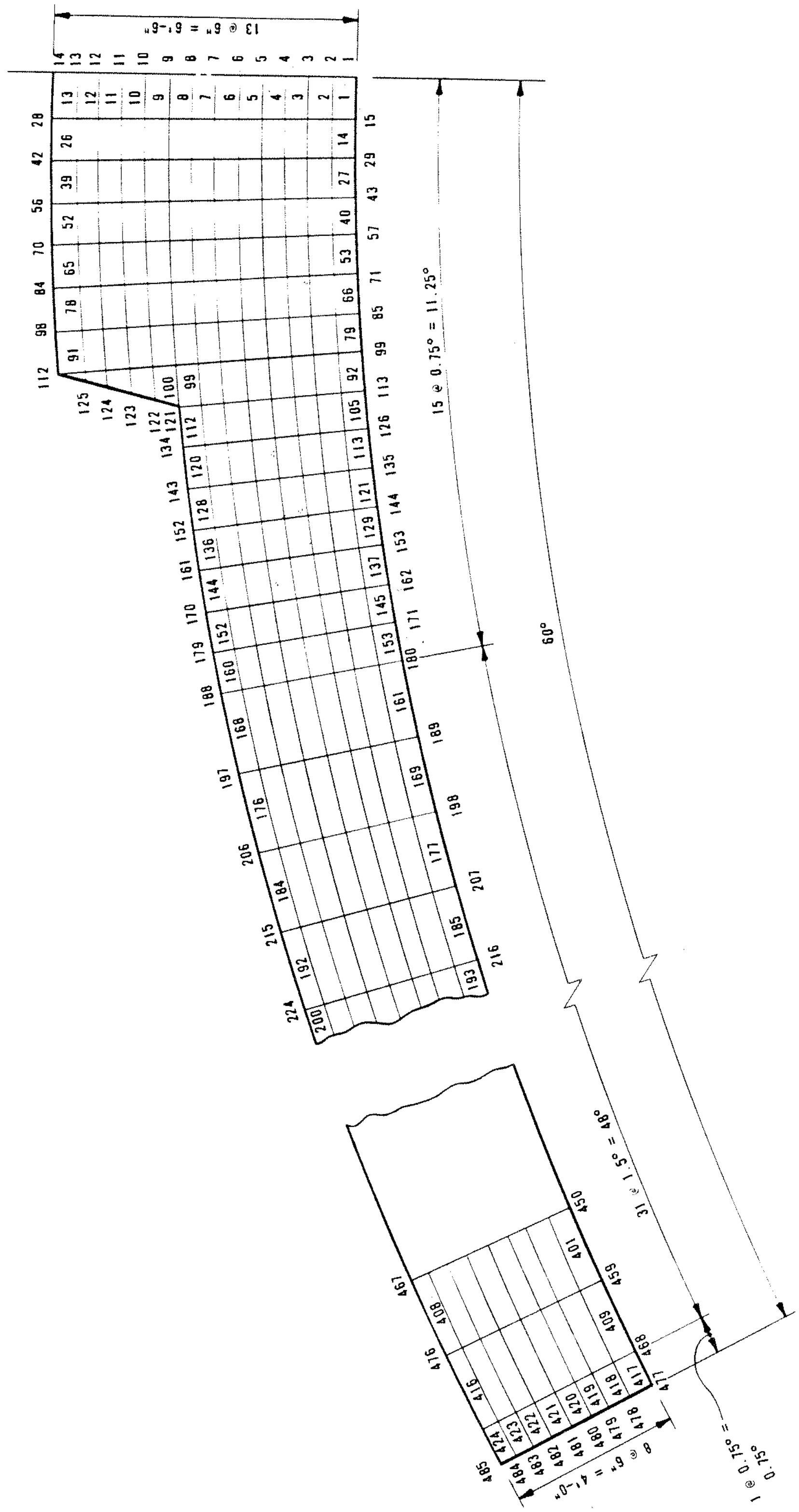
Figure 3.8-27



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Finite Element Model of
Accesss Openings

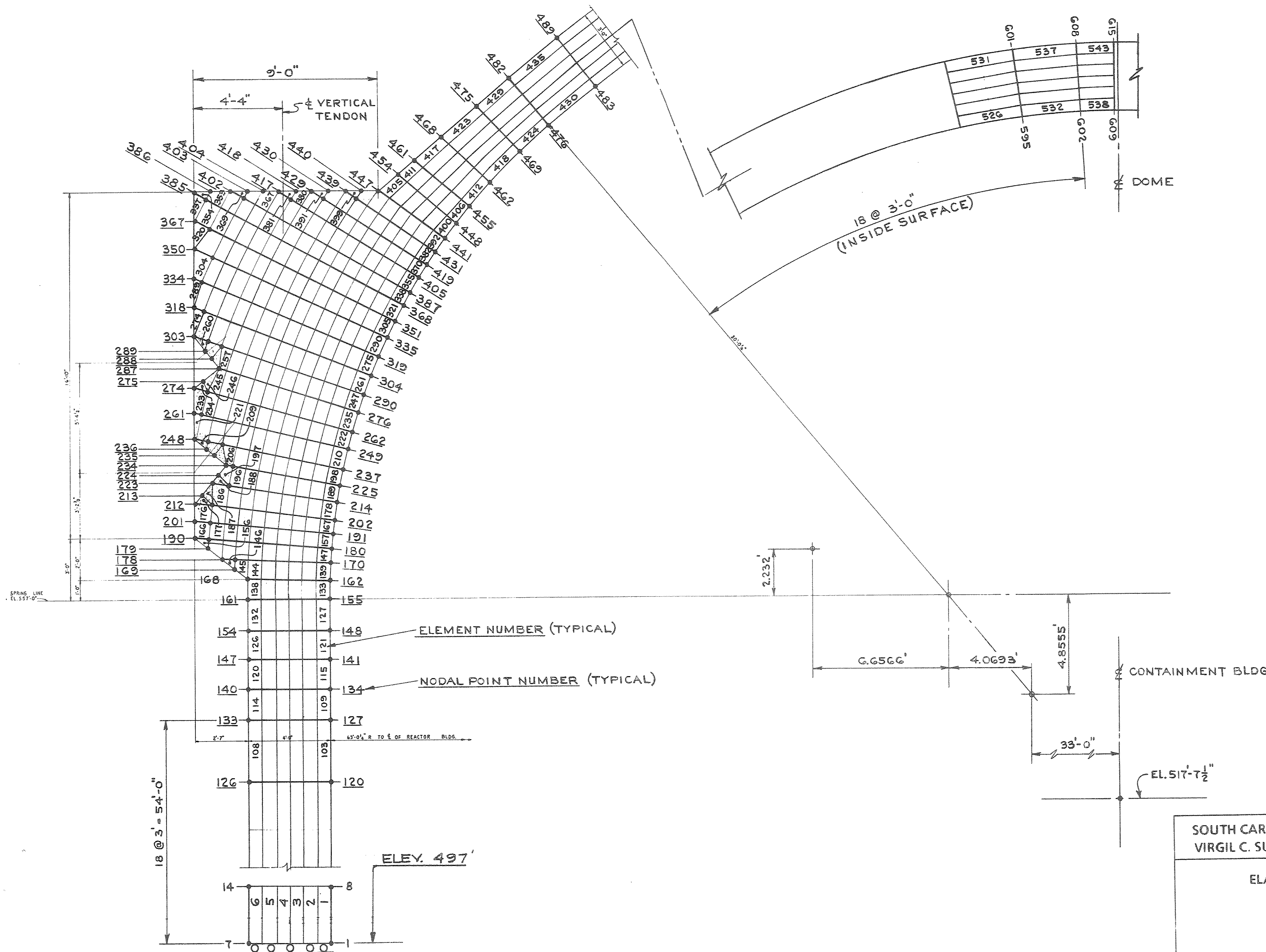
Figure 3.8-28



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Plain Stress Finite Element Model
For Buttress Analysis

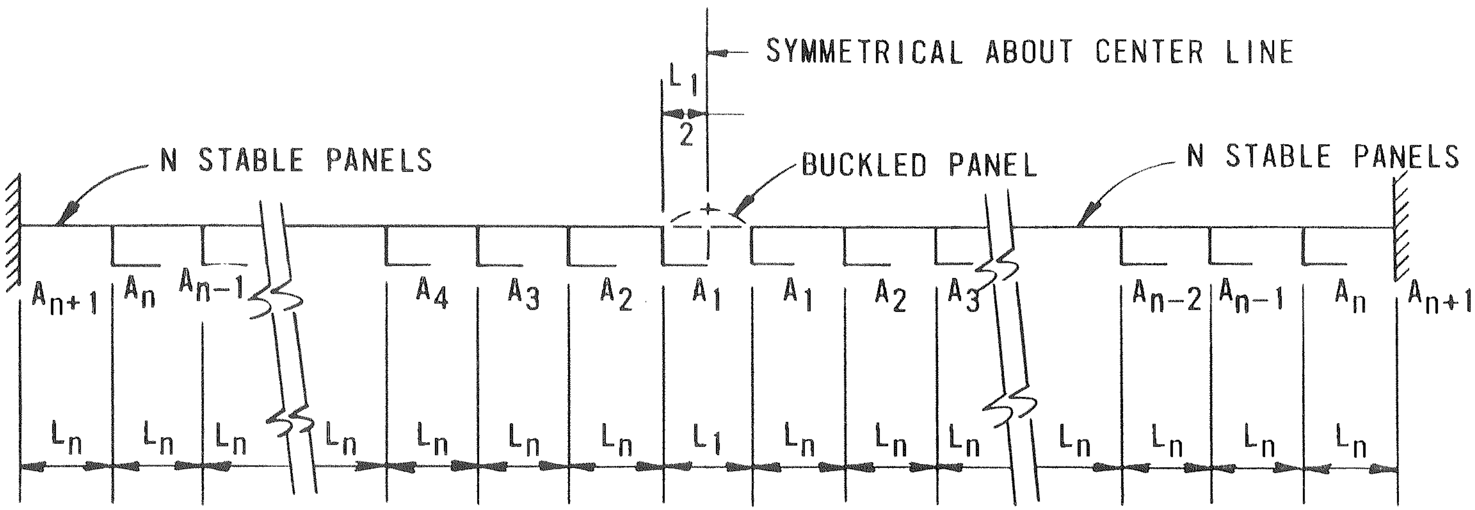
Figure 3.8-29



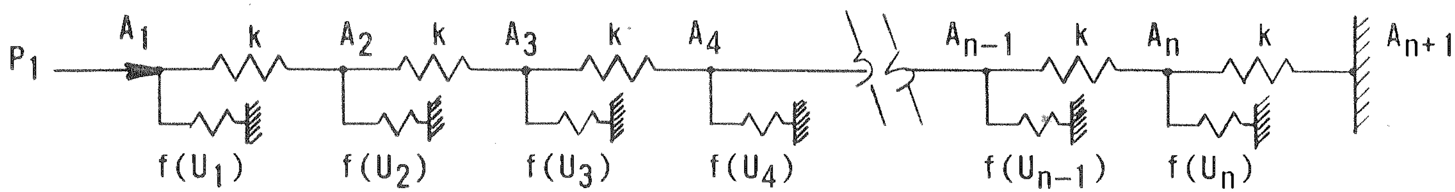
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

ELAD Model For Ring Girder Analysis

Figure 3.8-30



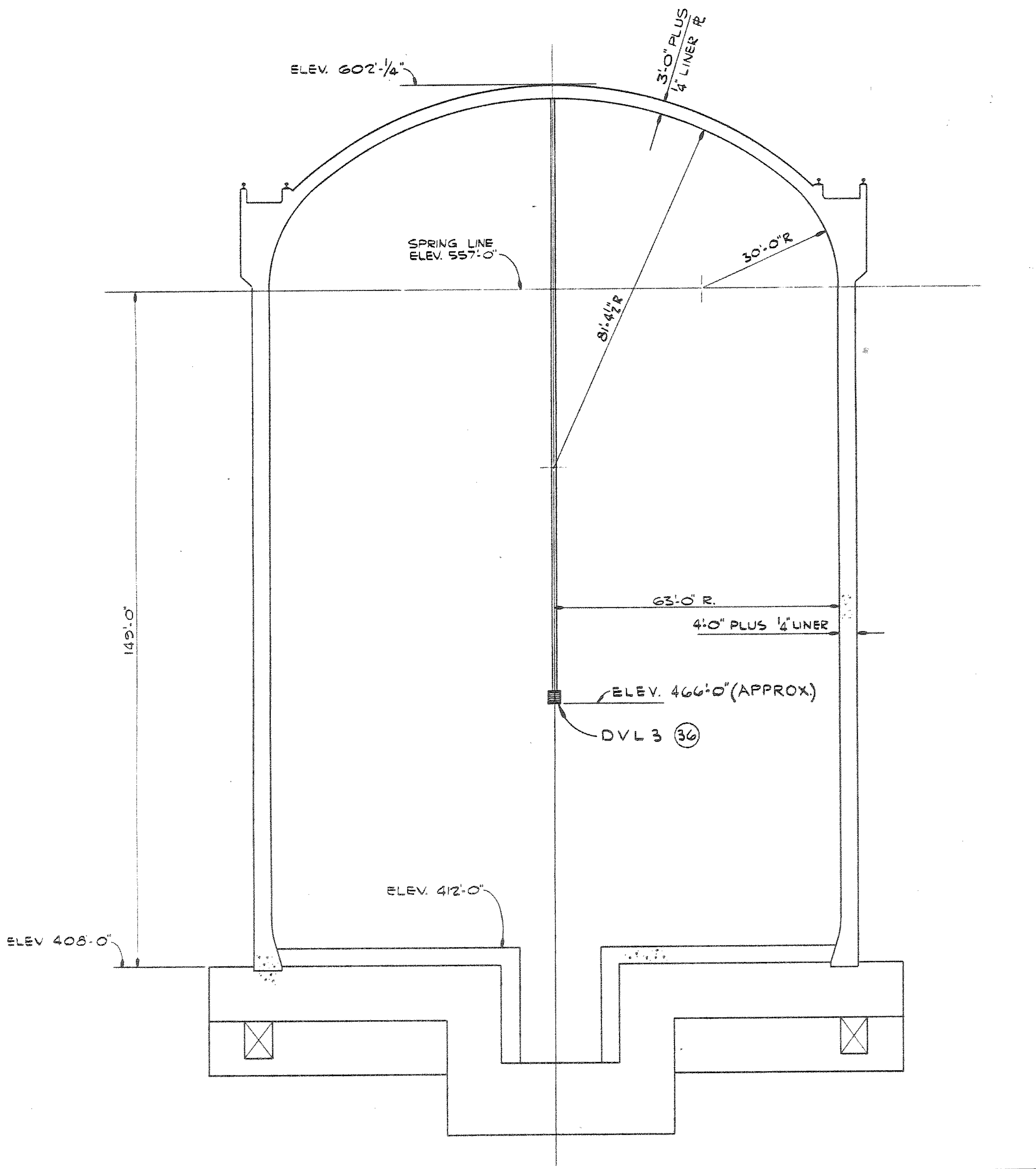
MODEL FOR ANALYSIS



SPRING-NODE REPRESENTATION OF MODEL

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

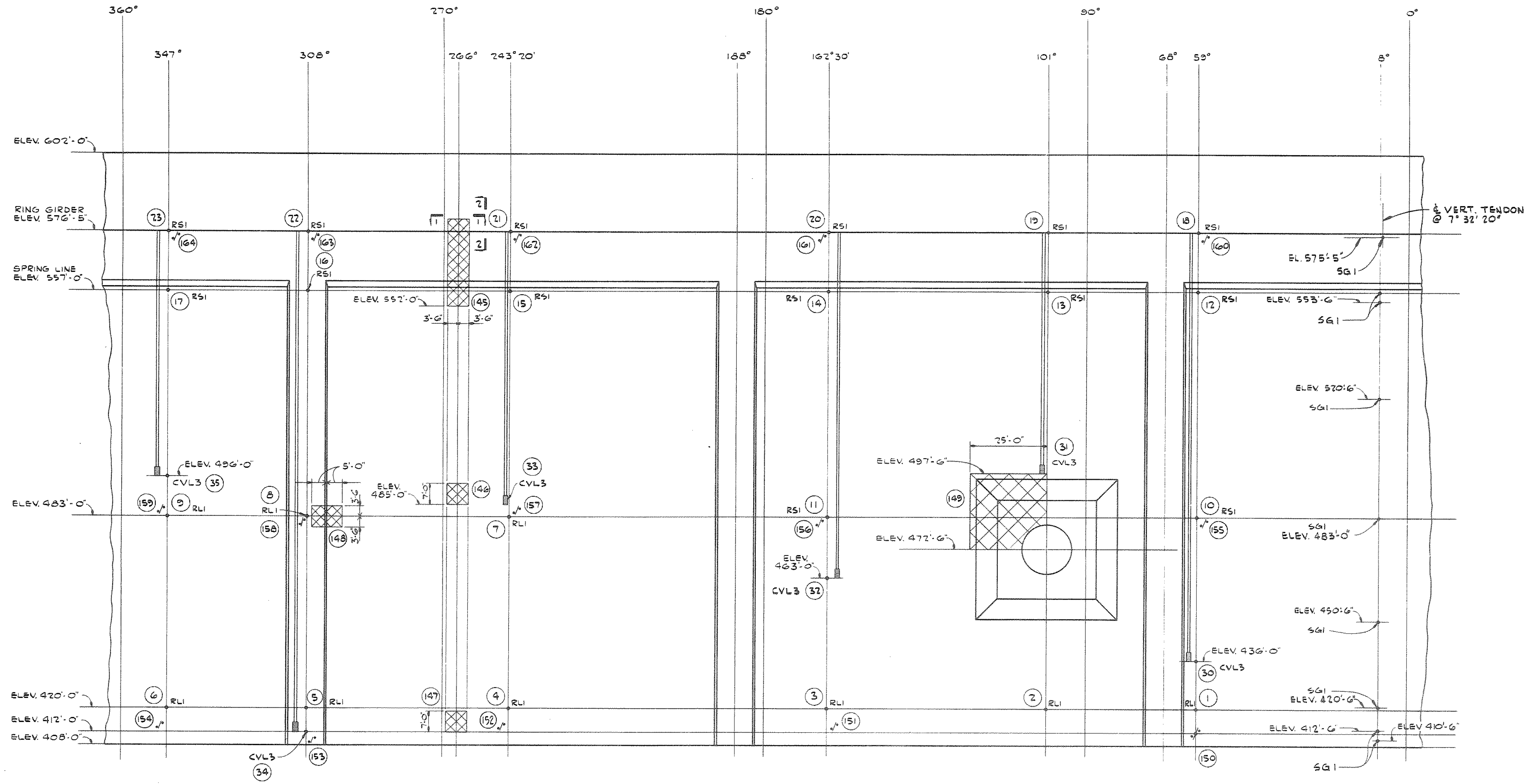
Liner - Anchorage
Model
Figure 3.8-30a

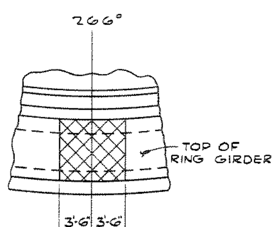


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

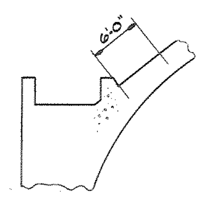
Structural Acceptance Test Instrumentation Reactor Building Cross Section and Location

Figure 3.8-31





SECTION 1-1



SECTION 2-2
FOR CRACK PATTERN

NOTES:-

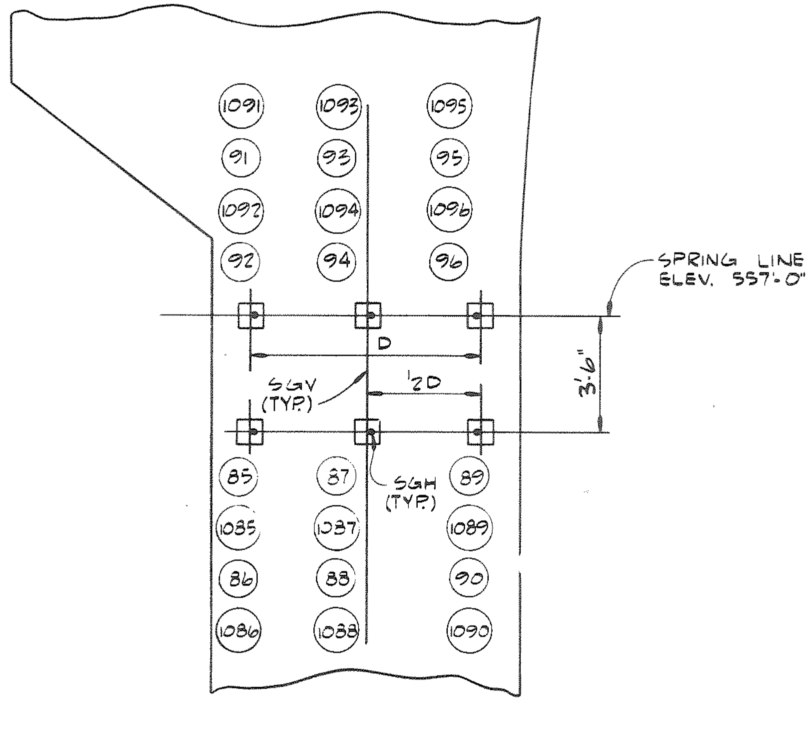
A. CYLINDER AND DOME RADIAL DISPLACEMENTS
   1. DCDT'S RL1
   2. SCALE AND TRANSITS RS1
B. EQUIPMENT ACCESS RADIAL, TANGENTIAL, AND VERTICAL DISPLACEMENT
   1. DCDT'S RL2, VL2, TL2, (RVTL2)
C. CYLINDER AND DOME VERTICAL DISPLACEMENT
   1. INVAR TAPE AND DCDT DVL3
   2. INVAR TAPES AND DCDT'S CVL3

NOTES:-(CONTINUED)

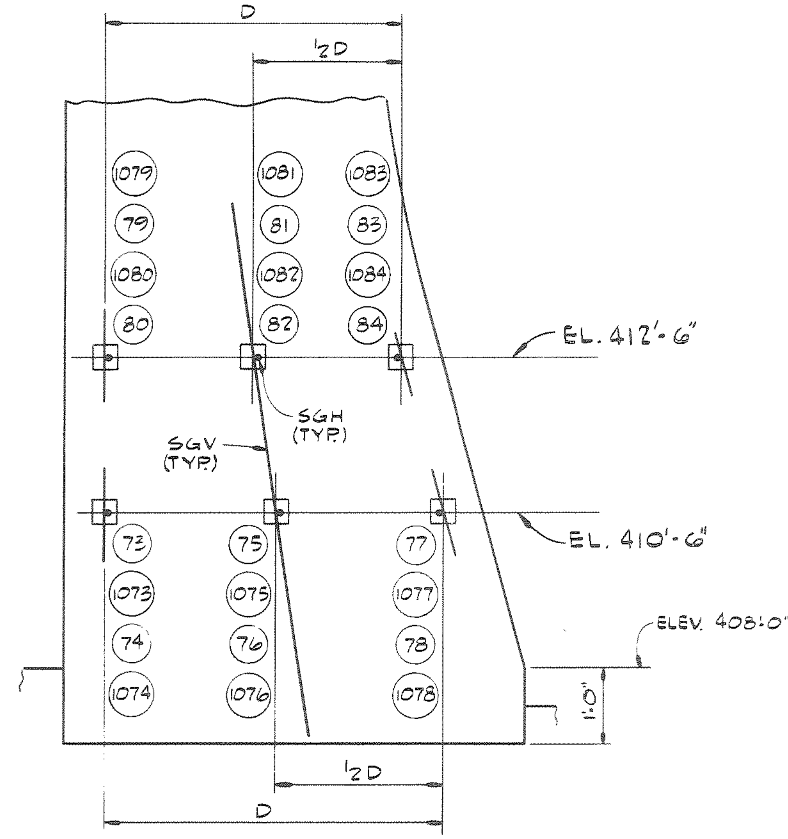
D. CRACK PATTERN
   1. SPECIAL PAINT
E. STRAIN MEASUREMENTS OF CYLINDER, EQUIPMENT ACCESS, RING GIRDER, AND CONCRETE UNDER VERTICAL TENDON
   1. STRAIN GAGES SG1
F. TEMPERATURE MEASUREMENTS
   1. THERMOCOUPLES

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

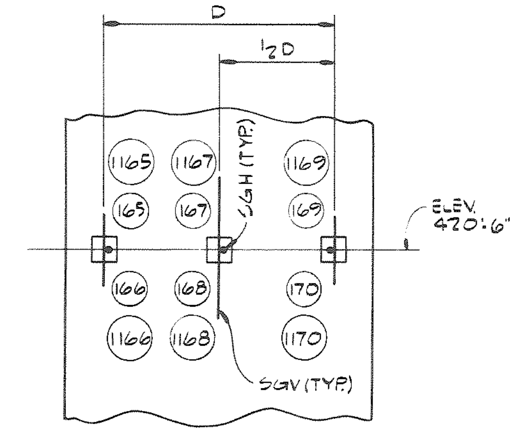
Structural Acceptance Test
Instrumentation - Layout and Location -
Reactor Building Exterior Stretchout
(GAI Dwg. No. D-400-223)
Figure 3.8-32



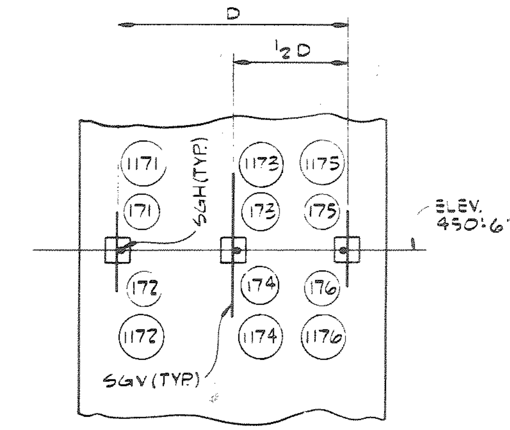
SECTION BELOW RING GIRDER
AT 8° AZIMUTH



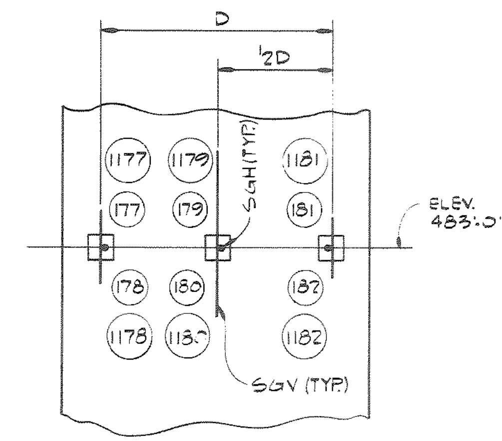
SECTION ABOVE BASE MAT
AT 8° AZIMUTH



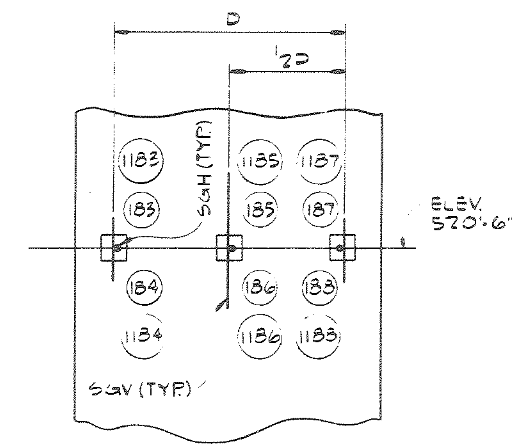
SECTION @ ELEV. 420'-6"
AT 8° AZIMUTH



SECTION @ ELEV. 450'-6"
AT 8° AZIMUTH



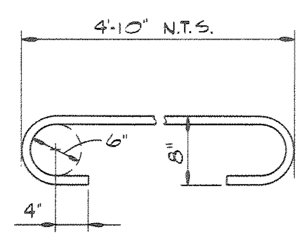
SECTION @ ELEV. 483'-0"
AT 8° AZIMUTH



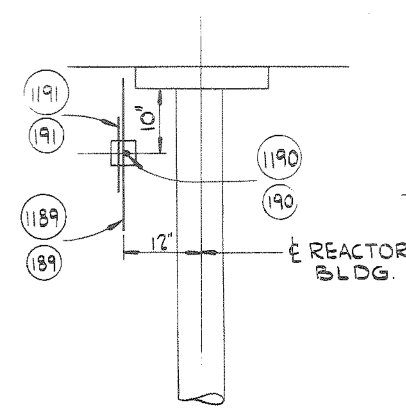
SECTION @ ELEV. 520'-6"
AT 8° AZIMUTH

NOTES

1. STRAIN GAGES SGI(SGV & SGH)
2. THE STRAIN GAGES ARE ATTACHED TO A SISTER REINFORCING BAR. SEE DETAIL I FOR SISTER REINFORCING BAR DETAILS.
3. THE SISTER REINFORCING BARS FOR THE INNER AND OUTER FACES WILL BE ATTACHED TO THE MAIN REINFORCEMENT BUT MAINTAIN MINIMUM CONCRETE COVER REQUIREMENTS.
4. THE MIDPOINT SISTER REINFORCING BARS SHALL BE SECURELY ATTACHED AT THEIR POSITION TO PREVENT MOVEMENT DURING PLACEMENT OF CONCRETE.



DETAIL I

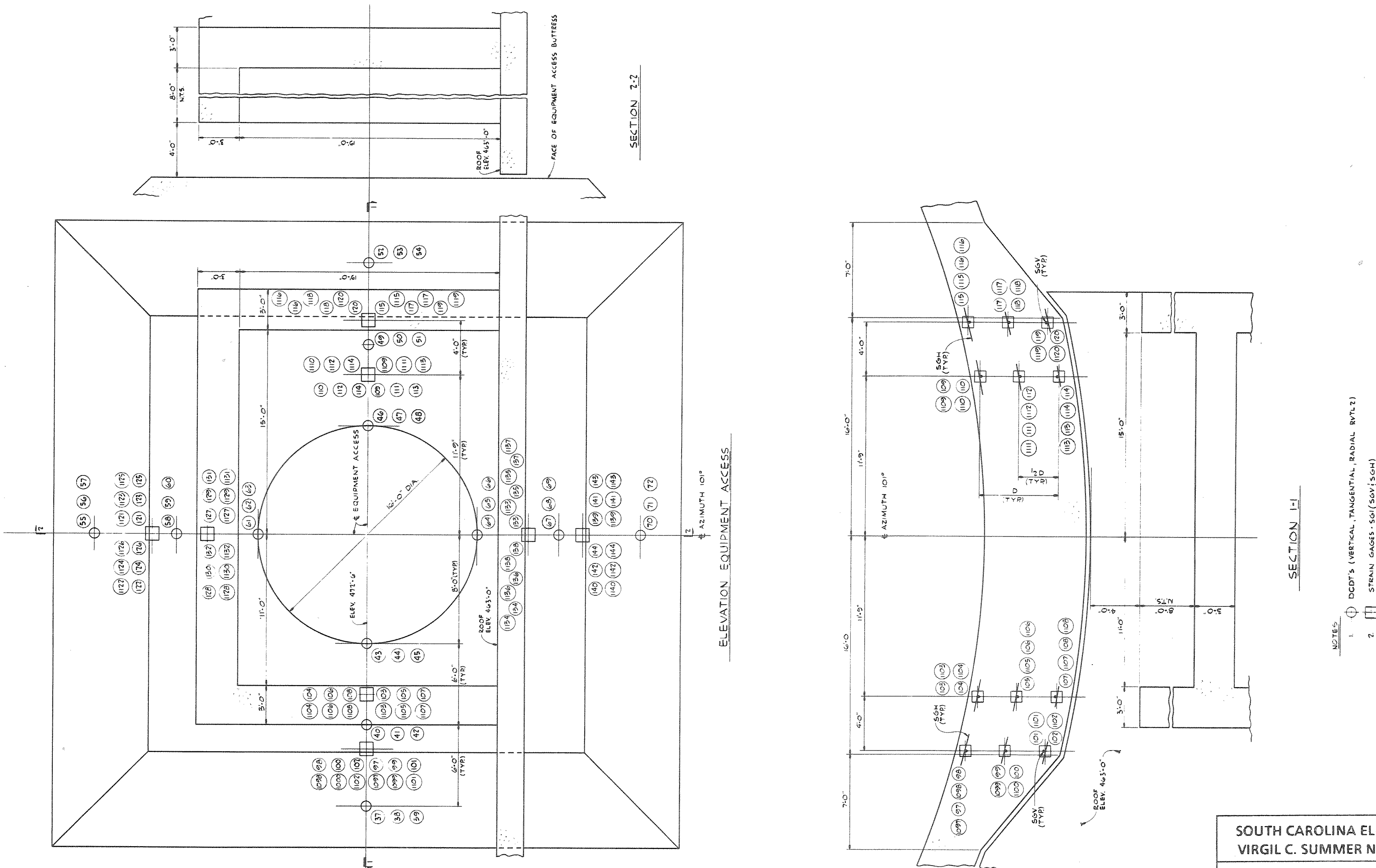


SECTION @ 7°32'20"
VERTICAL TENDON

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Structural Acceptance Test
Instrumentation - Layout and Location -
Reactor Building Section at 8° Azimuth

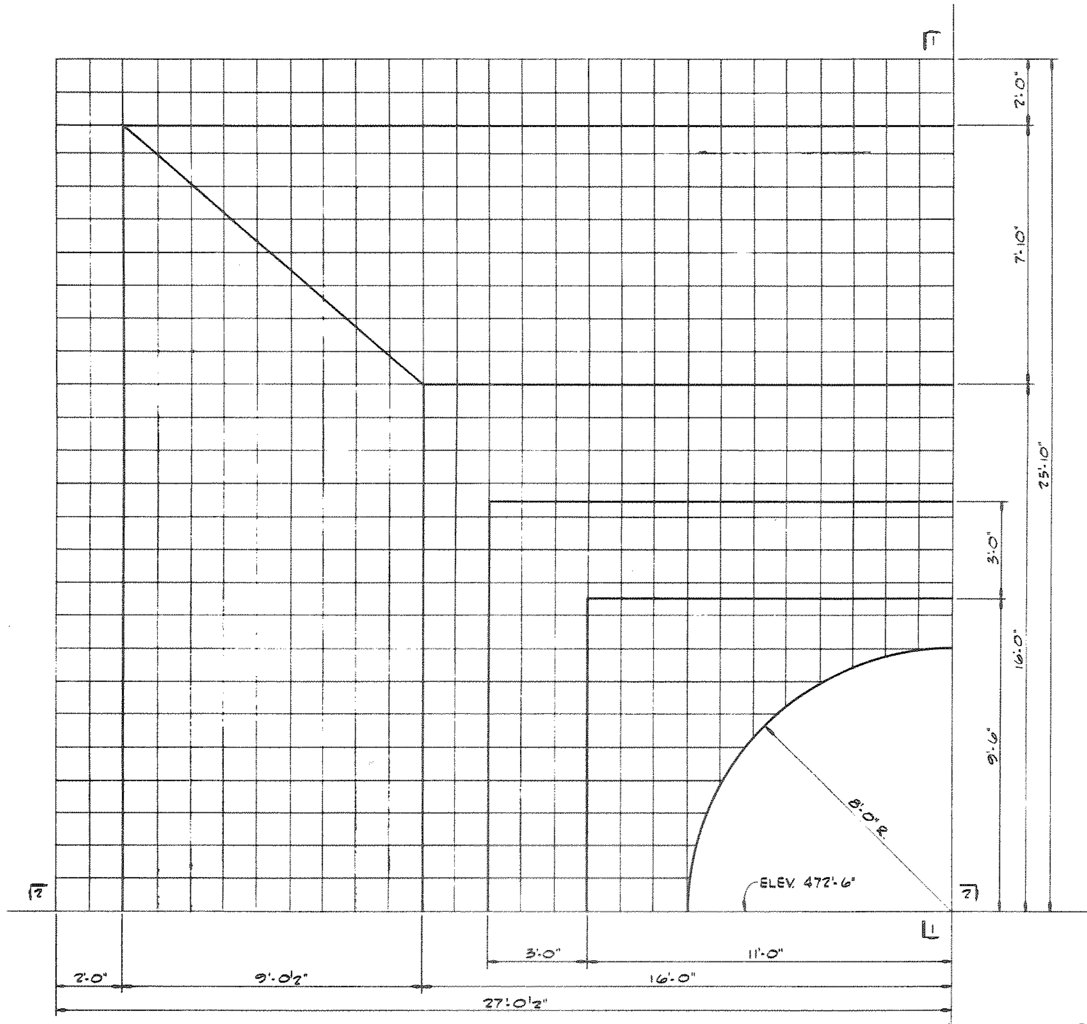
Figure 3.8-33



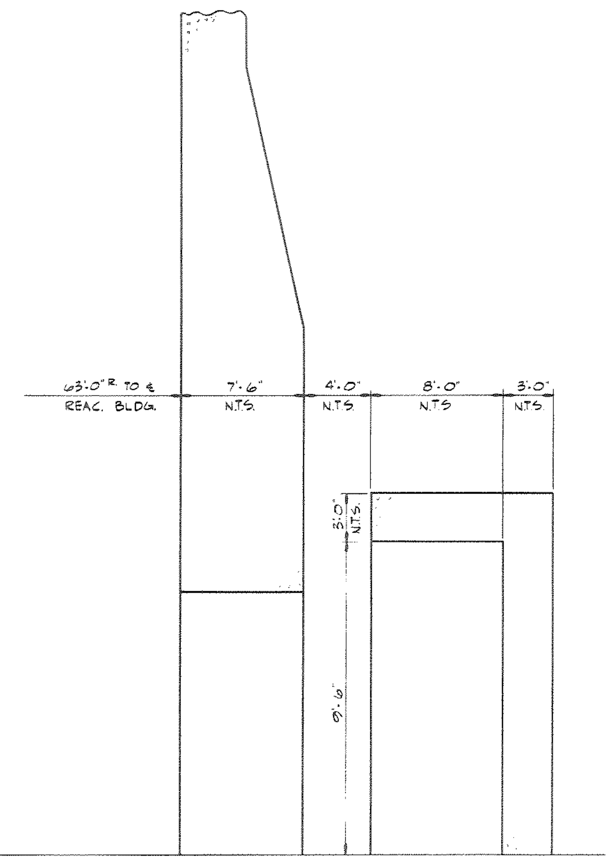
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Structural Acceptance Test
Instrumentation - Layout and Location
Equipment Access

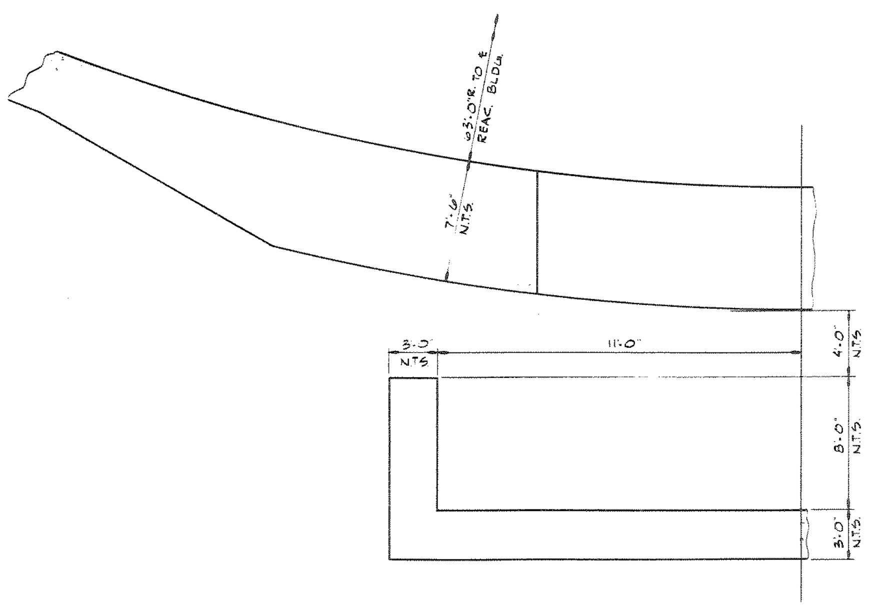
Figure 3.8-34



AREA No. 149
DEVELOPED



SECTION 1-1



SECTION 2-2

NOTES

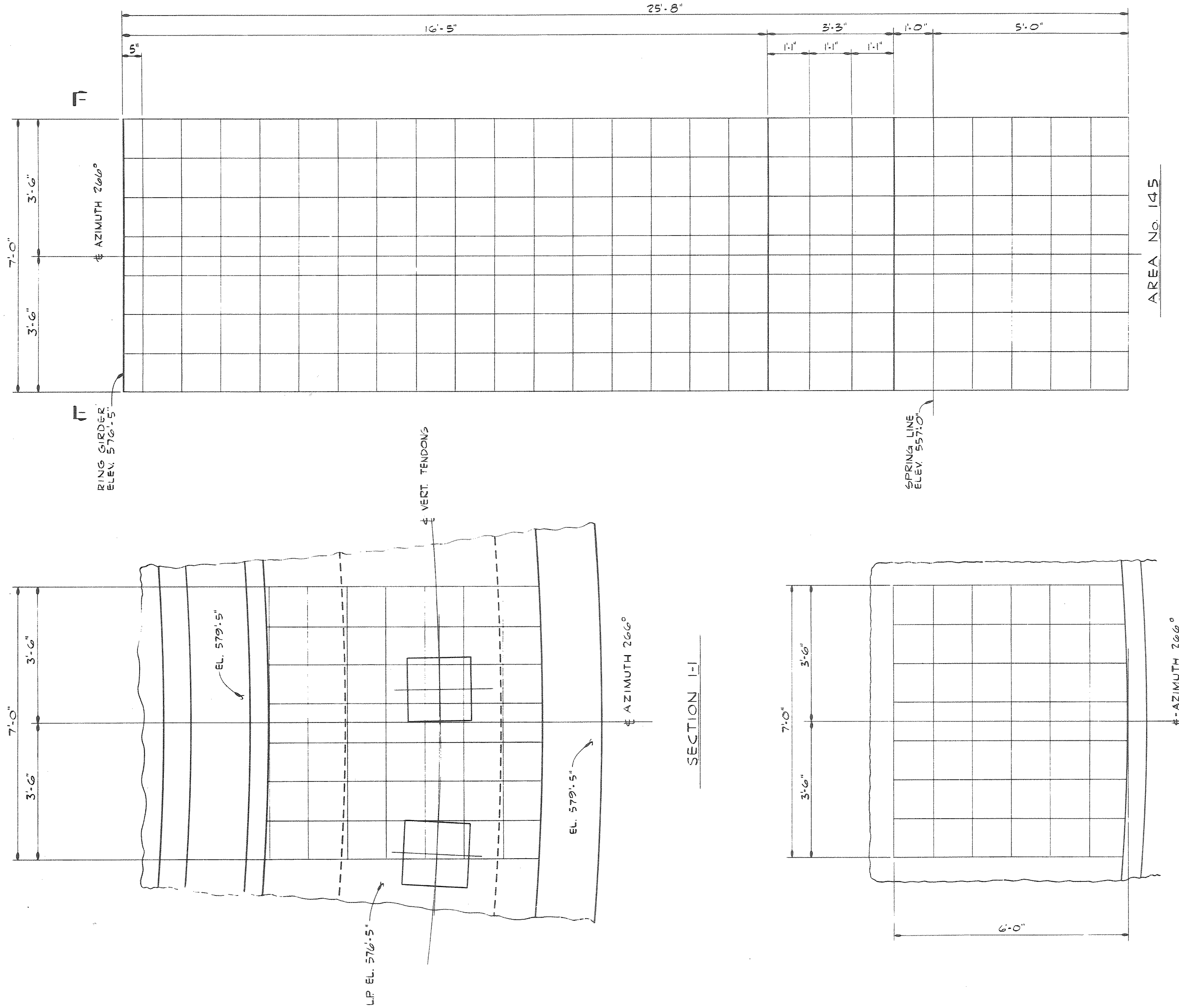
1. 1'-0" x 1'-0" GRID SHOWN TO AID SKETCHING OF CRACK PATTERNS.

2. CRACK PATTERN WILL ESSENTIALLY BE THAT OF EXISTING HAIRLINE SHRINKAGE CRACKS. CRACK WIDTHS SHOULD BE LESS THAN 0.010

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Structural Acceptance Test
Instrumentation - Layout and Location
Crack Patterns - Equipment Access

Figure 3.8-35

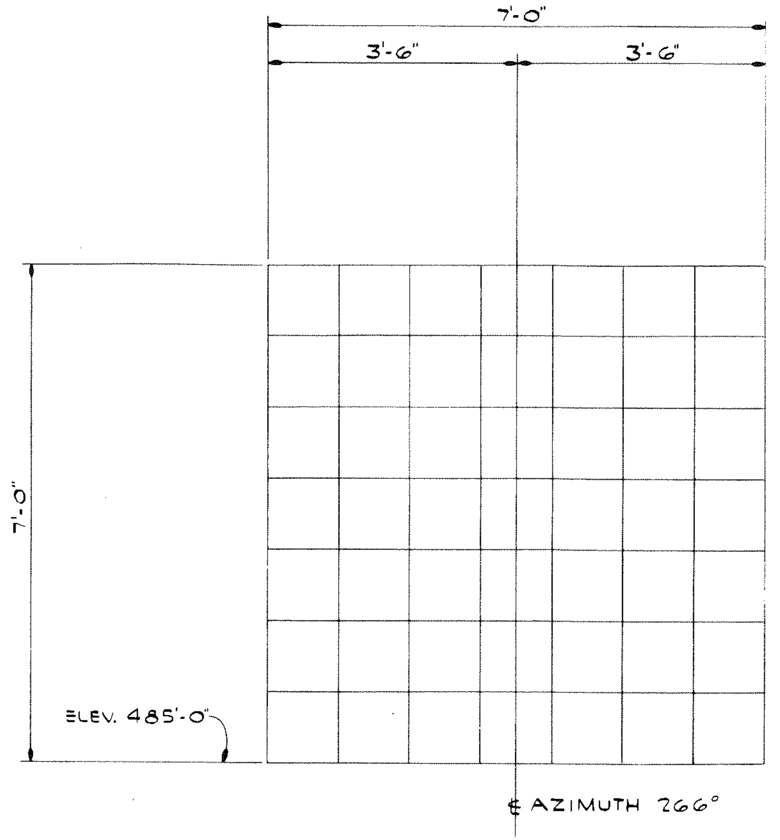


NOTES:
1. 1'-0"x1'-0" GRIDS SHOWN TO AID SKETCHING OF CRACK PATTERNS, UNLESS SHOWN OTHERWISE
2. CRACK PATTERN WILL ESSENTIALLY BE THAT OF EXISTING HAIRLINE SHRINKAGE CRACKS. CRACK WIDTHS SHOULD BE LESS THAN 0.010 INCH.

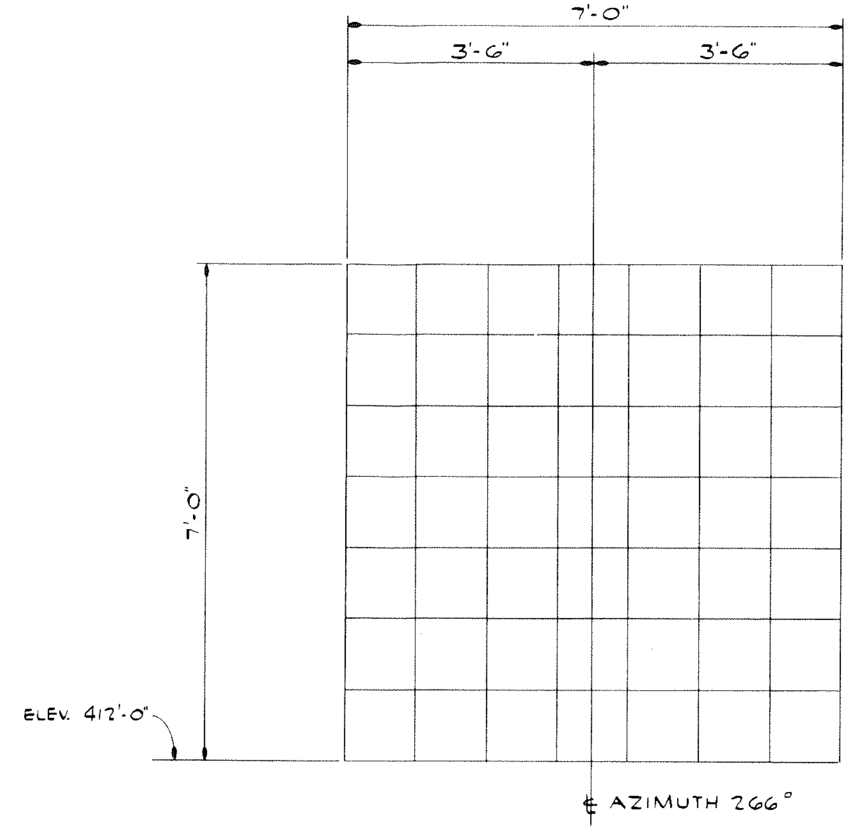
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Structural Acceptance Test
Instrumentation Layout and Location
Crack Patterns - Wall and Dome Junction

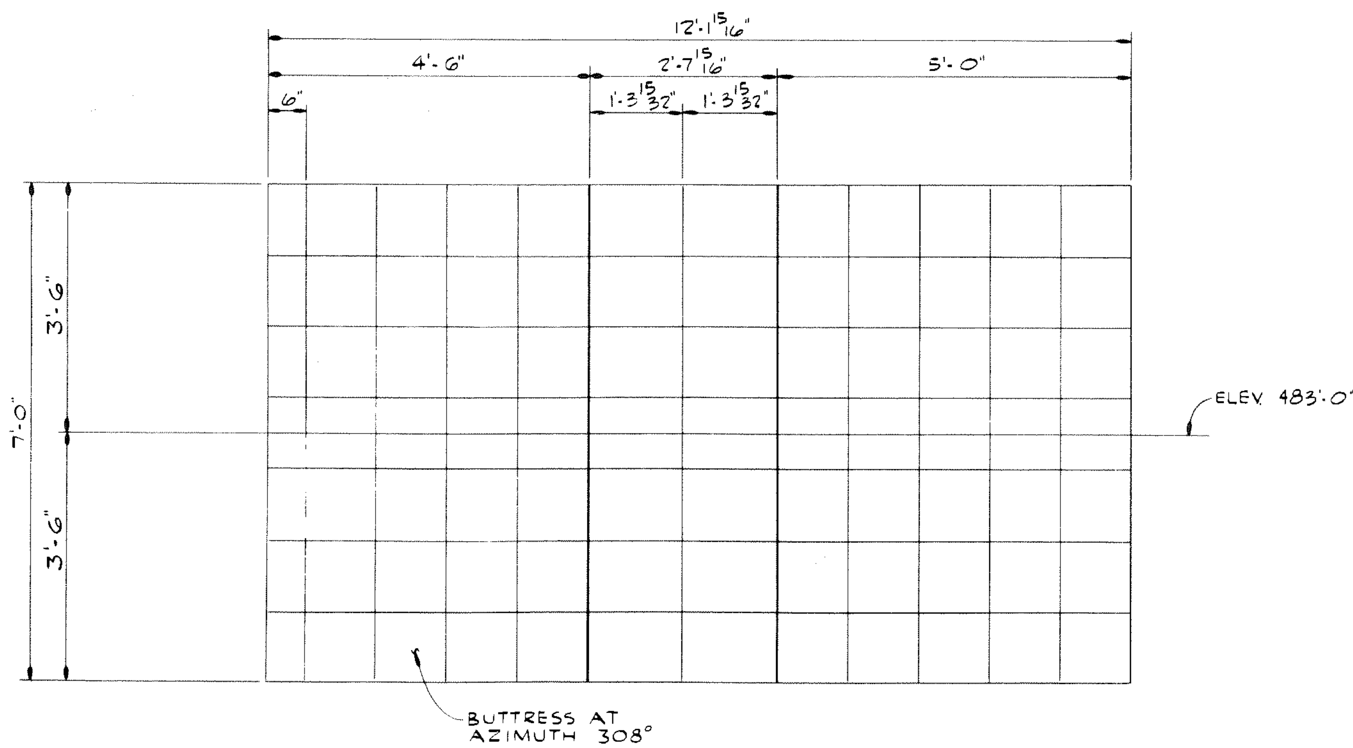
Figure 3.8-36



AREA No. 146



AREA No. 147



AREA No. 148

NOTES:

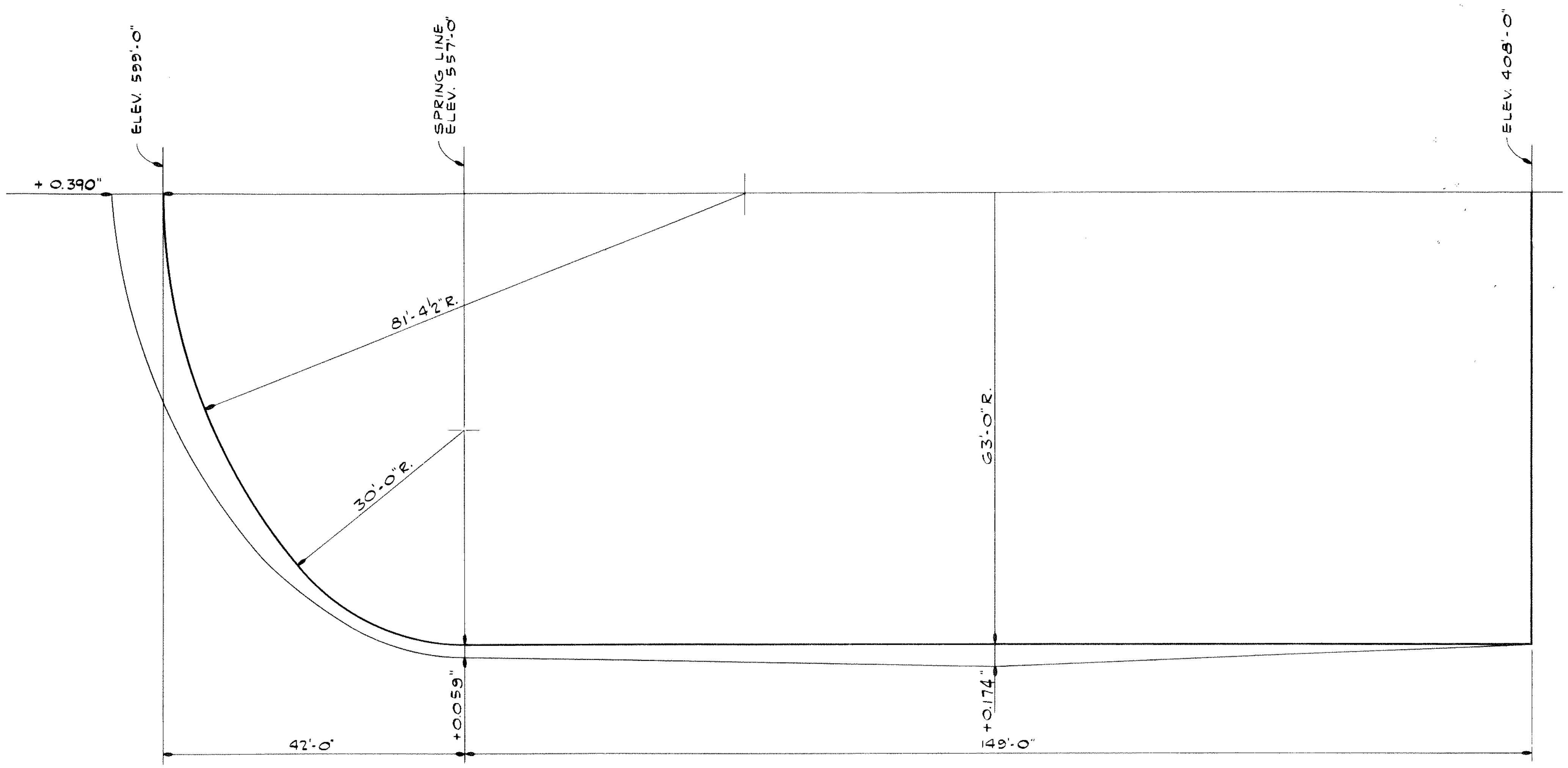
1. 1'-0" × 1'-0" GRID SHOWN TO AID SKETCHING OF CRACK PATTERN

2. CRACK PATTERN WILL ESSENTIALLY BE THAT OF EXISTING HAIRLINE SHRINKAGE CRACKS. CRACK WIDTHS SHOULD BE LESS THAN 0.010 INCH.

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Structural Acceptance Test
Instrumentation Layout and Location
Crack Patterns - Walls

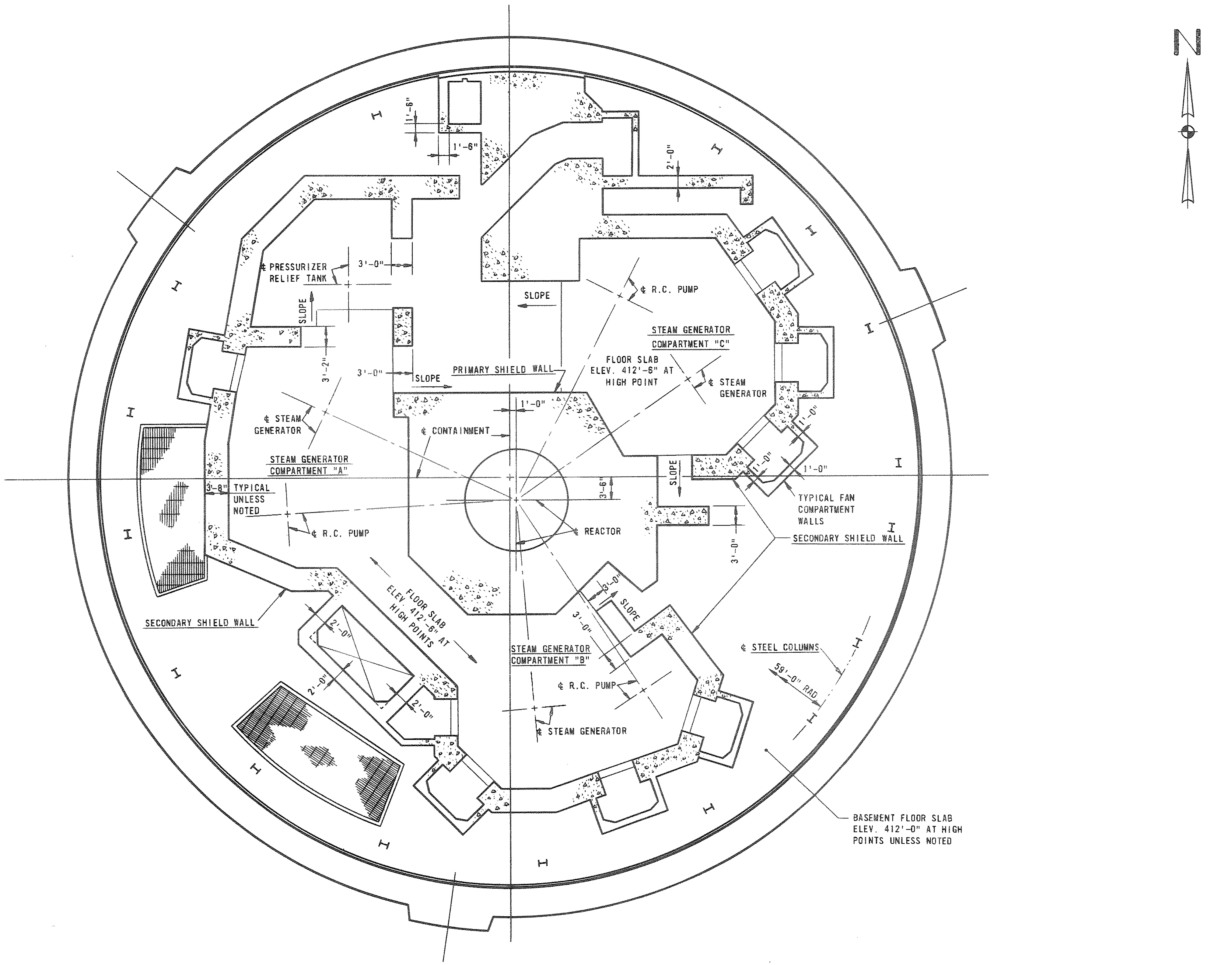
Figure 3.8-37



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Structural Acceptance Test Typical Radial Displacements - Reactor Building

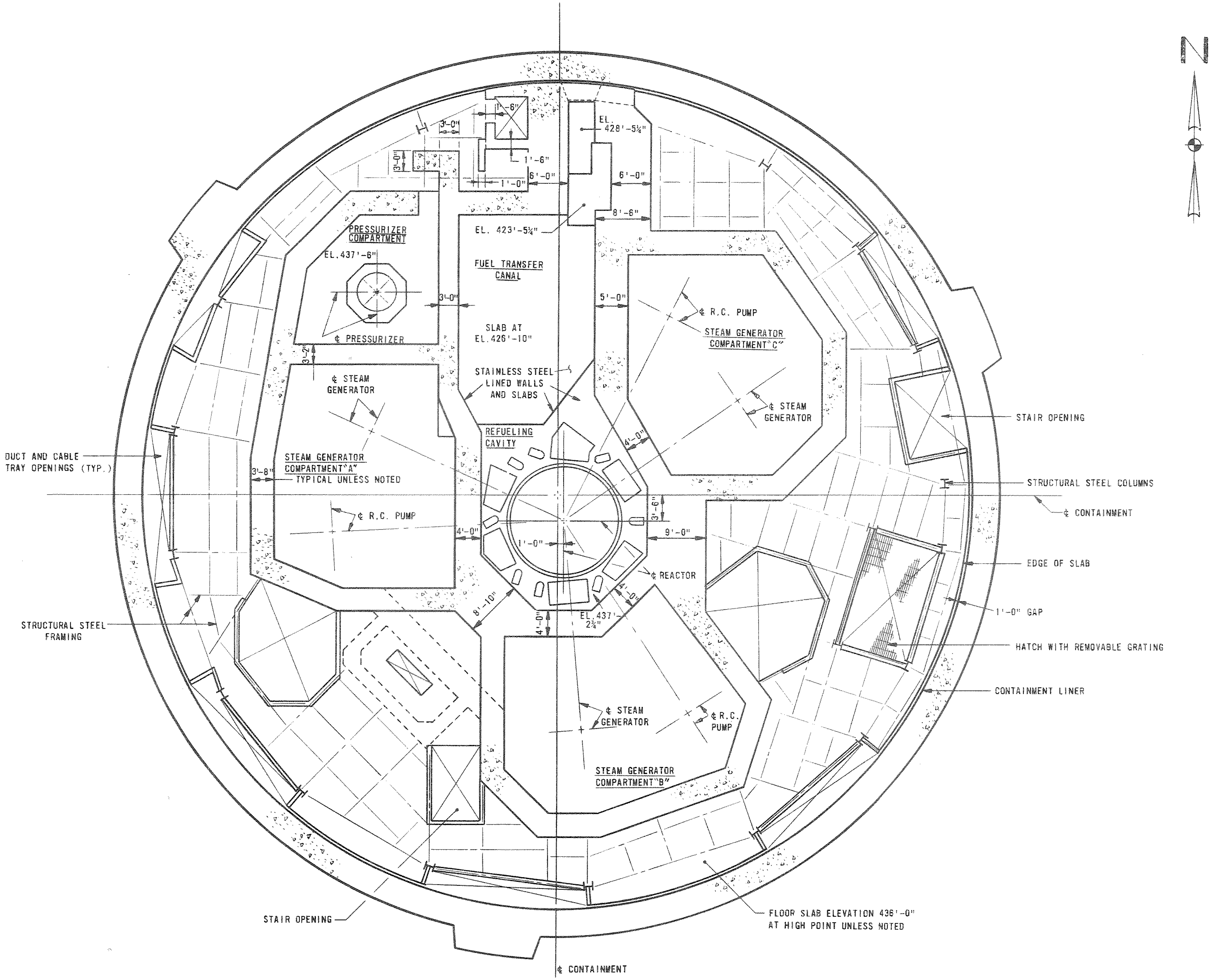
Figure 3.8-38



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Interior Structures Plan Above Elevation 412' - 0"

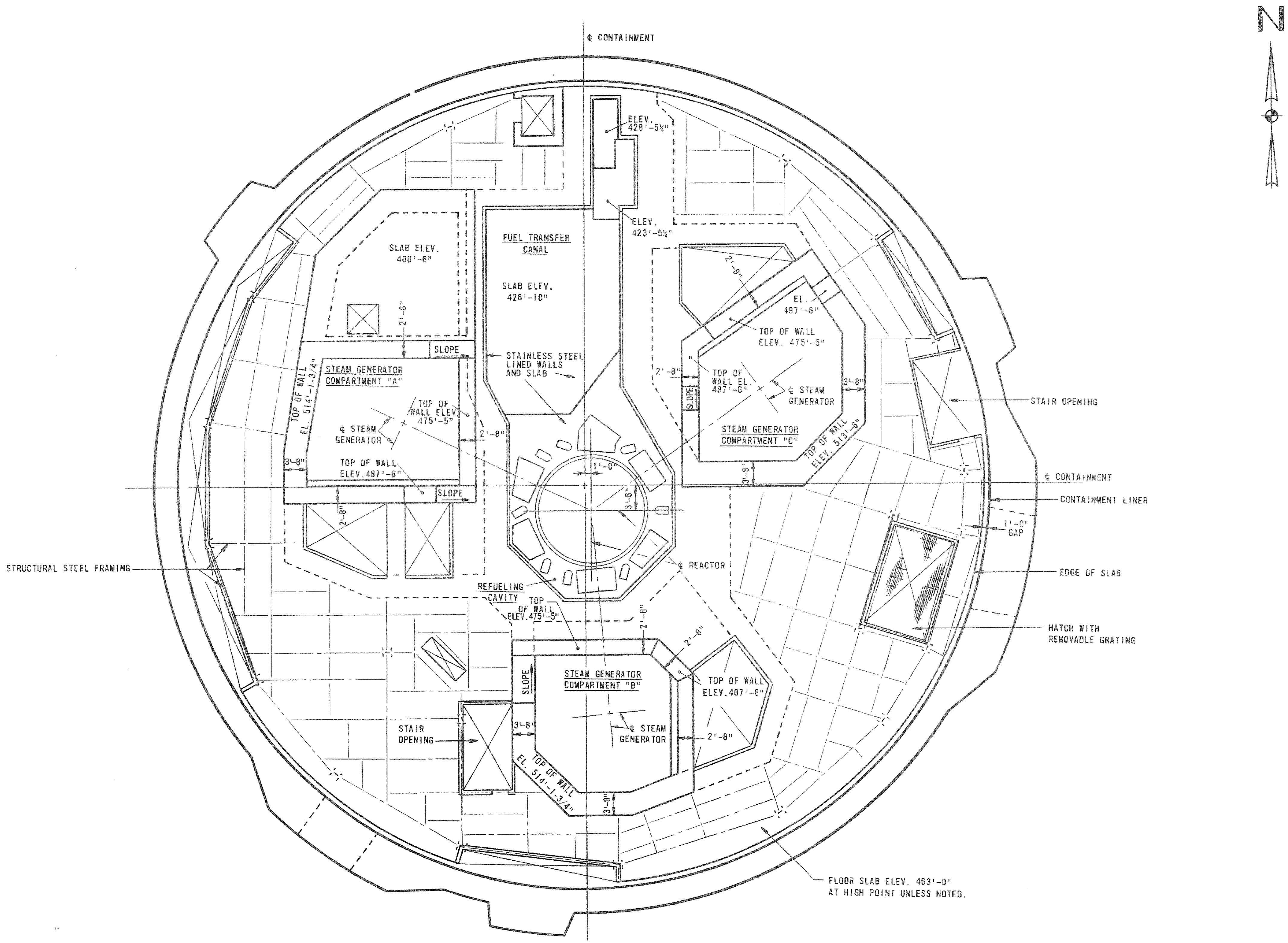
Figure 3.8-39



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Interior Structures Plan
Above Elevation 436' - 0"

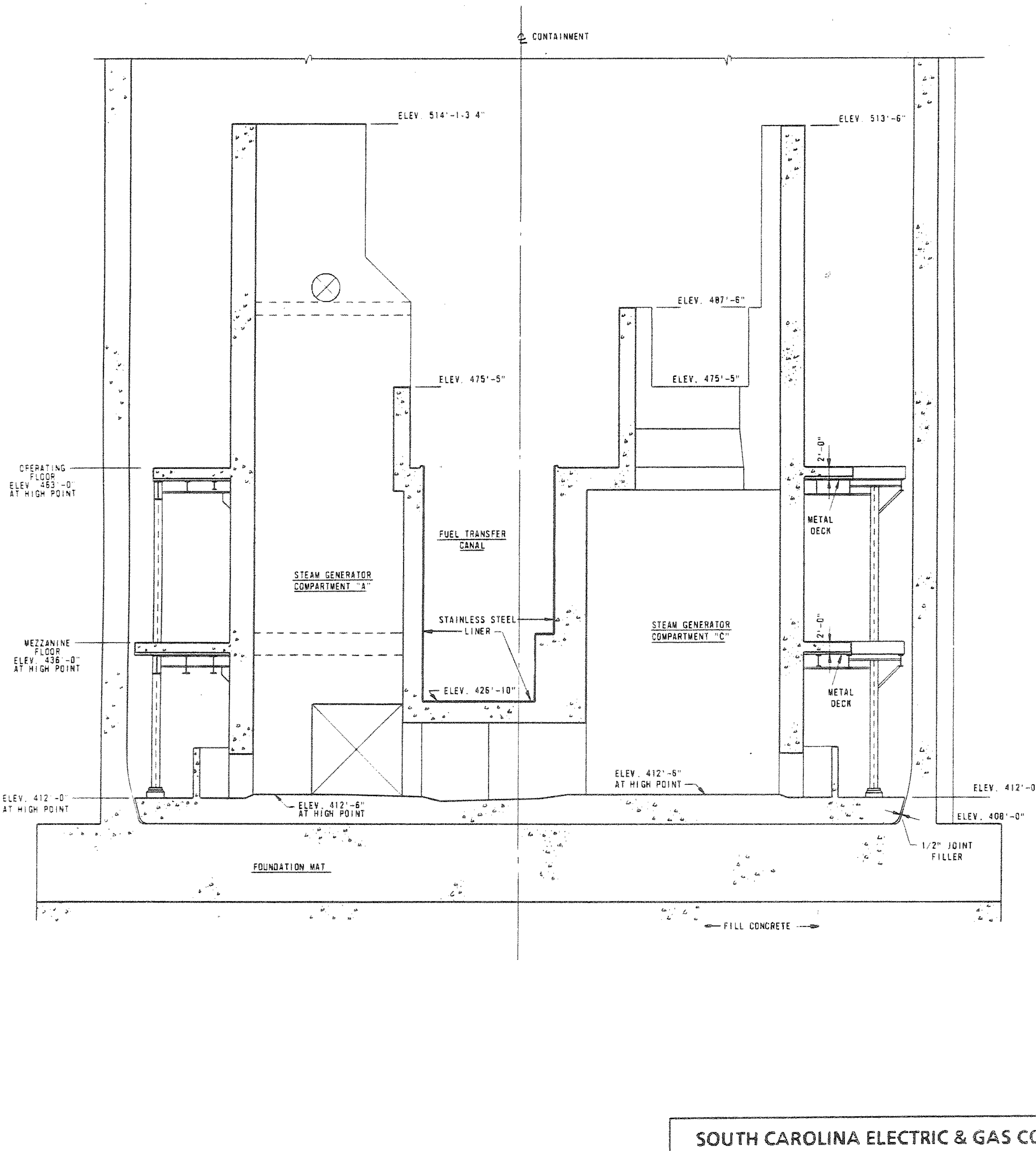
Figure 3.8-40



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Interior Structures Plan Above Elevation 463' - 0"

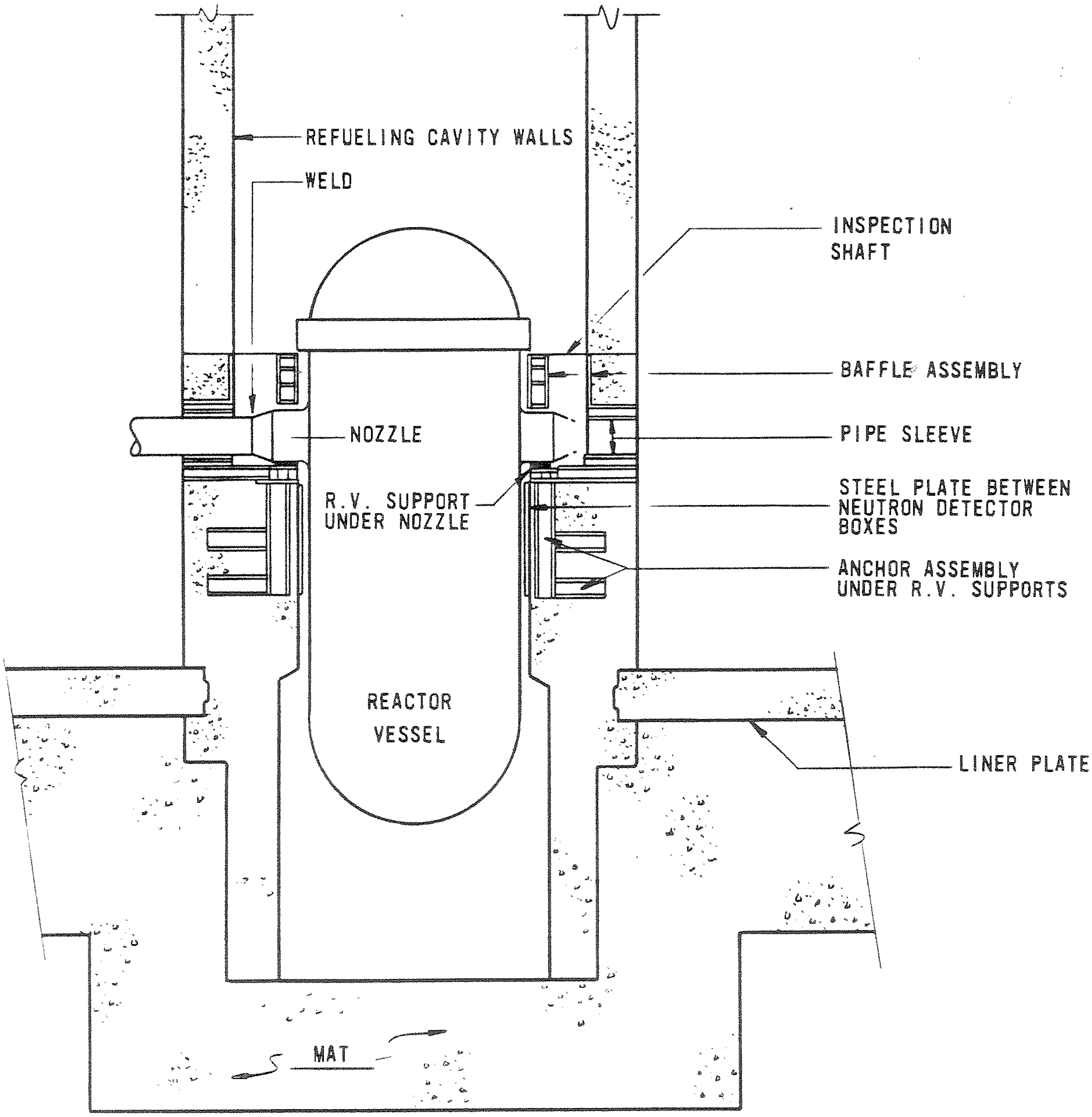
Figure 3.8-41



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Section Through Reactor Building
Looking North

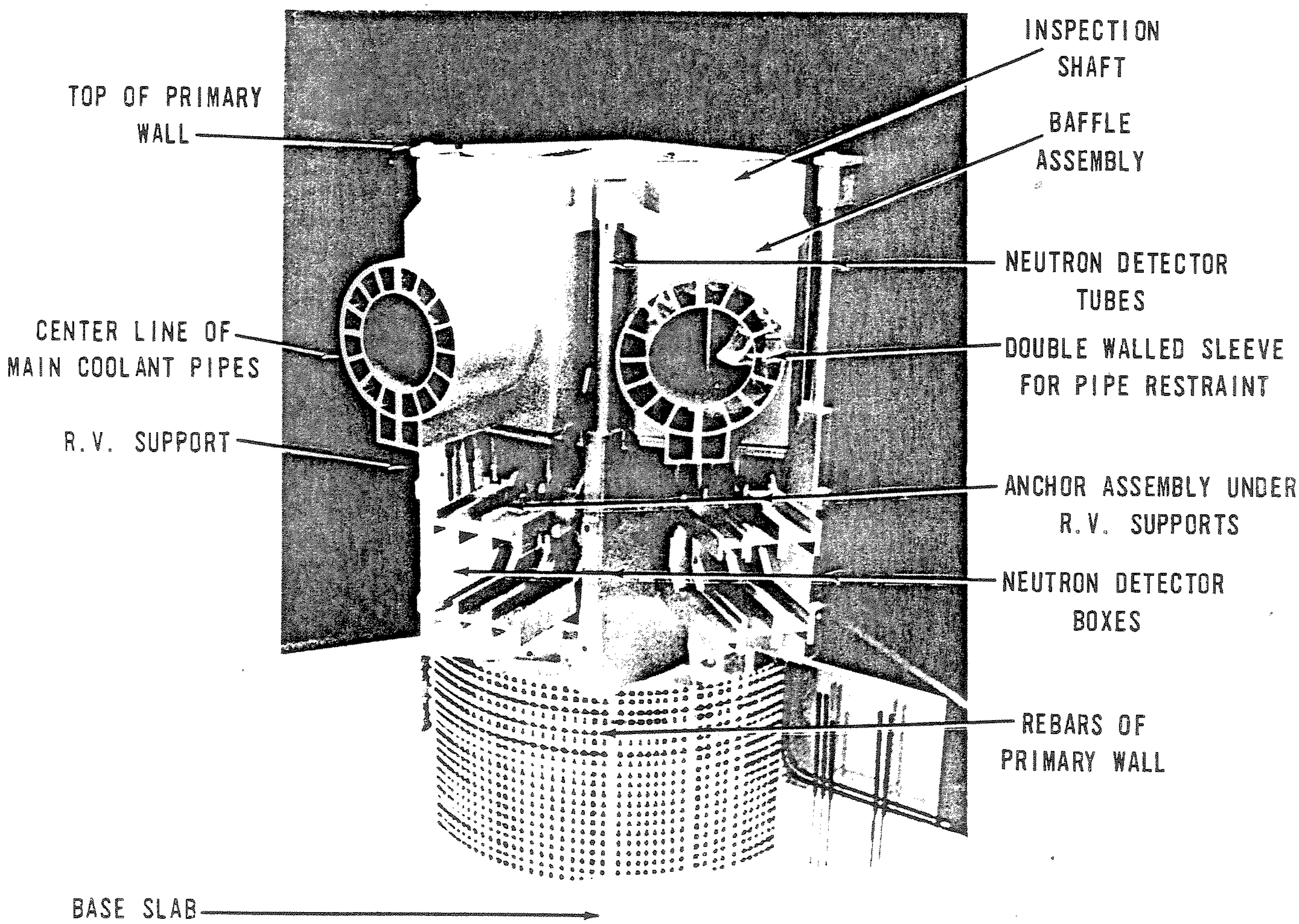
Figure 3.8-42



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Section Through
Primary Shield Wall

Figure 3.8-43

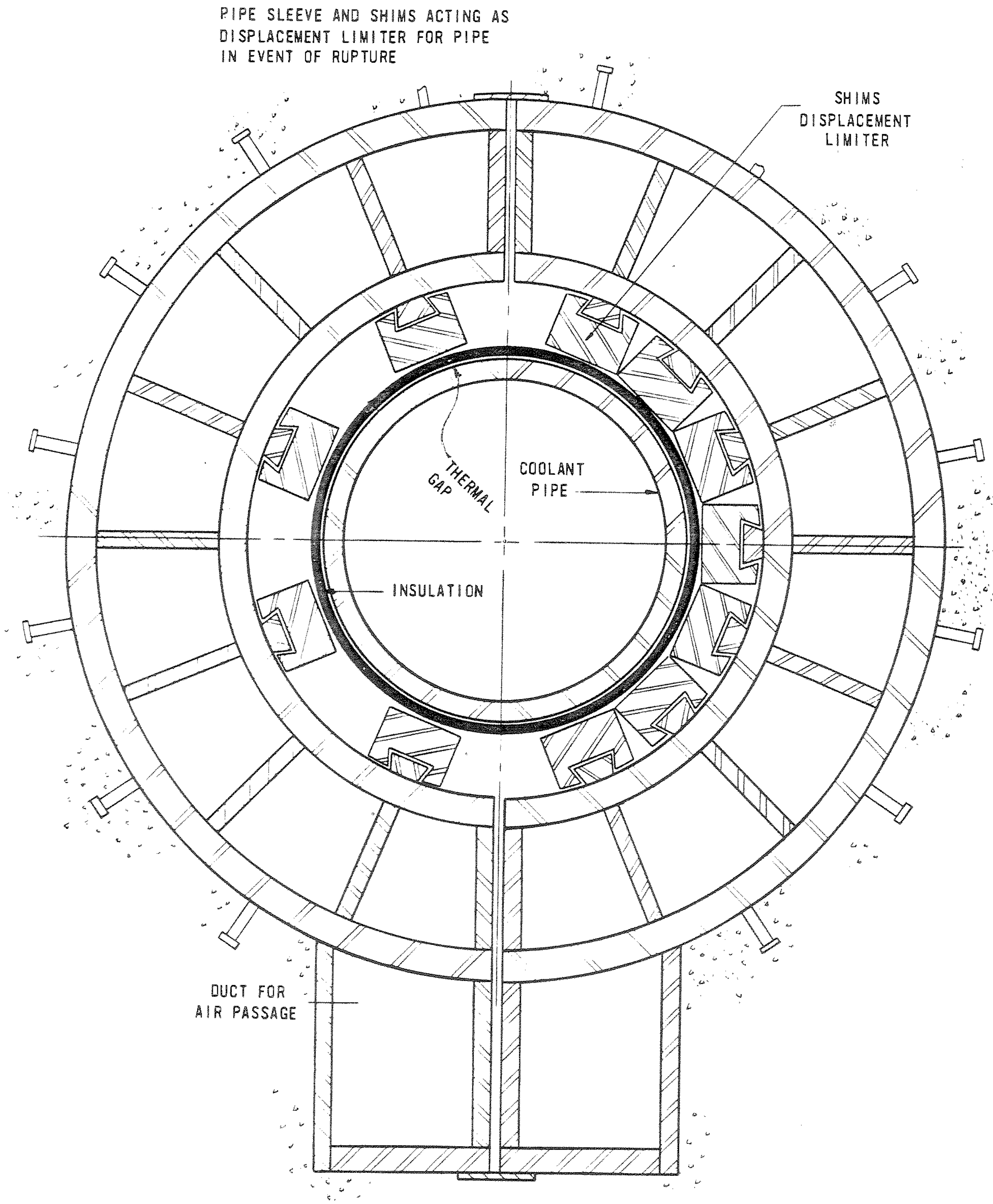


(CONCRETE NOT SHOWN FOR CLARITY)

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Outside View of
Primary Wall

Figure 3.8-44

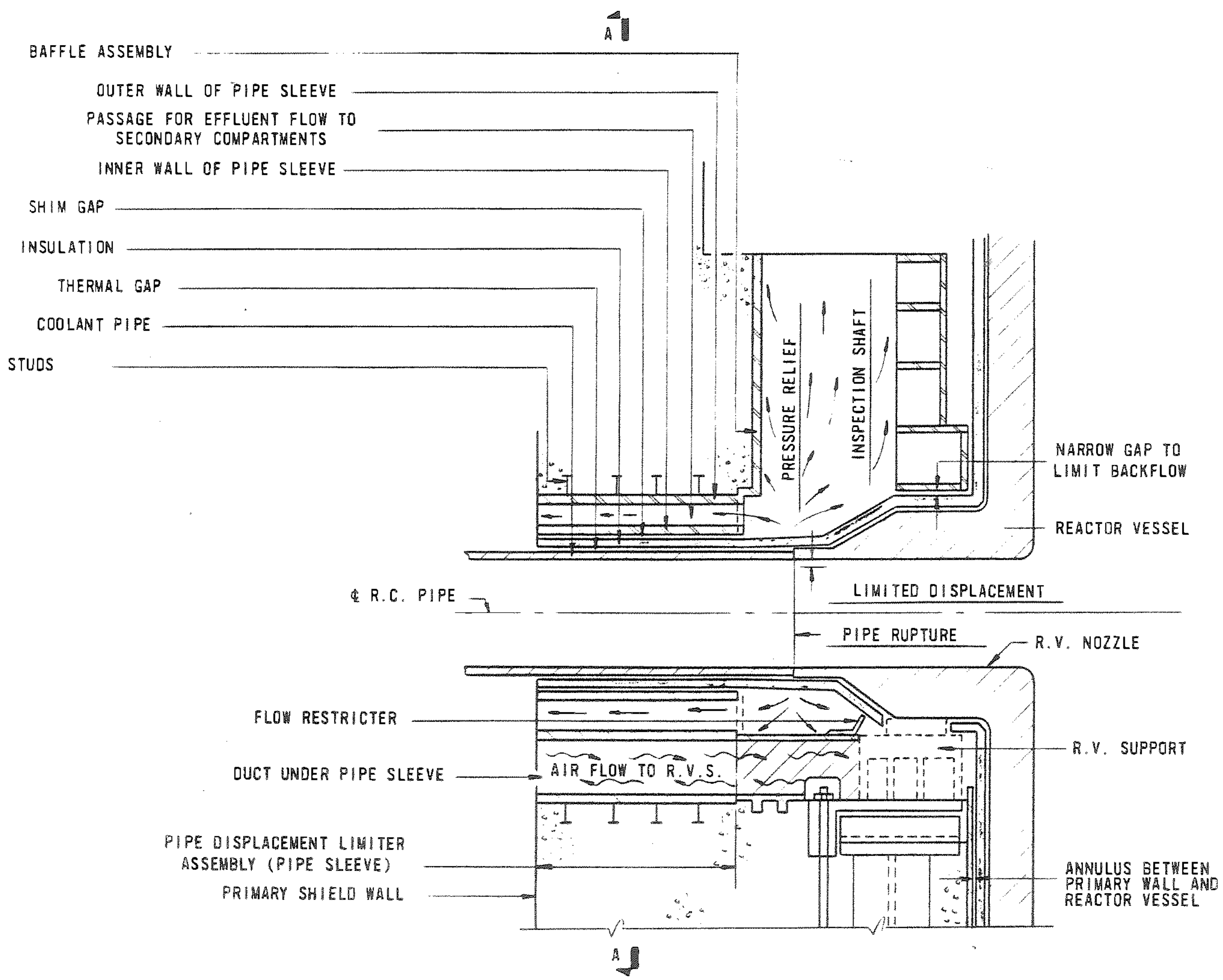


SECTION A-A

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Primary Coolant
Pipe Sleeve

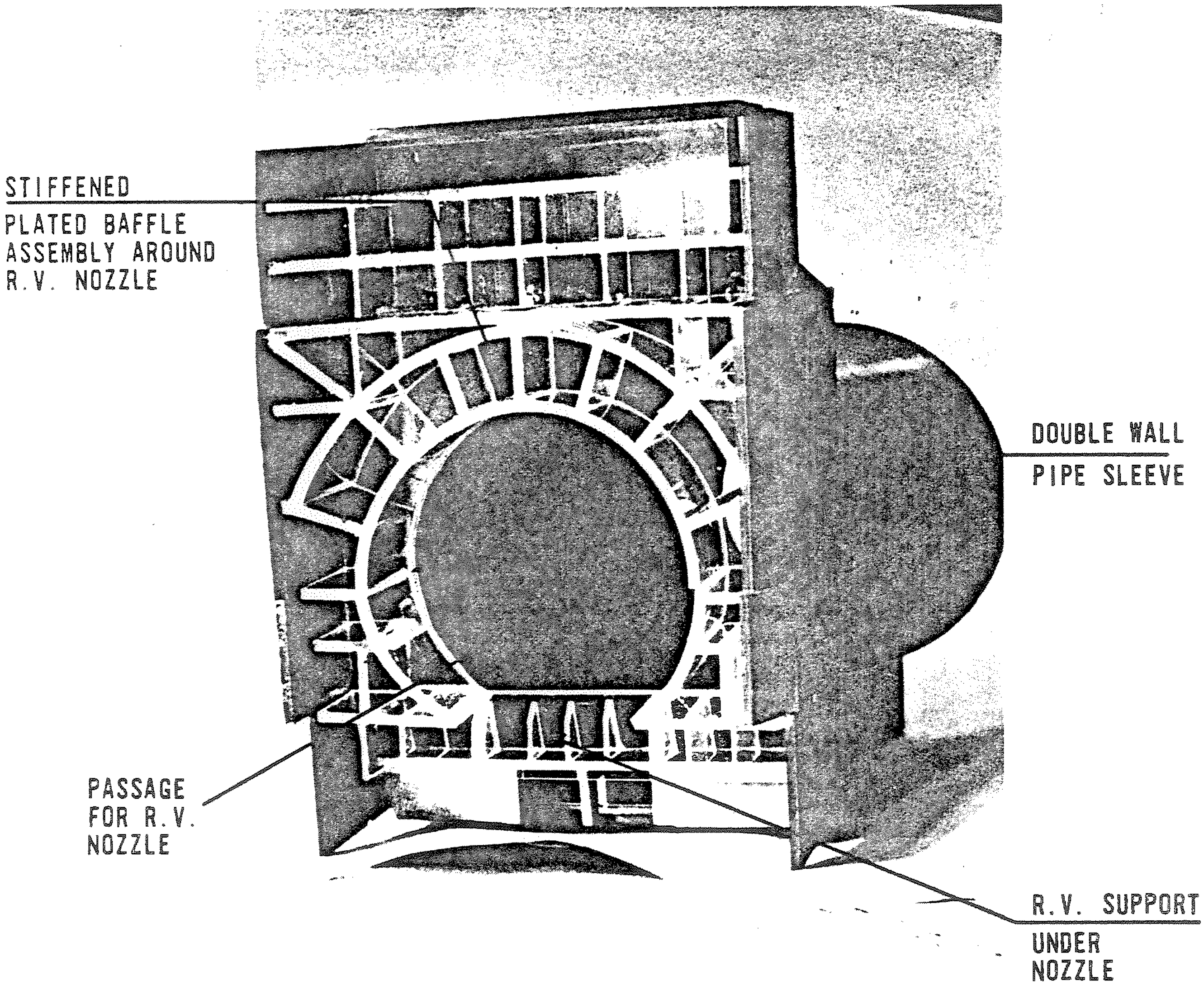
Figure 3.8-45



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Section Through Baffle
and Sleeve

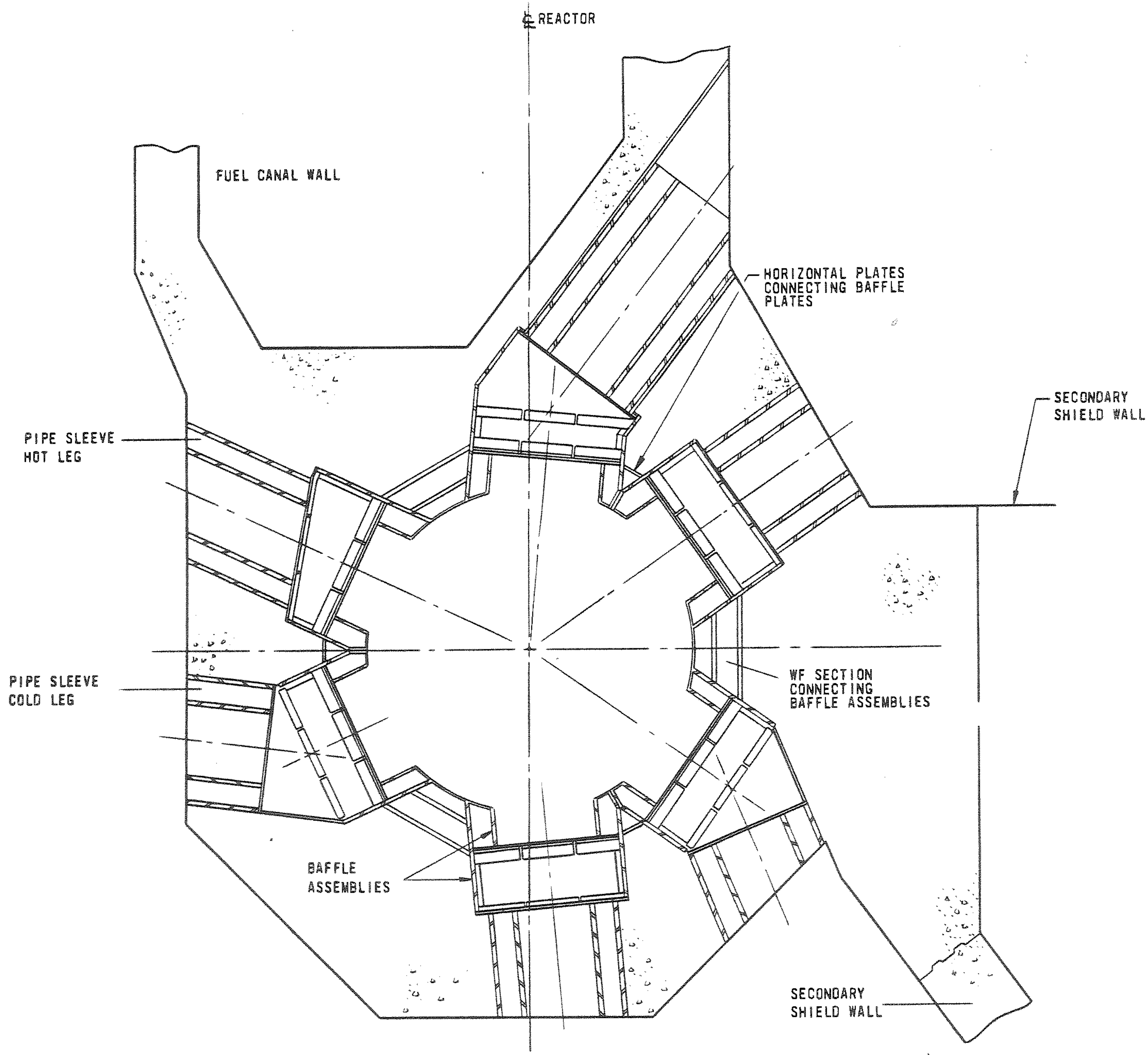
Figure 3.8-46



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Inside View of Baffle Assembly Around Reactor Vessel Nozzle

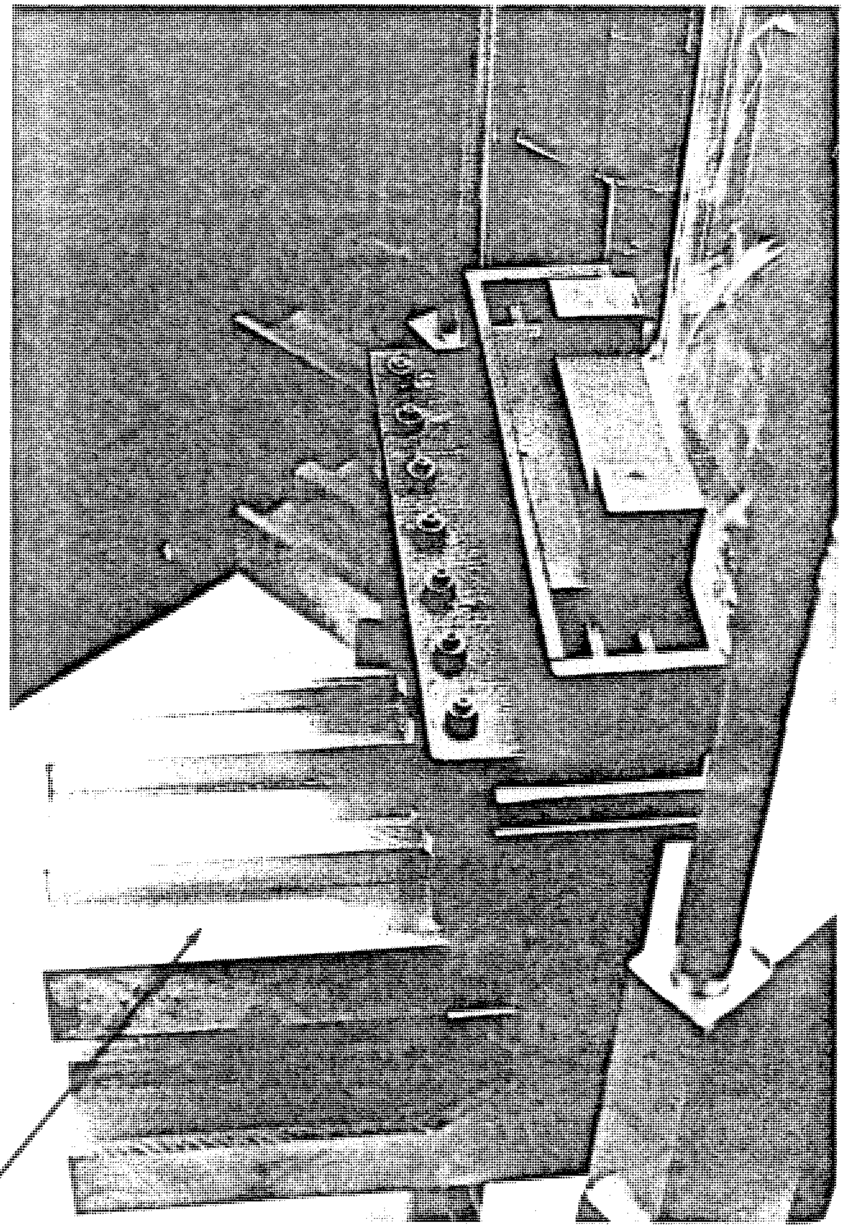
Figure 3.8-47



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Plan View of
Baffle Assemblies

Figure 3.8-48

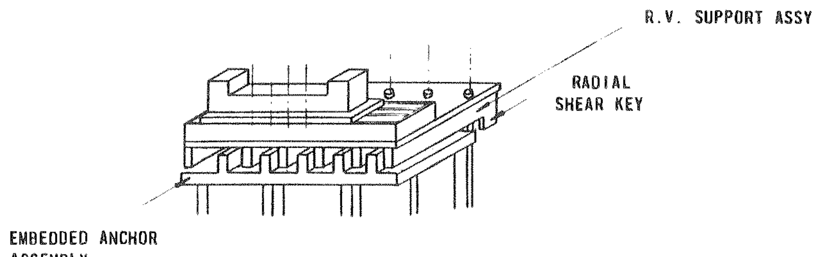


WIDE
FLANGE
ASSEMBLY

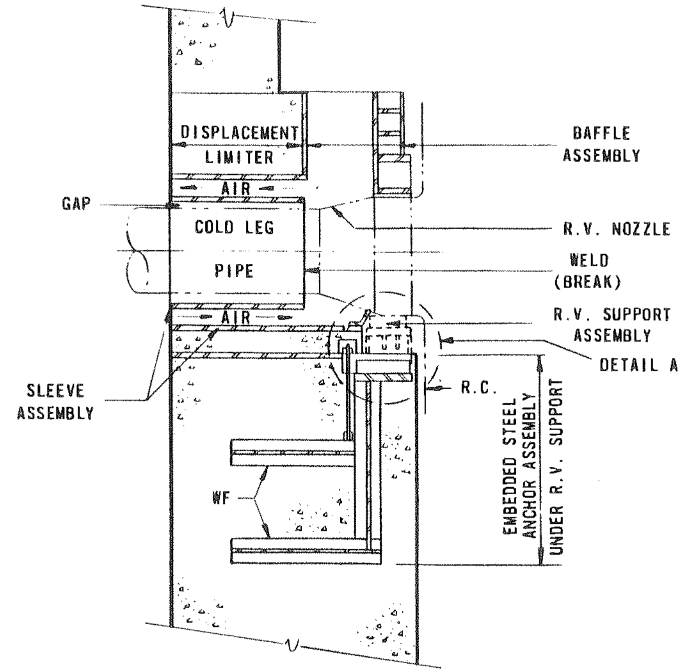
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Support and Wide Flange Anchor Assembly

Figure 3.8-49



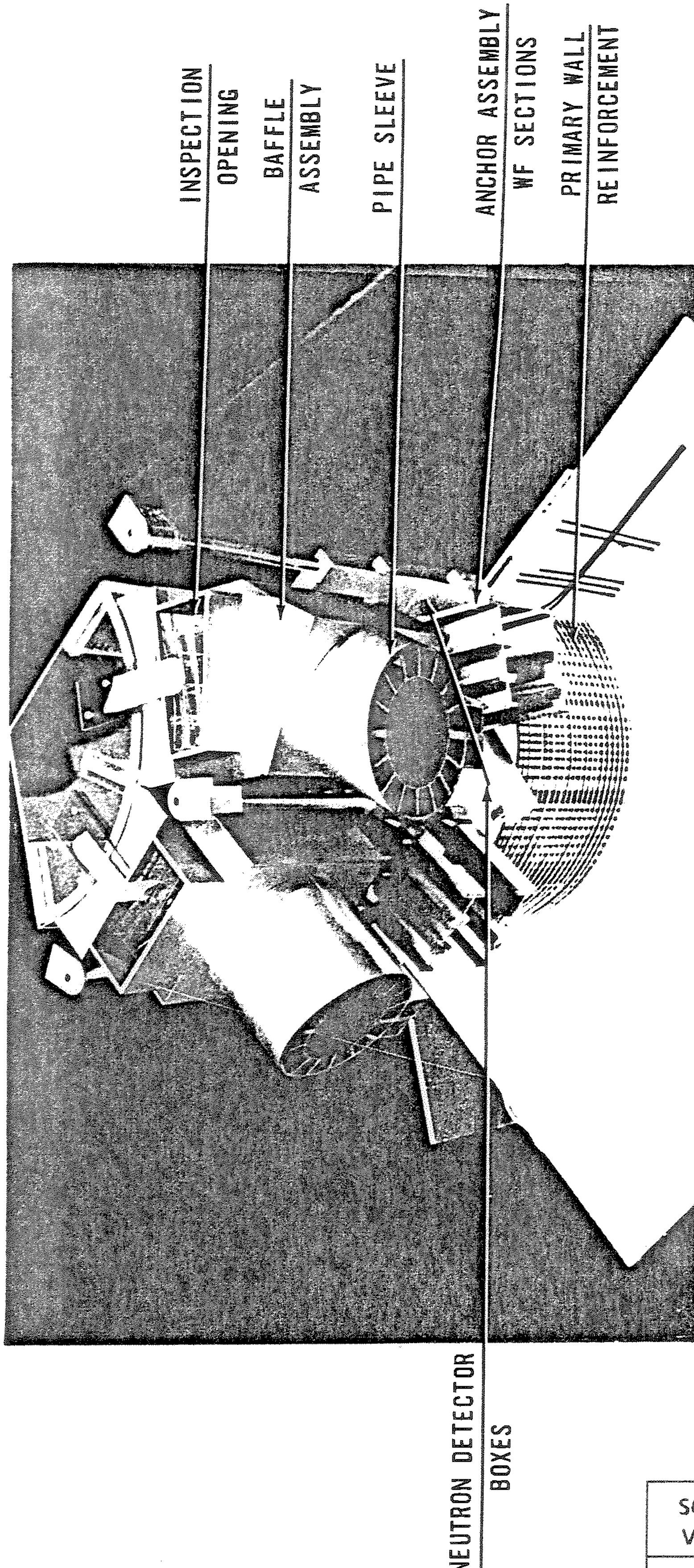
DETAIL A



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Pressure Vessel
Anchorage Assembly

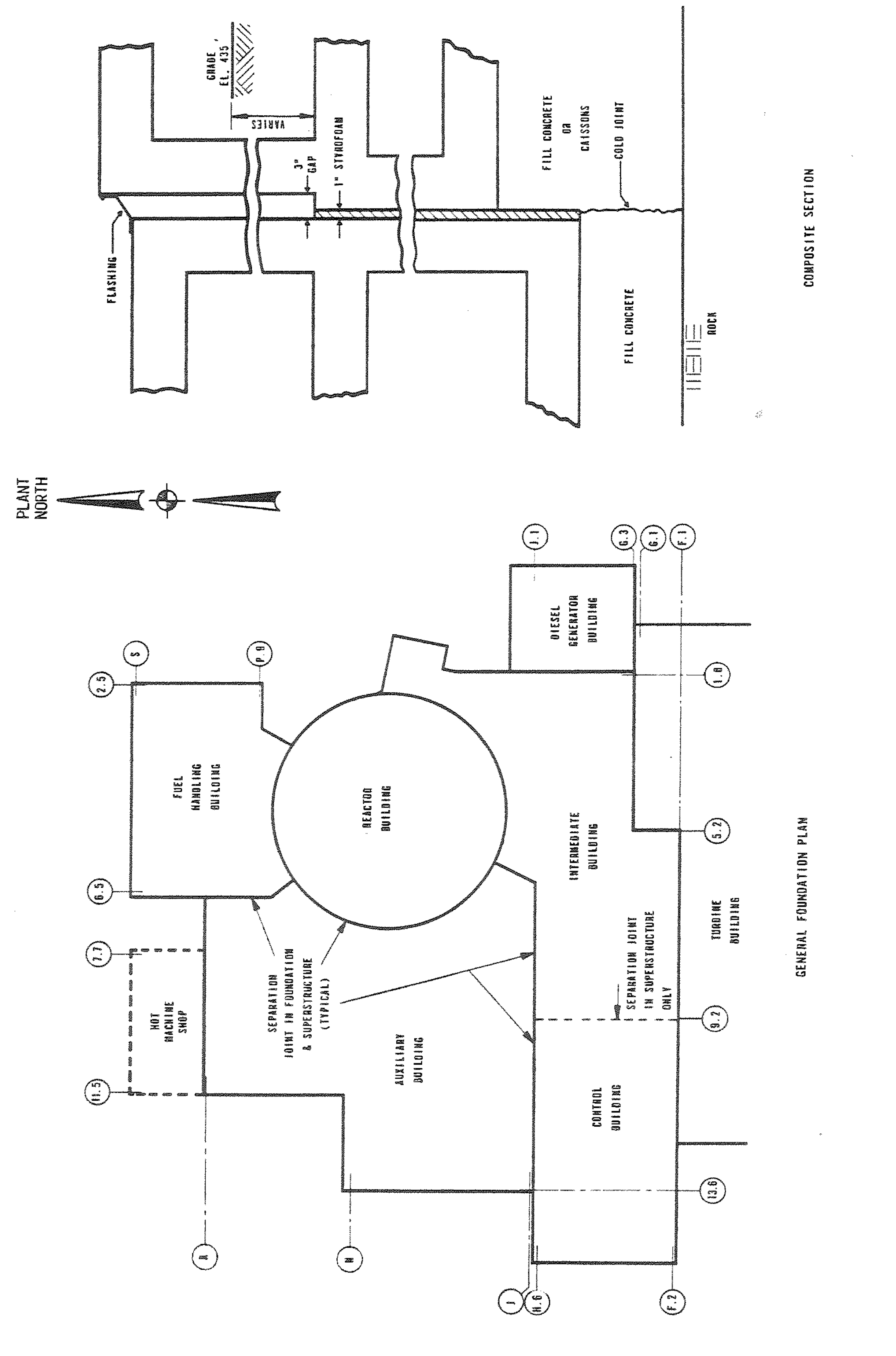
Figure 3.8-50



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Outside View of
Primary Wall

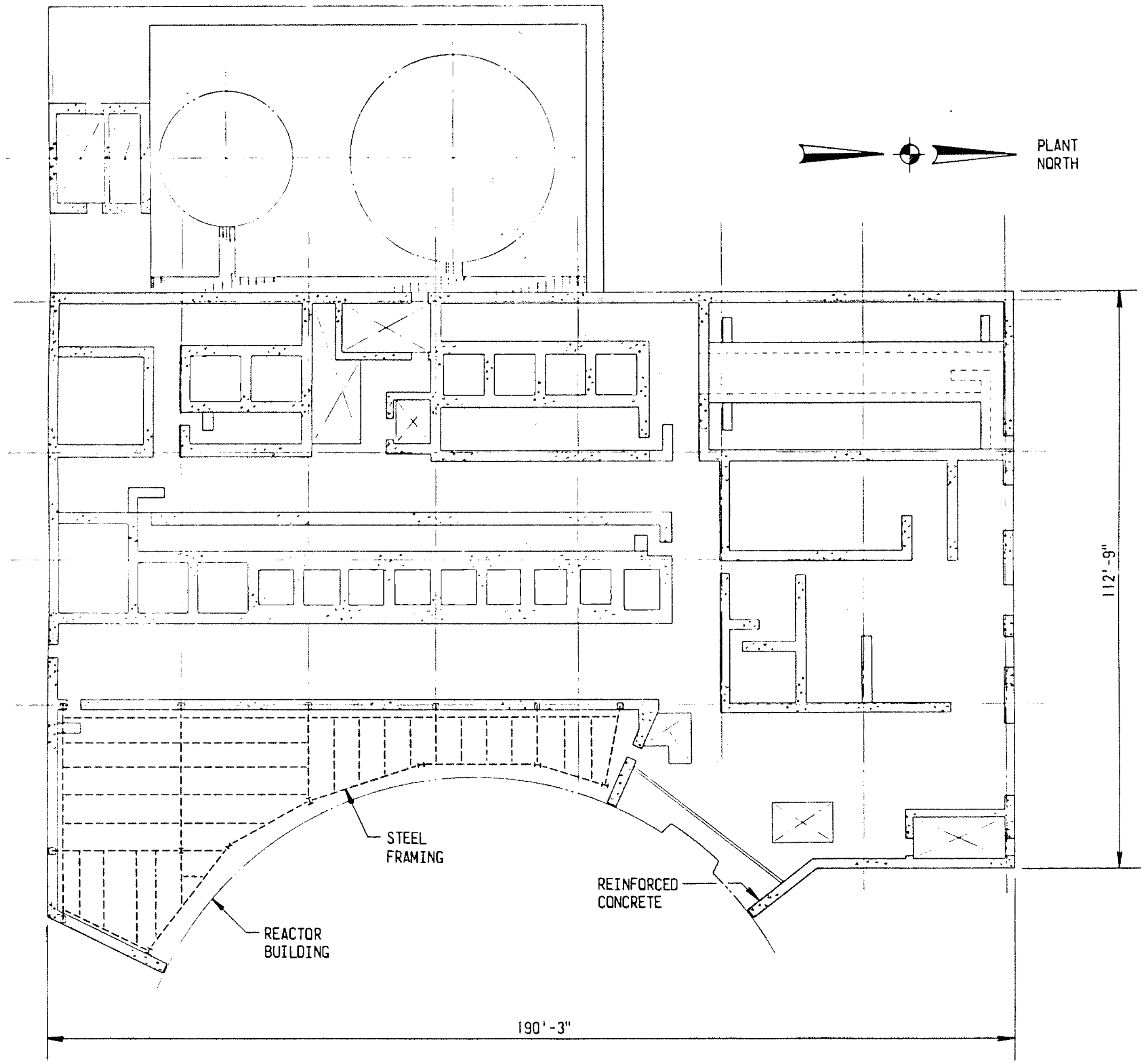
Figure 3.8-51



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

General Foundation Plan
and Composite Section

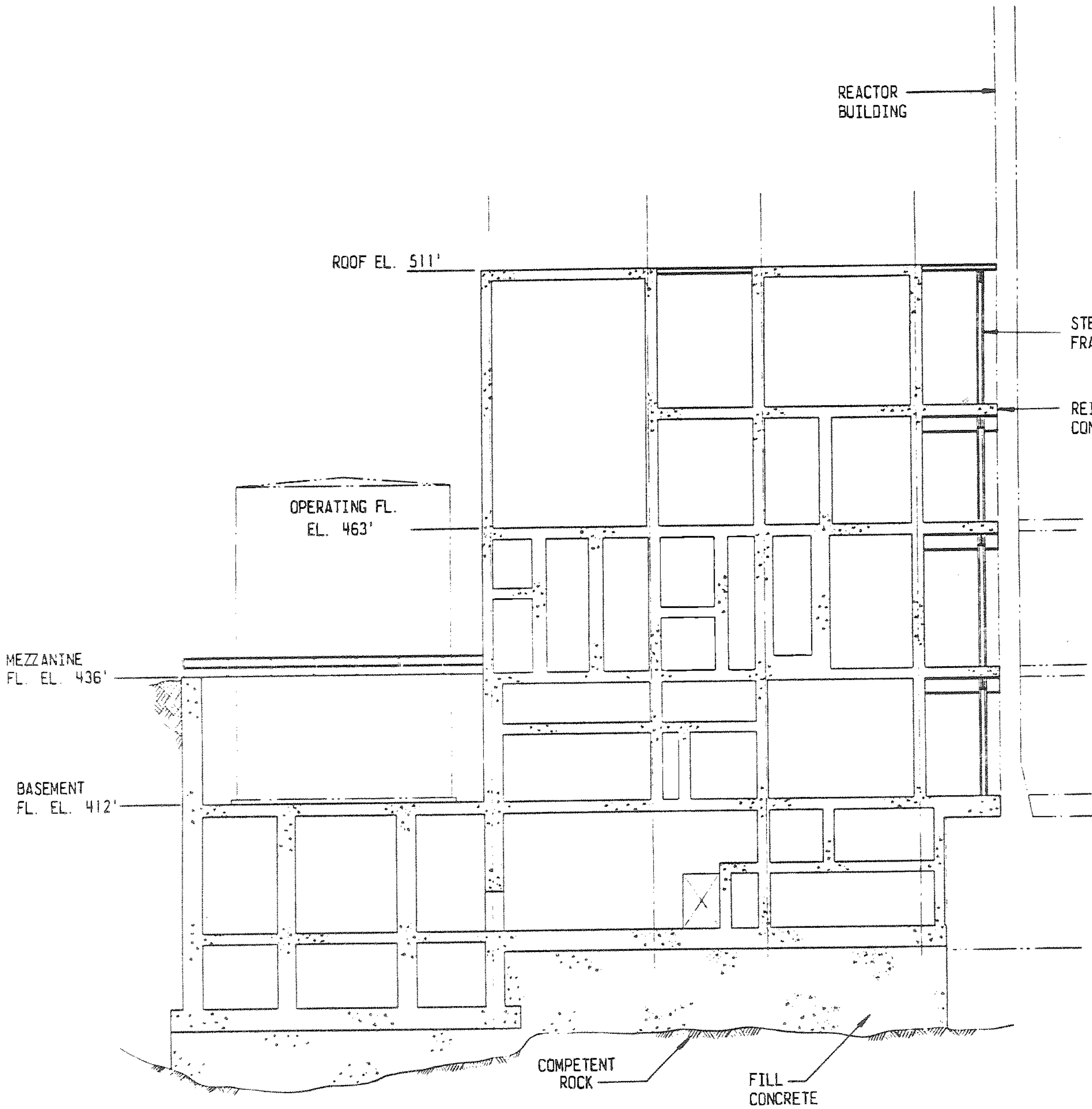
Figure 3.8-52



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Plan Auxiliary Building Mezzanine
Floor Elevation 436' - 0"

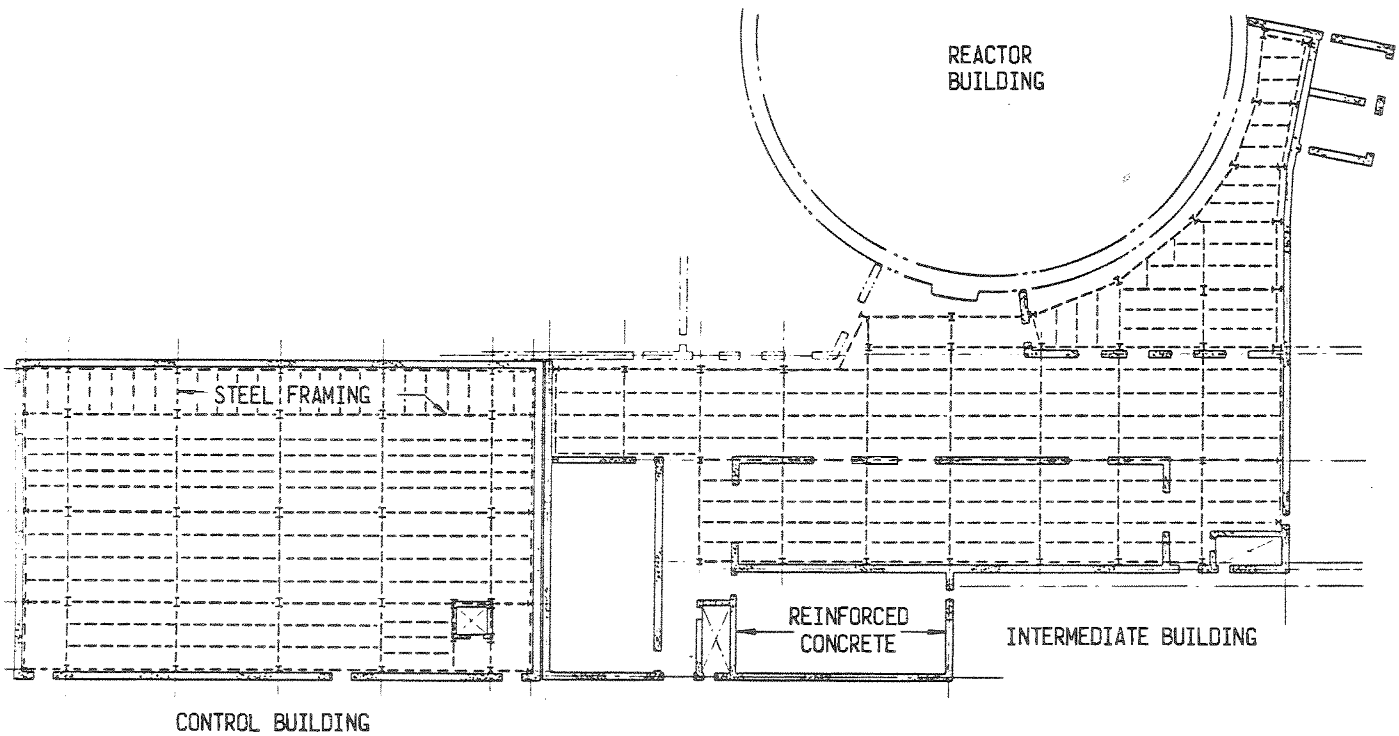
Figure 3.8-53



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Section Looking North Through Auxiliary Building

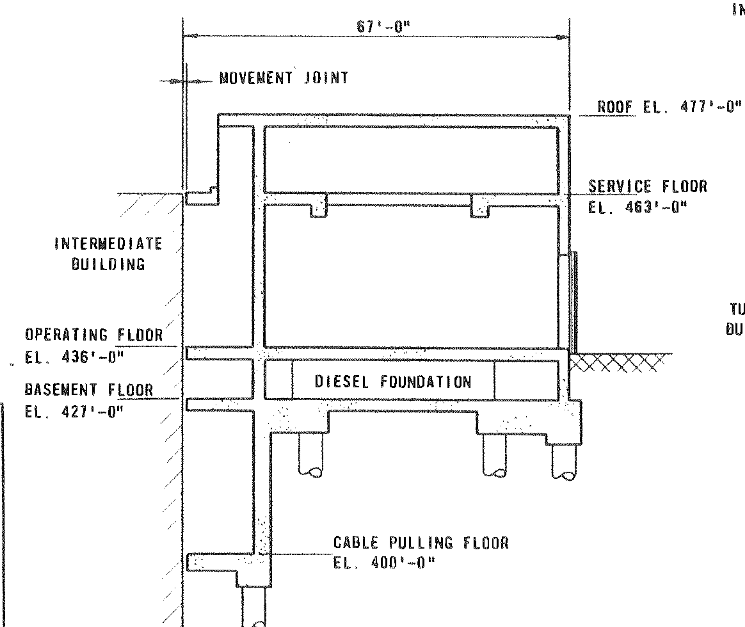
Figure 3.8-54



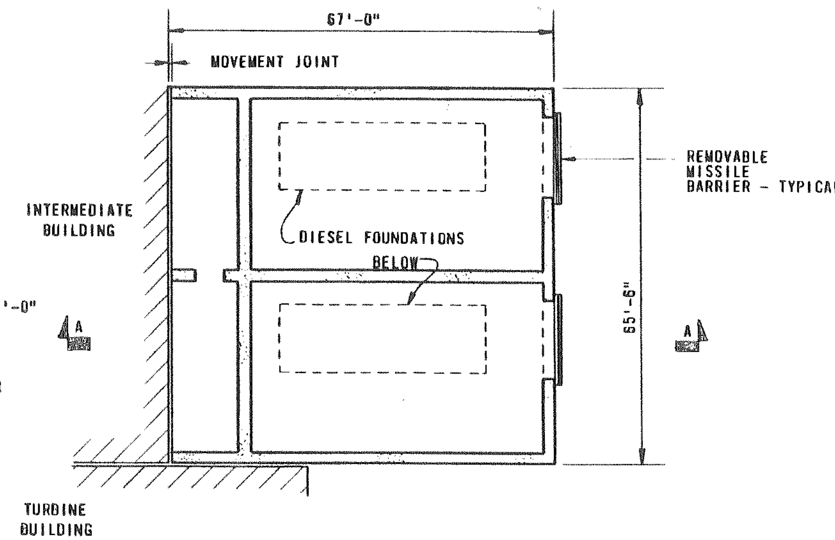
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Plan Control Building and Intermediate Building Mezzanine Floor Elevation 436' - 0"

Figure 3.8-55



SECTION A-A



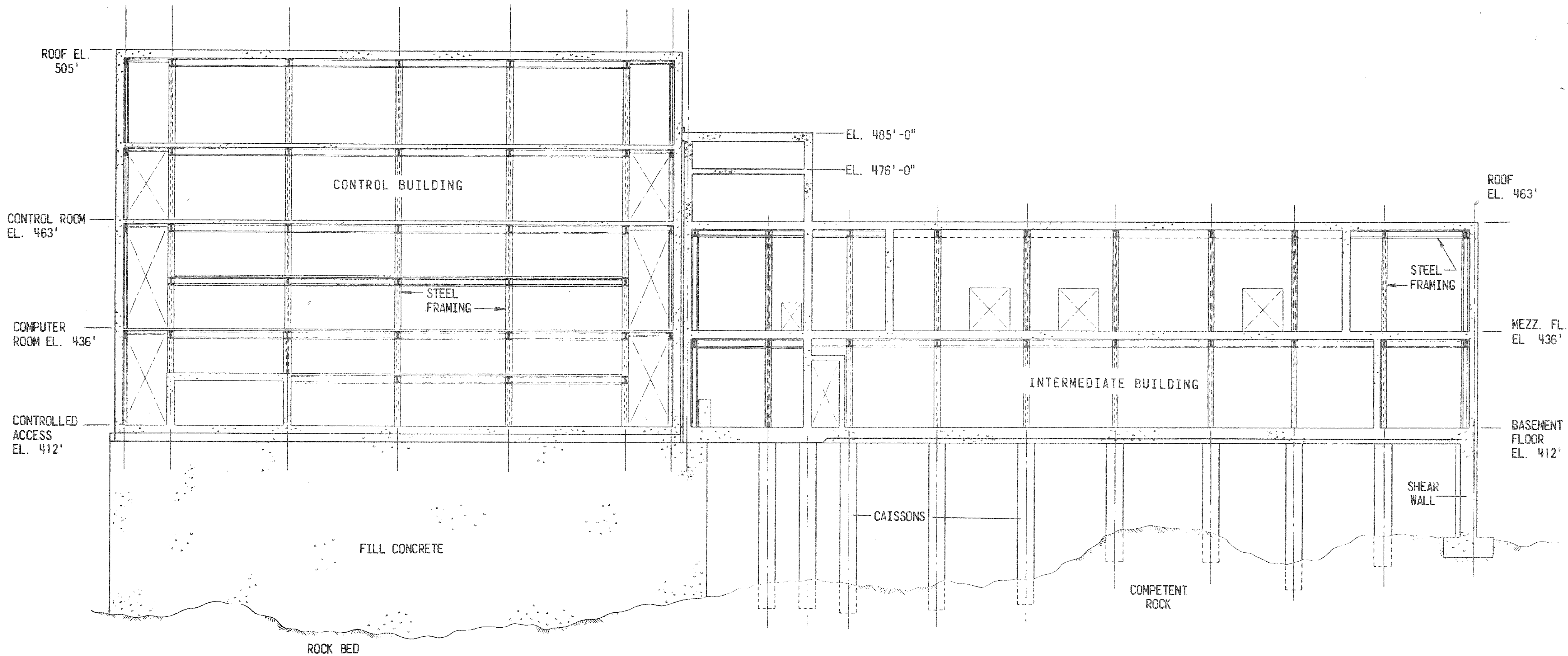
PLAN - OPERATING FLOOR

EL. 436'-0"

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Diesel Generator Building Plan and Foundation

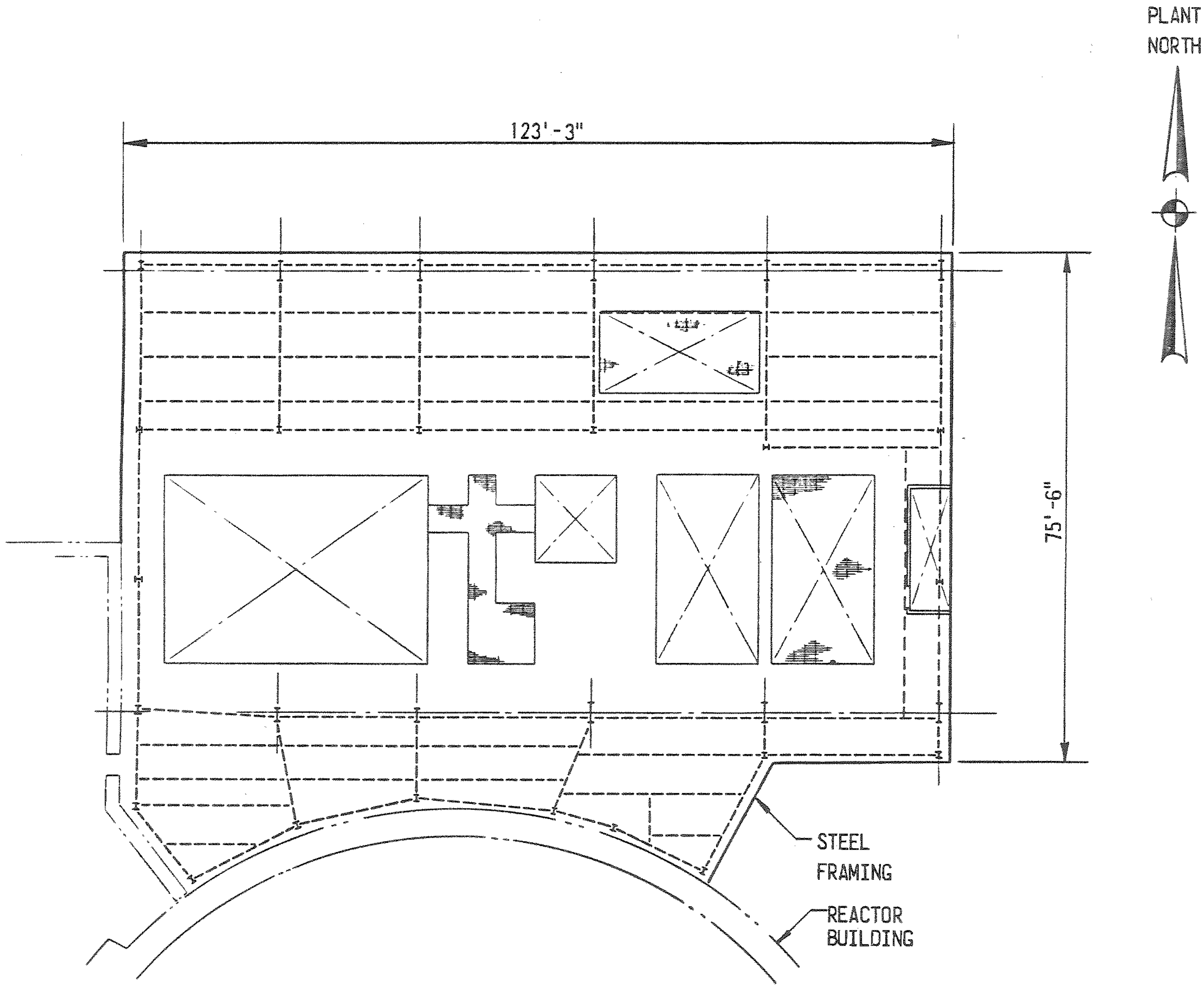
Figure 3.8-56



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Section Through Intermediate Building and Control Building Looking North

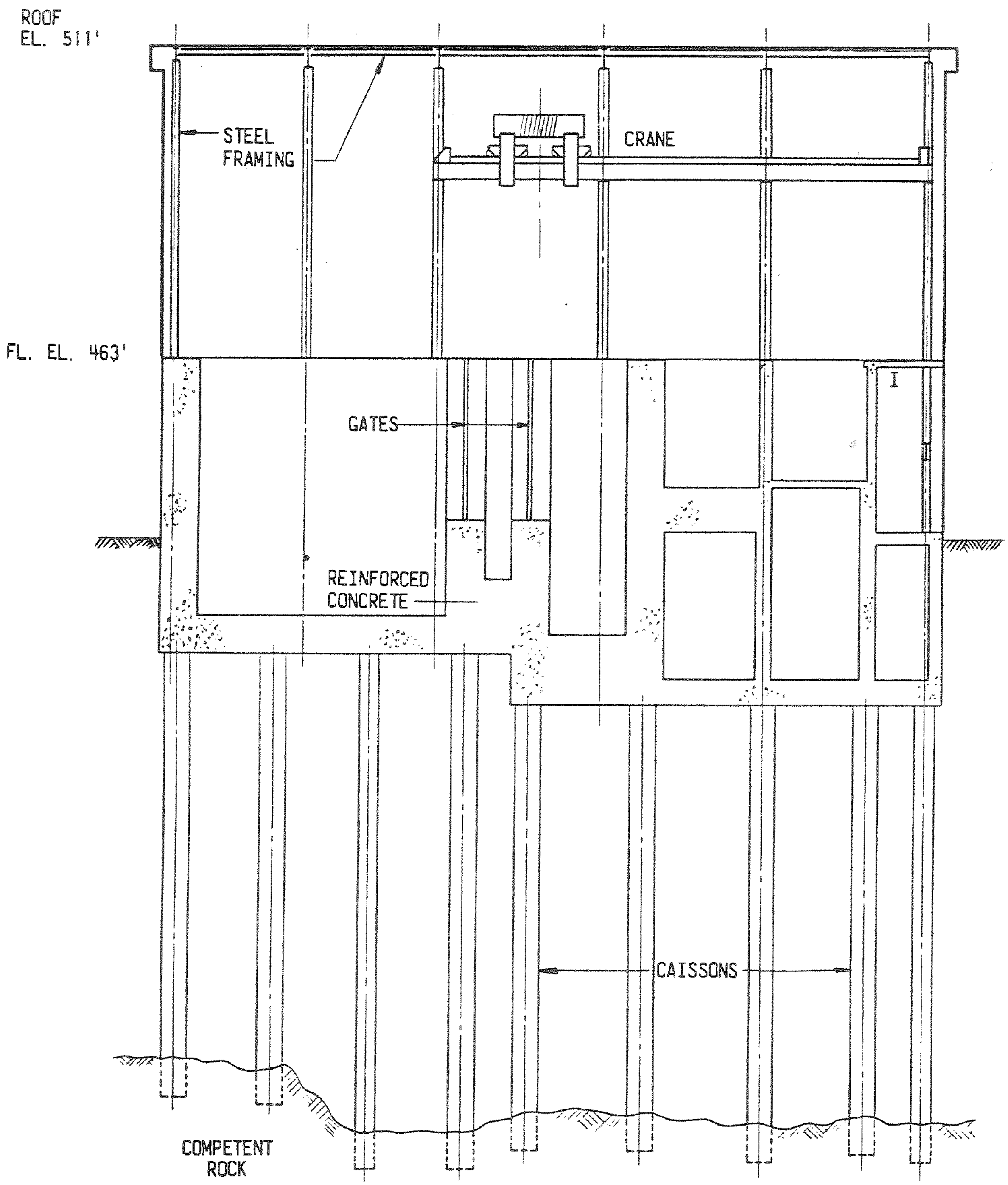
Figure 3.8-57



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Plan Fuel Handling Building
Operating Floor

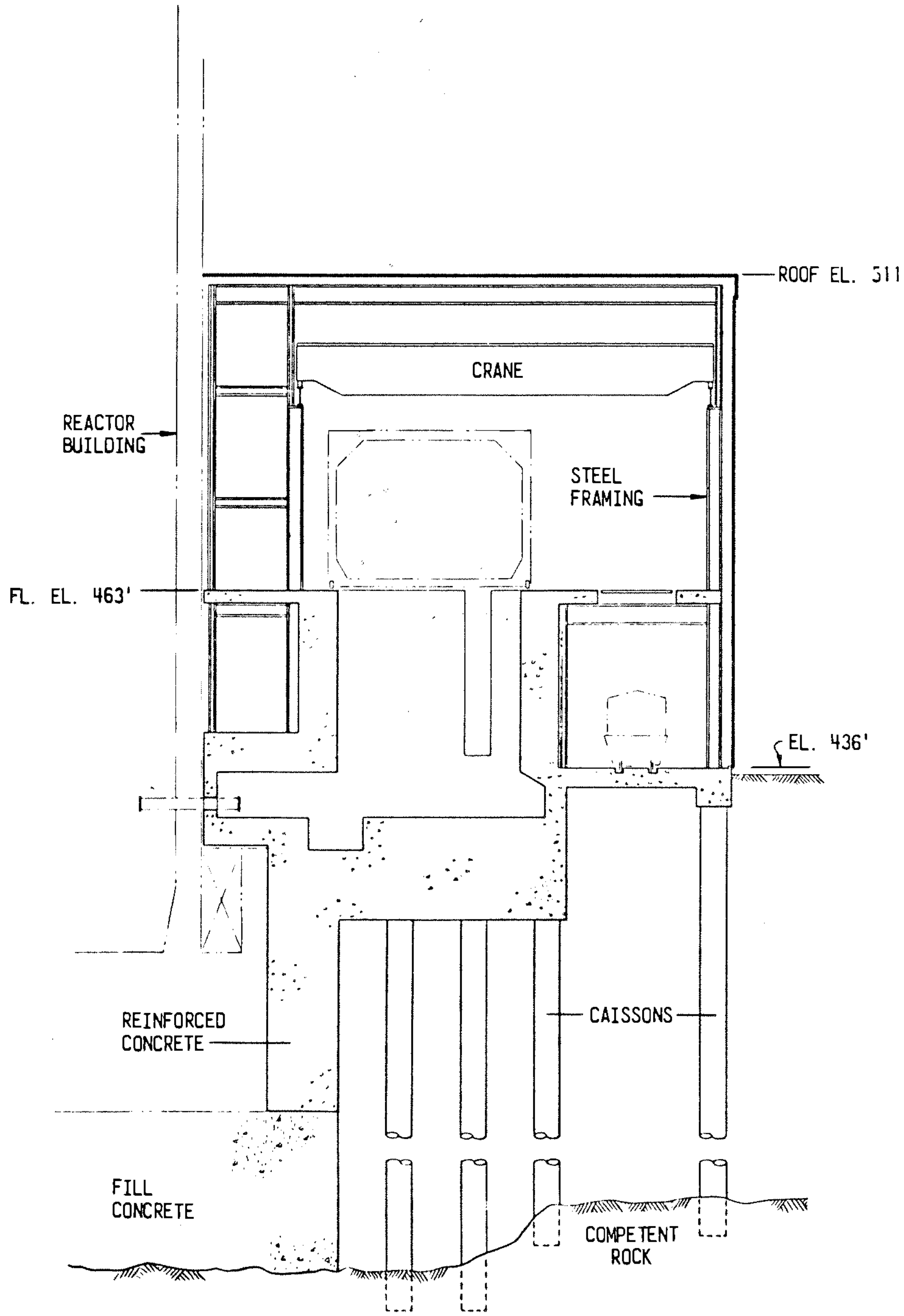
Figure 3.8-58



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Section Through Fuel Handling Building
Looking North

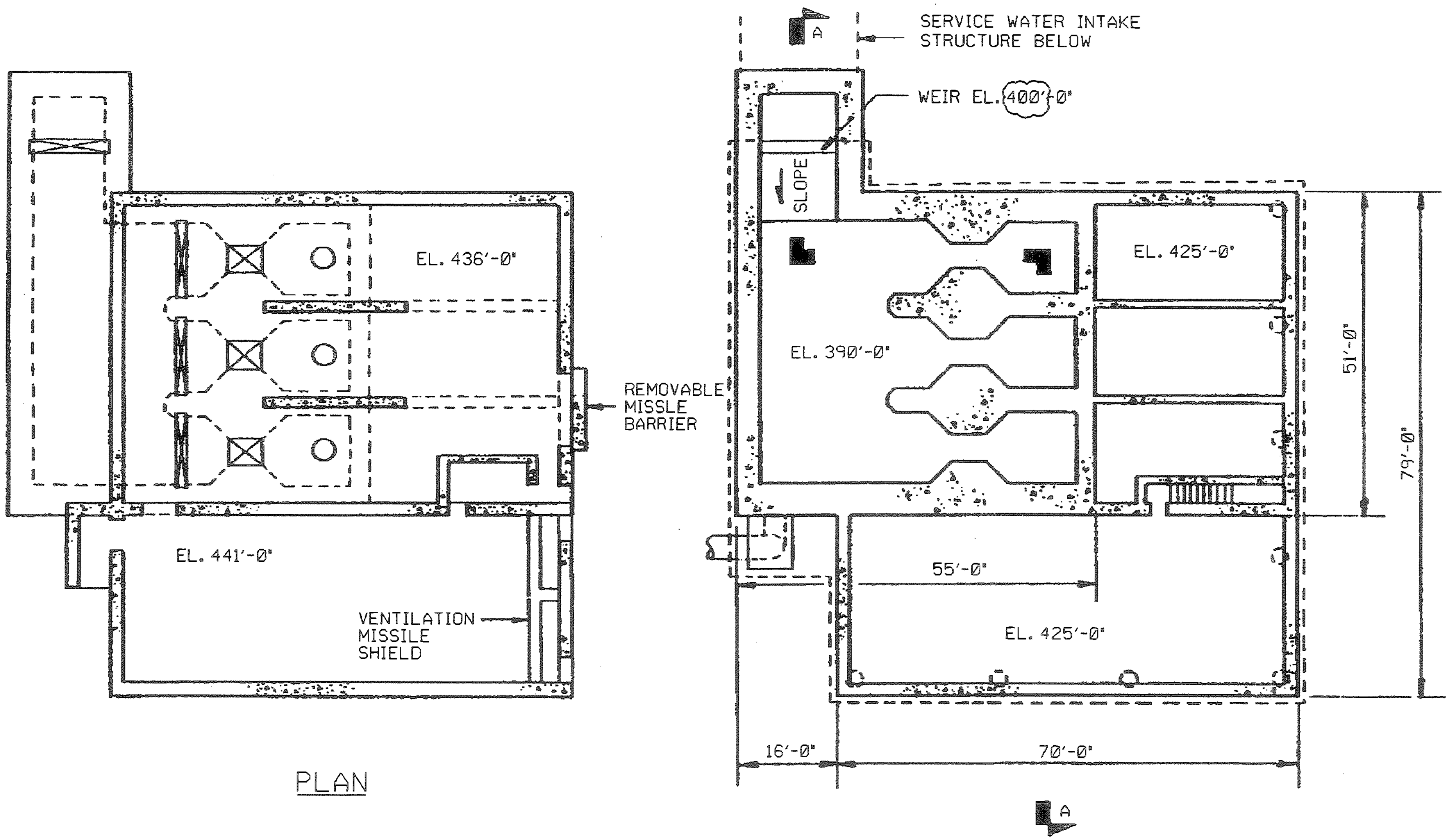
Figure 3.8-59



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Section Through Fuel Handling Building
Looking West

Figure 3.8-60



SERVICE WATER PUMP HOUSE

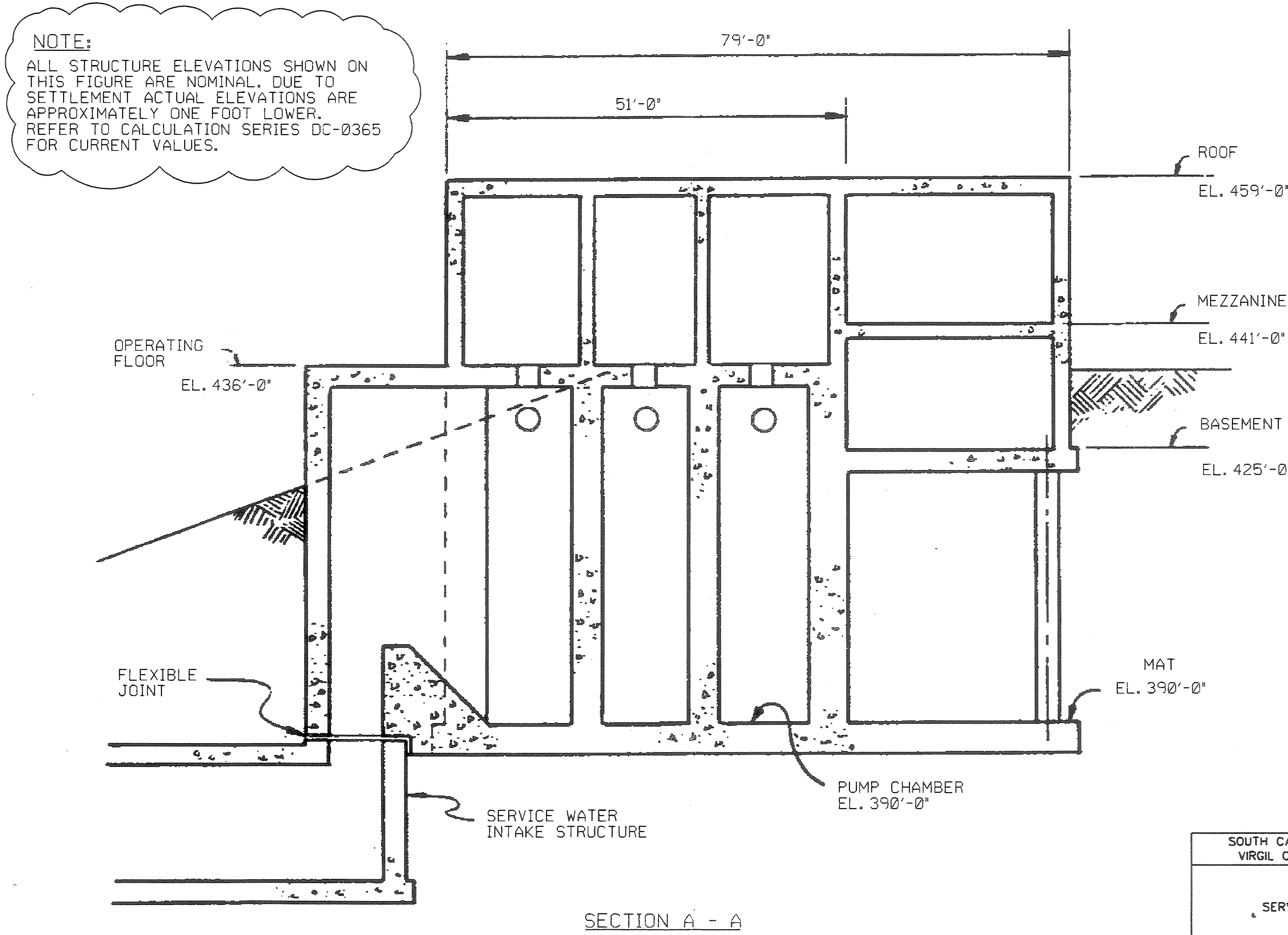
NOTE:

REFER TO FIGURE 3.8-62 FOR NOTE
ON SPECIFIED ELEVATIONS.

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

PLAN SERVICE WATER
PUMPHOUSE

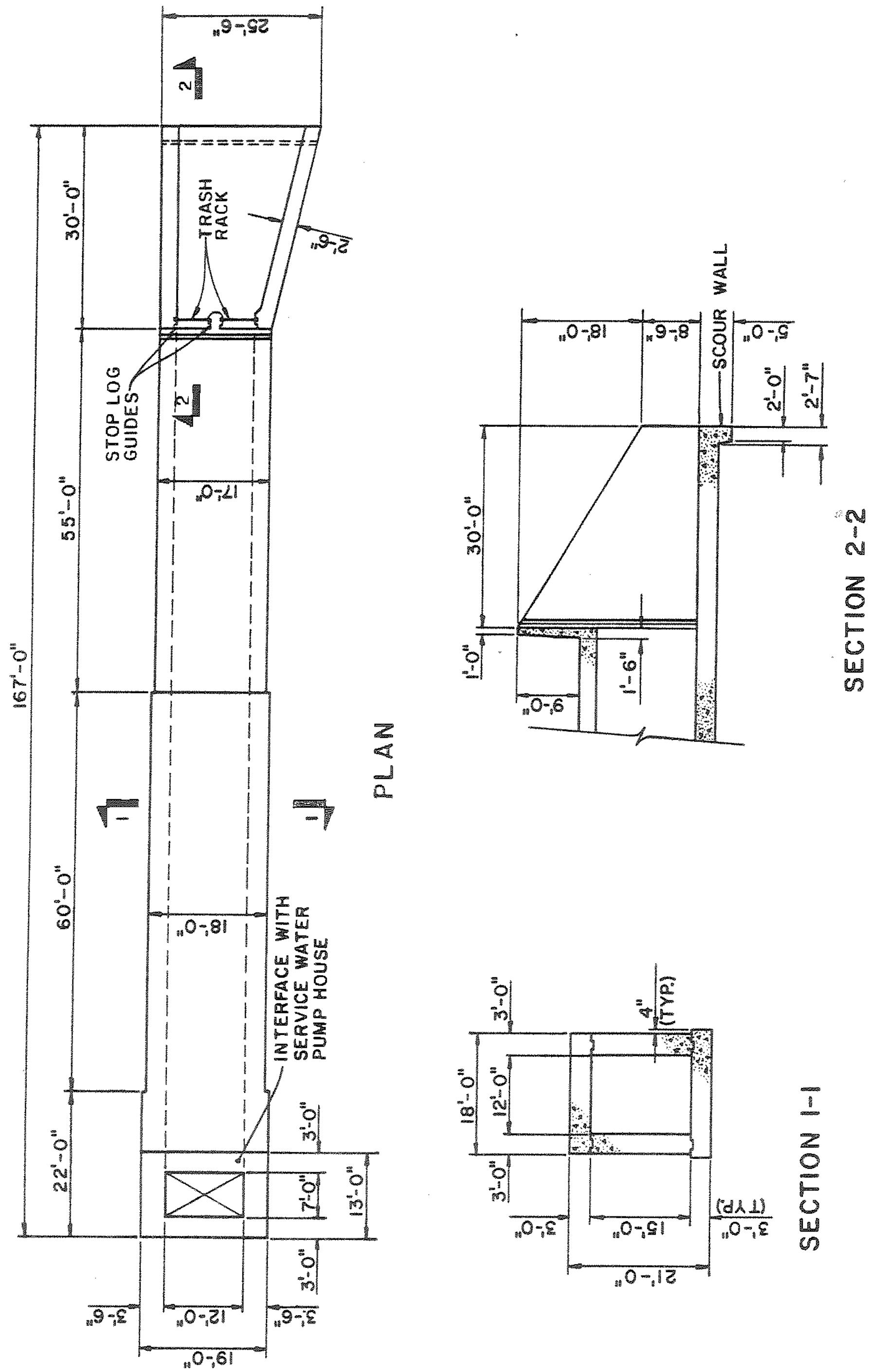
Figure 3.8-61



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

SERVICE WATER PUMPHOUSE
SECTION

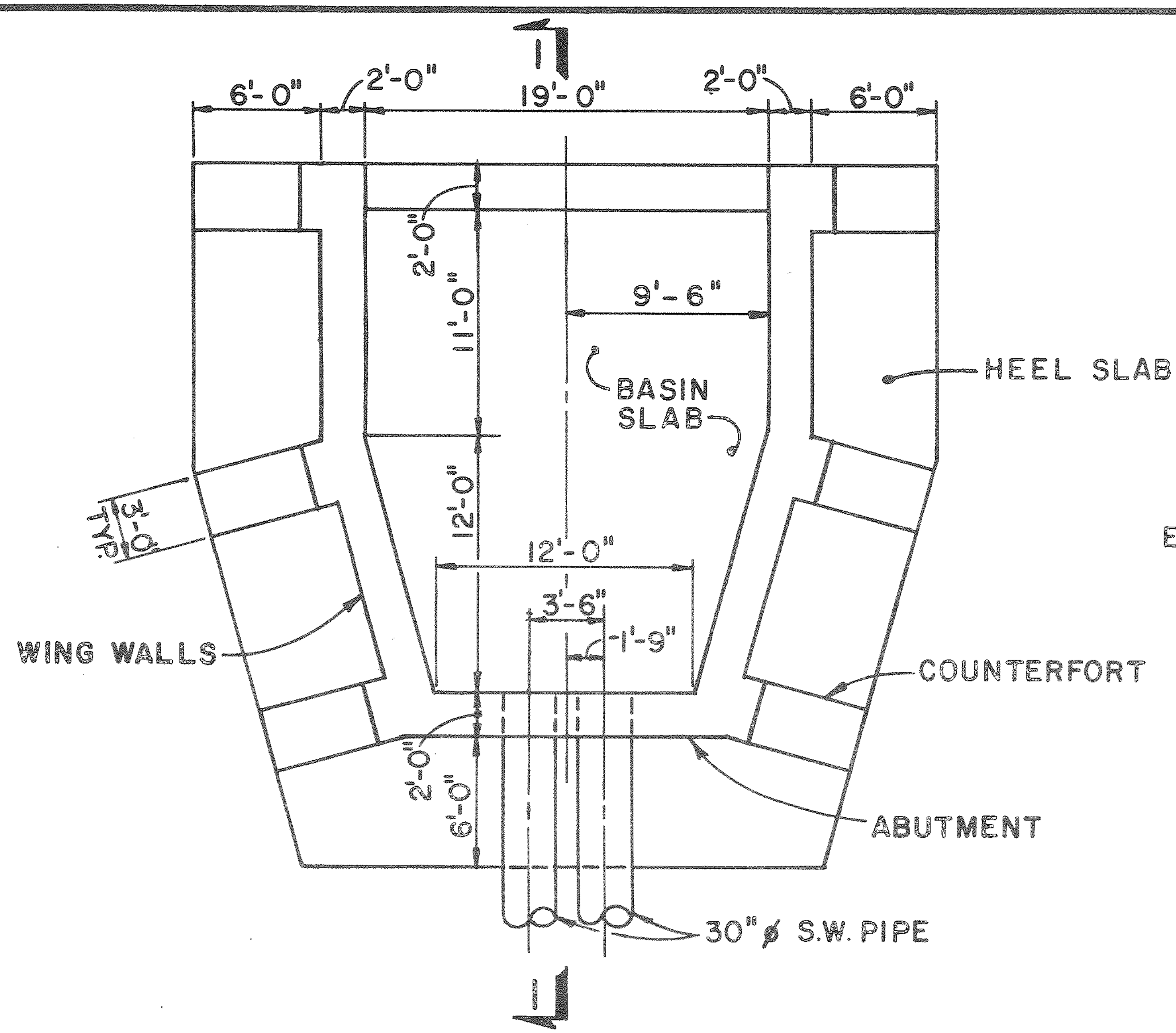
Figure 3.8-62



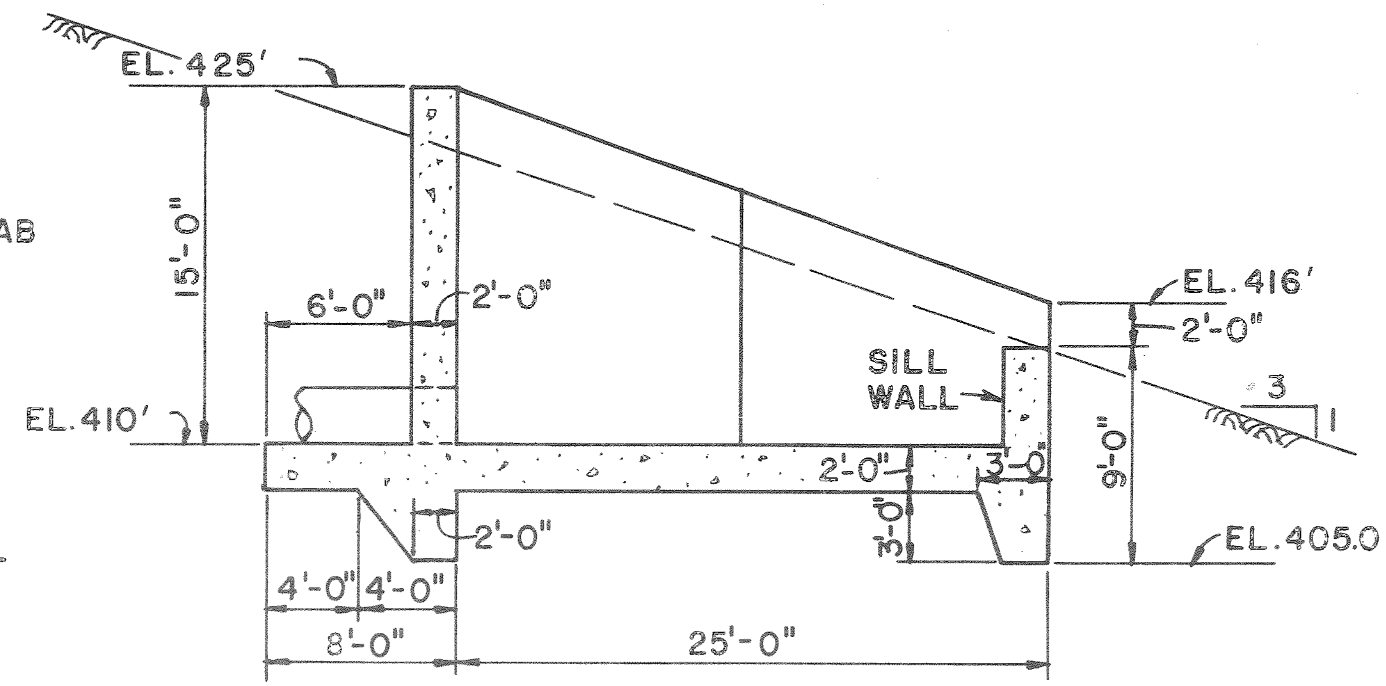
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Service Water Intake Structure

Figure 3.8-63



PLAN
SERVICE WATER
DISCHARGE STRUCTURE

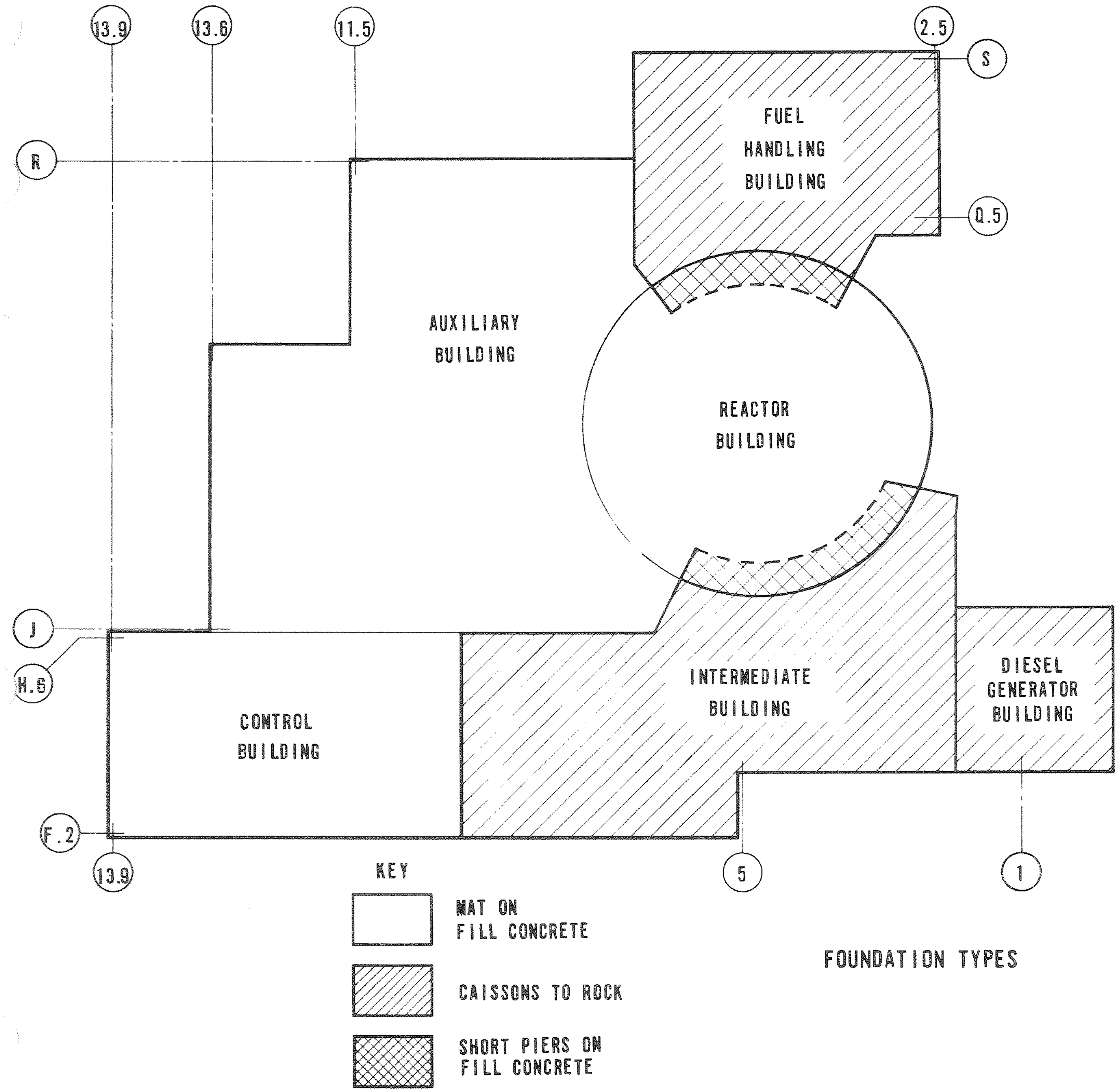


SECTION 1—1

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Service Water Discharge Structure

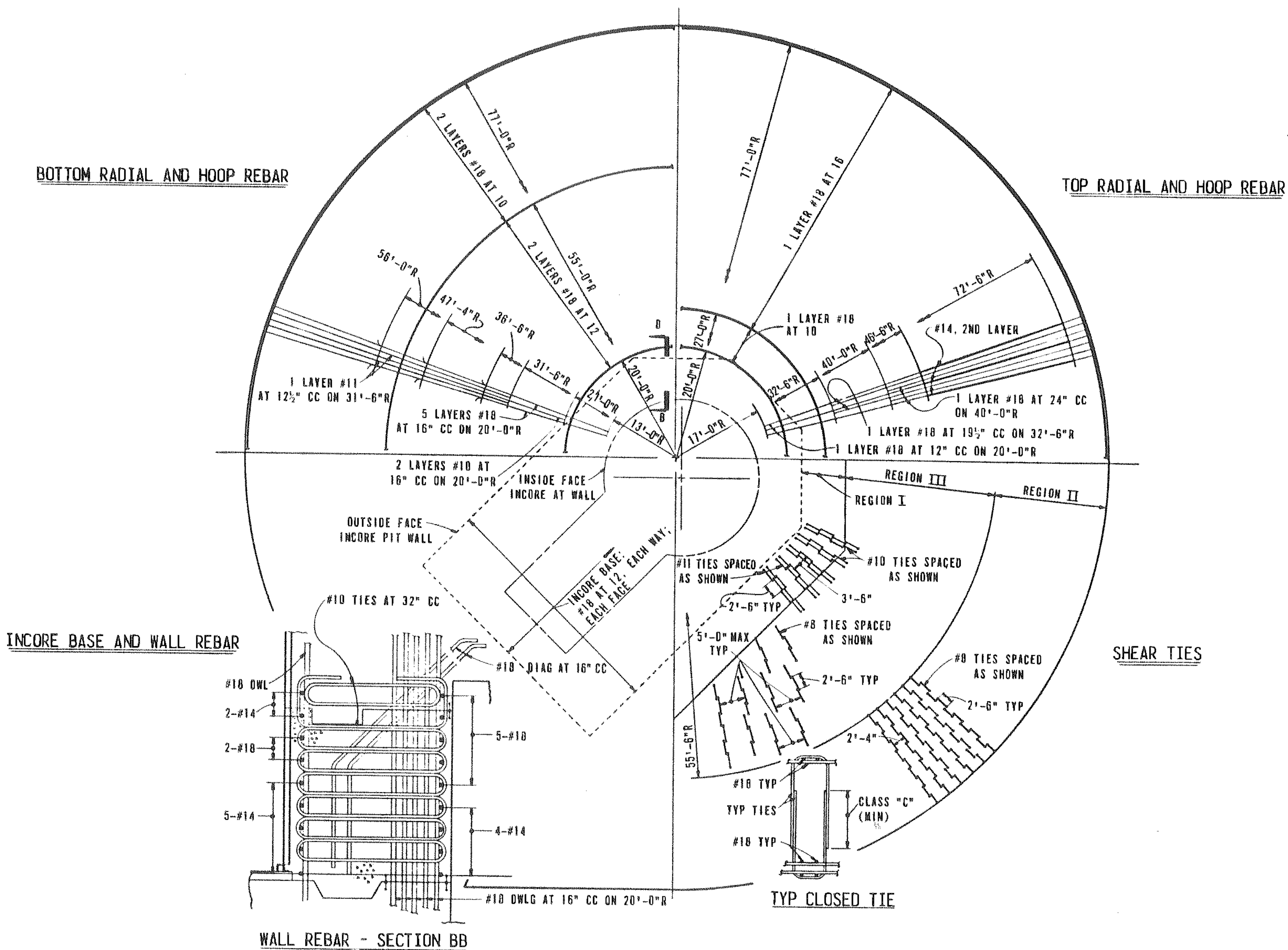
Figure 3.8-64



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Foundation Types

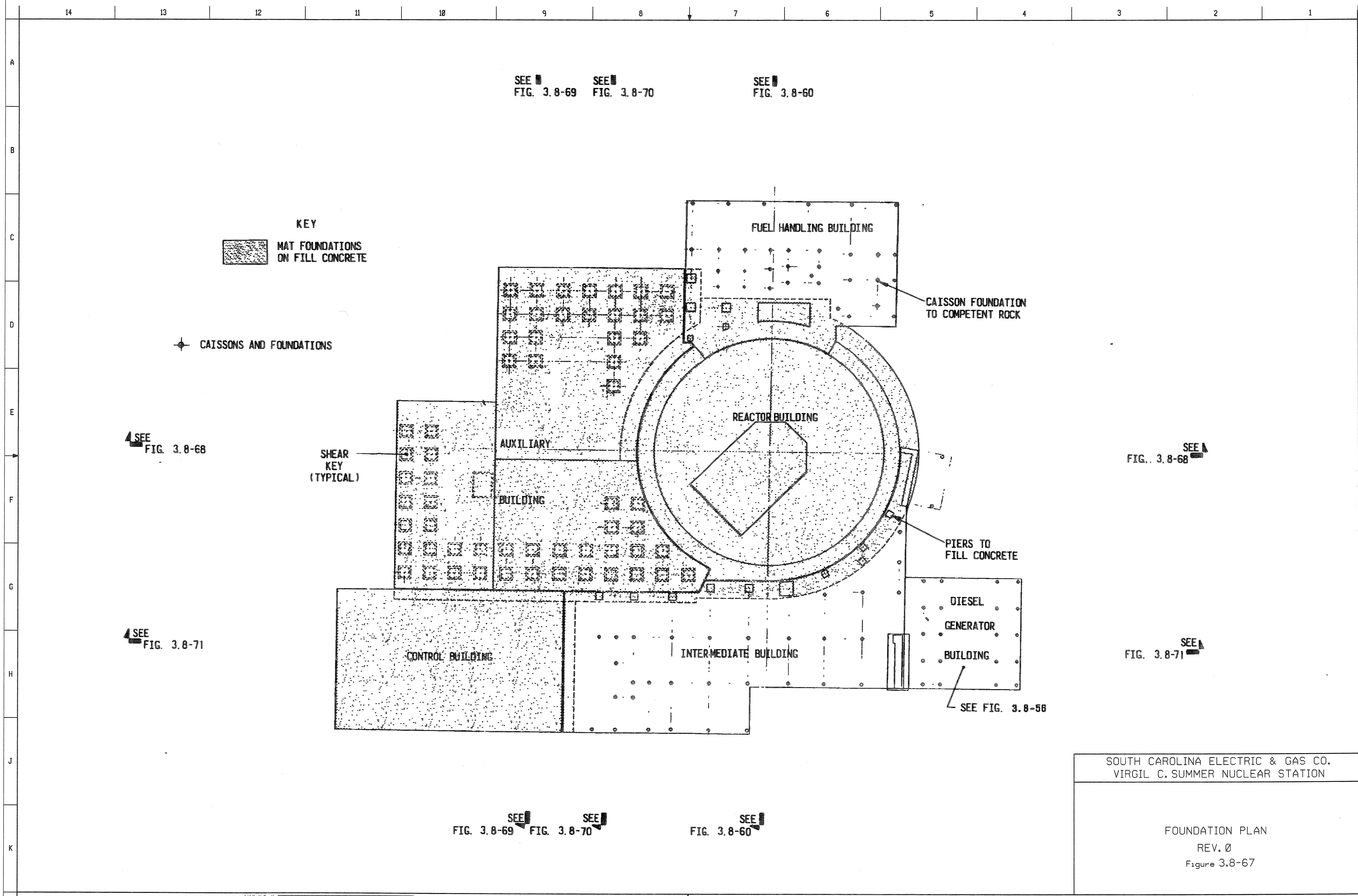
Figure 3.8-65



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGL C. SUMMER NUCLEAR STATION

Reactor Building Structural Mat Reinforcement

Figure 3.8-66

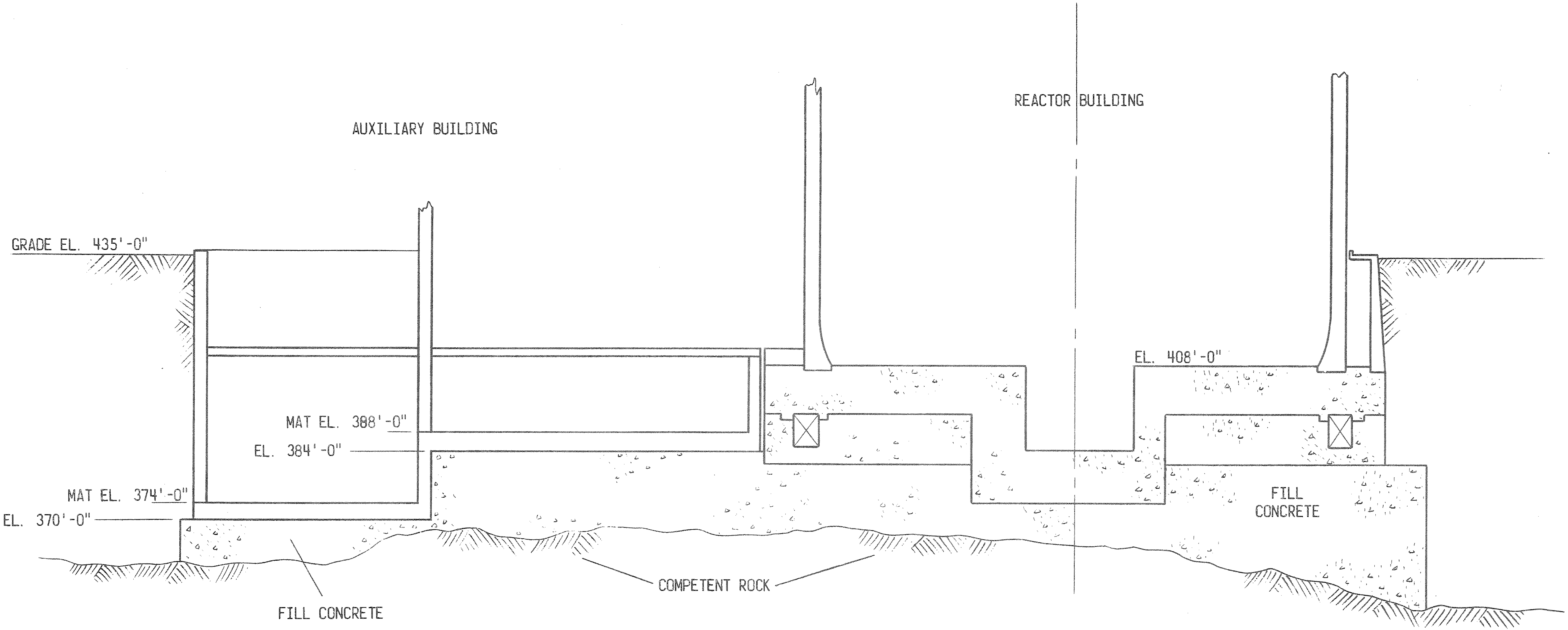


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

FOUNDATION PLAN

REV. 0

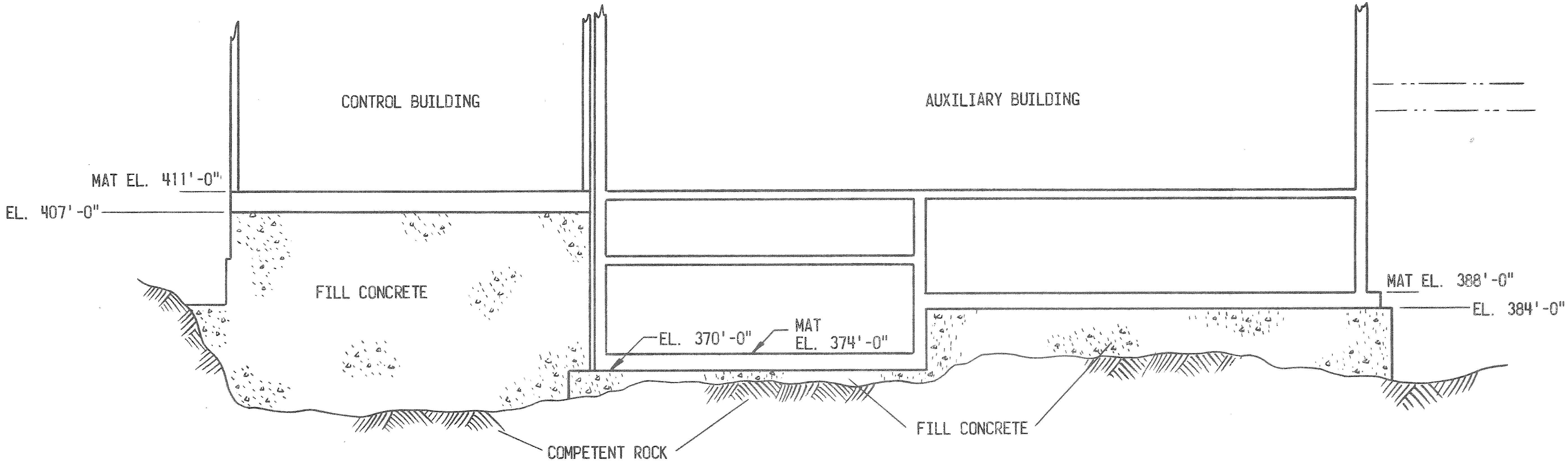
Figure 3.8-67



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Foundation Section Through Reactor Building and Auxiliary Building Looking North

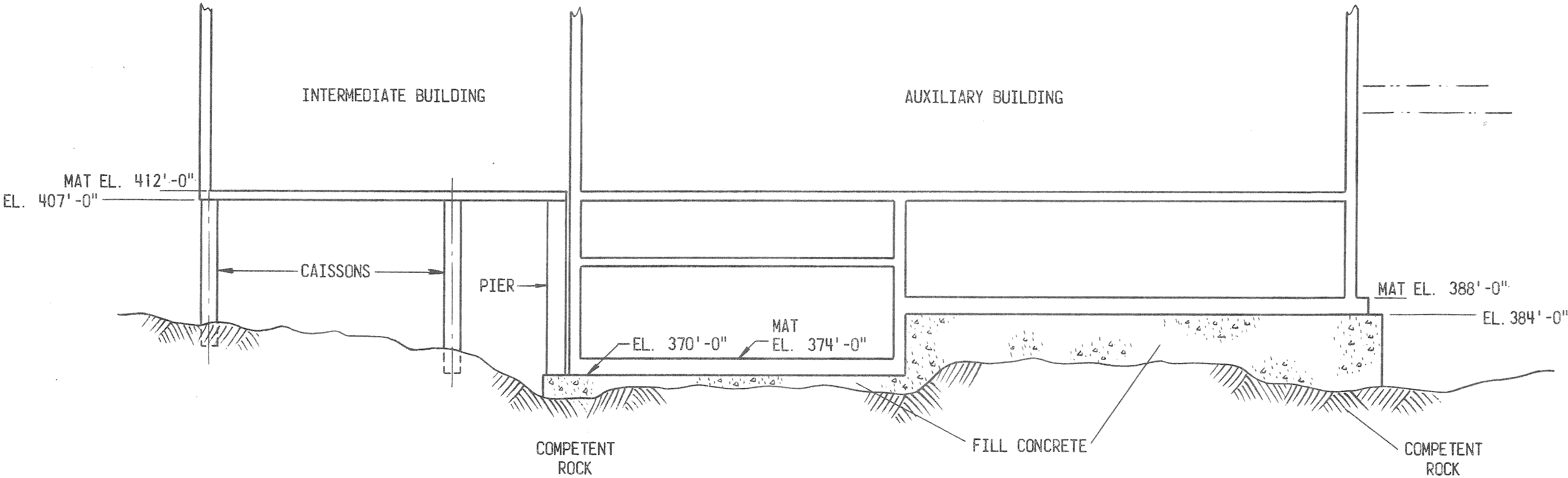
Figure 3.8-68



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Foundation Section Through Auxiliary Building and Control Building Looking West

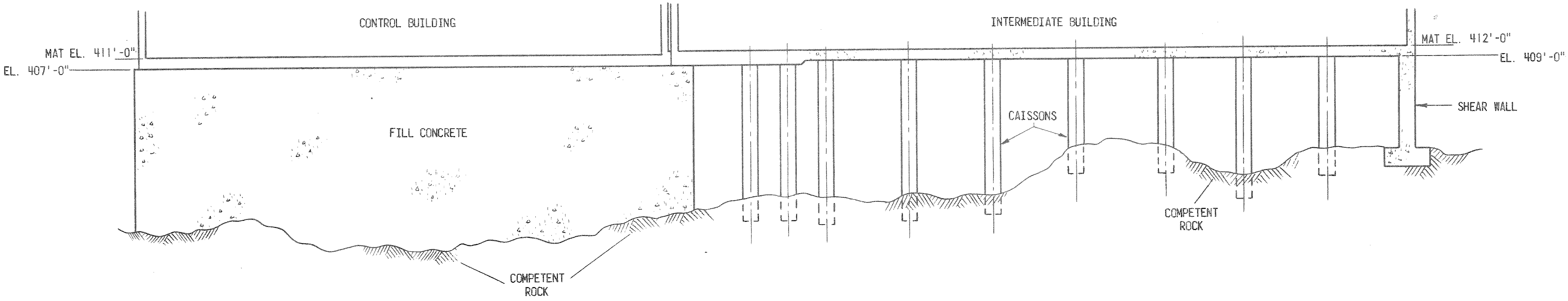
Figure 3.8-69



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Foundation Section Through Auxiliary Building and Intermediate Building Looking West

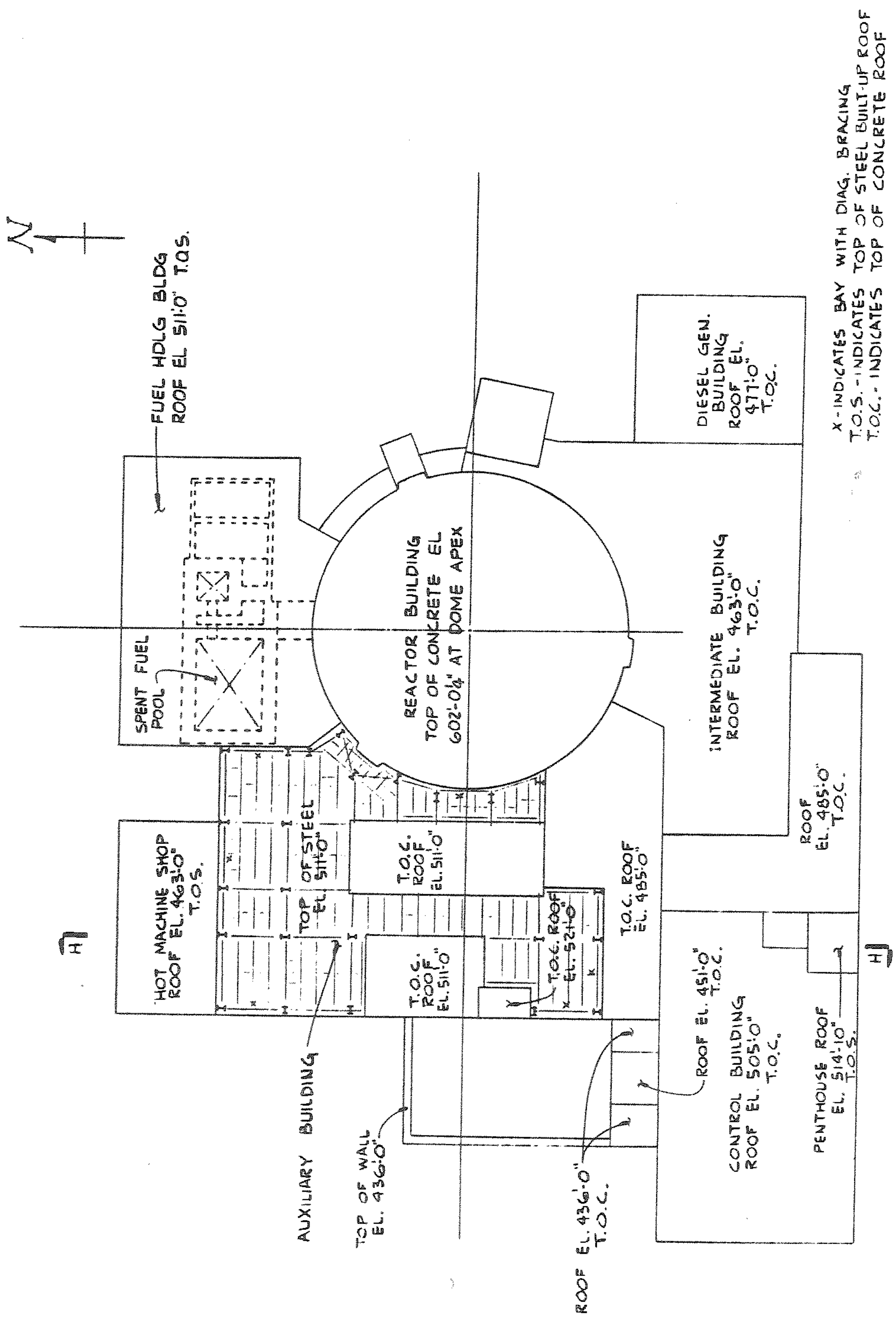
Figure 3.8-70



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Foundation Section Through Control Building and Intermediate Building Looking North

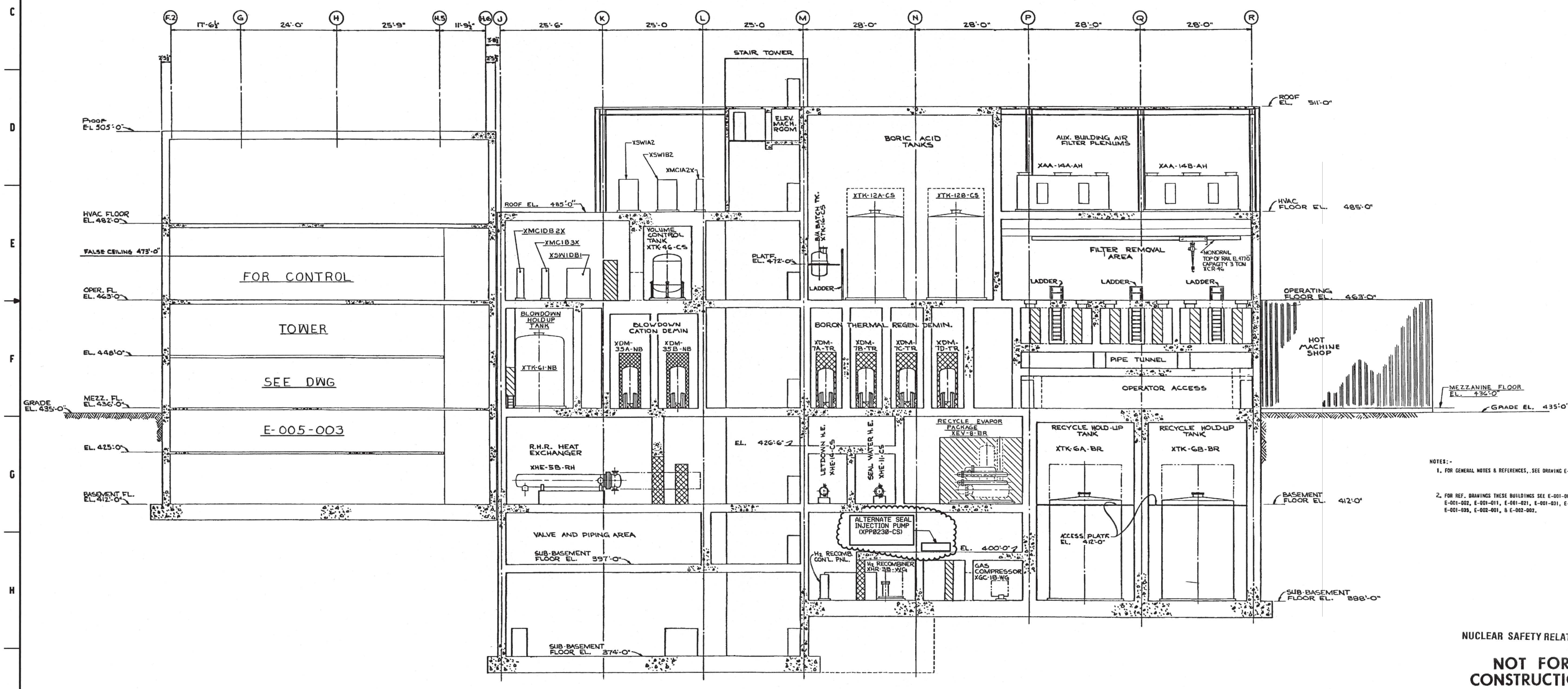
Figure 3.8-71



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Auxiliary Building Roof Steel Structure
With Surrounding Buildings

Figure 3.8-72



SECTION H-H
LOOKING WEST

NOTES:-

1. FOR GENERAL NOTES & REFERENCES, SEE DRAWING E-013-001.

2. FOR REF. DRAWINGS THESE BUILDINGS SEE E-001-001, E-001-002, E-001-011, E-001-021, E-001-031, E-001-034, E-001-035, E-002-001, & E-002-002.

NUCLEAR SAFETY RELATED

NOT FOR CONSTRUCTION

RN 11-027

PENDING/HOLD CHG
Use with reference to
ECR: 50493
DATE: 10/24/2009
INITIALS: TME

FSAR Figure 3.8–73

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION

PLANT LAYOUT FINAL
AUXILIARY & CONTROL BUILDING
SECTION H-H, LOOKING WEST

SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
9	11/12/12	JMR	REVISED PER ECR-50780C	MGR	DR
8	9/12/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	E-002-006		9

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 DYNAMIC SYSTEM ANALYSIS AND TESTING

3.9.1.1 Vibration Operational Test Program

Piping vibration tests are performed during the initial test program to comply with the recommendations of Regulatory Guide 1.68 (see Appendix 3A). Criteria for these tests satisfy the requirements of the applicable portions of the ASME Code, Section III. These tests are scheduled to commence with the heatup for hot functional testing and continue through hot functional testing and the cooldown thereafter. Further testing may be conducted during the power ascension test program. The scheduling of these tests allows system operation at normal operating temperatures and pressures.

This program encompasses all ASME Class 1, 2, and 3 piping systems, all other high energy piping systems inside Seismic Category I structures, high energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and Seismic Category I portions of moderate energy piping systems located outside containment. Valves, piping, and supports associated with the components in these systems will be visually checked for abnormal vibration during the normal course of the preoperational test program. The specific conditions under which vibration will be observed are as follows where applicable.

1. Normal flow mode.
2. Minimum flow mode.
3. Maximum flow mode.
4. Startup.
5. Shutdown.
6. Other specific transient or operating conditions which might be expected to produce abnormal vibration or pressure pulsations.

Specific attention will be directed toward evaluating possible vibration problems during the performance of the following transients:

1. Start and stop reactor coolant pumps with associated operation of valves (closures/openings) in primary reactor coolant piping systems.
2. Start and stop RHR pumps with normal operation (closures/openings) of the associated valves in RHR piping systems.
3. Operation of high pressure safety injection piping system and charging system.

4. Operation of pressurizer relief valves and the associated discharge piping system.

5. Start and stop emergency feedwater pump with normal operation (closures/openings) of the associated valves in the emergency feedwater piping system.

6. Main steam turbine stop valve trip.

7. Main steam relief valve blowdown.

8. Condenser and atmospheric dump valve opening.

Acceptance criteria will include those stipulated in the ASME Code, Section III, Subparagraph NB3622.3, which states that vibration effects in piping systems shall be visually observed and, where questionable, shall be measured. If excessive vibration is measured under the above transient loadings, restraints will be installed or relocated to reduce the vibration to acceptable levels.

Determination of "questionable vibration" levels rests entirely in the judgment of a qualified observer. Typical qualification for this observer will include an engineering degree from an accredited institution, or equivalent, as well as a minimum of five years experience in the field of piping/pipe support engineering. His judgment will be based upon the operational transients which have been defined in the ASME Design Specification for the applicable piping system, presence of existing pipe support/restraint hardware provided to mitigate the consequences of these transient loads, and relative flexibility of the piping system.

If, in the opinion of the qualified observer, vibration levels are considered acceptable, no vibration measurements will be required. If vibration levels are deemed questionable, suitable instrumentation will be provided at representative locations to measure either the resulting displacement, acceleration, or pressure rise during a subsequent test of the piping system. This output will be integrated into, or compared with, existing stress calculations to evaluate the effect of this measured transient on the piping/pipe restraint system. If this transient is found to produce a negligible effect, no system revisions will be made. If the stress levels exceed the design basis, detailed computer calculations will be performed to thoroughly evaluate the system response to this transient condition. Additionally, an inspection of the support/restraint system hardware will be made to assess its adequacy.

In the event supports are added or relocated during this process, they will be considered as "design deviations" and handled in accordance with the approved quality assurance program, thereby assuring the evaluation of such deviations in the final system analysis.

3.9.1.2 Dynamic Testing Procedures

Table 3.9-0 lists mechanical equipment required for safe shutdown of the plant, as well as the method of seismic qualification. Also, this table indicates whether single or multiple axis input forcing functions were used.

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 Sections 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 analysis which demonstrates steam generator tube integrity for the combined LOCA plus SSE is included in WCAP 7832, "Evaluation of Steam Generator Tube, Tube Sheet and Divider Plate Under Combined LOCA Plus SSE Conditions", December 1973.

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 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

3.9.1.3.1 Analysis Methods

The reactor internals are modeled dynamically for:

1. Loads produced by a pipe rupture of the reactor coolant loop for both cold and hot leg breaks,

2. Response due to a safe shutdown earthquake (SSE), and

3. Combination of LOCA and SSE.

The analysis methods for reactor internals are discussed in Section 3.9.3 except for seismic analysis. Seismic analysis of the reactor vessel and its internals are described in Section 3.7.

3.9.1.3.2 Preoperational Tests

The program used to establish the integrity of reactor internals has evolved from extensive design analysis, model testing, and post hot functional inspection. Additionally, full size reactors have been instrumented[1] to measure dynamic behavior 02-01
including a Virgil C. Summer Nuclear Station size plant and the measurements have been compared with predicted values.

The reactor instrumentation program was instituted as part of a basic philosophy of instrumenting the internals of the "first-of-a-kind" of the current nuclear steam supply system designs for power plants. These data provide added assurance of the adequacy of the internals design and assisting in the development of increased 02-01
capability for the prediction of the dynamic behavior of pressurized water reactor (PWR) internals. The "first-of-a-kind" plants that have been instrumented are R. E. Ginna (two loops), H. B. Robinson No. 2 (three loops), Indian Point Unit II (four loops), and Trojan (neutron panels and 17 x 17 style internals).

The H. B. Robinson No. 2 reactor has been established as the prototype for the Westinghouse three-loop plant internals verification program. Subsequent three-loop plants are similar in design. Experience with other reactors indicates that plants of similar designs behave in a similar manner. For these reasons an instrumentation program was conducted on the H. B. Robinson No. 2 to confirm the behavior of the reactor internal components.

The only significant differences between the Virgil C. Summer Nuclear Station internals and the H. B. Robinson No. 2 internals are the replacement of the annular thermal shield with neutron shield panels and the substitution of 17 x 17 fuel assemblies for 15 x 15 assemblies.

The replacement of the thermal shield with segmented neutron shield panels results in a reduction of the flow induced vibrations of the reactor core structures. This conclusion was confirmed in tests with a 1/22nd scale mode.[2,3] The flow test was first conducted 02-01
on a model with a thermal shield and then on a model with neutron shield panels. The results indicated that the vibration levels of the internals were low and levels on the neutron shield panel were negligible. Appendix B on Reference [2] presents the test results. Reference [4] justifies in more detail the comparison of the relative effects of replacing the annular thermal shield with neutron shielding pads.

There is no change in the configuration of the reactor internals core support structures from the 15 x 15 fuel assembly configuration due to the incorporation of the 17 x 17 fuel assembly. The mechanical properties of the 17 x 17 fuel assembly, such as fuel assembly weight and beam stiffness, are virtually identical to the 15 x 15 fuel assembly; therefore, their input to the reactor internals core support structures is the same and the response of the total reactor internals core support structural model will not change.

The change to a 17 x 17 core configuration results in the use of newly designed guide tubes which are stronger and more rigid than the 15 x 15 guide tubes and hence are less susceptible to flow-induced vibration. The remainder of the core structure design has not been changed, and consequently remains identical to the prototype which has been tested and proven to be well within design expectations and limits.

The Portland General Electric Company Trojan plant internals have been 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 the Virgil C. Summer Nuclear Station.

The Three Loop Internals Assurance Program conducted on H. B. Robinson No. 2, supplemented by the Trojan data on neutron panels and 17 x 17, jointly satisfy the intent of Regulatory Guide 1.20.

The core support structures 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 (see Appendix 3A). This examination will include the points in Figure 3.9-1, summarized as follows:

1. All major load bearing elements of the reactor internals relied upon to retain the core structure in place.

2. The lateral, vertical, and torsional restraints provided within the vessel.

3. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

4. Those other locations on the reactor internals components that are similar to those which were examined on the prototype H. B. Robinson No. 2 design.

The inside of the vessel is inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign materials are in evidence.

1. Lower Internals

   A particularly close inspection will be 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. Examine bearing surfaces for any shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.

d. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

e. Baffle assembly locking devices. Check for lockweld integrity.

f. Lower barrel to core support girth weld.

g. Neutron shield panel screw locking devices and dowel pin cover plate welds. Examine the interface surfaces for evidence of tightness and for lockweld integrity.

h. Radial support key welds.

i. Insert screw locking devices. Examine soundness of lockwelds.

j. Core support columns and instrumentation guide tubes. Check all the joints for tightness and soundness of the locking devices.

k. Secondary core support assembling welds.

l. Lower radial support keys and inserts (Examine for any shadow marks, burnishing, buffing, or scoring. Check the integrity of the lockwelds). These members supply the radial and torsion constraint of the internals at the bottom relative to the reactor vessel while permitting axial growth between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing, or shadowing marks that would indicate pressure loading and relative motion between the two parts. Some scoring of engaging surfaces is also possible and acceptable.

m. Gaps at baffle joints. (Check for gaps between baffle and top former, and at baffle to baffle joints.)

2. Upper Internals

A particularly close inspection will be 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. Examine for any shadow marks, burnishing, buffing, or scoring. Check the locking devices for integrity of lockwelds.

e. Thermocouple conduit gusset and clamp welds.

f. Thermocouple end plugs. (Check for tightness.)

g. 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 will be 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 will be 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.

3.9.1.3.3 Startup Tests

Vibration monitoring in accordance with Appendix "A" Section B.1.J of Regulatory Guide 1.20 and Section D.1.P of Regulatory Guide 1.68 is not necessary based on guidance given in Section 2.2.2.c of Regulatory Guide 1.20. This section allows the vibration testing to be conducted without real or dummy fuel assemblies if it can be shown by analytical or experimental means that such conditions will yield conservative results. An analysis of the testing described in 3.9.1.3.2 above, and the testing performed during hot functional testing prior to fuel loading and without any dummy fuel assemblies has been shown to yield conservative results. On this basis, no additional testing will be performed after core loading. 02-01

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 the plant site. Extensive instrumentation programs to measure vibration of reactor internals (including prototype units of various reactors) have been carried out during preoperational flow tests.

From scale model tests, information on stresses, displacements, flow distribution and fluctuating differential pressures is obtained. Studies have been performed[1] to verify the validity and to determine the prediction accuracy of models for determining reactor internals vibration due to flow excitation. 02-01

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 both hybrid real-time and digital techniques to determine the approximate frequency and phase content. In some structural components, the spectral content of the signals include nearly discrete-frequency or very narrow-band frequency, usually due to excitation by the main coolant pumps and other components that reflect the response of the structure at a natural frequency to broad band, mechanically or flow-induced excitation. Damping factors are also obtained from wave analyses.

In general, the determination of internals responses proceeds as follows. Frequencies and spring constants are obtained analytically and these values are confirmed with test results. Theoretical and experimental studies have provided information on the added apparent mass of the water, which has the effect of decreasing the natural frequency of the component. Damping coefficients are established experimentally and forcing functions are characterized from previous studies including those discussed above. Once these factors are established, the response can be computed analytically. In addition, the responses of important reactor structures are measured during preoperational reactor tests and frequencies and mode shapes of the structures are obtained. Once all of the dynamic parameters are obtained as explained above, the forcing functions can be estimated. When combined, these studies provide indications of the internal behavior during reactor operation.

Pre and post hot functional inspection results, in the case of plants similar to prototypes, serve to confirm preditions that the internals are well-behaved. Any gross motion or undue wear would be evident following the application of approximately 10^7 cycles of vibration expected during the test period.

3.9.1.5 Analysis Methods Under LOCA Loadings

1. 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 the blowdown transient. The dynamic analysis can employ the displacement method, lumped parameters, stiffness matrix formulations, and assumes that all components behave in a linearly elastic manner.

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. Analytical methods are discussed in more detail in Section 3.9.3.

2. Reactor Coolant Loop Analysis

The description of the methods and procedures that will be used to compute the dynamic response of the reactor coolant loop for a loss of coolant accident is present in Section 5.2.1.

3.9.1.6 Analytical Methods for ASME Code Class 1 Components

Analytical methods for ASME Code Class 1 components are discussed in Section 5.2.1.

3.9.2 ASME CODE CLASS 2 AND 3 COMPONENTS

3.9.2.1 Plant Conditions and Design Loading Combinations

For balance of plant scope, plant operating conditions are correlated with component operating conditions in Table 3.9-1. The plant operating condition categories establish related sets of design loading combinations that are applied to the design of systems and components.

For the components and equipment supplied by the NSSS vendor the design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9.2.2 Design Loading Combinations

For balance of plant scope, design loading combinations for Code Class 2 and 3 components are based upon the categorizations of component operating condition which are derived from the plant operating conditions. Stress limits and specific loading combinations for the basic types of mechanical components, such as valves, vessels, piping, and pumps, are presented in Table 3.9-2.

The design loading combinations for ASME Code Class 2 and 3 components and supports provided by the NSSS vendor are given in Table 3.9-3. The design loading combinations are categorized with respect to Normal, Upset, and Faulted Conditions. Stress limits for each of the loading combinations are component oriented and are presented in Tables 3.9-4, 3.9-5, and 3.9-6 for tanks, pumps, and valves, respectively.

3.9.2.3 Design Stress Limits

For balance of plant scope, design stress limits which allow inelastic deformation are not permitted for Code Class 2 and 3 components, except for a limited number of situations involving piping subjected to jet impingement forces or pipe rupture loadings. Table 3.9-2 provides detailed information concerning these exceptions.

The design stress limits established for the components provided by the NSSS vendor are sufficiently low to assure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading combinations, are component oriented and are presented in Tables 3.9-4 through 3.9-6.

Class 2 and 3 components within Westinghouse scope are designed and analyzed using umbrella nozzle loads. In the component analysis, the effects of these umbrella nozzle loads are combined by absolute sum with the effects of the SSE. The loads applied to the component from the piping system due to all applicable loading combinations do not exceed the loads for which the component is designed.

Combination of responses (e.g. stress, strain, moment, shear or displacement) of the SSE and the dynamic loads associated with the faulted plant condition (including LOCA) for Class 1, 2, and 3 components within the balance of plant scope is performed by the SRSS method. Engineering acceptance of this method is based upon the extremely low order of probability for the simultaneous occurrence of the SSE and a faulted plant condition. This is coupled with the fact that the SRSS combination of two time varying loads is adequately represented by this method. The SRSS methodology for treatment of multiple time varying loads is generally established in the evaluation of the three spatial earthquakes. Based upon these low probabilities, a highly satisfactory margin of safety is obtained using this method of combination.

3.9.2.4 Analytical and Empirical Methods for Design of Pumps and Valves

3.9.2.4.1 Balance of Plant Scope

A list that identifies all active Code Class 2 and 3 pumps is provided by Table 3.9-7. A similar list for active Code Class 1, 2, and 3 valves is presented by Table 3.9-8.

Stringent criteria are employed to ensure that the operability of active pumps and valves is not compromised during or following specified plant events. These criteria are specified for each active component and are included in ASME design specification or other contractual documents.

Provisions to ensure operability include specification of loading combinations and stress limits, as discussed in Section 3.9.2.3, for active pumps and valves.

In addition to compliance with the design limits specified, it is specified that assurance of operability under all design loading combinations be provided by test and/or analysis. In the performance of tests or analyses to demonstrate operability, the structural interactions of entire valve-operator and pump-motor assemblies are considered.

When superposition of test results for other than the combined loading condition is proposed, the applicability of such a procedure is demonstrated. Qualification of components by analysis only is permitted when testing is not feasible.

When proof of operability is shown by analysis, a deformation (interference) analysis is performed. The purpose of this deformation analysis is to show that all moving parts are uninhibited in movement due to deformation of components caused by any loading conditions described in Section 3.9.2.1.

3.9.2.4.2 Components and Equipment Supplied by the NSSS Vendor

The design methods and stress limits for pumps and valves are described above. These limits, selected following the Code intent, are sufficiently low to provide assurance that no gross deformation will occur in active [(1)] components and that the active components will operate as required following the event. The limits established for inactive [(2)] 02-01
components are intended to assure that violation of the pressure retaining boundary will not occur. Additional requirements are provided in Section 3.9.4.

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(1) Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

(2) Inactive components are those whose operability is not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.

Active pumps and valves are listed in Tables 3.9-7 and 3.9-8, respectively.

3.9.2.5 Design and Installation Criteria, Pressure Relieving Devices

1. Code Class 1 Components

   Design and installation criteria for Code Class 1 pressure relieving devices are discussed in Section 5.2.2.

2. Code Class 2 Components, Main Steam Safety Valves

   Safety-relief valves are mounted on the main steam lines from each steam generator, as described in Section 10.3. These safety valves are provided in accordance with the ASME Code, Section III, and are capable of relieving at least 110 percent of rated main steam flow. The valves are mounted in a manner that minimizes the reaction moments about the mounting flanges. The valves discharge into an open vent system which is sized to permit free passage of relieved steam to the atmosphere without development of excessive backpressure.

   The most severe combinations of applicable loads relative to main steam piping are presented in Section 3.9.2.1. These loading combinations are addressed in the analysis of main steam piping and mounting nozzles. The dynamic load factor for any individual main steam safety valve opening is determined to be 1.42.

3. Code Class 2 and 3 Components, Other than Main Steam

   The safety-relief valves on the sodium hydroxide storage tank (Code Class 3) are 1-1/2 inch valves. The valve reaction forces and moments are inputs to the associated piping stress analysis.

3.9.2.6 Stress Levels for Category I Components

Methods used to analyze Seismic Category I systems are discussed in Sections 3.9.1, 3.9.2, and 5.2.

3.9.2.7 Field Run Piping System

All Seismic Category I piping is fabricated from detail dimensioned engineering drawings. Section 3.7.3.8 describes the simplified procedure for the design of this piping.

3.9.2.8 Class 2 and 3 Component Supports

3.9.2.8.1 Balance of Plant Scope

1. Piping Supports

   Piping supports for safety-related, ASME Code Class 2 and 3 piping are designed and analyzed in accordance with the requirements of the ASME Code, Section III, Subsection NF, Winter 1973 Addenda, except for supports that function as a snubber. The components within a pipe support that function as a snubber for safety related, ASME Code Class 2 and 3 piping are designed and analyzed in accordance with the requirements of the ASME Code, Section III, Subsection NF, Winter 1976 Addenda. Except for snubber components, the use of Subsection NF of the Code represents a voluntary commitment since the contract date for piping and its supports predates mandatory application of Subsection NF. 02-01

   Loading combinations and stress limits are presented in Table 3.9-2. Support deformations are limited to assure piping system function and integrity.

   A set of recommended installation instructions, as well as detailed design drawings, is provided to the pipe support installer to facilitate correct installation procedures. Control of this function is in accordance with the quality control procedures of the installer. Additionally, in accordance with the testing requirements section of Article NF-6000 of the ASME Code, Section III, Winter, 1973, Addenda the piping designer will check the hot and cold settings during the plant startup and test phase to ensure both proper installation in accordance with the design drawing and that sufficient travel is available to allow for free thermal expansion of the piping.

   These components are accessible for inspection. There are a number of inspection platforms provided throughout the plant which would offer access to snubbers used on major pieces of equipment. Any snubbers found to be inoperable or maladjusted during the plant startup and operational test phase will be either readjusted, repaired, or replaced with spare components.

   A tabulation listing concerning snubbers utilized for support of safety related systems and components information is maintained for all pipe supports/restraints for the plant.

   All mechanical snubbers have "certified" load capacities on file in accordance with the requirements of Article NF of the ASME Code.

2. Pump and Vessel Supports

   Supports for pumps and vessels are designed and analyzed in accordance with the requirements of the ASME Code, Section III, Subsection NF, as applicable. Component supports not under the jurisdiction of Subsection NF by virtue of the contract date for the component and its supports are analyzed and designed to assure the integrity of the supports under loading combinations specified for the component.

   Loading combinations and stress limits are presented in Table 3.9-2. Support deformations are limited to assure the pressure boundary integrity and operability (for active components).

3. Valve Supports

   Except for special cases, valves are supported by the piping in which they are mounted. Details regarding load combinations and stress limits for piping are discussed in Section 3.9.2.2 and 3.9.2.3. Piping supports are discussed in Item 1 above.

   In special cases, such as large, massive valves, additional supports are provided as required. Design of these valve supports satisfies the same requirements as the piping supports.

3.9.2.8.2 Component Supports Provided by the NSSS Vendor

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.

3.9.2.8.3 Support Design Criteria

Structural bolting of support components within the NF code boundary are designed in accordance with Appendix XIII of ASME Section II, Winter 1973 Addenda or later code edition, or the American Institute of Steel Construction (AISC) manual, Seventh Edition or later.

The design of pipe support components not covered by ASME Code, Section III, Subsection NF shall be in accordance with the AISC Manual of Steel Construction Seventh Edition or later.

3.9.3 COMPONENTS NOT COVERED BY ASME CODE

3.9.3.1 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 PWR plant internals has been determined. The following mechanical functional performance requirements apply:

1. Following the LOCA, the basic operational or functional requirement to be met for the reactor internals is that the plant shall be shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

2. 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 assure 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.

3. 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.

4. The rod cluster control guide tube deflections are limited to ensure operability of the control rods.

5. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited.

6. The reactor has mechanical provisions which are sufficient to maintain the design functionality of the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from the LOCA operating conditions (References [5] and [6]).

7. The core internals are designed to withstand mechanical loads arising from operating basis earthquake, SSE, and pipe ruptures (References [6], [7], and [8]).

3.9.3.1.1 Faulted Conditions

The following events are considered in the faulted conditions category:

1. Loads produced by a rupture of the largest branch line piping, excluding the main (primary) coolant loop from consideration through crediting Leak-Before-Break (LBB) for both cases: a cold leg branch line and a hot leg branch line. 02-01

   The branch lines analyzed for V. C. Summer are the accumulator line and the pressurizer surge line. The methods of analysis adopted are related to the type of accident assumed (cold leg break or hot leg break). Note that throughout this description, hot or cold leg break refers to where the broken branch lines were attached to the main coolant loop. The size of the postulated break and its location along the primary loop piping are determined by the size and location of the branch line piping, as a result of crediting Leak-Before-Break exclusion of the main (primary) coolant loop for consideration with respect to LOCA forces.

2. Response due to a SSE.

3. Combination of SSE and LOCA.

Maximum stress intensities are compared to allowable stresses for each of the above conditions. When fatigue is of concern, the applicable stress concentrate factors are utilized and peak stresses are used to establish the usage factor. Elastic analysis is utilized to obtain the response of the structure and the stress analysis on each component is performed on an elastic basis. For faulted conditions stresses are 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. The reference to the use of inelastic techniques is applicable to the guide tube evaluations via test results, as well as to the core barrel inward and outward expansions. The guide tube tests and results are described in Reference [13]. The evaluation of the inward deflections of the core barrel is discussed in Reference [14]. The outward barrel deflection analysis is described in Reference [12]. For the outward barrel deflection, inelastic behavior is confined to a small region of the girth weld where the stresses exceed yield; independent hand calculations, evaluating the elongation of this region, indicated that the residual barrel deflection does not exceed the limit given in Table 4.2-1.

The criteria for establishing the deformation limits given in Table 4.2-1 reflect the mechanical functional performance requirements, which are discussed in Section 3.9.3.1. Guide tube deflections are limited to allow control rod insertion. Inward upper barrel deflections are limited to prevent barrel contact with the nearest guide tube. Outward upper barrel deflections are limited in order to maintain an adequate annulus for ECCS flow between the vessel inner diameter and the core barrel outer diameter.

The number of full cycle variations of significant LOCA loadings is quite small; therefore, resonance and high cycle endurance limit considerations are not of concern. Dynamic amplification for LOCA impulse-like loads and, therefore, maximum system response is considered. The elastic limit allowable stresses are used to compare with the result of the analysis. No inelastic stress limits are used.

The above described analyses show that the stresses and deflections which would result following a faulted condition are less than those which would adversely affect the integrity of the structures. Also, the natural and applied frequencies are such that resonance problems should not occur.

The use of unirradiated stress and elongation limits (Paragraph 4.2.2.5) to justify these conclusions is valid because the points of interest, listed in Table 4.2-1, are in low fluence, low irradiated zones. In these areas, the total fluence level for 54 full power years is less than 1.5×10^{21} n/cm^2 (E > 1 Mev). Below this level, data indicates that the decrease in elongation of stainless steel is such that the use of unirradiated limits is valid. RN 04-019

3.9.3.2 Reactor Internals Response Under Blowdown and Seismic Excitation

A loss of coolant accident would result from a rupture of reactor coolant piping. During the blowdown of the coolant, critical components of the core are subjected to vertical and horizontal excitation as a result of rarefaction waves propagating inside the reactor vessel. For these 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 pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A one millisecond severance time is taken as the limiting case.

In the case of the hot leg break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefaction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic instability (buckling) or large deflections of the upper core barrel or both is the possible response of the barrel during hot leg blowdown. In addition to the above effects, the hot leg break results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg break, a rarefaction wave propagates along a reactor inlet piping arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg break, the initial steady-state hydraulic lift forces (upward) decrease rapidly (within a few milliseconds) and then increase in the downward direction. These cause the reactor core and lower support structure to move initially downward.

If a simultaneous seismic event with the intensity of the SSE is postulated with the loss of coolant accident, the imposed loading on the internals component may be additive in certain cases and therefore the combined loading must be considered.

In general, however, the loading imposed by the SSE is small compared to the blowdown loading.

3.9.3.3 Acceptance Criteria

The criteria for acceptability in regard to mechanical integrity analyses is that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and stability of the parts in addition to a stress criterion to assure integrity of the components.

1. Allowable Deflection and Stability Criteria

   For the loss of coolant plus the SSE condition, deflections of critical internal structures are limited to the values given in Section 4.2.2.

   In a hypothesized downward vertical displacement of the internals, energy absorbing devices limit the displacement after contacting the vessel bottom head, ensuring that the geometry of the core remains intact.

2. Upper Barrel

   The upper barrel deformation has the following limits:

   a. To ensure a shutdown and cooldown of the core during blowdown, the basic requirement is a limitation on the outward deflection of the barrel at the locations of the inlet nozzles connected to the unbroken lines. A large outward deflection of the barrel in front of the inlet nozzles, accompanied with permanent strains, could close the inlet area and stop 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 the "no loss of function" limit, could impair the efficiency of the emergency core cooling system.

   b. To assure rod insertion and to avoid disturbing the control rod cluster guide structure, the barrel should not interfere with the guide tubes. This condition also requires a stability check to assure that the barrel will not buckle under the accident loads.

3. Control Rod Cluster Guide Tubes

   The guide tubes in the upper core support package house the control rods. The deflection limits were established from tests and are provided in Section 4.2.2.

4. Fuel Assembly

   The limitations for this case are related to the stability of the thimbles in the upper end. The upper end of the thimbles must not experience stresses above the allowable dynamic compressive stresses. Any buckling of the upper end of the thimbles must not experience stresses above the allowable dynamic compressive stresses. Any buckling of the upper end of the thimbles due to axial compression could distort the guide line and thereby affect the free fall of the control rod.

5. Upper Package

   The local vertical deformation of the upper core plate, where a guide tube is located, is limited so as to prevent the guide tubes from undergoing compression.

6. Allowable Stress Criteria

   The allowable stress limits during the LOCA used for the core support structures are based on the limits specified in Section 5.2.1.3. This section defines various criteria based upon their corresponding method of analysis. To account for multi-axial stresses, the von Mises theory is also considered.

3.9.3.4 Methods of Analysis

The internal 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 very large requiring many different techniques and methods, both static and dynamic. The more important and relevant methods are presented as an overview in Section 3.9.1 and summarized in the following.

3.9.3.5 Blowdown Forces Due to Cold and Hot Leg Break

The MULTIFLEX 3.0 (Reference [17]) computer program, an enhancement and extension of the NRC-approved MULTIFLEX 1.0 computer program (Reference [9]) was employed to generate the blowdown thermal-hydraulic transient in the primary reactor coolant system due to a postulated pipe rupture, or Loss-Of-Coolant-Accident (LOCA) in both the reactor coolant system hot and cold legs. It considers subcooled, transition and two-phase (saturated) blowdown regimes, employing the method of characteristics to solve the conservation laws, assuming one dimensional flow and a homogeneous liquid-vapor mixture. With its ability to model flow branches and a large number of nodes, MULTIFLEX 3.0 has the required flexibility to represent various flow passages within the primary reactor coolant system. The reactor coolant system is divided into subregions in which the fluid flows along longitudinal axes. While each subregion is regarded as an equivalent pipe, a complex network of these equivalent pipes is used to represent the entire primary RCS. RN 09-022 RN 01-068 RN 09-022

A coupled fluid-structure interaction is incorporated into the analysis by accounting for the deflection of the constraining boundaries, which are represented by separate spring-mass oscillator systems. The reactor core barrel is modeled as an equivalent beam with the structural properties of the core barrel in a plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into 10 segments, with each segment consisting of 3 walls. Mass and stiffness matrices that are obtained from an independent modal analysis of the reactor core barrel are applied in the equations of structural vibration at each of the 10 mass points locations. Horizontal forces are then calculated by applying the spatial pressure variation to the wall area at each of the elevations representative of the 10 mass points of the beam model. The resultant core barrel motion is then translated into an equivalent change in flow area in each downcomer annulus flow channel. At every time increment, MULTIFLEX iterates between the hydraulic and structural subroutines for each location confined by a flexible wall. RN 01-068

Predictions of this code have been compared with numerous test data (Reference [11]) and the results show good agreement in both the subcooled and the saturated blowdown regimes.

1. FORCE Models for Blowdown

   The MULTIFLEX 3.0 blowdown code evaluates the pressure and velocity transients at various 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 (Reference [9]) which utilize a detailed geometric description in evaluating the loadings on the reactor internals. RN 09-022

Each reactor component for which FORCE2 calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of: RN 09-022

a. The pressure differential across the element.

b. Flow stagnation on, and unrecovered orifice losses across the element.

c. Friction losses along the element.

Input to the code, in addition to the MULTIFLEX 3.0 calculated blowdown 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. RN 09-022

In addition to the vertical forces calculated by FORCE2 (Reference [9]), the horizontal forces on the vessel, core barrel, and thermal shield are calculated by LATFORC (Reference [9]). 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 10 different elevations. The total lateral force is calculated by summing the forces over the ten elevations. RN 09-022

The mechanical analysis has been performed using conservative assumptions in order to obtain results with extra margin. Some of the most significant are:

a. 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.

b. The multi-mass model described is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the horizontal and vertical direction.

The summary of the mechanical analysis follows: RN 09-022

2. Transverse and Vertical Excitation Model for Blowdown and Seismic Events

The mathematical model of the reactor pressure vessel (RPV) is a three-dimensional nonlinear finite element model which represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model consists of three concentric structural submodels connected by nonlinear impact elements and stiffness matrices. The first submodel, represents the reactor vessel shell and associated components.

The reactor vessel is restrained by the reactor vessel supports and by the attached primary coolant piping. Each reactor vessel support is modeled by a linear horizontal stiffness and a vertical impact element. The attached piping is represented by a stiffness matrix.

The second submodel, represents the reactor core barrel (RCB), neutron panels, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost submodel, represents the upper support plate, guide tubes, support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and second by stiffness matrices and nonlinear elements.

Fluid-structure or hydro-elastic interaction is included in the reactor pressure vessel model for seismic evaluation. The horizontal hydro-elastic interaction is significant in the cylindrical fluid flow region between the core barrel and reactor vessel (the downcomer). Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel and reactor vessel shell. RN 09-022

The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel shell 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 a 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 RCB allows inclusion of radii variations along the RCB height and approximates the effects of RCB beam deformation. These mass matrices were inserted between selected nodes on the core barrel and reactor vessel shell.

This method of modeling of the reactor internals for LOCA and seismic dynamic analyses has been approved by the USNRC (Reference [18]).

The appropriate forcing functions are applied simultaneously and independently. The results from the program give the forces, displacements, and deflections as functions of time for all the reactor internals components. Reactor internals response to both hot and cold leg pipe rupture was analyzed.

3. Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

a. Core Barrel

For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse. The barrel is then analyzed for dynamic buckling using these conditions and the following conservative assumptions: 02-01

(1) The effect of the fluid environment is neglected (water stiffening is not considered);

(2) The shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

(1) The upper core barrel is treated as a simply supported cylindrical shell of constant thickness between the upper flange weldment and the lower core barrel weldment without taking credit for the supports at the barrel midspan offered by the outlet nozzles. This assumption leads to conservative deflection estimates of the upper core barrel.

(2) The upper core barrel is analyzed as a shell with four variable sections to model the support flange, upper barrel, reduced weld section, and a portion of the lower core barrel.

(3) The barrel with the core and thermal shielding pads, is analyzed as a beam fixed at the top, and elastically supported, at the lower radial support and the dynamic response is obtained.

b. Guide Tubes

The guide tubes in closest proximity to the outlet nozzle of the ruptured loop are the most severely loaded during a blowdown. The transverse guide tube forces decrease with increasing distance from the ruptured nozzle location.

All of the guide tubes are designed to maintain the function of the control rods for a break size of 144 in^2 and smaller. No credit for the function of the control rods is assumed for break size areas above 144 in^2. However, the design of the guide tube will permit control rod operation in all but four control rod positions which is sufficient to maintain the core in a subcritical configuration for break sizes up to a double ended hot leg break. This double ended hot leg break imposes the limiting lateral guide tube loading.

Upper support columns located close to the broken nozzle during hot leg break will be subjected to transverse loads due to cross flow.
The loads applied to the columns were computed with a similar method to the one used for the guide tubes, i.e., taking into consideration the increase in flow across the column during the accident. The columns were studied as beams with variable sections and the resulting stresses were obtained using the reduced section modulus and appropriate stress risers for the various sections.

3.9.3.6 Methods and Results of Blowdown Analysis (Mechanical)

The results obtained from the nonlinear analysis indicate that during blowdown, the relative displacement between the components will close the gaps and consequently the structures will impinge on each other. The effects of the gaps that exist between vessel and barrel, between fuel assemblies, between fuel assemblies and baffle plates, and between the control rods and their guide paths are considered in the analysis. References [5], [12] and [18] provide further details of the blowdown method used in the analysis of the reactor internals. RN 09-022

Results of these analyses indicate that both static and dynamic stress intensities are within acceptable limits. In addition, the cumulative fatigue usage factor is also within the allowable usage factor of unity.

The stresses due to the SSE (vertical and horizontal components) are combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection.

These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to assure control rod insertion. For the guide tubes deflected above the no loss of function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9.3.7 Control Rod Drive Mechanisms

The control rod drive mechanisms are Class 1 components designed to meet the stresses of the ASME Boiler and Pressure Vessel Code and, therefore, are presented in Section 4.2.

3.9.3.8 Bolt Loads In Bolted Connections

Typical bolted connection/base plate designs, representing the majority of such designs developed for use on the Virgil C. Summer Nuclear Station, can be classified as follows:

a. Group I - Modular Embedments

   A series of four different configurations of steel base plates and Nelson studs. The plates conform to ASTM A36; the studs, to ASTM A108-58T.

b. Group II - Random Embedments

   A 12 inch square plate conforming to the same material requirements identified in Item 1 above for Group I embedments.

c. Group III - Drilled-in Expansion Bolt Anchored Plates

   Steel base plates having a variety of sizes and thicknesses anchored to the concrete with expansion bolts manufactured by Hilti, Inc. The expansion bolt material conforms to AISI-1144 and/or AISI-1038

   Maxi-Bolts, manufactured by Drillco, are another type of drilled-in anchorage system used to secure steel base plates of various size and thickness. These anchors have an undercut design and are typically utilized in applications which require a higher anchor capacity. MaxiBolt materials conform to ASTM A193, Grade B7. RN 07-024

These designs are illustrated by Figure 3.9-2.

Maximum allowable design loads for Group I and Group II designs are as plotted on Figures 3.9-3 and 3.9-4, respectively. These curves include consideration of the relationship between tensile loads, shears, permissible load eccentricities, and fixture types for modular and random embedments. RN 07-024

Group III Hilti designs were based upon a straight line interaction diagram relating tensile loads and shear loads and upon the assumption that the allowable bolt capacities in tension and shear were 25 percent of failure loads for hilti kwik bolts reported by Abbot A. Hanks, Inc., Testing Laboratories in March 1977 and published in Reference [2]. The allowable loads for hilti kwik bolts are 25 percent of the ultimate tension and shear loads published by Hilti. RN 07-024 02-01

Group III Maxi-Bolt designs were based upon ACI 349 Appendix B criteria for ductile anchor failure and the 1986 VCS Maxi-Bolt qualification testing program. Allowable tension and shear loads specified in design documents consider minimum concrete strength (3,000 psi), minimum specified ultimate bolt strength, and bolt pre-tensioning. Load factors are in accordance with applicable FSAR loading combinations. Appropriate reduction factors take into account anchor spacing, edge distance, and / or concrete thickness where applicable. RN 07-024

For all three design groups, the original design considered the plates to be rigid.

The base plate and bolt designs have been reexamined to evaluate the effects of plate flexibility, bolt stiffness, and potential prying action. Procedures were developed for the analysis of plates and anchorages for moment and axial load as shown by Figures 3.9-5 through 3.9-8. The indicated equations were derived from consideration of statics and deflection compatibility for both tension loads and moments. The equations permit calculation of the plate prying force and, subsequently, forces in the anchors and stresses in the plates. For the loading case of tension normal to the plate surface, a single equation can be solved directly for the prying force. For the loading case of moment applied to the plate surface, 6 equations were derived, the simultaneous solution of which yields the prying force. For both cases, criteria have been formulated to determine whether or not prying exists. That determination being a function primarily of the geometry of the detail, the preload and the magnitude of the applied loads.

Using the results of the analysis described by Figures 3.9-5 through 3.9-8, a variety of factors of safety and maximum axial stresses in the bolts were determined. For Group I plates the location of the applied load was chosen to produce the maximum stress in the studs. The resulting maximum stud tension was then combined with the shear on the stud, using the appropriate interaction equation, to determine the factor of safety. The interaction equation for evaluating the factor of safety for Group I and Group II embedments is, as defined in Reference [15]:

$$\left(\frac{F.S.\,x\,P}{P_{ult}}\right)^{\frac{5}{3}} + \left(\frac{F.\,S.\,x\,S}{S_u}\right)^{\frac{5}{3}} \leq 1$$

For plates anchored with expansion bolts the equation is, as defined in Reference [16]. 02-01

$$\frac{F.S.\,x\,T}{T_o} + \frac{F.\,S.\,x\,V}{V_o} \leq 1$$ 02-01

In the above equations P and T are applied tensile loads, S and V are applied shear loads and P_{ult}, T_o, S_u and V_o are ultimate tensile and shear capacities. Because the tension in the stud controls the overall design of the anchorage, the stress in the studs is used as the basis for determining the factor of safety of Group I and Group II embedments. The results of this determination are summarized in Table 3.9-9 in which only the minimum factor of safety for the stud or bolt, determined during the investigation, is reported. 02-01

Reactions due to dynamic loads are obtained from a dynamic analysis of the piping system using a recognized and verified computer program. Pipe support reactions are generated as an output of the computer program and are used in the load summary for the design of the individual pipe supports. Since a dynamic analysis is performed, the pipe support reactions generated already include dynamic effects. Therefore, a dynamic load factor is not required.

The governing load for computing factors of safety for expansion bolts is the maximum support load including SSE.

Where appropriate, an equivalent static analysis is used in lieu of a dynamic analysis and a dynamic load factor is established. The dynamic load factor is computed by using the natural frequencies of the structures and the characteristics of the forcing function. These calculations are based upon classical techniques presented in such texts as <u>Introduction to Structural Dynamics</u> by Biggs.

Application of the analytical methods described in Figures 3.9-5 through 3.9-8 resulted in the conclusion that, for plates secured to a concrete surface with expansion bolts, prying forces were considerably less prevalent and, where present, less significant than for plates anchored with Nelson studs. For instance, for Group I plates approximately 35 to 70 percent of the total force on the bolt was due to prying. For Group II plates the prying force accounted for 20 to 30 percent of the total force. For Group III plates prying forces were detected in only 3 of the 94 plates analyzed and in these three cases the prying force increased the total force in the bolt by an average of 25 percent.

The difference in behavior between stud anchored plates and expansion bolt anchored plates, as affected by prying, is attributable to the greater axial stiffness of the stud with respect to the bolt anchorage. For the stud, stiffness is considered to be equal to AE/L and is, therefore, a totally elastic property of the stud. For the expansion bolt, the stiffness is determined from the expression AE/2.5L, where the factor 1/2.5 is applied to account for the reduction in axial stiffness due to tensile elongation of the concrete and the anchor. The resulting stiffness is less than that for an embedded stud but greater than that which would be determined directly from the initial slope of the force-displacement curve representing a concrete pullout test of a nonpreloaded anchor. Figures 3.9-9 through 3.9-12 show typical force-displacement curves, as provided by Hilti, Inc., for several different bolt sizes. (Abbot A. Hanks Report No. 8785, January 30, 1974).

As indicated in the preceding discussion, Figures 3.9-9 through 3.9-12 provide experimentally determined force-displacement pull out curves for Hilti bolts used in the Virgil C. Summer Nuclear Station. This data is supplemented by the data presented by Table 3.9-10 which summarizes a series of additional pull out tests performed by the expansion bolt manufacturer. These latter tests consistently indicated a failure in the anchor itself rather than a concrete failure. Nelson stud anchorages were also designed to ensure a ductile failure, including the use of 7/8" by 8-3/16" studs which Nelson indicates to be the most ductile of their headed anchors.

3.9.4 OPERABILITY ASSURANCE

Equipment for the Virgil C. Summer Nuclear Station 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

The requirements of Section III of the ASME Code were adhered to in the design of active 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 Code were adhered to in the design of Code Class 1 manually operated globe valves and check valves, 2 inches in size or less.

Class 2 and 3 active valves were also designed to the requirements of the ASME Code. In addition, an analysis of the extended structure was performed with loads of 3.0 g in the horizontal and 2.0 g in the vertical direction simultaneously. 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 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 Section III of the ASME Code. The stress levels in the pumps did not exceed those provided in Table 3.9-5. Forces resulting from seismic accelerations in the horizontal and vertical directions are included in the analysis of the pumps and their supports. To eliminate any amplification of the seismic floor accelerations in the pump support structure, the supports were designed to have natural frequencies in excess of 33 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.

3.9.5 REFERENCES

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TABLE 3.9-0

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM (1)	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION (4)	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION (5)
AH	XAA-1A, B; XAA-2A, B	Reactor Building Cooling Unit (4 Units)	a	1. Assembly - analysis	1. Lowest natural frequency determined to be 12.75 Hz. Dynamic seismic analysis, response spectrum input.
				2. Damper - analysis	2. Natural frequency determined to be 25.32 Hz. Static seismic analysis.
				3. Fan and Motor - analysis	3. Natural frequency determined to be 57.47 Hz. Static seismic analysis.
				4. Damper actuator	4. (2)
				5. Limit switch - test	5. Single frequency, single axis test. (1 to 35 Hz no resonance frequency determined in test range).

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM [(1)]	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION [(4)]	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION [(5)]	
AH	XAH-12A, B	Control Room Normal Supply Cooling Unit (2 Units)	a	1. Cabinet-analysis	1. Natural frequency determined to be 31 Hz. Seismic modal analysis.	02-01
				2. Dampers-test	2. Multiple frequency, multi-axis test (1 to 40 Hz).	
				3. Cooling coil - analysis	3. Natural frequency determined to be 31.5 Hz. Static seismic analysis.	02-01
				4. Fan - analysis	4. Natural frequency determined. Static seismic analysis.	02-01
				5. Motor	5. Certificate of conformance with IEEE-344 furnished by motor vendor. Qualification report on file at vendor's office for audit. Vendor considers qualification report as proprietary.	

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM [1]	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION [4]	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION [5]	
AH	XAH-13A, B	Relay Room Air Handling Unit (2 Units)	a	(6)	(6)	
AH	XFN-30A, B	Control Room Emergency Fan (2 fans)	a	Analysis	Natural frequency determined to be 61.1 Hz. Static seismic analysis.	02-01
AH	XFN-39A, B	Battery Room Exhaust Fan (2 fans)	a	Analysis	Natural frequency determined to be 50.8 Hz. Static seismic analysis.	
AH	XFN-80A, B	Service Water Pumphouse Supply Fan (2 fans)	a	Analysis	Natural frequency determined to be 56.8 Hz. Static seismic analysis.	
AH	XDP-12A, B; 13A, B; 18A, B;19A, B; 21A, B; 22A, B; 23A, B; 24A, B; 35A, B; 39A, B; 45; 72A, B; 73A, B; 74A, B; 88A, B; 89A, B; 96; 99A, B; 100A, B; 103A, B; 106; 112A, B; 113A, B; 133A, B; 129	Pneumatic Actuated Dampers	a	Type test	Biaxial, multifrequency test, response spectrum input. Type tests included damper, actuator and limit switches and solenoid valves.	02-01

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM [1]	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION [4]	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION [5]
AH	XDP-152, 153, 155	Electric Motor Actuated Dampers for Battery Rooms A and B (4 Units)	a	(2)	(2)
CC	XPP-1A, B, C	Component Cooling Water Pump (3 pumps)	a	1. Pump and motor analyses 2. Pump piping - test	1. Natural frequency determined to be 36.5 Hz. Static analysis. 2. Single sinusoidal test frequency, single axis input. Natural frequency is greater than 30 Hz.
CC	XHE-2A, B	Component Cooling Water Heat Exchanger (2 heat exchangers)	a	Analysis	Lowest natural frequency determined to be 31.5 Hz. Static seismic analysis.
CO	XTK-8	Condensate Storage Tank	b	Analysis	Hydrodynamic frequency determined to be 0.252 Hz. Seismic modal analysis (TID-25021). Response spectrum input.
CS	XPP-13A, B	Boric Acid Transfer Pump (2 pumps)	a	Analysis	Natural frequency determined greater than 30 Hz. Static seismic analysis.
CS	XPP-43A, B, C	Charging Pumps (3 pumps)	a	Analysis	Natural frequency determined greater than 30 Hz. Static seismic analysis.

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM (1)	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION (4)	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION (5)
CS	XTK-12A, B	Boric Acid Tank (2 tanks)	a	Analysis	Hydrodynamic frequency determined to be 0.464 Hz. Seismic modal analysis (TID-25021). Response spectrum input.
DG	XPP-4A, B; XPP-141A, B	Diesel Generator Fuel Oil Transfer (4 pumps)	a	Analysis	Natural Frequency determined to be 486 Hz. Static seismic analysis.
DG	XTK-9A, B, C, D	Diesel Generator Starting Air Skid (2 skids)	a	Analysis	Lowest skid natural frequency determined to be 20.7 Hz. Seismic modal analysis of starting air equipment.
DG	XTK-20A, B	Diesel Generator Fuel Oil Day Tank (2 tanks)	a	Analysis	Natural frequency determined to be 44.5 Hz. Hydrodynamic frequency determined to be 1.47 Hz. Seismic modal analysis (TID-7024). Static seismic analysis.
DG	XTK-53A, B	Diesel Generator Fuel Oil Storage Tank (2 tanks)	a	Analysis	Natural frequency determined to be 66.2 Hz. Static seismic analysis.
DG	XEG-1A, B	Diesel Generators and Associated Equipment (2 units)	a	1. Diesel engine and mechanical equipment - analysis 2. Electrical and air starting controls - tests	1. Modal analysis. 2. Multiple frequency, multi-axis testing.

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM (1)	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION (4)	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION (5)	
DG	XHD-13A, B, C, D	Diesel Generator Air Intake Filter Silencer (4 units)	a	Analysis	Natural frequency determined to be 67.1 Hz. Seismic modal analysis.	02-01
DG	XNA-7A, B	Diesel Generator Exhaust Muffler (2 mufflers)	a	Analysis	1. Lowest shell natural frequency determined to be 30 Hz. Static seismic analysis.	02-01
					2. Internals lowest frequency determined to be 4.4 Hz. Seismic modal analysis.	
EF	XPP-8	Turbine Driven Emergency Feedwater Pump	b	1. Pump and Turbine - analysis	1. Natural frequency determined to be 64.7 Hz. Static seismic analysis.	02-01
				2. Pump and Turbine piping appurtenances - test	2. Single sinusoidal test frequency, single axis input. Natural frequency is greater than 30 Hz.	
EF	XPP-21A, B	Motor Driven Emergency Feedwater Pump (2 pumps)	b	1. Pump - analysis	1. Natural frequency determined to be 47.6 Hz. Static seismic analysis.	
				2. Pump piping appurtenances - test	2. Single sinusoidal test frequency, single axis input. Natural frequency is greater than 30 Hz.	

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM [(1)]	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION [(4)]	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION [(5)]
EF	IFV-3531 IFV-3536 IFV-3541 IFV-3546 IFV-3551 IFV-3556	Control Valve (6 Valves)	b	Analysis/Test	Static test of unbraced extended structure (valve bonnet, actuator, snubbers, and appurtenances) resulted in natural frequency of 20 Hz (x-axis) and 21 Hz (y-axis). Multifrequency sweep resulted in choosing 19, 24, 26, and 30 Hz (x-z axis) and 19, 22, and 30 Hz. (y-z axis) for sine beat test. Seismic test was biaxial. See also, Note 3.
FW	XVG-1611A, B, C	Main Feedwater Isolation Valves} (3 valves)	b	1. Valve and actuator - analysis	1. Natural frequency determined to be 57.48 Hz. Static seismic analysis. Natural frequencies in horizontal directions were 22 and 24 Hertz, no resonant frequencies found below 33 Hertz in vertical direction. Multifrequency and sine beat tests performed. Sine beat test performed at 22, 24, and 33 Hertz.
				2. Actuator test	2. Multiple frequency, biaxial seismic test.

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM [1]	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION [4]	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION [5]	
MS	IPV-2000 IPV-2010 IPV-2020	Control Valve (3 valves)	b	Analysis/Test	Static test of unbalanced extended structure (valve bonnet, actuator, snubbers and appurtenances) resulted in natural frequency of 23 Hz. (x-axis) and 21 Hz. (y-axis) Multifrequency sweep resulted in choosing 23, 27.5 and 29 Hz. (x-z axis) and 21, 25, and 29 Hz. (y-z axis) for sine beat test. Seismic test was biaxial. See also, Note 3.	02-01
MS	IFV-2030	Control Valve	b	Analysis/Test	Static test of unbalanced extended structure (valve bonnet, actuator, snubbers and appurtenances) resulted in natural frequency of 26 Hz. (x-axis) and 25 Hz. (y-axis) Multifrequency sweep resulted in choosing 23.5, 26, 29, 33.5 and 40 Hz. (x-z axis) and 25, 29, 33.5 and 40 Hz (y-z axis) for sine beat test. Seismic test was biaxial. See also, Note 3.	02-01

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM [(1)]	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION [(4)]	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION [(5)]
MS	XVM-2801A, B, C	Main Steam Isolation Valve (3 valves)	b	Analysis/Test	Natural frequency determined to be 58.2 Hz. Static seismic analysis. Static seismic load test.
MS	XVS-2806A - XVS-2806N, XVS-2806P	Main Steam Safety Valves (15 valves)	b	Test	Single frequency, single axis seismic test. Natural frequency determined to 37-38 Hz.
RC	XTK-24	Pressurizer Assembly	a	Analysis	Natural frequency determined. Dynamic seismic analysis.
RC	XSG-2A, B, C	Steam Generator (3 units)	a	Analysis	Natural frequency determined. Dynamic seismic analysis.
RH	XPP-31A, B	Residual Heat Removal Pump (2 pumps)	c	Analysis	Natural frequency determined greater than 30 Hz. Static seismic analysis.
RH	XHE-5A, B	Residual Heat Removal Heat Exchanger (2 heat exchangers)	c	Analysis	Natural frequency determined. Dynamic seismic analysis.
SF	XTK-25	Refueling Water Storage Tank	a	Analysis	Hydrodynamic frequency determined to be 0.275 Hz. Seismic modal analysis (TID-25021). Response spectrum input.

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM [1]	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION [4]	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION [5]	
SW	XPP-39A, B, C	Service Water Pump (3 pumps)	a	Analysis	Natural frequency determined to be 14 Hz. Seismic modal analysis. Response spectrum input.	
SW	XPP-45A, B	Service Water Booster Pump (2 pumps)	a	Analysis	Natural frequency determined to be 36.6 Hz. Static seismic analysis.	02-01
VL	XAH-4A, B	RHR/Spray Pump Room Cooling Unit (2 units)	c	(6)	(6)	
VL	XAH-1A, B; XAH-2	Charging Pump Room Cooling Units (3 units)	a	(6)	(6)	02-01
VL	XAH-11A, B	Emergency Feedwater Pump Area Cooling Unit (2 units)	b	(6)	(6)	
VL	XAH-24A, B	Battery Room Air Handling Unit (2 units)	a	(6)	(6)	
VL	XAH-9A, B	Service Water Booster Pump Area Cooling Unit (2 units)	a	(6)	(6)	

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SYSTEM (1)	EQUIP TAG NO.	DESCRIPTION	SAFE SHUTDOWN OPERATION (4)	METHOD OF SEISMIC QUALIFICATION	DESCRIPTION OF SEISMIC QUALIFICATION (5)	
VL	XAH-6, XAH-8	ESF Switchgear Room Cooling Unit (2 units)	a	(6)	(6)	
VL	XAH-19A, B	Speed Switch Room Cooling Unit (2 units)	a	(6)	(6)	
VU	XPP-48A, B, C	HVAC Chilled Water Pump (3 pumps)	a	Analysis	Natural frequency determined to be 37 Hz. Static seismic analysis.	RN 11-018
VU	XHX-1A, B, C	HVAC Mechanical Chillers (3 chillers)	a	Test	Multiple frequency, multi-axis test (1 to 40 Hz. range).	
		Power Actuated Valves (other than control valves and special valves listed above)	a	Analysis/Test	In general, valves (see Table 3.9-8) were seismically qualified by analysis. Natural frequency was determined and a static seismic analysis was performed. Selected valves and/or prototypes were tested by a static seismic load test or by a single frequency, single axis seismic test.	02-01

TABLE 3.9-0 (Continued)

SEISMIC QUALIFICATION OF MECHANICAL EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

NOTES:

1. System:

AH - Air handling (HVAC)	MS - Main Steam
CC - Component Cooling Water	RC - Reactor Coolant
CO - Condensate	RH - Residual Heat Removal
CS - Chemical and Volume Control	SF - Spent Fuel Cooling
DG - Diesel Generator Services	SW - Service Water
EF - Emergency Feedwater	VL - Local Ventilating and Cooling
FW - Feedwater	VU - Chilled Water

02-01

2. Vendor data not yet available - to be provided later.

3. Seismic analysis of each type of control valve. Prototype of each control valve type was seismically tested. Multiple frequency, multi-axis seismic test.

4. Equipment is required to maintain the plant in the following condition:

   a. Hot stand-by and cold shutdown

   b. Hot stand-by

   c. Cold shutdown

5. Types of analysis:

   a. Static

   b. Equivalent static (also known as static coefficient analysis)

   c. Dynamic (also known as seismic modal analysis), input to analysis can be either response spectrum or time-history.

6. Qualified by similarity to XAH-12A, B-AH which was seismically analyzed/tested. XAH-12A, B-AH is the “worst case” for seismic design.

TABLE 3.9-1

BALANCE OF PLANT COMPONENT AND PLANT OPERATING CONDITIONS

Operating Condition or Initiating Event	Plant Condition	System and Component Condition
Startup	Normal	Normal
Standby	Normal	Normal
Part Load	Normal	Normal
Full Load	Normal	Normal
Shutdown	Normal	Normal
Uncontrolled RCC Assembly Withdrawal at Power	Upset	Upset
Uncontrolled RCC Assembly Withdrawal from a Subcritical Condition	Upset	Upset
RCC Assembly Misalignment	Upset	Upset
Chemical and Volume Control System Malfunction	Upset	Upset
Practical Loss of Forced Reactor Coolant Flow	Upset	Upset
Startup of an Inactive Reactor Coolant Loop	Upset	Upset
Loss of External Electrical Load	Upset	Upset

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TABLE 3.9-1 (Continued)

BALANCE OF PLANT COMPONENT AND PLANT OPERATING CONDITIONS

Operating Condition or Initiating Event	Plant Condition	System and Component Condition
Turbine Trip	Upset	Upset
Loss of Normal Feedwater	Upset	Upset
Station Blackout	Upset	Upset
Excessive Heat Removal Due to Feedwater System Malfunction	Upset	Upset
Excessive Load Increase	Upset	Upset
Slow Loss of Reactor Coolant which Actuates Emergency Core Flooding	Emergency	Upset
Minor Secondary System Pipe Break	Emergency	Upset
Loading Fuel Assembly into Improper Position	Emergency	Upset
Complete Loss of Forced Reactor Coolant Flow	Emergency	Upset
Waste Gas Decay Tank Rupture	Emergency	Upset
Loss of Coolant Accident	Faulted	Emergency or Faulted (1)
Rupture of a Steam Pipe	Faulted	Emergency or Faulted (1)

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TABLE 3.9-1 (Continued)

BALANCE OF PLANT COMPONENT AND PLANT OPERATING CONDITIONS

Operating Condition or Initiating Event	Plant Condition	System and Component Condition
Steam Generator Tube Rupture	Faulted	Emergency
Single Reactor Coolant Pump Locked Rotor	Faulted	Emergency
Fuel Handling Accident	Faulted	Emergency
Rupture of a Control Rod Drive Mechanism Housing	Faulted	Emergency

(1) Faulted component conditions have been considered for components in a run of pipe subject to pipe rupture between pipe anchors and/or pipe rupture restraints.

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TABLE 3.9-2

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings
Nonactive ASME Code Class 2 and 3 Valves	Normal	Pr not exceeded, per ANSI B16.5, 1968 (see Note 1 for definition of Pr)	Internal pressure (normal) Deadweight Normal piping loads at valve ends
	Upset	Note 1.a	Internal pressure (upset) Deadweight Upset piping loads at valve ends Operating Basis Earthquake (OBE)
	Emergency	Note 1.b	Internal pressure (normal) Deadweight Emergency piping loads at valve ends Safe Shutdown Earthquake (SSE)
Active ASME Code Class 2 and 3 Valves	Normal	Pr not exceeded, per ANSI, B16.5, 1968	Internal pressure (normal) Deadweight Normal piping loads at valve ends 02-01
	Upset	Note 2.a	Internal pressure (upset) Deadweight Upset I piping loads at valve ends OBE

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TABLE 3.9-2 (Continued)

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings	
Active ASME Code Class 2 and 3 Valves (Cont'd)	Upset	Note 2.b	Internal pressure (upset) Deadweight Upset II piping loads at valve ends	
	Emergency	Note 2.c	Internal pressure (normal) Deadweight Emergency piping loads at valve ends SSE	
	Faulted(i)	Note 2.c	Same as emergency, above, except that faulted piping loads are applied at valve ends	
ASME Code Class 2 and 3 Vessels, Designed in Accordance with Division 1 of Section VIII of the ASME Code	Normal	In accordance with the ASME Code, Section VIII, Division 1	Internal pressure (normal) Nozzle loads from attached piping (normal) Concentrated loads from supports	02-01
	Upset	Notes 3.a and d	Internal pressure (upset) Nozzle loads from attached piping (upset) Concentrated loads from supports OBE	

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TABLE 3.9-2 (Continued)

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings	
ASME Code Class 2 and 3 Vessels, Designed in Accordance with Division 1 of Section VIII of the ASME Code (Cont'd)	Upset	Notes 3.b and d	Internal pressure (Upset) Nozzle loads from attached piping (upset) Concentrated loads from supports	02-01
	Emergency	Notes 3.c and d	Internal pressure (normal) Nozzle loads from attached piping (emergency) Concentrated loads from supports SSE	
ASME Code Class 2 Vessels, Designed in accordance with Division 2 of Section VIII of the ASME Code	Normal	In accordance with the ASME Code, Section VIII, Division 2	Internal pressure (normal) Nozzle loads from attached piping (normal) Concentrated loads from supports	
	Upset	Note 4.a	Internal pressure (upset) Nozzle loads from attached piping (upset) Concentrated loads from supports OBE	

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings
ASME Code Class 2 Vessels, Designed in Accordance with Division 2 of Section VIII of the ASME Code (Cont'd)	Upset	Note 4.b	Internal pressure (Upset) Nozzle loads from attached piping (upset) Concentrated loads from supports
	Emergency	Note 4.c	Internal pressure (Normal) Nozzle loads from attached piping (emergency) Concentrated loads from supports SSE
ASME Code Class 2 and 3 Piping	Normal	In accordance with the ASME Code, Section III, for Normal design conditions	Deadweight Longitudinal pressure stress (normal) Thermal expansion stress Soil loads (buried piping only)
	Upset	Note 5.a	Deadweight Longitudinal pressure stress (upset) Thermal expansion stress OBE (ii) Flow transients (ii) Soil loads and vehicle loads (buried piping only) Safety valve reactions, as applicable (ii)

02-01

TABLE 3.9-2 (Continued)

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings	
ASME Code Class 2 and 3 Piping (Cont'd)	Upset	Notes 5.b and d	Deadweight Longitudinal pressure stress (upset) Flow transients Soil loads and vehicle loads (buried piping only) Safety valve reactions, as applicable (ii) Jet loadings	
	Emergency	Note 5.c	Deadweight Longitudinal pressure stress (normal) Flow transients (ii) Soil loads (buried piping only) Jet loadings SSE (ii)	02-01
	Faulted(iii)	Note 5.c	Deadweight Longitudinal pressure stress (normal) Flow transients (ii) Soil loads (buried piping only) Pipe rupture loads SSE (ii)	02-01
Nonactive ASME Code Class 2 and 3 Pumps	Normal	In accordance with the ASME Code, Section III	Internal pressure (normal) Nozzle loads from attached piping (normal) Concentrated loads from supports	

TABLE 3.9-2 (Continued)

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings
Nonactive ASME Code Class 2 and 3 Pumps (Cont'd)	Upset	Note 6.a	Internal pressure (upset) Nozzle loads from attached piping (upset) Concentrated loads from supports OBE
	Upset	Note 6.b	Internal pressure (upset) Nozzle loads from attached piping (upset) Concentrated loads from supports
	Emergency	Note 6.c	Internal pressure (normal) Nozzle loads from attached piping (emergency) Concentrated loads from supports SSE
Active ASME Code Class 2 and 3 Pumps	Normal	In accordance with the ASME Code, Section III	Internal pressure (normal) Nozzle loads from attached piping (normal) Support loads 02-01
	Upset	Notes 7.a and d	Internal pressure (upset) Nozzle loads from attached piping (upset) Support loads 02-01 OBE

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TABLE 3.9-2 (Continued)

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings	
Active ASME Code Class 2 and 3 Pumps (Cont'd)	Upset	Notes 7.b and d	Internal pressure (upset) Nozzle loads from attached piping (upset) Support loads	02-01
	Emergency	Notes 7.c and d	Internal pressure (normal) Nozzle loads from attached piping (emergency) Support loads SSE	02-01
Code Class 2 and 3 Pipe Supports	Pipe support design conditions are consistent with the design conditions established for the piping	In accordance with the ASME Code, Section III, Subsection NF, Winter 1973 Addenda	Deadweight Superimposed loads and reactions Dynamic loads (seismic, flow transient) Thermal expansion Anchor and support movements Environmental loads	02-01 02-01
Code Class 2 and 3 Supports for Pumps and Vessels	Same as for supported component	In accordance with the ASME Code, Section III, Subsection NF, as applicable	Deadweight Superimposed loads and reactions Dynamic loads (seismic, flow transient) Thermal expansion Anchor and support movements Environmental loads	02-01 02-01

TABLE 3.9-2 (Continued)

BALANCE OF PLANT COMPONENT LOADING CONDITIONS

Component	Component Condition	Design Limits and Loading Combinations	Types of Loadings	
Code Class 2 and 3 Snubbers	Snubber design conditions are consistent with the design conditions established for the piping	In accordance with the ASME Code, Section III, Subsection NF, Winter 1976 Addenda	Dynamic loads (seismic, flow transient)	02-01

NOTES: 02-01

(i) Faulted component condition is specified for active valves in a run of pipe subject to a postulated pipe rupture. Operability has been demonstrated through test and analysis.

(ii) The following loads, if applicable, may be combined on a root mean square basis:

1. Flow transients.

2. OBE or SSE Peak loads.

3. Dynamic portion of safety valve reactions.

(iii) Faulted piping conditions are postulated for portions of piping systems subject to postulated pipe rupture.

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NOTES TO TABLE 3.9-2

1. Nonactive ASME Code Class 2 and 3 Valves

   The design of Code Class 2 and 3 valves encompasses the use of pressure temperature ratings. The design limits are in terms of Pr which is the primary pressure rating corresponding to the maximum transient temperature for each plant condition as specified in Articles NC-3511 and ND-3511 of the ASME Code, Section III, for Code Class 2 and 3 valves, respectively. To assure pressure retaining integrity the limits for Pr are set as follows:

   a. The primary pressure rating, Pr, is not exceeded by more than 10 percent when the component is subject to concurrent loadings associated with either:

      (1) The normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.

      (2) Loading conditions associated with the emergency plant condition.

   b. Pr is not exceeded by more than 20 percent when the component is subject to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

2. Active ASME Code Class 2 and 3 Valves

   To provide pressure retaining integrity and assurance of operability for active valves of Code Class 2 and 3, Pr is not exceeded for the combinations of loading delineated.

   a. The primary pressure rating, Pr, is not exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.

   b. The primary pressure rating, Pr, is not exceeded when the component is subject to loadings associated with the emergency plant condition.

   c. The primary pressure rating, Pr, is not exceeded when the component is subject to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

3. ASME Code Class 2 and 3 Vessels (designed in accordance with Division 1 of Section VIII of the ASME Code) (See Item d, below).

   To provide assurance of pressure retaining integrity for Code Class 2 and 3 vessels (designed in accordance with Division 1 of Section VIII of the ASME Code) the allowable stress value, S, (as specified in Appendix 1 of Section III of the ASME Code) is not exceeded by more than 10 percent when the component is subjected to the loading combinations identified by items a and b, below, and is not exceeded by more than 50 percent when the component is subjected to the loading combinations identified by item c, below.

   a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.

   b. Loadings associated with the emergency plant condition.

   c. The allowable stress value, S is not exceeded by more than 50 percent when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

   d. When a more detailed analysis is performed, Division 1 vessels satisfy, as a minimum, equations (1) and (2), below. Equation (1) is applicable to items a and b, above. Equation (2) is applicable to item c, above.

$$\sigma M < 1.1\ S > \sigma \frac{\sigma_M + \sigma_b}{1.5} \qquad (1)$$

02-01

$$\sigma M < 1.5\ S > \sigma \frac{\sigma_M + \sigma_b}{1.5} \qquad (2)$$

Where:

σ_M = Primary membrane stress.

σ_b = Primary bending stress.

S = Allowable stress value as specified in Appendix 1 of Section III of the ASME Code.

TABLE 3.9-2 (Continued)

4. ASME Code Class 2 Vessels (designed in accordance with Division 2 of Section VIII of the ASME Code.)

   To provide assurance of pressure retaining integrity for Code Class 2 Vessels, (designed in accordance with Division 2 of Section VIII of the ASME Code) the upset, emergency and faulted operating condition category design limits of Article NB-3200 of Section III of the ASME Code are not exceeded when the component is subjected to the following combinations:

   a. The design limits specified in Article NB-3223 of the ASME Code are not exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.

   b. The design limits specified in Article NB-3224 of the ASME Code are not exceeded when the component is subjected to loadings associated with the emergency plant condition.

   c. The design limits specified in Article NB-3225 of the ASME Code are not exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

5. ASME Code Class 2 and 3 Piping

   To provide assurance of pressure retaining integrity for Code Class 2 and 3 piping, the design limits specified in NC-3611.1(b)(4)(c)(b)(1) of the Winter 1972 Addenda to Section III of the ASME Code (i.e., 1.2Sh) are not exceeded when the piping is subjected to the loading combinations identified in items a and b, below. However, for short sections of piping exposed to jet impingement from postulated cracks or breaks in adjacent piping, a stress limit of 1.5Sh may be used.

   The design limits specified in NC-3611.1(b)(4)(c)(b)(2) of the Winter 1972 Addenda to Section III of the ASME Code (i.e. 1.8Sh) are not exceeded when the piping is subjected to the loading combinations identified in item c below. However, for short sections of piping exposed to jet impingement from postulated cracks or breaks in adjacent piping, a stress limit of 2.4Sh may be used.

Whenever the 2.4Sh allowable stress limit is employed for Class 2 and 3 piping, an evaluation of the allowable collapse load will be performed in accordance with Appendix F to the ASME Code, Section III, Winter, 1972, Addenda. It is recognized that neither the "collapse load" nor the "plastic instability load," as defined by the ASME Code, refer to geometrical instability. However, test results are available which do provide a basis for evaluating geometrical stability of fittings (elbows and tees), as well as straight pipe, at stress levels associated with the ASME Code "collapse" or "plastic instability" load limit. Additionally, test results reveal that, for thickness to radius ratios (t/r) less than 0.08, the mode of collapse is not in the form of geometrical instability at the fitting or discontinuity. The piping systems evaluated generally have t/r ratios less than 0.08.

a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.

b. Loadings associated with the emergency plant conditions (see Note 1).

c. Concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

Thermal expansion effects of piping are not evaluated for loadings associated with emergency or faulted plant conditions. Therefore, only Equation 9 of Article NC-3651 of Section III of the ASME Code is applied for the loading combinations identified in items b and c, above. Thermal expansion effects are evaluated for the loading combinations identified in item a, above.

6. Nonactive ASME Code Class 2 and 3 pumps

In order to assure pressure retaining integrity for nonactive Code Class 2 and 3 pumps, the primary membrane stress is not exceeded by more than 10 percent of S (as specified in Appendix 1 of Section III of the ASME Code) and the sum of the primary membrane plus primary bending stresses is not exceeded by more than 65 percent of S when the component is subjected to the following load combinations:

a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.

b. Loadings associated with the emergency plant condition.

c. In addition, the primary membrane stress is not exceeded by more than 20 percent of S, and the sum of the primary membrane and primary bending stresses is not exceeded by more than 80 percent of S when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

7. Active ASME Code Class 2 and 3 Pumps

To provide increased assurance that unacceptable deformations affecting operability of active Code Class 2 and 3 pumps do not result, the primary membrane stress does not exceed S (as specified in Appendix 1 of Section III of the ASME Code) and the sum of the primary membrane plus primary bending stresses is not exceeded by more than 50 percent of S when the component is subjected to the following loading combinations: (See item d, below).

a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of SSE.

b. Loadings associated with the emergency plant condition.

c. Concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

d. The design limits given below are not exceeded for the applicable loading combinations. Analysis and/or testing confirms that operability is not impaired when the component is designed to these limits.

The primary membrane stress is not exceeded by more than 10 percent of S and the sum of the primary membrane plus primary bending stresses is not exceeded by more than 65 percent of S when the component is subjected to the following combinations:

(1) Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50 percent of the SSE.

(2) Loadings associated with the emergency plant condition.

The primary membrane stress is not exceeded by more than 20 percent of S and the sum of the primary membrane and primary bending stresses is not exceeded by more than 80 percent of S which the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the SSE and the dynamic system loadings associated with the faulted plant condition.

TABLE 3.9-3

DESIGN LOADING COMBINATIONS FOR NSSS SUPPLIED
ASME CODE CLASS 2 AND 3 COMPONENTS AND SUPPORTS

Conditions Classification	Loading Combination
Design and Normal	Design pressure Design temperature, [1] Dead weight, nozzle loads [2]
Upset	Upset condition pressure, Upset condition metal temperature, [1] deadweight, OBE, nozzle loads [2]
Faulted	Faulted condition pressure, faulted condition metal temperature, [1] deadweight, SSE, nozzle loads [2]

(1) Temperature is used to determine allowable stress only.

(2) Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

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TABLE 3.9-4

STRESS CRITERIA FOR NSSS SUPPLIED SAFETY-RELATED ASME CLASS 2 AND CLASS 3 TANKS

Condition	Stress Limits
Design and Normal	The vessel shall conform to the requirements of ASME Section VIII, Division 1.
Upset	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S
Faulted	$\sigma_m \leq 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S

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TABLE 3.9-5

DESIGN CRITERIA FOR NSSS SUPPLIED PUMPS

Condition	Design Criteria
Design and Normal	ASME Section III Subsection NC-3400 and ND-3400
Upset	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S
Faulted	$\sigma m \leq 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S

TABLE 3.9-6

STRESS CRITERIA FOR NSSS SUPPLIED SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VALVES

Condition	Stress Limits (Notes 1-5)	P_{max} (Note 6)
Design and Normal	Valve bodies shall conform to the requirements of ASME Section III, NC-3500 (or ND-3500)	
Upset	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1
Faulted	$\sigma m \leq 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S	1.5

RN 01-113

NOTES:

1. Stress analysis is not required when both the following conditions are satisfied: (1) section modulus and area of every plane, normal to the flow, through the region defined as the valve body crotch is at least 110% of those for the piping connected (or joined) to the valve body inlet and outlet nozzles; and, (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the acceptance criteria ratio shall be 110% multiplied by the ratio of S_{pipe}/S_{valve}. If unable to comply with this requirement, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

TABLE 3.9-6 (Continued)

NOTES (Continued)

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

5. These rules do not apply to Class 2 and 3 relief valves. Relief valves are designed in accordance with the ASME Code, Section III requirements.

6. The maximum pressure resulting from upset or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.

TABLE 3.9-7

ACTIVE CODE CLASS 2 AND 3 PUMPS

Balance of Plant Pumps

Pump Tag Number	System (1)	Location (2)	Safety Class	FSAR Figure	Method of Qualification (3)	
						RN 01-113
XPP0001A	CC	IB	2b	9.2-4	A&T	
XPP0001B	CC	IB	2b	9.2-4	A&T	
XPP0001C	CC	IB	2b	9.2-4	A&T	
XPP0004A	DG	DB	2b	9.5-2	A	
XPP0004B	DG	DB	2b	9.5-2	A	
XPP0141A	DG	DB	2b	9.5-2	A	
XPP0141B	DG	DB	2b	9.5-2	A	
XPP0008	EF	IB	2b	10.4-16	A&T	
XPP0021A	EF	IB	2b	10.4-16	A&T	
XPP0021B	EF	IB	2b	10.4-16	A&T	
XPP0038A	SP	AB	2a	6.2-46	A	
XPP0038B	SP	AB	2a	6.2-46	A	
XPP0039A	SW	SW	2b	9.2-1	A	RN 01-113
XPP0039B	SW	SW	2b	9.2-1	A	
XPP0039C	SW	SW	2b	9.2-1	A	
XPP0045A	SW	IB	2b	9.2-1	A	
XPP0045B	SW	IB	2b	9.2-1	A	
XPP0048A	VU	IB	2b	9.4-22	A	
XPP0048B	VU	IB	2b	9.4-22	A	
XPP0048C	VU	IB	2b	9.4-22	A	
NSSS Pumps						
Charging Pumps						
XPP0043A	CS	AB	2a	6.3-1	A	
XPP0043B	CS	AB	2a	6.3-1	A	
XPP0043C	CS	AB	2a	6.3-1	A	
RHR Pumps						
XXP0031A	RH	AB	2a	6.3-1	A	RN 01-113
XXP0031B	RH	AB	2a	6.3-1	A	RN 11-027 RN 16-018

TABLE 3.9-7 (Continued)

<u>NOTES</u>:

1. Systems are as follows:

   CC - Component Cooling Water
   CS - Chemical and Volume Control
   DG - Diesel Generator
   EF - Emergency Feedwater
   RH - Residual Heat Removal
   SP - Reactor Building Spray
   SW - Service Water
   VU - HVAC Chilled Water

2. Locations are as follows:

   IB - Intermediate Building
   DB - Diesel Generating Building
   AB - Auxiliary Building
   SW - Service Water Pumphouse

3. Methods of qualification are as follows:

   A - Analysis
   T - Test

RN 11-027

RN 11-027

RN 01-113

RN 11-027
RN 16-018

TABLE 3.9-8

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]	
XVG00265-AS	AS	IB	6	3	N/A	S	A	RN 04-008 17-029
XVG00273-AS	AS	IB	6	3	N/A	S	A	
XVG0503A	BD	AB	3	2	10.4-13	AO	A	
XVG0503B	BD	AB	3	2	10.4-13	AO	A	
XVG0503C	BD	AB	3	2	10.4-13	AO	A	
XVB9503A	CC	AB	20	3	9.2-4	MO	A	
XVB9503B	CC	AB	20	3	9.2-4	MO	A	
XVB9524A	CC	AB	16	3	9.2-4	MO	A	
XVB9524B	CC	AB	16	3	9.2-4	MO	A	
XVB9525A	CC	AB	16	3	9.2-4	MO	A	
XVB9525B	CC	AB	16	3	9.2-4	MO	A	
XVB9526A	CC	IB	16	3	9.2-4	MO	A	
XVB9526B	CC	IB	16	3	9.2-4	MO	A	
XVG9568	CC	AB	8	2	9.2-5	MO	A	
XVC9570	CC	RB	8	2	9.2-5	N	A	
XVG9600	CC	IB	3	2	9.2-5	MO	A	
XVC9602	CC	RB	3	2	9.2-5	N	A	98-01
XVG9605	CC	RB	8	2	9.2-5	MO	A	
XVG9606	CC	AB	8	2	9.2-5	MO	A	
XVG9625	CC	AB	8	3	9.2-5	MO	A	
XVG9626	CC	AB	8	3	9.2-5	MO	A	
XVG9627A	CC	IB	4	3	9.2-4	AO	A	
XVG9627B	CC	IB	4	3	9.2-4	AO	A	98-01
XVG9684A	CC	AB	2	3	9.2-7	AO	A	
XVG9684B	CC	AB	2	3	9.2-7	AO	A	
XVG9684C	CC	AB	2	3	9.2-7	AO	A	
XVC9632	CC	AB	8	3	9.2-5	N	A	
XVC9680A	CC	IB	4	3	9.2-4	N	A	
XVC9680B	CC	IB	4	3	9.2-4	N	A	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]
XVC9633	CC	AB	8	3	9.2-5	N	A
XVC9682A	CC	IB	24	3	9.2-4	N	A
XVC9682B	CC	IB	24	3	9.2-4	N	A
XVC9682C	CC	IB	24	3	9.2-4	N	A
XVB9687A	CC	IB	16	3	9.2-4	MO	A
XVB9687B	CC	IB	16	3	9.2-4	MO	A
XVC970A,B	DG	DB	2	3	9.5-2	N	A
XVC971A,D	DG	YD	3	3	9.5-2	N	A
XVC972A,B	DG	DB	2	3	9.5-2	N	A
XVC10977A,B	DG	DB	3/4	3	9.5-6	N	A
XVC10978A,B	DG	DB	3/4	3	9.5-6	N	A
XVG1001A	EF	IB	6	3	10.4-16	MO	A
XVG1001B	EF	IB	6	3	10.4-16	MO	A
XVG1002	EF	IB	8	3	10.4-16	MO	A
XVG1008	EF	IB	8	3	10.4-16	MO	A
XVC1009A	EF	PR	4	2	10.4-16	AO	A
XVC1009B	EF	PR	4	2	10.4-16	AO	A
XVC1009C	EF	PR	4	2	10.4-16	AO	A
XVC1013A	EF	IB	6	3	10.4-16	N	A
XVC1013B	EF	IB	6	3	10.4-16	N	A
XVC1014	EF	IB	8	3	10.4-16	N	A
XVC01048A	EF	IB	4	3	10.4-16	N	A
XVC01048B	EF	IB	4	3	10.4-16	N	A
XVC1016	EF	IB	4	3	10.4-16	N	A
XVK1019A	EF	IB	4	2	10.4-16	N	A
XVK1019B	EF	IB	4	2	10.4-16	N	A
XVK1019C	EF	IB	4	2	10.4-16	N	A
XVK1020A	EF	IB	4	2	10.4-16	N	A
XVK1020B	EF	IB	4	2	10.4-16	N	A
XVK1020C	EF	IB	4	2	10.4-16	N	A
XVC1022A	EF	IB	8	3	10.4-16	N	A
XVC1022B	EF	IB	8	3	10.4-16	N	A

99-01

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]	
XVC1023A	EF	IB	2	3	10.4-16	N	A	
XVC1023B	EF	IB	2	3	10.4-16	N	A	
XVC1024	EF	IB	3	3	10.4-16	N	A	
XVC1034A	EF	IB	6	3	10.4-16	N	A	
XVC1034B	EF	IB	6	3	10.4-16	N	A	
XVG1037A	EF	IB	8	3	10.4-16	MO	A	A 98-01
XVG1037B	EF	IB	8	3	10.4-16	MO	A	RN
IFV3531	EF	IB	4	2	10.4-16	AO	A & T	14-029
IFV3536	EF	IB	4	2	10.4-16	AO	A & T	
IFV3541	EF	IB	4	2	10.4-16	AO	A & T	
IFV3546	EF	IB	4	2	10.4-16	AO	A & T	
IFV3551	EF	IB	4	2	10.4-16	AO	A & T	
IFV3556	EF	IB	4	2	10.4-16	AO	A & T	RN
XVM01072A	EF	IB	4	3	10.4-16	N	A	96-041
XVM01072B	EF	IB	4	3	10.4-16	N	A	12-030
XVG6797	FS	PR	4	2	9.5-1,Sht.3	MO	A	
XVG1611A	FW	PR	18	2	10.4-12	AO	A	A 02-01
XVG1611B	FW	PR	18	2	10.4-12	AO	A	
XVG1611C	FW	PR	18	2	10.4-12	AO	A	
XVK1633A	FW	PR	1-1/2	2	10.4-12	MO	A	
XVK1633B	FW	PR	1-1/2	2	10.4-12	MO	A	
XVK1633C	FW	PR	1-1/2	2	10.4-12	MO	A	A 02-01
XVC1684A	FW	PR	18	2	10.4-12	N	A & T	
XVC1684B	FW	PR	18	2	10.4-12	N	A & T	
XVC1684C	FW	PR	18	2	10.4-12	N	A & T	
XVX6050A	HR	RB	3/8	2	6.2-58	S	T	
XVX6050B	HR	RB	3/8	2	6.2-58	S	T	
XVX6051A	HR	RB	3/8	2	6.2-58	S	T	
XVX6051B	HR	RB	3/8	2	6.2-58	S	T	
XVX6051C	HR	RB	3/8	2	6.2-58	S	T	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]	
XVX6052A	HR	PR	3/8	2	6.2-58	S	T	
XVX6052B	HR	FH	3/8	2	6.2-58	S	T	
XVX6053A	HR	PR	3/8	2	6.2-58	S	T	
XVX6053B	HR	FH	3/8	2	6.2-58	S	T	
XVX6054	HR	PR	3/8	2	6.2-58	S	T	
XVG6056	HR	RB	6	2	6.2-58	AO	A	
XVG6057	HR	FH	6	2	6.2-58	AO	A	
XVG6066	HR	RB	6	2	6.2-58	AO	A	
XVG6067	HR	PR	6	2	6.2-58	AO	A	02-01
XVT2660	IA	AB	2	2	9.3-3	AO	T	
XVT2661	IA	RB	2	2	9.3-3	N	T	
XVT2662A	IA	AB	6	2	9.3-3	AO	T	
XVT2662B	IA	RB	6	2	9.3-3	AO	A	
IPV2000	MS	IB	8	2	10.3-1	AO	A & T	
IPV2010	MS	IB	8	2	10.3-1	AO	A & T	
IPV2020	MS	IB	8	2	10.3-1	AO	A & T	
IFV2030	MS	IB	4	3	10.3-1	AO	A & T	
XVM2801A	MS	IB	32	2	10.3-1	AO	A & T	
XVM2801B	MS	IB	32	2	10.3-1	AO	A & T	
XVM2801C	MS	IB	32	2	10.3-1	AO	A & T	
XVG2802A	MS	IB	4	2	10.3-1	MO	A	
XVG2802B	MS	IB	4	2	10.3-1	MO	A	
XVS2806A	MS	IB	6x10	2	10.3-1	N	A	
XVS2806B	MS	IB	6x10	2	10.3-1	N	A	
XVS2806C	MS	IB	6x10	2	10.3-1	N	A	
XVS2806D	MS	IB	6x10	2	10.3-1	N	A	
XVS2806E	MS	IB	6x10	2	10.3-1	N	A	98-01
XVS2806F	MS	IB	6x10	2	10.3-1	N	A	
XVS2806G	MS	IB	6x10	2	10.3-1	N	A	
XVS2806H	MS	IB	6x10	2	10.3-1	N	A	
XVS2806I	MS	IB	6x10	2	10.3-1	N	A	
XVS2806J	MS	IB	6x10	2	10.3-1	N	A	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]	
XVS2806K	MS	IB	6x10	2	10.3-1	N	A	98-01
XVS2806L	MS	IB	6x10	2	10.3-1	N	A	
XVS2806M	MS	IB	6x10	2	10.3-1	N	A	
XVS2806N	MS	IB	6x10	2	10.3-1	N	A	
XVS2806P	MS	IB	6x10	2	10.3-1	N	A	
XVT2843A	MS	IB	1-1/2	2	10.3-1	AO	A & T	
XVT2843B	MS	IB	1-1/2	2	10.3-1	AO	A & T	
XVT2843C	MS	IB	1-1/2	2	10.3-1	AO	A & T	
XVT2877A	MS	PR	1-1/2	2	10.3-1	AO	A & T	
XVT2877B	MS	PR	1-1/2	2	10.3-1	AO	A & T	
XVT2869A	MS	IB	4	2	10.3-1	AO	T	
XVT2869B	MS	IB	4	2	10.3-1	AO	T	
XVT2869C	MS	IB	4	2	10.3-1	AO	T	
XVC2876A	MS	IB	4	3	10.3-1	N	A	
XVC2876B	MS	IB	4	3	10.3-1	N	A	
XVD6242A	ND	RB	3	2	9.3-12	AO	T	
XVD6242B	ND	AB	3	2	9.3-12	AO	T	
XVG6701	SF	YD	3	2	9.1-3	N	A	RN 15-007
XVG3001A	SP	AB	12	2	6.2-46	MO	A	
XVG3001B	SP	AB	12	2	6.2-46	MO	A	
XVG3002A	SP	AB	3	3	6.2-46	MO	A	
XVG3002B	SP	AB	3	3	6.2-46	MO	A	
XVG3003A	SP	PR	10	2	6.2-46	MO	T	
XVG3003B	SP	PR	10	2	6.2-46	MO	A	
XVG3004A	SP	AB	12	2	6.2-46	MO	A	
XVG3004B	SP	AB	12	2	6.2-46	MO	A	
XVG3005A	SP	AB	12	2	6.2-46	MO	A	
XVG3005B	SP	AB	12	2	6.2-46	MO	A	
XVC3006A	SP	AB	12	2	6.2-46	N	A	
XVC3006B	SP	AB	12	2	6.2-46	N	A	
XVC3009A	SP	RB	10	2	6.2-46	N	A	
XVC3009B	SP	RB	10	2	6.2-46	N	A	
XVC3013A	SP	AB	3	2	6.2-46	N	A	
XVC3013B	SP	AB	3	2	6.2-46	N	A	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]	
XVA9311A	SS	RB	1	2	9.3-4	AO	A	
XVA9311B	SS	PR	1	2	9.3-4	AO	A	
XVA9312A	SS	RB	1	2	9.3-4	AO	A	
XVA9312B	SS	PR	1	2	9.3-4	AO	A	
XVT9339	SS	RB	3/8	2	9.3-20	S	A	
XVT9341	SS	AB	3/8	2	9.3-20	S	A	
XVX9356A	SS	RB	3/8	2	9.3-4	S	A & T	
XVX9356B	SS	RB	3/8	2	9.3-4	S	A & T	02-01
XVX9357	SS	AB	3/8	2	9.3-4	S	A & T	
XVV3014A	SP	AB	2x2	3	6.2-46	N	T	
XVV3014B	SP	AB	2x2	3	6.2-46	N	T	
XVX9364B	SS	RB	3/8	2	9.3-4	S	A & T	
XVX9364C	SS	RB	3/8	2	9.3-4	S	A & T	
XVX9365B	SS	AB	3/8	2	9.3-4	S	A & T	
XVX9365C	SS	IB	3/8	2	9.3-4	S	A & T	02-01
XVX9387	SS	AB	3/8	2	9.3-4	S	A & T	
XVX9398A	SS	AB	1	2	9.3-4	S	A & T	02-01
XVX9398B	SS	IB	1	2	9.3-4	S	A & T	
XVX9398C	SS	IB	1	2	9.3-4	S	A & T	
XVG3103A	SW	PR	16	2	9.2-2 SH. 2	MO	A	
XVG3103B	SW	IB	16	2	9.2-2 SH. 4	MO	A	
XVB3106A	SW	PR	16	2	9.2-2 SH. 2	MO	A	
XVB3106B	SW	IB	16	2	9.2-2 SH. 4	MO	A	
XVB3107A	SW	PR	16	3	9.2-2 SH. 2	AO	A	RN
XVB3107B	SW	FH	16	3	9.2-2 SH. 4	AO	A	08-008
XVG3109A	SW	RB	10	2	9.2-2 SH. 2	MO	A	
XVG3109B	SW	RB	10	2	9.2-2 SH. 2	MO	A	RN
XVG3109C	SW	RB	10	2	9.2-2 SH. 4	MO	A	09-002
XVG3109D	SW	RB	10	2	9.2-2 SH. 4	MO	A	
XVB3110A	SW	PR	12	2	9.2-2 SH. 2	MO	A	
XVB3110B	SW	IB	12	2	9.2-2 SH. 4	MO	A	
XVG3111A	SW	PR	12	3	9.2-2 SH. 2	MO	A	
XVG3111B	SW	IB	12	3	9.2-2 SH. 4	MO	A	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]	
XVG3112A	SW	PR	12	3	9.2-2 SH. 2	MO	A	RN 09-002
XVG3112B	SW	IB	12	3	9.2-2 SH. 4	MO	A	
XVC3115A	SW	SW	24	3	9.2-1	N	A	
XVC3115B	SW	SW	24	3	9.2-1	N	A	
XVC3115C	SW	SW	24	3	9.2-1	N	A	
XVB3116A	SW	SW	24	3	9.2-1	MO	A	
XVB3116B	SW	SW	24	3	9.2-1	MO	A	
XVB3116C	SW	SW	24	3	9.2-1	MO	A	
XVC3119A	SW	SW	8	3	9.2-2 SH. 1	N	A	
XVC3119B	SW	SW	8	3	9.2-2 SH. 3	N	A	
XVB3126A	SW	IB	6	3	9.2-2 SH. 1	MO	T	
XVB3126B	SW	IB	6	3	9.2-2 SH. 3	MO	T	
XVB3128A	SW	IB	6	3	9.2-2 SH. 1	MO	T	
XVB3128C	SW	IB	6	3	9.2-2 SH. 1	MO	T	
XVC3130A	SW	YD	30	3	9.2-2 SH. 1	N	A	RN 09-002
XVC3130B	SW	YD	30	3	9.2-2 SH. 3	N	A	
XVC3135A	SW	IB	16	3	9.2-2 SH. 2	N	A	
XVC3135B	SW	IB	16	3	9.2-2 SH. 4	N	A	
XVC3136A	SW	PR	12	3	9.2-2 SH. 2	N	A	
XVC3136B	SW	FH	12	3	9.2-2 SH. 4	N	A	
XVC3137A	SW	RB	16	2	9.2-2 SH. 2	N	A	
XVC3137B	SW	RB	16	2	9.2-2 SH. 4	N	A	RN 04-039
XVT3164	SW	RB	2	3	9.2-2 SH. 2	AO	A	
XVT3165	SW	RB	2	3	9.2-2 SH. 2	AO	A	RN 09-002
XVC3168	SW	RB	2	3	9.2-2 SH. 2	N	A	
XVT3169	SW	RB	2	2	9.2-2 SH. 2	AO	A	
XVV13143A	SW	PR	6	3	9.2-2 SH. 2	N	A	RN 13-032
XVV13143B	SW	FH	6	3	9.2-2 SH. 4	N	A	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

Balance of Plant Valves

99-01

Valve Tag Number [1]	System [2]	Building [3]	Size (in)	Code Class	FSAR Figure	Operator Type [4]	Method of Qualification [5]
XVC6461A	VU	IB	6	2	9.4-22	N	A
XVC6461B	VU	IB	6	2	9.4-22	N	A
XVC6461C	VU	IB	6	2	9.4-22	N	A
XVT8095A,B	RC	RB	2	1	5.1-1	M	A
XVT8096A,B	RC	RB	2	1	5.1-1	M	A
XVB-1A	AH	FH	36	2	9.4-28	AO	T
XVB-1B	AH	RB	36	2	9.4-28	AO	T
XVB-2A	AH	FH	36	2	9.4-28	AO	T
XVB-2B	AH	RB	36	2	9.4-28	AO	T

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

NSSS Valves

Valve Location Number	System	Building	Size (in) [6]	Safety Class	Figure No.	Actuator	Type	Method of Qualification [5]
1-8381	CS	RB	3	2a	9.3-16	---	Check	A *
1-8152	CS	AB	3	2a	9.3-16	Air	Globe	A *
1-8112	CS	RB	2	2a	9.3-16	Motor	Globe	A *
1-8100	CS	AB	2	2a	9.3-16	Motor	Globe	A *
1-8149A,B,C	CS	RB	2	2a	9.3-16	Air	Globe	A *
1-8154	CS	RB	1	1	9.3-16	Air	Globe	A *
1-8153	CS	RB	1	1	9.3-16	Air	Globe	A *
1-LCV-460	CS	RB	3	1	9.3-16	Air	Globe	A *
1-LCV-459	CS	RB	3	1	9.3-16	Air	Globe	A *
1-8379	CS	RB	3	1	9.3-16	---	Check	A *
1-8346	CS	RB	3	1	9.3-16	---	Check	A *
1-8378	CS	RB	3	1	9.3-16	---	Check	A *
1-8347	CS	RB	3	1	9.3-16	---	Check	A *
1-8377	CS	RB	2	1	9.3-16	---	Check	A
1-8367A,B,C	CS	RB	1-1/2	1	9.3-16	---	Check	A
1-8348A,B,C	CS	RB	1-1/2	1	9.3-16	---	Check	A
1-8368A,B,C	CS	RB	1-1/2	2a	9.3-16	---	Check	T
1-8145	CS	RB	2	1	9.3-16	Air	Globe	A *
1-8102A,B,C	CS	PR	1-1/2	2a	9.3-16	Motor	Globe	A *
1-LCV-115C,E	CS	AB	4	2a	9.3-16	Motor	Gate	A *
1-LCV-115B,D	CS	AB	8	2a	9.3-16	Motor	Gate	A *
1-8104	CS	AB	2	2a	9.3-16	Motor	Globe	A *
1-8480A,B,C	CS	AB	2	2a	9.3-16	---	Check	A
1-8442	CS	AB	2	2a	9.3-16	---	Check	A
1-8130A,B	CS	AB	8	2a	9.3-16	Motor	Gate	A
1-8131A,B	CS	AB	8	2a	9.3-16	Motor	Gate	A
1-8109A,B,C	CS	AB	2	2a	9.3-16	Motor	Globe	A
1-8481A,B,C	CS	AB	3	2a	9.3-16	---	Check	A
1-8132A,B	CS	AB	4	2a	9.3-16	Motor	Gate	A
1-8133A,B	CS	AB	4	2a	9.3-16	Motor	Gate	A

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

NSSS Valves

Valve Location Number	System	Building	Size (in) [6]	Safety Class	Figure No.	Actuator	Type	Method of Qualification [5]	
1-8106	CS	AB	3	2a	9.3-16	Motor	Gate	A	
1-8107	CS	AB	3	2a	9.3-16	Motor	Gate	A	
1-8108	CS	AB	3	2a	9.3-16	Motor	Gate	A	
1-8117	CS	RB	2	2a	9.3-16	---	Relief	A	
1-8314A,B	CS	AB	2	2b	9.3-16	---	Check	A	00-01
XVC18529	CS	AB	1-1/2	2a	9.3-16	---	Check	A	RN
1-8010A,B,C	RCS	RB	6	1	5.1-1	---	Relief	A	12-004
1-8047	RCS	RB	1	2a	5.1-1	Air	Diaphragm	A	
1-8033	RCS	AB	1	2a	5.1-1	Air	Diaphragm	A	
1-8046	RCS	RB	3	2a	5.1-1	---	Check	A	
1-8028	RCS	AB	3	2a	5.1-1	Air	Diaphragm	A	
1-8716A,B	RHR	AB	10	2a	5.5-4	---	Check	A	
1-FCV-602A,B	RHR	AB	3	2a	5.5-4	Motor	Gate	A	99-01
1-8706A,B	RHR	AB	8	2a	5.5-4	Motor	Gate	A	
1-8708A,B	RHR	RB	3	2a	5.5-4	---	Relief	A	
1-8701A,B	RHR	RB	12	1	5.5-4	Motor	Gate	A	
1-8702A,B	RHR	RB	12	1	5.5-4	Motor	Gate	A	
1-8993A,B,C	SI	RB	6	1	6.3-1	---	Check	A	
1-8992A,B,C	SI	RB	2	1	6.3-1	---	Check	A	
1-8990A,B,C	SI	RB	2	1	6.3-1	---	Check	A	
1-8988A,B	SI	RB	6	1	6.3-1	---	Check	A	
1-8884	SI	PR	3	2a	6.3-1	Motor	Gate	A	
1-8885	SI	PR	3	2a	6.3-1	Motor	Gate	A	99-01
1-8886	SI	PR	3	2a	6.3-1	Motor	Gate	A	
1-8995A,B,C	SI	RB	2	1	6.3-1	---	Check	A	
1-8997A,B,C	SI	RB	2	1	6.3-1	---	Check	A	
1-8998A,B,C	SI	RB	6	1	6.3-1	---	Check	A	
1-8801A,B	SI	FH	3	2a	6.3-1	Motor	Gate	A	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

NSSS Valves

Valve Location Number	System	Building	Size (in) [6]	Safety Class	Figure No.	Actuator	Type	Method of Qualification [5]	
1-8947	SI	RB	1	2a	6.3-1	---	Check	T	
1-8880	SI	PR	1	2a	6.3-1	Air	Globe	A	
1-8956A,B,C	SI	RB	12	1	6.3-1	---	Check	A	
1-8948A,B,C	SI	RB	12	1	6.3-1	---	Check	A	
1-8961	SI	PR	3/4	2a	6.3-1	Air	Globe	A	99-01
1-8871	SI	RB	3/4	2a	6.3-1	Air	Globe	A	
1-8926	SI	AB	8	2a	6.3-1	---	Check	A	
1-8958A,B	SI	AB	14	2a	6.3-1	---	Check	A	
1-8809A,B	SI	AB	14	2a	6.3-1	Motor	Gate	A	
1-8811A,B	SI	AB	14	2a	6.3-1	Motor	Gate	A	
1-8812A,B	SI	AB	14	2a	6.3-1	Motor	Gate	A	
1-8887A,B	SI	PR	10	2a	6.3-1	Motor	Gate	A	99-01
1-8861	SI	RB	1	2a	6.3-1	---	Check	A	
1-8860	SI	PR	1	2a	6.3-1	Air	Globe	A	
1-8889	SI	PR	10	2a	6.3-1	Motor	Gate	A	99-01
1-8888A,B	SI	PR	10	2a	6.3-1	Motor	Gate	A	
1-8974A,B	SI	RB	10	2a	6.3-1	---	Check	A	
1-8973A,B,C	SI	RB	6	1	6.3-1	---	Check	A	
1-7136	WL	AB	3	2a	11.2-2	Air	Diaphragm	A	
1-7170	WL	RB	3	2a	11.2-2	Air	Diaphragm	A	
1-7150	WL	AB	3/4	2a	11.2-2	Air	Diaphragm	A	
1-7126	WL	RB	3/4	2a	11.2-2	Air	Diaphragm	A	

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

NOTES TO TABLE 3.9-8

[1] Valve types are as follows:

a. Control Valves

IFV - Flow Control
ILV - Level Control
IPV - Pressure Control
ITV - Temperature Control
IVV - Vapor Vent Control

b. Noncontrol Valves

XVA - Ball
XVB - Butterfly
XVC - Check
XVD - Diaphragm
XVG - Gate
XVK - Stop/Check
XVM - Special
XVN - Needle
XVP - Plug
XVR - Relief
XVS - Safety
XVT - Globe
XVV - Vacuum Breaker
XVX - Solenoid Valve

[2] Systems are as follows:

AS - Auxiliary Steam RN 17-029
BD - Steam Generator Blowdown
CC - Component Cooling Water
CS - Chemical and Volume Control System
DG - Diesel Generator
EF - Emergency Feedwater
FS - Fire Service
FW - Feedwater
HR - Post Accident Hydrogen Removal
IA - Instrument Air
MS - Main Steam
ND - Nuclear Drain
NG - Nitrogen Blanket
RCS - Reactor Coolant
RHR - Residual Heat Removal
SA - Station Service and Instrument Air
SF - Spent Fuel Cooling RN 15-007
SI - Safety Injection

TABLE 3.9-8 (Continued)

ACTIVE CODE CLASS 1, 2, AND 3 VALVES

NOTES TO TABLE 3.9-8

[2] Systems are as follows: (Continued)

SP - Reactor Building Spray
SS - Nuclear Sampling
SW - Service Water
VU - HVAC Chilled Water
WL - Liquid Waste
AH - Air Handling

[3] Buildings are as follows:

AB - Auxiliary Building
IB - Intermediate Building
RB - Reactor Building
PR - Penetration Access Area
CB - Control Building
DB - Diesel Generator Building
FH - Fuel Handling Building
YD - Yard Area

[4] Operator types are as follows:

AO - Air Operated
MO - Motor Operated
PH - Pneumatic Hydraulic
S - Solenoid
N - No External Operator

[5] Methods of qualification are as follows:

A - Analysis
T - Test

[6] The 2 inch and under check valves shown in the CS and SI systems were bought to meet NSSS design criteria but were not bought as part of the NSSS package.

TABLE 3.9-9

FACTORS OF SAFETY AND BOLT STRESSES FOR LINEAR COMPONENT SUPPORTS

Connection Plate	Bolt Size & Type	Minimum Factor of Safety vs. Failure	Max. Axial Stress in Bolt	Bolt Capacity Criteria
Group I Modular Embedment	7/8” ϕ x 8-3/16 Nelson Headed Stud	4.2	7.4 ksi	Nelson Catalog
Group II Random Embedment	7/8” ϕ x 8-3/16 Nelson Headed Stud	2.5	11 ksi	Nelson Catalog
Group III Typical Expansion Bolt Anchored Plates	5/8” ϕ x 6 Hilti 3/4” ϕ x 7Hilti 1” ϕ x 9 Hilti 1-1/4” ϕ x 12 Hilti	2.98 [1]	Not determined	Hilti Catalog

NOTE:

[1] Of the 94 expansion bolt anchorages investigated, 84 had factors of safety in excess of 4, and none had a factor of safety less than 2.98. Calculations indicate that a factor of safety of 4 is not demonstrated for all cases. This is generally attributed to the difference in methodology of the analysis described in Figures 3.9-5 through 3.9-8, with respect to the original analysis. The existence of prying force did contribute to the reduction in factor of safety in two cases. In either case the factors of safety less than 4 were determined to exist only under upset or faulted load combinations.

TABLE 3.9-10

KWIK BOLT TESTS

Kwik-Bolt Size Dia. In.	Kwik-Bolt Size Length In.	Kwik-Bolt Embed-ment In.	Initial Torque Ft-Lbs	Turns of Anchor Nut	Time of Day at Instal-lation Day 2	Time of Day at Test-ing Day 2	Torque Reading at Testing Ft-Lbs	Torque Loss Ft-Lbs	Load at Initial Displacement Gage Read. psi	Load at Initial Displacement Force lbs	Subsequent Torques Applied Ft-Lbs	Ultimate Load Gage Read. psi	Ultimate Load Force lbs	Mode of Failure
5/8	8 1/2	7 1/2	70	1/2	8:30am	2:00pm	70	0	400	2668		1400	9415	Wedges pulled over end of anchor
									500	3335	100			(Anchor failure)
									850	5670	140			
5/8	8 1/2	7 1/2	70	1	8:30am	3:00pm	65	5	600	3335		1800	12000	Wedges pulled over end of anchor
									1000	6670	120			(Anchor failure)
5/8	8 1/2	7 1/2	70	1/2	8:30am	3:20pm	60	10	300	2001		1500	10125	Wedges pulled over end of anchor
									700	4669	100			(Anchor failure)
									800	5336	110			
												Avg.	10513	
3/4	10	9	120	2	8:40am	11:30am	110	10	400	2668		2500	17100	Wedges pulled over end of anchor
									500	3335	170			(Anchor failure)
									1500	10125	240			
3/4	10	9	120	1/2	8:40am	1:00pm	80	40	500	3335		2600	17750	Wedges pulled over end of anchor
									900	6003	180			(Anchor failure)
									1300	8710	220			
3/4	10	9	120	3/4	2:00pm	10:10am	120	0	700	4669		2900	20000	Wedges pulled over end of anchor
									1400	9415	180			(Anchor failure)
												Avg.	18283	

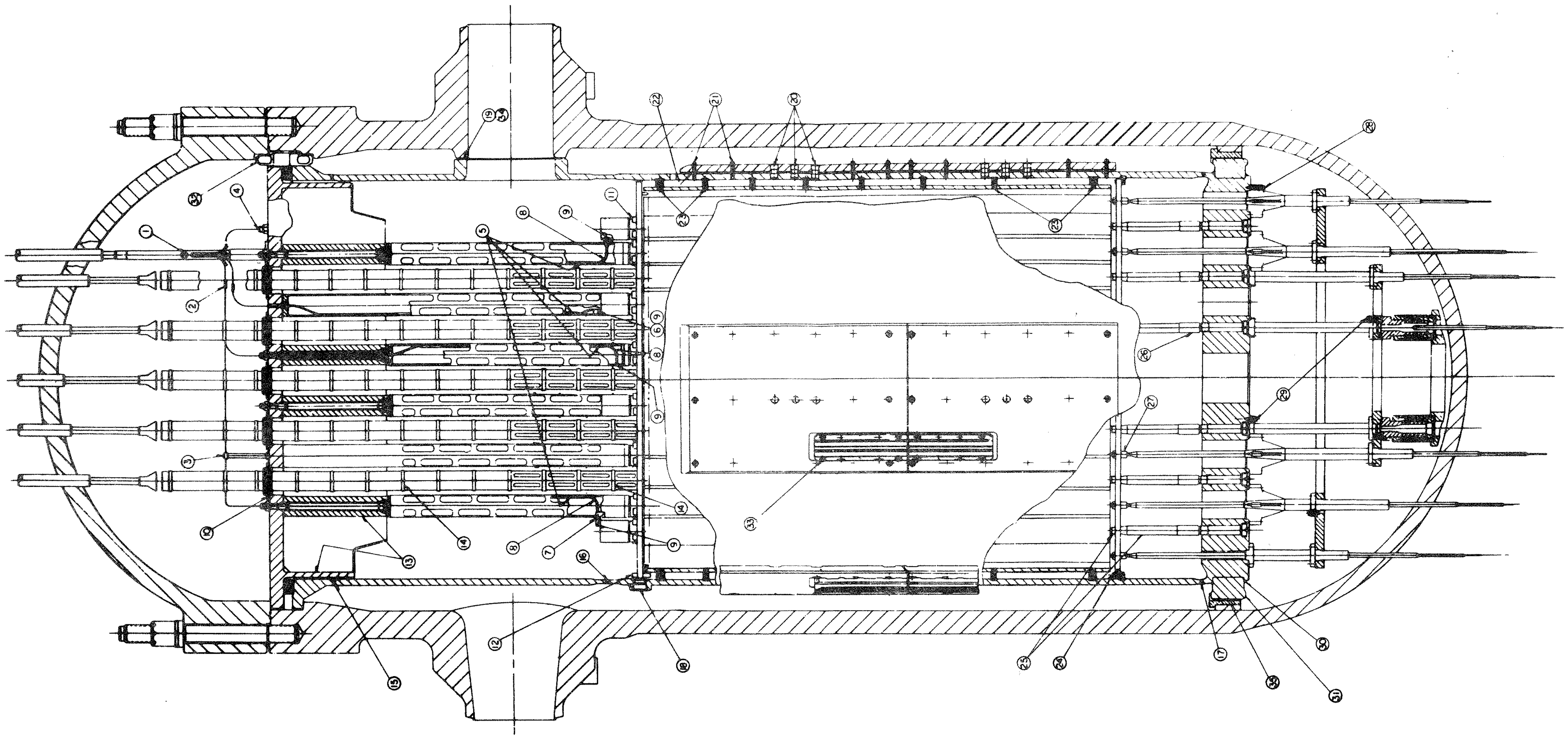
RN 01-113

TABLE 3.9-10 (Continued)

KWIK BOLT TESTS

Kwik-Bolt Size		Kwik-Bolt Embed-ment	Initial Torque	Turns of Anchor Nut	Time of Day at Instal-lation Day 2	Time of Day at Test-ing Day 2	Torque Reading at Testing	Torque Loss	Load at Initial Displacement		Subsequent Torques Applied	Ultimate Load		Mode of Failure
Dia.	Length								Gage Read.	Force		Gage Read.	Force	
In.	In.	In.	Ft-Lbs				Ft-Lbs	Ft-Lbs	psi	lbs	Ft-Lbs	psi	lbs	
1	12	10	240	2 1/2	9:30am	1:00pm	200	40	800	5336		2800	19250	Wedges pulled over end of anchor
									1400	9415	290			(Anchor failure)
1	12	10	240	2	9:30am	1:00pm	190	50	1000	6670		3500	24125	Wedges pulled over end of anchor
									1400	9415	290			(Anchor failure)
1	12	10	240	2	11:00am	1:00pm	240	0	850	5670		2800	19250	Wedges pulled over end of anchor
									1450	9751	290			(Anchor failure)
												Avg.	20875	
1 1/4	12	10 1/2	400	5	10:45am	2:00pm	320	80	1300	8710		2400	16375	Wedges pulled over end of anchor
									1600 2600	10750 17750	480 530			(Anchor failure)
1 1/4	12	10 1/2	400	5 1/2	10:45am	2:00pm	270	130	1150	7694		3000	20650	Wedges pulled over end of anchor
									1900	13250	480			(Anchor failure)
1 1/4	12	10 1/2	400	6	10:45am	2:00pm	320	80	1200	8028		4200	29000	Wedges pulled over end of anchor
									2300	15625	480			(Anchor failure)
												Avg.	22008	

RN 01-113



	FEATURES TO BE EXAMINED
1	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 (THERMOCOUPLE SUPPORT HARDWARE.)
8	RIGIDITY OF EXPOSED PORTION OF THERMOCOUPLE CONDUIT RUNS, AT ACCESSIBLE LOCATIONS. (INSIDE SUPPORT COLUMNS LOWER END.)
9	RIGIDNESS 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.

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Vibration Checkout Functional Test Inspection Points

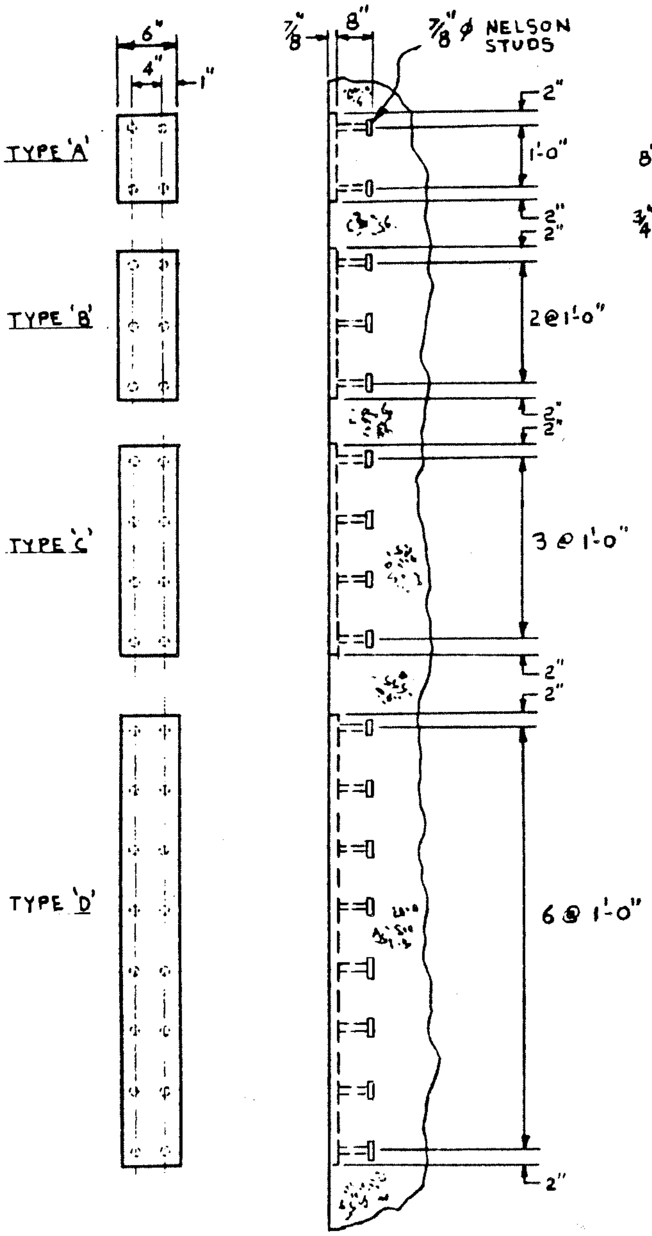
Figure 3.9-1

TYPICAL BASE PLATE DESIGNS FOR PIPE SUPPORTS

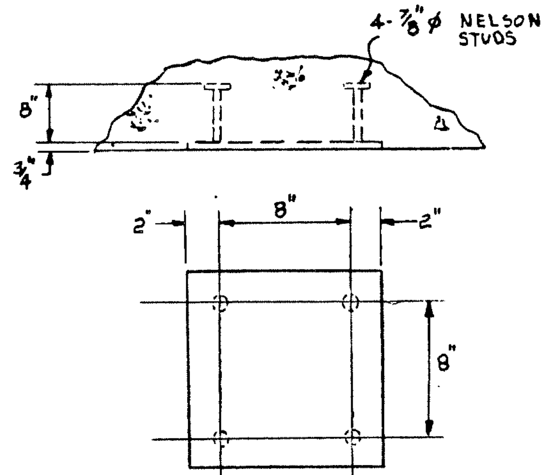
I. MODULAR EMBEDMENTS - (TYPE A, B, C, D)

II. RANDOM EMBEDMENTS - (TYPE K)

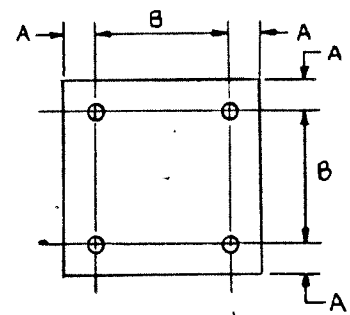
III. HILTI BASE PLATE



I. MODULAR EMBEDMENTS



II. TYPE 'K'



III. HILTI BASE ℘

TYPICAL HILTI-PLATE SIZES			
℘ THICKNESS	A	B	HILTI SIZE
1/2"	1½" TO 3"	6"	5/8" x 6"
3/4"	1½" TO 3"	8"	3/4" x 7"
3/4"	1½" TO 3"	10"	1" x 9"
1"	1½" TO 3"	10"	1" x 9"
1¼"	1½" TO 3"	12"	1¼" x 12"

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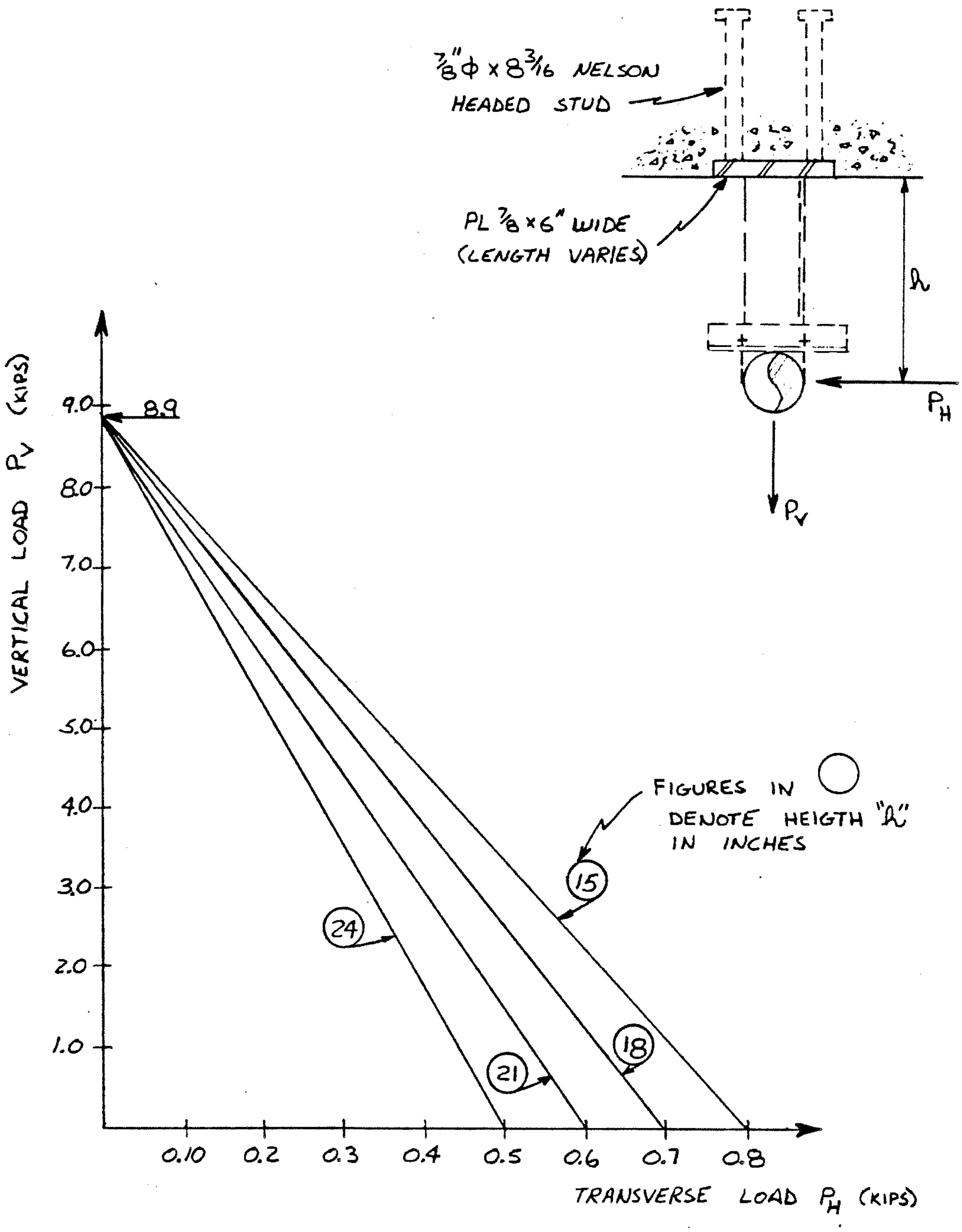
Typical Base Plate Designs for Pipe Supports

Figure 3.9-2



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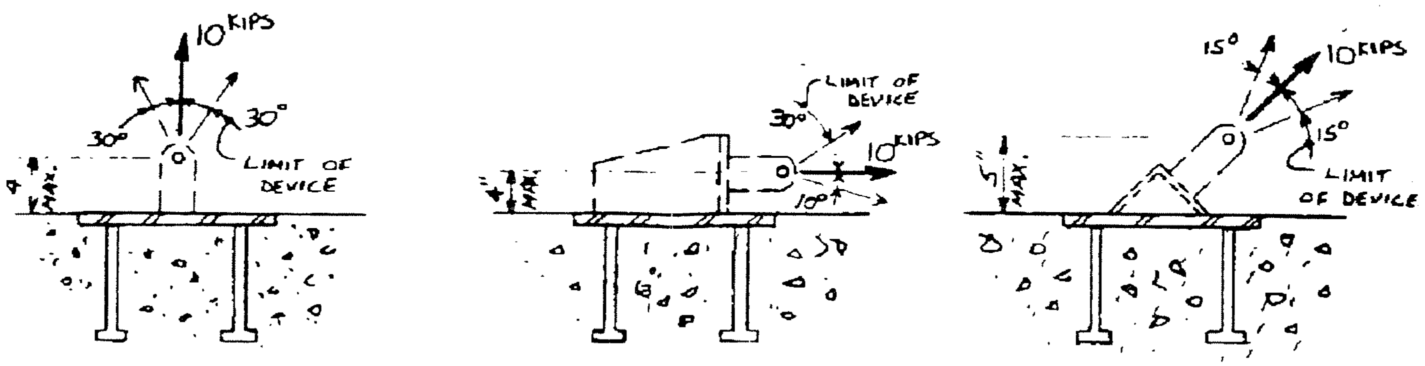
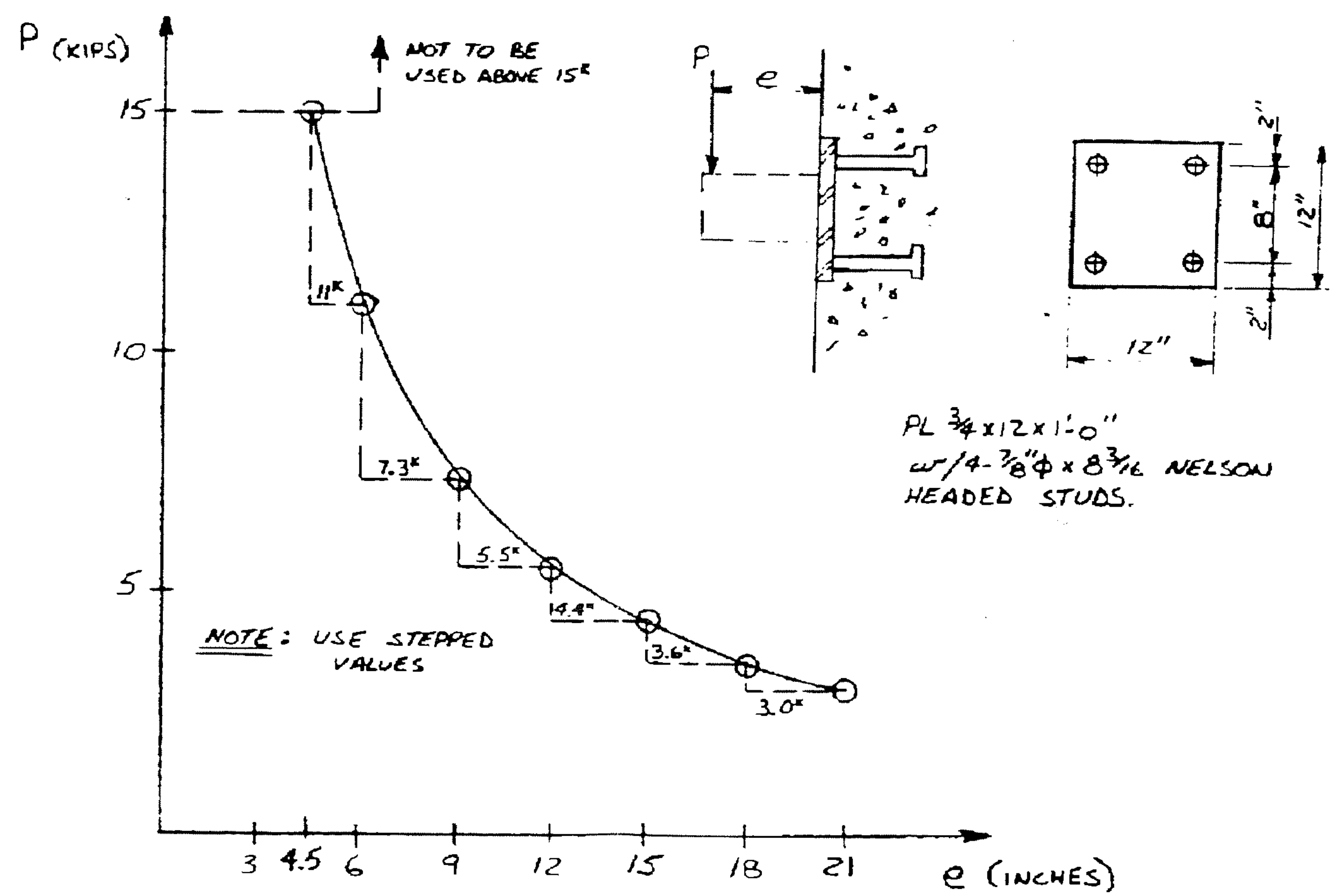
Allowable Transverse Load Vs. Allowable Vertical Load for Modular Embedment Plates

Figure 3.9-3



ALLOWABLE LOAD FOR CLEVIS TYPE FIXTURE ON RANDOM EMBEDMENT PLATE



ALLOWABLE LOADS ON RANDOM EMBEDMENT PLATES FOR VARIED FIXTURE TYPES

AMENDMENT 02-01
MAY 2002

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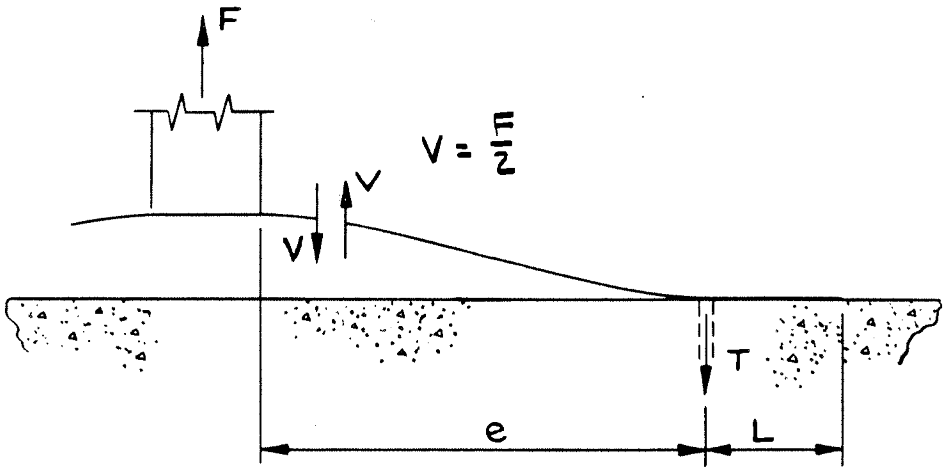
ALLOWABLE LOADS ON RANDOM EMBEDMENT PLATES

REV. Ø

Figure 3.9-4

CONDITION 1



WHEN $(T_i - V)L \geq V\frac{e}{2}$, ANCHOR LOAD $T = T_i$ = PRELOAD IN THE ANCHOR.

PRYING FORCE = 0

T = TOTAL LOAD IN ANCHOR

T_i = PRELOAD IN ANCHOR

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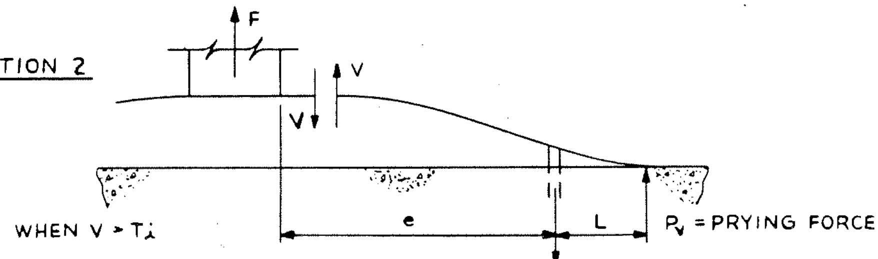
Analysis for Prying Action Under Tensile Load - No Prying Force

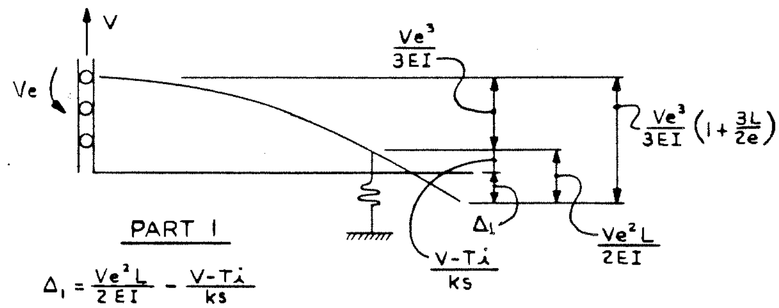
Figure 3.9-5

CONDITION 2



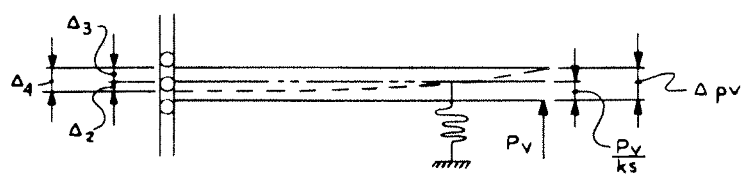
PROBLEM

THE PROBLEM IS SOLVED BY SUPERPOSITION, USING PARTS 1 AND 2 BELOW



PART 1

$$\Delta_1 = \frac{Ve^2L}{2EI} - \frac{V-Ti}{ks}$$



PART 2

$$\Delta_3 = \Delta_4 - \Delta_2 = \frac{P_v L^2 e}{EI} + \frac{P_v L^3}{3EI} + \frac{6}{5}\,\frac{P_v L}{AG}$$

$$\Delta pv = \Delta_3 + \frac{P_v}{ks} = \frac{P_v L^2 e}{EI} + \frac{P_v L^3}{3EI} + \frac{6}{5}\,\frac{P_v L}{AG} + \frac{P_v}{ks}$$

(EQ. 1) EQUATING $\Delta_1 = \Delta pv$; $P_v = \left[\frac{Ve^2L}{2EI} - (V-Ti)\frac{1}{ks}\right] \div \left[\frac{1}{ks} + \frac{L^2e}{EI} + \frac{L^3}{3EI} + \frac{6}{5}\,\frac{L}{AG}\right]$

IF $\frac{Ve^2L}{2EI} < \frac{V-Ti}{ks}$ *NO PRYING EXISTS AND T = V*

IF $\frac{Ve^2L}{2EI} > \frac{V-Ti}{ks}$ *SOLUTION OF EQ-1 YEILDS THE PRYING FORCE.* P_v.

TOTAL BOLT FORCE IS $V + P_v$

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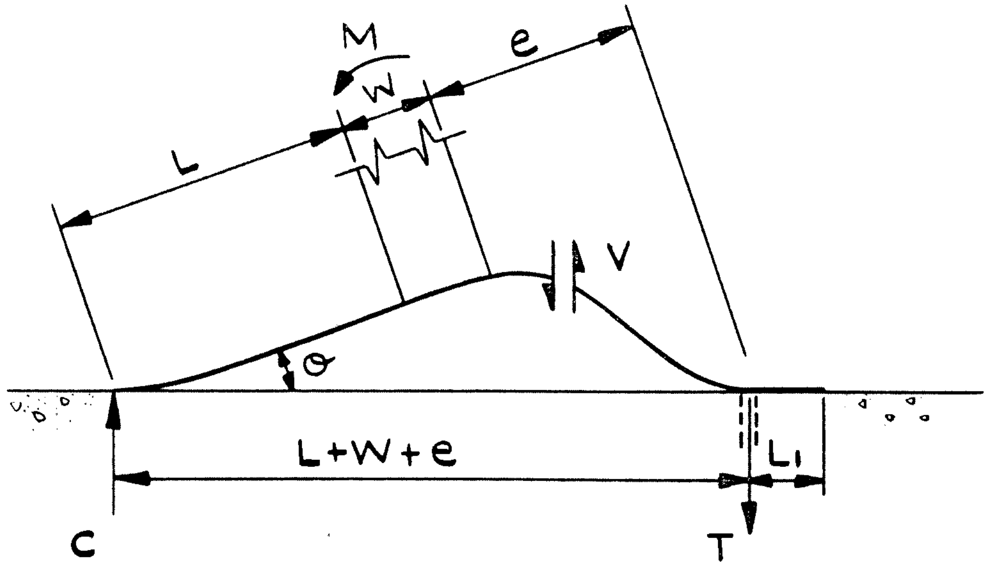
Analysis for Prying Action Under Tensile Load - Prying Forces Possible

Figure 3.9-6

CONDITION 1: NO TENSION ANCHOR ROTATION OR DISPLACEMENT



T = TOTAL LOAD IN ANCHOR

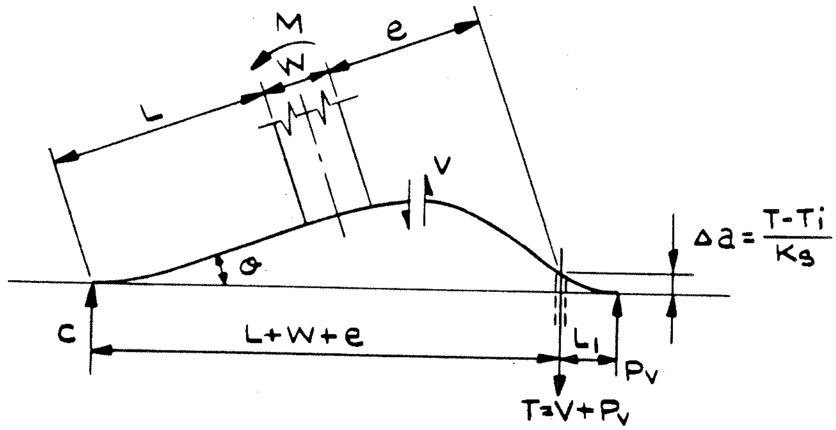
T_i = PRELOAD IN ANCHOR

FOR $M \leq \left(\frac{T_i L_1}{e/2 + L_1}\right)(L+W+e)$, $T = T_i$

PRYING FORCE = 0

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Analysis for Prying Action Under Applied Moment - No Prying Forces Figure 3.9-7

CONDITION 2 : TENSION ANCHOR ROTATION AND DISPLACEMENT



T = TOTAL LOAD IN ANCHOR
T_i = PRELOAD IN ANCHOR

$$T = V + P_V$$

$$\Delta a = \frac{T - T_i}{K_s}$$

$$\theta = \frac{VL^2}{2EI_p}$$

$$L = \frac{3}{\theta}\left\{\left[\Delta a - \frac{P_V L_1 e^2}{2EI} + \frac{Ve^3}{3EI}\right] - \theta W - \theta e\right\}$$

$$V = \frac{M + P_V L_1}{e + W + L}$$

$$P_V = \frac{\frac{Ve^2 L_1}{2EI} - \frac{V - T_i}{K_s} - \theta L_1}{\frac{1}{K_s} + \frac{L_1^2 e}{EI} + \frac{L_1^3}{3EI} + \frac{6}{5}\frac{L_1}{AG}}$$

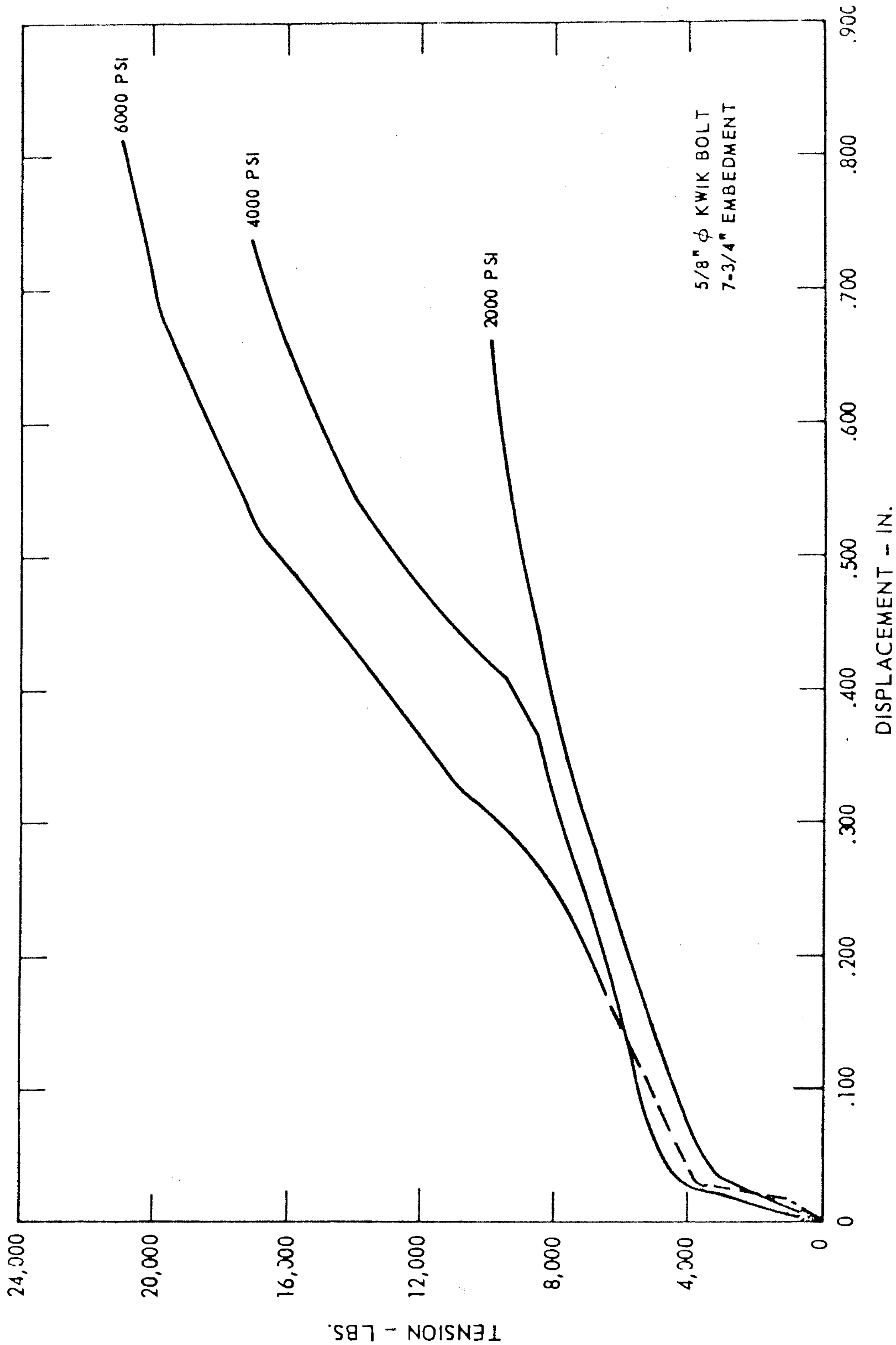
IF $\frac{V - T_i}{K_s} \geq \frac{Ve^2 L_1}{2EI} - \theta L_1$, NO PRYING EXISTS AND $T = V = C$

IF $\frac{V - T_i}{K_s} < \frac{Ve^2 L_1}{2EI} - \theta L_1$, SIMULTANEOUS SOLUTION OF THE SIX EQUATIONS GIVEN ABOVE WILL YIELD THE LOCATION OF THE COMPRESSIVE FORCE C AND THE MAGNITUDE OF THE PRYING FORCE P_V

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analysis for Prying Action Under Applied Moment - Prying Forces Possible

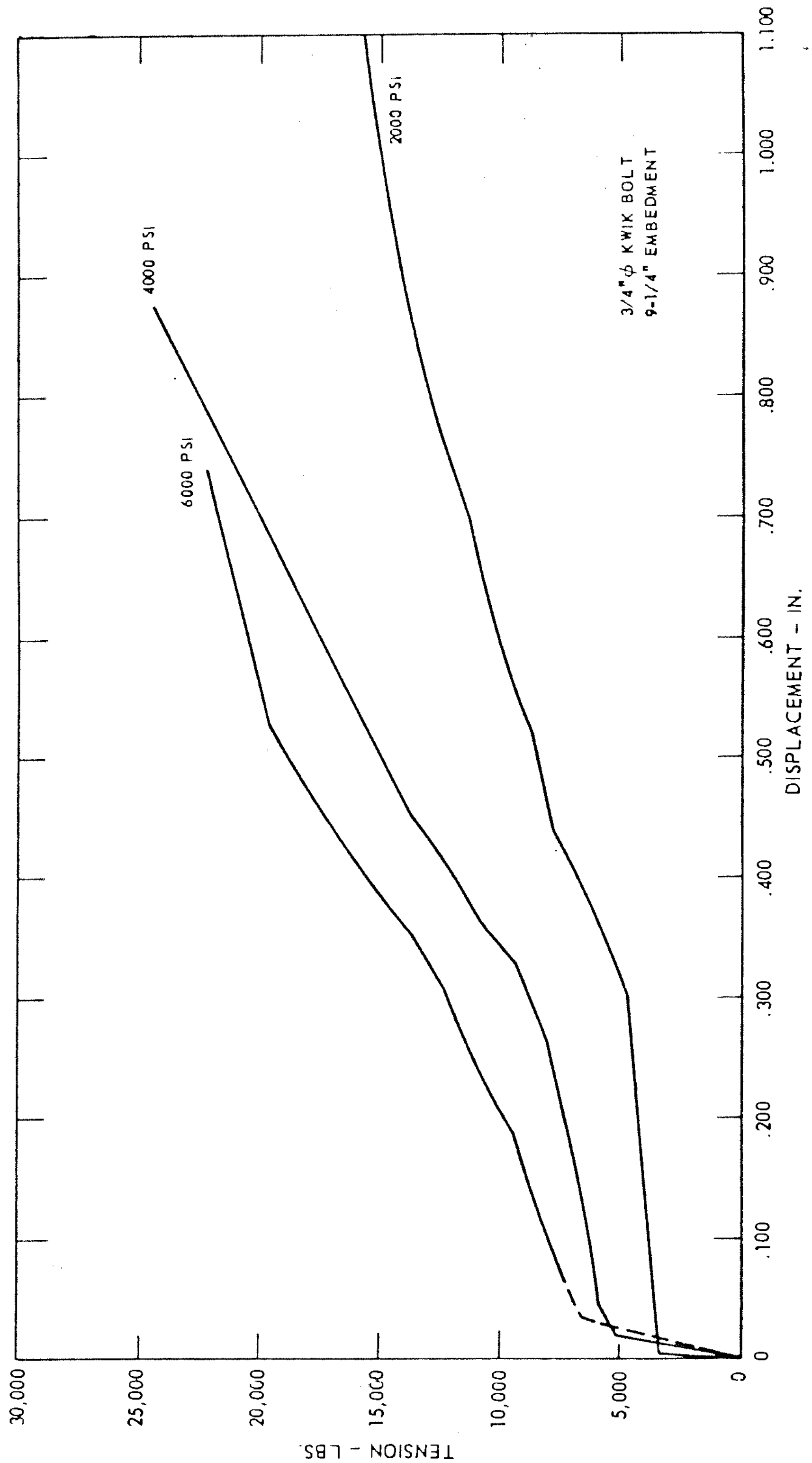
Figure 3.9-8



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Force - Displacement Pull Out Curve
for 5/8" Hilti Kwik Bolt and
7 - 3/4" Embedment

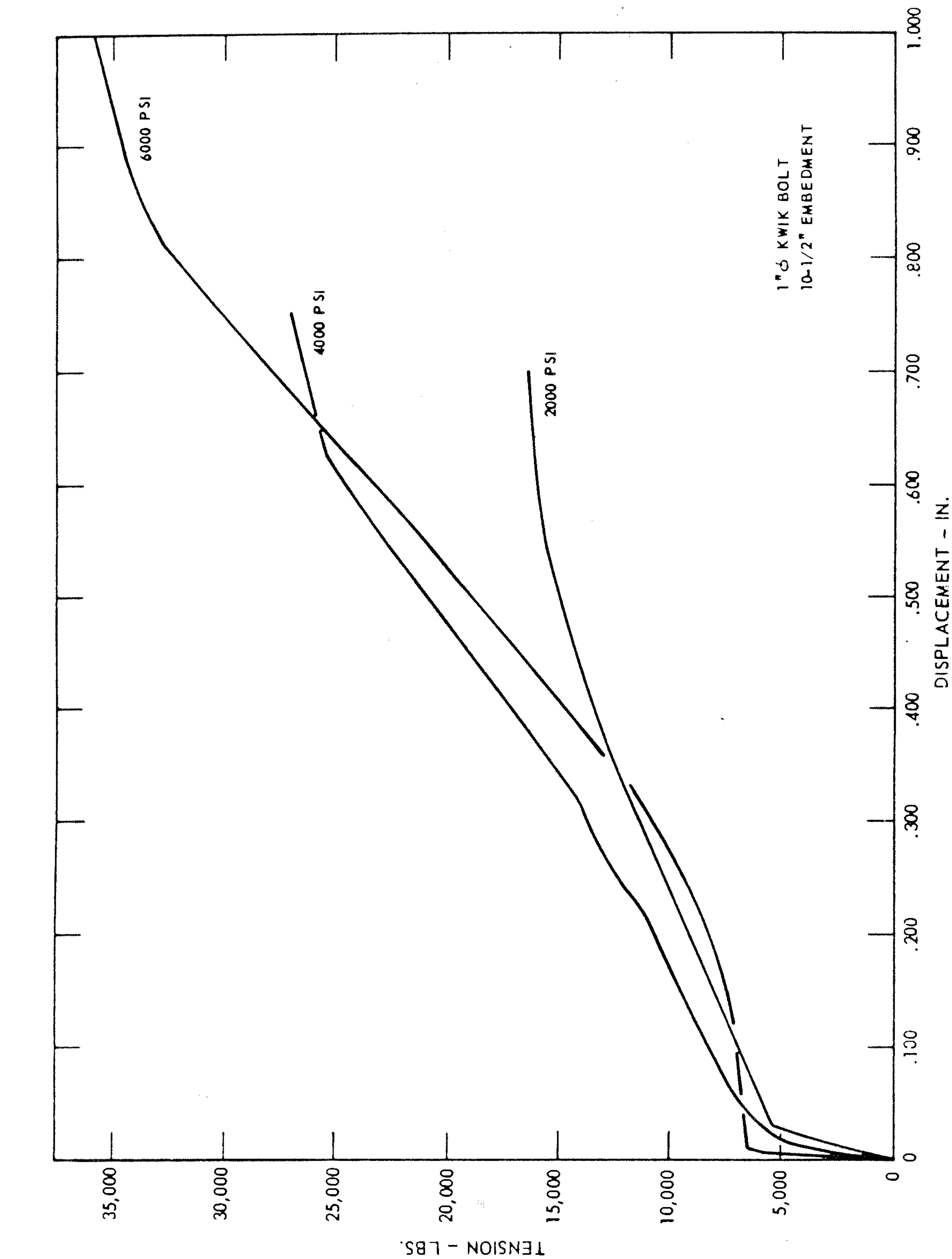
Figure 3.9-9



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Force - Displacement Pull Out Curve
for 3/4" Hilti Kwik Bolt and
9 - 1/4" Embedment

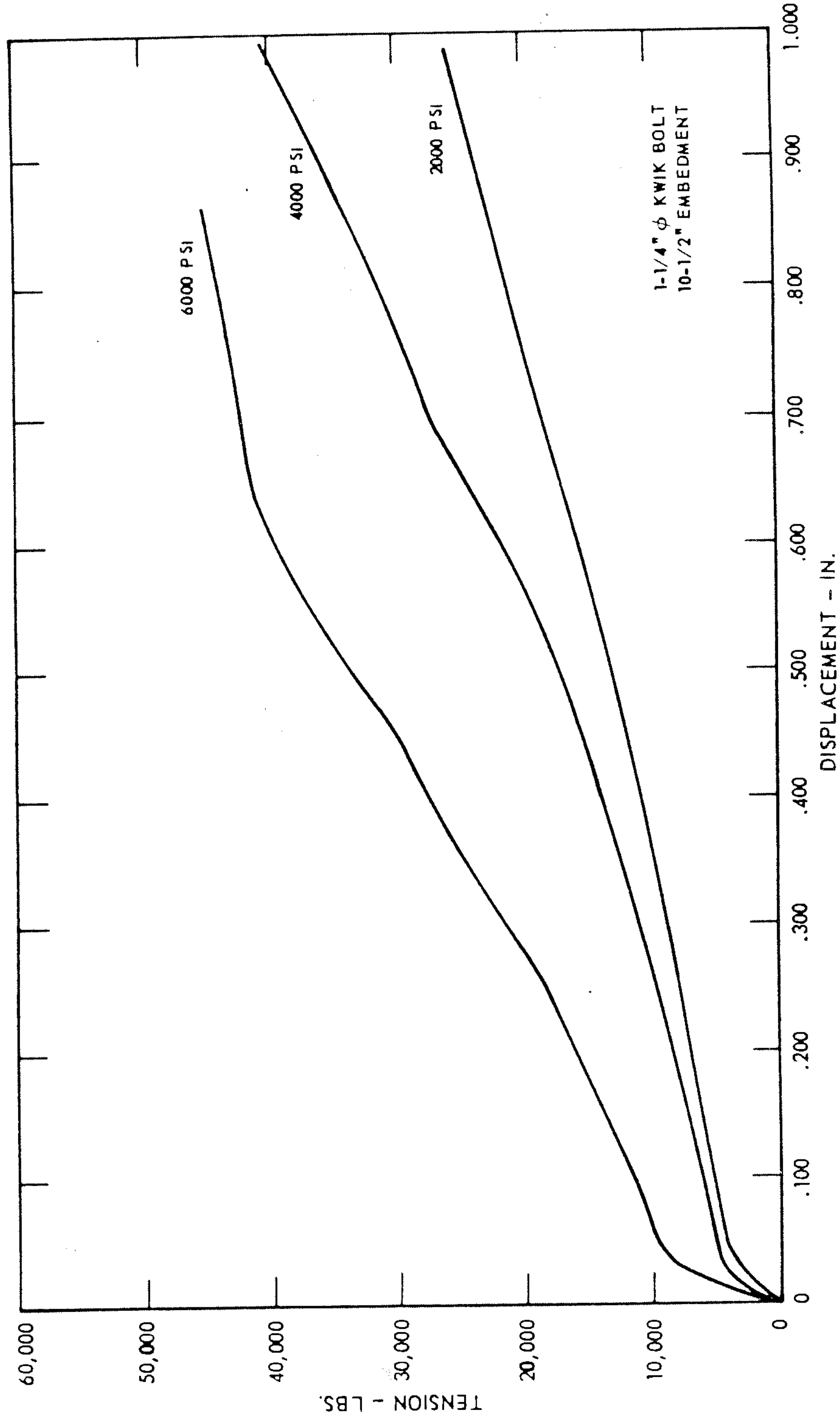
Figure 3.9-10



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Force - Displacement Pull Out Curve
for 1" Hilti Kwik Bolt and
10 - 1/2" Embedment

Figure 3.9-11



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Force - Displacement Pull - Out Curve
for 1-1/4" Hilti Kwik Bolt and
10-1/2" Embedment

Figure 3.9-12

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY 1 INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC QUALIFICATION CRITERIA

The safety-related instrumentation and electrical equipment that is seismically qualified is identified in Table 3.10-1. The safety-related instrumentation and electrical equipment within the scope of the Nuclear Steam Supply System (NSSS) and which requires seismic qualification is identified in Table 3.10-2. This equipment is designed to withstand the combined effect of normal operating loads acting simultaneously with horizontal and vertical components of the safe shutdown earthquake (SSE) (described in Section 3.7.1) without loss of function or structural integrity.

Suppliers of safety-related equipment are provided with the response spectrum curves for the SSE loading applicable for the equipment location. The response spectrum curves for the various locations within the plant are developed from the input ground motion, as described in Section 3.7.1.

Equipment suppliers are required to submit test data and/or calculations to demonstrate that their equipment does not suffer loss of function during or after seismic loading due to an SSE, as described in Section 3.7.5.

The seismic qualification of safety-related equipment, systems, and components may be accomplished in various ways. Three of the methods used are as follows:

1. Determine equipment performance through analysis (see Section 3.10.2.1.1).

2. Test the equipment under simulated seismic conditions (see Section 3.10.2.1.2).

3. Qualification by combined test and analysis.

The choice is based upon the practicality of the method for the type, size, shape, and complexity of the equipment under consideration, as well as the reliability of the conclusions resulting from the analysis, test, or combination.

Data submitted by the equipment suppliers is required to include justification for the use of the method chosen.

The NSSS supplier has previously type tested and qualified items 1 through 8, of the equipment listed in Table 3.10-2 in accordance with IEEE-344 [1]. Item 9 is discussed in Section 3.10.2. Reference [2] presents the testing procedures used to qualify NSSS equipment by type testing. Seismic qualification testing of this equipment in accordance with IEEE-344 [1] is documented in References [3] through [9]. Reference [10] presents the theory and practice, as well as justification for the use of single axis sine beat test inputs used in the seismic qualification of electrical equipment. In addition, it is noted that the NSSS supplier has conducted the seismic qualification demonstration test program outlined in Reference [11] (also see References [9], [12], [13], [14], and [15]).

For the seismic qualification of NSSS electrical equipment outside of the Reactor Building, this demonstration test program, in conjunction with the justification for the use of single axis sine beat tests presented in Reference [10] indicates that the original tests, documented in References [3] through [9], satisfy the requirements of IEEE-344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The general method showing the multi-frequency, multi-directional test inputs used in the demonstration test program is presented in Reference [12].

The peak accelerations used in the type testing are conservative values that are checked against those derived by structural analyses of SSE loadings. For the SSE, there may be permanent deformation of the equipment provided the capability to perform its function is maintained.

Seismic qualification of Category 1 instrumentation and electrical equipment meets the requirements of Criterion 2 of the 1971 GDC. 00-01

3.10.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

3.10.2.1 <u>Balance of Plant Equipment</u>

3.10.2.1.1 Seismic Analysis

1. Equipment is individually modeled as a multi-degree-of-freedom, lumped-mass system with mass-free interconnections. Three different models are used to determine the seismic response for the vertical and two horizontal directions of excitation.

   The number of mass points chosen in the analysis is selected to yield an accurate prediction of the response of the equipment to a seismic disturbance.

   The natural frequencies and mode shapes are determined for the equipment in the mounted, ready for service condition. Consideration is given to the relationship of the fundamental natural frequency of the equipment to the rigid frequency (i.e., greater, equal to, or less). The equipment supplier then performs a structural analysis of the equipment to ensure that material stress limits are not exceeded.

2. Equipment is analyzed statically when the fundamental frequency of the equipment is greater than the rigid frequency. In the static analysis, seismic forces on each component of the equipment is obtained by multiplying the lumped mass of each mass point by the appropriate maximum floor acceleration. The maximum floor acceleration is obtained from the high frequency end of the response spectra envelope. Response spectra curves are discussed in Section 3.7.2.

3. For cable trays, tray supports, and similar structures, the structures are analyzed statically using peak response accelerations. A conservatively low damping value is used when determining the peak response acceleration. Structural properties for the cable trays are determined by load tests and the properties of the members for supporting structures are obtained from Reference [16] and other similar documents. The load on a structure is determined by multiplying the peak acceleration by the mass of the structure. The resulting force is distributed in proportion to the mass of the structure and is combined with dead loads. A stress analysis is then performed on the structure to ensure its structural adequacy.

4. The majority of the instrument sensors are qualified to a conservatively high acceleration level which envelopes both frequency and acceleration for all plant locations. The remainder of the instrument sensors are qualified by one of the above methods. The values selected are 1.5 g for wall mounted instruments and 3 g for pipe mounted instruments. Except for pipe mounted instruments, instruments are mounted either on wall racks or floor stands. Table 3.10-3 identifies pipe mounted instruments and their locations. Instrument racks and stands are stiffened to preclude excessive amplification. For pipe mounted instruments, the attachment between the instrument and associated piping is designed to be rigid. The instrument and seismic qualification level will be included in the piping system design specification.

5. If by analysis, it is determined that the fundamental frequency of a valve assembly is greater than 30 Hz, a static seismic structural and deflection/clearance analysis is performed to show the structural and functional adequacy of the valve assembly.

3.10.2.1.2 Testing Under Simulated Seismic Conditions

Test methods used include the following:

1. Random vibration test.

2. Sinusoidal sweep test.

3. Continuous sine test.

4. Sine beat test.

5. Decaying sinusoidal test.

6. Short time sinusoidal test.

7. Multi-frequency biaxial test.

Testing is performed by subjecting devices and assemblies to vibratory motion which simulates seismic conditions predicted for the equipment mounting location during an SSE. Assemblies are tested with devices in operating condition except in the case of complex assemblies, such as control panels, switchgear, etc., where testing requirements dictate that devices be inoperative. Such testing demonstrates the ability of equipment to perform its intended function under seismic conditions or, in the case of complex assemblies, qualifies the assemblies and provides data for device mountings to be used as input information when testing individual component devices. Such data is in the form of a response spectrum or other equivalent forms.

Assemblies or component devices being tested are mounted for testing in a manner simulating the intended service mounting. Due to different characteristics of various driving mechanisms and the interaction between heavy equipment and the shake table, the input to the shake table may be different from the motions at the equipment mounting. Hence, the input to the equipment is based upon the motions at the equipment mounting instead of the input to the shake table.

Vertical and horizontal inputs are applied simultaneously unless it can be demonstrated that horizontal and vertical responses are uncoupled. Maximum vibratory accelerations at the equipment mounting are equal to or greater than the maximum floor acceleration.

3.10.2.2 Nuclear Steam Supply System Equipment

NSSS Seismic Category-1 instrumentation and electrical equipment was seismically qualified by type testing using sine beat inputs to each of three perpendicular axes independently applied according to the procedures of IEEE-344 [1], Section 3.2. At the time of this testing, which is reported in References [3] through [9], implementation of the IEEE-344 [1] testing method fulfilled all seismic qualification requirements. The results show that there are no electrical irregularities that would leave the plant in an unsafe condition. In addition, as noted in Section 3.10.1, a demonstration test program was conducted which, when considered in conjunction with those tests presented in References [3] through [8] and[17], results in satisfying the requirements of IEEE-344 1975.

In the reported tests, the equipment operated properly during and after testing with equivalent ground accelerations at zero period ranging up to 0.4 g and higher. The sine beat inputs were applied not only at the equipment natural frequencies but also at many frequencies (spaced at about 1/2 octave) below 33 Hz to ensure that the equipment would function normally regardless of uncertainties of building or equipment natural frequencies. The sine beat test is severe because it excites the resonant response of the equipment, thereby producing the most damaging effect to the components. This test not only excites the component to motion greater than the input but also produces fatigue damage well above that produced by seismic disturbances. This method assumes that building natural frequency coincides with that of the equipment and is as conservative as the one proposed by the NRC Staff. Any possible coupling effect loses importance when compared to the excitation of components at sensitive frequencies as is done by the sine beat test. This test, therefore, provides more positive proof of equipment capability than the simultaneous random input test which, because of phase relationships, could result in less severe application of the seismic input.

The nuclear instrumentation system power range neutron detector is qualified for seismic environments by testing at acceleration levels greater than those expected at their location during a seismic disturbance as defined in Sections 2.5 and 3.7.

The power range detector has been vibration tested in both the transverse (horizontal) direction and the longitudinal (vertical) direction. The pressure and differential pressure transmitters are qualified by multi- axial, multifrequency testing. Seismic inputs for this testing were developed using the methods described in WCAP 8687.

The Westinghouse ISD 7300 series process control cabinets were originally qualified using single-axis sine beat test methods and results reported in Reference [7]. To resolve NRC concerns regarding the ability of the process bistables to change state during a postulated seismic event and to address NRC questions relative to multi-axis, multifrequency effects, a typical 7300 channel was tested using bi-axial, multifrequency inputs. These inputs were developed in accordance with Reference [12]. Results of the testing were submitted to the NRC as Reference [15]. Seismic qualification of the solid state protection system was performed using single-axis sine beat test methods and results provided in References [5] and [6].

Initial testing on the Nuclear Instrumentation System (NIS) was performed using single-axis sine beat methods and reported in References [3] and [5]. The NIS was also tested to verify the ability of the bi-stables to change state during a seismic event. This testing was performed using multifrequency bi-axial inputs; however, unlike the process control equipment bi-stable testing, a single (out of four) NIS cabinet was tested. Electrical operability of the bi-stables was confirmed. Results of this testing were submitted to the NRC as Reference [14]. The safeguards test cabinets were qualified by single-axis sine beat testing and results are Reported in Reference [18]. The narrow range resistance temperature detectors were tested using single-axis sinusoidal techniques. Results of this testing are documented in Reference [9] (approved by NRC). Seismic qualification of the type DS reactor trip switchgear is scheduled to begin in late 1978. Unless limited by facility configuration, the intent is to utilize bi-axial, multifrequency test methods.

Original testing of the instrument supply inverters (static inverters) was performed using single-axis sine beat methods and results reported in References [3] and [5]. To satisfy NRC concerns regarding the original test at less than full load, the static inverter was seismically retested at the approximate normal operation and accident power loading condition using bi-axial sine beat inputs. Results of this testing were submitted to the NRC as Reference [13]. Westinghouse supplied indicators and recorders used for post accident monitoring are qualified using multi-axial multifrequency test methods per WCAP 8687.

Equipment for the Virgil C. Summer Nuclear Station is procured on a similar basis to that which is qualified. Any design change in the equipment is evaluated to determine if the changes were of a nature that could affect the results of the seismic tests or would require the equipment to be requalified for seismic integrity.

The criteria and verification procedure employed to account for the possible amplified design loads (frequency and amplitude) for NSSS safety-related instrumentation and electrical equipment is presented in Reference [2], Appendix B of Reference [3] and Section 4 of Reference [10].

3.10.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Testing of electrical equipment and instrumentation supports is performed in the manner described in Section 3.10.2.

3.10.4 OPERATING LICENSE REVIEW

The documentation for the equipment demonstrates that the equipment satisfies performance requirements before, during and after subjection to the seismic accelerations for which the equipment is qualified in accordance with the seismic criteria specified.

3.10.4.1 Analytical Data

In those cases where proof of performance was obtained by analytical means, a report is prepared. This report is presented in a step-by-step form which is readily auditable by persons skilled in such analyses. The report includes the following information:

1. The equipment identification and description.

2. Summary of analytical results.

3. Load criteria and assumptions.

4. Methods of analysis.

5. Calculations.

6. Certification of the report by an independent reviewer.

3.10.4.2 Test Data

When proof of performance is obtained through testing, a report similar to that described in Section 3.10.4.1 is prepared. This report includes the following information:

1. Equipment identification.

2. Test facility.

   a. Location.

   b. Test equipment.

   c. Test method.

   d. Test data (as a minimum: date, frequency, amplitudes, and duration of test).

   e. Results and conclusions.

   f. Certification of the report by an independent reviewer.

3.10.5 REFERENCES

1. Institute of Electrical and Electronics Engineers, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," IEEE-344-1971.

2. Morrone, A., "Seismic Vibration Testing with Sine Beats," WCAP-7558, October, 1971.

3. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment," WCAP-7397-L (Proprietary), February, 1970 and WCAP-7817 (Non-Proprietary), December, 1971.

4. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)," WCAP-7397-L, Supplement 1 (Proprietary), January, 1971 and WCAP-7817, Supplement 1 (Non-Proprietary), December, 1971.

5. Potochnik, L. M., "Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)," WCAP-7817, Supplement 2, December, 1971.

6. Vogeding, E. L., "Seismic Testing of Electric and Control Equipment (Westinghouse Solid State Protection System), (Low Seismic Plants)," WCAP-7817, Supplement 3, December, 1971.

7. Reid, J. B., "Seismic Testing of Electric and Control Equipment (Westinghouse NUCANA 7300 Series), (Low Seismic Plants)," WCAP-7817, Supplement 4, November, 1972.

8. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel), (Low Seismic Plants)," WCAP-7817, Supplement 5, March, 1974.

9. Bower, J. S. and Drexler, J. E., "Equipment Qualification Test Report WEED Resistance Temperature Detectors," WCAP 8687, Supplement 2 - ESOA, March 1989.

10. Fischer, E. G. and Jarecki, S. J., "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974," WCAP-8373, August, 1974.

11. Letter NS-CE-692, dated July 10, 1975, from C. Eicheldinger (Westinghouse) to D. B. Vassello (NRC).

12. Jarecki, S. J., "General Method of Developing Multi-Frequency Bi-axial Test Inputs for Bistables," WCAP-8624 (Proprietary) and WCAP-L8695 (Non-Proprietary), September 1975.

13. Figenbaum, E. K., "Seismic Testing of Electrical and Control Equipment, Static Inverter and Instrument Bus Distribution Panel," WCAP-7821, Supplement 2, Addendum 1, October, 1975.

14. Jarecki, S. J., et al., "Seismic Operability Demonstration Testing of the Nuclear Instrumentation System Bistables," WCAP-8830 (Proprietary) and WCAP-8831 (Non-Proprietary), October, 1976.

15. Jarecki, S. J., et al., "Seismic Operability Demonstration Testing of the WISD 7300 Series Process Instrumentation system Bistables," WCAP-8828 (Proprietary) and WCAP-8829 (Non-Proprietary), October, 1976.

16. American Institute of Steel Construction, "Manual of Steel Construction," 7th Edition.

17. "Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 7 (Proprietary) and WCAP-7820, Supplement 7 (Non-Proprietary), August, 1977.

18. Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment for Low Seismic Plants," WCAP-7817, Supplement 7, September, 1976.

TABLE 3.10-1

METHODS AND PROCEDURES FOR SEISMIC QUALIFICATION OF BOP SEISMIC CATEGORY 1 EQUIPMENT AND ASSOCIATED SUPPORTS

Equipment	Method
Valve Operators (actuator) [1]	See the applicable EQDP, EQF and/or SQF.
7200 Volt Switchgear [1]	See the applicable EQDP, EQF and/or SQF. 02-01
480 Volt Unit Substations [1]	See the applicable EQDP, EQF and/or SQF.
480 Volt Vital System Transformers	See the applicable EQDP, EQF and/or SQF.
Motor Control Centers [1]	See the applicable EQDP, EQF and/or SQF.
Motors for Safety Class Pumps and Fans [1]	See the applicable EQDP, EQF and/or SQF.
Battery Chargers [1]	See the applicable EQDP, EQF and/or SQF.
Diesel Generators [1]	See the applicable EQDP, EQF and/or SQF.
Cable Tray and Cable Tray Hangers [1]	Analysis. Peak response acceleration of structures vector sum combination of stress
Electrical Containment Penetrations [1]	See the applicable EQDP, EQF and/or SQF.
Distribution Panels [1]	See the applicable EQDP, EQF and/or SQF.
Batteries and Battery Racks [1]	See the applicable EQDP, EQF and/or SQF.
Speed Switches (7200 volt) [1]	See the applicable EQDP, EQF and/or SQF.
Transfer Switches (7200 volt) [3]	See the applicable EQDP, EQF and/or SQF.
Transfer Switches (480 volt) [3]	See the applicable EQDP, EQF and/or SQF.
Pressure Transmitters [1]	See the applicable EQDP, EQF and/or SQF.
Level Transmitters [1]	See the applicable EQDP, EQF and/or SQF.
Flow Transmitters [1]	See the applicable EQDP, EQF and/or SQF.
Temperature Sensors [1]	Type test. Single frequency; single axis test.
Control Board Switch Modules [1]	See the applicable EQDP, EQF and/or SQF.
ESF Loading Sequence [1] Control Panels	See the applicable EQDP, EQF and/or SQF.
Reactor Protection Under-frequency and Voltage Relay Panels	Type test and analysis. Random frequency with super-imposed sine beats; multi-axis test.
Main Control Board [1] [5]	See the applicable EQDP, EQF and/or SQF.
Heating, Ventilation and Air Conditioning Control Panel [1] [5]	See the applicable EQDP, EQF and/or SQF.

TABLE 3.10-1 (Continued)

METHODS AND PROCEDURES FOR SEISMIC QUALIFICATION OF BOP SEISMIC CATEGORY 1 EQUIPMENT AND ASSOCIATED SUPPORTS

Equipment	Method
Balance of Plant Instrument Panels	See the applicable EQDP, EQF and/or SQF.
Control Room Evacuation Panel	See the applicable EQDP, EQF and/or SQF.
HVAC Mechanical Water Chiller Control Panels and Motors [1]	See the applicable EQDP, EQF and/or SQF.
Radiation Monitoring Control Panel	See the applicable EQDP, EQF and/or SQF.
Termination Panels [1]	See the applicable EQDP, EQF and/or SQF.
Earthquake Instrumentation	See the applicable EQDP, EQF and/or SQF.
Hydrogen Analyzer Panels	See the applicable EQDP, EQF and/or SQF.
Radiation Monitors; RM-L1, RM-A2, RM-G7 [6]	See the applicable EQDP, EQF and/or SQF.
Radiation Monitors; RM-G18, RM-G17A&B [6]	See the applicable EQDP, EQF and/or SQF.
Level Switches, leak detection	Type test. Random frequency, multi-axis. 00-01
Flow Switches, leak detection	Type test. Single frequency, single axis.
TSC Isolation Cabinets	See the applicable EQDP, EQF and/or SQF.
RTD/I Converters	See the applicable EQDP, EQF and/or SQF.
Isolation Fuses and Fuse Blocks, within Heat Trace Panels	See the applicable EQDP, EQF and/or SQF.

[1] Equipment required for safe shutdown.

[2] Deleted

[3] Electrical continuity required for safe shutdown but transfer capability not required.

[4] Deleted

[5] The control boards are seismically qualified. Non-nuclear safety class components mounted on the boards are not required to be seismically qualified but the control boards are so designed that failure of non-nuclear safety class class components does not degrade any seismically qualified components.

[6] Pertinent requirements of Regulatory Guide 4.15 Revision 1 apply to the quality assurance program.

TABLE 3.10-2

IDENTIFICATION OF NUCLEAR STEAM SUPPLY SYSTEM SEISMIC CATEGORY I INSTRUMENTATION, ELECTRICAL EQUIPMENT AND SUPPORTS

	Item	Method	
1.	Pressure Transmitters ** and Differential Pressure Transmitters **	Multi-axial, multi-frequency	
2.	Process Control Equipment Cabinets **	Single axis sine beat, biaxial multi-frequency	
3.	NSSS Solid State Protection System Cabinets	Single axis sine beat	
4.	Nuclear Instrumentation System Cabinets	Single axis sine beat, biaxial multi-frequency	
5.	Safeguards Test Racks	Single axis sine beat	
6.	Resistance Temperature Detectors **	Single, sinusoidial	
7.	Instrument Supply Inverters **	Single axis sine beat, biaxial sine beat***	RN 17-022
8.	Reactor Trip Switchgear	Multiaxis, multi-frequency	
9.	Power Range Neutron Detectors	Single axis sinusoidal	
10.	Post Accident Monitoring Equipment (Indicators ** and Recorders)	Multi-axis, multi frequency	
11.	Post Accident Electric Hydrogen Recombiners	Single axis sine beat for recombiners, bi-axial sine beat for control panel	
12.	Core Subcooling Monitor	Exempt from qualification per NRC acceptance of the SCE&G response to Generic Letter 82-28	00-01
13.	Critical System Leak Monitoring System	Removed per MRF-20206	
14.	Reactor Vessel Level Instruments	Single axis sine beat, bi-axial multi-frequency	
15.	Pressurizer Safety Valve Flow Monitor	Sine sweep test; bi-axial multi-frequency test.	

** Required for safe shutdown (assuming normal operation and not post accident conditions).

***Replacement Inverters qualified via multi-axis multi-frequency testing. RN 17-022

TABLE 3.10-3

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System [1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
ITE4480	SW	Cooling water to reactor building air handling heat exchange loop A, temperature element, computer input T4480A, post accident monitor, 100 ohm RTD	SW; -; 425'-0"	D-302-222; E-303-010; E-811-037	
ITW4481	SW	Service water supply header A, thermowell, local test	SW; -; 425'-0"	D-302-221; E-303-010; E-811-037	RN 11-018
ITE4510	SW	Cooling water to reactor building air handling heat exchange loop B, temperature element, computer input T4510A, post accident monitor, 100 ohm RTD	SW; -; 425'-0"	D-302-222; E-303-010; E-811-037	
ITW4511	SW	Service water supply header B, thermowell, local test	SW; -; 425'-0"	D-302-221; E-303-010; E-811-037	RN 11-018
ITW4477	SW	Service water to diesel generator cooler A, thermowell, local test	DG; H.4-1; 436'-0"	D-302-222; E-303-022; E-811-047	
ITE4478	SW	Service water from diesel generator cooler A, temperature element, type E thermocouple	DG; H.4-1; 436'-0"	D-302-222; E-303-022; E-811-047	
ITW4507	SW	Service water to diesel generator cooler B, thermowell, local test	DG; J.1-1; 436'-0"	D-302-222; E-303-022; E-811-047	
ITE4508	SW	Service water from diesel generator cooler B, temperature element, type E thermocouple	DG; J.1-1; 436'-0"	D-302-222; E-303-022; E-811-047	02-01

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System [1]	Description	Bldg; [2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg
ITE2094	MS	Main steam, temperature element, computer input T2094A, 100 ohm RTD	IB; H4-4.4; 436'-0"	D-302-012; E-304-013; E-811-019
ITE3307A	FW	30" main feedwater header, temperature element, computer input, 100 ohm RTD	IB; G4-5.9; 436'-0"	D-302-083; E-304-084; -
ITE3307B	FW	30" main feedwater header, temperature element, computer input, 100 ohm RTD	IB; G4-5.9; 436'-0"	D-302-083; E-304-084; -
ITE3318	FW	Feedwater to steam generator A, temperature element, computer input T0418A, 100 ohm RTD	RB; RC-13; 436'-0"	D-302-083; E-304-085; E-811-004
ITE3320	FW	Feedwater to steam generator A, temperature element, type E thermocouple	AB; L-7.6; 436'-0"	D-302-083; E-304-085; E-811-013
ITE3322	FW	Feedwater to steam generator A, temperature element, computer input, 100 ohm RTD	RB; Q-4; 456'-0"	D-302-083; E-304-085; E-811-004
ITE3328	FW	Feedwater to steam generator B, temperature element, computer input T0438A, 100 ohm RTD	RB; RC-8; 436'-0"	D-302-083; E-304-085; E-811-004
ITE3330	FW	Feedwater to steam generator B, temperature element, computer input, type E thermocouple	IB; J1-4.4; 436'-0"	D-302-083; E-304-085; E-811-019

99-01

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System [1]	Description	Bldg; [2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
ITE3332	FW	Feedwater to steam generator B, temperature element, computer input, 100 ohm RTD	RB; RC-8; 453'-0"	D-302-083; E-304-085; E-811-004	02-01
ITE3338	FW	Feedwater to steam generator C, temperature element, computer input T0458A, 100 ohm RTD	RB; RC-7; 436'-0"	D-302-083; E-304-085; E-811-004	
ITE3340	FW	Feedwater to steam generator C, temperature element, computer input, type E thermocouple	IB; J6-2.8; 436'-0"	D-302-083; E-304-085; E-811-019	
ITE3342	FW	Feedwater to steam generator C, temperature element, computer input, 100 ohm RTD	RB; Q-1; 436'-0"	D-302-083; E-304-085; E-811-004	RN 12-030
ITW4470	SW	Service water to component cooling heat exchanger A, thermowell, local test	IB; H4-3.6 -	D-302-222; E-304-253; E-811-017	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System [1]	Description	Bldg; [2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg
ITE4471	SW	Service water from component cooling heat exchanger A, temperature element, type E thermocouple	IB; H4-4.4; 412'-0"	D-302-222; E-304-253; E-811-017
ITW4500	SW	Service water to component cooling heat exchanger B, thermowell, local test	IB; J1-3.6; 412'-0"	D-302-222; E-304-253; E-811-017
ITE4501	SW	Service water from component cooling heat exchanger B, temperature element, type E thermocouple	IB; J1-3.6; 412'-0"	D-302-222; E-304-253; E-811-017
ITE4467	SW	Cooling water from reactor building air handling heat exchange loop A, temperature element, computer input T4467A, post accident monitor, 100 ohm RTD	AB; M-7.7; 463'-0"	D-302-222; E-304-255; E-811-015
ITW4469	SW	Cooling water from reactor building air handling heat exchange loop A, temperature thermowell, local test	AB; M-7.7; 463'-0"	D-302-222; E-304-255; E-811-015
ITW4473	SW	Service water to HVAC mechanical water chiller A, thermowell, local test	IB; F1-5.2; 412'-0"	D-302-222; E-304-256; E-811-018
ITE4474	SW	Service water from HVAC mechanical water chiller A, temperature element, type E thermocouple	IB; F1-5.2; 412'-0"	D-302-222; E-304-256; E-811-018
ITW4482	SW	Service water to HVAC mechanical water chiller C, thermowell, local test	IB; F1-5.9; 412'-0"	D-302-222; E-304-256; E-811-018

99-01

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System [1]	Description	Bldg; [2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
ITE4484	SW	Service water from HVAC mechanical water chiller C, temperature element, type E thermocouple	IB; F1-5.9; 412'-0"	D-302-222; E-304-256; E-811-018	
ITE4497	SW	Cooling water from reactor building air handling heat exchange loop B, temperature element, computer input T4497A, post accident monitor, 100 ohm RTD	FB; Q-5.4; 463'-0"	D-302-222; E-304-256; E-811-034	
ITW4499	SW	Cooling water from reactor building air handling heat exchange loop B, thermowell, local test	FB; Q5-4.1; 463'-0"	D-302-222; E-304-256; E-811-034	
ITW4503	SW	Service water to HVAC mechanical water chiller B, thermowell, load test	IB; F1-6.8; 412'-0"	D-302-222; E-304-256; E-811-018	
ITE4504	SW	Service water from HVAC mechanical water chiller B, temperature element, type E thermocouple	IB; F1-6.8; 412'-0"	D-302-222; E-304-256; E-811-018	
ITE4514	SW	Service water from HVAC mechanical water chiller C, temperature element, type E thermocouple	IB; F1-5.9; 412'-0"	D-302-222; E-304-256; E-811-018	
ITW0410	RC	Loop 1 cold leg thermal well, TW-410	RB; RC-13; 412'-0"	E-302-601; E-304-601; -	RN 98-80
ITW0412	RC	Loop 1 cold leg well, TW-412	RB; RC-13; 412'-0"	E-302-601; E-304-601; E-811-003	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
ITW0412-1	RC	Loop 1 hot leg well, TW-412-1	RB; RC-15; 412'-0"	E-302-603; E-304-601; E-811-003	
ITW0412-2	RC	Loop 1 hot leg well, TW-412-2	RB; RC-15; 412'-0"	E-302-603; E-304-601; E-811-003	
ITW0412-3	RC	Loop 1 hot leg well, TW-412-3	RB; RC-15; 412'-0"	E-302-603; E-304-601; E-811-003	
ITW0413	RC	Loop 1 hot leg well, TW-413	RB; RC-15; 412'-0"	E-302-601; E-304-601; -	RN 98-80
ITW0420	RC	Loop 2 cold leg well, TW-420	RB; RC-8; 412'-0"	E-302-601; E-304-601; -	RN 98-80
ITW0422	RC	Loop 2 cold leg well, TW-422	RB; RC-8; 412'-0"	E-302-601; E-304-601; E-811-003	
ITW0422-1	RC	Loop 2 hot leg well, TW-422-1	RB; RC-9; 412'-0"	E-302-604; E-304-601; E-811-003	
ITW0422-2	RC	Loop 2 hot leg well, TW-422-2	RB; RC-9; 412'-0"	E-302-604; E-304-601; E-811-003	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	99-01
ITW0422-3	RC	Loop 2 hot leg well, TW-422-3	RB; RC-9; 412'-0"	E-302-604; E-304-601; E-811-003	
ITW0423	RC	Loop 2 hot leg well, TW-423	RB; RC-9; 412'-0"	E-302-601; E-304-601; -	RN 98-80
ITW0430	RC	Loop 3 cold leg well, TW-430	RB; RC-1; 412'-0"	E-302-601; E-304-601; -	RB 98-80
ITW0432	RC	Loop 3 cold leg well, TW-432	RB; RC-1; 412'-0"	E-302-601; E-304-601; E-811-003	
ITW0432-1	RC	Loop 3 hot leg well, TW-432-1	RB; RC-2; 412'-0"	E-302-605; E-304-601; E-811-003	
ITW0432-2	RC	Loop 3 hot leg well, TW-432-2	RB; RC-2; 412'-0"	E-302-605; E-304-601; E-811-003	
ITW0432-3	RC	Loop 3 hot leg well, TW-432-3	RB; RC-2; 412'-0	E-302-605; E-304-601; E-811-003	
ITW0433	RC	Loop 3 hot leg well, TW-433	RB; RC-2; 412'-0"	E-302-601; E-304-601; -	RN 98-80

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

99-01

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
ITW0450	RC	Pressurizer surge line well, TW-450	RB; RC-15; 412'-0"	E-302-602; E-304-601; -	RN 98-80
ITW0451	RC	Pressurizer spray line well, TW-451	RB; RC-15; 436'-0"	E-302-602; E-304-601; -	RN 98-80
ITW0452	RC	Pressurizer spray line well, TW-452	RB; RC-15; 436'-0"	E-302-602; E-304-601; -	RN 98-80
IPI7000	CC	Component cooling pump A, suction pressure indicator, local indication	IB; H4-3.6; 412'-0"	D-302-611; E-304-611; E-811-017	
IPI7002	CC	Component cooling pump A, discharge pressure indicator, local indication	IB; H7-3.6; 412'-0"	D-302-611; E-304-611; E-811-017	
IPI7010	CC	Component cooling pump B, suction pressure indicator, local indication	IB; H4-6.8; 412'-0"	D-302-611; E-304-611; E-811-018	
IPI7012	CC	Component cooling pump B, discharge pressure indicator, local indication	IB; H4-5.9; 412'-0"	D-302-611; E-304-611; E-811-018	
IPI7020	CC	Component cooling pump C, suction pressure indicator, local indication	IB; H4-5.9; 412'-0"	D-302-611; E-304-611; E-811-018	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
IPI7022	CC	Component cooling pump C, discharge pressure indicator, local indication	IB; H4-5.2; 412'-0"	D-302-611; E-304-611; E-811-018	
IPI7101	CC	Component cooling booster pump, suction header pressure indicator, local indication	IB; J-8.3; 412'-0"	D-302-612; E-304-611; E-811-012	
ITE7047	CC	Component cooling water from residual heat removal heat exchanger B	AB; L-11-5; 412'-0"	D-302-611; E-304-612; E-811-011	98-01
ITW7183	CC	Component cooling water to radioactive sample coolers, thermowell, local test	AB; L-9.5; 412'-0"	D-302-613; E-304-612; E-811-011	
ITW7203	CC	Component cooling water to spent fuel heat exchanger, thermowell, local test	AB; R-8-8; 388'-0"	D-302-613; E-304-613; E-811-007	
ITW7204	CC	Component cooling water to waste gas compressor, thermowell, local test	AB; N-8.8; 388'-0"	D-302-613; E-304-613; E-811-007	98-01
ITE7037	CC	Component cooling water from residual heat removal heat exchanger A	AB; K-11.5; 436'-0"	D-302-611; E-304-614; E-811-013	98-01
ITW7052	CC	Component cooling water from component cooling heat exchanger A, thermowell	IB; G4-4.4; 412'-0"	D-302-611; E-304-614; E-811-017	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
ITW7062	CC	Component cooling water from component cooling heat exchanger B, thermowell	IB; J1-5.2; 412'-0"	D-302-611; E-304-614; E-811-017	99-01
IPI7103A	CC	Component cooling booster pump A, discharge pressure indicator, local indication	IB; J-8.3; 412'-0"	D-302-612; E-304-614; E-811-018	
IPI7103B	CC	Component cooling booster pump B, discharge pressure indicator, local indication	IB; J-8.3; 412'-0"	D-302-612; E-304-614; E-811-018	99-01
IPI7103C	CC	Component cooling booster pump C, discharge pressure indicator, local indication	IB; J-8.3; 412'-0"	D-302-612; E-304-614; E-811-018	
ITW7234	CC	Component cooling water to recycle evaporator package, thermowell, local test	AB: N-9.5; 412'-0"	D-302-613; E-304-614: E-811-010	
ITE7188	CC	Component cooling water from seal water heat exchanger, temperature element, type E thermocouple	AB: N-9.5; 412'-0"	D-302-613; E-304-615; E-811-010	
ITW7193	CC	Component cooling water to letdown and seal water heat exchangers, thermowell, local test	AB; M-9.5; 412'-0"	D-302-613; E-304-615; E-811-010	
ITE7196	CC	Component cooling water from letdown heat exchanger, temperature element, type E thermocouple	AB; N-9.5; 412'-0"	D-302-613; E-304-615; E-811-010	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

99-01

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg
ITE7201	CC	Component cooling water from waste gas compressor A, temperature element, type E thermocouple	AB; N-9.5; 388'-0"	D-302-613; E-304-615; E-811-007
ITE7206	CC	Component cooling water from spent fuel pool heat exchanger A, temperature element, type E thermocouple	AB; Q-7.7; 388'-0"	D-302-613; E-304-615; E-811-007
ITE7211	CC	Component cooling water from waste gas compressor B, temperature element, type E thermocouple	AB; N-11.5; 388'-0"	D-302-613; E-304-615; E-811-007
ITE7216	CC	Component cooling water from spent fuel pool heat exchanger B, temperature element, type E thermocouple	AB; R-6.6; 388'-0"	D-302-613; E-304-615; E-811-007
ITE7221	CC	Component cooling water from waste processing system recombiner A, temperature element, type E thermocouple	AB; N-9.5; 388'-0"	D-302-613; E-304-615; E-811-007
ITE7226	CC	Component cooling water from recycle evaporator package, temperature element, type E thermocouple	AB; P-9.5; 412'-0"	D-302-613; E-304-615; E-811-010
ITI7227	CC	Component cooling water from recycle evaporator package, temperature indicator, local indication, well furnished by Pyco	AB; P-9.5; 412'-0"	D-302-613; E-304-615; E-811-010
ITE7231	CC	Component Cooling water from waste processing system recombiner B, temperature element, type E thermocouple	AB; N-11.5; 388'-0"	D-302-613; E-304-615; E-811-007

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg; [2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
ITE7236	CC	Component cooling water from waste evaporator package, temperature element, type E thermocouple	AB; M-8.8; 412'-0"	D-302-613; E-304-615; E-811-010	99-01
ITW7244	CC	Component cooling water to waste evaporator package, thermowell, local test	AB; N-8.8: 412'-0"	D-302-613; D-304-615; E-811-010	
ITE7246	CC	Component cooling water from residual heat removal pump A seal cooling, temperature element, type E thermocouple	AB; J-8.8; 374'-0"	D-302-614; E-304-615; E-811-006	
ITE7256	CC	Component cooling water from residual heat removal pump B seal cooling, temperature element, type E thermocouple	AB; L-8.8; 374'-0"	D-302-614; E-304-615; E-811-006	
ITW7107	CC	Component cooling water to reactor coolant pump thermal barrier cooler, thermowell, local test	IB; J6-3.6; 436'-0"	D-302-612; E-304-616; E-811-019	
ITE7108	CC	Component cooling water from excess letdown heat exchanger, temperature element, 100 ohm RTD	RB; RC-15; 436'-0"	D-302-612; E-304-617; E-811-004	
ITW7112	CC	Component cooling water to excess letdown heat exchanger and reactor coolant drain tank heat exchanger, thermowell, local test	RB; RC-13; 436'-0"	D-302-612; E-304-617; E-811-004	98-01
ITE7118	CC	Component cooling water from reactor coolant drain tank heat exchanger, temperature element, 100 ohm platinum RTD	RB; RC-17; 412'-0"	D-302-612; E-304-618; E-811-003	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
ITE7128	CC	Component cooling water from reactor coolant pump A upper bearing, temperature element, 100 ohm platinum RTD	RB; RC-12; 436'-0"	D-302-612; E-304-618; E-811-004	99-01
ITE7134	CC	Component cooling water from reactor coolant pump A lower bearing, temperature element, 100 ohm platinum RTD	RB; RC-12; 436'-0"	D-302-612; E-304-618; E-811-004	
ITE7140	CC	Component cooling water from reactor coolant pump A thermal barrier, temperature element, 100 ohm platinum RTD	RB; RC-12; 436'-0"	D-302-612; E-304-618; E-811-004	
ITE7148	CC	Component cooling water from reactor coolant pump B upper bearing, temperature element, 100 ohm platinum RTD	RB; RC-8; 436'-0"	D-302-612; E-304-618; E-811-004	
ITE7154	CC	Component cooling water from reactor coolant pump B lower bearing, temperature element, 100 ohm platinum RTD	RB; RC-8; 436'-0"	D-302-612; E-304-618; E-811-004	
ITE7160	CC	Component cooling water from reactor coolant pump B thermal barrier, temperature element, 100 ohm platinum RTD	RB; RC-8; 436'-0"	D-302-612; E-304-618; E-811-004	
ITE7168	CC	Component cooling water from reactor coolant pump C upper bearing, temperature element, 100 ohm platinum RTD	RB; RC-3; 436'-0"	D-302-612; E-304-618; E-811-004	
ITE7174	CC	Component cooling water from reactor coolant pump C lower bearing, temperature element, 100 ohm platinum RTD	RB; RC-3; 436'-0"	D-302-612; E-304-618; E-811-004	02-01

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
ITE7180	CC	Component cooling water from reactor coolant pump C thermal barrier, temperature element, 100 ohm platinum RTD	RB; RC-3; 436’-0"	D-302-612; E-304-618; E-811-004	
ITW0604A	RH	Residual heat removal pump, thermowell, TW-604A	AB; J-8.8; 374’-0"	E-302-641; E-304-644; -	RN 98-80
ITW0604B	RH	Residual heat removal pump, thermowell, TW-604B	AB; L-7.7; 374’-0"	E-302-641; E-304-644; -	RN 98-80
ITW0606A	RH	Residual heat exchanger outlet, thermowell, TW-606A	AB; K-9.5; 412’-0"	E-302-641; E-304-645; -	RN 98-80
ITW0606B	RH	Residual heat exchanger outlet, thermowell, TW-606B	AB; K-11.5; 412’-0"	E-302-641; E-304-645; -	RN 98-80
IPI7441	SF	Spent fuel cooling pump A, suction strainer pressure indicator, local indication	AB; R-6.6; 412’-0"	D-302-651; E-304-656; E-811-010	
IPI7442	SF	Spent fuel cooling pump A, suction pressure indicator, local indication	AB; R-6.6; 412’-0"	D-302-651; E-304-656; E-811-010	
IPI7444	SF	Spent fuel cooling pump A, discharge pressure indicator, local indication	AB; R-6.6; 412’-0"	D-302-651; E-304-656; E-811-010	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg; [2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
IPI7451	SF	Spent fuel cooling pump B, suction strainer pressure indicator, local indication	AB; R-7.7; 412'-0"	D-302-651; E-304-656; E-811-010	99-01
IPI7452	SF	Spent fuel cooling pump B, suction pressure indicator, local indication	AB; R-7.7; 412'-0"	D-302-651; E-304-656; E-811-010	99-01
IPI7454	SF	Spent fuel cooling pump B, discharge pressure indicator, local indication	AB; R-7.2; 412'-0"	D-302-651; E-304-656; E-811-010	99-01
IPI7362	SP	Reactor building spray pump A, suction pressure indicator, local indication	AB; J-8.8; 374'-0"	D-302-661; E-304-667; E-811-006	
IPI7366	SP	Reactor building spray pump A, discharge pressure indicator, local indication	AB; J-8.8; 374'-0"	D-302-661; E-304-667; E-811-006	
IPI7372	SP	Reactor building spray pump B, suction pressure indicator, local indication	AB; L-8.8; 374'-0"	D-302-661; E-304-667; E-811-006	
IPI7376	SP	Reactor building spray pump B, discharge pressure indicator, local indication	AB; L-8.8; 374'-0"	D-302-661; E-304-667; E-811-006	
ITW0139	CS	Excess letdown heat exchanger, thermowell, TW-139	RB; RC-17; 412'-0"	E-302-673; E-304-671; -	RN 98-80

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	99-01
ITW0140	CS	Regenerative heat exchanger letdown, thermowell, TW-140	RB; RC-1; 412'-0"	E-302-673; E-304-671; -	RN 98-80
ITW0144	CS	Letdown heat exchanger outlet, thermowell, TW-144	AB; M-9.5; 426'-6"	E-302-674; E-304-674; E-811-010	RN 98-80
ITW0381	TR	Letdown reheat heat exchanger outlet, thermowell, TW-381	AB; M-8.8; 426'-6"	E-302-676; E-304-674; -	RN 98-80
ITW0386	TR	Chemical and volume control system return header, test well, TW-386	AB; K-8.8; 426'-6"	E-302-676; E-304-675; -	RN 98-80
ITW0389	TR	Ion exchange outlet, thermowell, TW-389	AB; J-8.8; 426'-6"	E-302-676; E-304-675; -	RN 98-80
ITW0133	CS	Seal water return, thermowell, TW-133	AB; N-7.7; 412'-0"	E-302-673; E-304-676; -	RN 98-80
ITW0136	CS	Seal water heat exchanger, thermowell, TW-136	AB; N-9.5; 426'-6"	E-302-675 E-304-681; -	RN 98-80
IPI7351	SP	Sodium hydroxide storage tank, pressure indicator, local indication	YD; K-11.5; 436'-0"	D-302-661; E-304-728; E-811-011	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
IFS1900A	LD	Reactor building cooling unit, leak detection flow switch, high flow alarm, south cooling units	RB; RC-9; 463'-0"	-; E-304-837; E-811-005	
IFS1900B	LD	Reactor building cooling unit, leak detection flow switch, high flow alarm, north cooling units	RB; RC-17; 463'-0"	-; E-304-837; E-811-005	99-01
ITW9011A	VU	Chilled water chiller A exit, thermowell, local test	IB; F1-5.2; 412'-0"	D-302-841; E-304-841; E-811-018	99-01
ITW9011B	VU	Chilled water chiller B exit, thermowell, local test	IB; F1-6.8; 412'-0"	D-302-841; E-304-841; E-811-018	99-01
ITE9013	VU	Chilled water chiller A exit, temperature element, 100 ohm RTD	IB; F1-5.2; 412'-0"	D-302-841; E-304-841; E-811-018	99-01
ITE9023	VU	Chilled water chiller B exit, temperature element, 100 ohm RTD	IB; F1-7.5; 412'-0"	D-302-841; E-304-841; E-811-018	99-01
ITE9033A	VU	Chilled water chiller C exit, temperature element, 100 ohm RTD	IB; F1-6.8; 412'-0"	D-302-841; E-304-841; E-811-018	99-01
ITE9033B	VU	Chilled water chiller C exit, temperature element, 100 ohm RTD	IB; F1-6.8; 412'-0"	D-302-841; E-304-841; E-811-018	99-01

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED
PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
IPI9007A	VU	Chilled water pump A suction, pressure indicator, local indication	IB; F1-5.9; 412'-0"	D-302-841; E-304-845; E-811-018	
IPI9007B	VU	Chilled water pump B suction, pressure indicator, local indication	IB; G4-5.9; 412'-0"	D-302-841; E-304-845; E-811-018	
IPI9009A	VU	Chilled water pump A discharge, pressure indicator, local indication	IB; F1-5.9; 412'-0"	D-302-841; E-304-845; E-811-018	
IPI9009B	VU	Chilled water pump B discharge, pressure indicator, local indication	IB; G4-5.9; 412'-0"	D-302-841; E-304-845; E-811-018	
ITE9017	VU	Chilled water supply header A, temperature element, computer input T9017S, 100 ohm RTD	IB; G4-5.9; 412'-0"	D-302-841; E-304-845; E-811-018	02-01
ITW9018A	VU	Chilled water supply header A, thermowell, local test	IB; G4-5.2; -	D-302-841; E-304-845; E-811-018	
ITW9018B	VU	Chilled water supply header B, thermowell, local test	IB; G4-7.5; 412'-0"	D-302-841; E-304-845; E-811-018	
ITE9027	VU	Chilled water supply header B, temperature element, computer input T9027A, 100 ohm RTD	IB; G4-7.5; 412'-0"	D-302-841; E-304-845; E-811-018	

TABLE 3.10-3 (Continued)

SEISMICALLY QUALIFIED

PIPE MOUNTED INSTRUMENTS

Instr. No.	System[1]	Description	Bldg;[2] Column No.; Elev	Flow Diagram; Piping Dwg; Location Dwg	
					99-01
IPI9028	VU	Chilled water pump C suction, pressure indicator, local indication	IB; G4-5.9; 412'-0"	D-302-841; E-304-845; E-811-018	
IPI9029	VU	Chilled water pump C discharge, pressure indicator, local indication	IB; G4-5.9; 412'-0"	D-302-841; E-304-845; E-811-018	02-01

NOTE [1] System:

CC - Component cooling water
CS - Chemical and volume control
EF - Emergency feedwater
FW - Main feedwater
MS - Main steam
RC - Reactor coolant
RH - Residual heat removal
SP - Reactor building spray
SW - Service water 02-01
TR - Thermal regeneration
VU - Chilled water 02-01

NOTE [2] Building:

AB - Auxiliary building
DG - Diesel generator building
FB - Fuel handling building
IB - Intermediate building
RB - Reactor building
SW - Service water pumphouse
YD - Yard

99-01

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section describes the program for environmental qualification of electrical equipment. The Virgil C. Summer Nuclear Station does not have a licensing commitment to environmentally qualify safety-related mechanical equipment to the level of detail required to qualify electrical equipment. However, mechanical equipment has been specified and procured to satisfy requirements which assure that it can withstand the normal, abnormal, accident, and post-accident conditions to which it may be subjected. The environmental qualification program for the V. C. Summer Nuclear Station identifies the electrical equipment to be qualified, defines the environmental conditions under normal, abnormal, accident, and post-accident operating conditions. The program also documents the qualification tests and analyses employed to demonstrate the equipment's capability to perform design safety functions including post-accident monitoring when exposed to normal, abnormal, accident, and post-accident environments as applicable. Seismic qualification is addressed in Section 3.10 for mechanical and electrical equipment.

3.11.1 ENVIRONMENTAL CONDITIONS AND EQUIPMENT IDENTIFICATION

This section identifies: 1) the environmental design basis for electrical equipment, including the definition of the normal, abnormal, accident, and post-accident environments, and 2) the systems and electrical equipment that are required to perform a design safety function, including Regulatory Guide 1.97 [24] monitoring.

3.11.1.1 Environmental Conditions

Electrical equipment location for environmental qualification purposes is defined by environmental zones. Various environmental zones are encountered within the plant's building(s) (i.e. Auxiliary Bldg., Containment Bldg., Intermediate Bldg., etc.). The environmental zone boundaries are shown on plant layout drawings, called "Environmental Zone Maps", as documented by drawings SS-021-001 through SS-021-017 [3]. The zone boundaries shown on drawings SS-021-001 through SS-021-017 were determined based on contiguous areas with similar environmental conditions.

An equipment qualification database with environmental zone information as zones documented by drawing S-021-018 [4] provides a list of the environmental zones and conditions, including the normal, abnormal, and accident (including post-accident) environmental conditions. Environmental data is provided for the temperature, pressure, relative humidity, and radiation parameters for each environmental condition postulated to occur within each zone. Definitions used in determining the environmental conditions are as follows:

a. Normal Conditions - planned, purposeful, unrestricted reactor operating modes that include startup, power range and hot standby (condenser available), shutdown, and refueling modes.

b. Abnormal Conditions - any deviation from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operation impairment; planned testing including preoperational tests are also considered abnormal conditions(loss of non-safety related HVAC is an example of an abnormal condition).

c. Accident Conditions - a single event not reasonably expected during the course of plant operations that has been hypothesized for analysis purposes or postulated from unlikely but possible situations or that has the potential to cause a release of radioactive material (a reactor coolant pressure boundary rupture may qualify as an accident; a fuel cladding defect does not). Postulated accident conditions include those resulting from: a Main Steam Line Break (MSLB) inside or outside of containment; a Loss of Coolant Accident (LOCA); a High Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC); or other Line Break Accidents. Accident conditions are calculated for a post-accident period sufficient to ensure that steady state conditions have been reached.

The environmental parameters listed in drawing S-021-018 are based on verified design calculations and do not include margins required in qualification testing or analyses as described in Section 3.11.2. The design basis used for preventing the loss of ventilation for some zones is discussed in Section 3.11.4. The basis for the estimated chemical and radiation environmental conditions is discussed in Section 3.11.5. Environmental conditions listed in drawing S-021-018, and environmental zone boundaries shown on Environmental Zone Maps, drawings SS-021-001 through SS-021-017, will be revised when a modification in equipment or components or the results of temperature monitoring affect environmental qualification conditions.

Technical Requirements for Quality Related (QR) steam propagation barriers / components which are required to maintain area environmental conditions for equipment qualification (EQ) purposes are provided in plant procedures which specify the technical criteria for design, procurement, installation/maintenance and inspection of these QR components. These barriers/components segregate EQ zones to ensure that the conditions for which the equipment qualification were based on are not exceeded. RN 17-011

To aid in the application of the qualification program acceptance criteria that is discussed in Section 3.11.2, the environmental zones listed in S-021-018 drawings have been classified as either a harsh or mild environment based on the following definitions:

Harsh Environments - Those zones or areas where the environmental conditions significantly exceed the normal or abnormal range as a result of a DBE. A harsh environment area is an area or zone of the plant where one or more of the following environmental service conditions exist:

- Radiation:

  TID (normal plus accident) > 1.0 E05 Rads (1.0 E04 for electronics) and the accident dose is greater than or equal to normal.

- Humidity

  Relative Humidity = 100%

- Pressure:

  ≥ 0.1 psig above atmospheric

- Temperature:

  The accident temperature is ≥ a 20% change from the normal maximum temperature.

Mild Environments - Those zones or areas where the environmental conditions do not significantly exceed the normal or abnormal range as a result of a DBE. A mild environment area is an area or zone of the plant where the following environmental service conditions exist:

- Radiation:

  TID (normal plus accident) ≤ 1.0 E05 Rads; or, if greater than 1.0 E05, the accident dose is less than TID normal. The 1.0 E05 limit is reduced to 1.0 E04 for electronics.

- Humidity:

  Relative Humidity < 100%

- Pressure:

  < 0.1 psig above atmospheric

- Temperature:

  The accident temperature is < a 20% change from the normal maximum temperature.

3.11.1.2 Equipment Identification

1. General

   Electrical equipment to be qualified includes equipment associated with systems that are essential to:

   a. Emergency reactor shutdown.

   b. Containment isolation.

c. Reactor core cooling.

d. Containment heat removal.

e. Reactor heat removal.

f. Preventing significant release of radioactive material to the environment, or

g. Provide Regulatory Guide 1.97 Category 1 and 2 indicating and post-accident monitoring. Non 1E, Regulatory Guide 1.97 Category 2 equipment installed in a geographic mild environment is excluded from the mild EQ Program

This is equipment that:

a. Performs the previous functions automatically,

b. Is used by the operator to perform these functions manually, or

c. The failure of which can prevent the satisfactory accomplishment of one or more of the previous safety functions.

"Safety-related" equipment is categorized in three groups by design safety function:

a. Safety-related electrical equipment - designated as "Class 1E" per IEEE Standard 308. [1]

b. Safety-related "Active" Mechanical Equipment - that equipment which must move or change position to perform its design safety function (examples are pumps, motor operated valves, and safety relief valves).

c. Safety-related "Passive" Mechanical Equipment - that equipment which must only maintain its pressure integrity to perform its design safety function (examples are tanks, heat exchangers, and manual valves).

The design safety functions for specific equipment items are discussed on a system basis in Chapters 3 (Sections 3.4, 3.5, and 3.6), 5, 6, 7, 8, 9, 12, and 15.

2. List of Equipment

The Equipment Qualification Database (EQDB) [2] identifies electrical equipment which requires environmental qualification. The Database provides specific informative data such as: Tag number, manufacturer, model number, purchase order number, equipment qualification documentation package number, etc. It is a computerized Database loaded on the VCSNS VAX System. Refer to Section 3.11.3.1 for discussion.

An overview of major safety-related equipment and components other than Class 1E equipment that are required to function during and/or subsequent to the design basis accidents are listed in Tables 3.11-1 for equipment and components located inside the Reactor Building, and 3.11-2 for equipment and components located outside the Reactor Building. The remainder of the information required to comply with General Design Criterion (GDC) 4 is discussed in Sections 3.11.1.1 through 3.11.3.3.

3. Equipment Categorization

   a. For environmental qualification, "Environmentally Qualified" electrical equipment is grouped into one or more categories or designations based on the equipment's functional design requirements. A breakdown of environmental categories and designations including their respective definitions are as follows:

      1. Category A1:

         Equipment that could experience the environmental conditions of design basis (LOCA) accidents for which it must function to mitigate said accident. It must be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.

      2. Category A2:

         Equipment that could experience the environmental conditions of design basis line break accidents, including Main Steam Line Break (MSLB), High Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC), and/or other line breaks for which it must function to mitigate said accidents. It must be qualified to demonstrate operability in its specifically applicable accident environment for the time required for accident mitigation with safety margin to failure.

      3. Category A1 *:

         Equipment that could experience increased radiation exposure due to Post-LOCA recirculation for which it must function to mitigate said accident. It must be qualified to demonstrate operability in the radiation environment for the time required for accident mitigation with safety margin to failure.

4. Category B1:

   Equipment that could experience environmental conditions of design basis (LOCA) accidents through which it need not function for mitigation of said accident, but through which it must not fail in a manner detrimental to plant safety or accident mitigation. It must be qualified to demonstrate the capability to withstand any LOCA accident environment for the time during which it must not fail with safety margin to failure.

5. Category B2:

   Equipment that could experience the environmental conditions of design basis line break accidents, including Main Steam Line Break (MSLB), High Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC), and/or other line breaks for which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation. It must be qualified to demonstrate the capability to withstand any such specifically applicable accident environment for the time which it must not fail with safety margin to failure.

6. Category B1 *:

   Equipment that could experience increased radiation exposure due to Post-LOCA recirculation through which it need not function for mitigation of said accident, but through which it must not fail in a manner detrimental to plant safety or accident mitigation. It must be qualified to demonstrate the capability to withstand any radiation environment for the time during which it must not fail with safety margin to failure.

7. Category C1:

   Equipment that could experience environmental conditions of design basis (LOCA) accidents through which it need not function for mitigation of said accident, and whose failure is deemed not detrimental to plant safety or accident mitigation. It deemed not detrimental to plant safety or accident mitigation. It need not be qualified for any LOCA accident environment, but must be qualified for the non-accident environment.

8. Category C2:

   Equipment that could experience the environmental conditions of design basis line break accidents, including Main Steam Line Break (MSLB), High Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC), and/other line breaks through which it need not function for mitigation of said accidents and whose failure is deemed not detrimental to plant safety or accident mitigation. It need not be qualified for these accident environments, but must be qualified for the non-accident environment.

9. Category C1 *:

   Equipment that could experience increased radiation exposure due to Post-LOCA recirculation through which it need not function for mitigation of said accident and whose failure is deemed not detrimental to plant safety or accident mitigation. It need not be qualified for any Post-LOCA harsh accident environment, but must be qualified for the non-accident environment. Also equipment that could experience increased radiation exposure due to Post-LOCA recirculation but has completed its functional requirement prior to the environment becoming harsh and whose failure (after completing any required function) is deemed not detrimental to plant safety or accident mitigation need not be qualified for any Post-LOCA harsh accident environment, but must be qualified for the non-accident and Post-LOCA mild accident environments.

10. Category D:

    Equipment that would not experience environmental conditions of design basis accidents and that must be qualified to demonstrate operability in the normal and abnormal service environment. This equipment is located outside containment.

11. Designation QR-H:

    Equipment that is quality related and requires harsh environment qualification per design documents.

12. Designation QR-M:

    Equipment that is quality related and requires mild environment qualification per design documents.

b. Environmental categories for Regulatory Guide 1.97 equipment are similar to those previously identified for other safety-related electrical equipment except that a prefix letter E is used.

   A breakdown of Regulatory Guide 1.97 environmental categories and respective definitions are as follows:

   1. Category EA1:

      Equipment that could experience the environmental conditions of design basis (LOCA) accidents for which it must function to provide Regulatory Guide 1.97 monitoring information to the operator. It must be qualified to demonstrate operability in the accident environment for the time required with safety margin to failure.

   2. Category EA2:

      Equipment that could experience the environmental conditions of design basis line break accidents, including Main Steam Line Break (MSLB), High Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC), and/or other line breaks for which it must function to provide Regulatory Guide 1.97 monitoring information to the operator. It must be qualified to demonstrate operability in its specific applicable accident environment for the time required with safety margin to failure.

   3. Category EA1:*

      Equipment that could experience increased radiation exposure due to Post-LOCA recirculation for which it must function to provide RG 1.97 monitoring information to the operator. It must be qualified to demonstrate operability in the radiation environment for the time required with safety margin to failure.

   4. Category EB1:

      Equipment that could experience environmental conditions of Design Basis (LOCA) accidents through which it need not function to provide RG 1.97 monitoring information to the operator, but through which it must not fail in a manner detrimental to any related RG 1.97 monitoring function. It must be qualified to demonstrate the capability to withstand any such accident environment for the time during which it must not fail with safety margin to failure.

5. Category EB2:

   Equipment that could experience the environmental conditions of design basis line break accidents, including Main Steam Line Break (MSLB), High Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC), and/or other line breaks for which it need not function to provide RG 1.97 monitoring information to the operator, but through which it must not fail in a manner detrimental to any related RG 1.97 monitoring function. It must be qualified to demonstrate the capability to withstand any such specifically applicable accident environment for the time during which it must not fail with safety margin to failure

6. Category EB1:*

   Equipment that could experience increased radiation exposure due to Post-LOCA recirculation through which it need not function to provide RG 1.97 monitoring information to the operator, but through which it must not fail in a manner detrimental to any related RG 1.97 monitoring function. It must be qualified to demonstrate the capability to withstand any such accident environment for the time during which it must not fail with safety margin to failure.

7. Category EC1:

   Equipment that could experience the environmental conditions of design basis (LOCA) accidents through which it need not function for Regulatory Guide 1.97 monitoring purposes and whose failure is not deemed detrimental to the Regulatory Guide 1.97 monitoring function. It need not be qualified for a LOCA accident environment, but must be qualified for the non-accident environment.

8. Category EC2:

   Equipment that could experience the environmental conditions of design basis line break accidents, including Main Steam Line Break (MSLB), High Energy Line Break/Superheated Blowdown Outside Containment (HELB/SBOC), and other line breaks through which it need not function for Regulatory Guide 1.97 monitoring purposes and whose failure is deemed not detrimental to the Regulatory Guide 1.97 monitoring function. It need not be qualified for these accident environments, but must be qualified for the non-accident environment.

9. Category EC1:*

   Equipment that could experience increased radiation exposure due to Post-LOCA recirculation through which it need not function to provide RG 1.97 monitoring information to the operator and whose failure is deemed not detrimental to the RG 1.97 monitoring function. It need not be qualified for any Post-LOCA harsh accident environment, but must be qualified for the non-accident environment.

   Also equipment that could experience increased radiation exposure due to Post-LOCA recirculation but has completed its RG 1.97 monitoring requirement prior to the environment becoming harsh, and whose failure (after completing any required function) is deemed not detrimental to the RG 1.97 monitoring function. It need not be qualified for any Post-LOCA harsh accident environment, but must be qualified for the non-accident and the Post-LOCA mild accident environment.

10. Category ED:

   Equipment that would not experience environmental conditions of design basis accidents and that must be qualified to demonstrate operability for its Regulatory Guide 1.97 monitoring function in its normal and abnormal service environment. This equipment is located outside containment.

3.11.2 ENVIRONMENTAL QUALIFICATION PROGRAM ACCEPTANCE CRITERIA

This section describes the environmental qualification program acceptance criteria that were employed to meet the following general requirements:

a. The equipment was designed to have the capability of performing its design safety functions under postulated normal, abnormal, accident, and post-accident environments for the length of time for which its function is required plus margin.

b. The equipment environmental capability was demonstrated by appropriate testing, analyses, and/or operating experience.

c. A quality assurance program meeting the requirements of 10CFR50, Appendix B, was established and implemented to provide assurance that all requirements have been satisfactorily accomplished.

The Virgil C. Summer Nuclear Station is committed to qualification of electrical equipment requiring harsh qualification in accordance with NUREG-0588 [5], Cat. II, which relates to IEEE 323-1971 [6] for the original plant design. However, some equipment was qualified to IEEE 323-1974 [7] (NUREG 0588, Cat. 1) requirements. New and replacement electrical equipment requiring harsh qualification is governed by the current regulations of 10CFR50.49. The status of electrical equipment qualification in accordance with the applicable IEEE Standards and Regulatory requirements is documented in Virgil C. Summer Nuclear Station Equipment Qualification Documentation Packages (EQDP's) for equipment which requires hard qualification and Equipment Qualification Files (EQF's) for equipment which requires mild qualification. Refer to Section 3.11.3 for further discussion.

3.11.2.1 Conformance with Regulatory Requirements

Conformance with General Design Criteria 1, 4, 23, and 50 is discussed in Section 3.1.

3.11.2.1.1 10CFR50 Appendix A Criterion 4 [8] - Environmental and Missile Design Bases

The scope of electrical equipment and the environmental requirements for GDC 4 are addressed in Section 3.11.1. Refer to Sections 3.5 and 3.6 for discussions related to missile protection.

Electrical equipment that is required to perform a design safety function is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents in accordance with GDC 4.

3.11.2.1.2 10CFR50 Appendix B [9]

The Electrical Equipment Environmental Qualification Program is in compliance with the Virgil C. Summer Nuclear Station Quality Assurance Program. The Quality Assurance Program meets the requirements of 10CFR50, Appendix B.

3.11.2.1.3 Regulatory Guide 1.89 [10]

Compliance with Regulatory Guide 1.89, which pertains to the qualification of Class 1E electrical equipment, is also discussed in Appendix 3A.

Class 1E electrical equipment is qualified in accordance with IEEE 323-1974, as endorsed by Regulatory Guide 1.89 with the following exceptions:

a. In cases where the qualification has been demonstrated in accordance with IEEE 323-1971, documentation of that qualification is maintained in the form of auditable file packages. Refer to Section 3.11.3.2 for discussion.

b. Specific criteria for assessing the acceptability of the original scope of the environmental qualification program for safety related electrical equipment requiring harsh qualification is provided by NUREG-0588 Category II, as related to IEEE 323-1971, or Category I as related to IEEE 323-1974.

Specific criteria for assessing the acceptability of the environmental qualification program for electrical equipment added or replaced in harsh environments are provided by 10CFR50.49 [11], dated January 17, 1983.

c. Mild environment equipment qualification by test and/or analysis is not a requirement of 10CFR50.49. Mild environment equipment qualification files are established to provide design, procurement, and maintenance information in a readily accessible form.

3.11.2.1.4 Regulatory Guides 1.30 [12], 1.40 [13], 1.63 [14], and 1.73 [15]

The detailed criteria contained in these documents as they relate to environmental qualification should be used in conjunction with the more comprehensive criteria of NUREG-0588 for evaluating the respective equipment environmental qualification.

Compliance with these Regulatory Guides is discussed in Appendix 3A.

3.11.2.1.5 Regulatory Guide 1.97 [24]

Regulatory Guide 1.97 imposes environmental qualification requirements on electrical components used for monitoring certain plant parameters after an accident. Design and qualification criteria for Regulatory Guide 1.97 instrumentation, as described in Table 1 of the Regulatory Guide, states that the instrumentation should be environmentally qualified in accordance with Regulatory Guide 1.89 and the methodology as described in NUREG-0588.

Regulatory Guide 1.97 describes three equipment categories for design and equipment qualification requirements. V. C. Summer Nuclear Station Regulatory Guide 1.97 category 1 and 2 equipment conforms to Regulatory Guide 1.89 environmental qualification requirements. Regulatory Guide 1.97 Category 3 equipment has no special regulatory requirements but the regulatory position is that the equipment should be of high quality commercial grade and should be selected to withstand the specified service environment.

3.11.2.2 <u>Qualification Methodologies for Safety-Related Electrical Equipment</u>

Safety-related electrical equipment including Reg. Guide 1.97 monitoring equipment requiring harsh environmental qualification supplied by Westinghouse under the NSSS contract is qualified as outlined in Section 3.11.2.2.1.

All other safety-related electrical equipment including Reg. Guide 1.97 monitoring equipment requiring harsh environmental qualification is qualified using the methodologies of Section 3.11.2.2.2. Safety-related electrical equipment requiring mild environmental qualification is qualified as discussed in Exception c to Regulatory Guide 1.89 (refer to Section 3.11.2.1.3).

The documentation of the application of the methodologies for the specific equipment identified in Section 3.11.1.2, to demonstrate qualification to the environmental conditions defined in Section 3.11.1.1, is presented in Section 3.11.3.

3.11.2.2.1 Qualification Tests and Analyses Applicable to the NSSS Electrical Equipment

Westinghouse has qualified its NSSS safety-related electrical equipment in accordance with IEEE-323-1971. The Westinghouse supplemental qualification program (Reference 16) is an NRC approved seismic and environmental qualification program, as stated in the Staff letter, D. B. Vassallo to C. Eicheldinger dated November 19, 1975. Mechanical equipment design basis considerations are described in Chapters 5, 6, 9, and 10. Mechanical and electrical components have been identified and classified relative to their safety classification in Section 3.2.

Comprehensive testing and/or analysis is conducted for those electrical equipment and components which are required to function during and subsequent to any of the design basis accidents and that experience hostile environments. The program consists of performance tests of individual pieces of equipment in the manufacturer’s shop, integrated tests of the system as a whole in the field, and periodic inspection and tests of the activation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime.

The initial qualification tests of individual components and the integrated tests of the systems as a whole complement each other to assure performance of the system as designed and to prove proper operation of the actuation circuitry. For engineered safeguard features (ESF) equipment located inside the Reactor Building, qualification testing and/or analysis is performed under the effects of the conservative post accident temperature, pressure, humidity, radiation, and chemical environment, when applicable. Routine periodic inspection and testing of ESF equipment is performed as outlined in Technical Specifications. RN 01-113

Chapter 6.0 describes the containment temperature and pressure response to various sizes of in-containment main steam line ruptures. Qualified equipment located inside the Reactor Building is required to provide protection in the unlikely event of one of these breaks.

For the larger steam line breaks evaluated, the steam line pressure instrumentation, which is located outside containment, will initiate safety injection on low steam line pressure.

3.11.2.2.2 Qualification Tests and Analyses Applicable to the BOP Electrical Equipment

Balance of plant electrical equipment, including cabling, is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the location of the equipment. The environmental conditions considered include those expected during normal operation, maintenance, testing, and, if applicable, post accident periods.

The ESF mechanical and electrical equipment and instrumentation associated with balance of plant systems inside the Reactor Building are designed to perform required functions under the conservative accident and post-accident temperature, pressure, humidity, radiation, and chemical conditions.

Where design of balance of plant equipment to withstand dynamic effects of missiles, pipe whip, and jet forces was impractical, barriers were designed to protect such equipment (refer to Sections 3.5 and 3.6).

Safety-related equipment located outdoors is either qualified for expected environmental conditions or is protected from such conditions.

Qualification of BOP electrical equipment is accomplished by type testing, analysis, and/or documented operating experience. Electrical equipment requiring harsh environmental qualification is qualified in accordance with IEEE 323 and ancillary daughter standards (e.g., IEEE Std. 317 [17], 334 [18], 382 [19], 383 [20]). Although type testing is the preferred method of qualification, equipment qualification usually involves some combination of the three methods. The qualification methods used depend on a number of factors, including:

1. Material used in construction of the equipment.

2. Applicable normal, abnormal, accident, and post-accident environmental conditions.

3. Operational requirements (during and after accidents).

4. Nature of safety function(s).

5. Size of equipment.

6. Dynamic characteristics of expected failure modes (structural or functional).

In general, analysis is used to supplement test data, although equipment requiring mild environmental qualification and simple components may lend themselves to analysis in lieu of full scale testing. The role of operating experience is generally limited to aiding in determining realistic performance goals.

Equipment samples selected for qualification are of the same basic design and materials as the equipment to be installed at the Virgil C. Summer Nuclear Station. The sample is manufactured using similar techniques and processes as those used for the installed equipment. Any significant variations or deviations are noted in the qualification results with justification provided as necessary.

The list of electrical equipment subject to environmental qualification is documented by the Equipment Qualification Database (EQDB). The EQDB lists the electrical equipment as identified in Section 3.11.1.2. NUREG-0588 sets forth NRC positions in implementation of IEEE 323-1971 and 1974 versions of the "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" and was used as the basis for assessing the acceptability of the original scope of the environmental qualification program. Electrical equipment for replacements or modifications to the plant is procured under the guidance of 10CFR50.49 and is qualified under the requirements of IEEE 323-1974 and Regulatory Guide 1.89, if the equipment requires harsh environmental qualification.

3.11.2.2.2.1 Main Steam Line Break Inside Containment Equipment Qualification

Main steam line breaks (MSLB) inside containment have previously been discussed in Section 6.2.

The composite temperature profiles for environmental zones subject to MSLB conditions are presented in drawing S-021-018. Electrical equipment located in these environmental zones and required to mitigate the MSLB accident has been qualified to these temperature profiles as documented in the applicable EQDP’s.

3.11.2.2.2.2 Main Steam Line Break Outside Containment Equipment Qualification

Main steam line breaks (MSLB) outside containment have previously been discussed in Section 3.6. However, as a result of Information Notice No. 84-90 [21], South Carolina Electric and Gas Company has elected to address the HELB/SBOC (High Energy Line Break/Superheated Blowdown Outside Containment) accident scenario as a Virgil C. Summer Plant requirement and as a modification to the previously postulated MSLB accident scenario. A complete analysis of the HELB/SBOC environmental conditions and equipment qualification is presented in Reference 22. An increase in temperature in certain areas of the East and West Penetration Access areas and the Intermediate Building is the only significant environmental change resulting from postulated HELB/SBOC ruptures relative to that of the MSLB accident environment previously evaluated and identified in Section 3.6. All other environmental conditions remain unchanged.

Steamline breaks resulting in HELB/SBOC can only occur in the 436 ft. floor elevation of the East and West Penetration Access areas, 436 ft. floor elevation of the Intermediate Building, and in the Turbine Building. Breaks postulated in the 4-inch steamlines supplying the turbine driven EFW Pump result in a break area too small to lead to a superheated steam discharge per Westinghouse WCAP-10961 [23]; therefore, the licensing analysis for the 4-inch line remains unchanged. HELB/SBOC resulting from postulated breaks in large steam lines need not be addressed for the Turbine Building since equipment in this area is not required in order to mitigate the consequences of an HELB/SBOC.

For each environmental zone in which the ambient temperature profile for the HELB/SBOC exceeds that previously established for a MSLB, a composite temperature profile was generated. The composite temperature profiles for these environmental zones are presented in drawing S-021-018. Electrical equipment located in these environmental zones and required to mitigate the HELB/SBOC accident has been qualified to these temperature profiles.

Qualification of the required equipment in superheat high energy line break environmental zones was accomplished by a series of detailed engineering analyses. In some cases, these analyses are based on specific as-built hardware and location specific transient heat transfer analyses. Qualification for each component is based on qualification test data, but in some cases this data has been extrapolated using acceptable analytical techniques. The evaluations specifically address the higher temperature effect of the HELB/SBOC accident and the capability of the equipment to withstand the accident conditions for the operating time required to perform its safety function. These evaluations have demonstrated that the required equipment as installed, is environmentally qualified for the postulated environmental conditions.

3.11.2.2.3 QUALIFICATION MAINTENANCE

Qualification is not a guarantee of performance for each component of a system. It is rather, assurance that the system can perform its safety function under all specified service conditions. Maintenance of the qualified status of harsh environment electrical equipment requires scheduled maintenance to prevent components from exceeding their qualified life, and periodic testing to locate components that may have failed, or be near failure.

Scheduled maintenance is also performed on mild environment electrical equipment to ensure proper operation of the equipment throughout the established service life.

Scheduled maintenance activities required to maintain harsh and mild environment equipment qualification is specified for all safety-related electrical equipment. These activities are documented in the Equipment Qualification Database.

3.11.3 QUALIFICATION TEST RESULTS

This section addresses the qualification test results applicable to Nuclear Steam Supply System and Balance of Plant Electrical equipment. This equipment is required to function under anticipated normal operating conditions, and/or is required to function to mitigate the consequences of design basis accidents, including LOCA and MSLB inside containment and MSLB outside containment. In addition to the qualification results provided in this section, a complete analysis of the High Energy Line Break/Superheated Blowdown Outside Containment accident environmental conditions, as discussed in Section 3.11.2.2.2.2, is presented in Reference 22.

The results of the qualification program for each type of electrical equipment are recorded in the applicable Equipment Qualification Documentation Packages (EQDP's) and/or Equipment Qualification Files (EQF's). The collection of various computer data files containing information relative to equipment qualification and qualified equipment are included in an Equipment Qualification Database (EQDB). Electrical equipment and data relative to environmental qualification are listed in the EQDB.

Westinghouse NSSS supplied safety-related equipment, which is required to function to mitigate the consequences of a postulated accident and which may be exposed to the elevated environmental conditions that may result from the accident, is qualified by Westinghouse under the 1971 version of IEEE-323. The Westinghouse supplemental qualification program (Reference 16) has been accepted by the NRC staff as meeting the requirements of IEEE 323-1971 in NRC letter D. B. Vassallo to C. Eicheldinger dated November 19, 1975.

A portion of the Westinghouse supplied electrical equipment has subsequently been upgraded to the qualification requirements of IEEE 323-1974, by WCAP 8587 and WCAP 8687. This equipment upgrade has been documented in their respective EQDP's and EQF's.

3.11.3.1 Qualified Equipment List

The "Qualified Equipment/Components and Materials" list provides a listing of equipment, components, and materials for which environmental qualification is maintained. The list includes items subject to both mild and harsh environments and comprises two sections of the EQDB. One section is arranged alphanumerically by equipment number while the other section is arranged alphanumerically by system.

3.11.3.2 Auditable File

The auditable files are arranged in equipment qualification documentation packages or equipment qualification files by Qualification File Number. The "Equipment Qualification File Index" (part of EQDB) provides a listing of the EQDP's and EQF's and specifies the components to which each qualification file applies.

3.11.3.3 Master Equipment List

The "Master Equipment List" is contained within the EQDB and provides a listing of equipment and materials to which the requirements of 10CFR50.49 for environmental qualification of electrical equipment subject to harsh environment applies.

3.11.4 LOSS OF VENTILATION

Safety-related electrical equipment and components, as identified in Section 3.11.1.2, are located in areas which are mechanically cooled or ventilated by safety-related or quality related HVAC systems.

Safety-related HVAC systems providing cooling or ventilation are designed to satisfy the following considerations:

1. Seismic Category I requirements.

2. Redundant active system components are provided, as required.

3. Independent and redundant Class 1E power sources are provided.

4. Arrangement is such that single failure of an active or passive component does not result in loss of required cooling function.

In certain cases, such as rooms housing only one train of safety-related electrical equipment, each room is serviced by a single air handling unit. In these cases the safety-related HVAC systems, such as those servicing the Residual Heat Removal/Reactor Building Spray Pump Rooms, the Charging/SI Pump Rooms, the ESF Switchgear and Speed Switch Rooms, and the Auxiliary Building Motor Control Center Rooms, are designed such that no single failure can cause loss of cooling to more than one room, and subsequently to more than one train of redundant safety-related electrical equipment.

Quality Related (QR) HVAC equipment provides cooling or ventilation for some areas of the plant, outside the Reactor Building, which contain Class 1E electrical equipment. Class 1E equipment in these areas was evaluated on a case by case basis and is designed to function in the unlikely event of abnormal environmental conditions caused by loss of non-safety related HVAC. Loss of QR HVAC does not have an immediate effect on, or correlation to, the performance of the Class 1E equipment safety function. However, since credit is taken for normal conditions of HVAC operation in determining qualified life of Class 1E equipment located in harsh environmental areas and in determining service life of Class 1E equipment located in mild environmental areas, loss of QR HVAC systems could have a long term effect on equipment life expectancy. Therefore, QR HVAC is procedurally controlled and operability of QR HVAC systems is monitored to ensure that air flow to these areas, and subsequently the environmental conditions upon which the Class 1E equipment qualification was based, is maintained.

Controls and electrical equipment necessary for operation of safety-related HVAC systems outside the Reactor Building, following a LOCA or high energy line break condition, are located such that they are not exposed to post-accident environmental conditions, or are designed to withstand these severe conditions. Controls and electrical equipment required by safety-related HVAC systems within the Reactor Building are capable of withstanding the worst case environmental conditions resulting from a DBA.

Environmental test reports describing qualification of ventilation and cooling equipment located inside and outside the Reactor Building are referenced in the applicable equipment qualification data packages, and equipment qualification files.

The preceding discussions result in the determination that loss of ventilation, although highly unlikely, will not prevent performance of the Class 1E equipment safety function or affect the environmental qualification status of the safety-related electrical equipment.

3.11.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

This section presents the justification for the estimated chemical and radiation environments of Section 3.11.1.

3.11.5.1 Chemical Environment

3.11.5.1.1 Normal Operation

Adverse chemical environmental conditions do not exist during normal plant operation.

3.11.5.1.2 Design Basis Accident

The chemical spray environment for which electrical equipment inside containment must be qualified is based upon a maximum operating time for the spray system of 24 hours based on IEEE 323 testing standards. During a design basis LOCA, the containment spray system will be operated for a minimum period of four hours and up to a maximum of 40 days as required to return containment pressure and temperature conditions to normal levels. Therefore, a period of up to 40 days has been used as a basis for judging the adequacy of electrical equipment qualification. RN 03-008 12-034

The chemical spray environment for which electrical equipment inside containment must be qualified is based upon the following post-accident operating envelopes and spray pH conditions:

Operating Period	Spray pH Range
0-2 hours (minimum)	8.7-10.5
2 hours – 40 days (maximum)	8.0-8.5

RN 03-008

These chemical spray environmental (pH) conditions are based upon analyzing the drawdown of the Refueling Water Storage Tank (RWST) and Sodium Hydroxide Storage Tank (SHST) to develop a buffered borated water solution in the spray header. The analysis is performed for the range of boron sources (RWST, RGS, SIS), and sodium hydroxide concentrations (20-22 wt/%) required by the Technical Specifications in conjunction with the design, normal, and degraded operating modes discussed in Section 6.2.2.2.1.2.

Depending upon the spray system operating mode, the results of the analysis yield a spray pH range of 8.8 to 10.1 during the drawdown of the RWST (~ 23 to 65 minutes post LOCA). At the completion of the RWST drawdown, the spray system operation is maintained by recirculation of the Reactor Building Sump Water (pH: 7.5 - 8.5) and the injection of any remaining NaOH from the SHST. Upon recirculation from the sump the spray system pH is maintained in the range of 8.7 to 10.5 until the SHST is emptied (approximately 10 to 40 minutes). Thereafter, the spray pH is equal to the sump water pH of 8.1 to 8.5. Refer to Section 6.2.2.3.1.4 for a detailed discussion of drawdown analysis. RN 03-008

3.11.5.2 Radiation Environment

3.11.5.2.1 Normal Operation

The design basis radiation sources and dose rates for various plant systems and equipment during normal plant operation are discussed in Chapters 11 and 12. The neutron and gamma radiation source terms and energy spectra data for major equipment is summarized in Tables 12.1-3 through 12.1-17. Based upon these source terms and plant shielding, the plant radiation exposure zones for normal operation, shutdown, and refueling are presented in Figures 12.1-1 through 12.1-20. The total integrated (over 40 years) radiation doses resulting for normal plant operation are given in EQDB drawing S-021-018.

3.11.5.2.2 Design Basis Accident

The design basis post accident radiation sources and doses for vital plant systems and equipment are discussed in Chapter 12A. The radiation sources are based upon NUREG 0737, Section II.B.2. The post-accident total integrated (over 1 year) gamma and beta doses addressing the requirements of NUREG-0588 are given in EQDB drawing S-021-018.

3.11.6 REFERENCES

1. Institute of Electrical and Electronics Engineers (IEEE), "Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations", Standard 308, dated 1971.

2. Virgil C. Summer Nuclear Station Equipment Qualification Database.

3. Virgil C. Summer Nuclear Station Environmental Zone Maps SS-021-001 through SS-021-017.

4. Virgil C. Summer Nuclear Station Equipment Qualification Database Environmental Zone Information drawing S-021-018.

5. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment".

6. Institute of Electrical and Electronics Engineers (IEEE), "General Guide for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations", Standard 323, dated 1971.

7. Institute of Electrical and Electronics Engineers (IEEE), "Standard for Qualifying Class 1E Equipment Nuclear Power Generating Stations", Standard 323, dated 1974.

8. 10 CFR Part 50, Appendix A - General Design Criterion 4, "Environmental and Missile Design Bases".

9. 10 CFR Part 50, Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

10. U.S. Nuclear Regulatory Commission Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants".

11. 10 CFR Part 50, Section 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants".

12. U.S. Nuclear Regulatory Commission Regulatory Guide 1.30, "Q/A Requirements for the Installation, Inspection and Testing of Instrument and Electric Equipment".

13. U.S. Nuclear Regulatory Commission Regulatory Guide 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants".

14. U.S. Nuclear Regulatory Commission Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light Water-Cooled Nuclear Power Plants".

15. U.S. Nuclear Regulatory Commission Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants".

16. Letter NS-CE-692, dated July 10, 1975, from C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC).

17. Institute of Electrical and Electronics Engineers (IEEE), "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations", Standard 317, dated 1972.

18. Institute of Electrical and Electronics Engineers (IEEE), "Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations", Standard 334, dated 1971.

19. Institute of Electrical and Electronics Engineers (IEEE), "Standard for Qualification of Safety-Related Valve Actuators", Standard 382, dated 1972.

20. Institute of Electrical and Electronics Engineers (IEEE), "Standard for Type Tests of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations", Standard 383, dated 1974.

21. "Main Steam Line Break Effect on Environmental Qualification of Equipment", Nuclear Regulatory Commission Information Notice No. 84-90, December 7, 1984.

22. "Evaluation to Address Environmental Qualification of Qualified Equipment Subjected to a High Energy (Steam) Line Break of Superheated Blowdown Outside Containment and a Main Steam Line Break (MSLB) Inside Containment", Gilbert Associates, Inc. Report No. 2616.

23. "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment", Westinghouse Topical Report, WCAP-10961, October 1985.

24. U.S. Nuclear Regulatory Commission Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident".

TABLE 3.11-0

NUCLEAR STEAM SUPPLY SYSTEM CLASS 1E EQUIPMENT
IN CONTAINMENT

TABLE 3.11-0A

BALANCE OF PLANT
CLASS 1E EQUIPMENT

This information is contained in the computerized Equipment Qualification Database loaded on the VAX System

TABLE 3.11-1

<u>EQUIPMENT AND COMPONENTS, OTHER THAN CLASS 1E, INSIDE THE REACTOR BUILDING REQUIRED TO FUNCTION DURING AND/OR AFTER AN ACCIDENT</u>

1. Containment Spray System

   Piping
   Spray Header and Nozzles

2. Reactor Building Cooling Units

3. Containment Isolation System

   Isolation Valves
   Mechanical Penetrations
   Air Locks and Hatches

4. Emergency Core Cooling System

   Accumulators
   Valves and Piping

NOTES:

1. Unless otherwise indicated, equipment is located outside the secondary shield wall.

2. Refer to drawing S-200-971, “Essential Equipment List”, which includes a listing of equipment and components, other than Class 1E, required for safe shutdown and/or design basis accident, condition IV, mitigation.

TABLE 3.11-2

EQUIPMENT AND COMPONENTS, OTHER THAN CLASS 1E, OUTSIDE THE REACTOR BUILDING REQUIRED TO FUNCTION DURING AND/OR AFTER AN ACCIDENT

1. Service Water System

   Pumps
   Valves and Piping
   Heat Exchangers

2. Component Cooling Water System

   Pumps
   Valves
   Heat Exchangers
   Surge Tank

3. Containment Isolation System

   Isolation Valves
   Mechanical Penetrations
   Air Locks and Hatches

4. Main Steam System

   Valves and Piping to Turbine Driven Emergency Feedwater Pump

5. HVAC Chilled Water System

   Pumps
   Valves and Piping
   Chillers

6. Building Ventilation and Cooling Systems for the Control Room, Relay Room; RHR/RB Spray Pump Rooms; Charging/SI Pump Rooms, Auxiliary Building Motor Control Center and Switchgear Areas; ESF Switchgear and Speed Switch Rooms, Battery Rooms, Service Water Booster Pump Areas, Emergency Feedwater Pump Rooms; Diesel Generator Building; and Service Water Pumphouse.

   Air Handling Units
   Filters
   Fans
   Ducts, Dampers, Valves and Piping

7. Emergency Core Cooling System

   Refueling Water Storage Tank
   Charging Pumps
   RHR Pumps
   RHR Heat Exchanger
   Valves and Piping (see Section 6.3)

TABLE 3.11-2 (Continued)

8. Reactor Building Spray System

   Reactor Building Spray Pump
   Valves and Piping
   Refueling Water Storage Tank
   Sodium Hydroxide Storage Tank

9. Spent Fuel Cooling System

   Pumps
   Heat Exchangers
   Valves and Piping

10. Emergency Feedwater System

   Condensate Storage Tank
   Motor Driven Emergency Feedwater Pumps
   Turbine Driven Emergency Feedwater Pump
   Valves and Piping

NOTE: Refer to drawing S-200-971, “Essential Equipment List”, which includes a listing of equipment and components, other than Class 1E, required for safe shutdown and/or design basis accident, condition IV, mitigation.

TABLE 3.11-3

POSTULATED ENVIRONMENTAL CONDITIONS

Table 3.11-3 environmental data have been combined in a series of controlled drawings.

An equipment qualification database with environmental zone information, as documented by EQDB drawing S-021-018, provides a list of the environmental zones and conditions, including the normal, abnormal, and accident (including post-accident) environmental conditions for each environmental zone.

3.12 CONTROL OF HEAVY LOADS

3.12.1 Introduction / Licensing Background

In 1978, Nuclear Regulatory Commission (NRC) staff initiated Generic Technical Activity Task A-36 to systematically examine licensing criteria and the adequacy of measures in effect at operating nuclear power plants to ensure the safe handling of heavy loads and to recommend necessary changes to those measures.

Following licensee input, the results of this evaluation were reported in NUREG-0612. The NRC staff concluded that existing measures to control the handling of heavy loads, although providing protection from certain potential problems did not adequately cover the major causes of load-handling accidents and that these measures should be revised.

In NUREG-0612, the NRC provided a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in Article 5.1.1, ensures that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed.

RN 02-024

The approach used to develop these guidelines for minimizing the potential for a load drop was based on defense in depth and is summarized as follows:

- Provide sufficient operator training, handling system design, load-handling instructions, and equipment inspection to assure reliable operation of load handling systems.

- Define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over, or near, irradiated fuel or safe shutdown equipment.

- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

The second portion of the objective, achieved through guidelines identified in NUREG-0612 Articles 5.1.2 through 5.1.5, would ensure that, for load-handling systems in areas where their failure might result in significant consequences, either (1) features are provided to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane); or (2) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small.

On 12/22/1980, the NRC issued a Generic Letter (GL 80-113) from D. G. Eisenhut requesting that the station review provisions for handling and control of heavy loads, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance with these guidelines. Generic Letter 81-07 provided supplemental information regarding the GL 80-113 response.

VCSNS responded in submittals which included GAI Report 2364, Control of Heavy Loads at Nuclear Power Plants – Virgil C. Summer Nuclear Station, Unit 1 (currently VCSNS Technical Report TR03920-002), and WCAP-10233, Evaluation of the Acceptability of the Reactor Vessel Head Lift Rig, Reactor Vessel Internals Lift Rig, Load Cell, and Load Cell Linkage to the Requirements of NUREG-0612.

In 1983, NRC consultant Idaho National Engineering Laboratory (EG&G Idaho, Inc.) performed a review of the station compliance with NUREG-0612, resulting in the issuance of two Technical Evaluation Reports (TER). The Phase I TER assessed conformance with NUREG-0612 Article 5.1.1. The Phase II TER reviewed compliance with Articles 5.1.2 through 5.1.5. Subsequent revisions to the GAI Report addressed the concerns identified in this evaluation.

On 5/23/1985, the NRC issued a Safety Evaluation Report, concluding that VCSNS had satisfied the Phase I guidelines of NUREG-0612. This correspondence and Revision 3 of GAI Report 2364 document the commitments made to the Nuclear Regulatory Commission. RN 02-024

The NRC issued Generic Letter 85-11, in June 1985, which stated in part ". . . . Based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action is not required . . . . .and Phase II of NUREG-0612 is considered completed."

In April 1996, the NRC issued IE Bulletin 96-02, to alert licensees to the importance of complying with existing regulatory guidelines associated with the control and handling of heavy loads. Addressees were asked to review plans and capabilities for handling heavy loads while the reactor is at power and to determine whether those activities are within the licensing basis. SCE&G responded (RC-96-0176) that all station activities were within the current licensing basis. The NRC subsequently closed this issue in 1998.

The NRC established generic safety issue, GI-186, in 1999 to investigate the need for additional regulation or guidance to address risk associated with increased frequency of moving spent fuel storage casks during power operation. NUREG-1774 documented a survey of operating experience prepared as part of the GI-186 investigation. In evaluating this data, the NRC staff determined that the frequency of load drops was indeed low and unlikely to justify additional regulations or guidance. Problems identified in the survey would be addressed by clarification and reemphasis of existing guidance on control of heavy loads.

As a result of the GI-186 investigations, the NRC issued Regulatory Issue Summary (RIS) 2005-25 in October 2005. This document reemphasized the NUREG-0612 guidelines on control of heavy loads and provided additional recommendations based upon operating experience and inspection information. These suggestions included:

- Evaluate the capability of rigging components and materials to withstand rigging errors.
- Evaluate the need to establish standardized calculation methodologies for heavy load drops.
- Endorsement of ASME NOG-1-2004.

In May 2007, Supplement 1 to RIS 2005-25 was issued to discuss remaining recommendations associated with GI-186 and to communicate regulatory expectations related to the safe handling of heavy loads. Topics presented include:

- Selection of lifting devices for use with single-failure-proof cranes.
- Application of ASME NOG-1 design criteria in satisfying NUREG-0554 guidelines.
- Industry consensus documents providing guidance for heavy load drop evaluations. RN 02-024
- FSAR inclusion of station reliance on single-failure-proof crane and/or load drop analysis.
- Application of 10CFR50.59 to procedure changes governing the handling and control of heavy loads (such as changes to motion restrictions, maximum height or weight, and medium present under the load).

NEI 08-05 was developed in response to RIS 2005-25 and RIS 2005-25 Supplement 1, discussing interpretation and implementation of the provided regulatory guidance. This effort was undertaken to ensure that heavy load lifts continue to be conducted safely and that actual station practices are accurately reflected in licensing bases. Topics include configuration risk management for heavy loads, Reactor Pressure Vessel Head (RPVH) load drop analysis, single-failure-proof crane equivalence for reactor pressure vessel head lifts, and updates to the FSAR regarding control of heavy loads.

3.12.2 SAFETY BASES

The safety bases employed to prevent or mitigate the consequences of an accidental heavy load drop are discussed within Technical Report TR03920-002 and includes (1) compliance with the Phase I guidelines of NUREG-0612, (2) reliance on a load drop analysis to demonstrate that consequences from a postulated RPVH load drop are within acceptable limits (i.e., core remains covered and cooling is available), and (3) use of a single-failure-proof fuel handling building crane. RN 11-041

3.12.3 SCOPE OF HEAVY LOAD-HANDLING SYSTEMS

NUREG-0612 describes a "heavy load" as any load that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool. At VCSNS, a heavy load is any load greater than 2,500 pounds.

The station's load-handling systems are identified within Technical Report TR03920-002. Initially, all permanently installed overhead handling systems were initially reviewed. Systems were then excluded from further consideration based upon capacity and physical separation from either spent fuel or equipment required for safe shutdown / decay heat removal.

The remaining overhead handling systems manipulate "NUREG-0612 heavy loads"; i.e., loads where a postulated drop could impact spent nuclear fuel, spent fuel in the reactor vessel, or equipment required for safe shutdown / decay heat removal. Specific load drops were then analyzed to evaluate consequences of a load drop accident. In some instances, facility modifications were performed to minimize or eliminate risk.

The load-handling systems used at VCSNS to lift NUREG-0612 heavy loads, as well as the description of the safe load paths, and postulated load drop analysis (where applicable), are included in Technical Report TR03920-002. RN 02-024

3.12.4 CONTROL OF HEAVY LOADS PROGRAM

The stations' program governing the control of NUREG-0612 heavy loads is delineated within Technical Report TR03920-002. Originally identified as GAI Report 2364, this document was submitted to the Nuclear Regulatory Commission as part of the station's NUREG-0612 response. On 5/23/1985, the NRC issued a Safety Evaluation Report concluding that VCSNS satisfied the Phase I guidelines of NUREG-0612.

The VCSNS program for the control of heavy loads consists of the following:

1. License commitments made in response to NUREG-0612, Phase I.

2. Safety bases for the handling of the Reactor Pressure Vessel Head (RPVH). Reliance on a load drop analysis, which includes the basis lift height, load weight, medium present under load, necessary for concluding satisfaction to acceptance criteria. RN 11-041

3. Safety bases for the handling of spent fuel casks within the Fuel Handling Building. RN 16-003

3.12.4.1 Commitments in Response to NUREG-0612, Phase I

RN 02-024

VCSNS license commitments made in response to NUREG-0612, Phase I are described within Technical Report TR03920-002, and are summarized below:

1. Safe load paths

Safe load paths have been identified for each NUREG-0612 heavy load-handling crane or hoist to minimize the possibility of a heavy load drop onto spent nuclear fuel, spent fuel in the reactor vessel, or equipment required for safe shutdown or decay heat removal. In those cases where a safe load path could not be defined, station procedures have been generated and / or design modifications made to minimize the consequences of an inadvertent load drop. Safe load paths are permanently marked, where practical.

2. Load-handling procedures

Special operating procedures have been prepared for NUREG-0612 heavy load handling devices, and where possible, incorporated into standard component maintenance procedures to define the handling of heavy loads.

3. Qualifications, training, and specified conduct of crane operators

Crane operators and riggers undertake an extensive training program, which meets or exceeds the requirements of ANSI B30.2-1976, Chapter 2-3. This training program incorporates the safe load path concept and station procedures covering the handling of NUREG-0612 heavy loads. RN 02-024

4. Special lifting devices

Special lifting devices used for the handling of NUREG-0612 heavy loads associated with servicing reactor vessel components (i.e., RPVH lift rig, RV internals lift rig, load cell, and load cell linkage) do not strictly comply with ANSI N14.6-1978. The equipment vendor has performed an evaluation (WCAP-10233) of these discrepancies and has recommended alternate methods for demonstrating equivalency with the ANSI requirements. These recommendations include a detailed inspection and testing program, which has been incorporated into station procedures. RN 16-003

Note that the original RPVH Lift Rig has been subsequently replaced with a special lifting device integrated into the Reactor Head Service Structure (A.K.A. Integrated Head Assembly (IHA)). This device complies with ANSI 14.6-1978 design requirements as documented in the IHA Design Report (Ref. 21). RN 16-003

5. Lifting devices that are not specially designed

Non-specifically designed lifting devices (i.e., slings) used for the handling of NUREG-0612 heavy loads do not strictly comply with NUREG-0612, Guideline 5; which states that these lifting devices should (1) be installed and used in accordance with the guidelines of ANSI B30.9-1971, and (2) be selected based upon the sum of static and dynamic loads (excluding SSE loads).

In the Phase I response to NUREG-0612, VCSNS stated that non-specifically designed lifting devices (i.e., slings) would be installed and used consistent with the guidelines of 29CFR1910.184. Additionally, the station evaluated the loading for each NUREG-0612 heavy load-handling crane or hoist, and determined dynamic loads to be a relatively small fraction (15% or less) of the static loads. VCSNS concluded that modifying the selection criteria for these lifting devices to accommodate such minor additional loads, would not have a substantial effect on overall load-handling reliability. Therefore, lifting device selection would be based upon static load.

The NRC found both these approaches to be consistent with the intent of the NUREG-0612, and acceptable.

6. Periodic inspection, and maintenance

   Cranes and rigging equipment used for the handling of NUREG-0612 heavy loads are maintained, tested, and inspected to the requirements of ANSI B30.2-1976, Chapter 2-2.

7. Crane Design

   Cranes used for the handling of NUREG-0612 heavy loads were procured and designed in accordance with CMAA Specification 70 and ANSI B30.2-1976. Chapter 2-1. RN 02-024

3.12.4.2 Reactor Pressure Vessel Head Lifting Procedures

To control the handling of the Reactor Pressure Vessel Head (RPVH), station procedures are used to direct the lift, movement, and replacement of the RPVH. These procedures establish limitations on load weight, lift height, safe load path, and medium present under the load. Limits are based on the current RPVH load drop analysis, which provides assurance that the core will remain covered and cooled in the event of a postulated reactor pressure vessel head drop.

The original load drop analysis for VC Summer, which was based upon a generic Westinghouse plant design, has been superseded with a plant specific evaluation, which also addresses concerns identified by the NRC in RIS 2005-25 and RIS 2005-25, Supplement 1. The current assessment considers the concentric drop of a 320,000 lbs. Reactor Pressure Vessel Head from a height of 35 feet above the Reactor Vessel Flange, through air (i.e. no water below). The evaluation utilizes Finite Element Analysis (FEA) methodology to determine postulated loads and displacements associated with the Head drop. Structural integrity of the Reactor Vessel, Vessel Support Assemblies, Main Loop RCS Piping and Concrete supporting structure are then assessed following application of these loads/displacements. Guidance and acceptance criteria for this type of evaluation are provided in NEI 08-05. RN 16-003

This plant specific analysis demonstrates that after the postulated Reactor Vessel Head Drop Accident:

1) The Reactor Core will remain covered with coolant, and

2) Coolant retaining components (such as the Reactor Vessel, Vessel Nozzles, Main Loop RCS piping and elbows) will remain sufficiently intact to provide cooling capability.

Consistent with the described RPVH Drop Analysis, the following limitations have been incorporated into appropriate station procedures. RN 16-003

- Maximum Lift Weight of 315,000 lbs. or 157.5 tons (conservative).
- Maximum Lift Height of 35 feet above the reactor vessel flange.

3.12.4.3 Single Failure Proof Crane for Spent Fuel Casks

The safe handling of spent fuel casks is addressed within Technical Report TR03920-002. The fuel handling building crane is a single-failure-proof crane having a single-failure-proof 125 ton main hoist. The fuel handling building crane meets the modified crane single-failure-proof criteria and guidelines of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants", Appendix C of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", and meets the NRC-accepted single-failure-proof crane requirements of Ederer Generic Licensing Topical Report EDR-1 (P)-A, Reference [23]. Use of a single-failure-proof fuel handling building crane facilitates spent fuel cask handling without the need to postulate or analyze dropped cask accidents. RN 11-041

3.12.5 SAFETY EVALUATION

VCSNS Technical Report TR03920-002 establishes policy for the safe handling and control of NUREG-0612 heavy loads. Based on the stations' response to Phase I of NUREG-0612, this program establishes measures which assure an adequate level of defense-in-depth for heavy loads handled in the vicinity of spent nuclear fuel, spent fuel in the reactor vessel, or equipment required for safe shutdown / decay heat removal.

Measures taken to minimize the potential for occurrence of a NUREG-0612 heavy load-handling accident include:

- Controls implemented by station commitments to NUREG-0612, Phase I guidelines make the risk of a load drop very unlikely. RN 02-024
- In the event of a postulated load drop, the consequences are acceptable, as demonstrated by load drop analysis. Station procedures reflect restrictions on load weight, lift height, and medium present under the load.
- Where applicable, risk associated with the movement of NUREG-0612 heavy loads conducted during power operation or shutdown conditions is a configuration management activity with administrative controls established in accordance with 10 CFR 50.65(a)(4). This applies to RPVH lifts, spent fuel cask lifts, and other NUREG-0612 heavy load lifts defined in the station procedures.

- Changes to the program controlling heavy loads, such as station maintenance activities which handle NUREG-0612 loads outside the bounds of previous analysis, modifications which add or impact overhead handling systems, or implementation of dry cask storage of spent nuclear fuel, will be evaluated in accordance with 10CFR50.59 to determine the existence of an unreviewed safety question and the need for license amendment.

3.12.6 REFERENCE DOCUMENTS

1. 10CFR50.59; Changes, Tests, and Experiments.

2. 29CFR1910.184; Slings

3. ANSI B30.2-1976; Overhead and Gantry Cranes

4. ANSI B30.9-1971; Slings

5. ANSI N14.6-1978; Standard for Special Lifting Devices for Shipping Containers weighing 10,000 pounds or more for Nuclear Material

6. ASME NOG-1-2004; Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)

7. CMAA Specification 70-1975; Specifications for Electric Overhead Traveling Cranes RN 02-024

8. EGM 2007-006; Enforcement Discretion for Heavy Load Handling Activities, 9/28/2007

9. Generic Letter 80-113; Control of Heavy Loads

10. Generic Letter 81-07; Control of Heavy Loads

11. Generic Letter 85-11; Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612

12. GI-186; Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants

13. IE Bulletin 96-02; Movement of Heavy Loads over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment

14. NEI 08-05; Industry Initiative on Control of Heavy Loads, 7/2008

15. NUREG-0554; Single-Failure-Proof Cranes for Nuclear Power Plants

16. NUREG-0612; Control of Heavy Loads

17. NUREG-1774; A Survey of Crane Operating Experience at U. S. Nuclear Power Plants from 1968 through 2002 RN 02-024

18. RIS 2005-25; Clarification of NRC Guidelines for the Control of Heavy Loads

19. RIS 2005-25 Supplement 1; Clarification of NRC Guidelines for the Control of Heavy Loads

20. TR03920-002, Control of Heavy Loads at Virgil C. Summer Nuclear Station, Unit 1 (former GAI Report No. 2364) RN 11-041

21. DC05900-004, Integrated Head Assembly Design Report RN 16-003

22. WCAP-10233, Revision 0; Evaluation of the Acceptability of the Reactor Vessel Head Lift Rig, Reactor Vessel Internals Lift Rig, Load Cell, and Load Cell Linkage to the Requirements of NUREG-0612 RN 02-024

23. TR03920-007, EDR-1 (P)-A, Generic Licensing Topical Report for Ederer's Nuclear Safety Related eXtra – Safety And Monitoring (X-SAM) Cranes, 10/8/82. RN 11-041

24. DC03920-026, Reactor Head Load Drop Analysis RN 16-003

APPENDIX 3A

CONFORMANCE WITH REGULATORY GUIDES

This appendix discusses conformance with Division 1 NRC Regulatory Guides applicable to the Virgil C. Summer Nuclear Station. Specific revision numbers and dates of issue are identified in the title of each guide.

1.1 NET POSITIVE SUCTION HEAD FOR ECCS AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS (REVISION 0; 11/70)

Regulatory Guide 1.1 recommends that the emergency core cooling and containment heat removal systems 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 Sections 6.3.2 and 6.2.2, 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. In addition to considering the static head and suction line pressure drop, the calculation of available NPSH in the recirculation mode does not take credit for subcooling, i.e., the vapor pressure of the liquid in the sump is assumed to be equal to the saturation pressure corresponding to the temperature of the sump fluid.

1.2 THERMAL SHOCK TO REACTOR PRESSURE VESSELS (REVISION 0; 12/70)

The NRC has withdrawn this Regulatory Guide which is superseded by Section 50.61 of 10 CFR 50 and Regulatory Guide 1.154.

1.3 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR BOILING WATER REACTORS (REVISION 2; 6/74)

Regulatory Guide 1.3 is not applicable to the Virgil C. Summer Nuclear Station since it uses a PWR.

1.4 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS (REVISION 2; 6/74)

The guidance of Regulatory Guide 1.4 including the use of the TID-14844 accident source term remains the licensing basis for the following:

1. Equipment qualification

2. NUREG-0737 evaluations other than Control Room Habitability Envelope (CRHE) and Technical Support Center (TSC) doses RN 12-034

3. Final Safety Analysis Report (FSAR) accidents not analyzed in accordance with Regulatory Guide 1.183 including the loss of offsite power and the waste gas tank rupture

The Virgil C. Summer Nuclear Station licensing basis incorporates a full implementation application of the Alternative Source Term (AST) methodology compliant with Regulatory Guide 1.183, which is used for the design basis case loss-of-coolant accident (LOCA), main steam line break (MSLB) accident, fuel handling accident (FHA), steam generator tube rupture (SGTR), reactor coolant pump (RCP) locked rotor accident (LRA) and the control rod ejection accident (CREA). RN 12-034

1.5 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS (REVISION 0; 3/71)

Regulatory Guide 1.5 is not applicable to the Virgil C. Summer Nuclear Station since it uses a PWR.

1.6 INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS (REVISION 0; 3/71)

The Virgil C. Summer Nuclear Station design complies with the recommendations of Regulatory Guide 1.6.

See Sections 8.1, 8.3.1.2, and 8.3.2.2 for details.

1.7 CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS OF COOLANT ACCIDENT (REVISION 0; 3/71; SUPPLEMENT 10/71)

The design guidance and assumptions of Regulatory Guide 1.7 are used for control of combustible gas concentrations in containment following a loss of coolant accident, as described in Sections 6.2.5 and 15.4.

1.8 PERSONNEL SELECTION AND TRAINING (REVISION 2)

The personnel selection and training program for the Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.8, Revision 2, Section D, by using accredited training programs as an alternative to Sections A and C.

1.9 SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES (REVISION 0; 3/71)

The Virgil C. Summer Nuclear Station standby power system is discussed in Sections 8.3.1.1 and 8.3.1.2 and complies with the recommendations of Regulatory Guide 1.9.

1.10 MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY 1 CONCRETE STRUCTURES (REVISION 1; 1/73)

Regulatory Guide 1.10 has been withdrawn per NRC letter of July 8, 1981. However, the Virgil C. Summer Nuclear Station still complies with the recommendations of Regulatory Guide 1.10 with the following clarifications:

1. Regulatory Position C.1 [paragraph 2, subitem (2)] is interpreted as follows - Each member or crew is subject to requalification if more than 7% of the completed splices fail to pass the visual inspection test or fail to pass the tensile tests.

2. Regulatory Position C.4 - Separate test cycles are established for mechanical splices in horizontal, vertical, and diagonal bars.

3. Regulatory Position C.4.b - Concerning test frequency for combinations of production and sister splices. One splice, either production or sister splice out of every 33 is sampled in lieu of 3 out of every 100.

4. Regulatory Position C.5.a - If any production splice fails to meet the tensile requirement of Regulatory Position C.3.a (tensile strength greater than 125% of yield strength of the reinforcement) and failure did not occur in the bar, the adjacent production splices on each side of the failed splice shall be tested. If any sister splice used for testing fails to meet the tensile requirement of Regulatory Position C.3.a and failure did not occur in the bar, two additional sister splices shall be tested. If either of these retests fails to meet the requirements, splicing by the crew performing the work represented by the failed splices shall be halted. Splicing shall not be resumed until the cause and extent of the failures have been determined, corrected, and resolved.

5. Regulatory Position C.5.b - If the average tensile strength of 15 consecutive samples fails to meet the requirement of Regulatory Position C.3.b, the average equaling or exceeding the ultimate tensile strength of the substandard splices shall be investigated, and the necessary corrective action taken.

   Mechanical (cadweld) splice testing and acceptance criteria are in accordance with the requirements of the ASME Code, Section III Division 2. Refer to Section 3.8.1.6.

1.11 INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT (REVISION 0; 3/71)

Instrument sensing lines that penetrate containment are provided with isolation valves in accordance with Regulatory Guide 1.11 except for four lines necessary for sensing Reactor Building wide range pressure. Redundant containment boundaries are provided on these Reactor Building pressure sensing lines by use of filled bellows as described in Section 6.2.4.

1.12 INSTRUMENT FOR EARTHQUAKES (REVISION 1; 4/74)

Seismic monitoring instrumentation provided in the Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.12 with the following clarifications and exceptions:

1. The frequency range of the response spectrum recorders is 2 to 25.4 Hertz;

2. Triaxial peak accelerographs have been removed from the plant and not replaced due to their inability to monitor seismic motions over background ambient vibration noise.

3. The recorder specified in Section 5.3 of ANSI N18.5-1974 has been upgraded with a solid-state recording/analysis system.

4. The seismic trigger specified in Section 5.4 of ANSI N18.5-1974 has been removed from service and replaced with solid-state actuation using the Reactor Building foundation mat accelerometer as the trigger sensor.

The maximum SSE foundation acceleration for the Virgil C. Summer Nuclear Station is 0.25g; therefore, seismic monitoring instrumentation is provided in accordance with position C.1 of the guide.

A description of the seismic instrumentation is presented in Section 3.7.4.

1.13 SPENT FUEL STORAGE FACILITY DESIGN-BASIS (REVISION 1;12/75)

Comparison of the Virgil C. Summer Nuclear Station with Regulatory Guide 1.13 Recommendations.

Regulatory Position	Virgil C. Summer Nuclear Station Design Features
1.	The safety class 2b spent fuel pool and the Seismic Category I Fuel Handling Building are designed to withstand the SSE without loss of function. 99-01
2.	The reinforced concrete walls and the spent fuel pool liner of the Fuel Handling Building were designed to prevent loss of watertight integrity due to tornadic winds or missiles generated by these winds, as specified in Table 3.5-5 (Requirement C2.(a) of Regulatory Guide 1.13.) These missiles are, however, capable of penetrating the exterior metal walls and roof panels of the Fuel Handling Building. The steel frame superstructure has been designed so that the loss of any one structural member will not adversely affect the building superstructure integrity.

Regulatory
Position Virgil C. Summer Nuclear Station Design Features

This redundancy in structural support was achieved by the use of moment connections. Studies have shown an extremely low probability that spent fuel will be contacted by the postulated tornado missiles.

Consider the wood plank and wood pole as potential missiles. Based upon the annual probability of a tornado strike, the difference in density between wood and water, and the probability of attaining critical missile flight trajectory and orientation, the total probability of either of these two missiles entering the spent fuel pool and subsequently coming into contact with the fuel storage racks is substantially less than 10^{-7}.

The height of the missile resistant concrete wall (approximately 28 feet above ground), as stated previously, prevents the utility pole, compact and passenger automobiles from being thrown into the spent fuel pool by a tornadic wind.

The only other missile listed in Table 3.5-5, that was considered to be a potential missile capable of causing fuel damage, is the 3 inch schedule 40 pipe. A conservative probability estimate indicates that, in the event of the 3 inch pipe entering the spent fuel pool and contacting the fuel racks, the resulting offsite doses can be maintained within the 10 CFR 100 limits. It also indicates that the requirement of position C.2(b) of Regulatory Guide 1.13 can be satisfied. The basis for such a conclusion can be described by the following procedure:

a. Determine the probability, P_1, of the 3 inch pipe, as a potential missile, entering the spent fuel pool.

b. Calculate the allowable missile velocity, v, at impact, beyond which fuel assemblies and/or fuel racks could experience damage and consequently cause significant release of radioactivity.

It is noted that the fuel assemblies are completely submerged in the stainless steel fuel racks of square cross section. The three inch pipe missile with the above impact velocity was found not capable of penetrating through the fuel rack and coming into contact with the enclosed fuel assemblies. Hence, the requirement of position C.2(b) of Regulatory Guide 1.13 is satisfied.

c. Determine the number of fuel assemblies which are allowed to be damaged by the missile for the resulting offsite doses to be within 10 CFR 100 limits. For conservatism, a total of four fuel assemblies are considered permissible.

Regulatory
Position Virgil C. Summer Nuclear Station Design Features

d. Determine the required missile flight trajectory (in both horizontal and vertical planes) relative to the spent fuel pool and the missile alignment with respect to its flight path, in entering the spent fuel pool to make potential contact with more than four fuel cells and to result in a reduced impact velocity higher than the allowable velocity, v, previously established. Then determine the probability, P_2, of such events occurring.

e. Combine the probabilities, P_1 and P_2, to obtain the total probability, P, of the postulated 3 inch pipe missile being thrown into the spent fuel pool and subsequently damaging more than four fuel assemblies due to the impact effects transmitted through the fuel racks. This probability, P, was found to be less than 10^{-7}, the recognized acceptable probability limit. RN 18-009

3. See response to Item 5

4. The building enclosing the spent fuel pool is a low leakage building. The Fuel Handling Building Ventilation System exhausts to the plant vent through charcoal and HEPA filters. However, the Fuel Handling Building charcoal exhaust system is not required during the movement of fuel in the spent fuel pool or during crane operation with loads over the pool. The fuel handling accident is analyzed in Chapter 15. Due to the use of Alternate Source Term methods, no credit for ventilation system operation is needed for dose calculations. 98-01 RN 18-009

5. In addition to electrical interlocks, the Fuel Handling Building crane is prevented from moving over the spent fuel pool by stops welded to the rails. Only the spent fuel bridge crane is used to lift items from the fuel pool. The pool is designed to withstand, without leakage which could uncover the fuel, the impact of the heaviest load carried by this crane.

6. The spent fuel pool cooling and purification lines are located such that they cannot drain the pool. The only drain line that conceivably drains the pool is the refueling cavity drain which is a small line with two isolation valves. The refueling cavity is isolated from the spent fuel pool by the fuel transfer tube valve and the fuel transfer canal gate, which are Seismic Category I equipment.

Regulatory
Position   Virgil C. Summer Nuclear Station Design Features

7. Spent fuel pool level and radiation levels are monitored and activate alarms in the control room if abnormal conditions are detected. Radiation Monitors are described in Sections 12.1.4 and 12.2.4. The radiation monitors do not actuate the filtration system, because the filtration system is not credited in the Chapter 15 accident analysis. RN 18-009

8. Normal makeup water is from the station demineralized water supply which is not designed to withstand the SSE. However, two safety class backup supplies are provided.

   These are the refueling water storage tank and the reactor makeup water storage tank which are Safety Class 2a and 2b respectively.

1.14 REACTOR COOLANT PUMP FLYWHEEL INTEGRITY (REVISION 1; 8/75)

The Virgil C. Summer Nuclear Station addresses the recommendations of Regulatory Guide 1.14 as described in Section 5.2.6 and below:

1. Post-Spin Inspection

   Westinghouse has shown in WCAP-8163, "Topical Report on Reactor Coolant Pump Integrity in LOCA," that the flywheel would not fail at 290% of normal speed for a flywheel flaw of 1.15 inches or less in length. Results for a double ended guillotine break at the pump discharge with full separation of pipe ends assumed, show the maximum overspeed to be less than 110% of normal speed. The maximum overspeed was calculated in WCAP-8163 to be about 280% of normal speed for the same postulated break, and an assumed instantaneous loss of power to the reactor coolant pump. In comparison with the overspeed presented above, the flywheel is tested at 125% of normal speed. Thus, the flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125% provided that no flaws greater than 1.15 inches are present. If the maximum speed were 125% of normal speed or less, the critical flaw size for failure would exceed 6 inches in length. Non-Destructive tests and critical dimension examinations are all performed before the spin tests. The inspection methods employed (described in WCAP-8163) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 inches for 290% of normal speed would be detected. Flaws in the flywheel will be recorded in the pre-spin inspection program (see WCAP-8163). Flaw growth attributable to the spin test (i.e., from a single reversal of stress, up to speed and back), under the most adverse conditions is about three orders of magnitude smaller than what nondestructive inspection techniques are capable of detecting. For these reasons, Westinghouse performs no post-spin inspections and believes that pre-spin test inspections are adequate.

2. Interference Fit Stresses and Excessive Deformation

   Much of Revision 1 deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because Westinghouse's design specifies a light interference fit between the flywheel and the shaft; at zero speed, the hoop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and at normal speed, the flywheel is not in contact with the shaft in the sense intended by Revision 1. Hence, the definition of "Excessive Deformation," as defined in Revision 1 of Regulatory Guide 1.14, is not applicable to the Westinghouse design since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft does not cause unbalance of the flywheel. Extensive Westinghouse experience with reactor coolant pump flywheels installed in this fashion has verified the adequacy of the design. Westinghouse's position is that combined primary stress levels, as defined in Revision 0 of Safety Guide 14 [C.2. (a) and (c)] are both conservative and proven and that no changes to these stress levels are necessary. Westinghouse designs to these stress limits and thus does not have permanent distortion of the flywheel bore at normal or spin test conditions.

3. Section B, Discussion of Cross Rolling Ratio of 1 to 3

   Cross Rolling Ratio - Westinghouse's position is that specification of a cross rolling ratio is unnecessary since past evaluations have shown that ASME SA-533-B Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotropy in the flywheel material by means of cross rolling is unnecessary since adequate margins of safety are provided by both flywheel material selection (ASME SA-533-B Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material.

4. Section C, Item 1.a Relative to Vacuum-Melting and Degassing Process or the Electroslag Process

   Vacuum Treatment - The requirements for vacuum melting and degassing process or the electroslag process are not essential in meeting the balance of the Regulatory Position nor do they, in themselves, ensure compliance with the overall Regulatory Position. The initial Safety Guide 14 stated that the "flywheel material should be produced by a process that minimized flaws in the material and improves its fracture toughness properties." This is accomplished by using SA-533 material including vacuum treatment.

5. Section C, Item 2b; Westinghouse interprets this paragraph to mean:

   Design Speed Definition

   Design speed should be 125% of normal speed or the speed to which the pump motor might be electrically driven by station turbine generator during anticipated transients, whichever is greater. Normal speed is defined as the synchronous speed of the a-c drive motor at 60 Hz.

1.15 TESTING OF REINFORCING BARS FOR CONCRETE STRUCTURES (REVISION 1; 12/72)

Regulatory Guide 1.15 has been withdrawn per NRC letter of July 8, 1981. However, the testing of reinforcing bars for concrete structures is still in compliance with the recommendations of Regulatory Guide 1.15. The testing requirements are discussed in Section 3.8.1.6.

1.16 REPORTING OF OPERATING INFORMATION (REVISION 4; 8/75)

Regulatory Guide 1.16 has been withdrawn per NRC Federal Register Notice 74 FR 40244 dated 11 August 2009. However, operating information will be reported in accordance with the recommendations of this guide. A discussion of this subject is given in Section 6.9 of the Technical Specifications. RN 09-032

1.17 PROTECTION OF NUCLEAR POWER PLANTS AGAINST INDUSTRIAL SABOTAGE (REVISION 1; 6/73)

The Virgil C. Summer Nuclear Station's Physical Security Plan generally complies with the recommendations of Regulatory Guide 1.17 with the following exceptions as clarified below:

1. Alarms are automatically reset after obtaining operator acknowledgment. No alarm point status changes or actuation of locks are made without CAS/SAS knowledge.

2. Intrusion alarms, emergency exit alarms, alarm systems, and line supervisory systems meet the level of performance and reliability indicated by GSA Federal Specification WA00450B (GSA-FSS) Section 3. RN 96-041

3. X-ray systems are tested every 7 days for operability and quarterly for performance.

4. Tamper switches on alarm sensors, premise control units, junction boxes, and line supervisory units are checked every 3 months to ensure proper operation and that tampering has not occurred.

5. a) On-site communications are tested for performance at the beginning of each work shift.

   b) Off-site communications are tested for performance once per day.

1.18 STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS (REVISION 1; 12/72) 00-01

Regulatory Guide 1.18 has been withdrawn per NRC Letter of July 8, 1981. However, the Virgil C. Summer Nuclear Station still complies with the recommendations of this guide as described in Section 3.8.1.

1.19 NON-DESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS (REVISION 1; 8/72)

Regulatory Guide 1.19 has been withdrawn per NRC letter of July 8, 1981. However, the Virgil C. Summer Nuclear Station still complies with the recommendations of Regulatory Guide 1.19 as described in Section 3.8.1.

1.20 COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING (REVISION 2; 5/76)

In accordance with Regulatory Guide 1.20 recommendations, the results of tests and inspections during hot functional testing are to be compared with design data and previous test data for verification that detrimental reactor internals vibrations do not exist. A description of these tests, inspections, and analysis is found in Section 3.9.1.

1.21 MEASURING, EVALUATING, AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (REVISION 1; 6/74)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.21 with the following clarification. Turbine Building vents and sumps and Intermediate Building vents are not considered "principal effluent discharge paths" and are not continuously monitored. Assessment of radiation doses to the public from radioactive effluents will be implemented in accordance with the Offsite Dose Calculation Manual as required by Technical Specification, Section 6.9.1.8.

Meteorological conditions will be reported as required by the Offsite Dose Calculation Manual.

1.22 PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS (REVISION 0; 2/72)

Protection system actuation functions include the capability for periodic testing which meets the recommendations of Regulatory Guide 1.22. The periodic testing is described in Sections 7.1.2.5, 7.2.2, and 7.3.2.

1.23 ONSITE METEOROLOGICAL PROGRAMS (REVISION 0; 2/72)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.23 with the following clarification. Every reasonable effort will be made to assure 90% annual data recovery for those meteorological measurements used in the final documentation of station operational environmental impact. These measurements are to be made to obtain one full year of data following station commercial operation. Efforts for 90% annual data recovery shall be continued after this period only for wind speed 10M, wind direction 10M, and differential temperature (10-61M) as other measurements cannot reasonably be required for the assessment of diffusion characteristics. Readouts for these parameters (wind speed 10M, wind direction 10M, and differential temperature 10-61M) will be made available in the control room.

1.24 ASSUMPTION USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR GAS STORAGE TANK FAILURE (REVISION 0; 3/72)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.24 with the following clarifications. The waste gas decay tank contents as described in Section 15.3.5 are based on the planned operation of the Virgil C. Summer Nuclear Station Gaseous Waste Processing System rather than the method of operation assumed by the Regulatory Guide. Actual site related diffusion characteristics are utilized rather than those specified in the 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 (REVISION 0; 3/72)

Regulatory Guide 1.25 is not applicable to the Virgil C. Summer Nuclear Station since it uses Regulatory Guide 1.183 for evaluating fuel handling accidents. RN 12-034

1.26 QUALITY GROUP CLASSIFICATION AND STANDARDS FOR WATER, STEAM, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS (REVISION 3; 2/76)

Quality groups classification of fluid system equipment for the Virgil C. Summer Nuclear Station is described in Section 3.2.2.

Nuclear Steam Supply System fluid system components important to safety are classified in accordance with the August 1970 Draft of ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" except that components of the accumulator subsystem are classified in accordance with the 1973 version of N18.2, as finally accepted by ANSI, and components of the Liquid and Gaseous Waste Processing Systems and the Boron Recycle System are classified in accordance with Regulatory Guide 1.143.

Classification by this means is an alternative acceptable method of meeting the intent of Regulatory Guide 1.26.

1.27 ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS (REVISION 2; 1/76)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.27. The transient and thermal-hydraulic model analyses of the service water pond satisfy the changes noted in this guide.

The details of our analyses and design bases can be found in Sections 2.4.8 and 9.2.5 and V. C. Summer Technical Report, TR 02230-014, "Service Water Pond Thermal Study," dated June, 2000. 02-01

1.28 QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION) (REVISION 3; 8/85)/(REVISION 4-2010-REACTOR HEAD REPLACEMENT) RN 11-040 16-003

The Virgil C. Summer Nuclear Station complies with the recommendations of this guide as described in Section 17.1 and as discussed below.

The Virgil C. Summer Nuclear Station commits to Regulatory Position C.2 and Table 1 only, along with NQA-1-1994 Basic Requirement 17 and Supplementary Requirement 17S-1 for quality assurance record types and retention requirements. RN 11-040

V. C. Summer commits to Revision 4-2010 of this Regulatory Guide for the design and fabrication of the Replacement Reactor Vessel Closure Head. The Replacement Reactor Vessel Closure Head design and fabrication is performed in accordance with Design Specifications prepared and maintained by Westinghouse. RN 16-003

1.29 SEISMIC DESIGN CLASSIFICATION (REVISION 2; FOR COMMENT 2/76)/(REVISION 4-2013-REACTOR HEAD REPLACEMENT) RN 16-003

Seismic classification of structures, systems, and components for the Virgil C. Summer Nuclear Station is as described in Sections 3.2.1 and 3.8.

The classification of components by safety class provides the means of establishing applicable a seismic design requirements of both components and systems. At the time the Virgil C. Summer Nuclear Station was designed, duplication by special seismic classification was unnecessary since American National Standard (ANSI) N18.2 was considered to establish seismic design requirements of systems having components classified as Safety Class 1, Safety Class 2, or Safety Class 3. The structures described in Section 3.8 are classified as Seismic Category I.

Classification by this means is an alternate acceptable method of meeting the intent of Regulatory Guide 1.29, since the design construction and quality assurance provided fulfill the recommendations of Regulatory Guide 1.29.

V. C. Summer Commits to Revision 4-2013 of this Regulatory Guide for the design and fabrication of the Replacement Reactor Vessel Closure Head. This Replacement Reactor Vessel Closure Head design and fabrication is performed by Westinghouse with Seismic Classification consistent with the original replaced components. RN 16-003

1.30 QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTS AND ELECTRICAL EQUIPMENT (REVISION 0; 8/72)

Delete commitment. The Virgil C. Summer Nuclear Station commitment to Regulatory Guide 1.30 is no longer needed with the adoption of NQA-1-1994 Subpart 2.4. NQA-1-1994 Subpart 2.4 is equivalent to the requirements of ANSI N45.2.4-1972. RN 11-040

1.31 CONTROL OF STAINLESS STEEL WELDING (Revision 1; 6/73)/ (Revision 4-2013-Reactor Head Replacement) RN 16-003

1. Welding Under Daniel's Scope

Regulatory Position	Compliance
1.a	Procedures are qualified with filler metal containing 5-12% delta ferrite. Procedure qualification tests are examined by magnetic instruments to ensure a minimum delta ferrite of 3% at the surface. The above procedures ensure that weld deposits contain between 5 and 12% delta ferrite for wrought structures and between 5 and 15% for duplex cast structures.
1.b	Chemical analysis is performed by the filler metal manufacturer. For procedure qualification chemical analyses are performed on undiluted weld deposits and delta ferrite content is predicted by using an applicable constitution diagram to demonstrate compliance with Regulatory Position 1.a above.
1.c	For procedure qualification delta ferrite content in weld metal is determined using magnetic measurement devices.
1.d	Heat input is controlled during production welding by specifying amperage voltage, and maximum interpass limits. Stringers are preferred welding method. However, to meet the intent of the regulatory position for speed of travel, monitoring, the following is implemented: 1. Weaving does not exceed 3 electrode diameters (uncoated) or the gas cup orifice inside diameter whichever is greater. 2. Daniel welder qualification procedures are reviewed to preclude excessive weaving. 3. QC spot checks production welds to ensure compliance with the procedure.
1.e	Bend tests are conducted and evaluated in accordance with ASME Section IX.
2.	The results of the destructive and nondestructive test required in Regulatory Position 1 above are included in the certified qualification test report. Results are reported on the Record of Welding Procedure Qualification Tests.

Regulatory
Position Compliance

3. The welding materials used for production welds meet the requirements of Section III of the ASME B&PV Code. In addition, each Lot or Heat of Welding material meets the requirements of Regulatory Positions 1.a and 1.b above. Additionally, delta ferrite content is determined by the filler metal manufacturer with a magnetic measuring instrument.

4. Daniel considers use of only single lots and heats of filler material to be impracticable. Filler material meets the requirements of Position 3 above. Traceability of filler metal to joint is maintained.

5. Measurements are taken at the surface using a magnetic instrument and indicate a minimum of 3%. Surface measurements of delta ferrite are performed on all welds with a thickness greater than 1 inch. For welding stainless steel piping with a weld thickness of 1 inch or less, a sampling program has been initiated to sample 20% of the first 500 welds completed. At the end of this sampling program an engineering evaluation shall be made to determine whether sampling continuance is warranted.

6. In the event that requirements of the commitment to meet Position 5 above are not met, an engineering evaluation shall be performed which may include metallographic examinations performed on weld metal samples cut in a plane transverse to the weld location and correlated with magnetic measurement transverse. Weld acceptance will be based on the absence of unacceptable fissures or cracks.

7. Production welding is monitored for essential variables in accordance with ASME Section IX, and as described in 1.d above.

2. Welding Under Westinghouse Scope

The control of stainless steel welding for material procured by Westinghouse is discussed in Section 5.2.5.7. The Westinghouse production weld verification program, as described in WCAP-8324-A, June 1975, is a satisfactory substitute for conformance with the NRC Interim Position on this regulatory guide. The results of the verification program are summarized and documented in WCAP-8693, January 1976.

Revision 4 is utilized by Westinghouse for the Replacement Reactor Vessel Closure Head Design Specifications. RN 16-003

Regulatory
Position Compliance

3. Welding Under Piping Fabrication Scope

Welding materials are tested by the fabricator using the specific process(es) and the maximum welding energy inputs to be employed in production welding. The tests are in accordance with the requirements of ASME Section III, NB-2430.

The following additional requirements apply to stainless steel welding:

a. Tests on stainless steel weld metal include delta ferrite determinations indicating between 5 and 12% for materials to weld wrought base metals and between 5 and 15% for materials to weld cast metals.

b. These ferrite determinations are made on undiluted weld metal from each heat or lot of welding material and are made by performing chemical analyses on the weld deposits and predicting the delta ferrite content using the "Schaeffler Constitution Diagram for Stainless Steel Weld Metal," verified by one of the magnetic measurement methods in sub-item c.

c. Ferrite measurements are made on production welds with a thickness greater than 1 inch. For production welds 1 inch or less in thickness a sampling program has been initiated to randomly sample 20% of the first 500 welds completed. The results of this sampling are reviewed by the OWNER/ENGINEER to determine whether the sampling plan is continued as is, or altered.

   Production weld determinations are made using a Magna gage, Ferrite Scope, Elecometer, or Equal, calibrated to known test welds whose ferrite contents have been determined by the Magna gage-Schaeffler method. Acceptable production welds have a minimum ferrite content of 3% measured at the surface along the centerline of the completed weld.

d. In the event that the requirement of subitem c for at least 3% of ferrite is not met, an engineering evaluation is performed which may include metallographic examinations of samples of weld metal cut in a plane transverse to the weld. Acceptance of the joints is based on the absence of unacceptable fissures or cracks.

Regulatory
Position Compliance

4. Welding Under Other Scopes

For safety-related ASME Section III stainless steel components purchased by SCE&G or GAI, welding requirements meeting the intent of Regulatory Guide 1.31 recommendations have been imposed upon manufacturers. Welding materials and welding procedures meet the requirements of ASME Code, Section III and IX.

5. The reactor coolant pump lower bearing oil cooler flexible hoses have been manufactured in accordance with Revision 3 of Regulatory Guide 1.31.

1.32 ELECTRIC SAFETY-RELATED POWER SYSTEMS FOR NUCLEAR POWER RB PLANTS (REVISION 2, 2/77)

The Virgil C. Summer Nuclear Station design complies with the recommendations of Regulatory Guide 1.32 as discussed below:

1(a). Availability of Offsite Power - Comply. Refer to Sections 8.2.1.1 and 8.3.1.2.1.

1(b). Battery Charger Supply - Comply. Refer to Section 8.3.2.1.5.2.

1(c). Battery Performance Discharge Tests - Comply, with the exception of battery service test "intervals not to exceed 18 months." Our service tests are performed during refueling outages with a nominal interval of 18 months. However, due to scheduling requirements within outage windows, the interval between tests can slightly exceed 18 months. In addition, per Tech Specs, service tests are not performed during refueling outages that require "performance discharge tests." Refer to Section 8.3.2.2.2. RN 98-140

1(d). Independence of Redundant Standby Sources - Refer to positions on Regulatory Guides 1.6 and 1.75 in Appendix 3A.

1(e). Connection of Non-Class 1E Equipment to Class 1E Systems - Refer to position on Regulatory Guide 1.75 in Appendix 3A.

1(f). Selection of Diesel Generator Set Capacity for Standby Power Supplies - Refer to position on Regulatory Guide 1.9 in Appendix 3A.

2(a). Shared Electric Systems for Multi-Unit Nuclear Power Plants - Refer to position on Regulatory Guide 1.81 in Appendix 3A.

2(b). Availability of Electric Power Sources - Refer to position on Regulatory Guide 1.93 in Appendix 3A.

1.33 QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION) (REVISION 2; 2/78)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.33 with the following exceptions and clarifications: RN 11-040

1. The plant has programmatic control requirements in place that make the biennial review process redundant. These programmatic controls were affected in an effort to ensure that plant instructions and procedures are reviewed for possible revision when pertinent source material is revised, therefore maintaining the procedures current. In addition to these controls, the Quality Systems group will perform a biennial Quality Assurance Audit of the procedural development program utilizing a representative sampling process, thereby providing verification that the controls are effective in maintaining procedures current. SCE&G believes that this approach better addresses the intent of the biennial review process and is more acceptable from both a technical and a practical perspective than a static two-year review process. RN 01-115

2. The plant takes exception to paragraph C.4 regarding the increased frequencies required for the performance of the Nonconformance, Surveillance Testing, and Unit Staff audits. The specified frequencies in C.4 are six months, twelve months, and twelve months respectively. The plant will audit these areas at a minimum frequency of " . . . within a period of (2) two years." using the guidance of the Standard Review Plan 17.2. This change allows more flexibility in the scheduling of audits and allocation resources. Also, since previous audits have not identified any significant deficiencies in the stated programs, the frequency change will not decrease the effectiveness of the audits. RN 01-115 The plant will audit these areas every two years. Audits shall be performed at the intervals designated for each audit area. Schedules shall be based on the month in which the audit starts. Two year audits may be extended not to exceed 25 percent of its interval. The maximum time between audits will not exceed 30 months. When an audit interval extension greater than one month is used, the next audit for that particular audit area will be scheduled from the original anniversary month rather from the month of the extended audit. RN 06-035

1.34 CONTROL OF ELECTROSLAG WELD PROPERTIES (REVISION 0; 12/72)

Electroslag welding is not used for safety-related components at Virgil C. Summer Nuclear Station.

1.35 INSERVICE INSPECTION OF UNGROUTED TENDONS IN PRE-STRESSED CONCRETE CONTAINMENT STRUCTURES (REVISION 3; 4/79)

The surveillance program for the Virgil C. Summer Nuclear Station containment prestressing system is in compliance with the recommendations of Regulatory Guide 1.35 with the following exceptions and clarifications:

In place of the Lower Limit and 90% Lower Limit defined in this Guide, the 95% Base Value and 90% Base Value, respectively, are used. The Base Value is the force predicted for a tendon at the time of the surveillance. The Base Value is equal to the original stressing force minus the losses described in Proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

In the calculation of the Base Value, zero tolerance has been applied to the losses. The losses are combined by considering the interaction of the tendon stress relaxation and concrete creep using the procedure described in "A Method for Predicting Prestress Losses in a Prestressed Concrete Structure" which appeared in the Prestressed Concrete Institute Journal, March/April 1972. The Surveillance program is discussed in Section 3.8.1 and the Virgil C. Summer Nuclear Station Technical Specifications. 99-01

Following the third (5th year) tendon surveillance, the total losses predicted for the 4th and subsequent surveillances are based upon non-interaction of loss sources which is conservative compared to the interaction method previously used. For the non-interaction method each contributor to loss in tendon prestress force is evaluated individually and combined by direct summation.

1.36 NON-METALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL (REVISION 0; 2/73)

Regulatory Guide 1.36 is not applicable for components within the Reactor Building since only stainless steel mirror insulation or a mass type encapsulated in stainless steel is used on austenitic stainless steel piping and equipment.

For austenitic stainless steel piping and components outside the Reactor Building, Regulatory Guide 1.36 is followed.

1.37 QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS (REVISION 1; 3/07) RN 11-040

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.37. Procurement orders apply cleaning requirements during fabrication and packaging of safety-related components so that equipment is delivered to the site in a properly cleaned condition. Site procedures for the operational phase meet the requirements of ASME NQA-1-1994, Part II, Subpart 2.1 and this guide. RN 11-040

1.38 QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS (REVISION 0; 3/73)

Delete commitment. The Virgil C. Summer Nuclear Station commitment to Regulatory Guide 1.38 is no longer needed with the adoption of NQA-1-1994 Subpart 2.2. NQA-1-1994 Subpart 2.2 is equivalent to the requirements of ANSI N45.2.2-1973. RN 11-040

1.39 HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS (REVISION 2, 9/77)

Delete commitment. The Virgil C. Summer Nuclear Station commitment to Regulatory Guide 1.39 is no longer required with the adoption of NQA-1-1994 Subpart 2.3. NQA-1-1994 Subpart 2.3 is equivalent to the requirements of ANSI N45.2.3-1973. RN 11-040

1.40 QUALIFICATION TESTS OF CONTINUOUS-DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS (REVISION 0; 3/73)

The specifications for continuous-duty Class I motors installed within the containment stipulate the recommendations of IEEE-334-71. The vendor's qualification tests are in compliance with the recommendations of Regulatory Guide 1.40.

1.41 PREOPERATIONAL TESTING OF REDUNDANT ON-SITE ELECTRIC POWER SYSTEMS TO VERIFY PROPER LOAD GROUP ASSIGNMENTS (REVISION 0; 3/73)

The Virgil C. Summer Nuclear Station preoperational testing program complies with the recommendations of Regulatory Guide 1.41.

1.42 INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHT-WATER-COOLED NUCLEAR POWER REACTORS (REVISION 1; 3/74)

The NRC staff has withdrawn this Regulatory Guide.

1.43 CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS (REVISION 0; 5/73)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.43 as discussed herein. To ensure guide compliance, welding processes that induce underclad cracking by generating excessive heating and promoting grain coarsening in the base metal are not used. Known affected components are restricted to the reactor vessel, the steam generator, and the pressurizer.

Virgil C. Summer endorses Revision 1-2011 of this Regulatory Guide for the design and fabrication of the Replacement Reactor Vessel Closure Head. RN 16-003

The reactor vessel heads and shell courses were constructed of SA-533 Grade B Class 1 plate material made to a fine grain practice. The closure head and vessel flanges and the primary nozzles were constructed of SA-508 Class 2 forging material.

This plate and forging material was clad utilizing the shielded metal arc and the two-wire submerged arc processes which are considered low heat input processes. Since the plate material and the low heat input clad processes used on forging material are not subject to restrictions by the guide, the vessel is in compliance with Regulatory Position C.1. Regulatory Position C.2 is not applicable in this case. The reactor vessel fabricator monitored and recorded the weld parameters to verify compliance with the parameters established by the procedure qualifications of Regulatory Position C.3.

The steam generator and the pressurizer parts which are clad are constructed of SA-533 Grade A Class 2 and SA-508 Class 2a steels. These materials are made to fine grain practice and welding is done with low heat input techniques.

1.44 CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL (REVISION 0; 5/73)

The Virgil C. Summer Nuclear Station complies with the recommendations of this guide as discussed herein. Specific items pertaining to this guide are discussed in Section 5.2.5. For those exceptions where sensitized material is used, justification is provided in WCAP-7735, "Topical Report-Compiled by W. S. Hazelton, August 1974, Sensitized Stainless Steel in Westinghouse PWR Steam Supply System (WNES Class 3)."

Compliance with separate guide positions is as follows:

1. The use of processing, packaging, and shipping controls and preoperational cleaning to preclude adverse effects of exposure to contaminants on stainless steel materials are in accordance with the guide. Additional information is given in discussions of Regulatory Guides 1.37 and 1.38.

2. Where practical, austenitic stainless steel starting materials are utilized in the final heat treated condition required by the respective ASME Code Section II material specification for the particular type or grade of alloy in accordance with the guide.

3. Compliance with this guide position is discussed in Section 5.2.5.3.

4. This subject is discussed in Section 5.2.5.5. To provide a more rapid reduction of the oxygen concentration by reaction with hydrazine an upper limit of 250° F is used. Startup operations provide for hydrazine additions after the temperature is about 225° F. Oxygen scavenging at 225° F is rapid and complete.

Components with stainless steel sensitized in the manner expected during component fabrication and installation will operate satisfactorily under normal plant chemistry conditions in pressurized water reactor systems, because chlorides, fluorides, and particularly oxygen, are controlled to very low levels (0.15 ppm each).

Delta ferrite control is discussed in the Regulatory Guide 1.31 position and Section 5.2.5.7.

5. The compliance with this regulatory position is discussed in Section 5.2.5.6. Also, delta ferrite control is discussed in the Regulatory Guide 1.31 position and Section 5.2.5.7.

6. Compliance with this guide position in application to Westinghouse scope items is discussed in Section 5.2.5.5. For welding under Daniel's scope, E-308L-16 and ER-108L filler material is used in stainless steel production welding. In addition, an intergranular corrosion test is performed for each stainless welding procedure qualification. The use of E-308-16 and E-308 filler material when specified by Westinghouse is acceptable in accordance with criteria specified in Section 5.2.5.5.

Virgil C. Summer endorses Revision 1-2011 of this Regulatory Guide for the design and fabrication of the Replacement Reactor Vessel Closure Head. RN 16-003

1.45 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS (REVISION 0; 5/73)

The Virgil C. Summer Nuclear Station meets the technical recommendation of Regulatory Guide 1.45. Diverse methods for determination of reactor coolant leakage are employed in the design of Virgil C. Summer Nuclear Station which follows the recommendations of Regulatory Guide 1.45. Specific methods for reactor coolant leakage detection are described in Section 5.2.7.

1.46 PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT (REVISION 0; 5/73)

The following information and exceptions are provided to clearly define the means of implementing the recommendations of Regulatory Guide 1.46. The modifications specified below reflect the recommendations of relevant standards as referenced. An exception to Regulatory Guide 1.46 Regulatory Position C.4.d is taken to specify the measures for restraint against pipe whipping as a result of the design basis breaks postulated to occur at the locations specified in accordance with the guide need not be provided for piping where both the following pipe normal operating conditions are met:

1. The maximum normal operating temperature is less than or equal to 200°F and;

2. The maximum normal operating pressure is less than or equal to 275 psig.

Normal operating conditions are defined as those during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods, but qualify as moderate-energy fluid systems for the major operational period. The above change is in agreement with the NRC's latest position as stated in "Branch Technical Position - MEB No. 1" dated September 23, 1974, on pipe Rupture Outside Containment, and definitions included in Regulatory Guide 1.XX Appendix D2 dated March 15, 1974. SCE&G concurs that such an exception constitutes a reasonable approach.

A second exception is taken to the criteria of Regulatory Guide 1.46 with respect to break area for a longitudinal pipe rupture. A postulated longitudinal rupture will be assumed to have an area of opening equal to the cross-sectional flow area of the affected pipe in lieu of two cross-sectional flow areas as suggested in Regulatory Guide 1.46. This assumption has been shown to be conservative. This change of criteria is in agreement with both Regulatory Guide 1.XX and ANSI N176.

Regulatory Guide 1.46 and the NRC's Mechanical Engineering Branch's Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," June 1987 differ significantly on the postulated orientation and location of longitudinal breaks. Regulatory Guide 1.46 requires longitudinal breaks to be postulated at all break locations in piping 4 inches nominal pipe size and larger and oriented at any point around the pipe circumference. MEB 3-1 does not recommend a longitudinal break to be postulated at terminal ends, unless longitudinal welds are used, and most importantly for longitudinal breaks states "Splits should be oriented (but not concurrently) at two diametrically-opposed points on the piping circumference such that the jet causes out-of-plane bending of the piping configuration." RN 02-016

Guidelines of MEB 3-1 are utilized to determine the location and orientation of longitudinal breaks inside the Reactor Building in preference to those of Regulatory Guide 1.46. Such use of MEB 3-1 is endorsed by NRC's Standard Review Plan, section 3.6.2, paragraph II.1.

Dynamic forces resulting from circumferential pipe breaks are assumed to cause whipping in any direction of either end of the ruptured pipe normal to the pipe axis unless the direction of whipping can be defined giving due consideration to the geometry of the ruptured pipe and the effects of a less than instantaneous pipe rupture. Regulatory Guide 1.46 makes no provisions for the defining of pipe whip direction.

Regulatory Guide 1.XX and ANSI N176 limit the direction of pipe whip to the plane defined by the piping geometry. The stated assumption while more liberal than that of Regulatory Guide 1.46, is more conservative than the position set forth in Regulatory Guide 1.XX and ANSI N176 and can be substantiated using simple mechanics of propulsion.

For NSSS scope of supply, the Virgil C. Summer Nuclear Station utilizes the Westinghouse generic break criteria as specified in WCAP-8083-P-A. An AEC letter dated May 22, 1974 from D. B. Vassallo to R. Salvatori concerning WCAP-8083-P-A accepts the WCAP criteria and presents a discussion of the six items identified in the acceptance letter. These provide a level of protection equivalent to that provided by the application of Regulatory Guide 1.46 criteria. The protection against dynamic effects associated with the postulated rupture of piping are discussed in Section 3.6.

1.47 BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS (REVISION 0; 5/73)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.47 for deliberate operator action as discussed in Section 7.1.2.6.

1.48 DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS (REVISION 0; 5/73)

The Virgil C. Summer Nuclear Station components are in compliance with the recommendations of Regulatory Guide 1.48. Additional design details are given in Section 3.9.

Equipment Under Gilbert Scope

The seismic specification requires qualification of safety-related equipment by test and/or analyses. The parent procurement specification specifies that qualification of equipment be demonstrated under loading combinations, and design limits delineated in Regulatory Guide 1.48. The loading combinations stress limits and assurance of operability requirements, as applicable, are indicated in each specification. A sample of the requirements for each category is shown below.

Specification Paragraphs to be Used When Addressing Regulatory Guide 1.48 Requirements

Bidder shall offer equipment for which "Loading Combination" are in compliance with the Gilbert specification. The bidder shall use the following design limits and load combinations when complying with the specification[1].

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(1) Included as preface for each category.

1. Nonactive ASME Code Class 2 and 3 Valves

   The design of Code Class 2 and 3 valves encompasses the use of pressure temperature ratings. The design limits given herein are in terms of Pr which is the primary pressure rating corresponding to the maximum transient temperature for each plant condition as specified in paragraphs NC-3511 and ND-3511 of the ASME Code Section III for Code Class 2 and 3 valves respectively. To assure pressure retaining integrity the limits for Pr are set as follows:

   a. The primary pressure rating Pr should not be exceeded more than 10% when the component is subject to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake or loadings conditions associated with the emergency plant condition.

   b. Pr should not be exceeded by more than 20% when the component is subject to concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shut-down earthquake, and the dynamic system loadings associated with the faulted plant condition.

2. Active ASME Code Class 2 and 3 Valves

   To provide pressure retaining integrity and assurance of operability for active valves of Code Class 2 and 3, Pr should not be exceeded for the combinations of loadings delineated herein. (See Note 2.1 and 2.2.)

   a. The primary pressure rating Pr should not be exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

   b. The primary pressure rating Pr should not be exceeded when the component is subject to loadings associated with the emergency plant condition.

   c. The primary pressure rating Pr should not be exceeded when the component is subject to concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

Note 2.1: It is strongly preferred that valves having electric, pneumatic, or hydraulic operators; or having position limit switches or other miscellaneous electrical or pneumatic control devices, be qualified by test rather than by analysis. Should testing of the complete assembly be unavailable, it is then preferred that the operators and control devices be qualified by test. Qualification of these components by analysis only is judged least desirable. There will be major evaluation considerations given this.

Note 2.2: If proof of operability is to be shown by analysis, it is suggested that a deformation (interference) analysis is performed. The purpose of this analysis would be to show that all moving parts are uninhibited in movement due to deformation of components caused by any loading conditions as described above.

3. Non-Active ASME Code Class 1 Pumps and Valves (Designed by Analysis)

   In order to assure pressure-retaining integrity for nonactive Code Class 1 pumps and valves, the upset, emergency, and faulted operating condition category design limits of NB-3200 of Section III of the ASME Code should not be exceeded when the component is subjected to the following loading combinations:

   a. The design limits specified in NB-3223 of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

   b. The design limits specified in NB-3224 of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

4. Non-Active ASME Code Class 1 Valves (Designed by Standard or Alternative Design Rules)

   Standard or alternative design rules, which encompass the use of pressure-temperature ratings, for Code Class 1 valves are specified by NB-3512 and NB-3513 of Section III of the ASME Code. The design limits given herein are in terms of Pr which is the primary pressure rating corresponding to the maximum transient temperature for each plant condition as specified in Tables NB-3531-1 to NB-3531-7 of Section III of the ASME Code. To assure pressure-retaining integrity, Pr should not be exceeded by more than 10, 20, and 50% when the component is subjected to the following combinations:

   a. Pr should not be exceeded by more than 10% when the component is subjected to concurrent loadings associated with either the normal plant conditions or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

   b Pr should not be exceeded by more than 20% when the component is subjected to the loadings associated with the emergency plant condition.

   c. Pr should not be exceeded by more than 50% when the component is subjected to concurrent loadings associated with the normal plant conditions, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

5. ASME Code Class 2 and 3 Vessels (Designed to Division 1 of Section VIII of the ASME Code) (See Note 5.1)

   To provide assurance of pressure-retaining integrity for Code Class 2 and 3 vessels (Designed to Division 1 of Section VIII of the ASME Code) the allowable stress value S (as specified in Appendix 1 of Section III of the ASME Code) should not be exceeded by more than 10% when the component is subjected to the loading combinations identified by items a and b below, and should not be exceeded by more than 50% when the component is subjected to the loading combinations identified by item c below.

   a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

   b. Loadings associated with the emergency plant condition.

   c. The allowable stress value S should not be exceeded by more than 50% when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

Note: 5.1 If a more detailed analysis is performed, Division 1 vessels should meet, as a minimum equations (1) and (2) below. Equation (1) is applicable to items a and b above. Equation (2) is applicable to item c, above.

(1) $\sigma_M < 1.1S > \dfrac{\sigma_M + \sigma_b}{1.5}$

(2) $\sigma_M < 1.5S > \dfrac{\sigma_M + \sigma_b}{1.5}$

Where:

σ_M = Primary Membrane Stress

σ_b = Primary Bending Stress

S = Allowable stress value as specified in Appendix 1 of Section III of the ASME Boiler and Pressure Vessel Code.

6. ASME Code Class 2 Vessels (Designed to Division 2 of Section VIII of the ASME Code)

   To provide assurance of pressure-retaining integrity for Code Class 2 Vessels, (Designed to Division 2 of Section VIII of the ASME Code) the upset, emergency, and faulted operating condition category design limits of NB-3200 of Section III of the ASME Code should not be exceeded when the component is subjected to the following loading combinations:

   a. The design limits specified in NB-3223 of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

   b. The design limits specified in NB-3224 of the ASME Code should not be exceeded when the component is subjected to loadings associated with the emergency plant condition.

   c. The design limits specified in NB-3225 of the ASME Code should not be exceeded when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

7. ASME Code Class 2 and 3 Piping

   To provide assurance of pressure retaining integrity for Code Class 2 and 3 piping, the design limits specified in NC3611.1(b)(4)(c)(b)(1) of the Winter 1972 Addenda to Section III of the ASME Code (i.e., 1.2 Sh) are not exceeded when the piping is subjected to the loading combinations identified in items a and b, below. However, for short sections of piping exposed to jet impingement from postulated cracks or breaks in adjacent piping, a stress limit of 1.5 Sh may be used.

   The design limits specified in NC3611.1(b)(4)(c)(b)(2) of the Winter 1972 Addenda to Section III of the ASME Code (i.e., 1.8 Sh) are not exceeded when the piping is subjected to the loading combinations identified in item c, below. However, for short sections of piping exposed to jet impingement from postulated cracks or breaks in adjacent piping, a stress limit of 2.4 Sh may be used.

   a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

   b. Loadings associated with the emergency plant conditions.

c. Concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

8. Non-Active ASME Code Class 2 and 3 Pumps

In order to assure pressure-retaining integrity for nonactive Code Class 2 and 3 pumps, the primary membrane stress should not be exceeded by more than 10% of S (as specified in Appendix 1 of Section III of the ASME Code) and the sum of the primary membrane plus primary bending stresses should not be exceeded by more than 65% of S when the components is subjected to the following load combinations:

a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

b. Loadings associated with the emergency plant condition.

In addition, the primary membrane stress should not be exceeded by more than 20% of S, and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 80% of S when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

9. Active ASME Code Class 2 and 3 Pumps

To provide increased assurance that unacceptable deformation affecting operability of active Code Class 2 and 3 pumps will not result, the primary membrane stress should not exceed S (as specified in Appendix 1 of Section III of the ASME Code) and the sum of the primary membrane plus primary bending stresses should not be exceeded by more than 50% of S when the component is subjected to the following loading combinations: (See notes 9.1 and 9.2.)

a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of safe shutdown earthquake.

b. Loadings associated with the emergency plant condition.

c. Concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

Note 9.1: The design limits given below may be used for the applicable loading combinations if stringent analysis and/or testing confirms that operability will not be impaired when the component is designed to these limits.

The primary membrane stress should not be exceeded by more than 10% of S and the sum of the primary membrane plus primary bending stresses should not be exceeded by more than 65% of S when the component is subjected to the following load combinations:

a. Concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of 50% of the safe shutdown earthquake.

b. Loadings associated with the emergency plant condition.

The primary membrane stress should not be exceeded by more than 20% of S, and the sum of the primary membrane and primary bending stresses should not be exceeded by more than 80% of S when the component is subjected to concurrent loadings associated with the normal plant condition, the vibratory motion of the safe shutdown earthquake, and the dynamic system loadings associated with the faulted plant condition.

Note 9.2: In addition to compliance with the design limits specified, it is strongly preferred that the assurance of operability under all design loading combinations be provided by test. In the performance of test or analysis to demonstrate operability, the structural interaction of the entire pump-motor assembly should be considered. Should this be unavailable, it is then preferred that the components, with any appurtenances attached thereto, be qualified by test. If superposition of test results for other than the combined loading condition is proposed, the applicability of such a procedure should be demonstrated. Qualification of these components by analysis only is judged least desirable. There will be major evaluation consideration given this.

<u>Westinghouse Supplied Equipment</u>

Westinghouse equipment was designed to ensure structural integrity and operability. However, it must be realized that the load combinations and stress limits that were used reflect AEC (NRC) recommendations that were in effect when the construction permit for this plant was issued and when the components were purchased and designed. Furthermore, the codes and procedures which were available when the components were purchased are based on conservative design requirements rather than detailed stress analyses. These codes and procedures have been widely used by the nuclear industry for the design of components which are installed in plants that are presently operating. A discussion of the stress limits and loading combinations is presented in Section 5.2.1 for Code Class 1 components and in Section 3.9.2 for Code Class 2 and 3 components. The operability of active components are discussed in Sections 5.2.1, 3.9.2, and 3.9.4.

Prior to installation, the valves are subjected to shell hydrostatic tests, seat leakage tests, and functional tests to show that the valves will open and close within the specified time limits when subjected to the design differential pressure. After installation the valves undergo cold hydrostatic tests, hot functional tests to verify operation, and periodic inservice inspection and operation to assure the continued ability of the valves to operate. Class 1 active valves are designed in accordance with the ASME Code, Section III. In addition to Class 2 and 3 active valves designed in accordance with the ASME Code Section III, there are some Class 2 and 3 valves within the pump and valve code. These are designed to the requirements of the ANSI B16.5 Code.

Active pumps are designed in accordance with the ASME Code Section III. The stress levels in the pump's pressure retaining parts do not exceed those allowed by the Code. Forces resulting from seismic accelerations in the horizontal and vertical directions are included in the analyses of the pumps and their supports. To eliminate any amplification of the seismic floor acceleration in the pump support structure, the supports are designed to have natural frequencies in excess of 30 Hertz.

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.

1.49 POWER LEVELS OF NUCLEAR PLANTS (REVISION 1; 12/73)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.49 since the power level is less than 3800 MWt.

1.50 CONTROL OF PREHEAT TEMPERATURES FOR WELDING OF LOW-ALLOY STEEL (REVISION 0; 5/73)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.50 with the following clarifications and exceptions:

1. Welding Under General Contractor's Scope

Regulatory Position	Methods Of Compliance
1	Low alloy steels do not include those designated as P-1 by the ASME Code, Section IX.
1.a	A minimum preheat and maximum preheat and maximum interpass temperature is specified by the appropriate welding procedure.
1.b	Procedures are qualified in accordance with the ASME Code, Section IX. In general, preheat conditions are not maintained until initiation of post weld heat treatment.

Regulatory
Position Methods Of Compliance

2 For production welds, the preheat temperature is not maintained until a post-weld heat treatment has been performed. The alternate position as stated in Regulatory Guide Position 4 is utilized.

3 Preheat and interpass temperatures are monitored by construction and QC personnel.

4 Acceptance of welds is determined by NDE after postweld heat treatment is complete.

2. Welding Under Westinghouse's Scope

Westinghouse considers that this guide applies to ASME Section III Class 1 components.

The Westinghouse practice for Class 1 components is in agreement with the positions of Regulatory Guide 1.50 except for Regulatory Position 1.b and 2. For Class 2 and 3 components, Westinghouse does not apply any of the Regulatory Guide 1.50 recommendations.

In the case of Regulatory Position C.1.b, the welding procedures are qualified within the preheat temperature ranges required by Section IX of the ASME Code. Westinghouse experience has shown excellent quality of welds using the ASME qualification procedures.

In the case of Regulatory Position C.2, the Westinghouse position is that this guide recommendation is both unnecessary and impractical. Code accepted low alloy steel welds have been and are being made under present Westinghouse specified procedures. It is not necessary to maintain the preheat temperature until a post weld heat treatment has been performed, as recommended by the guide, in the case of large components.

In the case of reactor vessel main structural welds, the practice of maintaining preheat until the intermediate, or final post weld heat treatment, has been followed by Westinghouse. In either case, the welds have shown high integrity.

Westinghouse practices are documented in WCAP-8577 which has been accepted by the NRC.

For the reactor vessel the soundness of the welds has been verified by extensive ultrasonic testing as permitted by Regulatory Position C.4.

Revision 1-2011 is used for the design and fabrication of the Replacement Reactor Vessel Closure Head by Westinghouse in the Design Specifications. RN 16-003

Regulatory Position | Methods Of Compliance

3. Welding Under Other's Scope

Low alloy steel items falling within the category of items of this guide are limited to the exhaust piping for the diesel engine for which piping is fabricated in accordance with guide welding recommendations.

1.51 INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS (REVISION 0; 5/75) RN 01-113

The NRC staff has withdrawn this regulatory guide.

1.52 DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ENGINEERED SAFETY FEATURE ATMOSPHERIC CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (REVISION 1; 7/76)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.52 with the following exceptions and clarifications. A comparison of the Guide's recommendations and our compliance for the Reactor Building HEPA filter system, Control Room emergency filter system, and Fuel Handling Building charcoal exhaust system is given in Section 6.5.1. Table 6.5.1 indicates portions of SCE&G position of Regulatory Guide 1.52 which comply with the recommendations of Revision 2.

1.53 APPLICATION OF THE SINGLE-FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS (REVISION 0: 6/73)

As described in Section 7.1.2.7 and 7.3.2.2, protection systems included in the design of the Virgil C. Summer Nuclear Station can tolerate single failure without degrading the system functional capabilities to unacceptable levels and are in compliance with the recommendation of Regulatory Guide 1.53. 00-01

1.54 QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER COOLED NUCLEAR POWER PLANTS (REVISION 0; 6/73)

The Virgil C. Summer Nuclear Station is in compliance with the recommendations of Regulatory Guide 1.54 with the following clarifications and exceptions.

For the Westinghouse scope of supply, Westinghouse employs process specifications and the Westinghouse Quality Assurance Program, including quality assurance surveillance and auditing, to provide adequate confidence that coating work within Westinghouse scope will perform satisfactorily in service.

An alternate method of compliance with this regulatory guide has been submitted to the NRC (via letter NS-CE-1352, dated February 1, 1977, to Mr. C. J. Heltemes, Jr., Quality Assurance Branch, NRC, from Mr. C. Eicheldinger, Westinghouse PWRSD, Nuclear Safety Department) and accepted (via letter, dated April 27, 1977, to Mr. C. Eicheldinger from Mr. C. J. Heltemes, Jr.).

1.55 CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (REVISION 0; 6/73)

Regulatory Guide 1.55 has been withdrawn per NRC letter of July 8, 1981. However, concrete placement for the Virgil C. Summer Nuclear Station is still in compliance with the recommendation of Regulatory Guide 1.55 with the following exceptions.

Creep tests for concrete are performed for the Reactor Building only. Loss of prestress through creep is not applicable to nonprestressed structures.

Concrete placement and testing are discussed in Section 3.8.1.

1.56 MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (REVISION 0; 6/73)

Regulatory Guide 1.56 is not applicable to the Virgil C. Summer Nuclear Station since it uses a PWR.

1.57 DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (REVISION 0; 6/73)

Regulatory Guide 1.57 is not applicable to the Virgil C. Summer Nuclear Station since a concrete-metal containment structure that relies on concrete for its structural integrity is used.

1.58 QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL (REVISION 1; 9/80)

Delete commitment. Regulatory Guide 1.58 has been withdrawn. The requirements of NQA-1-1994 Basic Requirement 2, Supplementary Requirement 2S-1, Supplementary Requirement 2S-2, and Appendix 2A-1, are equivalent to the requirements of ANSI N45.2.6-1978. RN 11-040

1.59 DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS (REVISION 1; FOR COMMENT; 4/76)

Regulatory Guide 1.59 applies to plants by riverside, streamside, on estuaries, on coastal plain, or at the Great Lakes. It does not presently apply to plants with man-made reservoirs for cooling systems. SCE&G, however, performed the analyses consistent with the guidelines presented in Appendix A of the Regulatory Guide. Flood design considerations are discussed in Sections 2.4.2, 2.4.3, and 2.4.4.

1.60 DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (REVISION 1; 12/73)

The construction permit for the Virgil C. Summer Nuclear Station was issued based on the seismic data given in the PSAR. Considerable investigation was performed by the AEC/DOL and the ACRS prior to issuance. Designs have been accomplished based on data which was submitted and approved at that time. The same data is presently utilized for equipment procurement and testing. Refer to Sections 2.5.2 and 3.7. Since the data given in the PSAR had AEC/DOL and ACRS approval, this constitutes an acceptable alternate to this regulatory guide.

1.61 DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (REVISION 0; 10/73)

The construction permit for the Virgil C. Summer Nuclear Station was issued based on the seismic data given in the PSAR. Considerable investigation was performed by the AEC/DOL and the ACRS prior to issuance. Designs have been accomplished based on data which was submitted and approved at that time. The same data is presently utilized for equipment procurement and testing. Refer to Sections 2.5.2 and 3.7. Since the data given in the PSAR had AEC/DOL and ACRS approval, this constitutes an acceptable alternate to this regulatory guide.

1.62 MANUAL INITIATION OF PROTECTION ACTIONS (REVISION 0; 10/73)

Manual initiation of protective actions at the systems level may be accomplished from the control room of the Virgil C. Summer Nuclear Station in compliance with the recommendations of Regulatory Guide 1.62. Drawings and descriptions of the manual initiation circuits are found in Chapters 7 and 8.

1.63 ELECTRICAL PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURE FOR WATER-COOLED NUCLEAR POWER PLANTS (REVISION 1; 5/77)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.63 with the following clarifications.

This guide states the regulatory position with respect to IEEE 317-1976. The GAI procurement specification (No. 559) references, and requires compliance with IEEE

317-1972, as applicable to the Virgil C. Summer Nuclear Station. In Section C.1 of the guide IEEE 279-1971 is referenced as a guide for circuit overload protection, but IEEE-279 does not specifically apply to either power circuits or overload protection. Penetration assemblies are qualified to maintain containment integrity with a single failure of any overcurrent protective device.

In order to meet the recommendations of Regulatory Guide 1.63, primary and backup overcurrent protective devices are used on Class 1E and non-Class 1E power and control circuits feeding through penetrations, or justification/analysis is performed demonstrating that primary and backup overcurrent protection is not required. A listing of these protective devices is found in Appendix 8G.

The penetration conductors have short-time overload and short circuit ratings consistent with the characteristics of the backup protective-device, assuming the failure of the primary protective device. For the reactor coolant pump circuits fed from 7.2 kV switchgear, the motor feeder protective relays are coordinated with, and backed up by, the bus protective relays. The control power for the trip coils of the reactor coolant pump breakers are supplied by the plant Class 1E battery which is independent of the station battery which provides control power for the 7200 volt bus supply breakers. This precludes loss of protection from a single loss of control power.

For circuits fed from 480 volt switchgear, the time delay and instantaneous overcurrent trips or the motor feeder air circuit breaker are coordinated with, and backed up by, either the overcurrent relays of the bus protective breaker or fuses in the motor feeder circuit. The 480 volt breakers are provided with solid state tripping devices which are powered by the fault current through individual power supply sensors and signal sensors. Therefore, independent DC control power is not required.

For circuits fed from motor control centers, the normal overcurrent protective devices are backed up by a thermal-magnetic current limiting circuit breaker added to each circuit.

For motor operated valves that have their overload protection device bypassed during safety injection (see discussion of Regulatory Guide 1.106), a thermal-magnetic breaker is substituted for the normal magnetic only circuit breaker. The thermal-magnetic breaker is backed up by a thermal-magnetic current limiting circuit breaker.

Control rod drive power circuits are protected by two sets of fuses which are integral with the rod drive control system. These fuses are sized to protect solid state components within the control system from overcurrent levels which are conservative compared to penetration conductor capabilities.

Power circuits feeding the pressurizer heaters are protected by the heater bank power distribution circuit fuses and are backed up by the power distribution panel main fuses. 00-01

Power circuits supplied from a-c and d-c power panels are protected with thermal-magnetic circuit breakers which are backed up by fuses in series with the breakers.

AC and DC control circuits and protection associated with electrical penetrations are divided into four basic categories:

Category 1 AC control circuits supplied by 480-120V control power transformers located in motor control centers are each protected with a fuse on the secondary side of the transformer. An analysis of data obtained from tests demonstrated that the available short circuit current on the secondary of these 480-120V MCC control transformers is limited by the transformers impedance and eventual destruction, which occurs before reaching the I^2t capabilities of the penetration conductors. Therefore, the transformer will provide back-up protection in the event that the fuse on the secondary side of the control transformer fails to clear a fault.

Category 2 Certain AC and DC control circuits which are justified by examination of a potential fault and the type of circuit shorted do not require primary and backup overcurrent protection because the potential effect of their associated faults is negligible. This category applies to low level analog circuits, circuits supplied by equipment whose internal power supplies and wiring are current limiting by design, and circuits whose postulated fault will affect circuit operation but will not create a short circuit condition because the circuit load is located outside the Reactor Building.

Category 3 The majority of DC control circuits (those circuits which are not identified as Category 2) are supplied from ungrounded DC power systems. Each circuit is protected with two fuses, one in each positive and negative leg of the circuit. One fuse provides backup protection for the other. Both fuses are sized to clear a fault before the I^2t capability of the penetration conductors is reached.

Category 4 The majority of AC control circuits (i.e., those circuits which are not identified as Category 1 and 2) are protected with primary and backup overcurrent protective devices in series. Both devices are sized to clear a fault before the I^2t capability of their associated penetration conductors is reached.

1.64 <u>QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS (REVISION 2; 6/76)</u>

Delete commitment. Regulatory Guide 1.64 has been withdrawn. The requirements of NQA-1-1994 Basic Requirement 3 and Supplementary Requirement 3S-1 are equivalent to the requirements of ANSI N45.2.11-1974 and Regulatory Guide 1.64. RN 11-040

1.65 MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS (REVISION 0; 10/73)

The reactor vessel closure nuts and washers were procured after the issuance of Regulatory Guide 1.65 and as result are in compliance with the recommendations of this guide. RN 15-021

Regulatory Guide 1.65 was issued after the Virgil C. Summer Nuclear Station reactor vessel stud bolting material was procured. However, this material directly meets the major portions of the guide. The only exceptions are that a maximum tensile strength of 170,000 psi was not actually specified although achieved, and the calibration for the radial ultrasonic testing was established on each stud. Each point of compliance and the exceptions are discussed in detail as follows: RN 15-021

1. Bolting Materials

The Virgil C. Summer Nuclear Station reactor vessel closure head studs were manufactured from SA-540 Grade B24 material. This material directly complies with Regulatory Position C.1.a of the guide. RN 08-030 14-003 15-021

The tensile and Charpy-V-notch impact data for the stud bolt bar stock is as follows: RN 15-021

	Room Temperature Ultimate Tensile Strength (range)	Charpy V-notch Impact Test (range) Energy at 10°F	Lateral Expansion
Stud bolt bar stock	156,000-163,150 psi	48-57 ft-lbs	27-35 mils

RN 14-003 15-021

Although, as noted, a maximum tensile strength was not specified, the maximum measured ultimate tensile strength data do not exceed 170,000 psi and therefore meet the Guide Position C.1.b(1) tensile strength criterion. The Charpy V-Notch impact testing was performed according to the ASME SA-370 standard, and the actual results are in excess of 45 ft-lbs and 25 mils lateral expansion. The tests were performed at 10°F rather than the higher temperatures allowed by Par. IV.A.4 of Appendix G to 10 CFR 50, since the tests were conducted to the ASME Code Section III addenda in effect at the time of procurement and prior to issuance of Appendix G of the Regulation. If tested at the allowed higher temperature, the impact energy requirements of 45 ft-lbs and 25 mils lateral expansion would have been met. The guide position C.1.b.(3) has also been met since the bolt materials are not plated and are lubricated according to the guide position.

2. Inspection

Review of the mill test data, ultrasonic testing specifications, and ultrasonic test procedures applicable to the Virgil C. Summer Nuclear Station reactor vessel stud bolting material indicates that the intent of the portion of the NRC Regulatory Guide 1.65, Section C.2, concerning ultrasonic testing has been met as described in the following:

The bolting materials were ultrasonically examined according to Westinghouse approved procedures which require that:

a. The 100% examination is conducted after heat treatment and prior to machining.

b. The material is scanned in both the radial and axial directions, per the ASME Code, Section III, Paragraphs NB-2584 and 2585, and ASME SA-388.

c. The calibration for the radial examination is based on a standard back reflection established in an indication-free area of each stud.

d. The calibration for the axial scan is based on a distance corrected reference level established on the responses from 3/8 in. diameter flat bottomed holes in a representative calibration block, per the ASME Code, Section III, Paragraph NB-2583.

e. The acceptance criteria for radial testing state that material containing discontinuities that produce an indication exceeding 20% of the calibration back reflection are unacceptable. For axial testing, material containing a discontinuity or discontinuities producing an indication or indications, equal to or greater than the primary distance amplitude curve reference line are unacceptable.

The above procedures comply with the guide recommendations except for procedure (c). In this case, the guide states that the calibration standard used to establish the first back reflection for the ultrasonic test should be based on sound representative material. The section of the standard should be based on a preliminary scan. Westinghouse believes that procedure (c) above is a more directly applicable procedure, and while not directly following the guide position, meets the intent of the guide.

Magnetic particle examination was performed on the studs and nuts after final heat treatment and threading per the ASME Code, Section III, Paragraph NB-2583.

3. Protection Against Corrosion

During venting and filling of the pressure vessel and while the head is removed, Westinghouse procedures require that the stud bolts, nuts, and washers and stud bolt holes in the vessel flange are protected from corrosion and contamination. Section 5.4.2.2, and Section 5.4.4.4 thus meeting Position C.3 of the guide. Design of the reactor vessel studs, nuts, and washers, providing protection against corrosion by allowing them to be completely removed during each refueling and placed in storage racks on the containment operating deck, as required by Westinghouse refueling procedures. The stud holes in the flange are sealed with special plugs before removing the reactor closure. Thus, the bolting materials and stud holes are never exposed to the borated refueling cavity water.

4. Inservice Inspection

The reactor vessel design permits inservice inspection per the Guide Position C.4 and as discussed further in Sections 5.4.4.4, and 5.7.1.

1.66 NON-DESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS (REVISION 0; 10/73)

The NRC Staff has withdrawn this Regulatory Guide.

1.67 INSTALLATION OF OVERPRESSURE PROTECTIVE DEVICES (REVISION 0; 10/73)

Safety and relief valves and associated piping and valve headers, within the scope of Regulatory Guide 1.67, installed at the Virgil C. Summer Nuclear Station have been designed, analyzed, and qualified in accordance with the recommendations of this guide.

1.68 PREOPERATIONAL AND INITIAL STARTUP TEST PROGRAMS FOR WATER-COOLED POWER REACTORS (REVISION 0; 11/73)

The Virgil C. Summer Nuclear Station test programs comply with the recommendations of Regulatory Guide 1.68. A detailed discussion of the preoperational and initial startup test program is given in Chapter 14.

It should be noted that the Virgil C. Summer Nuclear Station plant computer does not perform any safety-related control or essential monitoring function nor is it required for the operation of the plant; therefore Item D.1.r of Appendix A is not applicable.

For Appendix item B.1.C. refer to Table 14.1-55 and Section 4.2.3.4.1. For Appendix C, item C, Fourth paragraph, use the following: Nuclear instruments should be calibrated. A neutron count rate (on the order of at least 1/2 counts per second) should register on startup channels before the startup begins, and the signal-to-noise ratio should be at least two. A conservative startup rate limit (no smaller than 30-seconds period) should be employed in obtaining low power. Power should be leveled after attaining criticality and before attaining sensible nuclear heat. Low-power testing should be performed at this level.

1.68.2 INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 1; 7/78)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.68.2. (See Table 14.1-79.)

1.69 CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS (REVISION 0; 12/73)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.69 as follows:

The recommended practices contained in ANSI N101.6-1972, have been incorporated into Regulatory Guide 1.69. The ANSI standard directs itself first to the nuclear shielding aspects and secondly to structural design and construction of these shields. This standard puts special emphasis on the use of high density concrete. However, for the Virgil C. Summer Nuclear Station, normal density concrete has been used in shielding calculations. The structural design and construction of the concrete shield walls follows the standards and recommended practices in the ANSI standard.

1.70 STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS (REVISION 1; 10/72)

Per discussions held among the NRC Staff, Westinghouse, Gilbert, and SCE&G on March 12, 1976 this FSAR was prepared in accordance with an updated Revision 1 (October 1972) Format.

Requirements for additional information which have been issued either as Regulatory Guides, Series 1.70.x, or as Revision 2 (September, 1975) format, have been met subject to informational availability.

1.71 WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY (REVISION 0; 12/73)

The recommendations of Regulatory Guide 1.71 for limited accessibility qualification or requalification, in addition to ASME Code Sections 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 are repetitive due to multiple production of similar components, and such welding is closely supervised.

Welding performed by Daniel (DDC) at the site is in accordance with the following:

Regulatory Position	Compliance
1	Determination of restricted access definitions done on case by case basis. Initial determination of obstructions within 12 to 14 inches for the purpose of potential access restriction identification is made and documented by a responsible QC inspector. The actual weld accessibility determination is made and documented by the Daniel welding engineer for each joint identified above. Such documentation includes the engineer's basis for each decision. When the weld is determined to actually qualify as a restricted access weld, then a qualified restricted access welder is used.
2.a	See above.
2.b	Procedures require Section IX compliance.
3	Production welding is monitored as necessary to assure requirements are satisfied.

1.72 SPRAY POND PLASTIC PIPING (REVISION 0; 12/73)

Regulatory Guide 1.72 does not apply to the Virgil C. Summer Nuclear Station since the design does not incorporate plastic piping.

1.73 QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS (REVISION 0; 1/74)

The Virgil C. Summer Nuclear Station complies with the recommendations Regulatory Guide 1.73 with the following clarifications:

For safety-related motor operated valves located inside containment, the recommendations of Regulatory Guide 1.73 with the exception that stem mounted limit switches are tested separately to the provisions of IEEE STD. 382-1972 are followed.

1.74 QUALITY ASSURANCE TERMS AND DEFINITIONS (REVISION 0; 2/74)

Delete commitment. Regulatory Guide 1.74 has been withdrawn. With the adoption of NQA-1-1994 Part I "Introduction" defining terms and definitions, ANSI N45.2.10-1973 is no longer needed. RN 11-040

1.75 PHYSICAL INDEPENDENCE OF ELECTRICAL SYSTEMS (REVISION 1; 1/75)

The Virgil C. Summer Nuclear Station design complies with the recommendations of Regulatory Guide 1.75 with the following clarifications:

1. The basis for physical separation is described in Section 8.3.1.4. Since issuance of the guide followed significant design completion of affected areas, Virgil C. Summer Nuclear Station will comply with Section 4.6.2 of IEEE 384-1974 by analysis as in alternative (1).

    a. A case-by-case analysis is described in Appendix 8C to demonstrate that circuits in a non-safety related tray that runs adjacent to a safety related tray will not degrade the separation levels between the circuits in the safety related tray and any redundant safety related circuits. Cases in which the adequacy of separation between a non-safety related tray and safety related tray(s) of a single channel cannot be demonstrated, will be separated by barriers as described in Appendix 8B.

    b. A separate analysis has been performed to identify cases of multiple violations where non-safety related cable trays do not meet the separation distance requirements between safety related cable trays of redundant channels in the same fire area. These cases were resolved by providing tray covers on some cable trays or assigning the circuit breaker protective devices of some cables to a controlled breaker surveillance program. Refer to FSAR Section 8.3.1.4.1, item 4 for details.

2. Regulatory Guide Position C.1, "Interruption devices actuated by fault current are not considered to be isolation devices." This policy is contrary to basic functional design bases.

    When non-safety related loads are fed from safety related buses, the following criteria apply to the Virgil C. Summer Nuclear Station design:

    a. 7200 volt and 480 volt switchgear - Non-Safety loads are automatically tripped upon receipt of an SI or undervoltage signal. The only exception to this is the supplies to the Technical Support Center which are provided with Class 1E circuit breakers and current limiting fuses for isolation through redundant diverse devices.

    b. Motor Control Centers - Isolation for Non-Safety loads is accomplished with the use of two diverse, Class 1E overcurrent protective devices in series or by load shedding where the contactor is controlled by an SI or undervoltage signal. Class 1E overcurrent devices include I-limiter thermal magnetic breakers, current limiting fuses, and magnetic breakers combined with starter thermal overloads.

c. Vital Distribution Panels and Class 1E DC System Distribution Panels - Isolation is accomplished with the use of a thermal magnetic breaker in series with a current limiting fuse. Both of these devices have been qualified for Class 1E application. This also provides diverse redundant isolation capability.

The diesel generator, batteries, and inverters have been sized to include these loads so there would be no concern for diesel generator, batteries, or inverter degradation. This design was incorporated to prevent the automatic loss of non-safety systems such as: Radiation Monitoring, Security, Heat Tracing, Annunciator Panels, Leak Detection Annunciation, Rod Position Indication, Control Room Lighting, and the power for the Technical Support Center.

Current limiting devices were included in the design to accommodate the Regulatory Guide's concern with the use of fault current actuated isolation devices.

Details of the electrical system physical independence and identification are discussed in Sections 8.3.1.4, 8.3.1.5 and in GAI - "Construction Guideline for Electrical Circuit Physical Separation" (Drawing S-200-926). 00-01

1.76 <u>DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (REVISION 0; 4/74)</u>

The Virgil C. Summer Nuclear Station meets the recommendations of Regulatory Guide 1.76 as discussed in Section 3.3.2.

1.77 <u>ASSUMPTION USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS (REVISION 0; 5/74)</u>

Virgil C. Summer Nuclear Station meets the recommendations of this guide as discussed in WCAP-7588, Revision 1, "An Evaluation of the Rod Ejection Accident in Westinghouse PWRs Using Spatial Kinetics Methods," which received Regulatory Staff approval in August 1973 and Section 15.4. Doses for the Control Rod Ejection Accident are evaluated in accordance with Regulatory Guide 1.183. RN 12-034

1.78 <u>ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (REVISION 0; 6/74)</u>

Consideration has been given to possible accidents involving hazardous chemicals in the vicinity of Virgil C. Summer Nuclear Station control room. These accidents are discussed in Sections 2.2 and 6.4. The threat to control room operators from hazardous chemicals is minimal; however, Virgil C. Summer Nuclear Station provides self-contained breathing apparatus of at least one-half hour capacity for each control room operator.

1.79 PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS (REVISION 1; 9/75)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.79 with the following exceptions and clarifications.

Integrated system verification tests are accomplished by simulating a safety injection signal concurrent with a loss of offsite power in one safety train. However, valves which if operated would have a detrimental effect on the subsequent commissioning of the plant are blocked from operation. Similarly, where full flow conditions cannot be achieved, pumps may be operated on minimum flow or bypass conditions.

1.80 PRE-OPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS (REVISION 0; 6/74)

Based on the fact that the failure mode of pneumatic devices from loss of air has been systematically designed to be in the preferred (safe) position, the instrument air system is not classified as being safety-related, therefore, Regulatory Guide 1.80 does not apply. Testing is performed to verify that operation of the instrument air system is consistent with instrument air system requirements described in Section 9.3.1.

1.81 SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS (REVISION 1; 1/75)

This Regulatory Guide is not applicable to the Virgil C. Summer Nuclear Station since the Virgil C. Summer Nuclear Station is not a multi-unit facility.

1.82 SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS (REVISION 0; 6/74)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.82 with the following clarifications.

The design of redundant Reactor Building recirculation sumps is discussed in Section 6.2.2.

As shown by Figure 3.8-17, the design of the recirculation sumps includes specific features for minimizing the potential for clogging of the screens. These design features include the following:

1. An outer trash rack to protect the fine screens from large pieces of debris.

2. The vertical orientation of the fine screens.

3. The establishment of low velocity settling areas for each of the four individual deep sumps.

The first low velocity settling area is established at the entrance to the 1/2 inch mesh screens by directing the downward flow path around each 6 foot by 6 foot standpipe to horizontal flow through the 1/2 inch screens. Based upon the available flow area around each 6 foot by 6 foot standpipe and the RHR pump design flow rate of 3750 gpm, the calculated fluid velocity at the entrance to the 1/2 inch mesh screens is 0.156 ft/sec.

The second low velocity settling area is provided between the 1/2 inch mesh screens and the 1/4 inch mesh screens. The location of the vertically mounted 1/4 inch mesh screens results in an upward flow path between the 1/2 inch and 1/4 inch mesh screens with the two required changes in flow direction to obtain horizontal flow through both sets of screens. Based upon the flow area between the screens and the RHR pump design flow rate (3750 gpm), the calculated fluid velocity in this second low velocity settling area is 0.42 ft/sec.

The total amount of fine screen in each recirculation sump standpipe provides enough total free area to ensure that the resultant pressure drop has no appreciable effect on the net positive suction head available to each RHR pump. For the postulated condition of partially clogged screens, the total pressure drop was conservatively calculated using only half of the free area of the screens. This calculated pressure drop for flow through both sets of screens is 0.2 feet for the RHR pump design flow rate of 3750 gpm. This pressure drop represents a reduction of only 0.77% in the net positive suction head available to the RHR pumps.

Additional features incorporated into the plant design to minimize clogging of the sumps are as follows:

1. Use of insulation described under Regulatory Guide 1.36 for equipment and piping inside the Reactor Building.

2. Use of coating systems as described under Regulatory Guide 1.54 for carbon steel and concrete surfaces.

1.83 INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES (REVISION 1; 7/75)

Originally, Virgil C. Summer Nuclear Station complied with the recommendations of Regulatory Guide 1.83 with exceptions for its in-service inspections of the steam generators. However, the NRC now endorses a more risk informed and performance-based approach to regulatory compliance and has withdrawn Regulatory Guide 1.83. VCSNS has adopted this approach by implementing the requirements of Technical Specifications 3/4.4.5, “Steam Generator Tube Integrity,” and 6.8.4.k, “Steam Generator Program.” These technical specifications are based on Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” and TSTF-449.

RN 10-016

1.84 DESIGN, FABRICATION, AND MATERIALS, CODE CASE ACCEPTABILITY, ASME SECTION III (LATEST REVISON PER 10 CFR 50.55a) RN 04-011 18-054

Regulatory Guides 1.84 and 1.85 (first effective date: July 1, 1974) were issued long after the issuance of the Construction Permit for the Virgil C. Summer Nuclear Station. Only ASME approved Code Cases are used for Code Class 1, 2, and 3 components. For Class 1 components, a discussion of Code Cases is contained in Section 5.2.1.4. AEC (NRC) approval of Code Cases for Class 2 and 3 components was not required and not obtained. After the regulatory guides' first effective date, Code Cases were reviewed against those specified in these regulatory guides. RN 16-003

The NRC staff reviews ASME BPV Section III Code Cases, rules upon the acceptability of each Code Case and publishes its findings in regulatory guides. The regulatory guides are revised periodically as new Code Cases are published by the ASME. The NRC incorporates by reference the regulatory guides listing acceptable and conditionally acceptable ASME Code Cases in 10 CFR 50.55a. The latest edition of 10 CFR 50.55a is available on the NRC's Public Web site. Licensees may use these Code Cases without requesting authorization from the NRC, provided that they are used with any identified limitation or condition. RN 18-054

1.85 CODE CASE ACCEPTABILITY ASME SECTION III MATERIALS RN 18-054

(Withdrawn 06/2004) Materials Code Case Acceptability - ASME Section III, Division 1 (Guidance incorporated into Revision 32 of Regulatory Guide 1.84, published 06/03).

1.86 TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS (REVISION 0; 6/74) RN 04-011

The termination of the operating license and subsequent decommissioning of the Virgil C. Summer Nuclear Station will be in accordance with regulations in effect at that time.

1.87 CONSTRUCTION OF CLASS 1 COMPONENTS IN ELEVATED TEMPERATURE REACTORS (SUPPLEMENT TO ASME SECTION III CODE CASES 1592, 1593, 1594, 1595, 1596) (REVISION 1; 6/75)

Regulatory Guide 1.87 does not apply to the Virgil C. Summer Nuclear Station.

1.88 COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS (REVISION 2; 10/76)

Delete commitment. Regulatory Guide 1.88 has been withdrawn. The adoption of NQA-1-1994 Basic Requirement 17, Supplementary Requirement 17S-1, and commitment to Regulatory Guide 1.28, Revision 3, Regulatory Position C.2, contain the requirements for quality assurance records and replaces ANSI N45.2.9-1974 and Regulatory Guide 1.88. RN 11-040

1.89 QUALIFICATION OF CLASS 1E EQUIPMENT FOR NUCLEAR POWER PLANTS (REVISION 0; 11/74)

Engineering design of the Virgil C. Summer Nuclear Station (VCSNS) was commenced and construction permit was granted prior to the issuance of Regulatory Guide 1.89. Therefore, the guidance of IEEE-323-1971 was used as the principal document in formulating the original environmental qualification programs for Class 1E equipment used in the Virgil C. Summer Nuclear Station.

NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" was issued in November of 1979 to provide the NRC staff positions regarding selected areas of environmental qualification of safety-related electrical equipment for plants committed to satisfy the requirements set forth in either the 1971 or 1974 version of the IEEE 323 standard.

Environmental qualification of safety-related electrical equipment purchased under the NSSS scope was based on the requirements of IEEE 323-1971 and the supplemental qualification program described in NS-CE-692 (7/10/75) from C. Eicheldinger to D. B. Vassallo of the Nuclear Regulatory Commission.

Environmental qualification of safety-related electrical equipment purchased under the BOP scope was based on the requirements of IEEE 323-1971 or IEEE 323-1974 and ancillary daughter standards (e.g., IEEE Stds. 317, 334, 382, 383) in accordance with the guidance provided by NUREG-0588. Environmental qualification of BOP electrical equipment was performed in accordance with NUREG-0588, Cat. II, which relates to IEEE 323-1971 for the original plant design. However, some equipment was qualified to IEEE 323-1974 (NUREG 0588, Cat. I) requirements. New and replacement electrical equipment is governed by the current regulations of 10 CFR 50.49 and the methodology described in Regulatory Guide 1.89 (Revision 1; 6/84).

In cases where qualification has been done to later revisions of the IEEE Standards, documentation of that qualification is maintained. The status of electrical qualification in accordance with the applicable IEEE Standards and Regulatory requirements is documented in VCSNS Equipment Qualification Documentation Packages (EQDP's) for equipment requiring harsh qualification and Equipment Qualifications Files (EQF's) for equipment requiring mild qualification. Refer to Section 3.11 for further discussion. However, 10 CFR 50.49 does not require safety-related electrical equipment located in mild environments to be qualified by test and/or analysis. Thus, electrical equipment requiring mild environmental qualification is considered qualified if it meets the requirements of Generic Letter 82-09, Item 4, and any applicable IEEE Standards.

1.90 INSERVICE INSPECTION OF PRE-STRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS (REVISION 0; 11/74)

Regulatory Guide 1.90 is not applicable to the Virgil C. Summer Nuclear Station since the tendons are greased not grouted.

1.91 EVALUATION OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANT SITES (REVISION 0; 1/75)

While not applicable to the Virgil C. Summer Nuclear Station due to implementation date requirements, the recommendations of Regulatory Guide 1.91 are met due to distance factors and relative explosive proximity.

1.92 COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS (REVISION 1; 2/76)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.92 and is discussed in Section 3.7.2.

1.93 AVAILABILITY OF ELECTRIC POWER SOURCES (REVISION 0; 12/74)

This guide is not directly applicable to the Virgil C. Summer Nuclear Station since the regulatory guide implementation date postdates Construction Permit SER issuance date, August 1972 for the Virgil C. Summer Nuclear Station. 99-01

The available sources of power are discussed in Chapter 8 and the affect on plant operation is discussed in the Technical Specifications and the station operating procedures.

1.94 QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS (REVISION 1; 4/76)

Delete commitment. The Virgil C. Summer Nuclear Station commitment to Regulatory Guide 1.94 is no longer required with the adoption of NQA-1-1994 Subpart 2.5. NQA-1-1994 Subpart 2.5 is equivalent to the requirements of ANSI N45.2.5-1974. RN 11-040

1.95 PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE (REVISION 0; 2/75)

The Virgil C. Summer Nuclear Station complies with Regulatory Guide 1.95. Compliance with Section C.2 is currently ensured through the onsite storage of chlorine in amounts less than 150 lbs. per tank. Should per tank storage in amounts in excess of 150 lbs. become necessary, the pertinent regulatory guide recommendations will be re-evaluated. Accidental chlorine release is discussed further in Sections 2.2 and 6.4.

1.96 DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTORS NUCLEAR POWER PLANTS (REVISION 1; 6/76)

Regulatory Guide 1.96 is not applicable to the Virgil C. Summer Nuclear Station since it uses a PWR.

1.97 INSTRUMENTATION FOR LIGHT-WATER COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT (REVISION 3; 12/83)

The Virgil C. Summer Nuclear Station meets the intent of Regulatory Guide 1.97, Revision 3. Specific details concerning Regulatory Guide 1.97 related instrumentation are provided in Section 7.5.

1.98 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFFGAS SYSTEM FAILURE IN A BOILING WATER REACTOR (REVISION 0; FOR COMMENT; 3/76)

Regulatory Guide 1.98 is not applicable to the Virgil C. Summer Nuclear Station since it uses a PWR.

1.99 RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS (REVISION 2; 5/88)

02-01

Regulatory Guide 1.99 is used to predict the effect of neutron radiation on reactor vessel materials to support implementation of Appendix G of 10 CFR Part 50.

1.100 SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS (REVISION 0; FOR COMMENT; 3/76)

Regulatory Guide 1.100 is not applicable to Virgil C. Summer Nuclear Station due to the regulatory guide implementation date.

Although Virgil C. Summer Nuclear Station is not committed to the requirements of IEEE 344-1975 for seismic equipment qualification (SEQ), it was recognized by the Nuclear Regulatory Commission (Reference SER Supplement 4, paragraph 3.10; August 1982) that the SEQ program met the requirements and recommendations of IEEE 344-1975 and the regulatory positions of Regulatory Guide 1.100. As such, Virgil C. Summer Nuclear Station has maintained a SEQ program which is in general conformance with IEEE 344-1975 and Regulatory Guide 1.100. 00-01

Seismic design of Category I instrumentation and electrical equipment is discussed in Section 3.10.

Revision 3-2009 of this Regulatory Guide is utilized by Westinghouse in the design and fabrication of the Replacement Reactor Vessel Closure Head as noted in the Design Specification. RN 16-003

1.101 EMERGENCY PLANNING FOR NUCLEAR POWER PLANTS (REVISION 0; FOR COMMENT; 11/75) 02-01

Regulatory Guide 1.101 was used as a guide in developing the contents of the Virgil C. Summer Nuclear Station Emergency Plan. For further information, see Section 13.3 and the "Virgil C. Summer Nuclear Station Emergency Plan."

By NRC letter from Mr. R. B Minogue, Director, Office of Standards Development, on September 24, 1980 this regulatory guide was withdrawn.

1.102 FLOOD PROTECTION FOR NUCLEAR POWER PLANTS (REVISION 0; FOR COMMENT; 10/75)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.102 as described in Sections 2.4, 3.4, and Technical Specifications. RN 99-136

1.103 POST-TENSIONED PRE-STRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS (REVISION 0; FOR COMMENT; 11/75)

Regulatory Guide 1.103 has been withdrawn per NRC letter of July 8, 1981. However, the Virgil C. Summer Nuclear Station still complies with the recommendations of Regulatory Guide 1.103 as issued for comments and is discussed in Section 3.8.1.

1.104 OVERHEAD CRANE HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS (REVISION 0; FOR COMMENT; 2/76)

Regulatory Guide 1.104 was issued by the NRC For Comment in February 1976, and was Withdrawn by the NRC on July 27, 1981 and replaced by NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants”. RN 11-041

1.105 INSTRUMENT SPANS AND SETPOINTS (REVISION 0; FOR COMMENT; 11/75)

The Virgil C. Summer Nuclear Station received its construction permit before the July 1, 1976 implementation date for Regulatory Guide 1.105. For this reason, the guide is not applicable to the Virgil C. Summer Nuclear Station.

South Carolina Electric and Gas Company concurs with the following comments made by Westinghouse on the "For Comment" issue of Regulatory Guide 1.105.

Detailed information on instrument spans and setpoints are included in the Technical Specifications. RN 99-136

Regulatory Guide 1.105, being circulated by the NRC for comments, is unduly restrictive for the allowable range of setpoint settings of the instrument span. Westinghouse will take exception to this guide in several areas as discussed below.

1. Item B, fourth paragraph: Westinghouse proposes to change "expected vibration and minor calibration variations" to "and environmental conditions."

2. Item C.1 is not consistent with current Westinghouse practice in determining setpoints. Currently, the inaccuracy of the instrument and the calibration uncertainty in measuring a parameter exclusive of allowances for instrument and setpoint drift are accounted for by the difference between the Technical Specification setpoint limit and the value of a parameter at which protective action is assumed to be initiated in the accident analyses. The difference between the setpoint and the Technical Specification setpoint limit is used to account for instrument and setpoint drift. Refer to Comment 8.

3. Item C.2 is unduly restrictive for the allowable range of setpoint settings which could only result in an unnecessary increase of the instrument range and corresponding decrease of the measurement accuracy, without benefit to safety. Item C.2 should be reworded as below.

   "The designer shall verify that the accuracy attainable at the chosen setpoint settings is adequate to meet the requirements of the safety analysis. In general, the setpoints should fall between 5% and 95% of the calibrated span of the instrument except for flow measurements based on differential pressure which should be between 25% and 95% of the calibrated span."

4. Item C.3: The range of the instrument should be based on the span required for safety, i.e., Item C.3 should not be interpreted to rule out narrow range transmitter. Nor should Item C.3 be interpreted as implying the instrument must work after the accident if it is not needed for a safety-related function.

5. Item C.4: Westinghouse recommends to delete the sentence beginning, "The instruments should not anneal...." Anything of this nature should be covered by the next sentence which references qualification programs.

6. Item C.5: Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint locking device should be required.

7. Item C.6: Westinghouse recommends to make changes as below.

   The assumptions used in selecting the setpoint values in Regulatory Position C.1, and the minimum margin with respect to the limiting safety system settings, calibration uncertainty, and instrument channel drift should be documented.

8. In the present NRC Standard Technical Specifications, the nominal instrument trip setpoints and allowable trip settings are listed in the tables. The allowable trip settings are provided for instrument and setpoint drift. The NRC should revise Section B, the fourth paragraph.

An additional comment on Item C.2 is: For setpoints that function only to alarm instrumentation or instrument power failure, the minimum setpoint of 5% or 25% or the maximum setpoint of 95% should not apply.

1.106 <u>THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR OPERATED VALVES (REVISION 0; FOR COMMENT; 11/75)</u>

The Virgil C. Summer Nuclear Station design complies with the recommendations of Regulatory Guide 1.106 in that motor operated valves operated by a safety injection signal in the event of a LOCA have their respective thermal overload protection devices bypassed by the same safety injection signal contact that initiates the valve operation. Valves that are not operated by a safety injection signal do not have this feature.

1.107 <u>QUALIFICATION FOR CEMENT GROUTING FOR PRE-STRESSING TENDONS IN CONTAINMENT STRUCTURES (REVISION 0; FOR COMMENT; 11/75)</u>

This regulatory guide does not apply to the Virgil C. Summer Nuclear Station since the tendon system is not cement grouted.

1.108 <u>PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS (REVISION 1; 8/77, INCLUDING ERRATA SHEET DATED 9/77)</u> 00-01

In response to question 423.36 the Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.108 positions C.2.a and C.2.b with the following clarification:

Item C.2(a)7 - The design of the Virgil C. Summer Nuclear Station Diesel Generator Fuel Oil System is such that each diesel generator has its own fuel oil system, each of which must meet the seven day storage requirement. Therefore, this item is not applicable.

Additionally, as of the effective date of Technical Specification Amendment 139 (March 30, 1998), the Virgil C. Summer Station takes exception the following provisions of Regulatory Guide 1.108:

* Reliability demonstration per Regulatory Position C.2.a.(9) - Reliability of the V. C. Summer Emergency Diesel Generators is monitored and ensured by the provisions of 10 CFR 50.65 (Maintenance Rule) and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

* Accelerated testing provisions of Regulatory Position C.2.d. - Accelerated testing, as a means of demonstrating reliability was removed from the Technical Specifications in accordance with Generic Letter 94-01. Monitoring the effectiveness of Emergency Diesel Generator maintenance and demonstrating reliability is governed by the provisions of 10 CFR 50.65 and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." RN 99-066

* Reporting requirements of Regulatory Position C.3.b - The 30 day Special Report requirement was removed from the Technical Specifications in accordance with Generic Letter 94-01. Reporting EDG performance problems is governed by the provisions of 10 CFR 50.72, "Immediate Reporting Requirements of Operating Nuclear Power Reactors" and 10 CFR 50.73, "Licensee Event Report System."

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 (REVISION 0; FOR COMMENT; 3/76)

The Virgil C. Summer Nuclear Station complies with the methodology and calculational procedures set forth in Regulatory Guide 1.109 as discussed in Chapter 11.

Pursuant to the September 4, 1975 annex to Appendix I of 10 CFR 50, no cost benefit analysis is required or warranted.

1.110 COSTS-BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS (REVISION 0; FOR COMMENT; 3/76)

Section D of the "For Comment" version of Regulatory Guide 1.110 provides for implementation of the guide in the evaluation of construction permit applications docketed after June 4, 1976, and thus this guidance is not applicable to the Virgil C. Summer Nuclear Station Operating License Application.

At the time of issuance of the Virgil C. Summer Nuclear Station Construction Permit, NRC evaluations of "as low as practicable" used the then proposed "staff" Appendix I and as indicated in the NRC's FES; therefore, requirements of the present Appendix I as amended are met.

Further detailed descriptions of equipment for the control of gaseous and liquid effluent are included in Chapter 11. Procedures for control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems are included in the Virgil C. Summer Nuclear Station Operating Procedures.

1.111 METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASES FROM LIGHT WATER COOLED REACTORS (REVISION 0; FOR COMMENT; 3/76)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.111 as indicated in Section 2.3.

1.112 CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT WATER COOLED POWER REACTORS (REVISION 0; FOR COMMENT; 4/76)

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.112 as discussed in Chapter 11.

1.113 ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I (REVISION 0; FOR COMMENT; 5/76)

The implementation date for Regulatory Guide 1.113; "For Comment" version, is after the issuance date of the Virgil C. Summer Nuclear Station Construction Permit and is not applicable for this reason. However, aquatic dispersion of effluents is addressed in Section 11.2.

1.114 GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT (REVISION 0; 2/76)

The Virgil C. Summer Nuclear Station operators will follow the guidance presented in Regulatory Guide 1.114.

1.115 PROTECTION AGAINST LOW TRAJECTORY TURBINE MISSILES (REVISION 0; FOR COMMENT; 3/76)

Current and historical turbine missile analyses are documented within VCSNS Technical Report TR03880-002. This report demonstrates that the hazard presented to an essential system or target from low trajectory turbine missiles is less than 10^{-8}/year, and that based on the single piece monoblock rotor there is no longer any wheel disc integrity concern. RN 11-015

1.116 QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS (REVISION 0-R; 6/76)

Delete commitment. The Virgil C. Summer Nuclear Station commitment to Regulatory Guide 1.116 is no longer required with the adoption of NQA-1-1994 Subpart 2.8. NQA-1-1994 Subpart 2.8 is equivalent to the requirements of ANSI N45.2.8-1975. RN 11-040

1.117 TORNADO DESIGN CLASSIFICATION (REVISION 0; FOR COMMENT; 6/76)

This guide is not applicable due to its implementation date. However, structures, systems, and components are constructed and/or protected against the design basis tornado (Regulatory Guide 1.76) and the resulting postulated missiles to ensure that (1) the plant can be safely shutdown and maintained in a safe condition (2) doses from the postulated related failures will be within acceptable limits. Tornado missile protection is presently being included for ESF equipment. For other equipment, where it is impractical due to present building design and layout, probability analysis is utilized to indicate conservatism. The Reactor Building has been designed to withstand the design basis tornado and the control room vents are provided with a tornado missile shield.

Design features utilized include redundancy, separation, barriers, or probability considerations (see Regulatory Guides 1.76 and 1.12).

1.118 PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS (REVISION 0; 6/76)

Section 7.1.2.11 outlines the provisions for dealing with periodic testing of electric power and protection systems in conformance with IEEE 338-1971. Chapter 14.0 outlines preoperational testing. Regulatory Guide 1.118 supplements IEEE 338-1975. Therefore, Regulatory Guide 1.118 is considered not applicable to the Virgil C. Summer Nuclear Station.

1.121 BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES (REVISION 0; 8/76)

Section XI, Subsection IWB-3521.1 of the ASME Code requires steam generator tube plugging or repair when a defect has penetrated 40% through the tube wall for tubes with a r/t ratio of > 8.70, and does not address specific degradation modes.
RG 1.121(draft) has been used as the basis for establishing tubing operability limits for specific degradation modes. While the Alloy 690 tubing in the Delta-75 steam generators are not expected to experience service related degradation, RG 1.121 can be used to assess the operability of specific non-destructive examination indications. The U. S. NRC has claimed that RG 1.121 forms the basis for the current Standard Technical Specifications steam generator tube plugging limit, however, detailed analyses performed by Westinghouse have routinely established acceptable defect

depths in excess of 40% throughwall (with eddy current uncertainty applied). The U. S. NRC has never accepted these alternate depth based analyses, with the exception of a few case by case submittals (such as pitting degradation at Indian Point Unit 2) applied to a limited number of tubes.

1.123 <u>QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS (REVISION 1; 7/77)</u>

Delete commitment. Regulatory Guide 1.123 has been withdrawn. NQA-1-1994 Basic Requirement 7 and Supplementary Requirement 7S-1 replaces the commitment to Regulatory Guide 1.123 and ANSI N45.2.13-1976. RN 11-040

1.137 <u>FUEL OIL SYSTEMS FOR STANDBY DIESEL GENERATORS (REVISION 1; 10/79)</u>

The Virgil C. Summer Nuclear Station complies with the recommendations of Regulatory Guide 1.137 as discussed below:

<u>Regulatory Position</u>	<u>Compliance</u>
C.1.a	Responses to separate regulatory guides are found in Appendix 3A. Commission regulations are addressed individually, as required by the regulation.
C.1.b	The Diesel Generator Fuel Oil System quality assurance meets the recommendations of Regulatory Guide 1.28, as discussed in Appendix 3A.
C.1.c	The fuel oil storage capacity requirements were calculated in accordance with the ANSI N195-1976 time dependent load method except the margin for Mode 1-4 for fuel inventory is 2% vs. the 10% margin indicated in ANSI N195-1976. The time dependent loads are bounded by those identified in Table 8.3-3. 02-01
C.1.d	The diesel manufacturer's criteria for locating the day tanks is to permit gravity drainage of excess fuel oil from the injectors. The engine driven and backup motor driven fuel oil pumps are gear type positive displacement pumps with suction lift capability. The day tank location provides adequate net positive suction head to the pumps.
C.1.e	Preservice and inservice inspection of the Diesel Generator Fuel Oil System will be performed in accordance with ASME Code Section XI.

Regulatory
Position   Compliance

C.1.f   Since no provisions have been made for heating of the fuel oil system, assurance that fuel oil will be supplied and ignited under the most severe environmental conditions expected at the site will be accomplished by conformance to the "Cloud Point" specifications.

C.1.g   The buried fuel oil piping is coal tar enameled and felt wrapped in accordance with AWWA C-203. Insulating flanges are provided between buried piping and indoor piping. The buried fuel oil storage tanks are external coated with coal tar epoxy.

The buried piping and tanks are provided with cathodic protection and the necessary corrosion test facilities.

C.1.h   The fire protection for the diesel generator is discussed in Section 9.2-3 and in the Fire Protection (FP) DBD. 00-01 RN 18-019

C.2.a   Diesel generator fuel oil is procured to an appropriate version of ASTM D975 to ensure the operability of the diesel generator and at a minimum meets the requirements of ASTM D975-81, "Standard Specification For Diesel Fuel Oils." RN 08-012

C.2.b   Prior to adding oil to the supply tank, a sample is taken and analyzed for 1) specific or API gravity, 2) clear and bright appearance (water and sediment), 3) kinematic viscosity at 40°C, and 4) flash point. Samples are also taken and analyzed for the remaining properties in ASTM D975-81, with results available within 30 days.

C.2.c   Sampling is performed manually in accordance with ASTM-D4057-81.

C.2.d, .e   Provisions have been incorporated into system design to provide for water removal and pipe flushing. Operational procedures will comply with the recommendations of the regulatory guide.

C.2.f   Provisions have been incorporated into system design to provide for removal of fuel oil from the storage tanks. Operational procedures will comply with the recommendations of the regulatory guide.

Regulatory Position	Compliance
C.2.g	No special design provisions have been provided to minimize turbulence of accumulated sediment in the bottom of the storage tank during filling. However, the fuel oil storage tank filling system features a filter/water separator which will help minimize the quantity of sediment in the bottom of the tank.
C.2.h	Refer to C.1.g. Appropriate surveillance procedures will be established, if required.

1.143 DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER COOLED NUCLEAR POWER PLANTS

The radioactive waste treatment and disposal systems at Virgil C. Summer were constructed to comply with the August 1970 Draft of ANSI N18.2 which in most cases is more restrictive than the guidelines of Regulatory Guide 1.143.

The intent of Regulatory Guide 1.143 is met by our systems. Future equipment procurement and modifications will meet or exceed the quality and code requirements of the Regulatory Guide. Specific exceptions to the provisions of 1.143 are as follows:

1. Discussion Section B

   The Boron Recycle System is reclassified to Non-Nuclear Safety grade and future equipment procurement and modifications will meet or exceed the code and quality requirements of Regulatory Guide 1.143. The basis for this reclassification is that this system is similar in function and design to the Liquid Waste Processing System. Accordingly, it can be considered in the same manner as the Liquid Waste Processing System in determining the appropriate safety classification. Support for this viewpoint is provided by the NRC's approval of a non-nuclear safety grade Boron Recycle System as documented in their safety evaluation report (NUREG-0491) on RESAR-414.

2. Position C 1.1.3

   Foundations and walls of structures that house the Liquid Radwaste System are designed to the seismic criteria as described in FSAR Section 3.7.

3. Position C 1.2

   The design features to prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor tanks are in accordance with ANSI N18.2-1973 (Section 5.6.4). In most cases curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system have been utilized.

4. Position C 2.1.3

   Portions of the Gaseous Radwaste System that are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems are designed to the seismic criteria as described in FSAR Section 3.7.

5. Position C 3.1.3

   Vendor supplied portable solid radwaste treatment systems will be located inside the Auxiliary Building which is designed to the seismic criteria as described in FSAR Section 3.7. Vendor supplied mobile solid radwaste treatment systems will be located in an area designed to prevent uncontrolled releases due to accidental spillage during processing in accordance with ANSI N18.2-1973 (Section 5.6.4).

6. Position C 5

   The applicable seismic design is described in FSAR Section 3.7.

7. Position C 6, "4.2.3.1(1)"

   Procurement documents shall be independently verified for conformance to the requirements of Regulatory Guide 1.143 by the individual(s) within the quality assurance organization.

8. As an alternate to ANSI B31.1 which covers piping and valves, ASME Section VIII may be substituted for the design of pressure relief valve arrangements. This option is based on the interpretation that ASME Section VIII code boundary may be extended up to and including the stop valve and relief valve as a part of the pressure vessel design. The stop valve itself, is covered by ANSI B31.1 since ASME Section VIII does not cover its design.

9. Regulatory Guide 1.143 specifies 30 minute hold time for leak testing of piping systems in addition to the required codes and standards. An exception to the Regulatory Guide 1.143 30 minute hold time is the 10 minute specified in ASME and ANSI codes without an additional 30 minutes. The basis for this exception is, all systems installed, maintained, and modified are tested in accordance with ASME Section III, Section XI and ANSI B-31-1 Codes. This exception will eliminate any confusion regarding test time and ANSI code requirements. 98-01

1.144 AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS (REVISION 0; 1/79)

Delete commitment. Regulatory Guide 1.144 has been withdrawn. The requirements of NQA-1-1994 Basic Requirements 7 and 18 and Supplementary Requirements 7S-1 and 18S-1 are adequate alternatives to the requirements of ANSI N45.2.12-1979 and Regulatory Guide 1.144. RN 11-040

1.146 QUALIFICATION OF QUALITY ASSURANCE PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS (REVISION 0; 8/80)

Delete commitment. Regulatory Guide 1.146 has been withdrawn. The requirements of NQA-1-1994 Basic Requirement 2 and Supplementary Requirement 2S-3 are adequate alternatives to the requirements of ANSI N45.2.23-1978 and Regulatory Guide 1.146. RN 11-040

1.147 INSERVICE INSPECTION CODE CASE ACCEPTABILITY ASME SECTION XI DIVISION 1, (LATEST REVISION PER 10 CFR 50.55a) RN 03-044 RN 16-029

Regulatory Guide 1.147 identifies the Code Cases that the NRC has determined to be acceptable alternatives to the applicable parts of Section XI. Licenses may use these Code Cases without requesting authorization from the NRC, provided that they are used with any identified limitation or modifications.

1.154 FORMAT AND CONTENT OF PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS FOR PRESSURIZED WATER REACTORS (REVISION 0; 1/87)

The NRC established screening criteria after extensive industry analysis identified the likelihood of a reactor vessel failure due to Pressurized Thermal Shock (PTS) events. The rule required that any plant that wishes to operate at values above the RT_{PTS} screening criterion must provide an extensive safety analysis. Westinghouse performed a study and supplied a report to SCE&G. The report, WCAP-15103, Revision 0 (September 1998), "Evaluation of Pressurized Thermal Shock for V. C. Summer," satisfies the requirement that current and projected values of RT_{PTS} for reactor vessel beltline region materials be evaluated and reported to the NRC. RT_{PTS} values contained in the report are based on actual plate and weld material chemistry data and are well below the NRC screening criteria values for pressurized thermal shock. The actual fluence levels (determined from surveillance capsule data) and projected levels for plant operation at 2900 MWt were used to calculate the RT_{PTS} values at 32 and 48 EFPY. The most limiting plate, A9154-1 was calculated to exhibition an RT_{PTS} value of 157°F at 48 EFPY. Using surveillance capsule data, the RT_{PTS} value for this plate was found to be 98°F. RN 00-001

1.155 STATION BLACKOUT (REVISION 0; 6/88) RN 00-001

The Virgil C. Summer Nuclear Station's Station Blackout Program meets the guidance provided by Regulatory Guide 1.155.

1.160 MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS (REVISION 3; 5/12) RN 12-038

The Virgil C. Summer Nuclear Station Maintenance Rule Program meets the guidance provided by Regulatory Guide 1.160.

1.183 ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (REVISION 0; 7/00) RN 12-034

The Virgil C. Summer Nuclear Station meets the guidance provided in Regulatory Guide 1.183.

1.189 FIRE PROTECTION FOR OPERATING NUCLEAR POWER PLANTS (REVISION 2; 10/09) RN 11-040

SCE&G implements quality requirements for the Fire Protection System in accordance with Regulatory Position 1.7, "Quality Assurance" with the exception of section 1.7.8, "Corrective Action." Corrective actions associated with Fire Protection will follow the guidance in QAPD Section 16, Corrective Action. RN 18-029

1.192 OPERATION AND MAINTENANCE CODE CASE ACCEPTABILITY, ASME OM CODE (LATEST REVISION PER 10 CFR 50.55a) RN 18-054 RN 03-044

The NRC staff reviews ASME OM Code Cases, rules upon the acceptability of each Code Case and publishes its findings in regulatory guides. The regulatory guides are revised periodically as new Code Cases are published by the ASME. The NRC incorporates by reference the regulatory guides listing acceptable and conditionally acceptable ASME Code Cases in 10 CFR 50.55a. The latest edition of 10 CFR 50.55a is available on the NRC's Public Web site. Licensees may use these Code Cases without requesting authorization from the NRC, provided that they are used with any identified limitation or condition. RN 18-054

1.194 ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL HABITABILITY ASSESSMENTS AT NUCLEAR PLANTS (REVISION 0; 6/03) RN 12-034

The Virgil C. Summer Nuclear Station meets the guidance provided in Regulatory Guide 1.194.