BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

NRC Docket Nos. 50-424, 50-425

In the Matter of

GEORGIA POWER COMPANY

SUPPLEMENT 6 TO

APPLICATION FOR LICENSE

UNDER THE ATOMIC ENERGY ACT OF 1954

AS AMENDED

FOR

ALVIN W. VOGTLE NUCLEAR PLANT

UNITS 1, 2

The Applicant, Georgia Power Company, hereby supplements its Application for a Construction Permit and Operating License, originally submitted on August 1, 1972, by the addition of supplementary material attached hereto.

By: C. F. Whitmer
Vice President

Sworn to and subscribed before me, this 21st day of August, 1979.

Notary Public
Notary Public, Georgia, State at Large
My Commission Expires Sept. 29, 1981

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INSTRUCTION SHEET-1
SUPPLEMENT NO. 6

POOR ORIGINAL

ALVIN W. VOGTLE NUCLEAR PLANT
PRELIMINARY
SAFETY ANALYSIS REPORT

DO NOT REMOVE EXISTING WHITE PAGES

Replace Table of Contents pages S2 i and S2 ii with pages S6 i and S6 ii; replace pages S2 xiii and S2 xiv with pages S6 xiii and S6 xiv; replace pages S2 xxiii and S2 xxiv with S6 xxiii and X6 xxiv.

Replace Table of Contents pages S2 1-i and S2 1-ii with S6 1-i and S6 1-ii.

Insert Table of Contents page S6 1-vii ahead of page 1-vii.

Insert Chapter 1, pages S6 1.1-1 and S6 1.1-2 ahead of page 1.1-1.

Insert Chapter 1, pages S6 1.2-9 through 1.2-12 ahead of page 1.2-9.

Insert Chapter 1, figures 1.2-9 and 1.2-10 ahead of figure 1.2-9.

Insert Table of Contents pages S6 3-xi, S6 3-xia and S6 3-xii ahead of page 3-xi; insert pages S6 3-xix through S6 3-xxii ahead of 3-xix.

Insert Chapter 3, pages S6 3.8-81 through S6 3.8-84 ahead of page 3.8-81; insert pages S6 3.8-91 and S6 3.8-92 ahead of page 3.8-91; insert page S6 3.8-95 and S6 3.8-96 ahead of 3.8-95 and insert page S6 3.8-123 and S6 3.8-124 ahead of page 3.8-123.

Insert figure 3.8-2 ahead of figure 3.8-2, insert figure 3.8-4 ahead of figure 3.8-4.

Insert figures 3.8-27 and 3.8-28; insert change sheet for figure 3.8-29 and figures 3.8-30 and 3.8-31 ahead of figure 3.8-27.

Insert Table of Contents pages S6 6-1 through S6 6-vi ahead of page 6-i; insert page S6 6-xi ahead of page 6-xi.

Insert Chapter 6, pages S6 6.1-1 through S6 6.1-2 a/b ahead of page 6.1-1.

Insert Chapter 6, page S6 6.2-5 ahead of page 6.2-5; insert page 6.2-51 and 6.2-6 ahead of page 6.2-51.

Insert Chapter 6, pages S6 6.6-1 and S6 6.6-6 ahead of page 6.6-1.

Insert Chapter 6, Change Sheets for figures 6.6-1 and 6.6-2 ahead of figure 6.6-1.

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ALVIN W. VOGTLE NUCLEAR PLANT
PRELIMINARY
SAFETY ANALYSIS REPORT

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Insert Table of Contents pages S6 7-i and S6 7-ii ahead of page 7-i.

Insert Chapter 7, page S6 7.3-1 and S6 7.3-2 ahead of page 7.3-1.

Insert Table of Contents pages S6 8-i and S6 8-ii ahead of page 8-i.

Insert Chapter 8, pages S6 8.1-3 and S6 8.1-4 ahead of page 8.1-3.

Insert Chapter 8, pages S6 8.3-3 through S6 8.3-5a in front of page 8.3-3.

Insert Table of Contents pages S6 9-vii and S6 9-viii ahead of page 9-vii.

Insert Chapter 9, page S6 9.5-11 ahead of page 9.5-11.

Insert Table of Contents page S6 12-i and S6 12-ii ahead of page 12-i.

Insert Chapter 12, page S6 12.2-1 and S6 12.2-2 ahead of page 12.2-1.

Insert Table of Contents pages S6 15-i through S6 15-iv ahead of Table of Contents page 15-i.

Insert Chapter 15, pages S6 15.4-1 through S6 15.4-10 ahead of page 15.4-1.

Insert Appendix 15B pages S6 15B-i through S6 15B-5 behind the divider tab for Appendix 15B and ahead of page 15B-i.

Insert Table of Contents page S6 16-iii and S6 16-iv ahead of 16-iii; insert page S6 16-xi and S6 16-xii ahead of page 16-xi.

Insert Chapter 16, page S6 16.3.3-3 and S6 16.3.3-4 ahead of page 16.3.3-3.

Insert Chapter 16, page S6 16.4.5-3 and S6 16.4.5-4 ahead of page 16.4.5-3.

Insert Chapter 16, page S6 16.5.2-1 and S6 16.5.2-2 ahead of page 16.5.2-1.

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Preliminary Safety Analysis Report (PSAR) complies with the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" issued by the Atomic Energy Commission in February 1972. For a discussion of the format of this report, refer to subsection 1.1.7, Organization of Contents.

1.1.1 LICENSE REQUESTED

This PSAR is submitted to support the application of the Georgia Power Company, herein referred to as the GPC, for a construction permit and operating license for a nuclear power plant designated as the Alvin W. Vogtle Nuclear Plant, herein referred to as the VNP.

1.1.2 PLANT UNITS

The application is for four units, each with a reactor core rated at a power level of 3411 MWt under section 103 of the Atomic Energy Act of 1954, as amended, and the regulations of the Atomic Energy Commission set forth in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50).

The plant will be constructed in 2 two-unit stations with the two units in each station essentially the same. Descriptions of one unit shall be interpreted as applying to both units in each station. Similarly, descriptions of one station shall be interpreted as applying to both stations.

Differences between the two units in a station and particularly structures, systems, and components that are shared between the two units are specified in the appropriate location in the PSAR. Similarly, differences between the two stations and any structures, systems, and components shared between the two stations, if any, are likewise specified.

1.1.3 PROPOSED PLANT LOCATION

The proposed location of the VNP is on the southwest side of the Savannah River approximately 23 river miles upstream from the intersection of the Savannah River and U.S. Highway

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No. 301, as shown on figures 1.2-1 and 1.2-2. The site is in the eastern sector of Burke County, Georgia, and across the river from Barnwell County, South Carolina. The VNP site is directly across the Savannah River from the AEC's Savannah River Plant (SRP).

1.1.4 CONTAINMENT TYPE

S6 The containment for each of the VNP units will be designed by Bechtel Corporation, consisting of a prestressed concrete, steel-lined, cylindrical structure with hemispherical dome, called the containment, which completely encloses the reactor coolant pressure boundary.

1.1.5 NUCLEAR STEAM SUPPLY SYSTEM

1.1.5.1 Reactor Type and Supplier

The nuclear steam supply system (NSSS) for each of the four VNP units is a pressurized water reactor (PWR). The Westinghouse Electric Corporation is contracted to design and supply units for the VNP.

1.1.5.2 Power Output

Each NSSS unit is rated or guaranteed at a net core power output of 3411 MWt plus 14 MWt net of heat from nonreactor sources, primarily pump heat, a total of 3425 MWt; the corresponding turbine-generator gross generator output is 1159 MWe. The design net core power output rating for each NSSS unit is 3565 MWt, the ultimate expected capability of the NSSS.

The turbine-generator unit of the steam and power conversion system will have the capability of generating a gross electrical output of 1210 MWe, the maximum calculated-not guaranteed, valves wide open condition, corresponding to a NSSS output of 3570 MWt.

Although the license application is for 3411 MWt net core per NSSS unit, all safety systems, including the containment and engineered safety features, are designed and evaluated for operation at a higher power level, 3565 MWt net core. This power rating is used in the analysis of all postulated accidents bearing significantly on the acceptability of the site. The thermal-hydraulic and nuclear aspects of the core have been evaluated on the basis of a core thermal output of 3411 MWt.

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1.2.6 EQUIPMENT BUILDING

The equipment building (EB) surrounds the containment from grade to the 270 foot level. The EB is a group of steel-framed structures with uninsulated metal siding and metal roof deck. It provides protection from the weather for equipment located within the building. Safety-related equipment located within the EB are physically protected from tornado missiles on an individual basis. A detailed description of the structure is provided in section 3.3.4.1.1. S6

1.2.7 SAFETY FEATURES

The safety features limit the potential radiation exposure to the public and to plant personnel following an accidental release of radioactive fission products from the reactor system, particularly as the result of a LOCA. These safety features function to localize, control, mitigate, and terminate such accidents, ensuring that 10 CFR 100 limits are not exceeded. The safety features consist of the following systems:

A. Emergency core cooling system

B. Containment spray system

C. Containment cooling system

D. Penetration room filtration system

E. Hydrogen recombiners and emergency purge system

1.2.7.1 Emergency Core Cooling System

The emergency core cooling system (ECCS) injects borated water into the reactor coolant system following a LOCA. This

provides cooling to limit core damage, metal-water reactions, and fission product release; and assures adequate shutdown margin regardless of temperature. The ECCS also provides continuous long term post-accident cooling of the core by recirculating borated water between the containment sump and the reactor core.

1.2.7.2 Containment Spray System

The containment spray system is one of two independent, full capacity systems with which each unit is equipped for cooling the containment atmosphere after the postulated LOCA.

The containment spray system supplies borated water to cool the containment atmosphere. The spray system in combination with two of the four containment air coolers (operating at reduced speed) is sized to provide adequate cooling with either or both of the two containment spray pumps in service. These pumps take suction from the refueling water storage tank. When the supply from the storage tank is depleted, suction of the pumps is aligned to pump water from the containment sump directly into the containment during the recirculation mode of operation. Sodium hydroxide is added to the spray to remove iodine from the containment atmosphere in the post-accident condition.

1.2.7.3 Containment Cooling System

The containment cooling system is the second of two independent, full capacity systems with which each unit is equipped to mix and cool the containment atmosphere.

1.2.7.4 Penetration Room Filtration System

The penetration room filtration system for each unit collects and processes potential containment penetration leakage to limit environmental activity levels following a LOCA and the subsequent pressure transient inside the containment.

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1.2.7.6 Hydrogen Recombiners and Emergency Purge System

Electrical hydrogen recombiners reduce the percentage of hydrogen in the post-LOCA containment atmosphere to below uncontrolled release mixture levels. Emergency purge of the containment supplements the recombiners and further reduces hydrogen concentration by feeding ambient air and bleeding containment atmosphere.

1.2.8 UNIT CONTROL

The reactor is controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. During steady-state operation, the reactor control system maintains a programmed average reactor coolant temperature that rises in proportion to the load. The combined actions of the control system, and steam bypass to the condenser maintain and strengthen steam relief station auxiliary load upon separation from the transmission system from full load.

The solid state protection logic system automatically initiates appropriate action whenever the parameters monitored by this system reach pre-established setpoints. This system acts to trip the reactor, actuate emergency core cooling, close containment isolation valves, and initiate the operation of other safety features systems.

1.2.9 PLANT ELECTRICAL POWER

The four main turbine-generators are each rated at 1350 MVA, 0.90 pf, 22,000 volts; they are 3-phase, 60-Hz, 1800-rpm hydrogen- and water-cooled units. The power from these units is delivered to the GPC 230-kV and 500-kV switchyards, which are connected to the system grid by 230-kV and 500-kV transmission lines. Termination points of these lines are shown in section 8.2.

Each unit has three separate sources of power for its auxiliaries. The three sources of power and associated electrical equipment ensure the functioning of all units without undue risk to the health and safety of the public.

The three sources for each unit are as follows:

A. The main turbine-generator supplies normal auxiliary loads during plant operation.

B. The two reserve auxiliary transformers supply the safety feature buses from the 230-kV switchyard.

C. The two, fast-starting diesel generators are connected to two safety feature buses. Upon loss of all offsite power, either diesel generator and its associated bus can shut the unit down safely under LOCA conditions.

Station batteries ensure a constant supply of power to vital instruments and controls.

Station power is distributed through redundant buses at 4160-, and 480-volt, ac, to buses and load centers. The dc load requirements for each unit are distributed through two 125-volt buses.

1.2.10 PLANT INSTRUMENTATION AND CONTROL SYSTEM

To avoid undue risk to the health and safety of the public, instrumentation and controls monitor and maintain neutron flux, primary coolant pressure, temperature, and control rod positions within prescribed operating ranges.

The non-nuclear regulating, process, and containment instrumentation measures temperatures, pressure, flow, and levels in the steam systems, containment, and auxiliary systems. Process variables required on a continuous basis for startup, operation, and shutdown of the unit are indicated, recorded, and controlled from the control room. The quality and types of process instrumentation provided ensure safe and orderly operation of all systems and processes over the full operating range of the plant.

Startup and shutdown of the reactor and adjustment of reactor power in response to turbine load demand, are provided by the reactor control system. The reactor is controlled by a combination of mechanically-driven control rods. The control system permits each unit to accept step load increases or decreases of 10 percent and ramp load increases or decreases of 5 percent per minute over the load range from 15 percent to, but not exceeding, 100 percent power under normal operating conditions, subject to xenon limitations.

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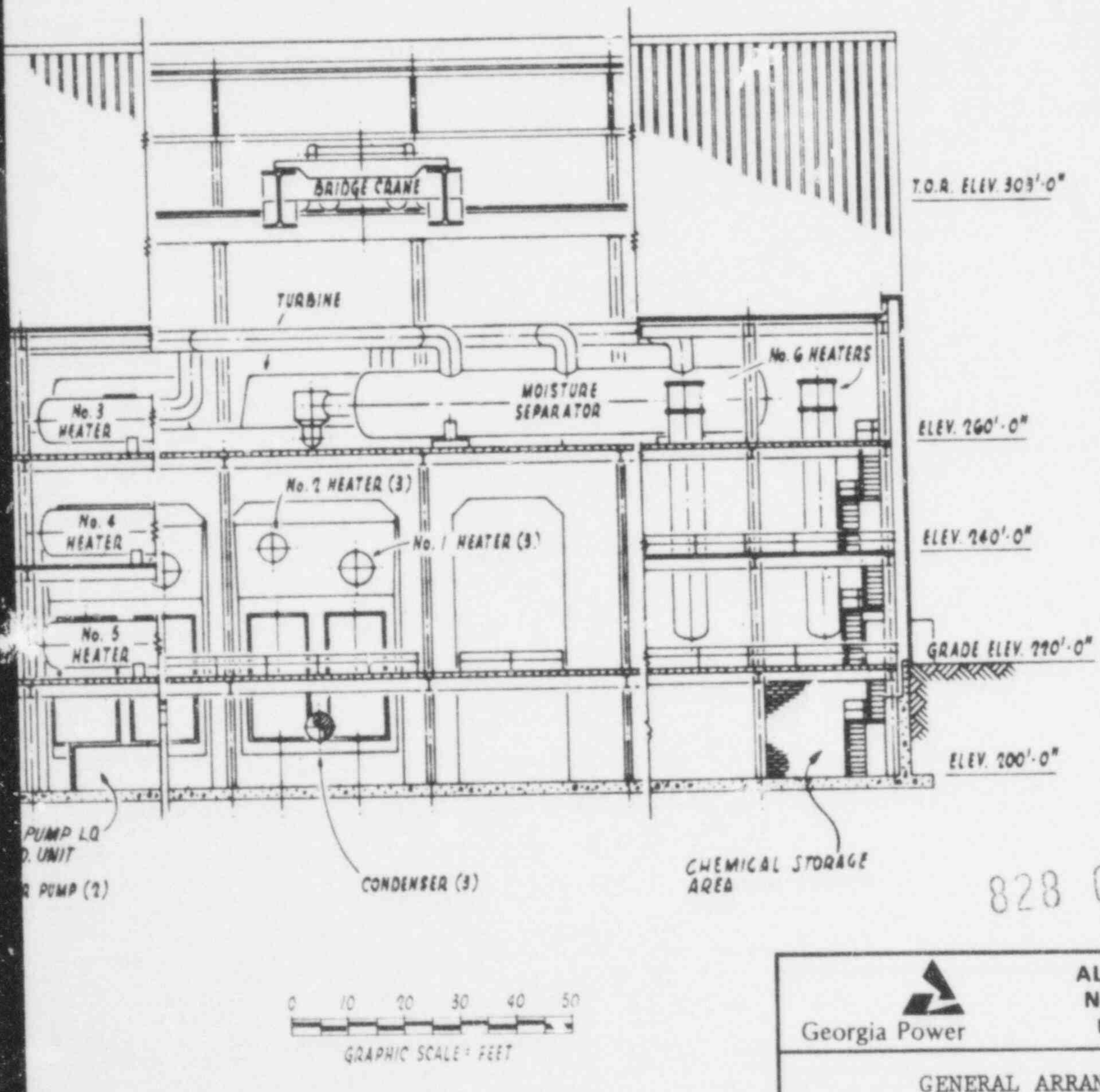

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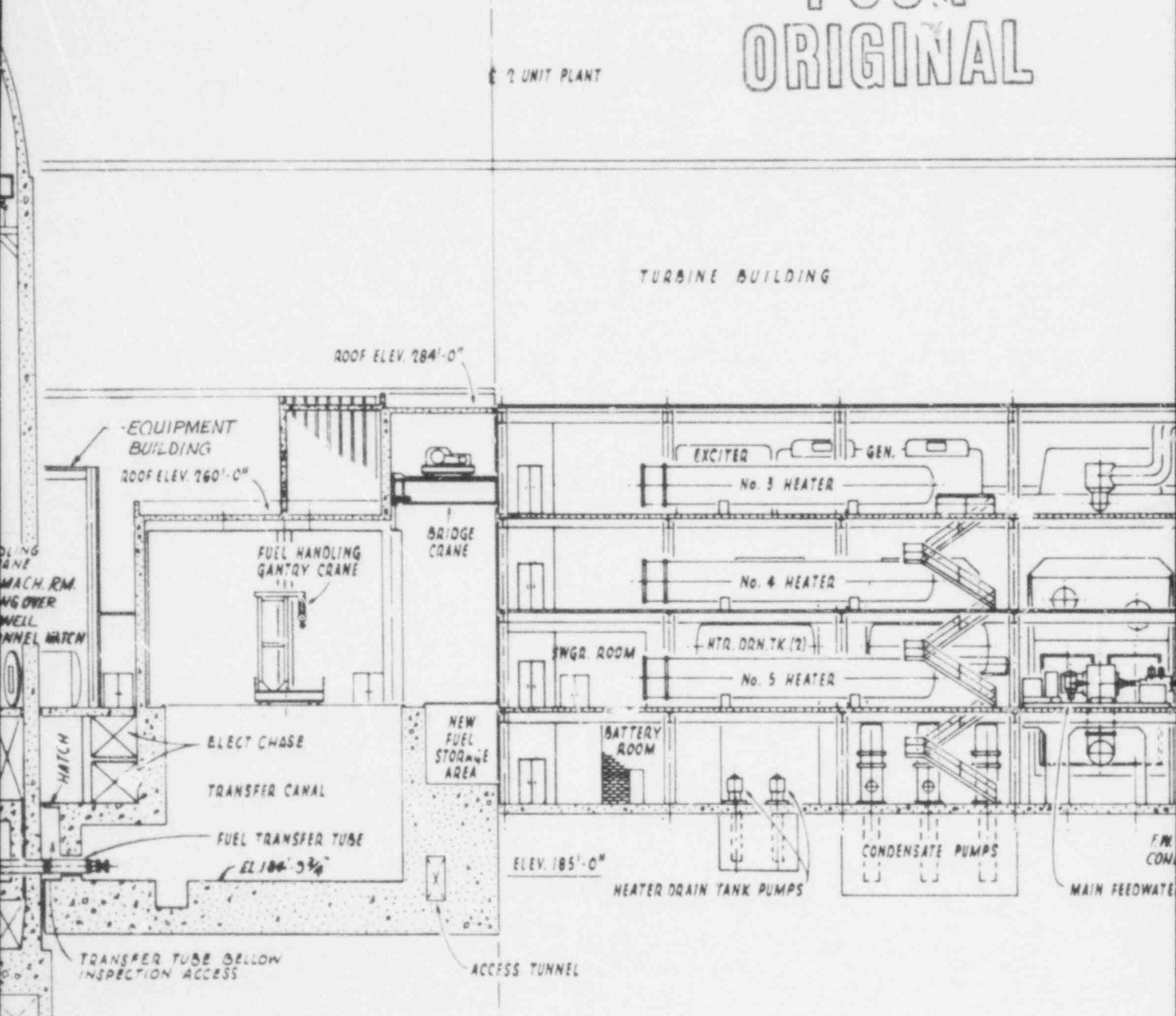
Georgia Power	ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2
GENERAL ARRANGEMENT SECT ELEVATION LOOKING NORTH	
FIGURE 1.2-9	


℄ 2 UNIT PLANT
TURBINE BUILDING
ROOF ELEV. 284'-0"
EQUIPMENT
BUILDING
ROOF ELEV. 260'-0"
FUEL HANDLING
GANTRY CRANE
BRIDGE
CRANE
EXCITER
GEN.
No. 3 HEATER
No. 4 HEATER
HTR. DRN. TK (2)
No. 5 HEATER
SWGR. ROOM
BATTERY
ROOM
NEW
FUEL
STORAGE
AREA
ELECT CHASE
TRANSFER CANAL
HATCH
FUEL TRANSFER TUBE
EL 184'-3¾"
ELEV. 185'-0"
CONDENSATE PUMPS
HEATER DRAIN TANK PUMPS
MAIN FEEDWATE
TRANSFER TUBE BELLOW
INSPECTION ACCESS
ACCESS TUNNEL
TENDON GALLERY

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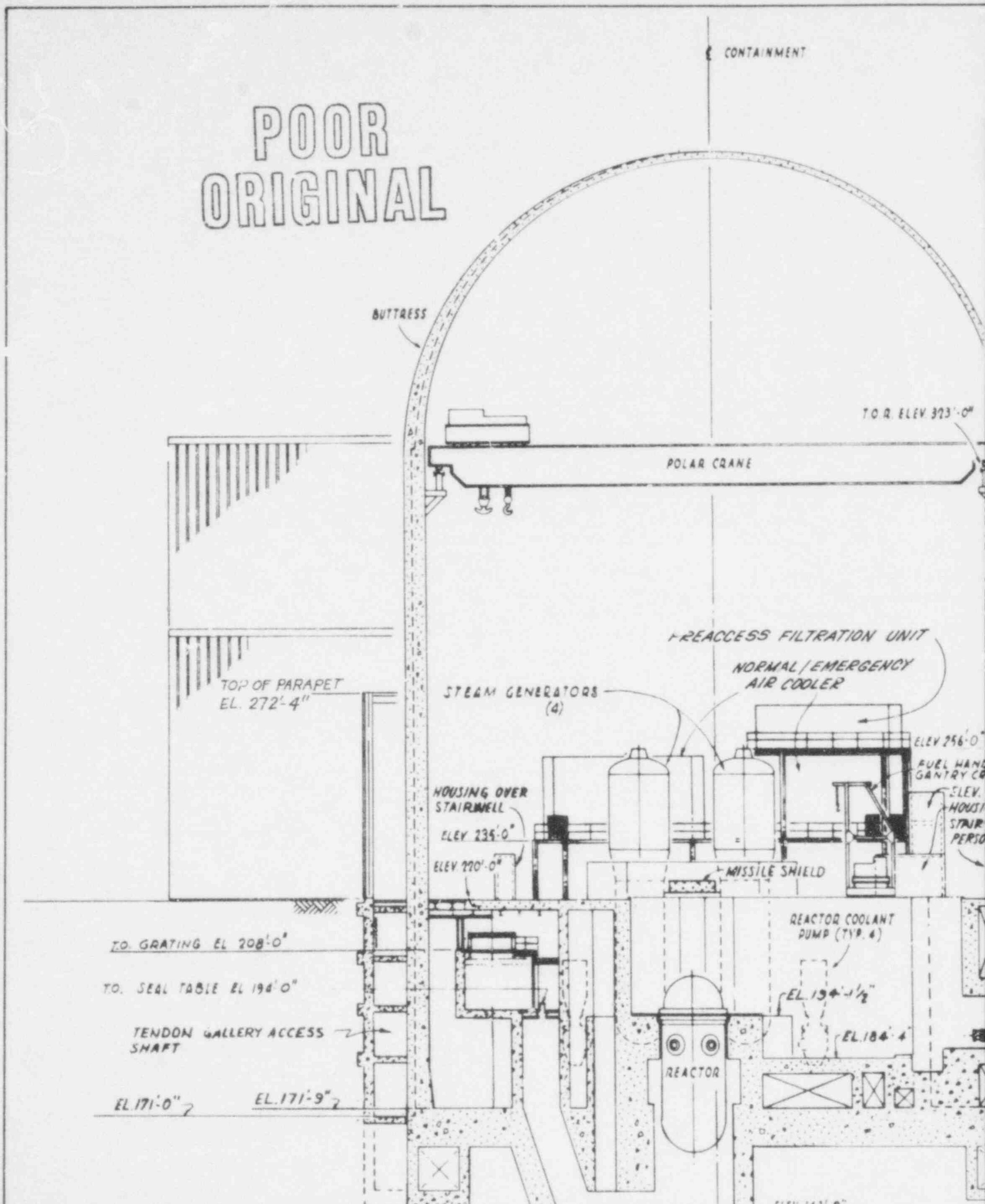

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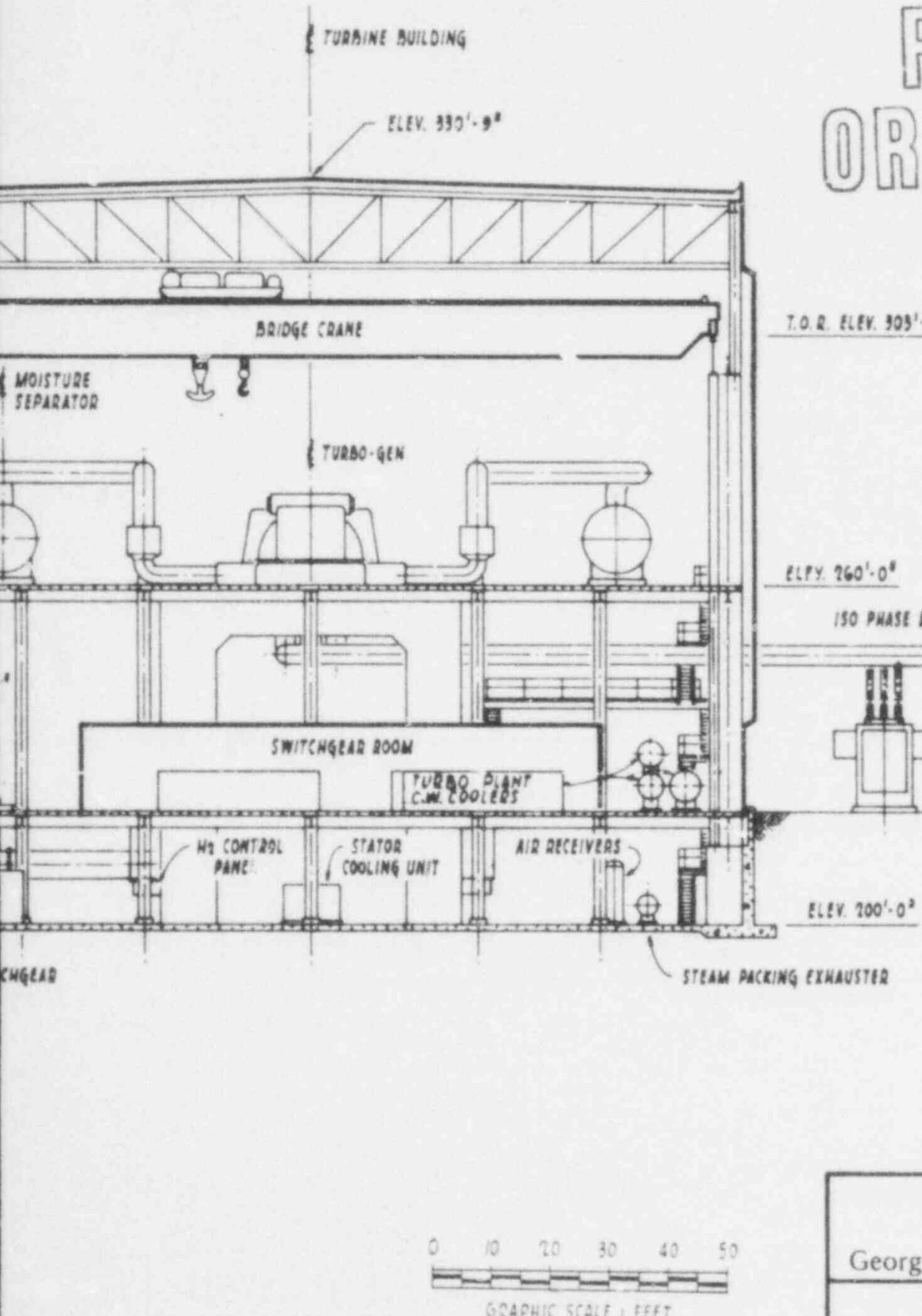

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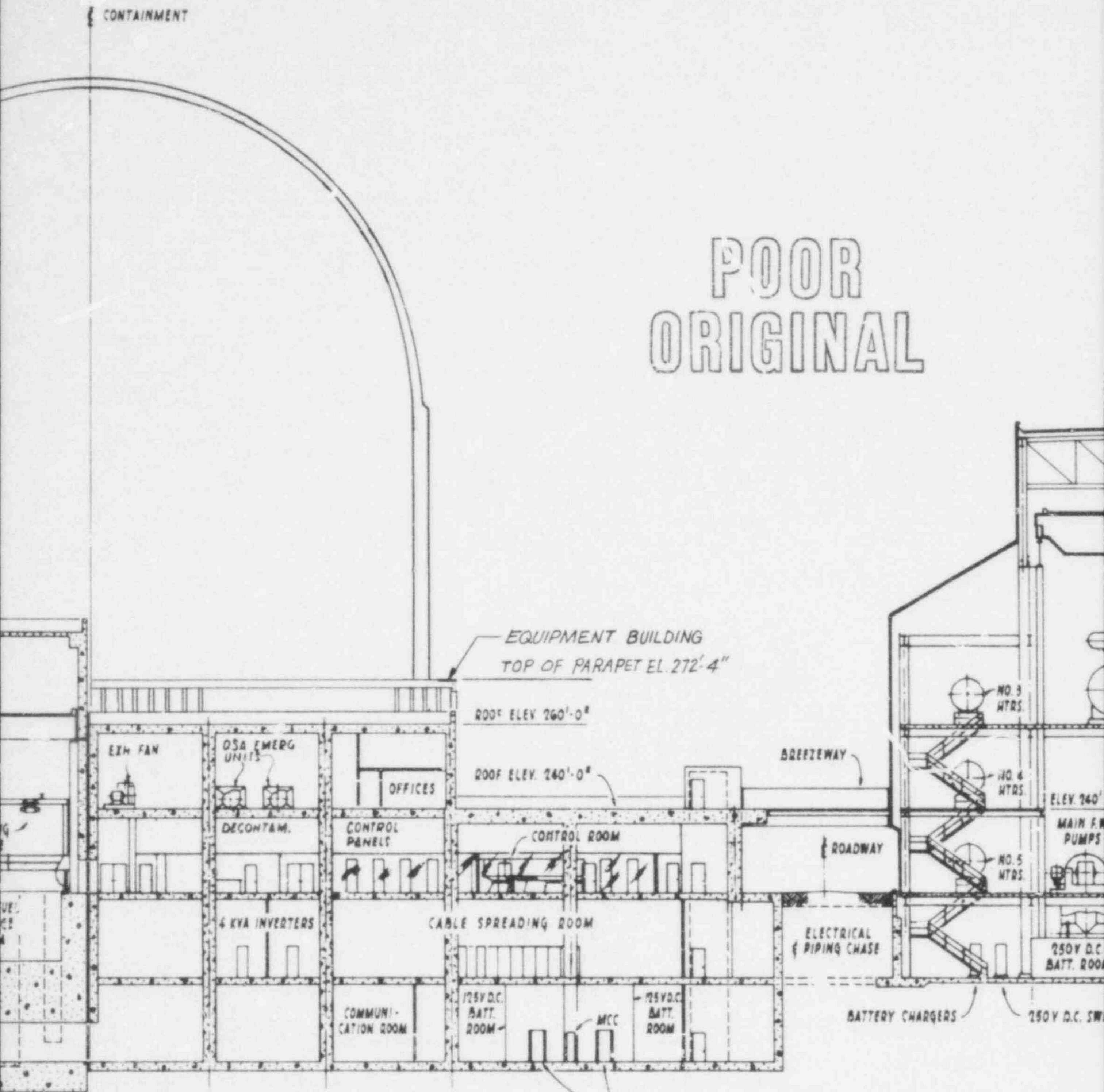
Georgia Power

ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2

GENERAL ARRANGEMENT
SECT ELEVATION LOOKING WEST

FIGURE 1.2-10


CONTAINMENT
EQUIPMENT BUILDING
TOP OF PARAPET EL. 272'-4"
ROOF ELEV. 260'-0"
ROOF ELEV. 240'-0"
EXH FAN
OSA EMERG UNITS
OFFICES
DECONTAM.
CONTROL PANELS
CONTROL ROOM
BREEZEWAY
ROADWAY
NO. 3 HTRS.
NO. 4 HTRS.
NO. 5 HTRS.
ELEV. 240'
4 KVA INVERTERS
CABLE SPREADING ROOM
ELECTRICAL & PIPING CHASE
COMMUNICATION ROOM
125V D.C. BATT. ROOM
MCC
125V D.C. BATT. ROOM
480V SWGR.
BATTERY CHARGERS

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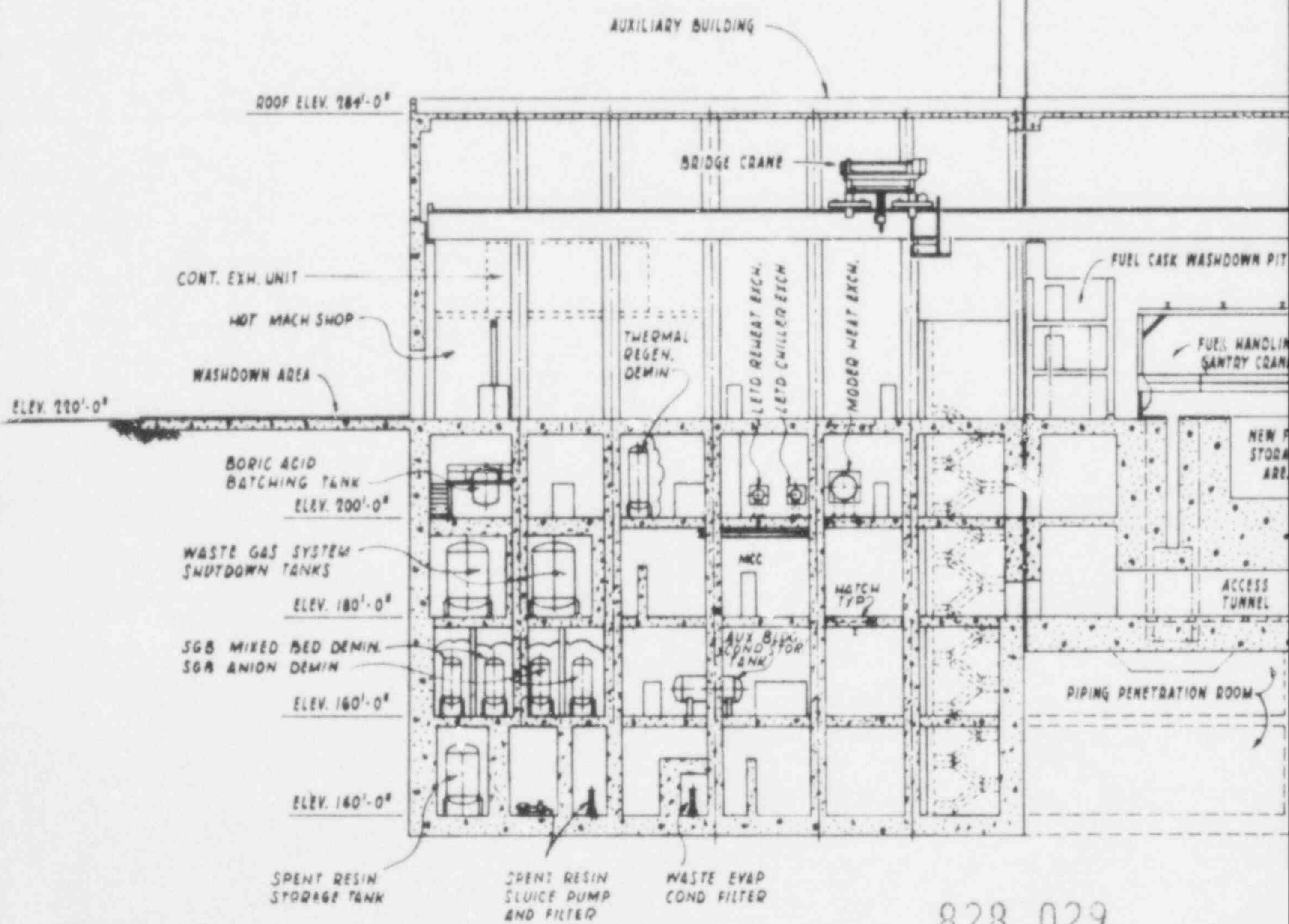

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3.8.3.6.5 Construction Procedure

The construction procedures are the same as described in paragraph 3.8.1.6.6.

3.8.3.6.6 Quality Control

The quality control requirements will be met as described in paragraph 3.8.1.6.7 and Chapter 17.

3.8.3.7 Testing and Inservice Surveillance Requirements

A formal program of testing and inservice surveillance is not planned for the internal structures. The internal structures are not directly related to the functioning of the containment concept, hence, no testing or surveillance is required.

3.8.4 OTHER CATEGORY I STRUCTURES

S6 Other Category I structures include equipment building, auxiliary building, control building, fuel handling building, diesel generator building, auxiliary feedwater pump building, nuclear service cooling tower, emergency cooling water wells, condensate storage water tank, reactor makeup storage tank, refueling water storage tank, diesel oil fuel storage tank, pipe and electrical cable tunnels and electrical duct banks.

3.8.4.1 <u>Description of the Structure</u>

S6 3.8.4.1.1 Equipment Building

The equipment building (EB) is a group of five steel-framed structures with uninsulated metal siding and roof deck. The EB completely surrounds the containment from grade to 270 foot level. There is no safety function associated with the S6 EB except that its failure shall not damage any safety-related equipment. The approximate center line to center line column dimensions of the EB are: length, 176 feet, width, 176 feet, and height, 50 feet above grade.

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3.8.4.1.2 Auxiliary Building

The auxiliary building is a multistory reinforced concrete structure located south of the fuel handling building with a building separation in between. This building has two wing structures (one east wing and one west wing) with a separation in between the wings and a combined foundation mat. It has a four-level basement extending 80 feet below grade. The highest portion of the structure is 64 feet above grade. The basement houses the radioactive waste treatment facilities, heat-exchangers pumps, and miscellaneous items. The remainder of the building contains other auxiliary nuclear equipment and associated facilities, hot machine shop, cask handling crane, heating, ventilating, and air conditioning facilities. Refer to figure 3.8-32 thru 3.8-34, inclusive, for the general layout and geometric description of the building. The cask handling crane complies with OSHA Subparagraph N - Materials Handling and Storage of 29 CFR 1910, section 1910.179.

3.8.4.1.3 Control Building

The control building is a multistory reinforced concrete structure located between the two containment structures with the turbine building at the north side and a separation between it and the fuel handling building which is located south of the control building. Two stories are above grade level and two stories are below grade. The control equipment, plant laboratories, decontamination facilities and locker room are on the grade level floor. The upper floor has office space and air conditioning equipment.

The remainder of the building contains the control room, switchgear, battery rooms, communications, computer and cable spreading rooms. Refer to figures 3.8-35 thru 3.8-37, inclusive, for the general layout and building size.

3.8.4.1.4 Fuel Handling Building

The fuel handling building is a reinforced concrete structure located at the center of the category I structures comprising the nuclear block for Units 1 and 2 of the Vogtle Nuclear Plant. The fuel building is physically separated from the surrounding structures and has an independent foundation mat. The building contains the new fuel storage area and two spent fuel pools, one for each reactor. The spent fuel pools have thick concrete walls and floor lined on the inside surfaces with stainless steel pla:es for leaktightness. The building superstructure consists of concrete walls and roof system.

The new and spent fuel bundles are stored in stainless steel racks spaced 21 inches center-to-center each way. The spent fuel pools are filled with borated water. Each pool is sized to hold fuel bundles for 1-2/3 reactor cores.

The fuel handling building has an overhead crane capable of handling such 'eavy loads as the fuel cask. Travel of this crane is preve. ed by design from moving over the spent fuel pools. Interlocks are provided to prevent the crane from moving over the new fuel area during cask handling operations. A fuel handling crane that runs on rails mounted on the operating floor is provided to handle the new and spent fuel assemblies.

Refer to figures 3.8-39 and 3.8-40 for a plan and sectional view of the building.

3.8.4.1.5 Nuclear Service Cooling Towers

The Vogtle Nuclear Power Plant has two nuclear service cooling towers per unit with a storage capacity for 3,640,000 gallons of water in each tower. The cooling towers are partially buried in the soil with a base elevation of 132 feet above datum. The towers are 77 feet apart at their closest point. The water depth in the tower reservoir is 80 feet.

The cooling tower is a concrete cylindrical shell with top of base slab at elevation 132.0 ft and the top of roof at elevation 750.0 ft. It has a total height of 118 ft and an outside diameter of 94 ft. The thickness of the wall is 3 ft, while that of the roof and base slab is 2 ft and 4 ft respectively. There are four fan cylinders extending 14 ft above the roof. At elevation 220 ft there are openings around the wall. The size of the openings is 12 ft by 10 ft with a separation of 2 ft. The fill supports are at elevation 227 ft, while those of cooling fans are at elevation 250 ft. Two

R_o Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

B. Severe Environmental Loads

Severe environmental loads are those loads to be infrequently encountered during the plant life. Included in this category are:

E Loads generated by the operating base earthquake (0.6 SSE).

W Loads generated by the design wind specified for the plant.

C. Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but are highly improbable. They include:

E' Loads generated by the safe shutdown earthquake and

W_t Loads generated by the design tornado specified for the plant. They include loads due to the tornado wind pressure, due to tornado-created differential pressures, and due to tornado-generated missiles.

D. Abnormal Loads

Abnormal loads are those loads generated by a postulated accident within a building and/or compartment thereof. Postulated accidents primarily include high energy pipe ruptures. Included in this category are the following:

P_a Pressure equivalent static load within or across a compartment and/or building, generated by the postulated accident, and including an appropriate dynamic load factor applied to the peak of the pressure-time curve.

T_a Thermal effects and loads generated by the postulated accident.

R_a Pipe reactions under thermal conditions generated by the postulated accident.

Y_r Reaction equivalent static load on the rupture high energy pipe during the postulated accident, and including an appropriate dynamic load factor applied to the peak of the reaction-time curve.

Y_j Jet impingement equivalent static load on a structure generated by a rupture of any high energy pipe during the postulated accident, and including an appropriate dynamic load factor applied to the peak of the jet-time load.

Y_m Missile impact equivalent static load on a structure generated by or during the postulated accident, and including an appropriate dynamic load factor applied to the peak of the missile impact-time curve.

3.8.4.3.8.2 Loading Combinations for Concrete and Steel Structures. For design load combinations for structures under this paragraph, refer to tables 3.8-7 and 3.8-8.

3.8.4.4 Design and Analysis Procedures

The Category I structures are designed to maintain elastic behavior when subjected to various loading combinations. Plastic design is not intended to be used for any Category I structures analysis.

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3.8.4.4.2 Auxiliary Building

The static analysis of the auxiliary building is performed using the theory of elastic frames. The equivalent frame method outlined by the ACI-318-71 code is used. If in certain

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areas the modeling of the slabs and walls by the equivalent frame method becomes inaccurate, the finite element will be used in these local areas. The computer program for the analysis is ICES-STRUDL (Structural Design Language), which is outlined in appendix 3F.

The analysis of the base slab is performed using plates on elastic foundation technique. Finite element representation of slabs is used. STRUDL uses the stiffness method for the analysis. All different loads are handled by STRUDL, including thermal and tornado loads. The thermal loads are applied as a rise in temperature or a gradient. The tornado loads are applied in three different ways. First a 300 mph applied as windward, leeward, and upward pressure. Second a 3 psi differential pressure between the inside and outside of the building for the building portions above ground. Third a missile impingement load as described in subsection 3.3.2.

The dynamic analysis of the auxiliary building is accomplished as follows:

A dynamic analysis is performed for the purpose of generating seismic response spectrum at different elevations. The mathematical model is a lumped-mass free-free stick model. The modeling technique is described in section 3.7.

Soil-structure interaction due to the embedment of the structure is discussed in section 3.7. The time dependent boundary condition described in paragraph 3.7.2.4 is applied at the appropriate nodes of the model to represent the interaction. This boundry condition is a function of accelerations.

The output of the dynamic analysis is the time history at different elevations of the structure. Once the time history is obtained, the response spectrum is developed for that elevation. The SPECTRA program, as described in appendix 3F, is used to generate the response spectrum curves which are required for the design of the equipment on the various floors.

The maximum inertial forces of the dynamic analysis are applied at the different floor elevations using a three dimensional model of the structure. The stresses in the structural members are then calculated for these forces.

3.8.4.4.3 Control Building

The static and dynamic analysis of the control building is the same as for the auxiliary building.

compound. A thick membrane is placed on the soil structure upon which the basin mat is poured. The waterproofing material is used to prevent any intrusion of ground water through the basin shell and mat. The interior surfaces are also waterproofed using a thermoplastic membrane cast in the concrete. The cooling tower and basin are embedded to a considerable depth. This embedment gives rise to additional resistance to overturning from lateral soil pressure. The soil structure interaction is accomplished by the LUSH program.

3.8.5.1.10 Equipment Building

The equipment building is supported on the adjacent buildings except for the equipment hatch area which is supported at grade level by means of a conventional slab and grade beam foundation system or a mat foundation. S6

3.8.1.1.11 Pipe and Electrical Cable Tunnels and Electrical Duct Banks

The design for the seismic Category I pipe and electrical cable tunnels and electrical duct banks is presently in development stage as stated in paragraph 3.8.4.4.8. For preliminary layout of the underground systems, refer to figures 3.5-1 and 3.5-2. The discussion as pertains to buried structures in this paragraph will be covered in the FSAR.

3.8.5.2 <u>Applicable Codes, Standards and Specifications</u>

Refer to paragraph 3.8.1.2 for the containment and to paragraph 3.8.3.2 for other Category I structures.

3.8.5.3 <u>Loads and Loading Conditions</u>

Containment foundation loads and loading combinations are discussed in paragraphs 3.8.1.3 and 3.8.3.3.

Foundation loads and loading combinations for other Category I structures are discussed in paragraph 3.8.4.3.

3.8.5.4 <u>Design and Analysis Procedures</u>

Design and analysis procedures for the containment including the base slab, are discussed in appendix 3R.

The basic techniques for analyzing the foundations of Category I structures are by the conventional methods, involving

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simplifying assumptions, such as are found in the Theory of concrete structures practice. Stresses resulting from local moments, torques, and concentrated reactions, and uniform loading are computed by these methods. These methods are further discussed in paragraph 3.8.3.4.

3.8.5.5 Structural Acceptance Criteria

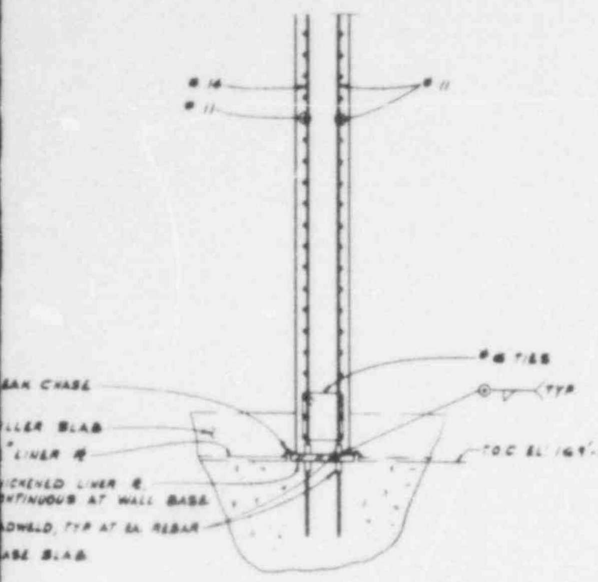
The foundations of all Category I buildings are designed to meet the same structural acceptance criteria as the buildings themselves. These criteria are discussed in paragraphs 3.8.1.5, 3.8.3.5 and 3.8.4.5. The limiting conditions for the foundation medium together with a comparison actual capacity and estimated structure loads are found in paragraphs 2.5.4.10 and 2.5.4.11.

3.8.5.6 Materials Quality Control and Special Construction Techniques

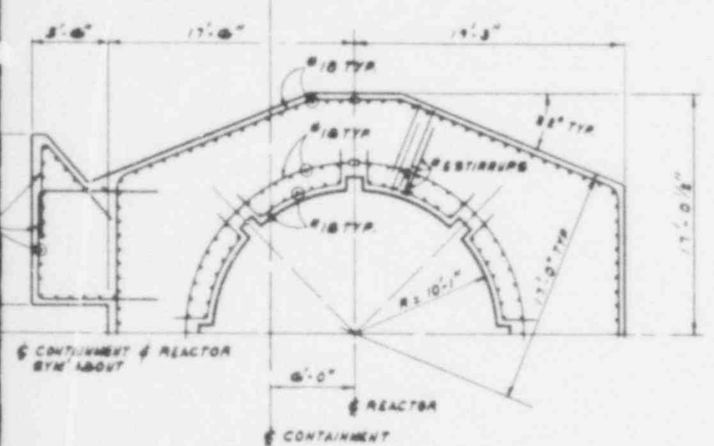
The foundations and concrete supports are constructed of concrete using proven methods common to heavy industrial construction. For further discussion, refer to paragraph 3.8.3.6.

3.8.5.7 Testing and Inservice Surveillance Requirements

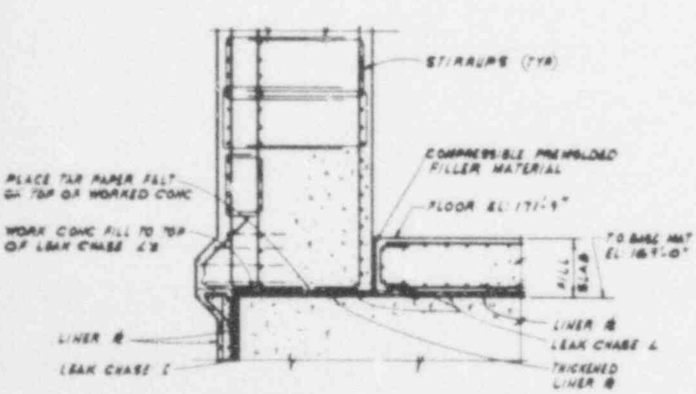
Testing and inservice surveillance are not required and are not planned for foundations of structures or for concrete supports. A discussion of the test program which serves as the basis for the Soils Investigation and Foundation Report in Chapter 2, appendix 2A.



SECONDARY SHIELD WALL SECTION



REACTOR CAVITY PLAN SECTION



PRIMARY SHIELD WALL SECTION

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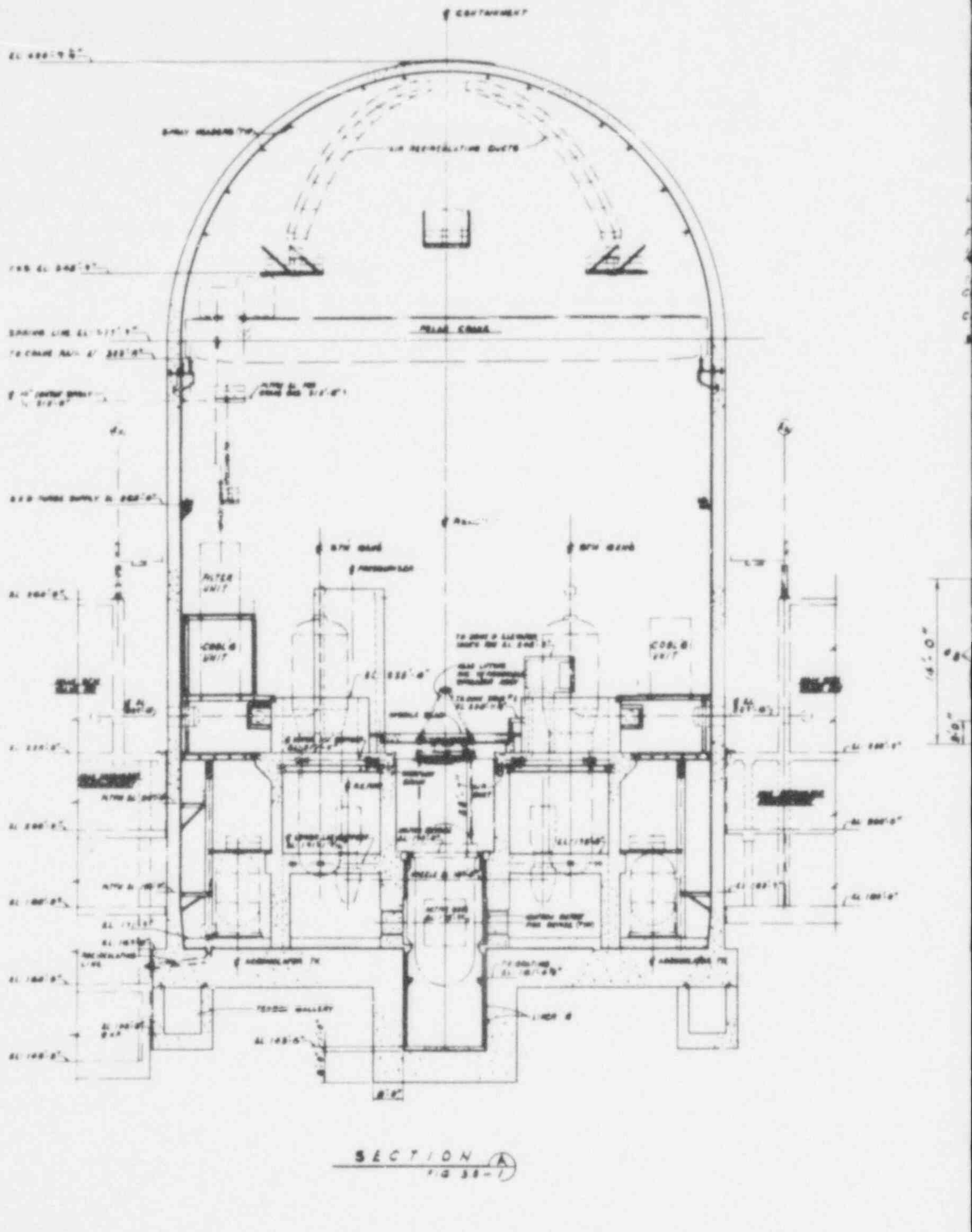
Georgia Power	ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2
INTERIOR STRUCTURE TYPICAL SECTIONS	
FIGURE 3.8-2	



SECTION A

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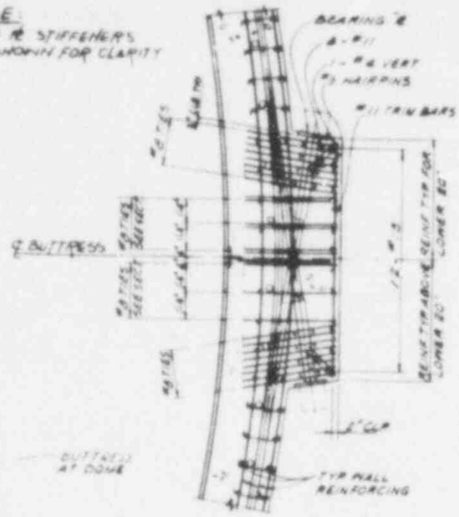
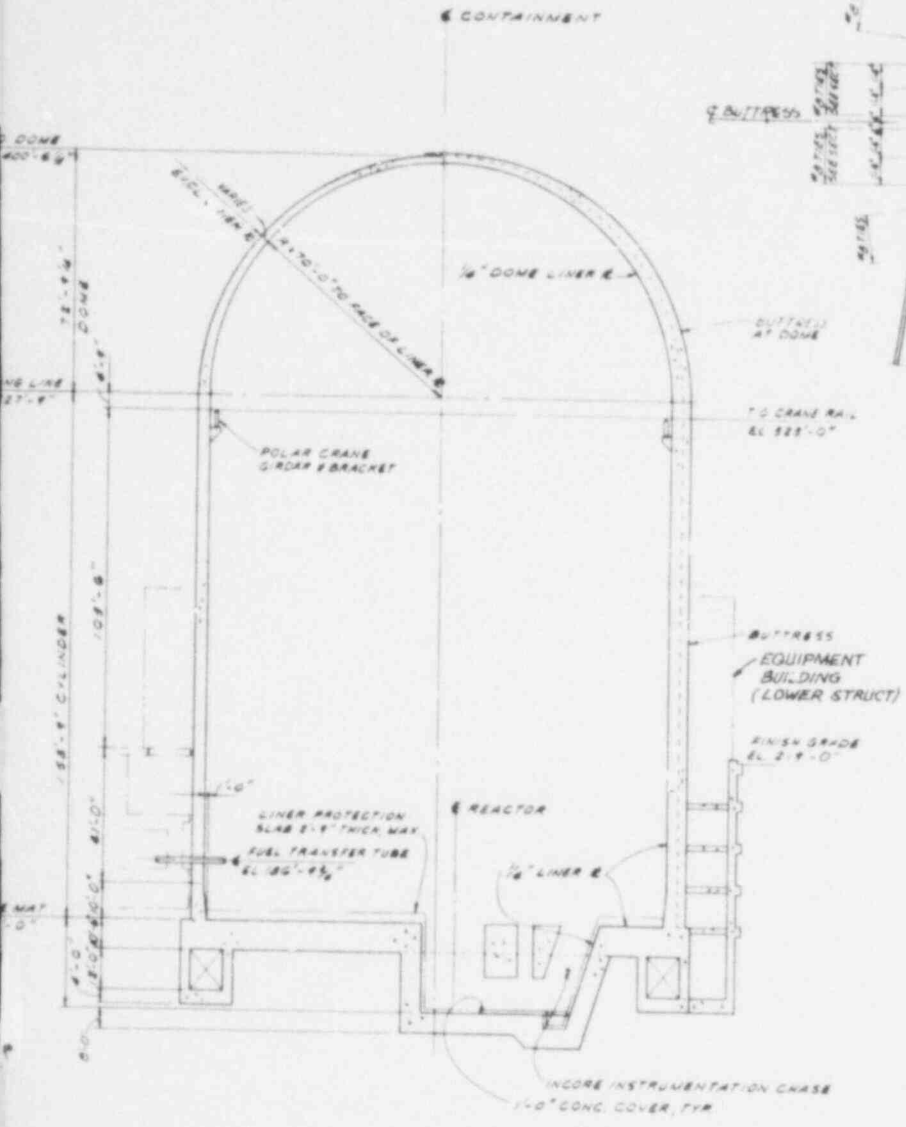
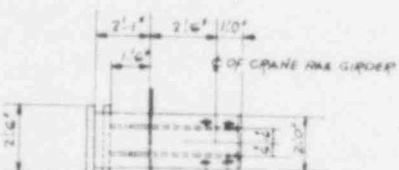
NOTE:
LINER & STIFFENERS NOT SHOWN FOR CLARITY



BUTTRESS REINFORCING



SECTION



PLAN B
1/2" = 1'-0"

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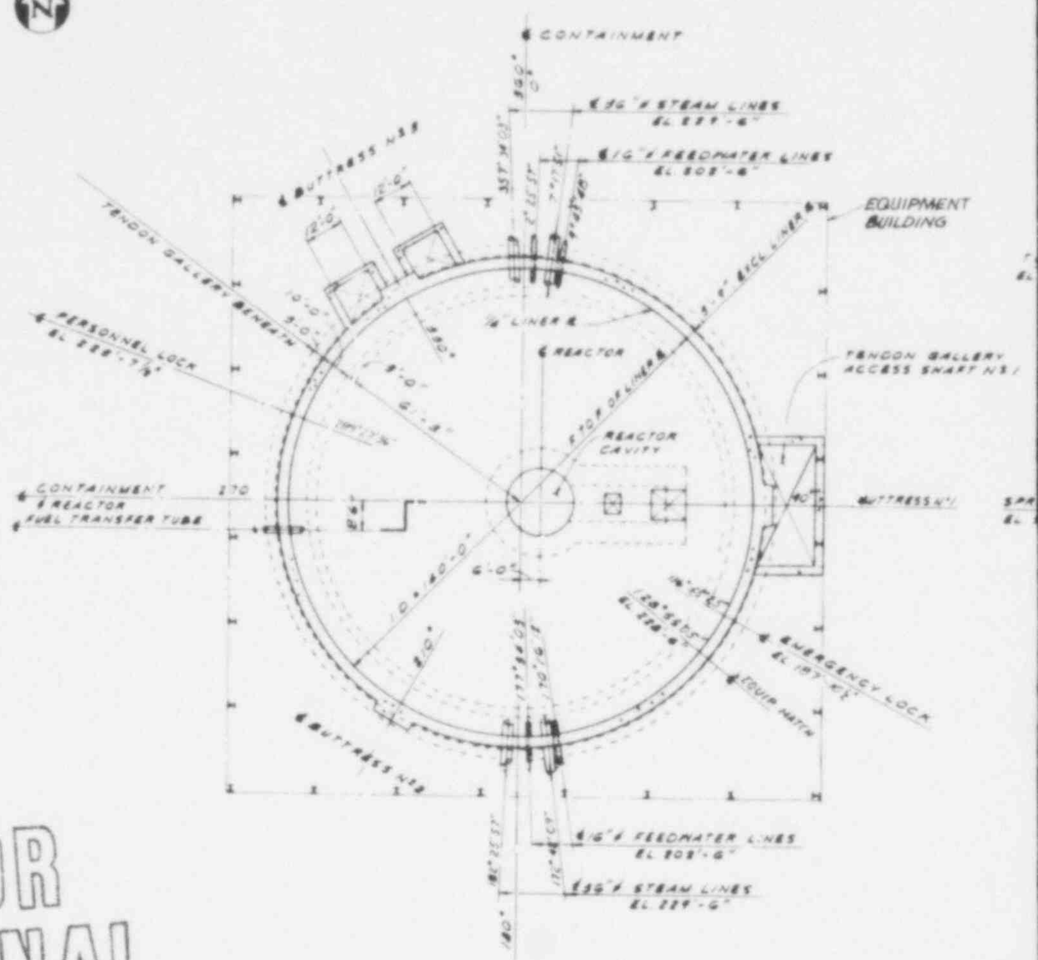
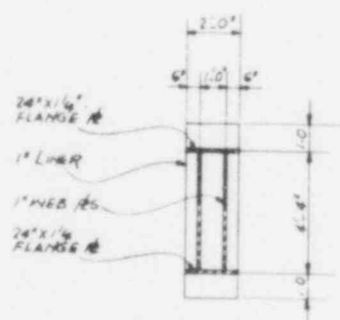
Georgia Power	ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2

CONTAINMENT GENERAL
ARRANGEMENT TYPICAL DETAILS

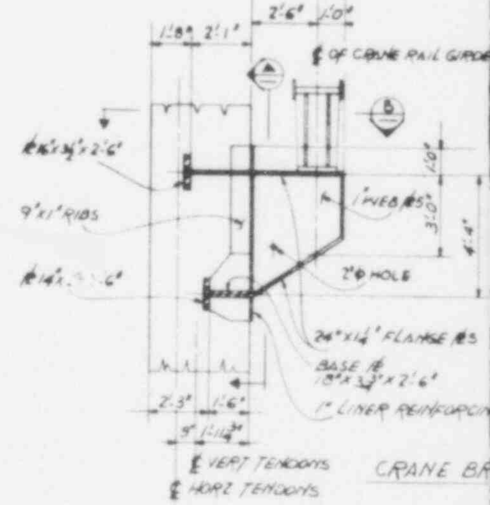
FIGURE 3.8-4



PLAN



SECTION (A)
½"=1'-0"



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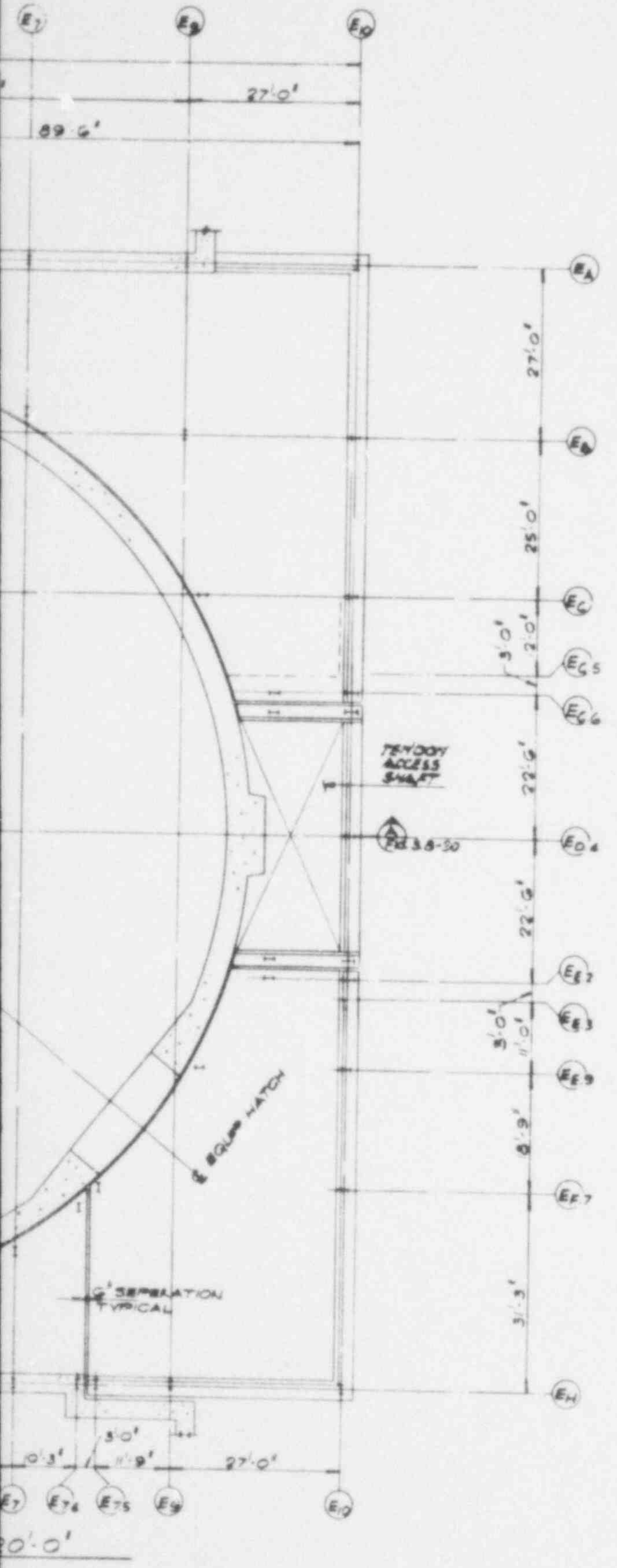

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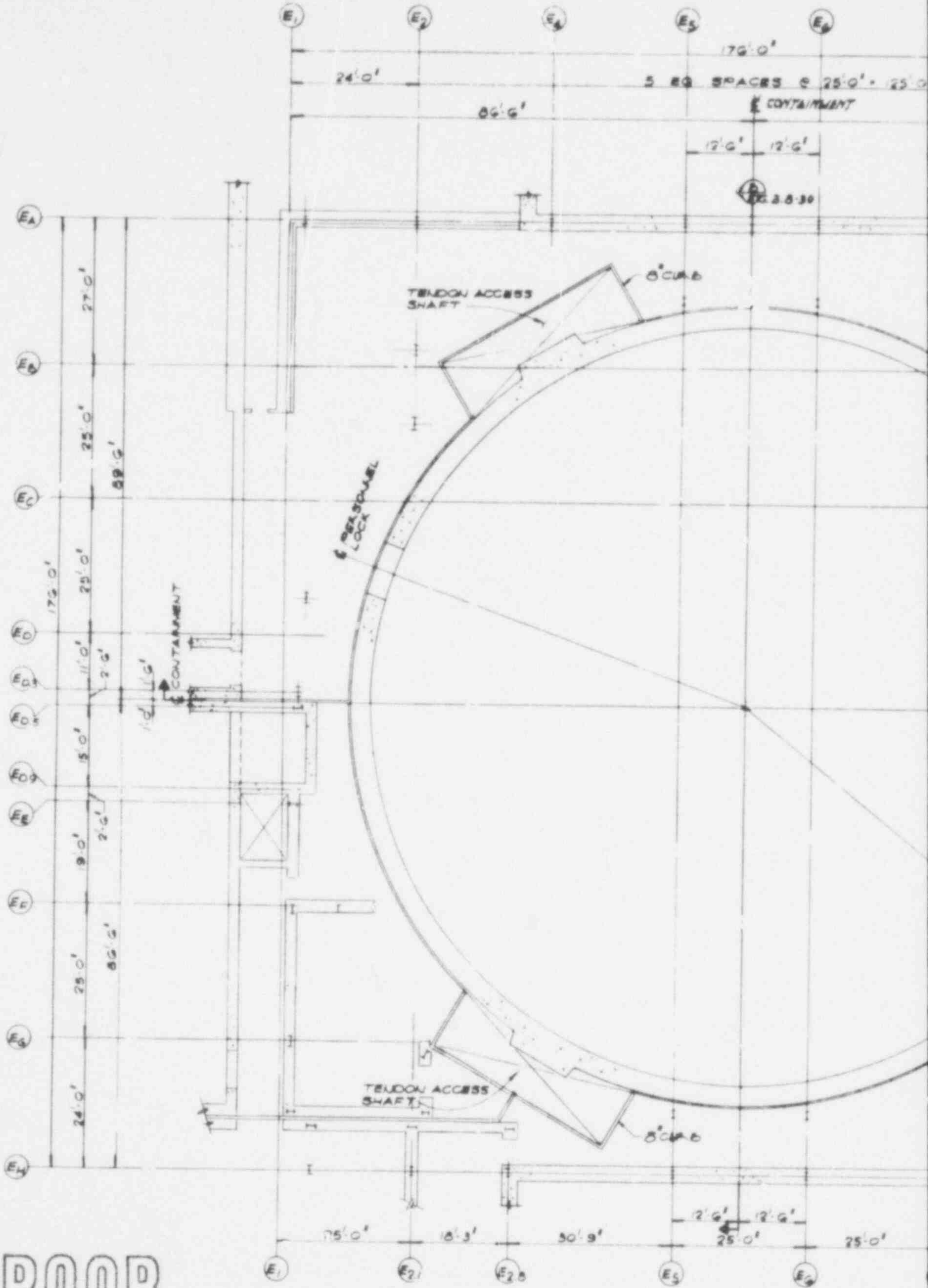
ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2

EQUIPMENT BUILDING
PLAN AT FINISH GRADE

FIGURE 3.8-27



PLAN AT ELEVATION 22

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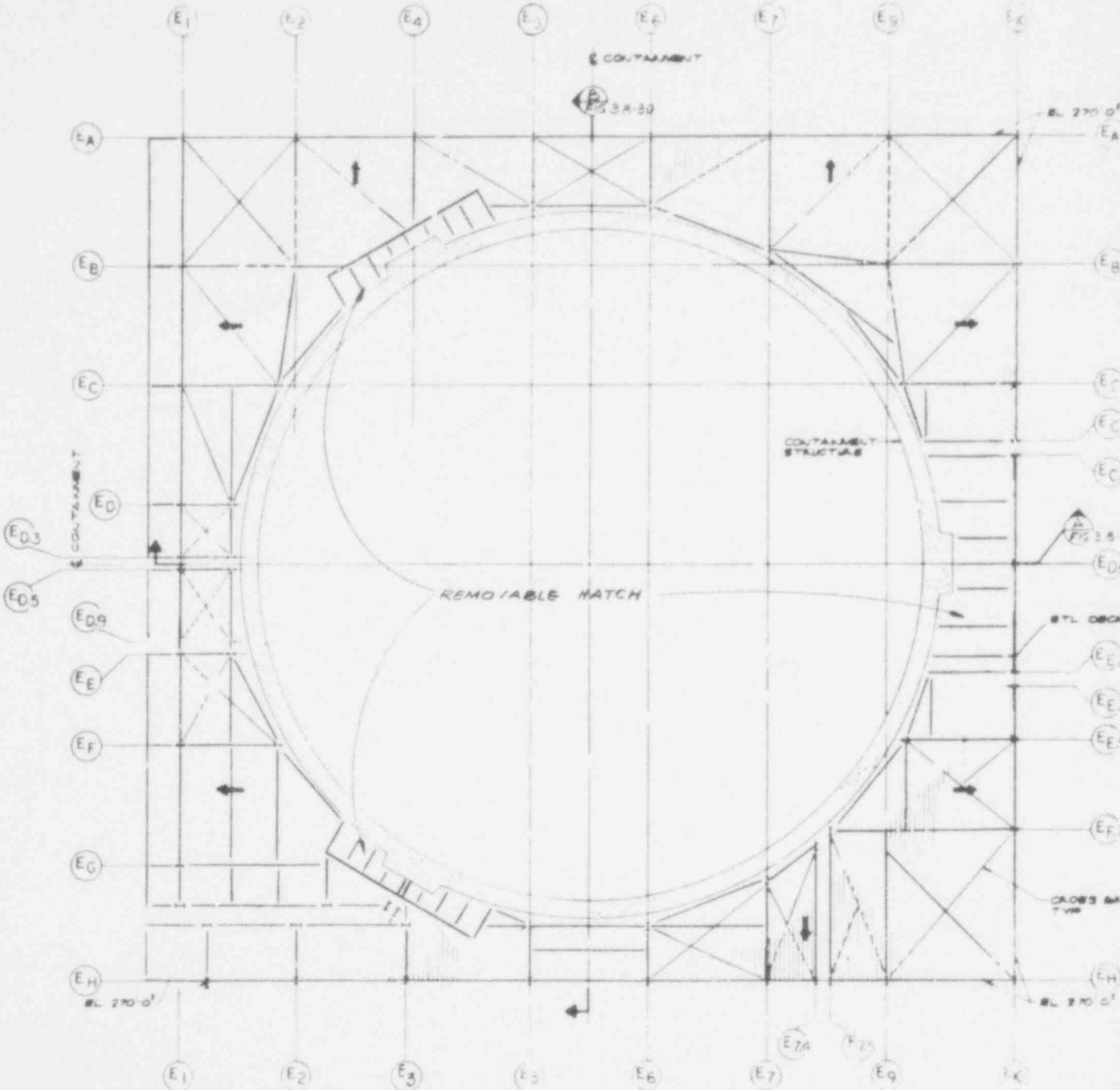
Georgia Power	ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2

EQUIPMENT BUILDING
ROOF FRAMING PLANS

FIGURE 3.8-28



ROOF FRAMING PLAN @ ELEVATION 270'-0"

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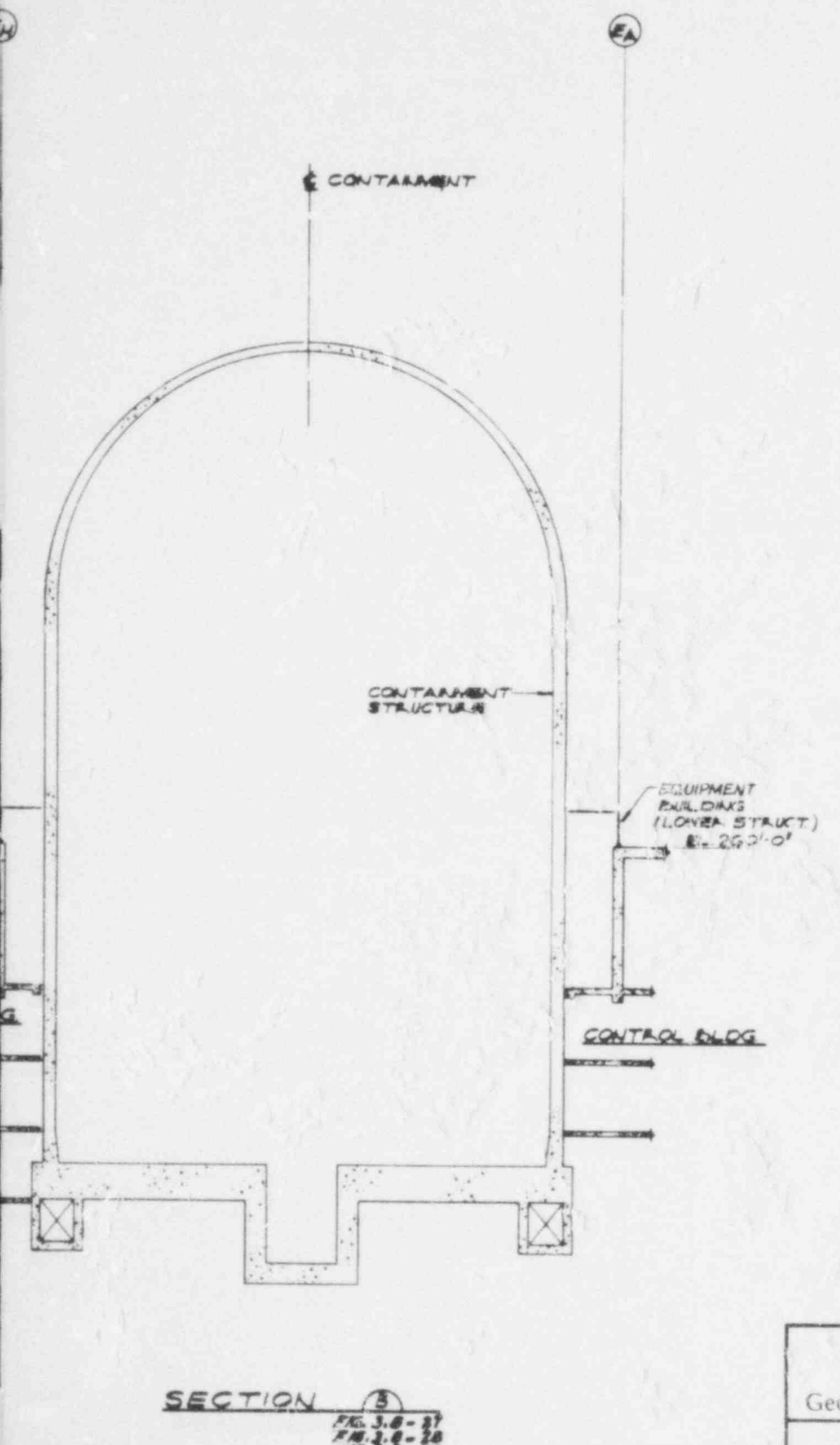
Figure 3.8-29



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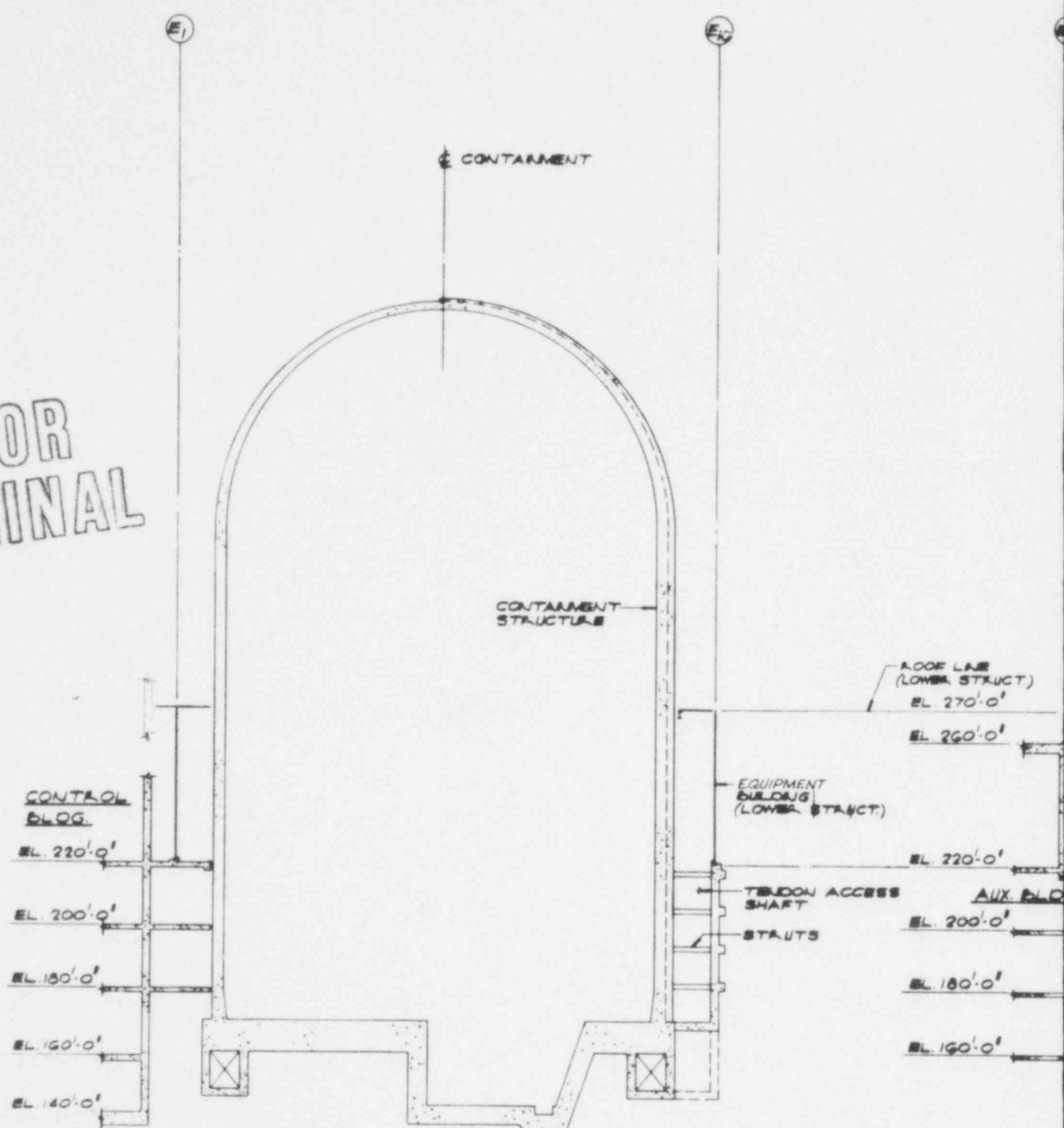
Georgia Power

ALVIN W. VOGTLE NUCLEAR PLANT UNITS 1 AND 2

EQUIPMENT BUILDING SECTIONAL ELEVATIONS

FIGURE 3.8-30



SECTION A
FIG. 3.8-27
FIG. 3.8-28
FIG. 3.8-29

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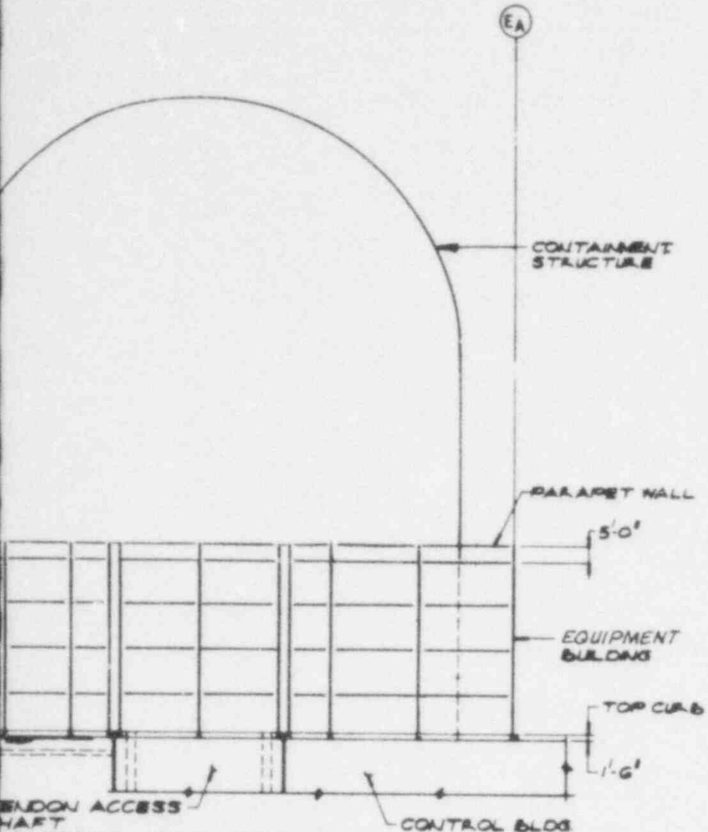
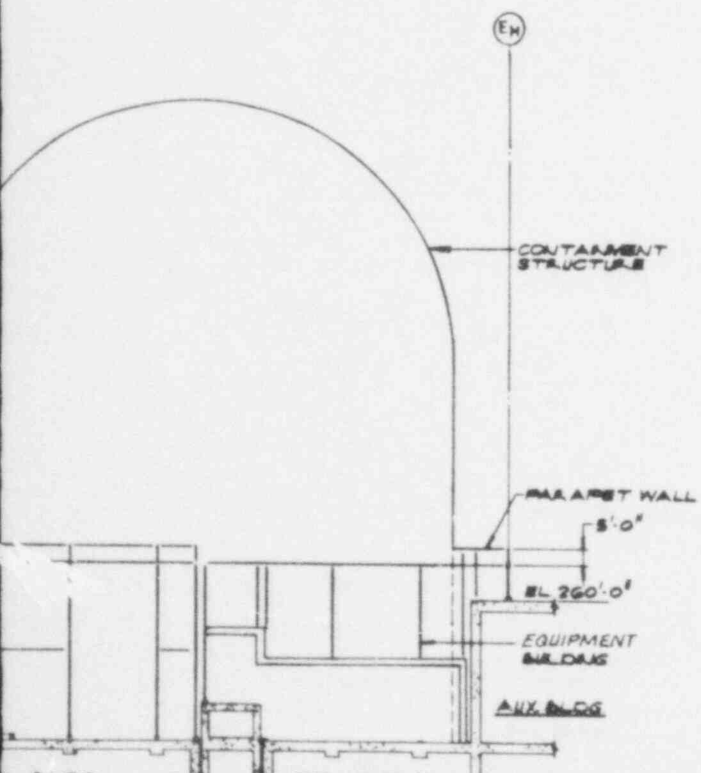


EXTERIOR EAST ELEVATION



XTERIOR WEST ELEVATION

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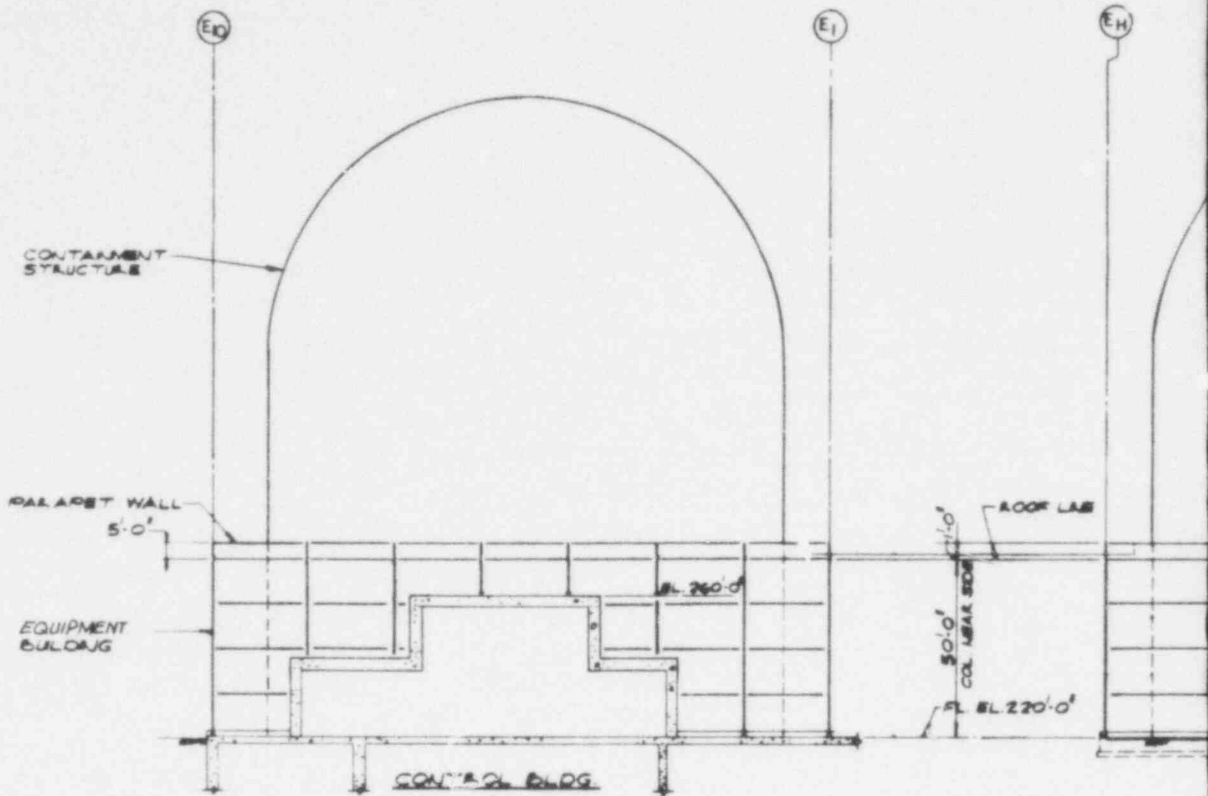
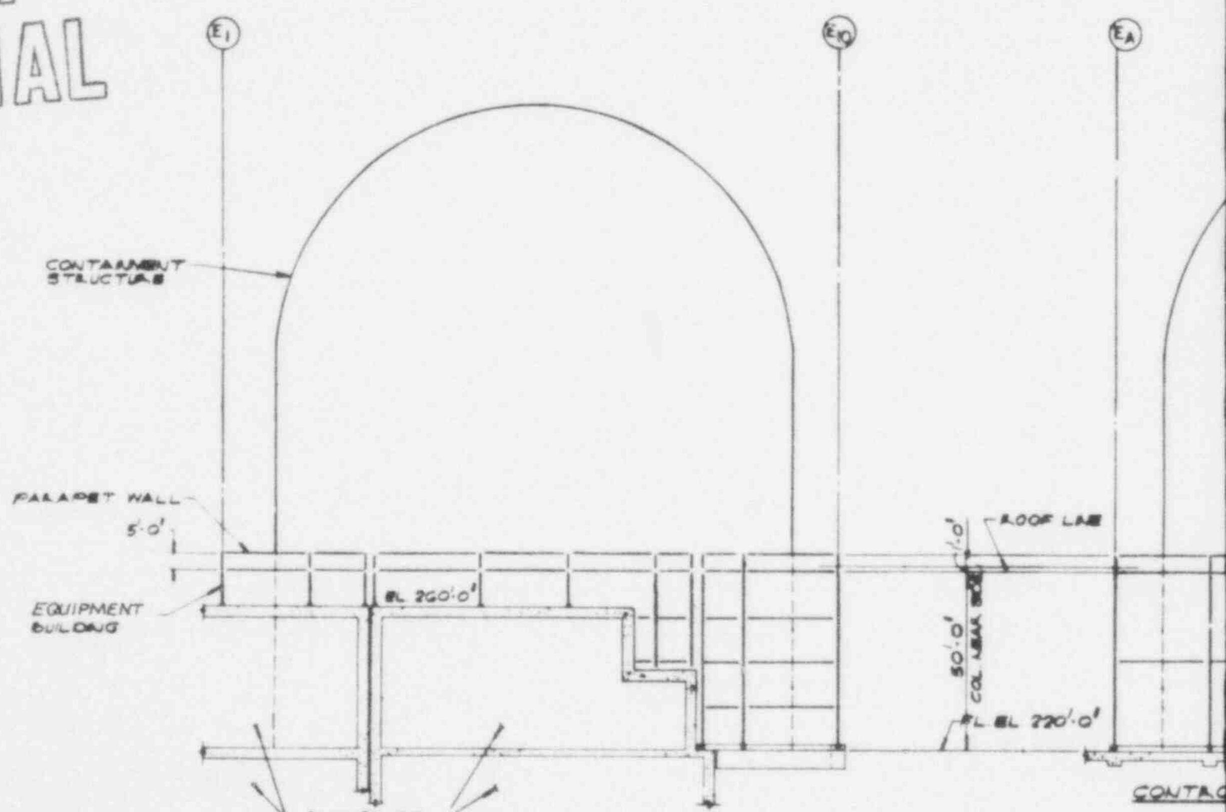
ALVIN W. VOGTLE
NUCLEAR PLANT
UNITS 1 AND 2

EQUIPMENT BUILDING
ELEVATIONS

FIGURE 3.8-31



EXTERIOR NORTH ELEVATION



EXTERIOR SOUTH ELEVATION

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CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 GENERAL

Safety features are designed to minimize the severity and to mitigate the consequences of Condition IV Limiting Faults (PSAR Section 15.4) by fulfilling the following safety functions under accident conditions:

1. Protect the fuel cladding
2. Ensure containment integrity
3. Minimize containment leakage
4. Remove fission products from the containment atmosphere.
5. Provide habitable atmosphere for operating personnel.
6. Limit radioactivity discharged to the atmosphere (and thence the environment)

6.1.1 SAFETY FEATURES SYSTEMS

The safety features systems provided to satisfy the functions listed above are as follows:

Containment isolation system (subsection 6.2.4)
Containment spray system (subsection 6.2.2)
Containment fan cooler system (subsection 6.2.2)
Containment air purification and cleanup system (subsection 6.2.3)
Emergency core cooling system (subsection 6.3)
Residual heat removal system (subsection 5.5)
Penetration room filtration system (subsection 6.5)
Combustible gas control in containment (subsection 6.2.5) S6
Engineered Safety Features Electrical Equipment Rooms HVAC (subsection 9.4.6)
Fuel Handling Building post accident filter exhaust system (subsection 9.4.5)
Control Room HVAC (subsection 9.4.1A)

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The first safety function is satisfied by the timely, continuous, and adequate supply of borated water to the reactor coolant system and, ultimately, the reactor core. This supply of water is provided by the emergency core cooling system. These systems provide high head (safety injection and centrifugal charging pumps), low head (residual heat removal pumps), injection and accumulator injection immediately following an incident, and low head/high head recirculation in the long term recovery period. The second and third safety

functions are satisfied by the provision of means for condensing the steam inside the containment, depressurizing the containment following an incident, and maintaining the containment at near atmospheric conditions for an extended period of time. The containment isolation system, spray system, fan
S6 cooler system, penetration room vent and filtration system and the electric hydrogen recombiners provide the means for satisfying these requirements.

The fourth safety function is satisfied by providing chemical additives and filters within the containment. These enhance the removal of radioactive iodine from the containment atmosphere following an incident. The containment air purification and cleanup systems are provided to meet this function.

The fifth and sixth safety features are satisfied by the various safety features filtration systems which conforms to Regulatory Guide 1.52 (see table 6.1-1).

The safety features systems are designed with sufficient redundancy to meet the general design criteria as discussed in section 3.1, 3.2 and appendix 6A. Electrical power for all safety features systems is provided both from offsite sources and from emergency onsite sources as described in sections 8.2 and 8.3, respectively.

Safety features are separated into two independent trains of equal capability. Either train can handle the entire emergency coolant injection and emergency cooling loads; either train can provide the entire containment isolation, containment cleanup, and containment leakage minimization functions. Each train has an independent onsite and offsite power source. Failure of either train cannot affect the other. Both trains are shown in figure 6.1-1, Safety Features Systems.

Some of the high and low pressure emergency injection systems use equipment that serves normal functions during normal plant operation or shutdown. Observation of their normal functioning provides monitoring of equipment availability and condition. In cases where equipment is used for emergencies only, systems are designed to permit periodic inspection and tests.

6.1.2 OPERATIONAL RELIABILITY

Operational reliability is achieved by using proven components and by conducting tests. All safety features systems are quality items meeting the requirements of 10 CFR 50 appendix B and seismically designed as discussed in chapter 3. Those

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Table 6.1-1

ATMOSPHERE CLEANUP SYSTEM
AIR FILTRATION AND ADSORPTION
UNITS - COMPLIANCE WITH
REGULATORY GUIDE 1.52

REGULATORY POSITION														
4e	4f	4g	4h	4i	4j	4k	4l	4m	5a	5b	5c	6a	6b	6c
X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X	X	X	X	X	X	X	X

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k	2l	2m	3a	3b	3c	3d	3e	3f	3g	3h	3i	3j	3k	3l	3m	4a	4b	4c	4d
X	X	NA	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	X	NA	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	X	NA	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	X	NA	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X

UNIT DESCRIPTION	REGULATORY POSITION															
	1a	1b	1c	1d	1e	2a	2b	2c	2d	2e	2f	2g	2h	2i	2j	2
Engineered Safety Features Electrical Equipment Room HVAC System	X	X	X	NA	X	X	X	X	NA	X	X	X	X	X	X	X
Fuel Handling Building Post Accident Filter Exhaust System	X	X	X	NA	X	X	X	X	NA	X	X	X	X	X	X	X
Piping and Electrical Penetration Room Filter Exhaust System	X	X	X	NA	NA	X	X	X	NA	X	X	X	X	X	X	X
Control Room HVAC System	X	X	X	NA	NA	X	X	X	NA	X	X	X	X	X	X	X

X = Compliance
N/A = Not Applicable

containment structures and the environment was considered to be adiabatic.

6.2.1.2 System Design

The analytical technique and design methods used to assure the integrity of the containment internal structure and subcompartments from the effects of LOCA are discussed in paragraph 3.8.1.4 and 3.8.3.4.

6.2.1.3 Design Evaluation

Table 6.2-2 provides detailed data on which the containment design is evaluated. Mass addition and energy flow into containment for the double ended pump suction break, the worst break, is given in table 6.2-2 sections IV and V. Mass addition and energy flow for the double ended cold leg and the double ended hot leg breaks are given in table 6.2-3 and 6.2-3a respectively.

6.2.1.3.1 Following an accident, the containment air may become radioactively contaminated. The isolation of the containment following an accident therefore is mandatory. During normal operation, air flowing intermittently from inside the containment to the outer atmosphere is filtered and monitored by the normal containment purge system. Following an accident, the normal purge system is removed from service. Containment leaktightness is accomplished by the redundant quick-closing purge isolation valves located at both supply and exhaust ducts to the containment. Additionally, the piping and electrical penetration filter exhaust systems maintain a subatmospheric pressure in the penetration rooms following a loss-of-coolant accident. These systems are discussed further under subsection 6.2.3 and section 6.5 respectively. S6

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Table 6.2-3a (Continued)

MASS AND ENERGY FLOW INTO CONTAINMENT
FOR A DOUBLE-ENDED HOT LEG BREAK

Time (sec)	Mass Rate (lbs/sec)	Energy Rate (Btu/sec)
19.8	0.	0.
20.9	0.	0.
22.18	1231.1	99.83 E+04
27.8	4365.5	158.70 E+04
50.	1928.3	105.82 E+04
*62.800	900.8	84.93 E+04
62.801	194.0	23.09 E+04
100.	176.0	20.93 E+04
200.	143.56	17.08 E+04
500.	104.67	12.46 E+04
1000.	79.20	9.43 E+04
2000.	60.60	7.20 E+04
5000.	42.05	5.36 E+04
10000.	35.78	4.26 E+04

* Time that quench front reaches 8 ft elevation in the core.

S6 6.2.1.3.2 The piping and electrical penetration filter exhaust systems are designed to operate during emergency situations. The systems are designed as seismic Category I. All equipment is provided with 100 percent redundancy, including electrical buses. Each train in these systems is connected to one emergency bus which assures operation following loss of offsite power.

6.2.1.3.3 Blowdown Containment Pressure Analysis

6.2.1.3.3.1 Results. The results of the pressure transient analysis of the containment for the blowdown phase of the loss-of-coolant accident are shown in figure 6.2-1 sheet 1 and 2. The cases examined in this analysis determine the effects of the full range of large reactor coolant break sizes up to and including a double ended rupture. Cases illustrating the sensitivity to break location are also shown. All of these cases show that the containment pressure will remain below design pressure with margin. This blowdown margin can be represented in terms of energy or pressure.

Figure 6.2-2, the plot of containment pressure versus internal energy of the containment atmosphere, can help quantize the available margin. Point A on this figure represents the blowdown pressure; point B corresponds to the containment design pressure. The increase in energy necessary to go from point A to point B in this figure represents the energy margin available in the containment design. Since energy transferred to the containment from the core is in the form of steam, the total transferred core energy corresponding to allowed energy addition can be calculated as follows:

$$Q_{core} = \frac{h_{fg}}{h_g} Q_{allowed}$$

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Q_{core} = Core energy release

h_{fg} = Latent heat of vaporization

h_g = Steam enthalpy

$Q_{allowed}$ = Permissible increase in containment energy

This allowable value of energy which could be transferred from the core to the containment without increasing the transient containment pressure to design pressure can be compared to the

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6.6 DELETED S6

S6 The contents of Pages 6.6-1a thru 6.6-13 have been deleted.

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Figure 6.6-1

This figure is deleted by Supplement No. 6

Figure 6.6-2

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CHAPTER 7

LIST OF TABLES

Tables for this chapter are presented in RESAR-3, chapter 7.

LIST OF FIGURES

Figures for this chapter are presented in RESAR-3, chapter 7.

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7.3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

This section is presented in RESAR-3, Section 7.3 with the following modifications and additions.

7.3.1.1.1.2

a. Per RESAR-3

b. Per RESAR-3

c. Containment air recirculation fans and filtration system which serve to cool the containment and limit the potential for release of fission products from the containment by reducing the pressure following an accident.

d. Nuclear service water pump which provides cooling water to the component cooling system heat exchangers and is thus the heat sink for containment cooling.

e. Auxiliary feedwater pumps.

f. Containment dome circulator fans which will prevent stratification of gases in the containment dome and the possible accumulation of a dangerous mixture of hydrogen and air from occurring.

g. Containment cooling units which serve to reduce pressure by cooling the containment following an accident.

h. Penetration room filtration system which will limit release to the environment of radioisotopes that have leaked from the containment into the penetration rooms. S6

7.3.1.1.2 Analog Circuitry

Numbers 1 through 4 are per RESAR-3.

5. Containment air coolers

The containment air coolers cooling water discharge flow is alarmed in the control room if the flow is low or if the differential (inlet versus outlet) flow is high or low. Cooling water flow through each cooler is recorded in the control room. The transmitters are outside the reactor containment. Local pressure and temperature indicators are provided outside the containment to monitor the cooler cooling water discharge flow. Air cooler air flow is monitored by an alarm function in the control room which is actuated by low discharge air flow. In addition a common radiation monitor exists for exit cooling water flow which will actuate a control room alarm in the event of high radiation. The faulty cooler can be detected locally by manually valving each cooler out in turn.

The instrument lines which sense cooling water flow do not penetrate the containment but are tapped into the process cooling water lines at orifice plates located outside of the automatic isolation valves which are external to the containment. These valves are installed in accordance with 10CFR50, Criterion 57. The low discharge air flow signal is provided by an electric switch contact.

6. (Per RESAR-3)

S6 7. Deleted

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CHAPTER 8

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Table 8.1-1

SAFETY LOADS AND FUNCTIONS

SAFETY LOADS	FUNCTION	POWER
Component Cooling Water Pumps	Provide cooling water to NSSS equipment	AC
Residual Heat Removal Pumps	Remove reactor heat during hot shutdown	AC
Charging Pumps	Provide emergency core cooling during emergency shutdown	AC
Safety Injection Pumps	Provide emergency core cooling during emergency shutdown	AC
Containment Spray Pump	Provide cooling spray in containment during LOCA	AC
Nuclear Service Cooling Water Pumps	Provide water for component cooling water system	AC
Containment Fans	For cooling containment after accident	AC
Spent Fuel Pool Coolant Pump	Cool Spent Fuel Pool	AC
Hydrogen Recombiner	Maintains a safe level of Hydrogen in Containment	AC
Emergency Air Conditioning	Maintains a safe air operating temperature	AC
Fire Pump	To provide water to extinguish fires	AC
Reactor Service Cooling Twr. I.D. Fans	Cool service cooling water	AC
Motor Control Centers	Provide power for M.O.V., small motors, fans, heaters, and small pumps associated with safety related equipment.	AC
Penetration Room Filtration & Vent	Minimize containment leakage	AC

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Table 8.1-1 (Continued)

SAFETY LOADS AND FUNCTIONS

SAFETY LOADS	FUNCTION	POWER
Fuel Handling Building Filtration & Vent	Mitigate the effects of fuel handling accident	AC
Containment H_2 Purge System	Control Post-LOCA H_2 Concentration	AC
Safeguard Logic Relay System	Initiates safety injection signal	DC
Reactor Protection Logic Relay System	Prevents reactor from operating in unsafe conditions	DC
Auxiliary Relay Cabinet	Auxiliary relays for process control and nuclear instrumentation	DC
Reactor Trip Switchgear	Trips reactor	DC
Waste Disposal Panel	Control panel for waste disposal system	DC
Instrument Power Supply Inverter	Supplies power to the vital instrument busses	DC
NSSS Solenoid Valves: CVCS, RCS SIS Sampling System	Controls flow of the NSSS (pneumatic valves with solenoid actuators)	DC
Distribution Panel	Supplies power to emergency lighting and protective relay panel	DC

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Table 8.3-1

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT AND LOSS OF OFFSITE POWER

Component	No. Installed Per Unit	HP Each Nameplate Rating	Auto. Sequence Start	Delay Time After Safeguard Signal is Actuated
Centrifugal charging pumps	2	600	Yes	10 Sec.
Safety injection pumps	2	400	Yes	15 Sec.
Residual heat removal pumps	2	400	Yes	20 Sec.
Containment spray pumps	2	400	Yes	25 Sec.
Component cooling water pumps	4	800	Yes	30 Sec.
Nuclear service water pumps	4	1100	Yes	35 Sec.
Valves (motor operated)	-	175	Yes	10 Sec.
Nuclear service cooling tower fans	4	125	Yes	10 Sec.
Battery chargers	3	45 kVA	Yes	10 Sec.
Control Room Outside Air Supply	2	10	Yes	15 Sec.
Control Room Air Conditioning	2	60	Yes	15 Sec.
Control Room Heating Coil	2	120	Yes	15 Sec.
Penetration Room Recirculation Exhaust	2	15	Yes	15 Sec.

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Table 8.3-1 (Continued)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR
LOSS-OF-COOLANT ACCIDENT AND LOSS OF OFFSITE POWER

Component	No. Installed Per Unit	HP Each Nameplate Rating	Auto. Sequence Start	Delay Time After Safeguard Signal is Actuated
Penetration Room Exhaust	2	1.5	Yes	15 Sec.
Containment Cooler Fan No. 1	2	150	Yes	15 Sec.
Containment Cooler Fan No. 2	2	150	Yes	15 Sec.
Hydrogen Recombiner	2	75 kW	Yes	15 Sec.
Post-Accident Containment Purge Exhaust Fan	2	1	Yes	15 Sec.
Post-Accident Containment Purge Supply Fan	2	0.75	Yes	15 Sec.
Cavity Cooling Supply	2	10	Yes	15 Sec.
Upper Dome Circulator	2	20	Yes	15 Sec.
Upper Dome Circulator	2	20	Yes	15 Sec.
Refueling Pool Vent Unit	2	3	Yes	15 Sec.
Spent Fuel Pool Post-Accident Exhaust	2	15	Yes	15 Sec.
Electrical Equipment Room Exhaust Air Conditioning	2	15	Yes	15 Sec.

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Table 8.3-1 (Continued)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS
FOR LOSS-OF-COOLANT ACCIDENT AND LOSS OF OFFSITE POWER

Component	No. Installed Per Unit	HP Each Nameplate Rating	Auto. Sequence Start	Delay Time After Safeguard Signal is Actuated
Battery Room Air Conditioning	2	1	Yes	15 Sec.
Emergency submerged deep well make-up pump	2	125	Yes	20 Sec.
Emergency lighting	-	40 kW	Yes	Immediately
Reciprocating charging pump	1	200	No*	-
Boric acid transfer pumps	3	15	No*	60 Min.
Pressurizer heaters	2 Banks/ Unit	350 kW/ Bank	No*	-
Boron injection tank heater	2	6 kW	No*	-
Heat tracing			No*	5 Min.
Instrumentation and control	-	5 kW	Yes	Immediately
Containment Ventilation fans			No*	
Spent Fuel Pool Cooling Pumps	1	250	No*	-
Auxiliary Feedwater Pumps	2	500	Yes	40 Sec.

*Manual start when required.

Train A = 7438 hp
Train B = 7438 hp

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Table 8.3-2

EMERGENCY ELECTRICAL LOADING REQUIREMENTS
FOR BLACKOUT CONDITIONS (Sheet 1 of 2)

Component	No. Installed Per Unit	HP Each Name-plate Rating	Blackout: Auto. Sequence Start	Blackout: Delay Time After Blackout
Centrifugal Charging Pumps	2	600	Yes	10 Sec.
Component Cooling Water Pumps	4	800	Yes	30 Sec.
Nuclear Service Water Pumps	4	1100	Yes	35 Sec.
Valves (motor operated)	-	175	Yes	10 Sec.
Instrumentation and Control	-	5 kW	Yes	Immediately
Nuclear Service Cooling Towers	4	125	Yes	10 Sec.
Battery Chargers	3	45 kVA	Yes	10 Sec.
Control Room Outside Air Supply	2	10	Yes	15 Sec.
Control Room Air Conditioning	2	60	Yes	15 Sec.
Control Room Heating	2	120	Yes	15 Sec.
Penetration Room Recirculation Exhaust	2	15	Yes	15 Sec.

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Table 8.3-2

EMERGENCY ELECTRICAL LOADING REQUIREMENTS
FOR BLACKOUT CONDITIONS (Sheet 2 of 3)

Component	No. Installed Per Unit	HP Each Name-plate Rating	Blackout Auto. Sequence Start	Blackout Delay Time After Blackout
Penetration Room Exhaust	2	1.5	Yes	15 Sec.
Containment Cooler Fan No. 1	2	150	Yes	15 Sec.
Containment Cooler Fan No. 2	2	150	Yes	15 Sec.
Hydrogen Recombiner	2	75 kW	Yes	15 Sec.
Post-Accident Containment Purge Exhaust Fan	2	1	Yes	15 Sec.
Post-Accident Containment Purge Supply Fan	2	0.75	Yes	15 Sec.
Cavity Cooling Supply	2	10	Yes	15 Sec.
Upper Dome Circulator	2	20	Yes	15 Sec.
Upper Dome Circulator	2	20	Yes	15 Sec.
Refueling Pool Vent Unit	2	3	Yes	15 Sec.
Spent Fuel Pool Post-Accident Exhaust	2	15	Yes	15 Sec.
Electrical Equipment Room Exhaust Air Conditioning	2	15	Yes	15 Sec.
Battery Room Air Conditioning	2	1	Yes	15 Sec.

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FIRE PROTECTION FOR SAFETY RELATED EQUIPMENT

Safety Related Equipment	Fire Protection System	Operation	Detection Device	Accessibility: Heat	Accessibility: Radiation	Accessibility: Toxic Combustion Products
Control room support area air conditioning unit	E, F	M	HD	0	0	0
Post-LOCA purge unit	E	M	HD, SD	0	0	0
Control room	E, F	M	HD, SD	0	0	0
Normal-emergency containment air cooling units	F	M	HD, SD	0, X	0, X	0
Component cooling water heat exchanger	C, H	A, M	S	0	0	0
Turbine driven auxiliary feedwater pump	C, H	A	S	0	0	0
Diesel driven auxiliary feedwater pump	D	A	HD, SD	0	0	0
4160 V switchgear	G	M	SD	0	0	0
Penetration rooms filtration system	E, F	M	HD	0	0	0
Cable spreading room	D	M	HD	0	0	P
Accumulator tanks	F	M	HD, SD	0, X	0, X	0
Component cooling water pumps	C, H	A, M	S		0	0
Emergency air conditioning system	E, F	M	HD	0	0	0
MCC and switchgear rooms	G	M	SD	0	0	P
125 VDC battery rooms	E	M	SD	0	0	0
Communication room	E	M	HD	0	0	0
Electric and air conditioning equipment room	G, E, F	M	HD, SD	0	0	P
RHR heat exchangers	C, H	A, M	S	0	0	0
Containment spray heat exchangers	C, H	A, M	S	0	0	0

Legend

Fire Protection System

C - Std. wet pipe sprinklers

D - Fixed pipe CO_2 system | S6

E - Portable CO_2 extintinguisher

F - Portable dry chemical extinguisher

G - CO_2 hose reels

H - Firewater hose station

Detection Devices

S - Sprinkler head

HD - Heat detector

SD - Smoke detector

Operation

A - Automatic

M - Manual

Accessibility

0 - No special protective device

P - Protective device provided

X - Limited access

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12.2 VENTILATION

12.2.1 DESIGN OBJECTIVES

The plant is designed to provide maximum safety and convenience for operating personnel with equipment arranged in zones so that areas of potential contamination are separated from clean areas. The plant ventilation systems are designed to provide a suitable environment for equipment and personnel.

12.2.2 DESIGN DESCRIPTION

12.2.2.1 Containment Building

Ventilation for the containment building is discussed in section 6.2.

12.2.2.2 Penetration Rooms

Ventilation for the penetration rooms is discussed in section 6.5.

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12.2.2.4 Control Room Area

Ventilation for the control room area is discussed in section 9.4.1.

12.2.2.5 Fuel Handling Building

Ventilation for the fuel handling building is discussed in section 9.4.5.

12.2.2.6 Radwaste Area

Ventilation for the radwaste area is discussed in section 9.4.3.

12.2.2.7 Battery Room

Ventilation for the battery room is discussed in section 9.4.6.

12.2.2.8 Diesel Generator Building

Ventilation for the diesel generator building is discussed in section 9.4.7.

12.2.2.9 Turbine Building

Ventilation for the turbine building is discussed in section 9.4.4.

12.2.2.10 Plant Vents

Plant vents in seismic Category I structures are designed for seismic Category I and for tornado conditions.

12.2.3 SOURCE TERMS

Design estimates of normal annual liquid leakage volumes for a single unit are presented in section 11.2. Airborne radioactivity will be calculated based on an expected distribution of leakage into the containment and auxiliary building. All noble gases and a portion of the halogens and particulates are assumed to become airborne.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

The ventilation airborne radioactivity monitoring system provides radiation measurements, indications, records, alarms and controls at selected locations to verify compliance with AEC applicable limits to detect and control abnorma occurrences within the plant.

Monitors are located in ventilation systems where personnel exposure to radiation is most likely, gaseous effluent streams, and in the process line of the waste gas processing system. A detailed description of the airborne radioactivity monitors is given in paragraph 11.4.2.2.2 and tabulated in table 11.4-3.

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15.4 CONDITION IV -- LIMITING FAULTS

Unless otherwise indicated, the applicable data for this section are presented in RESAR-3, section 15.4. The information for the four-loop plant is applicable to the VNP.

NOTE: The following paragraph should be read in place of the first paragraph on page 15.4-23 of RESAR-3.

Figure 15.4-29 shows the required total peaking factor at the license application power rating to meet the AEC Interim Acceptance Criteria for ECCS as a function of calculated peak containment pressure for a double ended cold leg break. The calculated peak containment pressure for the double ended cold leg break is reported in section 6.2.1. Using this pressure in figure 15.4-29, the maximum allowable linear power and total peaking factor at a core power of 3411 MWt for which the ECCS will meet the AEC Interim Acceptance Criteria can be obtained. Note that the peaking factor in figure 15.4-29 is based on ECCS analysis at containment pressure values of 90 percent of the calculated peak containment pressure for blowdown and 80 percent of the peak for reflood, as specified by the AEC Interim Policy Statement.

NOTE: The first paragraph in RESAR-3, page 15.4-42 should read as follows:

Core Power and Reactor Coolant System Transient

Figure 15.4-37 shows the reactor coolant system transient and core heat flux following a main steam pipe rupture (complete severance of a pipe) outside the containment, downstream of the flow measuring nozzle at initial no load condition (case a). The break assumed is the largest break which can occur anywhere outside the containment either upstream or downstream of the isolation valves. Offsite power is assumed available such that full reactor coolant flow exists. The transient shown assumes a steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by high differential pressure between any steam line and the remaining steam lines or by high steam flow signals in coincidence with either low-low reactor coolant system temperature or low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the bi-directional isolation valves in the steam lines by the high steam flow signals in coincidence with either low reactor coolant system temperature or low steam line pressure. Even with the failure of one valve, release is limited to no more than 5 seconds for

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the other steam generators while the one generator blows down. The steam line isolation valves are designed to be fully closed in less than 5 seconds.

The following paragraphs on environmental consequences of postulated accidents are added to RESAR-3, section 15.4.

15.4.1.3 Radiological Consequences of a Postulated Loss Of Coolant Accident (See Subsection 6.2.1.3).

The results of analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss of coolant accident do not result in doses which exceed the guideline values specified in 10 CFR 100.

The analyses performed is based on Regulatory Guide 1.4, Revision 2. The parameters used for this analysis are listed in table 15.4-11. In addition, an evaluation of the offsite dose resulting from purging the containment for hydrogen control, and an evaluation of the offsite doses resulting from recirculation loop leakage, are presented in subsection 15.4.1.3.9.

15.4.1.3.1 Fission Product Release to the Containment

The calculation of potential offsite doses resulting from a
S6 loss of coolant accident are based on the conservative fission product releases recommended by Regulatory Guide 1.4.

One hundred percent of the core noble gas inventory and 25 percent of the core iodine inventory is assumed to be immediately available for leakage from the primary containment. Ninety one percent of the halogen activity available for release is assumed to be in elemental form, 4 percent in methyl form, and 5 percent in particulate form. The total core noble gas and iodine inventories are given in table 15.4-11A, while the activity in the containment atmosphere immediately following the LOCA and available for leakage is shown in table 15.4-11B.

15.4.1.3.2 Containment Model

The activity released from the containment was calculated with a two-volume model to represent sprayed and unsprayed regions of the containment.

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The initial instantaneous release of fission products to the containment is assumed to be the only source of activity in the containment. The activity change with respect to time in each of the two well-mixed volumes is described by the following equations:

$$\frac{da_1}{dt} = -\sum_{j=1}^{k_1} \lambda_{1,j} a_1 - \frac{Q_{12}}{V_1} a_1 + \frac{Q_{21}}{V_2} a_2 \qquad (15.4\text{-}1)$$

$$\frac{da_2}{dt} = -\sum_{j=1}^{k_2} \lambda_{2,j} a_2 - \frac{Q_{21}}{V_2} a_2 + \frac{Q_{12}}{V_1} a_1 \qquad (15.4\text{-}2)$$

where:

a_1, a_2 are the fission product activities of a species in volumes 1 and 2, respectively (Curies) S6

Q_{12}, Q_{21} are the transfer rates between the two volumes (ft^3/hr)

V_1, V_2 are the volumes of the unsprayed and sprayed regions of the containment, respectively (ft^3)

$\lambda_{1,j}$ $\lambda_{2,j}$ are the removal coefficients due to the jth removal process in volume 1 and 2, respectively (hrs^{-1})

k_1, k_2 are the number of removal processes applicable in volumes 1 and 2, respectively

The transfer rate between the sprayed and unsprayed regions was assumed to be limited to the forced convection induced by the fan-cooler units. The flow rate per fan-cooler unit, and the number of units assumed in operation are summarized in table 15.4-11. This assumed minimum flow rate conservatively neglects the effect of natural convection, steam condensation, and diffusion although these effects are expected to enhance the mixing rate between the sprayed and unsprayed volumes.

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15.4.1.3.3 Modeling of Removal Process

For fission products other than iodine, the removal processes considered are radioactive decay and leakage from the containment. The decay constants used in the calculations are listed in appendix 15B.

The fission product iodine is assumed to be present in the containment atmosphere in elemental, organic, and particulate form. It is assumed that 91 percent of the iodine available for leakage from the containment is in elemental (i.e., I_2 vapor) form, 4 percent is assumed to be in the form of organic iodine compounds (e.g., methyl iodide), and 5 percent is assumed to be adsorbed on airborne particulate matter. For this analysis it was conservatively assumed to subject particulates to a removal rate of 0.5 hr^{-1} until the particulate iodine in the containment is reduced by a factor of 100 and to subject organic iodines to no removal process other than radioactive decay and leakage from the containment.

S6 The effectiveness of the containment spray for the removal of the elemental form of iodine, and the model used to determine the iodine removal efficiency for this application are described in subsection 6.2.3.3.2 and appendix 6A. The spray iodine removal process is considered only in the upper containment volume, which corresponds to the sprayed regions of the containment. A spray removal rate of 20.5 hr^{-1} is assumed until the airborne elemental iodine in the containment is reduced by a factor of 100. After this time, the spray removal rate is assumed to be zero.

The modeling of the unsprayed region (lower containment volume) as a separate volume eliminates the need to adjust the spray rate constant (λs) for the effect of the unsprayed regions.

15.4.1.3.4 Containment Leak Rate

The primary containment leak rate used in this analysis is the design-basis leak rate specified in the Technical Specifications. For the first 24 hours following the accident, the leak rate was assumed to be 0.3 percent per day and the leak rate was assumed to be 0.15 percent per day for the remainder of the 30-day period.

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15.4.1.3.5 Penetration Rooms and Associated Safety Grade Filtration and Exhaust Systems

The majority (about 85 percent) of penetrations through the containment wall occur within the electrical and piping penetration rooms. These rooms are contained within the fuel handling building, auxiliary building, and control building and are serviced by the safety-grade electrical and piping penetration room filtration and exhaust systems, and the rooms are at a negative pressure when the filtration systems are in operation. All ECCS piping which potentially recirculates contaminated fluids following an accident are completely routed within the containment or the piping penetration room. Any airborne radioactivity released as a result of leakage from these piping systems outside the containment will be collected and filtered prior to release to the environment. Changing the design from the enclosure building to the equipment building and deletion of the enclosure building filtration and exhaust system does not alter the capability to filter potential airborne radioactivity due to leakage or transfer of post - accident containment fluids. S6

In the calculation of offsite doses, no credit has been taken for filtration of airborne containment leakage even though approximately 85 percent of the containment penetrations are within the negative pressure and filtration boundary. Thus, the majority of the most likely containment liner leak locations are situated in such a manner that filtration of potential airborne radioactivity occurs before release.

15.4.1.3.6 Deleted

15.4.1.3.7 Deleted

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Table 15.4-11

PARAMETERS USED IN LOCA ANALYSES

	REGULATORY GUIDE 1.4 ANALYSIS
Core thermal power	3565 MWt
Containment free volume	2.75×10^6 ft^3
Sprayed containment free volume	2.145×10^6 ft^3
Unsprayed containment free volume	6.05×10^5 ft^3
Primary containment deck fan flow rate	85,000 cfm
Number of deck fans assumed operating	4 of 8
Activity released to containment and available for release	
Noble gases	100% of core inventory
*Iodines	25% of core inventory
Form of iodine activity in primary containment available for release	
Element iodine	91%
Methyl iodine	4%
Particulate iodine	5%
Primary containment leak rate	0.30% per day (0-24 hours)
	0.15% per day (1-30 days)
Containment spray removal	
Elemental iodine	20.5 hr^{-1}
Particulates	0.5 hr^{-1}
Meteorology	See Table 15.B-2

*The activity available for release reflects the assumption of 50 percent plateout on containment surfaces.

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Table 15.4-11A

IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE

ISOTOPE	CORE ACTIVITY (Curies)
I-131	8.80×10^7
I-132	1.34×10^8
I-133	1.97×10^8
I-134	2.31×10^8
I-135	1.79×10^8
Xe-131m	6.68×10^5
Xe-133	2.03×10^8
Xe-133m	5.16×10^6
Xe-135	5.55×10^7
Xe-135m	5.46×10^7
Xe-138	1.79×10^8
Kr-83m	1.64×10^7
Kr-85	9.99×10^5
Kr-85m	3.95×10^7
Kr-87	7.59×10^7
Kr-88	1.08×10^8
Kr-89	1.40×10^8

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Table 15.4-11B

NOBLE GAS AND IODINE INVENTORY IN THE CONTAINMENT ATMOSPHERE IMMEDIATELY AFTER LOCA AND AVAILABLE FOR LEAKAGE

ACTIVITY (Ci)

ISOTOPE	REGULATORY GUIDE 1.4 ANALYSIS
I-131	2.20×10^7
I-132	3.35×10^7
I-133	4.93×10^7
I-134	5.78×10^7
I-135	4.48×10^7
Xe-131m	6.68×10^5
Xe-133	2.03×10^8
Xe-133m	5.16×10^6
Xe-135	5.55×10^7
Xe-135m	5.46×10^7
Xe-138	1.79×10^8
Kr-83m	1.64×10^7
Kr-85	9.99×10^5
Kr-85m	3.95×10^7
Kr-87	7.59×10^7
Kr-88	1.08×10^8
Kr-89	1.10×10^8

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15.4.1.3.8 Results

S6 The iodine and noble gas activity releases to atmosphere for the Regulatory Guide 1.4 analysis are given in table 15.4-11C. The gamma and thyroid doses for the loss of coolant accident at the site boundary and the low population zone are given in table 15.4-12. The doses are based on the atmospheric dilution factors and dose models given in Appendix 15B. The dose limits for this accident are defined in 10 CFR 100 (25 rem whole body and 300 rem thyroid). Doses for the Regulatory Guide 1.4 analysis are well within the 10 CFR 100 guidelines.

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Table 15.4-11C

ACTIVITY RELEASES TO ATMOSPHERE FROM LOSS OF COOLANT ACCIDENT

Regulatory Guide 1.4 Analysis Activity Release (Ci)

ISOTOPE	0-2 HR	2-8 HR	8-24 HR	1-4 DAYS	4-30 DAYS
I-131	7.59(+2)	9.59(+2)	2.09(+3)	3.99(+3)	1.19(+4)
I-132	9.76(+2)	4.01(+2)	6.6(+1)	2.80(-1)	1.4(-10)
I-133	1.67(+3)	1.87(+3)	2.96(+3)	1.91(+3)	1.93(+2)
I-134	1.35(+3)	1.22(+2)	8.30(-1)	1.2(-6)	2.1(-31)
I-135	1.45(+3)	1.22(+3)	9.72(+2)	1.14(+2)	6.8(-2)
Xe-133	5.06(+4)	1.48(+5)	3.72(+5)	6.60(+5)	1.31(+6)
Xe-133m	1.28(+3)	3.64(+3)	8.43(+3)	1.11(+4)	7.30(+3)
Xe-135	1.28(+4)	2.86(+4)	3.50(+4)	7.40(+3)	3.22(+1)
Xe-135m	2.56(+3)	1.24(+1)	1.39(-6)	2.00(-28)	NEGLIGIBLE
Xe-138	9.07(+3)	6.80(+1)	2.80(-5)	1.30(-22)	NEGLIBIBLE
Kr-85	4.22(+2)	1.26(+3)	3.37(+3)	7.55(+3)	6.39(+4)
Kr-85m	1.43(+4)	2.34(+4)	1.35(+4)	5.75(+2)	6.10(-3)
Kr-87	1.94(+4)	9.43(+3)	3.68(+2)	2.90(-2)	2.27(-19)
Kr-88	3.60(+4)	4.30(+4)	1.21(+4)	1.13(+2)	1.71(-6)

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Note: 6.21(-5) = 6.21×10^{-5}

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Table 15.4-12

POTENTIAL OFFSITE DOSES DUE TO ACCIDENTS
VOGTLE NUCLEAR PLANT

		DOSE (2 HOURS) AT EXCLUSION AREA BOUNDARY (1060 Meters)		DOSE (COURSE OF ACCIDENT) AT LOW POPULATION ZONE (3218) meters)	
Postulated	PSAR Section	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
LOCA	15.4				
Containment Leak					
Conservative		93.3	7.43(+0)	103	5.47(+0)

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APPENDIX 15B

DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15B.1 INTRODUCTION

This section identifies the models used to calculate the off-site radiological doses that would result from releases of radioactivity due to various postulated accidents.

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15B.2 ASSUMPTIONS

The following assumptions are basic to both the model for the
gamma doses due to immersion in a semi-infinite cloud of radio- S6
activity, and the model for the thyroid dose due to inhalation
of radioiodine: S6

a. Direct radiation from the source point is negligible compared to gamma radiation due to immersion in the semi-infinite radioactive cloud. S6

b. All radioactivity releases are treated as ground level releases regardless of the point of discharge.

c. The dose receptor is a standard man, as defined by the International Commission on Radiological Protection (ICRP) (Reference 1). S6

d. Radioactive decay from the point of release to the dose receptor is neglected.

e. Isotopic data such as decay rates and decay energy emissions are taken from Reference 2.

15B.3 GAMMA DOSE

The gamma dose delivered to a dose receptor is obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane, i.e., an "infinite semi-spherical cloud." The concentration of radioactive material within this cloud is taken to be uniform, and equal to the maximum centerline ground level concentration that would exist in the cloud at the appropriate distance from the point of release.

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The gamma dose is a result of external gamma radiation. Equations describing a semi-infinite cloud were used to calculate the doses for a given time period as follows: (Reference 3).

$$\text{Gamma Dose} = 0.25 \; \frac{\chi}{Q} \cdot \Sigma A_{R_i} \cdot \overline{E}_{\gamma_i} \qquad (15B\text{-}1)$$

where:

S6 A_{R_i} is the activity of isotope i released during a given time period (Curies)

$\frac{\chi}{Q}$ is the atmospheric dilution factor for a given time period (table 15B-2) (sec/m^3)

$\overline{E}_{\gamma_i}$ is the average gamma radiation energy emitted by isotope i per disintegration (meV/disintegration)

15B.4 THYROID INHALATION DOSE

The thyroid dose for a given time period t is obtained from the following expression: (Reference 4)

$$D = \chi/Q \cdot B \cdot \Sigma Q_i \cdot DCF_i \qquad (15B\text{-}2)$$

where:

D = thyroid inhalation dose, rem

$(\chi/Q)_t$ = site dispersion factor for time interval t, sec/m^3

B = brea[illegible]g rate for time interval t, m^3/sec

Q_i = total activity of iodine isotope i released in time period t, Curies

$(DCF)_i$ = dose conversion factor for iodine isotope i, rem/Curie inhaled

S6 The isotopic data and "standard man" data are given in table 15B-1. The gamma energies, E_γ, on table 15B-1 include the X-rays and annihilation gamma rays if they are prominent in the electromagnetic spectrum.

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15B.5 REFERENCES

1. "Report of ICRP Committee II on Permissive Dose for Internal Radiation (1959)," Health Physics Volume 3, pp. 30, 146-153, 1970. S6

2. Lederer, C. M., Hollander, J. M., Perlman, I. "Table of Isotopes, Sixth Edition".

3. Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," USNRC, June 1974. S6

4. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," USNRC, October 1977.

Table 15B-1

PHYSICAL DATA FOR ISOTOPES

ISOTOPE	DECAY CONSTANT** (hr-1)	GAMMA ENERGY** (Mev/Disint.)	DOSE CONVERSION FACTOR* (rem/Curie)
I-131	3.587×10^{-3}	0.371	1.49×10^{6}
I-132	3.01×10^{-1}	2.400	1.43×10^{4}
I-133	3.33×10^{-2}	0.477	2.69×10^{5}
I-134	7.87×10^{-1}	1.94	3.73×10^{3}
I-135	1.04×10^{-1}	1.78	5.6×10^{4}
Xe-133	5.48×10^{-3}	0.03	-
Xe-133m	1.28×10^{-2}	0.0326	-
Xe-135	7.53×10^{-2}	0.246	-
Xe-135m	2.67×10^{0}	0.422	-
Xe-138	2.45×10^{0}	2.87	-
Kr-85	7.35×10^{-6}	0.00211	-
Kr-85m	1.59×10^{-1}	0.151	-
Kr-87	5.47×10^{-1}	1.37	-
Kr-88	2.50×10^{-1}	1.74	-

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BREATHING RATES

Time Period (Hours)	Breathing Rates (m^3/sec)
0 - 8	3.47×10^{-4}
8 - 24	1.75×10^{-4}
24 - 720	2.32×10^{-4}

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* Reference 4

** Reference 2

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Table 15B-2

ACCIDENT ATMOSPHERIC DILUTION FACTORS (X/Q)* AT EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE FOR THE VOGTLE NUCLEAR PLANT

TIME PERIOD (hours)	EXCLUSION AREA BOUNDARY (1060 Meters)	LOW POPULATION ZONE (3218 Meters)
0-2	1.6(-4)	6.3(-5)
0-8		3.2(-5)
8-24		2.2(-5)
24-96		1.0(-5)
96-720		3.3(-6)

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* X/Q values, expressed in sec/m^3, are based on the guidelines set forth in the August 1978 draft of Regulatory Guide 1.XXX. The site data used was taken during the recent year (1977-1978).

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NOTE: 1.6(-4) = 1.6×10^{-4}

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Table 15B-2

ACCIDENT ATMOSPHERIC DILUTION FACTORS (X/Q)* AT EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE FOR THE VOGTLE NUCLEAR PLANT

TIME PERIOD (hours)	EXCLUSION AREA BOUNDARY (1060 Meters)	LOW POPULATION ZONE (3218 Meters)
0-2	1.6(-4)	6.3(-5)
0-8		3.2(-5)
8-24		2.2(-5)
24-96		1.0(-5)
96-720		3.3(-6)

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* X/Q values, expressed in sec/m^3, are based on the guidelines set forth in the August 1978 draft of Regulatory Guide 1.XXX. The site data used was taken during the recent year (1977-1978).

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NOTE: 1.6(-4) = 1.6×10^{-4}

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5. One residual heat exchanger may be out of service provided that it is restored to operable status within (FSAR) days. If one pump is out of service per the previous paragraph, it must be the one in the same train as the out-of-service heat exchanger.

6. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to operable status within (FSAR) hours and all valves in the system that provide duplicate function are demonstrated to be operable.

7. During normal operation, one boron injection tank heater channel and/or one channel of line heat tracing may be inoperable for an extended period.

3.3.3.2 Containment Cooling and Iodine Removal Systems

A. The reactor shall not be made critical, except for low power physics tests, unless the following conditions are met:

1. The spray additive tank contains not less than 3200 gallons of solution with hydroxide concentration of not less than 30 percent by weight.

2. Two containment spray trains, including containment spray pumps, piping, and valves shall be operable.

3. Four fan cooler units including all essential valves, controls, dampers, and piping associated with these units shall be operable.

B. During power operation the requirements of specification 3.3.3.2.1 may be modified to allow the following components to be inoperable. If the components are not restored to meet the requirements of specification 3.3.3.2.1 within the time period specified below, the reactor shall be placed in the hot shutdown condition. If the requirements of specification 3.3.3.2.1 are not satisfied within an additional 48 hours, the reactor shall be placed in

the cold shutdown condition using normal operating procedures.

1. One containment spray train may be out of service, provided immediate attention is directed to making repairs and the train can be restored to operable status within (FSAR) hours. The other containment spray train shall be tested as specified in specification 4.5 to demonstrate operability prior to initiating repair of the inoperable system, or

2. One containment fan cooler unit may be out of service, provided immediate attention is directed to making repairs and the fan cooler unit can be restored to operable status within (FSAR) hours. Two of the remaining fan cooler units shall be tested as specified in specification 4.5 to demonstrate operability prior to initiating repair of the inoperable unit.

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2. Acceptable Criteria

a. System tests will be considered satisfactory if visual observations indicate all components of each system have functioned properly.

b. Containment spray pump tests will be considered satisfactory at miniflow condition if the pumps reach their maximum required flow. The discharge pressure and corresponding flow rate determine a point on the head curve; and the pumps operate for at least fifteen minutes.

c. A test of a motor operated valve shall be considered satisfactory if its limit switch operates a light on the main control board demonstrating that the valve has stroked.

d. The test of the containment spray nozzles shall be considered satisfactory if air flow or smoke through the nozzles indicates that the nozzles are not plugged.

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E. Penetration Room Filtration System

System tests shall be made at (FSAR) intervals while the unit is operating. These tests shall consist of visual inspection, a flow measurement using the pitot tube installed at the outlet of each filter subsystem, and pressure drop measurements across each filter bank. Visual inspection shall include inspection of general conditions for evidence of: water, oil, or other foreign material; gasket deterioration; and adhesive deterioration in the HEPA units. These tests will be performed within (FSAR) hours after startup from cold shutdown.

The test will be considered satisfactory if visual inspection reveals no abnormal conditions, flow is equal to design flow or higher, no unusual or excessive noise or vibration exists when the fan motors are operating, and the pressure drop across any filter bank does not exceed two times the pressure drop when new.

4.5.3.2 Component Tests

A. Pumps

1. Safety injection pumps, residual heat removal pumps, containment spray pumps, and the auxiliary component cooling water pumps shall be started at intervals not greater than (FSAR).

2. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least (FSAR) minutes.

B. Valves

1. Each boron injection tank outlet valve shall be cycled by operator action with the pumps shut down at intervals not greater than once every refueling.

2. Each spray additive valve shall be cycled by operator action with the pumps shut down at intervals not greater than once every refueling.

5.2 CONTAINMENT

5.2.1 CONTAINMENT

5.2.1.1 Shielding S6

The containment completely encloses the reactor and reactor coolant system and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the reactor coolant system occurs. The structure provides biological shielding for both normal and accident situations.

5.2.1.2 Internal Pressure, Temperature, and Vacuum

The containment is designed for an internal pressure of 47 psig and an internal temperature of 271 F. The containment is also structurally designed to withstand an internal vacuum of 3.0 psig.

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5.2.3 PENETRATION ROOM

The penetration room filtration system collects, controls, and minimizes the release of radioactive materials from the containment to the environment following a design basis accident. Air from the penetration rooms is drawn through the filter subsystem consisting of prefilters, high efficiency particulate filters, and charcoal filters in service. The air is then discharged through the vent. The system provides for enhanced dispersion of fission products by mixing of containment penetration leakage with all areas of the penetration

rooms, increased holdup time within the penetration rooms, improved cleanup of fission products by filtration, and a slightly negative pressure in the penetration rooms to insure inleakage.

5.2.4 PENETRATIONS

5.2.4.1 Seals

Penetration assemblies are seal welded to the containment liner. Access openings, electrical penetration canisters, and fuel transfer tube covers are equipped with double seals. Containment purge penetrations and containment atmosphere sampling penetrations are equipped with double valves having resilient seating surface.

5.2.4.2 Leakage Barriers

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is minimized by a double barrier so that no single credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the containment, and various types of isolation valves.

5.2.5 POST LOCA CONTAINMENT PRESSURE SUPPRESSION SYSTEMS

5.2.5.1 Internal Spray Systems

The containment has redundant internal spray systems capable of providing a distributed borated water spray of at least 2,600 gal./min. During the initial period of spray operation, sodium hydroxide is added to the spray water to increase the removal of iodine from the containment atmosphere.

5.2.5.2 Cooling Systems

The containment has redundant containment cooling systems, which consist of four containment air coolers (each consisting of a fan and a water-cooled heat exchanger) with total heat removal capability of twice the design requirement under conditions following LOCA.