TABLE OF CONTENTS

CHAPTER 1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

CHAPTER 1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This updated Final Safety Analysis Report (FSAR) complies with the Standard Format and Content of Safety Analysis Reports (Revision 2) issued by the Nuclear Regulatory Commission (NRC) in September 1975 and 10CFR50.71(e). The original FSAR, as amended, is considered to be the licensing basis for the plant. The updated FSAR will be the reference document for purposes of communications with the NRC such as reporting of deviations from conditions as stated in the FSAR and for evaluations requiring the FSAR, such as 10CFR50.59. Reference to the "FSAR" by this document, by plant directives, and by pertinent Entergy Operations manuals will be understood to reference the updated FSAR. This approach is consistent with the guidance provided in Generic Letter 81-06 entitled "Periodic Updating of Final Safety Analysis Reports (FSARs)" dated February 26, 1981.

A discussion of the format of the updated FSAR is presented in subsection 1.1.9.

1.1.1 Type of License Required

[HISTORICAL INFORMATION] [The original FSAR was submitted in support of the application of Mississippi Power & Light Company** for a license to operate a two-unit nuclear power facility at a core thermal power level of 3833 MWt, each, the power level equivalent to 100 percent of the design steam flow. This application was submitted under Section 103 (b) of the Atomic Energy Act of 1954, as amended, and the regulation of the Nuclear Regulatory Commission set forth in Part 50 of Title 10 to the Code of Federal Regulations (10CFR50).

In December of 1979 construction of Grand Gulf Unit 2 (NRC Docket Number 50-417) was deferred in order to concentrate resources on the completion of Unit 1. After Unit 1 had received its Commercial Operating License, Entergy Operations, Inc. formally requested the NRC to revoke the Construction Permit and officially cancel the second unit at the Grand Gulf Nuclear Station. The Construction Permit for Grand Gulf Unit 2 was formally revoked by the NRC in August 1991.]

1.1.2 Identification of Applicant

[HISTORICAL INFORMATION] [The Grand Gulf Nuclear Station is owned by System Energy Resources, Inc. (SERI*) and South Mississippi Electric Power Association (SMEPA). SMEPA maintains a 10% interest in GGNS and, of its original 90% ownership share, SERI maintains 77.23% ownership interest. The remaining 12.77% interest is owned by equity investors: Textron Financial Corporation and Resources Capital Management Corporation, and is leased back to SERI. SERI and SMEPA pay costs associated with their respective ownership or leased interests. Entergy Operations, Inc. (Entergy Operations) operates GGNS. SERI and Entergy Operations are wholly owned subsidiaries of Entergy Corporation, a registered public utility holding company. Mississippi Power & Light Company (MP&L**) originally assumed responsibility for design, construction, and operation of the facility and acted as an agent for SMEPA. On December 20, 1986, SERI assumed responsibility for the control and performance of licensed activities from MP&L. On June 6, 1990 Entergy Operations assumed responsibility for the control and performance of licensed activities from SERI. As a part of the final transfer, Entergy Operations assumed responsibility for commitments originally made by MP&L and SERI. In those cases in the FSAR where Entergy Operations has either present or future responsibility, reference is made to "Grand Gulf Nuclear Station," "GGNS," or "Entergy Operations", with no mention of MP&L or SERI. In 1996, MP&L changed its name to Entergy Mississippi, Inc., however, to address certain historical information where a reference to Entergy Operations could cause confusion, "MP&L" or "SERI" is used to represent situations where either MP&L or SERI originally had responsibility or made commitments but where Entergy Operations is now responsible. In the cases where Entergy Mississippi, Inc. has responsibility (such as offsite power), references are made to "Entergy Mississippi, Inc".]

* System Energy Resources, Inc. was originally named Middle South Energy, Inc. The name was changed to System Energy Resources, Inc. in 1986.

**Mississippi Power & Light Company (MP&L) changed its name to Entergy Mississippi, Inc. as approved by Amendment 127 to the facility operating license. Historical references to MP&L are contained in the FSAR.

1.1.3 Number of Plant Units

[HISTORICAL INFORMATION] [This application was submitted for both Units 1 and 2 of the Grand Gulf Nuclear Station which were docketed in November 1972 on NRC Docket Numbers 50-416 and 50-417, respectively.

In December of 1979 construction of Grand Gulf Unit 2 (NRC Docket Number 50-417) was deferred in order to concentrate resources on the completion of Unit 1. After Unit 1 had received its Commercial Operating License, Entergy Operations, Inc. formally requested the NRC to revoke the Construction Permit and officially cancel the second unit at the Grand Gulf Nuclear Station. The Construction Permit for Grand Gulf Unit 2 was formally revoked by the NRC in August 1991.]

1.1.4 Description of Location

[HISTORICAL INFORMATION] [The facility is located in Claiborne County, Mississippi, on the east side of the Mississippi River approximately 25 miles south of Vicksburg and 37 miles north-northeast of Natchez, Mississippi.]

1.1.5 Type of Nuclear Steam Supply System

[HISTORICAL INFORMATION] [Grand Gulf has a BWR-6 boiling water reactor (251-inch vessel with 800 fuel assemblies) designed and supplied by General Electric Company.]

1.1.6 Type of Containment

[HISTORICAL INFORMATION] [The Grand Gulf containment is the Mark III BWR containment incorporating the drywell/pressure suppression concept. The containment is a steel-lined reinforced concrete structure designed by Bechtel Power Corporation.]

1.1.7 Core Thermal Power Levels

The information presented in this updated FSAR pertains to the Grand Gulf reactor with a rated power level of 4408 Mwt. This power level represents a 15% increase from the original license of 3833 Mwt. The station utilizes a single-cycle forced circulation boiling water reactor (BWR) provided by General Electric-Hitachi (GEH). The heat balance for rated power is shown in Figure 1.1-1. The station is designed to operate at a gross electrical power output of approximately 1523.5 MWe.

1.1.8 Scheduled Completion and Operation Dates

[HISTORICAL INFORMATION] [The fuel loading for Unit 1 was completed in August 1982. Commercial operation for Unit 1 was declared on July 1, 1985. Construction of Unit 2 was deferred in December 1979 in order to concentrate resources on completion of Unit 1. After completion of Unit 1, Entergy Operations, Inc. formally requested the NRC to revoke the Unit 2 Construction Permit (NRC Docket Number 50-417). The Unit 2 Construction Permit was revoked in August 1991.]

1.1.9 Organization of Contents

1.1.9.1 Subdivisions

The updated FSAR is organized into 18 chapters, each of which consists of a number of sections that are numerically identified by two numerals separated by a decimal (e.g., 3.4 is the fourth section of Chapter 3). Further subdivisions are referred to as subsections.

1.1.9.2 Standard Format

The updated FSAR has been written to comply with the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 2) as issued by the Nuclear Regulatory Commission in October 1975. The updated FSAR uses the same chapter, section, and subsection headings as those used in the standard format except in cases where this format is not applicable to plant design. Where appropriate, the updated FSAR is subdivided beyond the extent of the standard format to isolate all information specifically requested in that document. Where information has been presented that is not specifically requested by the standard format and this information is identified numerically (chapter, section, or subsection), this information is presented under the appropriate general headings as a subdivision containing information specifically requested by the standard format. (For example, subsection 1.1.9 is not requested in the standard format. Since it apparently belonged in Section 1.1, it was placed after the eight subsections containing information requested by the standard format).

1.1.9.3 References

References to another location in the updated FSAR are made by chapter or section number. References to another document are indicated by the notation (Ref. 1). The reference section is located at the end of the applicable text and before any tables in the section.

1.1.9.4 Tables and Figures

Tabulations of data are designated as "tables." They are identified by the section number, followed by a number according to its order of mention in the section (e.g., Table 3.3-5 is the fifth table of Section 3.3). Tables are located at the end of the applicable section. Drawings, sketches, curves, graphs, and engineering diagrams are all identified as "figures" and are numbered according to the order of mention in the section (Figure 3.4-2 is the second figure of Section 3.4). Figures are located at the end of the applicable section.

1.1.9.5 Numbering of Pages

Pages are numbered sequentially within each section. For example, 1.1-2 is the second page of Section 1.1. When it becomes necessary during revision of this updated FSAR to insert a page(s) between two existing pages within a section, letters will be used (for example, to insert two pages between 3.2-4 and 3.2-5, the following page sequence would appear: 3.2-4, 3.2-4a, 3.2-4b, 3.2-5).

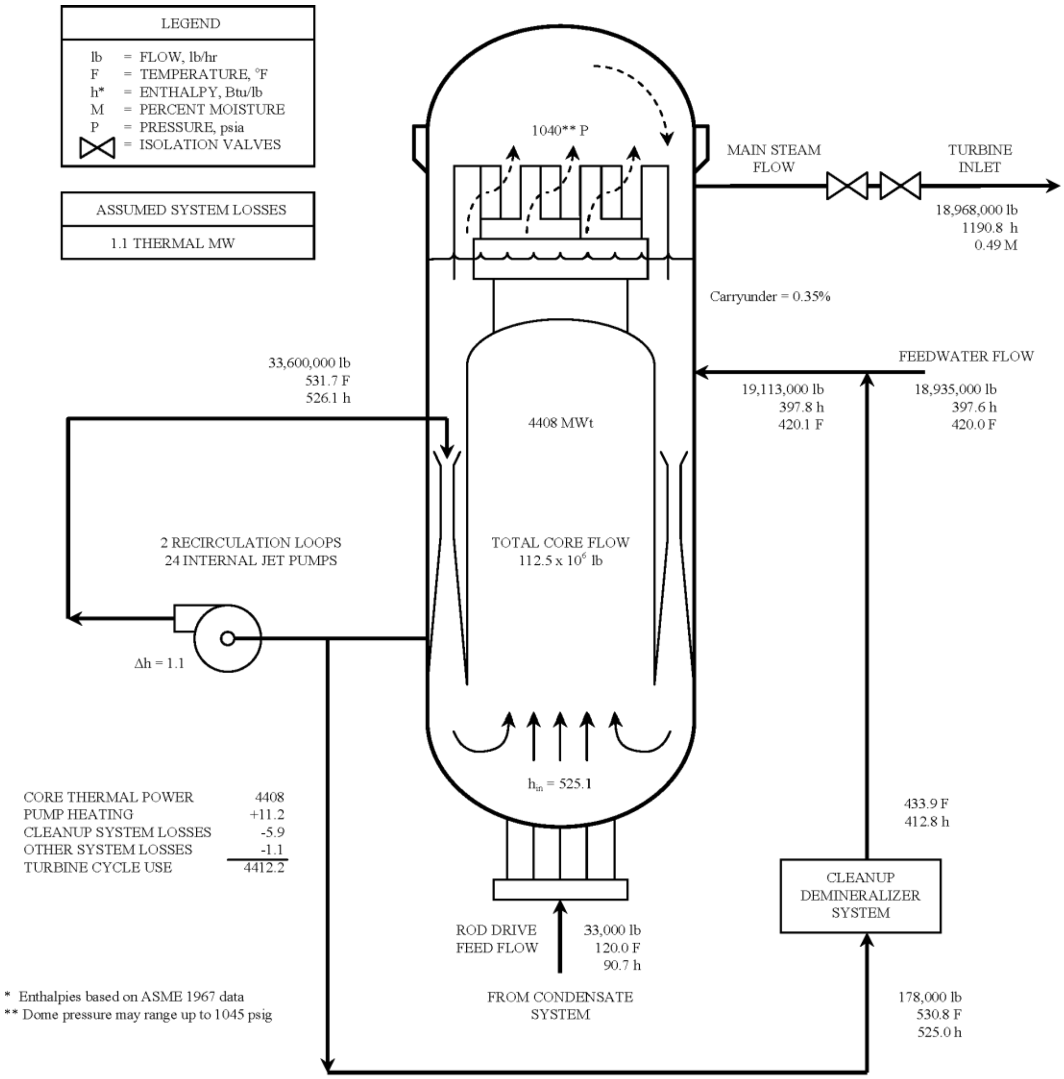
1.1.9.6 Revising the Updated FSAR

When it becomes necessary to submit additional information or to revise information presently contained in the updated FSAR, the following procedures will be followed:

a. When a change is made to the updated FSAR text, those pages affected will be marked with the page change identification (date of revision or change number or both) and a change indicator (e.g. vertical line) drawn in the margin adjacent to the portion actually changed. Further revising of previously revised sections will delete the original labeled vertical change bar if the entire portion is revised.

b. Figures will be revised by indicating the page change identification (date of revision or change number or both) on the Figure.

c. Revisions containing updated information shall be submitted on a replacement-page basis and shall be accompanied by a list which identifies the current pages of the FSAR following page replacement.



GRAND GULF NUCLEAR STATION UNIT 1 UPDATED FINAL SAFETY ANALYSIS REPORT	HEAT BALANCE AT RATED POWER AND CORE FLOW FIGURE 1.1-1

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Principal Design Criteria

The principal design criteria are presented in two ways. First, they are classified as either a power generation function or a safety function. Second, they are grouped according to system. Although the distinctions between power generation or safety functions are not always clear cut and are sometimes overlapping, the functional classification facilitates safety analyses, while the grouping by system facilitates the understanding of both the system function and design.

1.2.1.1 General Design Criteria

1.2.1.1.1 Power Generation Design Criteria

a. The station is designed to produce steam for direct use in a turbine-generator unit.

b. Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients.

c. Backup heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

d. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity throughout the range of normal operational conditions and abnormal operational transients.

e. The fuel cladding accommodates, without loss of integrity, the pressures generated by fission gases released from fuel material throughout the design life of the fuel.

f. Control equipment is provided to allow the reactor to respond automatically to minor load changes, major load changes, and abnormal operational transients.

g. Reactor power level is manually controllable.

h. Control of the reactor is possible from a single location.

i. Reactor controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions.

j. Interlocks or other automatic equipment are provided as backup to procedural controls to avoid conditions requiring the functioning of nuclear safety systems or engineering safety features.

1.2.1.1.2 Safety Design Criteria

a. The station is designed, fabricated, constructed, and operated in such a way that the normal release of radioactive materials to the environment is significantly less than the requirements of 10 CFR 20.

b. The station is designed, fabricated, erected, and operated in such a way that the release of radioactive materials to the environment resulting from abnormal transients and accidents is less than the requirements of 10 CFR 100, 10 CFR 50.67 and 10 CFR 50 GDC 19.

c. The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.

d. The reactor is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the reactor with other appropriate plant systems.

e. Gaseous, liquid, and solid waste disposal facilities are designed so the discharge of radioactive effluents and offsite shipment of radioactive materials can be made in accordance with applicable regulations.

f. The design provides means by which plant operators are alerted when limits on the release of radioactive material are approached.

g. Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered by plant safety analysis.

h. Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operations.

i. Those portions of the nuclear system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material containment barrier following abnormal operational transients and accidents.

j. Nuclear safety systems function to assure that no damage to the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients and accidents.

k. Where positive, precise action is immediately required in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel.

l. Essential safety actions are provided by equipment of sufficient redundance and independence that no single failure of active components can prevent the required actions. For systems or components to which IEEE 2791971, "Criteria for Protection Systems for Nuclear Power Generating Stations," applies, single failures of both active and passive electrical components are considered in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

m. Provisions are made for control of active components of nuclear safety systems and engineered safety features from the control room.

n. Nuclear safety systems and engineered safety features are designed to permit demonstration of their functional performance requirements. The ability and the extent that systems can be tested during operation is discussed further in each individual system subsection.

o. The design of nuclear safety systems and engineered safety features includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site.

p. Standby electrical power sources have sufficient capacity to power all nuclear safety systems and engineered safety features requiring electrical power.

q. Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

r. A containment is provided that completely encloses the reactor system, drywell, and suppression pool. The containment employs the pressure suppression concept.

s. It is possible to test primary containment integrity and leak tightness at periodic intervals.

t. A secondary containment is provided that completely encloses the primary containment. This secondary containment provides a method for controlling the release of radioactive materials from the primary containment.

u. The primary containment and secondary containment, in conjunction with other engineered safety features, limit radiological effects of accidents resulting in the release of radioactive material to the containment volumes to less than the requirements of 10 CFR 100.

v. Provisions are made for removing energy from the primary containment as necessary, to maintain the integrity of the containment system following accidents that release energy to the containment.

w. Piping that penetrates the primary containment and could serve as a path for the uncontrolled release of radioactive material to the environs is isolated whenever such uncontrolled radioactive material release is threatened. Such isolation is effected in time to limit radiological effects to less than the requirements of 10 CFR 100.

x. Emergency core cooling systems are provided to limit fuel cladding temperature to less than that which could cause fragmentation in the event of a loss-of-coolant accident.

y. The emergency core cooling systems provide for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.

z. Operation of the emergency core cooling systems is initiated automatically when required, regardless of the availability of offsite power supplies and the normal generating system of the station.

aa. The control room is shielded against radiation so that continued occupancy under accident conditions is possible.

bb. In the event that the control room becomes inaccessible, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing the local controls and equipment that are available outside the control room.

cc. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any normal operating condition and subsequently to maintain the shutdown condition.

dd. Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel.

1.2.1.2 System Criteria

The principal design criteria for particular systems are listed in the following subsections.

1.2.1.2.1 Nuclear System Criteria

a. The fuel cladding is designed to retain integrity as a radioactive material barrier throughout the design power range. The fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

b. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity throughout any abnormal operational transient.

c. Those portions of the nuclear system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material barrier during normal operation and following abnormal operational transients and accidents.

d. Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational transients. The capacity of such systems is adequate to prevent fuel cladding damage.

e. Heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage. The reactor is capable of being shut down automatically in sufficient time to permit decay heat removal systems to become effective following loss of operation of normal heat removal systems.

f. The reactor core and reactivity control system are designed so that control rod action is capable of bringing the core subcritical and maintaining it so, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

g. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.

h. The nuclear system is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

1.2.1.2.2 Power Conversion Systems Criteria

Components of the power conversion systems are designed to perform the following basic objectives.

a. Produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater, with a major portion of its gases and particulate impurities removed.

b. Assure that any fission products or radioactivity associated with the steam and condensate during normal operation are safely contained inside the system or are released under controlled conditions in accordance with waste disposal procedures.

1.2.1.2.3 Electrical Power Systems Criteria

Sufficient normal and standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances. The power sources are adequate to accomplish all required essential safety actions under postulated design-bases accident conditions.

1.2.1.2.4 Radwaste System Criteria

a. The gaseous and liquid radwaste systems are designed to minimize the release of radioactive effluents from the station to the environs. Such releases as may be necessary during normal operations are limited to values that meet the requirements of applicable regulations including 10 CFR 20 and 10 CFR 50.

b. The solid radwaste disposal systems are designed so that inplant processing and offsite shipments are in accordance with all applicable regulations, including 10 CFR 20, 10 CFR 71, and 49 CFR 171 through 179, as appropriate.

c. The system's design provides means by which station operations personnel are alerted whenever specified limits on the release of radioactive material may be approached.

1.2.1.2.5 Auxiliary Systems Criteria

a. Fuel handling and storage facilities are designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel. Provisions are made for maintaining the cleanliness of spent fuel cooling and shielding water.

b. Auxiliary systems which are required for safe shutdown or to mitigate the consequences of an accident are designed to function during normal and/or accident conditions.

c. Auxiliary systems that are not required to effect safe shutdown of the reactor or maintain it in a safe condition are designed such that a failure of these systems shall not prevent the essential auxiliary systems from performing their design functions.

1.2.1.2.6 Nuclear Safety Systems and Engineered Safety Features Criteria

Principal design criteria for nuclear safety systems and engineered safety features are as follows:

a. These criteria correspond to criteria j through q, x through z, bb and cc in subsection 1.2.1.1.2.

b. Standby electrical power sources have sufficient capacity to power engineered safety features requiring electrical power.

c. Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

d. In the event that the control room is inaccessible, it is possible to bring the reactor from power range operation to a cold shutdown condition by manipulating controls and equipment that are available outside the control room.

e. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any operating condition and subsequently to maintain the shutdown condition.

1.2.1.2.7 Process Control Systems Criteria

The principal design criteria for the process control systems are discussed in this subsection.

1.2.1.2.7.1 Nuclear System Process Control Criteria

a. Control equipment is provided to allow the reactor to respond automatically to load changes within design limits.

b. It is possible to control the reactor power level manually.

c. Control of the nuclear system is possible from a central location.

d. Nuclear systems process controls and alarms are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

e. Interlocks or other automatic equipment are provided as a backup to procedural controls to avoid conditions requiring the actuation of engineered safety features.

1.2.1.2.7.2 Power Conversion Systems Process Control Criteria

a. Control equipment is provided to control the reactor pressure throughout its operating range.

b. The turbine is able to respond automatically to minor changes in load.

c. Control equipment in the feedwater system maintains the water level in the reactor vessel at the optimum level required by steam separators.

d. Control of the power conversion equipment is possible from a central location.

e. Interlocks or other automatic equipment are provided in addition to procedural controls to avoid conditions requiring the actuation of engineered safety features.

1.2.1.2.7.3 Electrical Power System Process Control Criteria

a. The Class IE power systems are designed as a three Division system. The ESF systems of any two of the three divisions provide for the minimum safety functions necessary to shutdown the unit and maintain it in a safe shutdown condition.

b. Protective relaying is used to detect and isolate faulted equipment from the system with a minimum of disturbance in the event of equipment failure.

c. Voltage relays or bistables are used on the emergency equipment buses to isolate these buses from the normal electrical system in the event of loss of offsite power and to initiate starting of the standby emergency power system diesel generators.

d. The standby emergency power diesel generators are started by control relays. The generators are also loaded by a control system to meet the existing emergency condition.

e. Electrically operated breakers are controllable from the control room.

f. Instruments for monitoring the operation of essential generators, transformers, and circuits are provided in the control room.

1.2.2 Plant Description

1.2.2.1 Site Characteristics

1.2.2.1.1 Location

[HISTORICAL INFORMATION] [Grand Gulf Nuclear Station is located in Claiborne County in southwestern Mississippi. The plant site is on the east side of the Mississippi River about 25 miles south of Vicksburg and 37 miles north-northeast of Natchez.] The Grand Gulf Military Park borders a portion of the north side of the plant site property, and the community of Grand Gulf is about 1-1/2 miles to the north. The town of Port Gibson is about 6 miles southeast of the plant site.

The site and its environs consist primarily of woodlands and farms. The total area of the plant site is approximately 2100 acres. Within this area are two lakes, Gin Lake and Hamilton Lake. These lakes were once the channel of the Mississippi River and average about 8 to 10 feet in depth.

The western half of the plant site consists of materials deposited by the Mississippi River and extends eastward from the river about 0.8 mile. This area is generally 55 to 75 feet above mean sea level (msl).

The eastern half of the plant site is rough and irregular with steep slopes and deep-cut stream valleys and drainage courses. Elevations in this portion of the plant site range from about 80 feet above msl to more than 200 feet above msl at the inland of the site. Elevations of about 400 feet above mean sea level occur on the hilltops east and northeast of the site.

The orientation of the principal plant structures on the site is shown in Figure 1.2-1.

1.2.2.1.2 Meteorology

[HISTORICAL INFORMATION] [The climate at the site is generally subtropical and humid in character, but is subject to important polar influence from time to time. Maximum rainfall in a 68-year period of record at Vicksburg, Mississippi, was 9.97 inches in 24 hours, and the maximum average monthly rainfall was about 16.5 inches. Prevailing winds are from the south-southeast. Maximum wind speeds at Municipal Airport, Jackson, Mississippi, in a 50-year period of record were 68 mph and occurred in March 1952.

During 92 years of record, 65 hurricanes, or post-hurricane path centerlines, passed within 100 miles of the site. There have been two damaging tornadoes in a 50-year period of record (1916-1966) within a 25-mile radius. This is typical of tornado frequency in the site region. An onsite meteorological measurement program was initiated in 1972 to provide data to assess limits to be set later on radioactive gas releases. Safety-related structures are design-ed for a maximum tornado load of 360 mph and wind load of 90 mph.]

1.2.2.1.3 Hydrology

[HISTORICAL INFORMATION] [The site for the Grand Gulf Nuclear Station is located on the east side of the Mississippi River in the vicinity of river mile 406 about 25 miles south of Vicksburg and 6 miles northwest of Port Gibson. It is bounded on the west by the Mississippi River and on the east by loessial bluffs (forming part of the hilly region which extends from Vicksburg to Baton Rouge). The Mississippi River floodplain adjacent to the site is relatively low and flat with elevations of 55 to 75 ft msl.

The plant site is located in the loessial uplands with a plant grade elevation of 132.5 ft msl. This elevation is well above the probable maximum flood (PMF) elevation in the Mississippi River. The design project flood (DPF) and 100-year flood elevation of the Mississippi River in the plant vicinity are at elevations of 96.2 and 93.1 ft msl, respectively.

The plant makeup and service water is supplied by a series of radial collector wells located in the floodplain parallel to the Mississippi River. These collector wells have been constructed by sinking cylindrical concrete caissons into the alluvial aquifer, sealing the bottom with a concrete plug, and projecting perforated pipes horizontally into the aquifer.

The principal ground water-bearing zones in the site vicinity are the Mississippi River alluvium, the terrace deposits, and the Catahoula formation.

The Mississippi River alluvium is the principal aquifer at the site and is the source of plant service water supply. The ground water is unconfined and the water level is generally controlled by the Mississippi River stage. The terrace deposits contain local permeable zones that yield several hundred gallons of water per minute. The regional water table occurs within the terrace deposits and adjacent Mississippi River alluvium; however, several perched water zones also occur within the terrace deposits. The Catahoula formation underlies the alluvium and terrace deposits and comprises a source of ground water for domestic wells. The ground water in the Catahoula formation occurs during semi-confined conditions.]

1.2.2.1.4 Geology

[HISTORICAL INFORMATION] [Surface material at the site is Pleistocene loess. This material erodes easily forming very steep slopes along stream channels. One such slope, along the Mississippi River floodplain, divides the site so that it lies in two subprovinces of the Central Gulf Coastal Plain physiographic province. The subprovinces are the Loess or Bluff Hills to the east and the Mississippi alluvial plain to the west.

The site is underlain by approximately 18,000 ft of Cretaceous through Cenozoic sands, gravels, clays, marls, claystones, sandstones, and limestones. These sediments were deposited on middle Jurassic evaporites, the parent material for salt domes found in the area. Regional dip is southward and becomes progressively steeper toward the Gulf Coast. As a result of the steepened dip, most formations tend to be wedge shaped, thickening coastward.

Several domal or structural uplift areas are found within the Gulf Coast Basin. The nearest of these, located about 50 miles east-northeast of the site, is the Jackson Dome. Formation of this structure began in the early Cretaceous period and ended in the middle Tertiary period. A salt dome has been formed as near as 8 miles from the site. The dome was formed from the late Cretaceous period through the Oligocene epoch. No nearer salt domes are known.

Petroleum exploration drilling near the site has generally been unsuccessful. Within a 6-mile radius of the site, 13 wildcat oil wells have been drilled; all were dry. The nearest of these was 3-1/2 miles from the site. At least 50 wells have been drilled in Claiborne County and only two have discovered hydrocarbons.]

1.2.2.1.5 Seismology and Design Response Spectra

[HISTORICAL INFORMATION] [The site area is not seismically active; however, distant earthquakes may have been felt there. The New Madrid, Missouri, earthquakes of 1811-1812, which occurred 325 miles north of the site, had maximum intensities of MM XI-XII. These events are conservatively estimated to have had a maximum intensity of MM VI at the site.

The largest event known to have occurred in the Gulf Coast Basin, not associated with a structure, is the strong intensity MM VI Donaldsonville, Louisiana, earthquake of October 19, 1930. If this earthquake occurred at the site, a peak acceleration of 0.07-0.10 g would result, according to the intensity-acceleration curves of Neumann (1954). A safe shutdown peak horizontal acceleration of 0.15 g and vertical acceleration of 0.10 g were selected for plant design giving additional conservatism. Design spectra for the safe shutdown earthquake with horizontal acceleration of 0.15 g and for a variety of damping values have been used for analysis of plant structures and equipment.]

1.2.2.1.6 Unusual Site Characteristics

There are no unusual site characteristics.

1.2.2.2 <u>General Arrangement of Structures and Equipment</u>

[HISTORICAL INFORMATION] [The principal buildings and structures include the containment structure, the turbine building, the auxiliary building, the control building, the diesel generator building, the standby service water cooling towers and basins, the enclosure building, the radwaste building, the auxiliary cooling tower, and the natural draft cooling tower.] A structure which houses the administration offices, clean machine shop, and guardhouse is provided. A building is also provided to store the site fire truck, foam chemicals, and miscellaneous fire fighting apparatus.

Bulk storage facilities are provided for hydrogen and oxygen in support of the hydrogen water chemistry system on the plant north end of the Unit 2 cooling tower basin. The bulk liquid hydrogen facility includes a 20,000 gallon cryogenic tank, cryogenic pumps, atmospheric vaporizers and gas storage tubes to supply high pressure gas to the hydrogen water chemistry, generator cooling and primary water tank blanket systems.

The bulk liquid oxygen facility includes a 9,000 gallon cryogenic tank and atmospheric vaporizers to supply low pressure gas to the hydrogen water chemistry system.

A Large Component Storage Building (LCSB) is located in the Northwest laydown area. This building houses components that were replaced during the GGNS EPU. The components include the steam dryer, both moisture separator reheaters, 9 feedwater heaters, both reactor feedpump turbines and their inner casings and the high pressure turbine rotor.

These buildings and structures are founded upon suitable material for their intended function. Structures essential to the safe operation and shutdown of the plant are designed to withstand more extreme loading conditions than normally considered in conventional nonnuclear design practice. The buildings and internal structures so designated are designed to provide protection as required from tornadoes, earthquakes, and the failure of equipment producing flooding, missiles, and pipe whip. Additional discussion of design considerations may be found in Chapter 3.

Location and orientation of the buildings on the site are shown in Figure 1.2-1. The general arrangement of the buildings and equipment locations is shown in Figures 1.2-2 through 1.2-16.

a. The containment structure, shown in Figures 1.2-2 through 1.2-8, is a seismic Category I structure which encloses the reactor coolant system, the drywell, suppression pool, upper pool, and some of the engineered safety feature systems and supporting systems. The functional design basis of the containment, including its penetrations and isolation valves, is to contain with adequate design margin the energy released from a design basis loss-of-coolant accident and to provide a leaktight barrier against the uncontrolled release of radioactivity to the environment, even assuming a partial loss of engineered safety features.

b. The turbine building, shown in Figures 1.2-2 through 1.2-8, houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building. There are safety-related instruments in the turbine building, but the building will not collapse onto or otherwise adversely affect the systems of which those instruments are a part in the event of a postulated accident.

c. The auxiliary building, shown in Figures 1.2-2 through 1.2-8, is a seismic Category I structure that contains safety systems, fuel storage and shipping equipment and necessary auxiliary support systems. Redundant safety trains in the auxiliary building and all other areas of the plant are separated and protected so that a loss of function of one train will not prevent the other train from performing its safety function.

d. The control building, shown in Figures 1.2-2 through 1.2-8, is a seismic Category I, multistoried, concrete and steel structure, in which many of the control and electrical systems, including required support systems directly related to safety or necessary for plant operations, are located.

e. The diesel generator building, shown in Figure 1.2-4 is seismic Category I and is constructed of reinforced concrete. The building contains the three diesel generators, three fuel oil day tanks, six starting air receivers-compressors, air intake vents and filters, mufflers, and controls. Each diesel generator and its associated equipment is in an individual room within the diesel generator building. The building interior and exterior walls that separate the diesel generators and associated equipment constitute a fire barrier wall having a 3-hour fire resistance rating.

f. The enclosure building, shown in Figure 1.2-8, is a limited leakage seismic Category I structure that encloses the upper portion of the containment above the auxiliary building roof level. The enclosure building provides a boundary for the standby gas treatment system, which maintains a negative pressure in the volume between the containment and enclosure building to ensure that leakage of radioactive materials from the containment is filtered prior to release to the environment in the unlikely event of a loss-of-coolant accident.

g. The radwaste building, shown in Figures 1.2-10 through 1.2-14, has six major areas; the collection tankage area, a processing area, a pipeway area, a personnel area, a solidification area, and a storage area. The radwaste systems process liquid, solid, and gaseous radioactive wastes generated by the plant.

h. The natural draft cooling tower is a concrete, natural draft, hyperbolic structure and is shown in Figure 1.2-15. The tower is designed to operate alone or in conjunction with the auxiliary cooling tower to dissipate all excess heat removed from the main condensers and accomplishes this function by the use of a spray network, a film type heat transfer surface, a tower basin, and circulating water pumps, piping, and valves.

i. The Ultimate Heat Sink (see Figure 1.2-1) is comprised of two separate, seismic Category I, mechanical draft cooling tower/pumphouse/basin structures. Each tower consists of four cells; each cell with a separate stack. Only four cells are required to support Unit 1 operation. The towers are constructed of a reinforced concrete frame with air intake louvers in the sides. Cells A and B of SSW cooling tower A and B contain ceramic fill blocks within the frame. Cells C and D of SSW cooling tower A and B contain stainless steel fill within the frame. Each tower is located over a separate concrete cooling water basin. Each pumphouse is located over the southwest corner of the basins, contains vertical wet pit pumps, and is provided with separate tornado missile protection walls on all sides, and on the roof.

j. The auxiliary cooling tower is a multi-cell mechanical draft fiberglass reinforced plastic structure with a concrete basin/foundation and is shown in Figure 1.2-16. The auxiliary cooling tower is designed to operate in conjunction with the natural draft cooling tower to dissipate excess heat removed from the main condensers. It accomplishes this function by the use of a spray network, a film type heat transfer surface, electric motor driven fans, a tower basin, a discharge flume connected to the natural draft cooling tower basin, piping, valves, and associated electric equipment contained in the auxiliary cooling tower power and control building.

1.2.2.3 Nuclear System

The nuclear system includes a direct cycle, forced circulation, General Electric boiling water reactor that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the rated power conditions is shown in Figure 1.1-1.

Extended Power Uprate

On September 8, 2010, Entergy requested approval of an amendment to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Operating License and Technical Specifications to increase the maximum reactor core power operating limit authorized in the Operating License from 3898 megawatts thermal (MWt) to 4408 MWt. The license amendment request included NEDC-33477P, "Safety Analysis Report for Grand Gulf Nuclear Station Constant Pressure Power Uprate," Revision 0(PUSAR), which the NRC also reviewed in conjunction with the amendment request. On July 18, 2012, the NRC approved the license amendment request.

1.2.2.3.1 Reactor Core and Control Rods

Fuel for the reactor core consists of slightly enriched uranium dioxide pellets sealed in Zircaloy-2 tubes. These tubes (or fuel rods) are assembled into individual fuel assemblies. Gross control of the core is achieved by movable, bottom-entry control rods. The control rods are cruciform in shape and are dispersed throughout the lattice of fuel assemblies. The control rods are positioned by individual control rod drives.

Each fuel assembly has several fuel rods with gadolinia (Gd_2O_3) mixed in solid solution with the UO_2. The Gd_2O_3 is a burnable poison which diminishes the reactivity of the fresh fuel. It is depleted as the fuel reaches the end of its first cycle.

A conservative limit of plastic strain is the design criterion used for fuel rod cladding failure. The peak linear heat generation for steadystate operation is well below the fuel damage limit even late in life. Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of a fuel loading.

1.2.2.3.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater, core sprays, and standby liquid control; the in-core instrumentation; and other components. The main connections to the vessel include steam lines, coolant recirculation lines, feedwater lines, control rod drive and in-core nuclear instrument housings, core spray lines, residual heat removal lines, standby liquid control line, core differential pressure line, jet pump pressure sensing lines, and water level instrumentation.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure in the steam space above the separators is 1040 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel (except for the top head, nozzles, and nozzle weld zones which are unclad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steam lines. Each steam line is provided with two isolation valves in series; one on each side of the containment barrier.

1.2.2.3.3 Reactor Recirculation System

The reactor recirculation system pumps reactor coolant through the core. This is accomplished by two recirculation loops external to the reactor vessel but inside the containment. Each external loop contains motor-operated maintenance valves and one hydraulically operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low frequency motor-generator set to control reactor power level through the effects of coolant flow rate on moderator void content.

The internal portion of the loop consists of the jet pumps which contain no moving parts. The jet pumps provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Any recirculation line break would still allow core flooding to approximately two-thirds of the core height - the level of the inlet of the jet pumps.

1.2.2.3.4 Residual Heat Removal System

The residual heat removal (RHR) system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

a. Removes decay and sensible heat during and after plant shutdown.

b. Injects water into the reactor vessel, following a loss-of-coolant accident, rapidly enough to reflood the core and maintain fuel cladding below fragmentation temperature independent of other core cooling systems. This is discussed in subsection 1.2.2.4.8, “Emergency Core Cooling Systems.”

c. Removes heat from the containment, following a loss-of-coolant accident, to limit the increase in containment pressure. This is accomplished by cooling and recirculating the suppression pool water (containment cooling) and by spraying the containment air space (containment spray) with suppression pool water.

1.2.2.3.5 Reactor Water Cleanup System

The reactor water cleanup system recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

1.2.2.3.6 Nuclear Leak Detection System

The nuclear leak detection system consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

a. Main steam lines

b. Reactor water cleanup (RWCU) system

c. Residual heat removal (RHR) system

d. Reactor core isolation cooling (RCIC) system

e. Fuel pool cooling and cleanup (FPCC) system

f. High pressure core spray (HPCS) system

g. Low pressure core spray (LPCS) system

h. Instrument lines

Small leaks generally are detected by temperature and pressure changes, fillup rate of drain sumps, and fission product concentration inside the containment. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

1.2.2.4 Nuclear Safety Systems and Engineered Safety Features

1.2.2.4.1 Reactor Protection System

The reactor protection system (RPS) initiates a rapid, automatic shutdown (scram) of the reactor. It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The reactor protection system overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single failure occurs.

1.2.2.4.2 Neutron Monitoring System

Although not all portions of the neutron monitoring system qualify as a nuclear safety system, those that provide high neutron flux signals to the reactor protection system do. The intermediate range monitors (IRMs) and average power range monitors (APRMs), which monitor neutron flux via in-core detectors, signal the reactor protection system to initiate a scram in time to prevent excessive fuel cladding damage as a result of overpower transients. The source range monitors (SRMs) prevent rod motion in the startup mode when certain conditions discussed in subsection 7.6.1.6.2d are not satisfied.

1.2.2.4.3 Control Rod Drive System

When a scram is initiated by the reactor protection system, the control rod drive system inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high pressure water stored in an accumulator in the hydraulic control unit or reactor pressure forces its control rod into the core.

1.2.2.4.4 Control Rod Drive Housing Supports

Control rod drive housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured. The supports prevent a nuclear excursion as a result of a housing failure and thus protect the fuel barrier.

1.2.2.4.5 Control Rod Velocity Limiter

A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from its control rod drive. This action limits the rate of reactivity insertion resulting from a rod drop accident. The limiters contain no moving parts.

1.2.2.4.6 Nuclear System Pressure Relief System

A pressure relief system consisting of safety/relief valves mounted on the main steam lines is provided to prevent excessive pressure inside the nuclear system following either abnormal operational transients or accidents.

1.2.2.4.7 Reactor Core Isolation Cooling System

The reactor core isolation cooling system (RCIC) provides makeup water to the reactor vessel when the vessel is isolated. The RCIC system uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel.

1.2.2.4.8 Emergency Core Cooling Systems (ESF System)

Four emergency core cooling systems are provided to maintain fuel cladding below fragmentation temperature in the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant. The systems are:

High pressure core spray (HPCS) system

Automatic depressurization (ADS)

Low pressure core spray (LPCS)

Low pressure coolant injection (LPCI), an operating mode of the residual heat removal system

a. <u>High Pressure Core Spray</u> - The HPCS system provides and maintains an adequate coolant inventory inside the reactor vessel to maintain fuel cladding temperatures below fragmentation temperature in the event of breaks in the reactor coolant pressure boundary. The system is initiated by either high pressure in the drywell or low water level in the vessel. It operates independently of all other systems over the entire range of pressure differences from greater than normal operating pressure to zero. The HPCS cooling decreases vessel pressure to enable the low pressure cooling systems to function. The HPCS system pump motor is powered by a diesel generator if auxiliary power is not available, and the system may also be used as a backup for the RCIC system.

b. Automatic Depressurization - The automatic depressurization system rapidly reduces reactor vessel pressure in a loss-of-coolant (LOCA) accident situation in which the HPCS system fails to maintain the reactor vessel water level. The depressurization provided by the system enables the low pressure emergency core cooling systems to deliver cooling water to the reactor vessel. The ADS uses some of the relief valves that are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open on conditions indicating both that a break in the reactor coolant pressure boundary has occurred and that the HPCS system is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value. The ADS will not be activated unless either the LPCS or LPCI pumps are operating. This is to ensure that adequate coolant will be available to maintain reactor water level after the depressurization.

c. Low Pressure Core Spray - The LPCS system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the reactor coolant pressure boundary but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water into each fuel channel. The LPCS loop functioning in conjunction with the ADS or HPCS can maintain the fuel cladding below the prescribed temperature limit following a loss-of-coolant accident.

d. Low Pressure Coolant Injection - Low pressure coolant injection is an operating mode of the residual heat removal (RHR) system, but is discussed here because the LPCI mode acts as an engineered safety feature in conjunction with the other emergency core cooling systems. LPCI uses the pump loops of the RHR to inject cooling water into the pressure vessel. LPCI is actuated by conditions indicating a breach in the reactor coolant pressure boundary, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation provides the capability of core reflooding, following a loss-of-coolant accident, in time to maintain the fuel cladding below the prescribed temperature limit.

1.2.2.4.9 Containment Systems

1.2.2.4.9.1 Containment Functional Design

The containment design for this plant has been given the name Mark III. This containment design incorporates the drywell/pressure suppression feature of previous BWR containment designs into a dry-containment type of structure.

In fulfilling its design basis as a fission product barrier in case of an accident, the Mark III containment is a low-leakage structure even at the elevated pressures that could follow a main steam line rupture or a recirculation line break.

The main features of the design include the following:

a. A drywell surrounding the reactor pressure vessel (RPV) and a large part of the reactor coolant pressure boundary

b. A suppression pool that serves as a heat sink during normal operational transients and accident conditions

c. A containment upper pool for shielding, refueling operations, and makeup to the suppression pool

d. A steel-lined reinforced concrete containment structure

The containment functional design is described in more detail in subsection 6.2.1.

1.2.2.4.9.2 RHR/Suppression Pool Cooling

The suppression pool cooling subsystem of RHR is placed in operation to limit the temperature of the water in the suppression pool following a design basis loss-of-coolant accident, to control the pool temperature during normal operation of the safety-relief valves and the RCIC system, and to reduce the pool temperature following an isolation transient. In the suppression pool cooling mode of operation, the RHR main system pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to the service water. The fluid is then discharged back to the suppression pool or the reactor pressure vessel.

1.2.2.4.9.3 RHR/Containment Spray (ESF System)

A containment spray system is provided to function, by automatic initiation, to condense steam which may bypass the suppression pool to prevent over-pressurization of the containment following a LOCA. The containment spray system consists of two redundant subsystems, each with its own full-capacity spray header. Each subsystem is supplied from a separate redundant RHR subsystem. The containment spray system also serves as an iodine removal system to reduce doses to the environment following a LOCA.

1.2.2.4.9.4 Combustible Gas Control (ESF System)

In the unlikely event of a loss-of-coolant accident, hydrogen and oxygen will be generated in the drywell and containment. The combustible gas control system will ensure that hydrogen concentrations are kept below the limits specified in NRC Regulatory Guide 1.7, Revision 1. For postulated degraded core accidents, the combustible gas control system will preclude the potential for local detonations and ensure the integrity of the containment. The systems used will include a drywell purge system, hydrogen control systems, and a backup containment purge system.

The drywell purge compressor also performs the function diluting the drywell source term with the containment and suppression pool environment by pressurizing the drywell and discharging the drywell source term through the drywell suppression pool vents. With the implementation of the alternative source term (Amendment 145), this dilution of drywell source term is no longer credited in the Equipment Qualification analysis which is presented in FSAR Section 3.11.

1.2.2.4.10 Containment and Reactor Vessel Isolation Control System (ESF System)

The containment and reactor vessel isolation control system automatically initiates closure of isolation valves to close off all process lines that are potential leakage paths for radioactive material to the environs. This action is taken upon indication of a breach in the reactor coolant pressure boundary.

1.2.2.4.10.1 Main Steam Line Isolation Valves

Although all pipelines that both penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities, the main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. Automatic isolation valves are provided in each main steam line. Each is powered by both air pressure and spring force. These valves fulfill the following objectives:

a. Prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the reactor vessel resulting from either a major leak from the steam piping outside the containment or a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel.

b. Limit the release of radioactive materials by isolating the reactor coolant pressure boundary in case of a gross release of radioactive materials from the fuel to the reactor cooling water and steam.

c. Limit the release of radioactive materials by closing the containment barrier in case of a major leak from the nuclear system inside the containment.

1.2.2.4.10.2 Main Steam Line Flow Restrictors

A venturi-type flow restrictor is installed in each steam line. These devices limit the loss of coolant from the reactor vessel before the main steam line isolation valves are closed in case of a main steam line break outside the containment.

1.2.2.4.11 Process Radiation Monitoring System

1.2.2.4.11.1 Main Steam Line Radiation Monitoring Subsystem

The main steam line radiation monitoring subsystem consists of four gamma radiation monitors located externally to the main steam lines just outside the containment. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors are used to initiate closure of the Rx water sample line drywell isolation valves and trip the mechanical vacuum pump and valves.

1.2.2.4.11.2 Ventilation Exhaust Radiation Monitoring System

The process ventilation radiation monitoring systems consist of a number of radiation monitors arranged to monitor the activity level of the air exhaust from the containment and drywell, auxiliary building fuel handling and pool sweep areas, and air intake into the control room.

1.2.2.4.12 Standby Gas Treatment System (ESF System)

The standby gas treatment system has been designed to minimize exfiltration of contaminated air from the enclosure building, the auxiliary building, and the containment following an accident or abnormal condition that could result in abnormally high airborne radioactivity in these areas.

All necessary equipment and surrounding structures have been designed to seismic Category I specifications.

All components of the standby gas treatment system will be operable during a loss of offsite power supply.

1.2.2.4.13 Auxiliary Building Isolation Control System

The auxiliary building isolation control system automatically initiates closure of isolation valves on selected lines that penetrate the auxiliary building to preserve the integrity of the standby gas treatment boundary. This action is taken upon indication of a breach in the reactor coolant pressure boundary.

1.2.2.4.14 Safety-Related Electrical Power Systems

Standby ac power is supplied by three diesel generators. Each engineered safety features (ESF) division is supplied by a separate diesel generator. There are no provisions for transferring ESF division buses between standby ac power supplies or supplying more than one ESF division from one diesel generator. The one-to-one relationship between diesel generator and ESF division ensures that a failure of one diesel generator can affect only one ESF division.

Three independent Class IE 125-volt dc systems exist, one per ESF division of the Class IE electric power system.

1.2.2.4.15 Standby Liquid Control System

Although not intended to provide prompt reactor shutdown, as the control rods are, the standby liquid control system provides a redundant, independent, and alternate way to bring the nuclear fission reaction to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition.

1.2.2.4.16 Safe Shutdown from Outside the Control Room

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by the use of the local controls and equipment that are available outside the control room.

1.2.2.4.17 Main Steam Line Isolation Valve Leakage Control System (ESF System)

The main steam line isolation valve leakage control system (MSIVLCS) is designed to minimize the fission products which could bypass the standby gas treatment system after a LOCA. This is accomplished by directing the leakage through the closed main steam line isolation valves to a space serviced by the Standby Gas Treatment System (SGTS).

1.2.2.4.18 Feedwater Leakage Control System (ESF System)

The feedwater leakage control system is designed to minimize the fission products which could bypass the SGTS after a LOCA. This is accomplished by filling the feedwater lines between the containment isolation valves with suppression pool water and maintaining a water seal at a pressure slightly higher than the containment pressure.

1.2.2.4.19 Suppression Pool Make-up System (ESF System)

The suppression pool make-up system provides water from the upper containment pool to the suppression pool by gravity flow following a LOCA. The quantity of water provided is sufficient to maintain required drywell upper-most vent coverage for all postulated accidents.

1.2.2.4.20 Control Room HVAC (ESF System)

The control room HVAC system provides an environment in the control room suitable for the operation of equipment necessary for the safe shutdown of the plant and will function in the event of a LOCA. The system shall protect the plant operators from the results of any accident which could impair their safety and therefore compromise the safety of the plant.

1.2.2.5 Power Conversion System

1.2.2.5.1 Turbine Generator

The turbine generator is an 1800-rpm, tandem-compound, six-flow, 46-inch last-stage buckets, reheat unit with electrohydraulic control (EHC) for normal operation. The EHC system is equipped with three independent levels of speed sensing. The approximate rating of the turbine generator is 1,352,907 kw.

The generator is a direct-driven, three-phase, 60-Hz, 22,000 volt, 1800-rpm, hydrogen cooled with water cooled stator and rotor windings, synchronous generator rated at 1600 MVA at 0.9 power factor, 75 psig hydrogen pressure, and 0.58 short-circuit ratio.

1.2.2.5.2 Main Steam System

The main steam system delivers steam from the nuclear boiler system via four 28-inch OD steam lines to the turbine generator, second-stage reheaters, steam jet air ejectors, offgas preheater, and to the reactor feed pump turbines, seal steam generators, and main condenser hotwell at startup and low loads.

1.2.2.5.3 Main Condenser

Steam from the low-pressure turbine is exhausted directly downward into the condenser shells through exhaust openings in the bottom of the turbine casings and is condensed. The condenser is a three-section, multipressure condenser, each section serving one double-flow, low-pressure turbine section. The condenser also serves as a heat sink for the turbine bypass system, feedwater heater and drain tank high-level dumps, relief valve discharges during transient conditions and reactor feed pump turbine exhausts.

1.2.2.5.4 Main Condenser Evacuation System

The main condenser evacuation system removes the noncondensable gases from the main condenser and exhausts them to the gaseous radwaste system. Two twin-element, two-stage, steam jet air ejectors (100-percent-capacity each), complete with intercondenser, are provided for air removal during normal operation. A mechanical vacuum pump is used during startup.

1.2.2.5.5 Turbine Gland Sealing System

The turbine gland sealing system provides clean, nonradioactive steam to the seals of the turbine valve stem glands and the turbine shaft glands. The sealing steam is supplied by a separate seal steam generator using condensate from the condensate storage tank during normal plant operation. The unit auxiliary boiler provides an auxiliary steam supply for startup and when the seal steam generator is not available. The seal steam condenser collects and condenses the air and steam mixture and discharges the air leakage to the turbine building vent, using a motor-driven exhauster. Contaminated gland seal heating steam is condensed in Feedwater Heater No. 4.

1.2.2.5.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under the control of the pressure controller. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The capacity of the turbine bypass system is 35% of the reactor rated steam flow. The pressure control system provides main turbine control valve and bypass valve position demands so as to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power level by changing reactor recirculation flow rate.

1.2.2.5.7 Circulating Water System

The circulating water system provides the main condenser with a continuous supply of cooling water to remove the heat rejected from the cycle. The circulating water system is a closed system utilizing a natural draft cooling tower and a mechanical draft auxiliary cooling tower. Two vertical motor-driven pumps circulate the cooling water from the cooling tower basin through the main condenser and then back to the cooling towers. Makeup water, to compensate for drift, blowdown, and evaporation losses, is supplied from the plant service water system.

1.2.2.5.8 Condensate and Feedwater Systems

Three condensate pumps take the deaerated condensate from the hotwell of the intermediate-pressure shell of the main condenser and deliver it, in turn, through the condensate full flow filters and the condensate demineralizers. Filtered and demineralizer effluent then passes to the three condensate booster pumps, and the condensate booster pumps then discharge through four stages of low-pressure feedwater heaters to the two turbine-driven reactor feed pumps. Drains from moisture-separator and reheaters, and the fifth- and sixth-stage feedwater heaters, are pumped forward by two heater drain pumps, and the drains from first-, second-, third-, and fourth-stage lower-pressure heaters are cascaded back to the main condenser. The reactor feed pumps discharge the total feedwater flow through the fifth- and the sixth-stage high-pressure feedwater heaters to the reactor. Contaminated gland sealing steam from the reactor feed pump turbines is condensed in the main condenser.

1.2.2.5.9 Condensate Cleanup System

The 133-percent-capacity condensate cleanup system consists of eight units of multiple deep-bed-type demineralizers (with two units as spares) that operate in parallel. The system also includes three precoat filters (precoat filter “B” abandoned in place)used for startup and normal operation.

The condensate cleanup system maintains the required purity of feedwater flowing to the reactor.

1.2.2.5.10 Hydrogen Water Chemistry System

A hydrogen water chemistry (HWC) system is provided to further reduce the susceptibility of reactor recirculation piping and reactor vessel internal materials to intergranular stress corrosion cracking. This is accomplished by injecting hydrogen into the condensate booster pump suction header to suppress the formation of radiolytic oxygen in the reactor coolant. Oxygen is injected into the offgas system to maintain a stoichiometric balance of hydrogen and oxygen entering the offgas recombiners.

1.2.2.6 Electrical Systems and Instrumentation Control

1.2.2.6.1 Electrical Power Systems

The station generator power is fed to a main step-up transformer bank through the isolated phase bus system that was modified for EPU. The main step-up transformer bank transforms the power generated at 20.9 kV (originally 22 kV) to 500 kV. Then it is fed to the switchyard where the distribution of power to the utility grid via the transmission lines and to the station for station ac power requirements takes place.

The switchyard is fed by three 500 kV transmission lines on separate right-of-ways. The station offsite power is fed by two 500 kV circuits from the switchyard and one independent 115 kV offsite circuit. Each 500 kV circuit feeds a service transformer which provides engineered safety features (ESF) and balance-of-plant transformers with 34.5 kV power for further voltage transformations. The independent 115 kV offsite circuit feeds a third ESF transformer. The ESF transformers provide only the ESF buses with 4.16 kV ac power, and the balance-of-plant transformers supply the 4.16 kV ac power, and the balance-of-plant transformers supply the 4.16, 6.9, and 13.8 kV ac power requirements for the balance-of-plant load groups. The ac power is also distributed to the ESF and balance-of-plant loads at 480 volts from the associated load centers and motor control centers.

Changes in electrical load demand associated with EPU are two nonsafety-related changes, the addition of Radial Wells and the Auxiliary Cooling Tower (ACT) expansion.

Three independent Class IE 125-volt dc systems exist, one per ESF division of the Class IE electric power system. For the balance-of-plant electric system, three 125-volt dc systems are provided; two of these are connected in series to provide a 250-volt dc system for large dc loads.

Standby ac power is supplied by three diesel generators. Each ESF division is supplied by a separate diesel generator. There are no provisions for transferring ESF division buses between standby ac power supplies or supplying more than one ESF division from one diesel generator. The one-to-one relationship between diesel generator and ESF division ensures that a failure of one diesel generator can affect only one ESF division.

1.2.2.6.2 Nuclear System Process Control and Instrumentation

1.2.2.6.2.1 Rod Control and Information System

The rod control and information system provides the means by which control rods are positioned from the control room for power control. The system operates valves in each hydraulic control unit to change control rod position. One control rod or a group of rods can be manipulated at a time. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.2.2.6.2.2 Recirculation Flow Control System

During normal power operation a variable position discharge valve is used to control flow. Adjusting this valve changes the coolant flow rate through the core and thereby changes the core power level. For startup and shutdown flow changes at lower power, the pump speed is changed by adjusting the frequency of the electrical power supply.

1.2.2.6.2.3 Neutron Monitoring System

The neutron monitoring system is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The source range monitors (SRMs) and the intermediate range monitors (IRMs) provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRMs) and average power range monitors (APRMs) allow assessment of local and overall flux conditions during power range operation. The traversing in-core probe system (TIP) provides a means to calibrate the individual LPRM sensors. The Neutron Monitoring System provides inputs to the Rod Control and Information System to initiate rod block trips if preset flux limits are exceeded, and inputs to the Reactor Protection System to initiate a scram if other limits are exceeded.

1.2.2.6.2.4 Refueling Interlocks

A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent an inadvertent criticality during refueling operations. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling platform, refueling platform main hoist, and control rods.

1.2.2.6.2.5 Reactor Vessel Instrumentation

In addition to instrumentation for the nuclear safety systems and engineered safety features, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and reactor vessel head inner seal ring leakage.

1.2.2.6.2.6 Core Performance Monitoring System

An on-line core performance monitoring system is provided to monitor and log process variables and to make certain analytical computations.

1.2.2.6.3 Power Conversion Systems Process Control and Instrumentation

1.2.2.6.3.1 Pressure Regulator and Turbine-Generator Control

The pressure controller maintains control of the turbine control valves and turbine bypass valves to allow proper generator and reactor response to system load demand changes while maintaining the nuclear system pressure essentially constant.

The turbine-generator speed-load controls act to maintain the turbine speed (generator frequency) constant and respond to load changes by adjusting the reactor recirculation flow control system and pressure control set point.

The turbine-generator speed-load controls can initiate rapid closure of the turbine control valves (rapid opening of the turbine bypass valves) to prevent turbine overspeed on loss of the generator electric load.

1.2.2.6.3.2 Feedwater Control System

The feedwater control system automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at predetermined levels. A conventional three element control system is used to accomplish this function.

1.2.2.7 Fuel Handling and Storage Systems

1.2.2.7.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient coolant and shielding are maintained to prevent overheating and excessive personnel exposure, respectively. The design of the fuel pool provides for corrosion resistance, adherence to seismic Category I requirements, and prevention of k_{eff} from exceeding 0.95 under flooded conditions. This subject is further discussed in Section 9.1.

1.2.2.7.2 Fuel Handling System

The fuel handling equipment includes a 125-ton cask and a 150-ton cask crane, new fuel bridge crane, fuel handling platform, fuel inspection stand, fuel preparation machine, fuel assembly transfer mechanism, containment refueling platform, containment polar crane, and other related tools for reactor servicing. All equipment conforms with applicable codes and standards. The principal function of the cask crane is to handle the spent fuel cask. The new fuel bridge crane normally transfers new fuel while in the fuel handling area until it is placed in the fuel preparation machine. The fuel handling platform transfers the fuel assemblies between the transfer pool, storage pools, and cask. Fuel assemblies are transferred through the transfer tube between the containment and the auxiliary building. The fuel assemblies inside the containment are handled by the refueling platform.

The disassembly and reassembly of the reactor head, internals, and drywell head during refueling is done using the containment polar crane.

All tools and servicing equipment necessary to meet the reactor general servicing requirements are designed for efficiency and safe serviceability.

1.2.2.8 Cooling Water and Auxiliary Systems

1.2.2.8.1 Standby Service Water System

The standby service water (SSW) system is designed to cool reactor auxiliaries essential to a safe reactor shutdown, to minimize the leakage of radioactive contamination from these auxiliaries to the environment, to provide a means of flooding the drywell and containment, and to provide a backup source of makeup water to the spent fuel pool. The system consists of two independent trains, each capable of cooling the engineered safety features following a LOCA and rejecting this heat to the atmosphere through one of the two redundant standby service water cooling towers. The system is designed to meet seismic Category I requirements. In the unlikely event that radioactive contamination occurs in either train, the radiation monitors of the system will alarm and permit the operator to isolate the portion of the system that is contaminated.

1.2.2.8.2 Component Cooling Water System

The component cooling water (CCW) system is a closed-loop system that provides parallel flow cooling to auxiliary equipment in the containment, drywell, and auxiliary buildings. The closed loop provides a barrier between contaminated systems and the service water discharged to the environment. Heat is removed from the closed loop by the plant service water system. The system has no safety-related function or required for a safe shutdown of the reactor and it is not designed to seismic Category I requirements. However, piping and valves associated with fuel pool heat exchangers and piping and valves forming a part of containment boundary are safety-related and designed to seismic Category I requirements. Radiation monitors are provided to detect contaminated leakage into the closed system.

1.2.2.8.3 Turbine Building Cooling Water System

This system is designed to cool the auxiliary plant equipment associated with the power conversion systems over the full range of normal plant operation.

1.2.2.8.4 Ultimate Heat Sink

The ultimate heat sink, consisting of the standby service water (SSW) system cooling towers and makeup basins, provides heat rejection and makeup water required for the dissipation of heat to permit the safe shutdown and cooldown of the plant and to maintain it in a safe shutdown condition. The SSW cooling towers are seismic Category I.

1.2.2.8.5 Condensate Storage and Transfer System

The condensate storage and transfer system maintains the required capacity and flow of the condensate for the RCIC and HPCS systems and maintains the required level in the condenser hotwell. The system also: Stores and transfers upper containment pool water during refueling, and cask storage pool water during fuel shipping cask loading; receives and stores the process effluent from the liquid radwaste system and provides makeup to other plant systems where required; provides storage space for the suppression pool water during plant shutdown, and provides condensate to the control rod drive (CRD) hydraulic system.

The system consists of a condensate storage tank, two condensate transfer pumps, and the necessary controls and instrumentation.

1.2.2.8.6 Makeup Water Treatment System

The makeup water treatment system furnishes suitable water as makeup for the plant. The permanent plant equipment consists of two trains, each containing a mixed bed cation exchanger, a mixed bed anion exchanger, and a charcoal filter. Connections are available for a mobile vendor supplied water treatment system.

1.2.2.8.7 Domestic Water System and Sanitary Waste Water System

The domestic water system provides the necessary supply of domestic water for the plant. Construction water is used as the domestic water system supply.

The sanitary waste water system is designed to maintain the sewer waste water quality in accordance with the applicable quality criteria limits.

1.2.2.8.8 Chilled Water Systems

Chilled water is produced by mechanical chilling units and supplied to area cooling units through closed recirculating piping systems. Chemical water treatment is provided for scale and corrosion control.

1.2.2.8.9 Compressed Air Systems

The service, instrument and plant air systems provide a continuous supply of compressed air of suitable quality and pressure for instrument control and general plant use. The plant air compressors discharge into their respective air receivers. The air is then distributed throughout the plant. Instrument air is additionally filtered and dried by plant air dryers prior to distribution throughout the plant.

1.2.2.8.10 Process Sampling Systems

The process sampling system is furnished to provide process information that is required to monitor plant and equipment performance and changes to operating parameters. Representative liquid and gas samples are taken automatically and/or manually during normal plant operation for laboratory or on-line analyses.

1.2.2.8.11 Plant Floor and Equipment Drainage

The floor and equipment drainage system is designed to collect liquid waste throughout the plant and discharge the radioactive and potentially radioactive waste to the radwaste system for processing. Separate drainage facilities are provided for nonradioactive waste.

The drainage system is also used to detect abnormal leakage in the emergency safety features rooms, the drywell, and containment.

1.2.2.8.12 Heating, Ventilating, and Air Conditioning Systems

The plant heating, ventilating, and air conditioning systems are designed to provide an environment with controlled temperature and humidity to ensure the comfort and safety of personnel and the integrity of plant equipment.

Plant heating, ventilating, and air conditioning systems serving engineered safety features equipment are designed with sufficient redundancy to ensure operation during emergency conditions.

1.2.2.8.13 Fire Protection System

The fire protection system is designed to provide an adequate supply of water or chemicals to points throughout the plant area where fire protection may be required. Diversified fire-alarm and fire-suppression types are selected to suit the particular areas or hazards being protected. The water for the system is taken from two 300,000-gallon tanks that are replenished automatically from the plant service water system. In addition to the tanks, the system consists of one electric-driven pump, two diesel engine-driven pumps, one jockey pump, and the associated piping, valves, and hydrants.

Chemical fire-fighting systems (CO_2 and Halon 1301) are also provided as additions to or in lieu of the water fire-fighting systems.

The necessary instrumentation and controls are provided for the proper operation of the fire-fighting systems and for fire detection and annunciation.

1.2.2.8.14 Communications Systems

Diverse systems have been provided for intra-plant and plant-to-offsite communication. A detailed description of the systems is provided in section 9.5.2.

1.2.2.8.15 Lighting Systems

The design of the lighting facilities is based on standards of the Illuminating Engineering Society. Special attention is given to areas where proper lighting is imperative during normal and emergency operations. The system design precludes the use of mercury vapor fixtures in the containment and the fuel handling area except where specifically evaluated and approved. The normal lighting systems are fed from the normal buses. Essential lighting fixtures are supplied by engineered safety features buses and are backed up by diesel-generator units. Emergency lighting fixtures are backed up by inverters off the station batteries and self-contained batteries. Normal operation and regular simulated offsite power-loss tests verify system integrity.

1.2.2.8.16 Diesel Generator Fuel-Oil System

The purpose of this system is to supply and store the fuel oil required to operate the diesel-generator units during post-LOCA maximum load demands. The principal design criteria associated with this system consist of the following:

a. Seven-day fuel oil capacity to meet the conditions above is provided for each diesel

b. Seismic Category I design

c. Missile protection

1.2.2.8.17 Auxiliary Steam System

An auxiliary steam system is provided to furnish a separate and independent steam supply. Process steam is generated in packaged, high voltage, electrode boilers and distributed through the plant by an auxiliary steam header. Auxiliary steam is required for condensate deaeration/heating, pump testing, and main turbine shaft seal steam during startup. Auxiliary steam is also used for plant heating and other miscellaneous plant processes.

1.2.2.8.18 Plant Service Water System

The plant service water (PSW) system is designed to cool plant auxiliaries that are not potential sources of radioactive contamination during normal operation, that are not required for safe reactor shutdown, and that can be efficiently cooled by raw well water. During refueling outages, PSW may provide cooling water for the alternate decay heat removal subsystem of RHR. The PSW system also provides makeup to the circulating water system and the water treatment system. The system draws water from the radial well system, pumps the coolant through the heat exchangers, and discharges to the circulating water system.

1.2.2.8.19 Containment Ventilation

The containment ventilation system consists of a normally operating containment ventilation system, a containment purge system, and a drywell purge system.

The containment ventilation system has been designed to provide a reliable source of fresh air, and to filter the containment air by recirculation through filter trains.

The containment purge system has been designed to purge the containment completely, when required, at a minimum rate of one air change per 5-hour period.

The drywell purge system has been designed either to purge the drywell at a minimum rate of one air change per hour or to serve as a drywell cleanup system for the removal of airborne contamination at a minimum recirculation rate of one air change per hour.

1.2.2.8.20 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system maintains acceptable temperature, clarity, and radioactivity levels of the water in the upper containment, fuel storage, and cask pools. The system includes two heat exchangers, each with the capacity for removing

15.0 x 10^6 Btu/hr from the pool with 140 F pool water and 95 F cooling water and having the capacity to pass the system flow or greater to maintain the desired purity level.

Detailed system operation is provided in Section 9.1.3.

1.2.2.9 Radioactive Waste Management

1.2.2.9.1 Gaseous Radwaste System

The purpose of the gaseous radwaste system is to process and control the release of gaseous radioactive wastes to the site environs so that the total radiation exposure to persons outside the controlled area does not exceed the maximum limits of the applicable 10 CFR 20 regulations even with some defective fuel rods.

The offgases from the main condenser are the major source of gaseous radioactive waste. The treatment of these gases includes volume reduction through a catalytic hydrogen-oxygen recombiner, water vapor removal through a condenser, decay of short-lived radioisotopes through a holdup line, further condensation and cooling filtration, adsorption of isotopes on activated charcoal beds, further filtration through high efficiency filters, and final releases.

Continuous radiation monitors are provided which indicate radioactive release from the reactor and from the charcoal absorbers. The radiation monitors are used to isolate the offgas system on high radioactivity in order to prevent gas of unacceptably high activity from release.

1.2.2.9.2 Liquid Waste System

The liquid waste system, consisting of equipment drain, floor drain, and chemical waste subsystems, is designed to collect and process waste generated throughout the plant. Processing of the waste is sufficient to allow recycle of the wastewater. Ties exist between all the subsystems to provide backup processing in the event of failure.

Continuous radiation monitors in the discharge line provide indications and records of radioactivity release and automatically discontinue flow in the event of high activity levels.

1.2.2.9.3 Solid Waste System

The solid waste system is designed to handle and dispose of solid waste produced by the plant. The waste, depending on activity and type, will be packaged for offsite shipment in accordance with all applicable regulations.

1.2.2.10 Radiation Monitoring and Control

1.2.2.10.1 Process Radiation Monitoring System

Process radiation monitoring systems are provided to monitor and control radioactivity in process and effluent streams and to activate appropriate alarms and controls.

A process radiation monitoring system is provided for indicating and recording radiation levels associated with plant process streams and effluent paths leading to the environment. All effluents from the plant which are potentially radioactive are monitored.

Process radiation monitoring is also discussed in subsections 7.6.1.2 and 12.3.4.

1.2.2.10.2 Area Radiation Monitoring System

The area radiation monitoring system functions to alert plant personnel of increasing or abnormally high radiation levels which could possibly result in inadvertent overexposure. The system consists of detectors located throughout the plant, along with local alarms, and has readout, alarming, and recording provisions in the control room.

1.2.2.10.3 Offsite Radiological Monitoring System

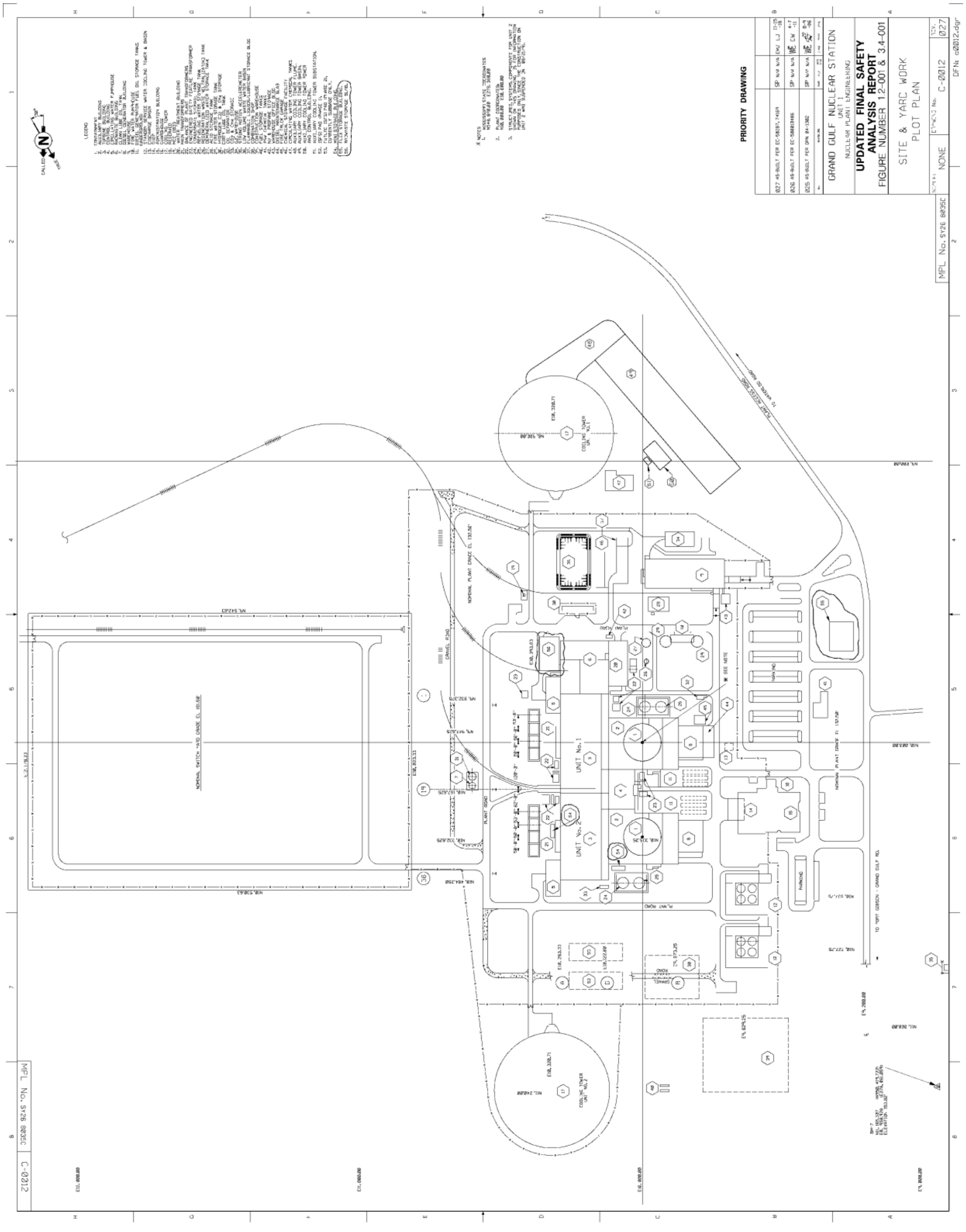
The important pathways to man are monitored by radiological measurements, including surveys, passive dosimeters, and samples collected for laboratory analyses. These include airborne, aquatic, and terrestrial pathways. The radiological monitoring program is implemented at least one year prior to reactor criticality. The program is designed to document background levels of direct radiation and concentrations of radionuclides that exist in aquatic and terrestrial ecosystems before and after plant operation and document the concentrations of radionuclides that could be attributable to operation of the Grand Gulf Nuclear Station.

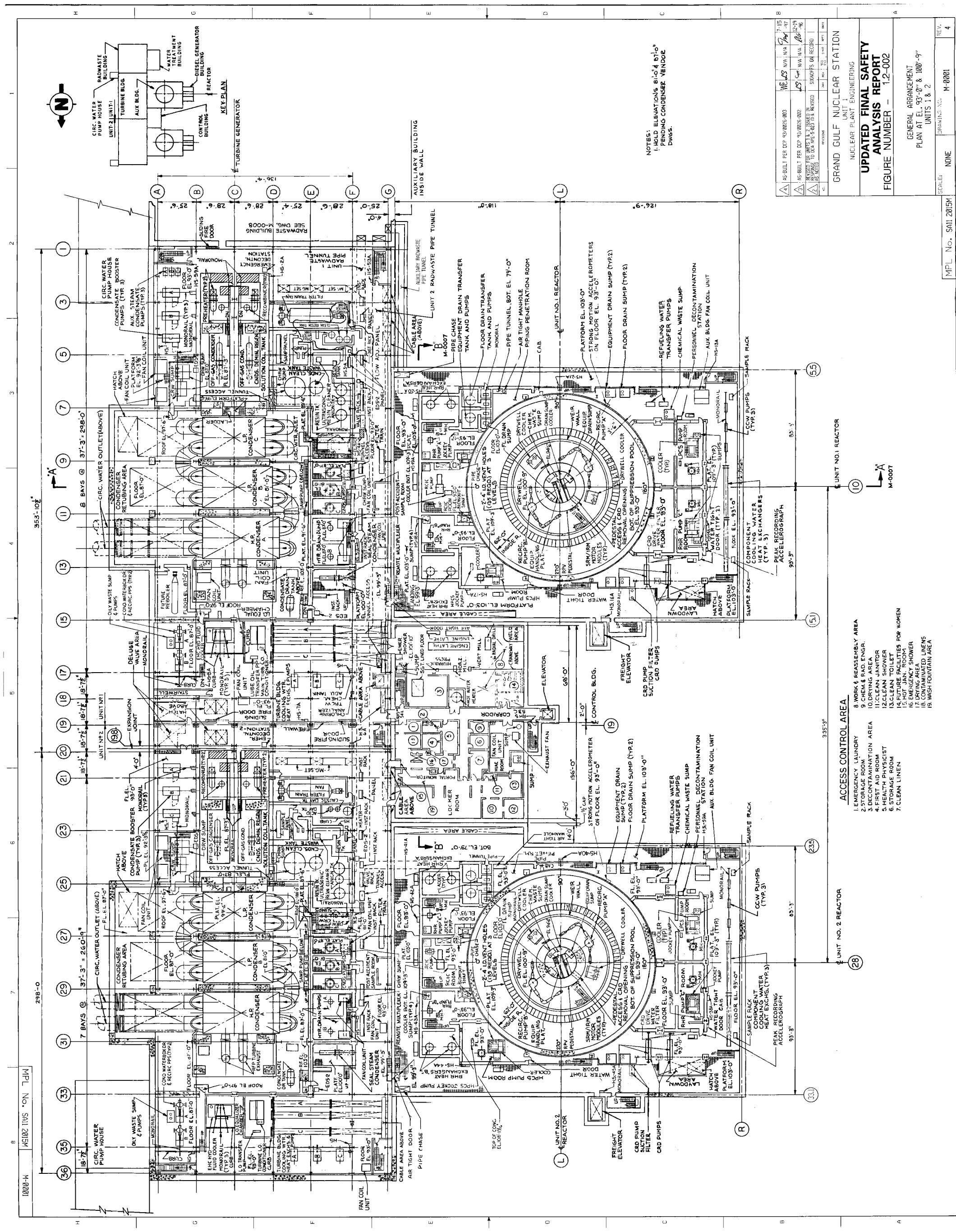
1.2.2.11 Particularly Difficult Engineering Problems

In general, particularly difficult engineering problems can be defined as those requiring development work or vendor testing to finalize the design. Such areas are discussed in Section 1.5.

1.2.2.12 Extrapolation of Technology

There are no significant extrapolations of technology incorporated in the Grand Gulf Nuclear Station.


PRIORITY DRAWING
GRAND GULF NUCLEAR STATION
UNIT 1
UPDATED FINAL SAFETY
ANALYSIS REPORT
FIGURE NUMBER 12-001 & 3.4-001
PLOT PLAN
C-0012


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 1.2-002
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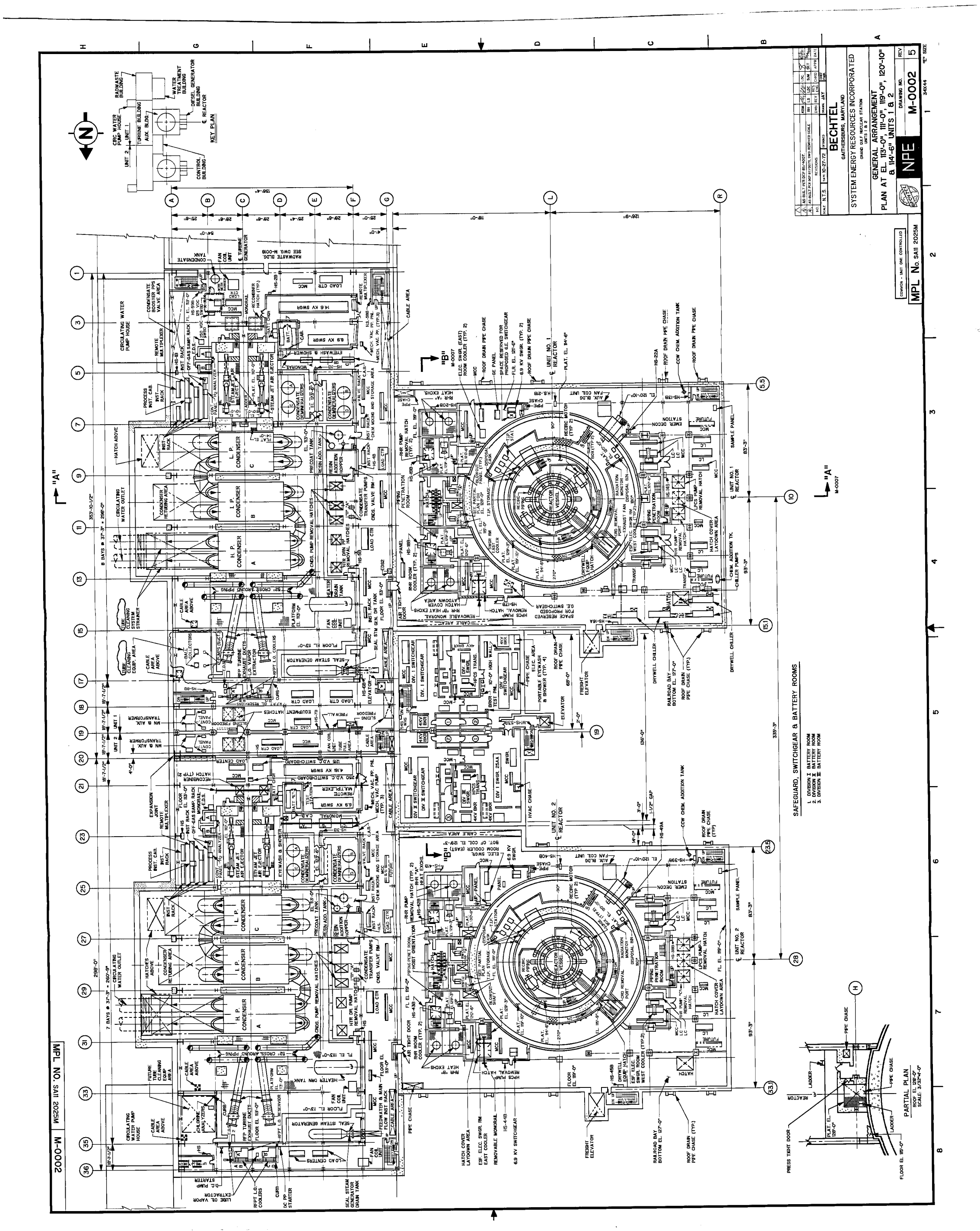
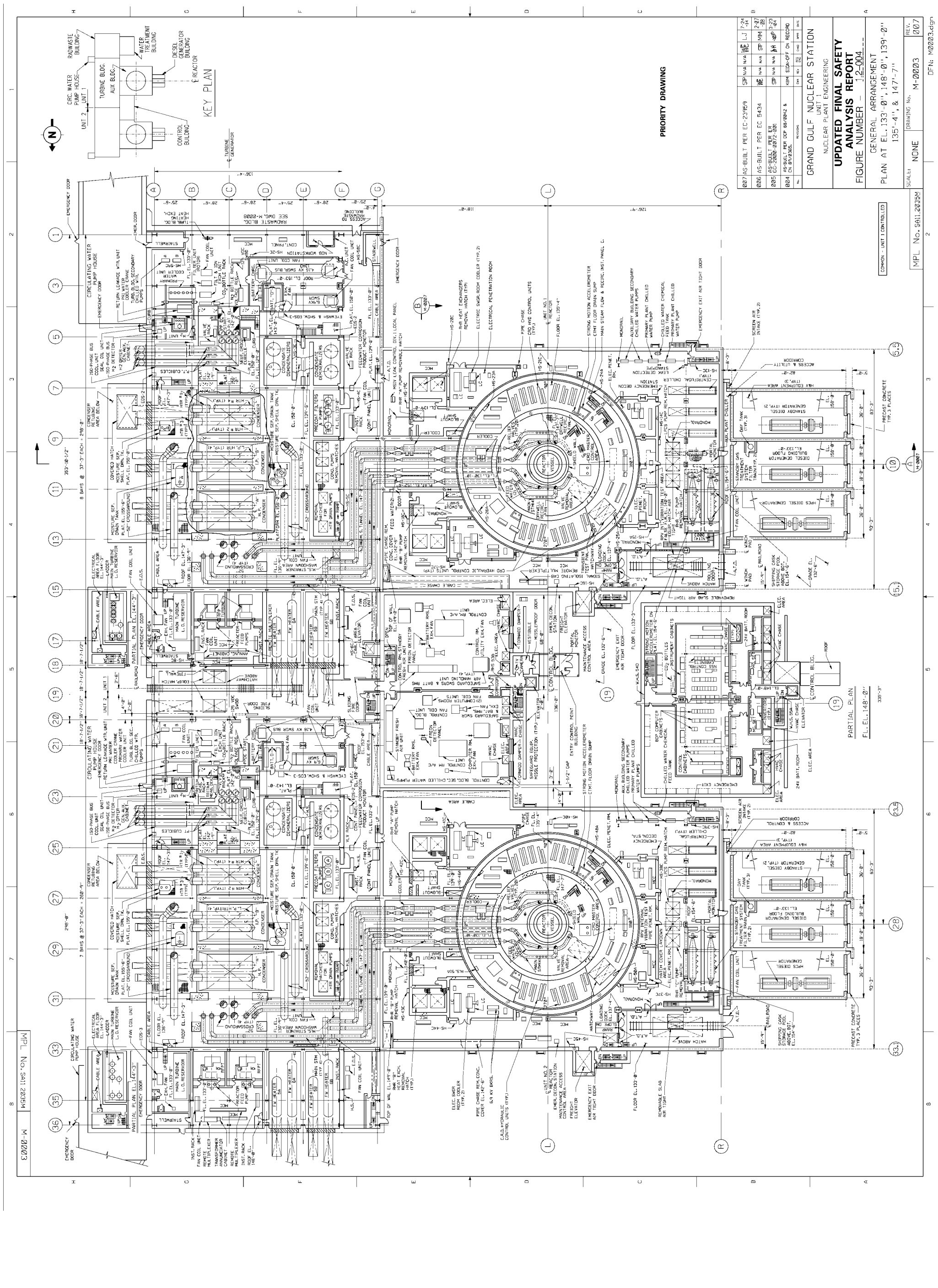
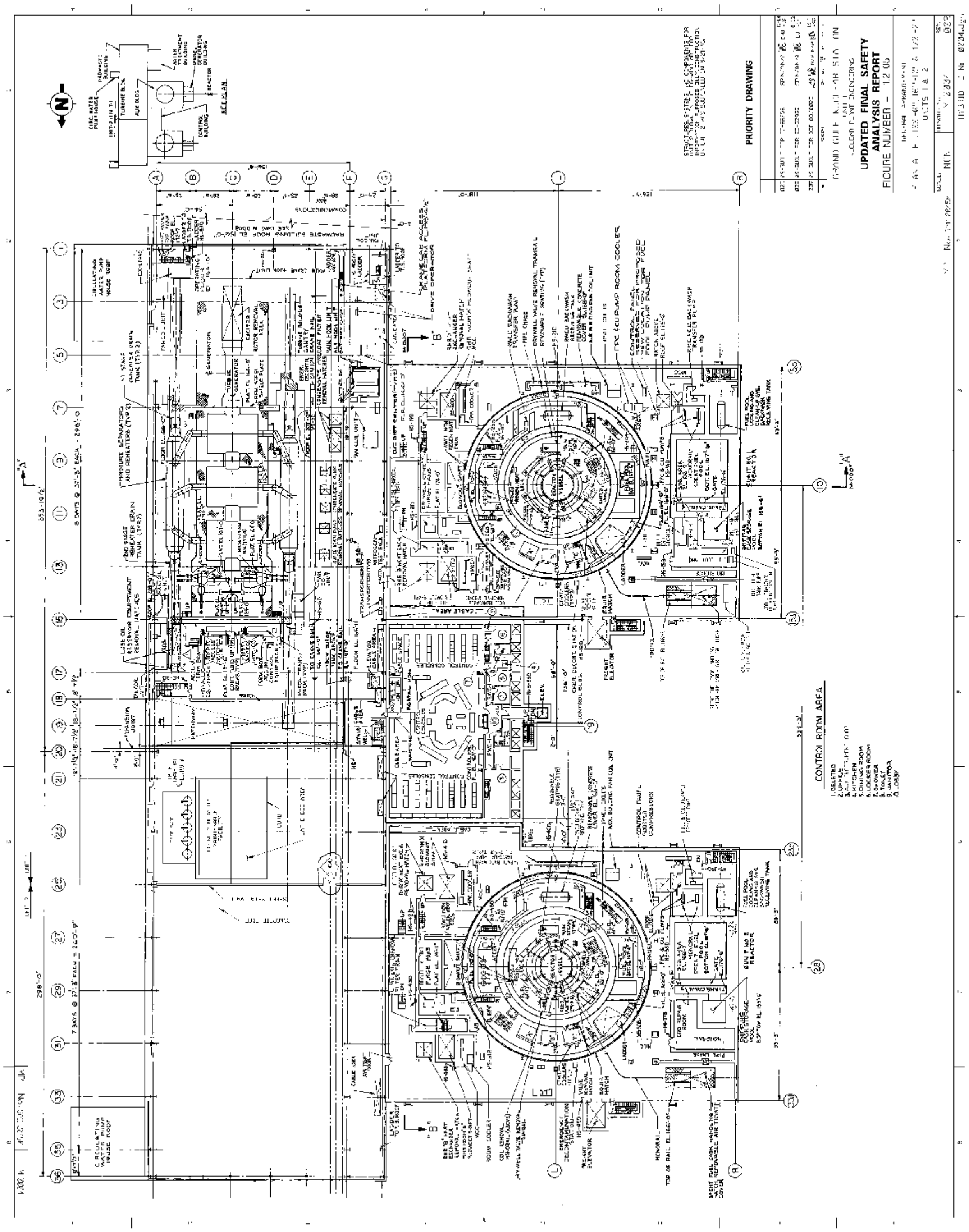
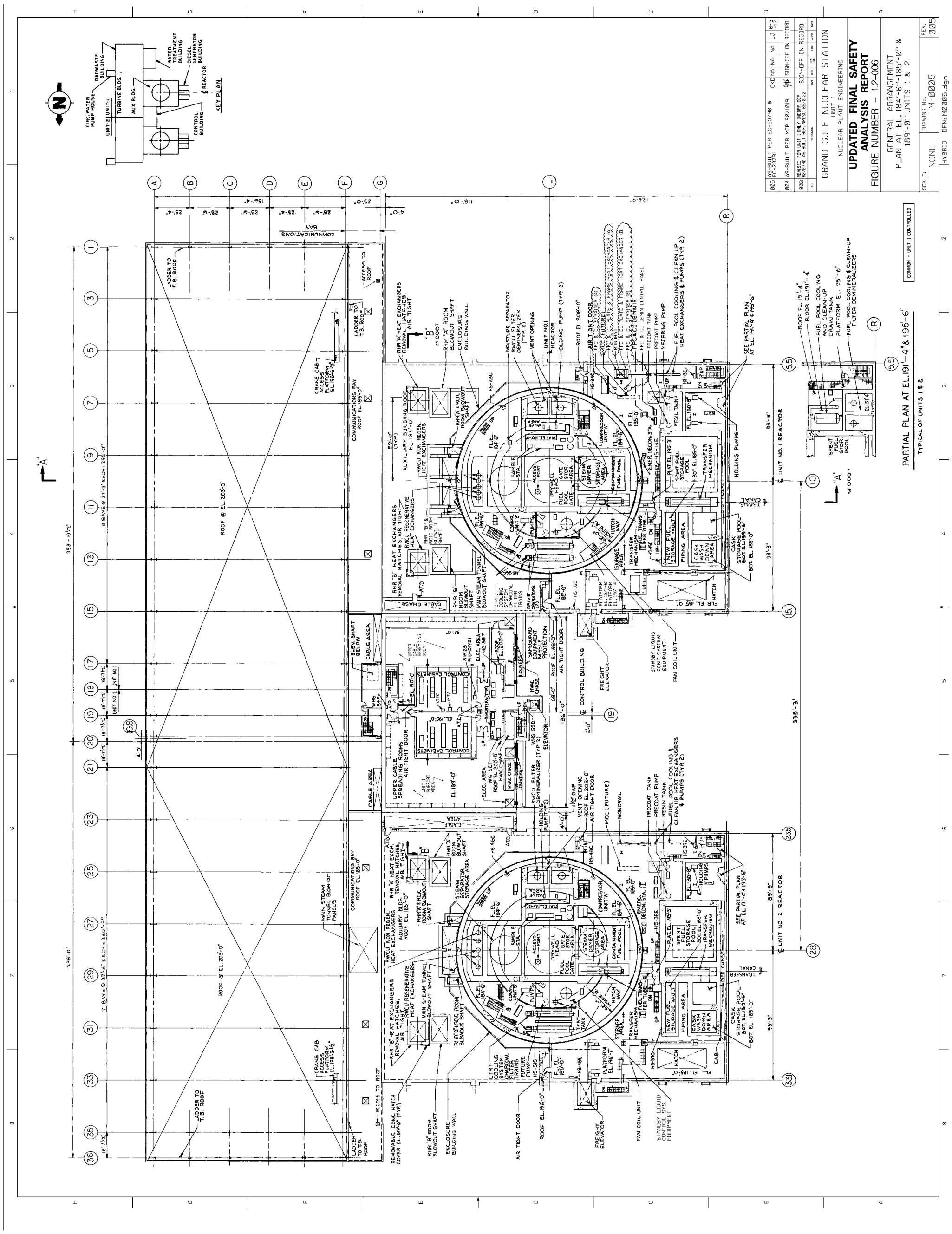
PLAN AT EL. 93'-0" & 100'-9"
UNITS 1 & 2
ACCESS CONTROL AREA

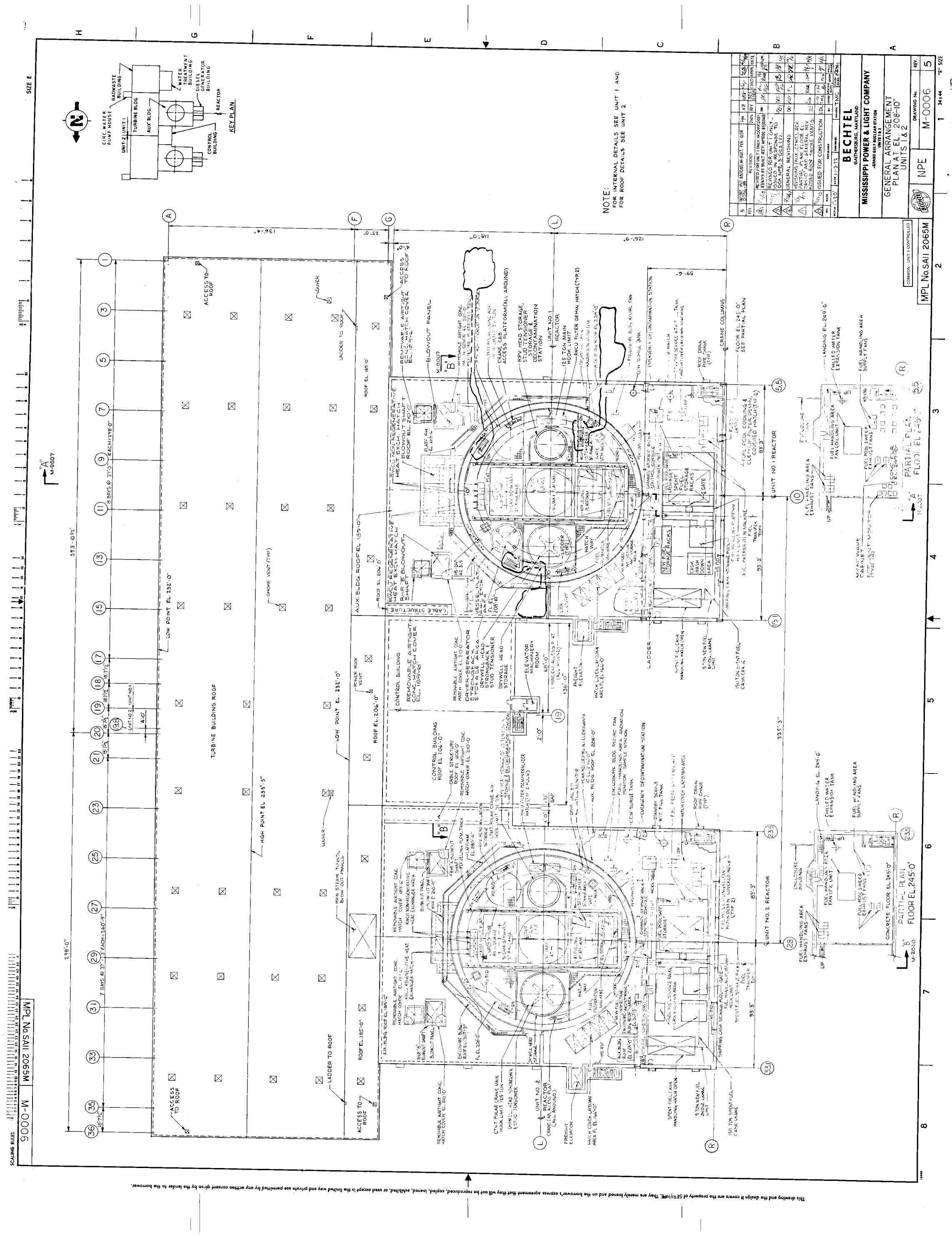


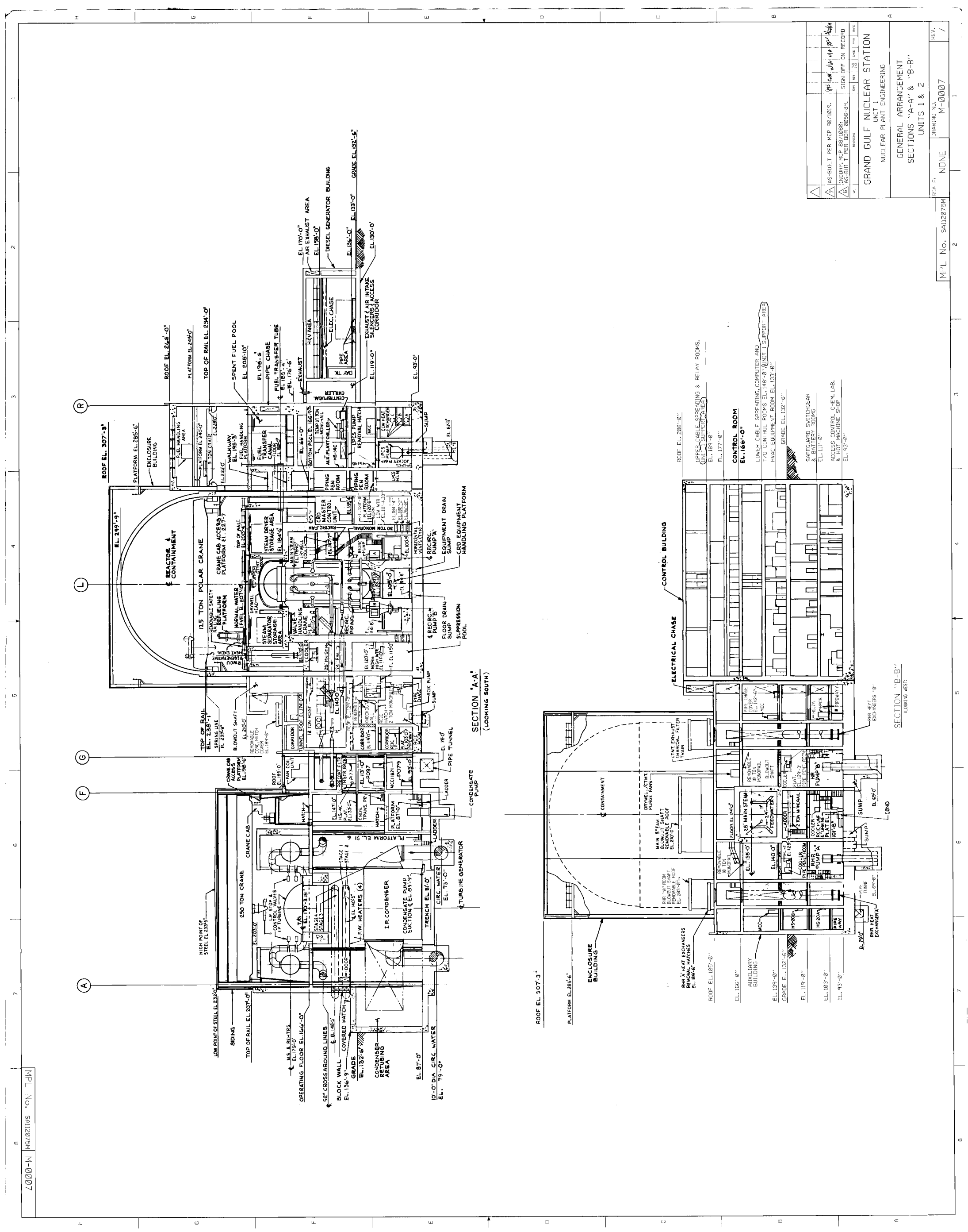


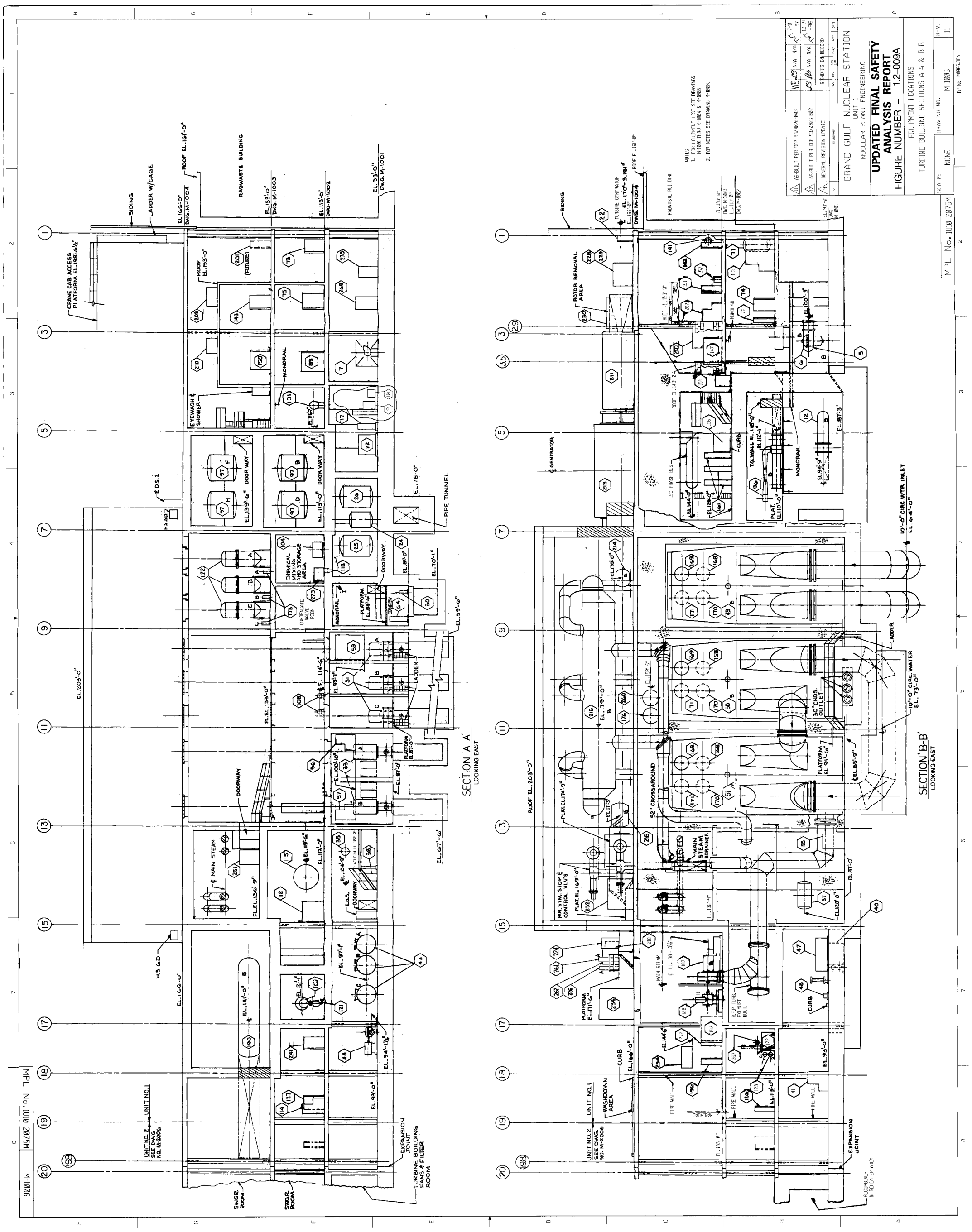


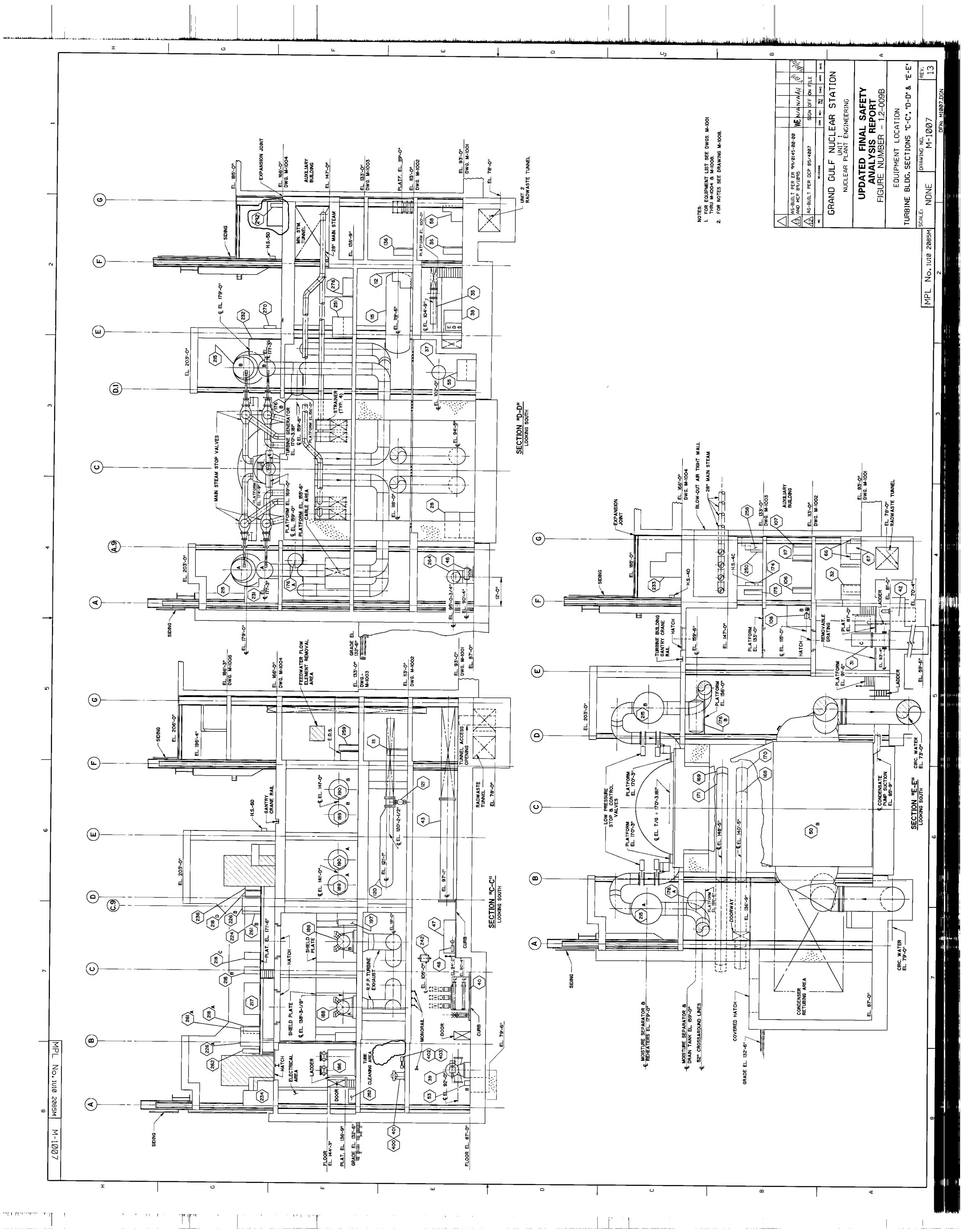
PRIORITY DRAWING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 1.2-06
CONTROL ROOM AREA

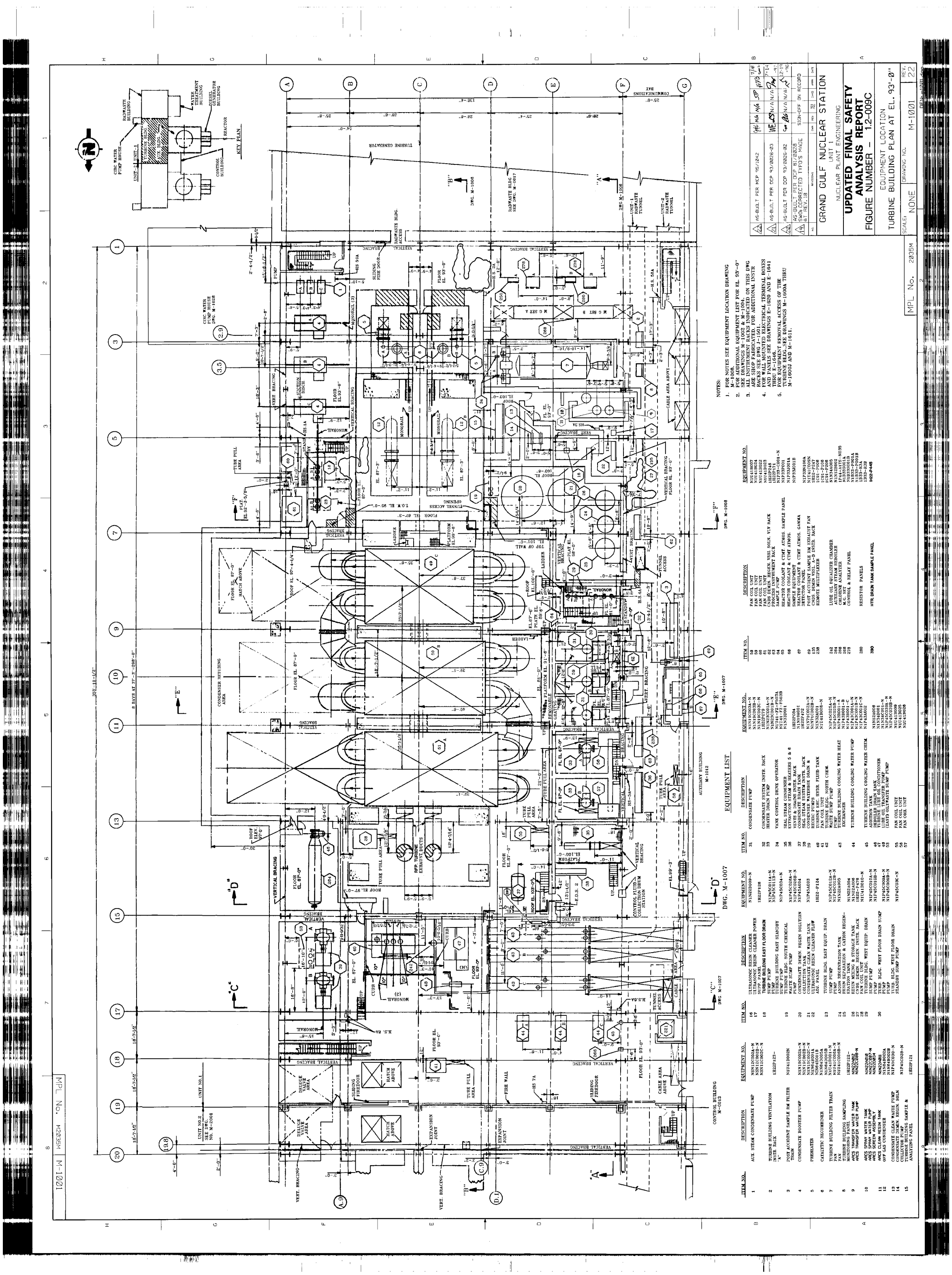


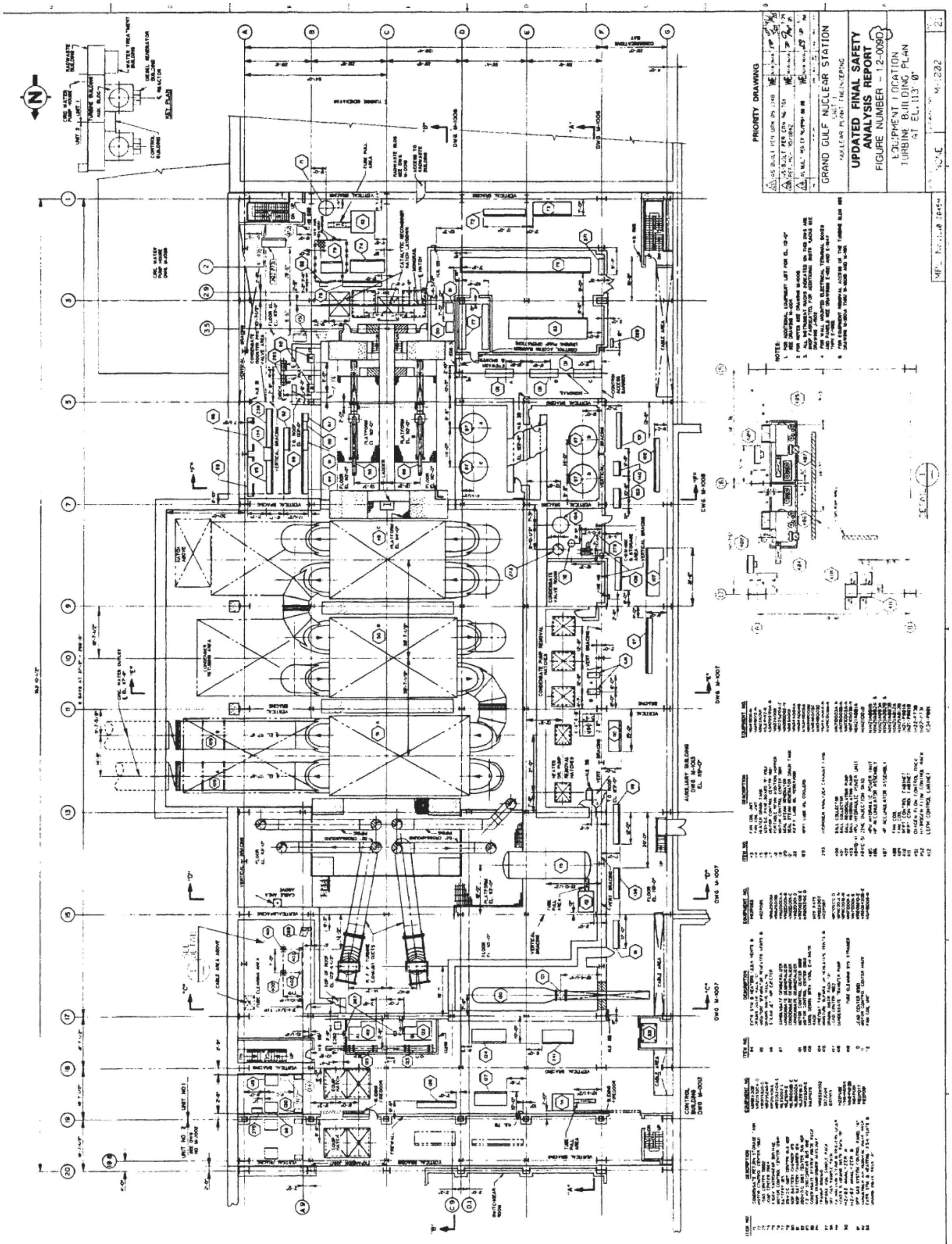

BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
GENERAL ARRANGEMENT
PLAN AT EL. 208'-10"
UNITS 1 & 2
M-0006
MPL No. SAII 2065M
KEY PLAN
TURBINE BUILDING ROOF
UNIT NO. 1 REACTOR
UNIT NO. 2 REACTOR


GRAND GULF NUCLEAR STATION
GENERAL ARRANGEMENT
SECTIONS "A-A" & "B-B"
UNITS 1 & 2
M-0007
SECTION "A-A"
SECTION "B-B"


GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 1.2-009A
EQUIPMENT LOCATIONS
TURBINE BUILDING SECTIONS A-A & B-B
SECTION 'A-A' LOOKING EAST
SECTION 'B-B' LOOKING EAST


GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 1.2-009B
EQUIPMENT LOCATION
TURBINE BLDG. SECTIONS 'C-C', 'D-D' & 'E-E'
SCALE: NONE
DRAWING NO. M-1007
REV. 13
AS-BUILT PER ER 99/0145-00-00 AND MCP 89/1095
AS-BUILT PER DCP 85/4007
SECTION "C-C" LOOKING SOUTH
SECTION "D-D" LOOKING SOUTH
SECTION "E-E" LOOKING SOUTH
MPL No. 1U10 2085M
DFN: M1007.DGN


GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 1.2-009C
EQUIPMENT LOCATION
TURBINE BUILDING PLAN AT EL. 93'-0"
M-1001
EQUIPMENT LIST
NOTES: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GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
TURBINE BUILDING PLAN

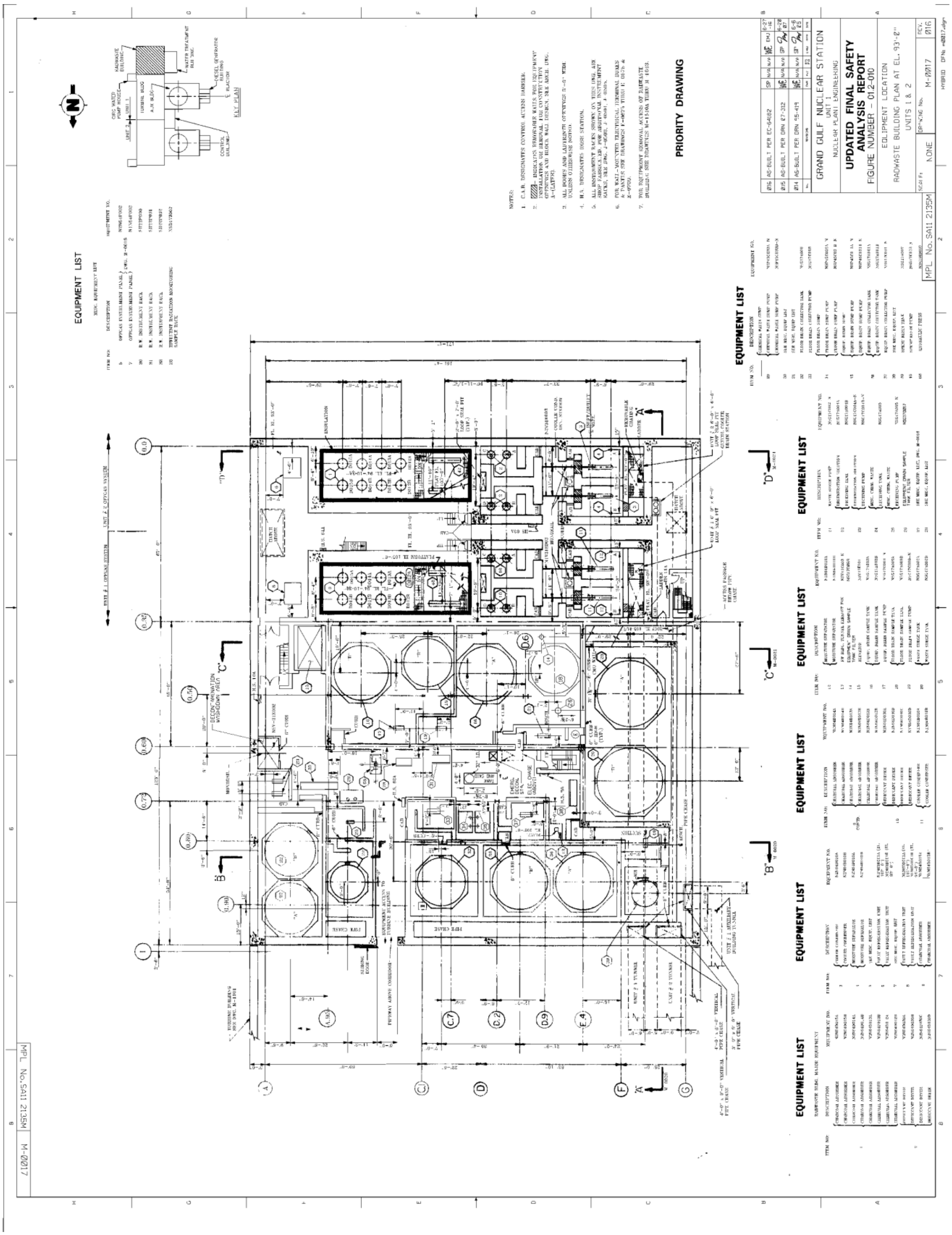
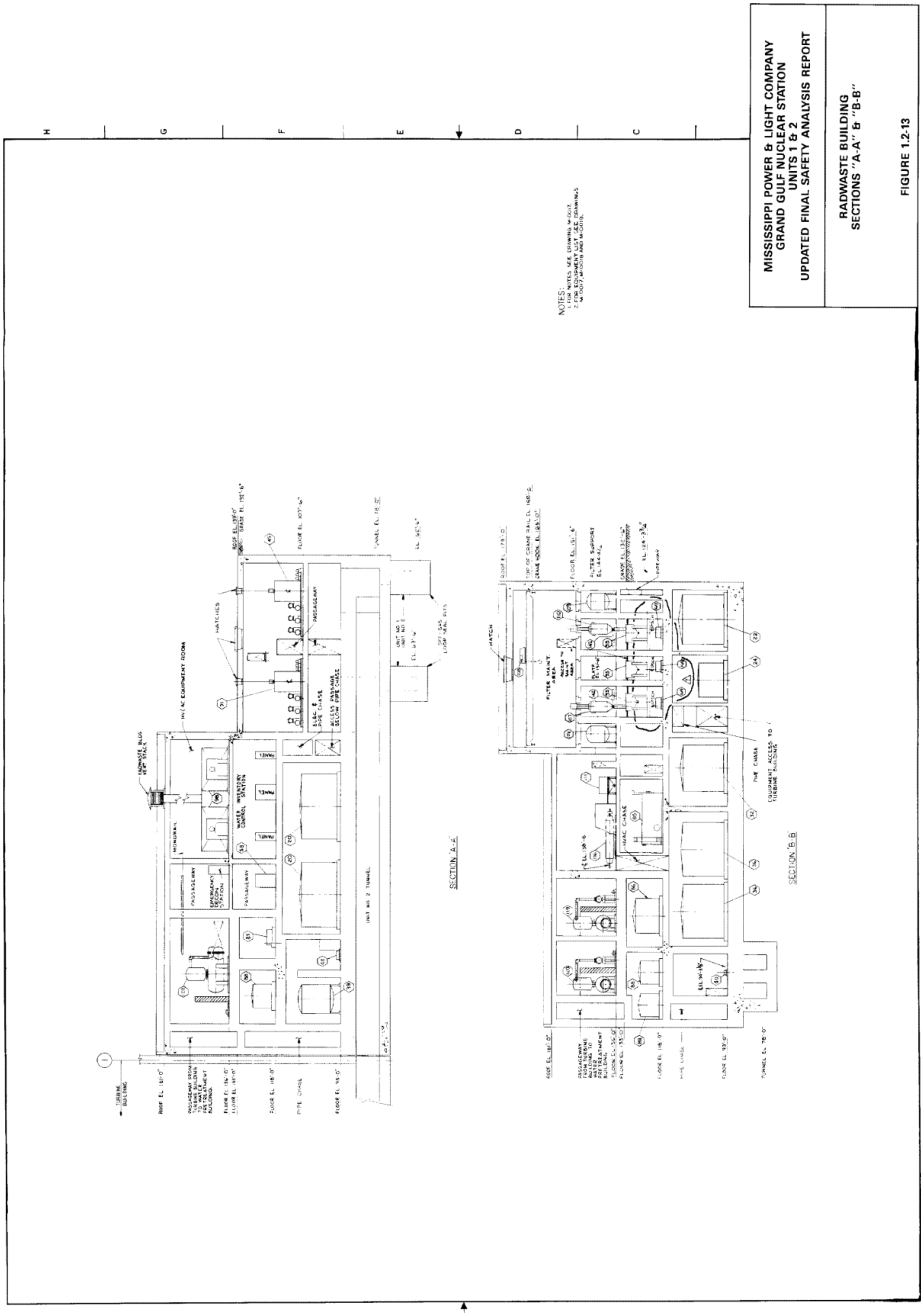

GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 012-010
EQUIPMENT LOCATION
RADWASTE BUILDING PLAN AT EL. 93'-0"
UNITS 1 & 2
M-0017
MPL No. SA11 2135M
PRIORITY DRAWING
EQUIPMENT LIST

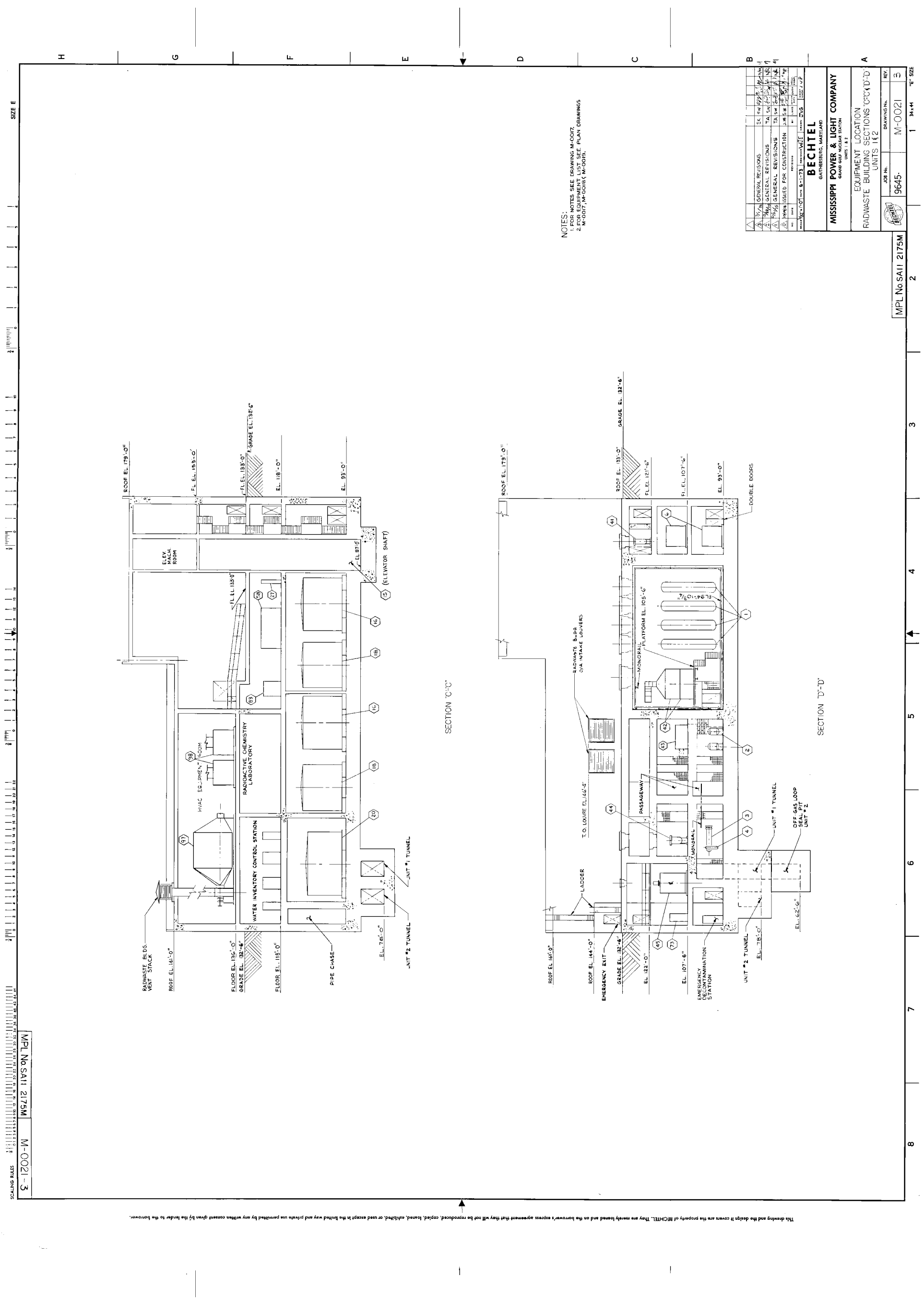
FIGURE 1.2-11: Deleted

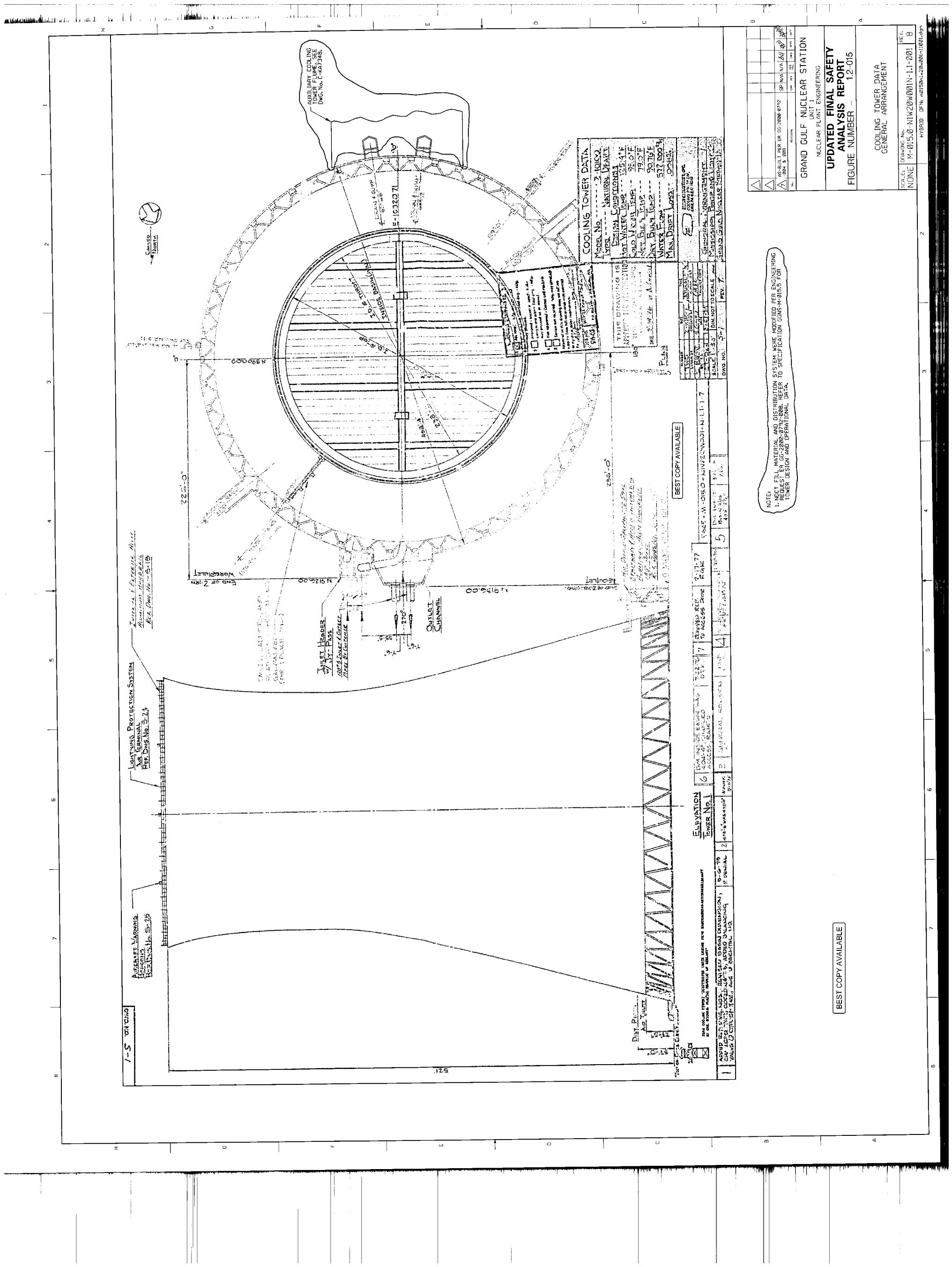
(See Figure 12.3-6)

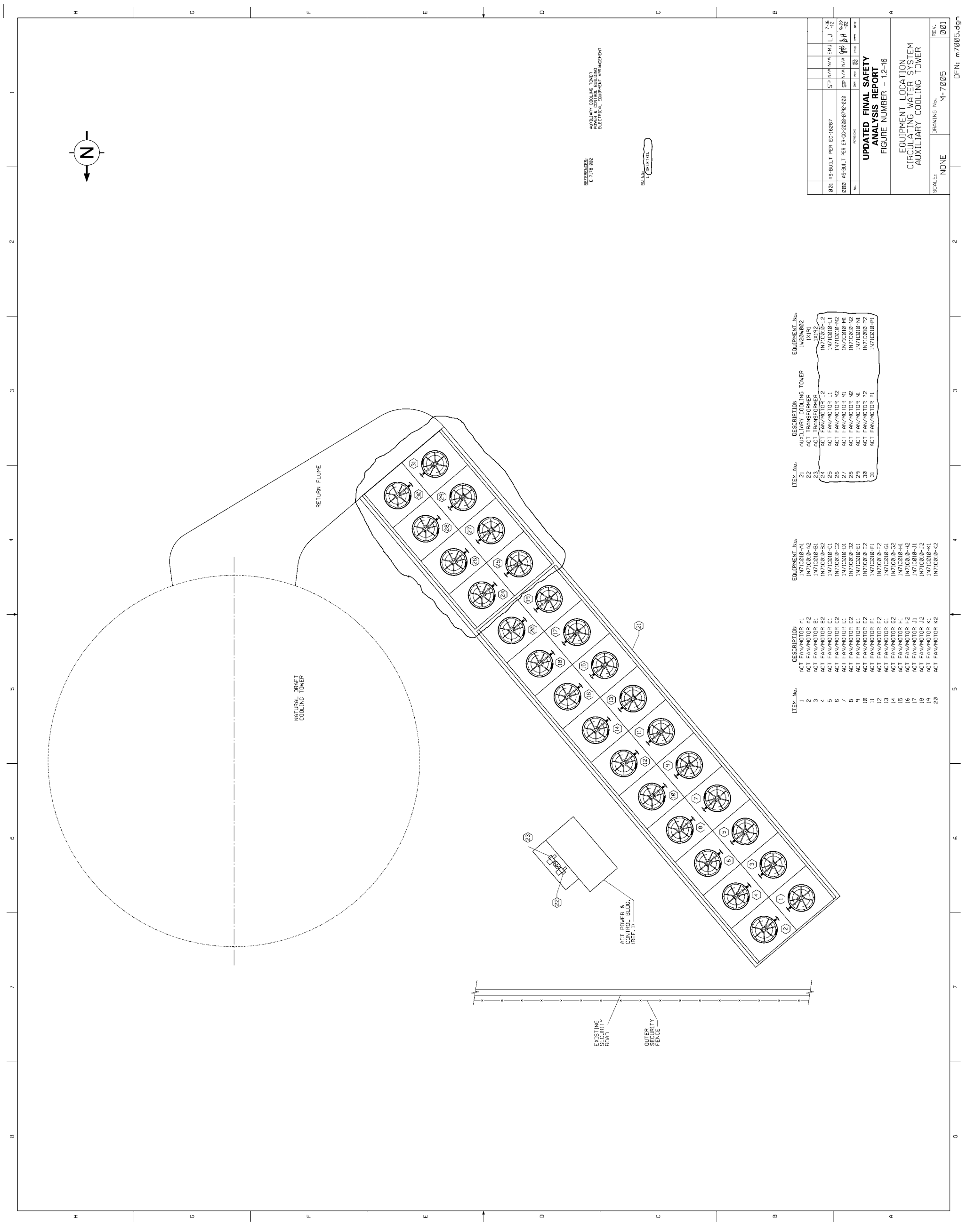
FIGURE 1.2-12: Deleted

(See Figure 12.3-7)


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
RADWASTE BUILDING
SECTIONS "A-A" & "B-B"
FIGURE 1.2-13
SECTION 'A-A'
SECTION 'B-B'




GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER 1.2-015
COOLING TOWER DATA
GENERAL ARRANGEMENT
COOLING TOWER DATA
BEST COPY AVAILABLE


001 AS-BUILT PER EC-16287
000 AS-BUILT PER ER-GG-2000-0792-000
UPDATED FINAL SAFETY
ANALYSIS REPORT
FIGURE NUMBER - 1.2-16
EQUIPMENT LOCATION
CIRCULATING WATER SYSTEM
AUXILIARY COOLING TOWER
SCALE: NONE
DRAWING No. M-7005
REV. 001
DFN: m7005.dgn
NATURAL DRAFT COOLING TOWER
RETURN FLUME
ACT POWER & CONTROL BLDG.
EXISTING SECURITY ROAD
OUTER SECURITY FENCE

1.3 COMPARISON TABLES

1.3.1 Comparisons with Similar Facility Designs

[HISTORICAL INFORMATION] [This subsection highlights the principal design features of the plant and compares its major features with other boiling water reactor facilities. The design of this facility is based on proven technology obtained during the development, design, construction; and operation of boiling water reactors of similar types. The data, performance, characteristics, and other information presented here represent a current, firm design. The comparisons presented here were considered valid at the time the operating license was issued.]

1.3.1.1 Nuclear Steam Supply System Design Characteristics

[HISTORICAL INFORMATION] [Table 1.3-1 summarizes the design and operating characteristics for the nuclear steam supply systems. Parameters are related to rated power output for a single plant unless otherwise noted.]

1.3.1.2 Power Conversion System Design Characteristics

[HISTORICAL INFORMATION] [Table 1.3-2 compares the power conversion system design characteristics.]

1.3.1.3 Engineered Safety Features Design Characteristics

[HISTORICAL INFORMATION] [Table 1.3-3 compares the engineered safety features design characteristics.]

1.3.1.4 Containment Design Characteristics

[HISTORICAL INFORMATION] [Table 1.3-4 compares the containment design characteristics.]

1.3.1.5 Radioactive Waste Management Systems Design Characteristics

[HISTORICAL INFORMATION] [Table 1.3-5 compares the radioactive waste management design characteristics.]

1.3.1.6 Structural Design Characteristics

[HISTORICAL INFORMATION] [Table 1.3-6 compares the structural design characteristics.]

1.3.1.7 Instrumentation and Electrical Systems Design Characteristics

[HISTORICAL INFORMATION] [Table 1.3-7 compares the electrical systems design characteristics. Table 7.1-2 compares the instrumentation and control systems design characteristics.]

1.3.2 Comparison of Final and Preliminary Information

All of the significant changes that have been made in the facility design since submission of the PSAR are listed in Table 1.3-8. Each item in Table 1.3-8 is cross-referenced to the appropriate portion of the FSAR.

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
THERMAL AND HYDRAULIC DESIGN				
(See Section 4.4)				
Rated power, MWt	3833	2436	2436	3579
Design power, MWt (ECCS design basis)	3993	2550	2550	3758
Steam flow rate, lb/hr	16.419×10^6	$10.03 \text{ X } 10^6$	$10.477 \text{ X } 10^6$	$15.396 \text{ X } 10^6$
Core coolant flow rate, lb/hr	112.5×10^6	$78.5 \text{ X } 10^6$	$78.5 \text{ X } 10^6$	$105.0 \text{ X } 10^6$
Feedwater flow rate, lb/hr	16.379×10^6	$10.445 \text{ X } 10^6$	10.477×10^6	15.358×10^6
System pressure, nominal in steam dome, psia	1045	1020	1020	1040
Average power density, kW/liter	54.1	51.2	50.51	56.0
Maximum thermal output, kW/ft	13.4	13.4	13.4	13.4
Average thermal output, kW/ft	5.92	7.11	5.45	6.04
Maximum heat flux, Btu/hr-ft^2	362,000	428,300	354,000	354,300
Average heat flux, Btu/hr-ft^2	159,700	159,700	143,900	159,600
Maximum UO_2 temperature, F	3430	4380	3325	3337
Average volumetric fuel temperature, F	1100	1100	1100	1100
Average cladding surface temperature, F	558	558	558	558
THERMAL AND HYDRAULIC DESIGN				

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
Minimum critical power ratio (MCPR)	1.23	1.9*	1.21	1.24
Coolant enthalpy at core inlet, Btu/lb	527.9	526.2	527.4	527.9
Core maximum exit voids within assemblies	76	79	75	76
Core average exit quality,% steam	14.7	12.9	13.6	14.9
Feedwater temperature, F	417	387.4	420	420
Design Power Peaking Factor				
(See Section 4.4)				
Maximum relative assembly power	1.40	1.40	1.40	1.40
Local peaking factor	1.13	1.24	1.24	1.13
Axial peaking factor	1.40	1.5	1.4	1.40
Total peaking factor	2.26	2.6	2.43	2.22
NUCLEAR DESIGN (First Core)				
(See Section 4.3)				
Water/UO_2 volume ratio (cold)	2.70	2.53	2.41	2.70
Reactivity with strongest control rod out, k_{eff}	<0.99	<0.99	<0.99	<0.99
*For Hatch, minimum critical heat flux ratio (MCHFR) was used.				
Moderator void coefficient				
Hot, no voids, $\Delta k/k$ - % void	-1.0×10^{-3}	-1.0×10^{-3}	-1.0×10^{-3}	-0.3×10^{-5}

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
At rated output, Δk/k - % void	-1.6×10^{-3}	-1.6×10^{-3}	-1.6×10^{-3}	-1.0×10^{-5}
Fuel temperature doppler coefficient				
At 68 F, Δk/k - F fuel	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}	-1.6×10^{-5}
Hot, no voids, Δk/k - F fuel	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}	-1.3×10^{-5}
At rated output, Δk/k - F fuel	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}	-1.2×10^{-5}
Initial average U-235 enrichment wt. %	1.70	2.23	1.90	1.90
Fuel average discharge exposure, MWd/short ton	15,000	19,000	15,053	13,000*
CORE MECHANICAL DESIGN (Initial GGNS Core)				
Fuel Assembly				
(See Section 4.2)				
Number of fuel assemblies	800	560	560	732
Fuel rod array	8 x 8	7 X 7	8 X 8	8 X 8
*Average - first core				
CORE MECHANICAL DESIGN (Continued)				
Overall dimensions, in.	176	176	176	176
Weight of UO_2 per assembly lb (pellet type)	458 (chamfered)	490.4 (undished) 483.4 (dished)	465.15	472 (Chamfered)

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
Weight of fuel assembly, lb	699 (including channel)	681 (undished) 675 (dished)	698	
Fuel Rods				
(See Section 4.2) Number per fuel assembly	62	49	63	63
Outside diameter, in.	0.483	0.563	0.493	0.493
Cladding thickness, in.	0.032	0.032	0.034	0.034
Gap, pellet to cladding, in.	0.0045	0.006	0.0045	0.009
Length of gas plenum, in.	10	16	14	12
Cladding material*	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2

*Free-standing loaded tubes

CORE MECHANICAL DESIGN (Continued)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
Fuel Pellets				
(See Section 4.2) Material	UO_2	UO_2	UO_2	UO_2
Density, % of theoretical	95	95	95	94
Diameter, in.	0.410	0.487	0.416	0.416
Length, in.	0.410	0.5	0.420	0.420

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
Fuel Channel				
(See Section 4.2)				
Overall dimension, length, in.	166.9	166.9	166.9	166.9
Thickness, in.	0.120	0.080	0.100	0.120
Cross section dimensions, in.	5.46 x 5.46	5.44 X 5.44	5.48 X 5.48	5.52 X 5.52
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core Assembly				
(See Section 4.2)				
Fuel weight as UO_2, lb.	366,400	272,850	260,538	345,500
Core diameter (equivalent), in.	191.5	160.2	160.2	183.2
Core height (active fuel), in.	150	144	146	148
CORE MECHANICAL DESIGN (Continued)				
Reactor Control System				
(See Chapters 4 and 7)				
Method of variation of reactor power	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 218-560**	**GESSAR BWR 6 238-732**
Number of movable control rods	193	137	137	177
Shape of movable control rods	Cruciform	Cruciform	Cruciform	Cruciform
Pitch of movable control rods	12.0	12.0	12.0	12.0
Control material in movable rods	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes
Type of control rod drives	Bottom entry locking piston	Bottom entry locking piston	Bottom entry locking piston	Bottom entry locking piston
CORE MECHANICAL DESIGN (Continued)				
Type of temporary reactivity control for initial core	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods
Incore Neutron Instrumentation				
(See Chapters 4 and 7)				
Number of incore neutron detectors (fixed)	176	124	124	164
Number of incore detector assemblies	44	31	31	41

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
Number of detectors per assembly	4	4	4	4
Number of flux mapping neutron detectors	5	4	4	5
Range (and number) of detectors				
Source range monitor	Source to 0.001% power (6)	Source to 0.001% power (4)	Source to 0.001% power (4)	Source to 0.001% power
Intermediate range monitor	0.001% to 10% power (8)	0.001% to 10% power (8)	0.001% to 10% power (8)	0.001% to 10% power
CORE MECHANICAL DESIGN (Continued)				
Local power range monitor	5% to 125% power (176)	5% to 125% power (124)	5% to 125% power (124)	5% to 125% power
Average power range monitor	2.5% to 125% power (8)*	2.5% to 125% power (6)*	2.5% to 125% power (6)*	2.5% to 125% power*
Number and type of incore neutron sources	7 Sb-Be	5 Sb-Be	5 Sb-Be	7 Sb-Se
REACTOR VESSEL DESIGN				

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
(See Section 5.3)				
Material	Low-alloy/ Steel stainless clad	Carbon steel/ stainless clad	Carbon steel/ stainless clad	Carbon steel/ stainless clad
Design pressure, psig	1250	1265	1250	1250
Design temperature, F	575	575	575	575
Inside diameter, ft-in.	20-11	18-2	18-2	19-10
Inside height, ft-in.	73-0	69-4	69-4	70-10

* Channels of monitors from LPRM detectors

REACTOR VESSEL DESIGN (Continued)				
Minimum base metal thickness (cylindrical section), in.	6.14	5.53	5.375	5.70
Minimum cladding thickness, in	1/8	1/8	1/8	1/8
Reactor Coolant Recirculation Design				
(See Chapter 5)				
Number of recirculation loops	2	2	2	2
Design pressure:				
Inlet leg, psig	1250	1148	1250	1250

TABLE 1.3-1: [HISTORICAL INFORMATION] COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
Outlet leg, psig	1625*; 1525**	1274	1675*; 1575**	1675*; 1575**
Design temperature, °F	575	562	575	575
Pipe diameter, in.	24	28	20	22/24
Pipe material, ANSI	304/316	304/316	304/316	304
Recirculation pump flow rate, gpm	44,900	42,200	33,880	35,400
Number of jet pumps in reactor	24	20	20	20

* Pump and discharge piping to and including discharge block valve

** Discharge piping from discharge block valve to vessel

MAIN STEAMLINES

(See Section 5.4)

Number of steamlines	4	4	4	4
Design pressure, psig	1250	1146	1250	1250
Design temperature, F	575	563	575	575
Pipe diameter, in.	28	24	24	26
Pipe material	Carbon steel	Carbon steel	Carbon steel	Carbon steel

TABLE 1.3-2: [HISTORICAL INFORMATION] COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

	GG	Bailly	Limerick	Zimmer
Turbine Generator (See Section 10.2)				
Net generator output (MW)	1331.5	626	1,092	835.9
Turbine cycle heat rate (Btu/KW-hr)	9815	9602	10,287	9959
Type/LSB length (line)	TC6F-46	TC4F/28	TC6F/38	TC4F/40
Cylinders (No.)	1-HP, 3-LP	1-HP, 2-LP	1-HP, 3-LP	1-HP, 2-LP
Steam Conditions at throttle valve				
Flow (lb/hr)	15.655×10^6	8.29×10^6	14.14×10^6	10.477×10^6
Pressure (psia)	997.86	965	965	965
Temperature (F)	544	510	540	540
Moisture Content (%)	0.66	0.40	0.40	0.40
Turbine cycle arrangement (See Section 10.4)				
Steam reheat stages (No.)	2	2	None	2
Feedwater heating stages (No.)	6	6	6	6
Strings of feedwater heaters (No.)	2-HP, 3-LP	2	3	2
Heaters in condenser necks (No.)	4	1	2	1
Heater drain system	Pumped forward	Pumped forward	Cascade	Pumped forward
Condensate pumps (No.)	3	3	3	3
Condensate booster pumps (No.)	3	3	None	3
Heater drain pumps (No.)	2	2	None	2
Reactor feed pumps (No.)	2	2	3	2
Main steam line				
Steam lines (No.)	4	4	4	4
Design pressure (psig)	1250	1250	1250	1250
Design temperature (F)	575	575	575	575
Pipe diameter (in.)	28	20	26	24
Pipe material	Carbon steel	Carbon steel	Carbon steel	Carbon steel
Main steam bypass capacity (%)	35	25	25	25

TABLE 1.3-2: [HISTORICAL INFORMATION] COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS (CONTINUED)

	GG	**Bailly**	**Limerick**	**Zimmer**
Final feedwater temperature (F)	417	420	420	420
Condenser (See Section 10.4)				
Type	Multiple pressure	Single pressure	Multiple pressure	Single pressure
Condenser shells (No.)	3	2	3	2
Design pressure (in. Hg abs)	3.62/2.91/2.37	3.2	2.81/3.56/4.67	3.5
Total condenser duty (Btu/hr)	8.506×10^9	4.25×10^9	7.8×10^9	7.053×10^9
Circulating water system (Section 10.4)				
Type	Closed/ND & Mech Draft cooling tower	Closed/ND cooling tower	Closed/ND cooling tower	Closed/ND cooling tower
Flow (gpm)	572,000	376,000	113,000 (each)	450,000
Circulating water pumps (No.)	2	2 (1/2 capacity)	4 (1/2 capacity)	3

TABLE 1.3-3: [HISTORICAL INFORMATION] COMPARISON OF ENGINEERED SAFETY FEATURES DESIGN CHARACTERISTICS

EMERGENCY CORE COOLING SYSTEMS	**GRAND GULF BWR 6 251-800**	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 218-560**	**GESSAR BWR 6 238-732**
(Systems sized on design power)				
Low Pressure Core Spray Systems (See Section 6.3)				
Number of loops	1	2	1	1
Flow rate, gpm	7115 at 128 psid	4625 at 120 psid	4725 at 119 psid	6000 at 122 psid
High Pressure Core Spray System (See Section 6.3)				
Number of loops	1	1[a]	1	1
Flow rate, gpm	1650 at 1147 psid	4250	1330 at 1110 psid	1465 at 1130 psid
	7115 at 200 psid		4725 at 200 psid	6000 at 200 psid
Automatic Depressurization System (See Section 6.3)				
Number of relief valves	8	7	7	8
Low Pressure Coolant Injection[b] (See Section 6.3)				

TABLE 1.3-3: [HISTORICAL INFORMATION] COMPARISON OF ENGINEERED SAFETY FEATURES DESIGN CHARACTERISTICS (CONTINUED)

EMERGENCY CORE COOLING SYSTEMS	**GRAND GULF BWR 6 251-800**	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 218-560**	**GESSAR BWR 6 238-732**
Number of loops	3	2	3	3
Number of pumps	3	4	3	3
Flow rate, gpm/pump	7450 at 24 psid	7700 at 20 psid	5050 at 20 psid	7100 at 20 psid
<u>AUXILIARY SYSTEMS</u>				
<u>Residual Heat Removal System</u> (See Section 5.4) Reactor Shutdown cooling Mode:				
Number of loops	2	2	2	2
Number of pumps	2	4	2	2
Flow rate, gpm/pump[c]	7450	7700	5050	7100
Duty, Btu/hr/heat exchanger[d]	50×10^6	32×10^6	30.8×10^6	45.0×10^6
Number of heat exchangers	2	2	2	2
Primary containment cooling mode:				

TABLE 1.3-3: [HISTORICAL INFORMATION] COMPARISON OF ENGINEERED SAFETY FEATURES DESIGN CHARACTERISTICS (CONTINUED)

EMERGENCY CORE COOLING SYSTEMS	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
Flow rate, gpm	7450[e]	30,800	5050[e]	7100[e]
Standby Service Water System (See Section 9.2)				
Flow rate, gpm/heat exchanger	25,300	8000	5000	7000
Number of pumps	3 (2 @ 12,000 gpm) (1 @ 1,300 gpm)	4	4	5
Reactor Core Isolation Cooling System (See Section 5.4)				
Flow rate, gpm	800 at 1120 psid	400 at 1120 psid	400 at 1120 psid	700 at 1120 psid
Fuel Pool Cooling and Cleanup System (See Section 9.1)				
Capacity, Btu/hr	15.0×10^6	5.7×10^6	6.6×10^6	11.8×10^6

Notes
[a]High-pressure cooling injection system utilized

TABLE 1.3-3: [HISTORICAL INFORMATION] COMPARISON OF ENGINEERED SAFETY FEATURES DESIGN CHARACTERISTICS (CONTINUED)

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
EMERGENCY CORE COOLING SYSTEMS				

[b]A mode of the RHR system

[c]Capacity during reactor flooding mode with more than one pump running

[d]Heat exchanger duty at 20 hours following reactor shutdown

[e]Flow per heat exchanger

TABLE 1.3-4: [HISTORICAL INFORMATION] COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS (See Chapter 3)

	Grand Gulf	Zimmer	Bailly	Limerick
Type	Mark III. Reinforced concrete containment, but with pressure suppression. Containment encloses drywell and suppression pool.	Mark II. Over-and-under primary containment, enclosing drywell and suppression pool. Enclosed by reactor building.	Mark II. Over-and-under primary containment, enclosing drywell and suppression pool. Enclosed by reactor building.	Mark II. Over-and-under primary containment, enclosing drywell and suppression pool. Enclosed by reactor building.
Leak rate (%/day)	0.35	0.5	0.5	0.5
Containment				
Construction	Reinforced concrete cylindrical structure (not prestressed) with hemispherical head; steel lined.	Not applicable	Not applicable	Not applicable
Internal design temperature (F)	185	Not applicable	Not applicable	Not applicable
Design pressure (psig)	15	Not applicable	Not applicable	Not applicable
Free (air) volume (cu ft)	1.40×10^6 (excluding drywell)	Not applicable	Not applicable	Not applicable
Drywell				
Construction	Reinforced concrete. Basically cylindrical; Flat concrete roof with a steel refueling head	Prestressed concrete. Drywell is frustum of a cone; steel lined.	Prestressed concrete. Drywell is frustum of a cone; steel lined.	Prestressed concrete. Drywell is frustum of a cone; steel lined.

TABLE 1.3-4: [HISTORICAL INFORMATION] COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS (See Chapter 3) (Continued)

	Grand Gulf	**Zimmer**	**Bailly**	**Limerick**
Internal design temperature (F)	330	340	340	340
Design pressure (psig)	30	+45, -2	+45, -2	+55, -5
Free (air) volume, total (cu ft)	270,000	287,000	263,800	390,450
Suppression Pool				
Construction	Reinforced concrete, steel lined. Basically cylindrical.	Prestressed concrete. Pool is cylindrical; steel lined.	Prestressed concrete. Pool is cylindrical; steel lined.	Prestressed concrete. Pool is cylindrical; steel lined.
Internal design temperature (F)	185	340	340	340
Design pressure (psig)	15	+45, -2	+45, -2	+55, -5
Water volume (cu ft)	136,000	106,000	73,500	122,400
Break area/total vent area	0.008	0.008	0.012	0.019

TABLE 1.3-5: [HISTORICAL INFORMATION] RADIOACTIVE WASTE MANAGEMENT SYSTEMS DESIGN CHARACTERISTICS

	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
GASEOUS RADWASTE			
(See Section 11.3)			
Design Bases, noble gases, µCi/sec	100,000 at 30 min	100,000 at 30 min	100,000 at 30 min
Process treatment	Chilled charcoal	Recombiner Ambient Charcoal	Chilled Charcoal
Number of beds	8	12	5
Design condenser In-leakage, cfm	40	40	12.5
Release point-height above ground, ft	31.5 (Radwaste Bldg)	394	172
LIQUID RADWASTE			
(See Section 11.2)			
Treatment of:			
1. Floor drains	Filtered, demineralized, evaporated, and returned to condensate storage	F, D, and R	F, E, and R
2. Equipment drains	Filtered, demineralized, evaporated, and returned to condensate storage	F, D, and R	F, D, and R

* See legend, Sheet 2

TABLE 1.3-5: [HISTORICAL INFORMATION] RADIOACTIVE WASTE MANAGEMENT SYSTEMS DESIGN CHARACTERISTICS (Continued)

LIQUID RADWASTE (Cont.)	**GRAND GULF BWR 6 251-800**	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 218-560**
3. Chemical drains	Neutralized, evaporated, and returned to equipment drain collector tank	F, discharged E, solid to radwaste	E, D, concentrates to solid radwaste distillate R
4. Laundry drains	NONE (Laundry will be processed offsite by an authorized contractor.)	Diluted and sent to circulating water discharge	Reverse osmosis discharge
5. Expected annual avg. release, µCi (excluding tritium)	110,000	20,000	10,900

*Legend:

D = demineralized
F = filtered
E = evaporator/concentrator
R = recycled, i.e., returned to condensate storage

TABLE 1.3-6: [HISTORICAL INFORMATION] COMPARISON OF STRUCTURAL DESIGN CHARACTERISTICS

	GRAND GULF	HATCH 1	ZIMMER	GESSAR
	BWR 6	BWR 4	BWR 5	BWR 6
Seismic Design	251-800	218-560	218-560	238-732
(See Sections 3.2 and 3.7)				
Operating Basis Earthquake				
- horizontal g	0.075	0.08	0.10	0.15
- vertical g	0.05	0.05	0.07	0.15
Safe shutdown earthquake				
- horizontal g	0.15	0.15	0.20	0.3
- vertical g	0.10	0.10	0.14	0.3
Wind Design				
(See Section 3.3)				
Maximum sustained - mph	90	105	90	130
Tornados (See Section 3.3)				
Translational - mph	60	60	60	70
Tangential - mph	300	300	300	290

TABLE 1.3-7: [HISTORICAL INFORMATION] COMPARISON OF ELECTRICAL SYSTEMS (See Chapter 8)

System	Grand Gulf	Bailly	Zimmer	Limerick
	(1 unit)	(1 unit)	(2 unit)	(2 unit)
Number of offsite circuits	3	9	4	4
Number of auxiliary power sources	3 service transformers (1 exclusively for esf)	3-1 unit auxiliary transformers 1 reserve auxiliary transformer 1 emergency reserve auxiliary transformer	2-1 unit auxiliary transformers 2 startup transformers	4-2 unit auxiliary 2 startup transformers
Number of preferred power circuits for esf buses	3	3 (except 2 for HPCS)	2	2
Number of esf buses per unit	3	3	3	4
Number of standby a-c power supplies	3 (1/esf bus)	3 (1/esf bus)	3 (1/esf bus)	4 (1/2) esf buses)
Number of 125 V d-c systems supplying buses	3 (1/esf bus)	3 (1/esf bus)	3 (1/esf bus)	4 (1/2 esf buses)
Sharing of standby power supplies and interconnections between safety buses	None	d-c buses interconnected	None	Diesels, batteries shared between corresponding buses of both units

Note: esf = engineered safety features

TABLE 1.3-8: SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Nuclear fuel	The number of water rods in each fuel bundle has been changed from 1 to 2. Five different U-235 enrichments are now used in the fuel assemblies instead of previous four types.	Improved fuel performance	4.2.2.3.2
Control rod drive position indication	Changed to 11 wire probe and solid state	Improved reliability and increased frequency of checking actual rod position	4.2
Feedwater sparger	The thermal sleeve was changed to provide improved slip fit design of sparger to nozzle.	To eliminate failure, leakage, and provide for possilbe inservice inspection.	5.3
Standby liquid control (SLC) system	Interlocks on the SLC system were revised.	To prevent inadvertent boron injection during system testing.	7.4.1.2, 9.3.5
RCIC system	Each component of the RCIC system has been made capable of functional testing during normal plant operation.	Improved testability	5.4

TABLE 1.3-8: SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR (CONTINUED)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Automatic depressurization system (ADS)	The interlocks on the automatic depressurization system were revised.	To meet IEEE-279 requirements	7.3.1.1
Leak detection system	The leak detection system was revised to upgrade the capability and incorporate the requirements of IEEE-279 Added additional monitors to increase adequacy of detection	To meet IEEE-279 and Reg. Guide 1.45 requirements	7.6.1.4
Control rod drive fast scram	Increased system pressure from 1750 to 2000 psi, enlarged insert/withdraw draw lines, and increased accumulator volume to provide faster scram time	Provides increased reactivity control, especially at end of fuel cycle. Provides increased thermal margin, and reduces amount of operation of steam relief	3.9.4.1, 4.6
Reactor Recirc. pump trip	Pumps tripped on signals from turbine control or stop valves upon generator load rejection or turbine trip	Reduces transient core flow and reactivity. Works with fast scram to provide increased thermal margin	4.6.4, 5.4.1, 7.6.1.8

TABLE 1.3-8: SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR (CONTINUED)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Fuel storage racks	Added 48 more fuel storage castings for use in spent fuel, new fuel and containment pool storage areas	Increases capacity to handle more onsite fuel storage	9.1.1, 9.1.2
Fuel Pool Cooling	Upgraded calculations to verify pool cooling system able to handle increased fuel storage	Provides for increased fuel capacity	9.1.3
Fuel Pool Cooling	Upgraded system (except for filter/demineralizer which can be isolated) to meet Seismic 1 classification	To meet Reg. Guide 1.13	9.1.3
High Pressure Core Spray System	Changed motor control center capacity to handle increased electrical loads	Design improvement	7.3
Reactor Protection System	Changes for control system instrument test ability. Changed from switches to transmitters and added calibration units	Provides improved testability reliability	7.2.2.2

TABLE 1.3-8: SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR (CONTINUED)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Gauged Control Rod Withdrawal	Changed logic and control rod drive hydraulic system to move groups of control rods. Added stabilizing hydraulic valves	Improves operating time for control maneuvering and startup	3.9.4.1, 4.6
Reactor In-Core Monitors	Changed replacement from top to bottom of core monitor entry	Improves time for replacement during outages	7.6.1.5
Reactor Recirc. Pump	Added vibration sensors to record and alarm when high shaft vibration encountered on pump or motor	Improves reliability	Ch 5
Reactor Recirc. Pump Motor Controls	Added Motor-Generator Sets to provide control for reduced flow during startup and shutdown	Provides improved operation	7.7.1.3
Reactor Recirc. System	Removed pump bypass lines for reduction of region potentially sensitive to stainless steel stress corrosion problems	Design Improvement	5.4.3
Feedwater Leakage Control System	Added system to plant	To eliminate through-line bypass leakage	6.7.2

TABLE 1.3-8: SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR (CONTINUED)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Switchyard	Changed configuration of 500 Kv switchyard to provide two 500 Kv offsite sources	Safety Evaluation Report (SER) by NRC	8.2.2
Guard Pipe Assemblies	Design criteria	To comply with BTPs MEB 3-1 and APCSB 3-1	3.6
Pipe Break Criteria	Design criteria	NRC requirement/design improvement	3.6
ISI	ISI criteria	ASME, Code, Section XI requirements	5.2.4, 6.6
Suppression Pool Clean-up System	Added system	To reduce doses inside containment	9.3.6
Tornado Missile Spectrum	Changed spectrum	To comply with SRP 3.5.1.4 Draft Rev 1	3.5.1.4
Containment Leakage	Raised from 0.1%/day to 0.35%/day	Improved meteorological data (see Section 2.3), also, taking credit for Iodine removal via containment sprays (see Section 6.5)	6.2.6
Radial Wells - Intake Structure	Added radial wells, deleted intake structure	To improve water quality of cooling water systems	9.2.10

TABLE 1.3-8: SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR (CONTINUED)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Service Water Discharge Line	Rerouted discharge line into the barge slip	Discharge line originally routed to the middle of the Mississippi River; changed to reduce hazards to shipping	Ch. 2
Auxiliary Building Isolation Valve Arrangement	Removed double isolation valves on smaller piping	Double valve isolation not necessary for SGTS operation based on single failure analysis	6.2.3
Control Room Inleakage	Increased allowable inleakage from 60 scfm to 263 scfm	Improved dose calculations	6.4
Iodine Removal Via Containment Spray	Iodine removal credit accounted for in dose calculations	Improved dose calculations	6.5
Turbine Building Ventilation Charcoal Filters	Deleted	Improved dose calculations show releases are within Appendix I requirements without these filters	9.4, 11.3

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 GGNS Project

[HISTORICAL INFORMATION] [The Grand Gulf Nuclear Station (GGNS) is owned or leased by System Energy Resources, Inc. (SERI) and South Mississippi Electric Power Association (SMEPA). GGNS is operated by Entergy Operations, Inc. (Entergy Operations). SERI and Entergy Operations are wholly owned subsidiaries of Entergy Corporation, formerly Middle South Utilities, Inc. SERI provides financing for construction and maintains title ownership of the facility. Entergy Operations assumes responsibility for design, construction, and operation of the facility.

MP&L, (Now Entergy Mississippi, Inc.), Middle South Energy, Inc. (now System Energy Resources, Inc.) and SMEPA were co-applicants in the licensing proceedings for GGNS Unit 1; Entergy Operations, SERI, Entergy Mississippi, Inc., and SMEPA are co-licensees. Entergy Operations, SERI, MP&L and SMEPA were co-applicants in the licensing proceedings for GGNS Unit 2 prior to cancellation of the Unit 2 construction permit.

During construction of GGNS Unit 1, MP&L did not maintain engineering and construction staffs but used reputable engineering and construction firms for these purposes. For the work covered by this FSAR, Bechtel Corporation was retained to provide engineering, procurement, quality assurance, and construction management services. The engineering firms and consultants used during construction of GGNS Unit 1 are given in the following subsections to Section 1.4. The current GGNS engineering staff is provided in Section 13.1.]

1.4.2 Architect Engineer

[HISTORICAL INFORMATION] [Bechtel Corporation has been continuously engaged in construction or engineering since 1898. For the last 35 years, Bechtel has been active in the fields of pipelines, petroleum, power generation and distribution, harbor development, mining and metallurgy, and chemical and industrial processing. The Bechtel organization has grown progressively to become one of the world's largest engineer-constructors for industrial facilities. Since the close of World War II, Bechtel Corporation has been responsible for the design of over 200 thermal power-generating units; this represents more than 115,000,000 kilowatts of new generating capacity, of which more than 65,000,000 kilowatts are nuclear. Bechtel Corporation is qualified to provide and does provide required services for station design, construction management, equipment procurement, construction, and startup.]

1.4.3 Nuclear Steam Supply System

[HISTORICAL INFORMATION] [The General Electric Company was awarded the contracts to design, fabricate, and deliver the single-cycle, boiling water nuclear steam supply system, to fabricate the first core of nuclear fuel, and to provide technical direction for the installation and startup of this equipment. General Electric has engaged in the development, design, construction, and operation of boiling water reactors (BWR) since 1955. Thus, General Electric has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of the reactors. See Table 1.4-1 for a list of nuclear facilities utilizing GE designed reactors.]

1.4.4 Turbine Generator Vendor

[HISTORICAL INFORMATION] [Allis-Chalmers Power Systems, Inc. (A-CPSI) has supplied the turbine generators and provided technical assistance for installation and startup of this equipment. The A-CPSI is a jointly owned company of Allis-Chalmers Corporation (A-CC) and Kraftwerk Union AG (KWU) of West Germany and employs the extensive experience and capabilities of both of its parent firms in the steam turbine generator and nuclear power field. The technology and design of A-CPSI turbine generators is provided by KWU under a license agreement. In the past 25 years, KWU and its parent firms, Siemens and AEG of Germany (at the beginning of 1977, Siemens purchased the AEG share of KWU and thus became the sole owner), have designed and built nearly 600 steam turbine generator units for fossil-fueled and nuclear power plants. At the present time, KWU and A-CPSI have, either in service or on order, a total of 40 nuclear turbine generators rated 350 Mw or larger for BWR and PWR applications. The design of the turbine generator for Grand Gulf has been based directly on similar KWU units in service or being manufactured at the present time. KWU and A-CC also have related experience in the design and construction of BWR and PWR reactors and complete turnkey nuclear power plants.

The Grand Gulf turbine generator was manufactured partly by A-CC and partly by KWU, and certain components (such as heat exchangers, pumps, motors, and prefabricated piping) have been procured directly by A-CPSI. Manufacturing by A-CC has been conducted primarily at its facilities in West Allis, Wisconsin, where many turbine generators and other equipment have been built. KWU has manufactured its portion at its Muelheim/Ruhr facilities, where turbines and generators are built; and in Erlangen, where electrical control equipment is designed and built.

Technical assistance for installation and startup has been provided by the A-CPSI Product Service Department, which is staffed with personnel trained and experienced in this work.]

1.4.5 Consultants

[HISTORICAL INFORMATION] [Woodward-Clyde Consultants has been retained to assist in evaluating the potential impact that the construction and operation of the nuclear facility has on the environment. They provide environmental consulting services in the areas of meteorology, demography, hydrology, biology, and radiological surveys.

Memphis State University provides, through its Center for Nuclear Studies, instructions which include a training program in Basic Reactor Fundamentals for employees of Entergy Operations as part of their qualifications for licensing as nuclear power plant operating and/or maintenance personnel.

Southern Nuclear Engineering, Inc. was retained to provide engineering consultant services in the areas of:

a. Plant site review

b. Writing of licensing documents

c. Review of licensing documents and Environmental Reports

d. Review of plant component and system designs

e. Design of special equipment and systems

f. Performing or checking calculations required in the design and/or licensing of the nuclear plant(s)

g. Presentation of expert witness testimony or technical information at licensing meetings or hearings and at Public Hearings

h. Design and/or operation of meteorological and other environmental stations

Engineering Data Systems, Inc. (EDS) has been retained to provide consulting engineering services as required by GGNS. These services are typically required in the following areas:

a. Review and assistance in the development of a QA program

b. Review of equipment specifications for QA/QC requirements

c. Review of equipment specifications for seismic requirements

d. Technical assistance in review of nuclear plant systems

EDS shall provide such other consulting engineering services as may be required by GGNS.

Eberline Instrument Corporation (EIC) provides consultation services for radiation exposure control related programs as requested by authorized personnel of Entergy Operations. Such consultation includes but is not necessarily limited to the following activities:

a. Assist with the development of an operating philosophy

b. Assist with the preparation of technical specifications dealing with in-plant exposure control or radioactivity released to the environment. Assist with the preparation of amendments to these technical specifications that will permit maximum flexibility consistent with the Entergy Operations operating philosophy and NRC requirements.

c. Review facility and equipment design to identify potential exposure control problems and suggest modifications that would help limit radiation exposures to “as low as reasonably achievable” (ALARA).

d. Assist in development of radiation protection training programs.

e. Provide backup radiological control personnel for non-routine activities.

f. Help specify instrumentation for radiation exposure control and effluent documentation.

EIC provides other such consulting engineering services as required by GGNS.

General Electric Co. (GEH), I&SE Division, provides consultation and nondestructive testing services in connection with the inservice inspection of the GGNS. This includes:

a. Performance of the inservice inspection.

b. Engineering consulting to assure that the inspection can be performed.

c. Data analysis and appropriate recommendations.

d. Preparation of a criterion used in the design of Class I and Class II systems.

e. Review and comment on drawings and specifications of pipes, welds, hangers, access provisions, insulation, shielding, etc.

f. Preparation of input to documents required for licensing.

g. Assistance to Entergy Operations, Inc. with the Nuclear Regulatory Commission in the area of inservice inspection.

h. Prepare or assist in preparation of bid specifications or requirements for inservice inspection.

i. Keep Entergy Operations updated on Code changes and new codes affecting GGNS and inservice inspection requirements.

Nuclear Services Corporation (NSC) provides engineering services as required by GGNS in the following typical areas:

a. Assist with the development of an operating philosophy

b. Assist with the preparation of technical specifications

c. Assist with preparation of procedures for routine operation, maintenance, inspection, and refueling activities and non-routine or emergency plans

Betz, Calgon, or other contract chemical service companies as required provide consultation services for water chemistry and other chemical analysis as GGNS deems necessary.]

TABLE 1.4-1: [HISTORICAL INFORMATION] COMMERCIAL NUCLEAR REACTORS COMPLETED, UNDER CONSTRUCTION OR IN DESIGN BY GENERAL ELECTRIC

Station	Utility	Rating (MWe)	Year of Order	Year of Startup
Dresden 1	Commonwealth Edison	200	1955	1960
Humboldt Bay	Pacific G&E	69	1958	1963
Kahl	Germany	15	1958	1961
Garigliano	Italy	150	1959	1964
Big Rock Point	Consumers Power	70	1959	1963
JPDR	Japan	11	1960	1963
KRB	Germany	237	1962	1967
Tarapur 1	India	190	1962	1969
Tarapur 2	India	190	1962	1969
GKN	Holland	52	1963	1968
Oyster Creek	JCP&L	640	1963	1969
Nine Mile Point 1	Niagara Mohawk	625	1963	1970
Dresden 2	Commonwealth Edison	809	1965	1970
Pilgrim	Boston Edison	644	1965	1972
Millstone 1	NUSCO	642	1965	1971
Tsuruga	Japan	340	1965	1970
Nuclenor	Spain	440	1965	1971
Fukushima 1	Japan	439	1966	1971
BKW KKM	Switzerland	306	1966	1972
Dresden 3	Commonwealth Edison	809	1966	1971
Monticello	Northern States	545	1966	1971
Quad Cities 1	Commonwealth Edison	800	1966	1972
Browns Ferry 1	TVA	1,098	1966	1974
Browns Ferry 2	TVA	1,098	1966	1974
Quad Cities 2	Commonwealth Edison	800	1966	1972

TABLE 1.4-1: [HISTORICAL INFORMATION] COMMERCIAL NUCLEAR REACTORS COMPLETED, UNDER CONSTRUCTION OR IN DESIGN BY GENERAL ELECTRIC (CONTINUED)

Station	Utility	Rating (MWe)	Year of Order	Year of Startup
Vermont Yankee	Vermont Yankee	514	1966	1972
Peach Bottom 2	Philadelphia Electric	1,065	1966	1974
Peach Bottom 3	Philadelphia Electric	1,065	1966	1974
Fitzpatrick	PASNY	821	1966	1975
Bailly	NIPSCO	660	1967	1977
Shoreham	LILCO	819	1967	1978
Cooper	Nebraska PPD	778	1967	1974
Browns Ferry 3	TVA	1,098	1967	1975
Limerick 1	Philadelphia Electric	1,098	1967	1981
Hatch 1	Georgia	786	1967	1975
Fukushima 2	Japan	762	1967	1974
Brunswick 1	Carolina P&L	821	1968	1976
Brunswick 2	Carolina P&L	821	1968	1975
Arnold	Iowa ELP	569	1968	1974
Fermi 2	Detroit Edison	1,123	1968	1979
Limerick 2	Philadelphia Electric	1,065	1969	1982
Hope Creek 1	PSE&G	1,067	1969	1981
Hope Creek 2	PSE&G	1,067	1969	1983
Zimmer 1	CCDPP	810	1969	1978
Chinshan 1	Taiwan	610	1969	1977
Caorso 1	Italy	827	1969	1975
Hatch 2	Georgia Power Company	795	1970	1978
La Salle 1	Commonwealth Edison	1,078	1970	1978
La Salle 2	Commonwealth Edison	1,078	1970	1979
Susquehanna 1	Pennsylvania P&L	1,052	1970	1980

TABLE 1.4-1: [HISTORICAL INFORMATION] COMMERCIAL NUCLEAR REACTORS COMPLETED, UNDER CONSTRUCTION OR IN DESIGN BY GENERAL ELECTRIC (CONTINUED)

Station	Utility	Rating (MWe)	Year of Order	Year of Startup
Susquehanna 2	Pennsylvania P&L	1,052	1970	1982
Chinshan 2	Taiwan	610	1970	1978
WPPSS 2	WPPSS	1,103	1971	1977
Nine Mile Point 2	Niagara Mohawk	1,080	1971	1979
Grand Gulf 1	Entergy Operations, Inc.	1,290	1971	1980
Kaiseraugst	Switzerland	915	1971	1978
Fukushima 6	Japan	1,135	1971	1976
Tokai 2	Japan	1,135	1971	1976
Riverbend 1	Gulf States	934	1972	1980
Riverbend 2	Gulf States	934	1972	1981
Perry 1	Cleveland Electric	1,205	1972	1979
Perry 2	Cleveland Electric	1,205	1972	1980
Douglas Point 1	PEPCO	1,178	1972	1985
Douglas Point 2	PEPCO	1,178	1972	1987
Hartsville 1	TVA	1,228	1972	1980
Hartsville 2	TVA	1,228	1972	1981
Hartsville 3	TVA	1,228	1972	1981
Hartsville 4	TVA	1,228	1972	1982
Laguna Verde 1	Mexico	660	1972	1977
Leibstadt	Switzerland	940	1972	1978
Kuosheng 1	Taiwan	992	1972	1978
Kuosheng 2	Taiwan	992	1972	1979
Clinton 1	Illinois Power	955	1973	1981
Clinton 2	Illinois Power	955	1973	1984
Montague 1	NUSCO	1,220	1973	1982

TABLE 1.4-1: [HISTORICAL INFORMATION] COMMERCIAL NUCLEAR REACTORS COMPLETED, UNDER CONSTRUCTION OR IN DESIGN BY GENERAL ELECTRIC (CONTINUED)

Station	Utility	Rating (MWe)	Year of Order	Year of Startup
Allens Creek 1	Houston L&P	1,150	1973	1980
Allens Creek 2	Houston L&P	1,150	1973	1982
Skagit 1	Puget SD	1,290	1973	1981
Skagit 2	Puget SD	1,290	1973	1983
Blackfox 1	Oklahoma	950	1973	1983
Blackfox 2	Oklahoma	950	1973	1985
Laguna Verde 2	Mexico	660	1973	1978
Enel 6	Italy	982	1974	1980
Enel 8	Italy	982	1974	1980

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

1.5.1 Current Development Programs

1.5.1.1 Instrumentation for Vibration

[HISTORICAL INFORMATION] [Vibration testing for reactor internals has been performed on virtually all GE-BWR plants. At the time of issue of NRC Regulatory Guide 1.20, test programs for compliance were instituted. The first BWR 6 plant of each size will be considered a prototype design and will be instrumented and subjected to both cold and hot, two-phase flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. Subsequent plants which have internals similar to those of the prototypes will be tested in compliance to the requirements of Regulatory Guide 1.20 to confirm the adequacy of the design with respect to vibration.]

1.5.1.2 Core Spray Distribution

[HISTORICAL INFORMATION] [Due to slight changes in core dimensions and core spray sparger geometry, the core spray flow distribution header has been tested to assure that each fuel assembly in the reactor core would receive adequate cooling water in the event of a LOCA. These tests are regarded as confirmatory only since the basic spray header design has been successfully tested over a wide range of similar geometrical conditions.

The tests demonstrate that each fuel assembly receives adequate cooling water flow for any spray system flow rate between the rated flow and the runout flow condition.

GEH has completed development of a core spray methodology, consisting of single nozzle tests in steam, computer calculations, and multiple nozzle tests in air, to calculate minimum bundle flow. Application of the methodology for Grand Gulf shows a minimum calculated bundle flow of 3.1 gpm. This compares to a minimum required bundle flow of approximately 1 gpm as described in the questions and answers to NEDO-10846, April 1973. Additional descriptions of the tests and computer codes may be found in NEDO-20566, Amendment 3, April 1977; NRC letter, "Review of General Electric Topical Report, NEDO-20566, Amendment 3, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K-Effect of Steam Environment of BWR Core Spray Distribution," June 13, 1978; and

NRC letter from P. S. Check to R. L. Tedesco, "Evaluation of NEDO-24712, Core Spray Distribution Methodology Confirmation Tests," September 8, 1980.]

1.5.1.3 Core Spray and Core Flooding Heat Transfer Effectiveness

[HISTORICAL INFORMATION] [Due to the incorporation of an 8 x 8 fuel rod array with unheated "water rods," tests have been conducted to demonstrate the effectiveness of ECCS in the new geometry.

These tests are regarded as confirmatory only, since the geometry change is very slight and the "water rods" provide an additional heat sink in the inside of the bundle which improves heat transfer effectiveness.

There are two distinct programs involving the core spray. Testing of the core spray distribution has been accomplished, and the Licensing Topical Report NEDO-10846, "BWR Core Spray Distribution," April, 1973, has been submitted. The other program concerns the testing of core spray and core flooding heat transfer effectiveness. The results of testing with stainless steel cladding were reported in the Licensing Topical Report NEDO-10801, "Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness," March, 1973. The results of testing using Zircaloy cladding were reported in the Licensing Topical Report, NEDO-20231, "Emergency Core Cooling Tests of an Internally Pressurized, Zircaloy Clad, 8x8 Simulated BWR Fuel Bundle," December, 1973.]

1.5.1.4 Verification of Pressure Suppression Design

[HISTORICAL INFORMATION] [The General Electric Company has conducted a large scale test program to verify the performance characteristics of the Mark III containment. Large scale testing was started in November 1973 following completion of a two-year small scale test program.

The large scale test program utilizes a facility which represents a segment of a Mark III containment. The original character of the programs was to be a confirmatory exercise to verify the short term analytical model. The scope of the total program included testing beyond design basis conditions to investigate the margins available in pressure suppression systems. As a result of this testing, GE proposed a new analytical model to evaluate the Mark

III design. This model is entitled “The General Electric Mark III Pressure Suppression Containment System Analytical Model,” and is described in NEDO-20533.

During early tests it was observed that containment structures could be subject to significant suppression pool hydrodynamic loads during blowdown. This resulted in several additional test series whose objective was to generate design basis loads to be incorporated in the design of the affected containment structures.

Eleven large scale test series have been completed to date. The primary objective of three series of these tests was to verify short-term analytical models for horizontal vents (and centerline submergences). The objectives of two others were to obtain scoping data regarding pool dynamic response and impact loads on structures located above the suppression pool. Other tests were designed to measure froth impingement loads on the Hydraulic Control Unit floor and to determine pool swell motion characteristics, to measure pool impact loads on representative containment structures, and to determine pool motion characteristics for large air mass fraction vent flows and to compare these scale results to the previous full scale air tests.

Additional tests will be conducted to indicate comparability of liquid blowdown to steam blowdowns and to investigate pool stratification and vent chugging effects.

Tests will be performed with the suppression pool at an initial elevated temperature to determine steam condensation characteristics under such conditions. A multi-vent series will be run to consider possible vent interactions. In plant testing of the safety relief valves to verify that the design basis safety relief valve discharge loads inside the suppression pool are adequately conservative will be performed on Grand Gulf Unit 1 prior to full power operation.]

1.5.1.5 Critical Heat Flux Testing

[HISTORICAL INFORMATION] [A program for Critical Heat Flux testing was established and was to be similar to that described in the report APED-5286, “Design Basis for Critical Heat Flux Condition in Boiling Water Reactors,” September 1966. Since that time, however, a new analysis has been performed and the GETAB program initiated. The results of that analysis and related testing is described in the approved Licensing Topical Report, NEDO-10958-A, “General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,” January 1977. These results and correlation are applicable to GEH Initial core fuel for cycle 1.

A similar program has been established by Exxon Nuclear Company (ENC), Inc. as part of the process of developing the capability to license fuel for nuclear reactor reloads. The result of this program is the XN-3 (Revision 1) critical heat flux correlation which is used to predict the onset of transition boiling. This effort is described in the approved licensing topical report XN-NF-512(P)(A) Revision 1 and XN-NF-512(P)(A) Revision 1 Supplement 1, “XN-3 Critical Power Correlation,” Exxon Nuclear Co., October 1982.]

1.5.1.6 Structural Testing

[HISTORICAL INFORMATION] [Although tests are being conducted to determine the effects of vibration on fuel assembly spacers and to determine the forces to which the assemblies are subjected during shipment, there is no special program at present concentrating on structural testing, and no topical report is anticipated.]

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 is a list of GE and Bechtel topical reports and any other reports of documents which are incorporated in whole or in part by reference in this FSAR and has been filed with the NRC.

Additional documents which are referenced in this FSAR are listed at the end of the sections in which they have been referenced.

TABLE 1.6-1: REFERENCED REPORTS

Report Number	Title	Referenced in FSAR Section
A.		
APED-4827	Maximum Two-Phase Vessel Blowdown from Pipes (April 1965)	6.2
APED-4986	Consequences of Operating Zircaloy-2 Clad Fuel Rods Above the Critical Heat Flux (October 1965)	4.2
APED-5286	Design Basis for Critical Heat Flux Condition in BWRs (September 1966)	1.5
APED-5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)	5.4
APED-5460	Design and Performance of General Electric BWR Jet Pumps (July 1968)	3.9
APED-5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November 1967)	4.6
APED-5640	Xenon Considerations in Design of Boiling Water Reactors (June 1968)	4.1, 4.3
APED-5652	Stability and Dynamic Performance of the General Electric Boiling Water Reactor	4.1
APED-5706	In-Core Neutron Monitoring System for General Electric Boiling Water Reactors (November 1968, Revised April 1969)	7.6.1.5, 7.7.1.7, 7.6.2.5

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
APED-5736	Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards (April 1969)	6.3
APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves (March 1969)	5.4
APED-5756	Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor (March 1969)	15.4
GEAP-10546	Theory Report for Creep-Plast Computer Program (January 1972)	4.1
GEAP-13112	Thermal Response and Cladding Performance of an Internally Pressurized, Zircaloy-Clad, Simulated BWR Bundle Cooled by Spray Under Loss-of-Coolant Conditions (April 1971)	4.2
NEDC-33477P	Safety Analysis Report for Grand Gulf Nuclear Station Constant Pressure Power Uprate (March 2012 as corrected June 2016)	Referenced in Chapters 1 through 12 & 15
NEDE-10313	PDA-Pipe Dynamic Analysis Program for Pipe Rupture Movement (Proprietary Filing)	3.6
NEDE-11146	Design Basis for New Gas System (July 1971) (Company Proprietary)	11.3
NEDE-20386	Fuel Channel Deflections	4.2

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
NEDE-21156	Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration (January 1976)	4.4
NEDE-21175-P BWR/6	Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (November 1976)	3.9
NEDE-21354-P	PWR Fuel Channel Mechanical Design and Deflection (September 1976)	3.9
NEDE-24196 (Proprietary)	Basis for BWR 6 8x8 Fuel Thermal Analysis Application, General Electric Information Report	4.4, 4.3
NEDE-23014	HEX 01 User's Manual (July 1976)	15.2
NEDM-10735	Densification Considerations in BWR Fuel Design and Performance (December 1972)	4.2
NEDO-10173	Current State of Knowledge, High Performance BWR Zircaloy-Clad U02 Fuel (May 1970)	4.2, 11.1
NEDO-10174	Consequences of a Postulated Fuel Blockage Incident in a Boiling Water Reactor (May 1970)	4.2
NEDO-10299	Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello (January 1971)	4.4
NEDO-10320	The General Electric Pressure Suppression Containment Analytical Model (April 1971) Supplement (May 1971)	6.2

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
NEDO-10329	Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors (April 1971) Supplement 1 (April 1971) Addenda (May 1971)	4.3
NEDO-10349	Analysis of Anticipated Transients Without Scram (March 1971)	15.8
NEDO-10466-A	Power Generation Control Complex (February 1979) and Addendum l (December 1979)	7.1.2.2
NEDO-10505	Experience with BWR Fuel Through September 1971 (May 1972)	4.2, 11.1
NEDO-10527	Rod Drop Accident Analysis for Large Boiling Water Reactors (March 1972) Supplement l (July 1972) Supplement 2 (January 1973	4.3, 15.4
NEDO-10585	Behavior of Iodine in Reactor Water During Plant Shutdown and Startup (August 1972)	15.6
NEDO-10602	Testing of Improved Jet Pumps for the BWR/6 Nuclear System (June 1972)	3.9
NEDO-10734	A General Justification for Classification of Effluent Treatment System Equipment as Group D (February 1973)	11.3
NEDO-10739	Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems (January 1973)	6.3

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
NEDO-10751	Experimental and Operational Confirmation of Offgas System Design Parameters (January 1973) (Company Proprietary)	11.3
NEDO-10801	Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness (March 1973)	1.5
NEDO-10802	Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor (February 1973)	4.4, 5.2, 15.1
NEDO-10846	BWR Core Spray Distribution (April 1973)	1.5
NEDO-10899	Chloride Control in BWR Coolants (June 1973)	5.2
NEDO-10905	High Pressure Core Spray Power Supply Unit	8.3.1.2
NEDO-10958	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application (November 1973)	4.3, 4.4, 15.0
NEDO-10958-A	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application (January 1977)	1.5, 15.4, 16.1
NEDO-10959	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application (November 1973)	15.0

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
NEDO-10977	Drywell Integrity Study: Investigation of Potential Cracking in BWR/6 Mark III Containment	6.3
NEDO-20231	Emergency Core Cooling Tests of an Internally Pressurized, Zircaloy-Clad, 8x8 Simulated BWR Fuel Bundle (December 1973)	1.5
NEDO-20340	Process Computer Performance Evaluation Accuracy (June 1974)	4.3
NEDO-20360	General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel (May 1975)	4.2, 15.4
NEDO-20360-IP	General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel (March 1976)	4.2
NEDO-20533	The General Electric Mark III Pressure Suppression Containment System Analytical Model (June 1974)	1.5
NEDO-20566	General Electric Company Model for Loss-of-Coolant Accident Analysis in Accordance with 10 CFR 50, Appendix K (January 1976)	3.9, 4.3, 6.3, 1.5
NEDO-20605 and NEDO-20606	Creep Collapse Analysis of BWR Fuel Using Safe Collapse Model (August 1974)	4.2
NEDO-20626	Studies of BWR Designs for Mitigation of Anticipated Transients without Scrams (October 1974)	15.8

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
NEDO-20626-1	Studies of BWR Designs for Mitigation of Anticipated Transients without Scrams (June 1975)	15.8
NEDO-20626-2	Studies of BWR Designs for Mitigation of Anticipated Transients without Scrams (July 1975)	15.8
NEDO-20913	Lattice Physics Methods (June 1975)	4.3
NEDO-20922	Experience with BWR Fuel Through September 1974 (June 1975)	4.2, 11.1
NEDO-20939	Lattice Physics Methods Verification (August 1975)	4.3
NEDO-20943	Urania-Gadolinia Nuclear Fuel Physical and Material Properties (January 1977)	4.2
NEDO-20944	BWR/4 and BWR/5 Fuel Design (October 1976)	4.1, 4.3
NEDO-20946	BWR Simulator Methods Verification (May 1976)	4.3
NEDO-20948-P BWR/6	Fuel Design (June 1976)	4.2
NEDO-20953	Three-Dimensional Boiling Water Reactor Core Simulator (May 1976)	15.4

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
NEDO-20964	Generation of Void and Doppler Reactivity Feedback for Application to BWR Plant Transient Analysis (August 1975)	4.3
NEDO-21142	Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAC Code and User's Guide (September 1978)	15.4, 15.6, 15.7
NEDO-21143	Conservative Radiological Accident Evaluation - The CO/NAC01 Code (March 1976)	15.4, 15.6, 15.7
NEDO-21159	Airborne Release from BWRs for Environment Impact Evaluations (March 1976)	11.1
NEDO-21174	BWR Fuel Channel Deflections	4.2
NEDO-21231	Banked Position Withdrawal Sequence (September 1976)	4.3
NEDO-21291	Group Notch Mode of the RSCS for Cooper (June 1976)	15.4
NEDO-21708	Radiation Effects in BWR Pressure Vessel Steels	5.3
NEDO-21985	Functional Capability Criteria for Essential Mark II Piping (September 1978)	3.9
NEDO-24083	Recirculation Pump Shaft Seal Leakage Analysis (November 1978)	5.5
NEDO-24142	Fast Scram Control Rod Drive	4.6

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
NEDO-24154	Qualification of the One-Dimensional Core Transient Model for BWR, NEDO-24154, October 1978	5A3.1.1
NEDO-24708A	Additional Information for NRC Staff Generic Report on BWRs, Volumes 1 and 2 (December 1980)	18.1.29
NEDO-24712	Core Spray Design Methodology Confirmation Tests (August 1979) Qualification Program (October 1978)	1.5
NEDO-26453	3D BWR Core Simulator (May 1976)	4.3
	Oyster Creek Station, FSAR Amendment 10	1.5
	"Summary Memorandum on Excursion Analysis Uncertainties," Dresden Nuclear Power Station, Unit 3, Plant Design Analysis Report Amendment 3	4.3, 15.0
	Hatch Nuclear Plant, Unit 1, PSAR Amendment 10, Appendix L; and Amendment 7.	15.5; 7.6.1.5
	Millstone Nuclear Power Station, PSAR Amendment 14	6.3
	Pilgrim Nuclear Power Station, PSAR Amendment 14	6.3
	Quad Cities Station, Units 1 and 2, PSAR Amendment 9	4.3

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
22A4365	Interim Containment Loads Report (ICLR), Mark III Containment, Revision 2	3.8 Appendix
B. Other Referenced Reports		
AE-RTL-788	Void Measurements in the Region of Subcooled and Low Quality Boiling (April 1966)	4.4
ANL-5621	Boiling Density in Vertical Rectangular Multichannel Sections with Natural Circulation (November 1956)	4.4
ANL-6385	Power-to-Void Transfer Functions (July 1961)	4.4
AGN-TM-407	AGN-GAM, and IBM 7090 Code to Calculate Spectra and Multigroup Constants (April 1965)	4.3
ANL-7460	Reactor Development Program Progress Report, p. 121-122 (June 1968)	4.3
ANL-7527	Reactor Development Program Progress Report, p. 132 (December 1968)	4.3
BNL-5826	THERMOS-A Thermalization Transport Code for Reactor Design (June 1961)	4.3
BNWL-340	"Computer Code Abstracts, Computer Code-HRG," Reactor Physics Dept., Technical Activities Quarterly Report, July, August, September, 1966 (October 15, 1966)	4.3

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
BHR/DER 70-1	Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor (March 1970)	11.1
BHR/DER 70-1	Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor (March 1970)	11.1
BMI-1163	Vapor Formation and Behavior in Boiling Heat Transfer (February 1957)	4.4
CF 59-6-47 (ORNL)	Removal of Fission Product Gases from Reactor Off-Gas Streams by Adsorption (June 11, 1959)	11.3
IDO-ITR-105	The Response of Waterlogged U02 Fuel Rods to Power Bursts (April 1969)	4.2
IN-ITR-111	The Effects of Cladding Material and Heat Treatment on the Response of Waterlogged U02 Fuel Rods to Power Bursts (January 1970)	4.2
STl-372-38	Kinetic Studies of Heterogeneous Water Reactors (April 1966)	4.4
TID-4500	Relap 3 - A Computer Program for Reactor Blowdown Analysis IN-1321 (June 1970)	3.6
WACP-6065	Melting Point of Irradiated Uranium Dioxide (February 1965)	4.2

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
WAPD-BT-19	A Method of Predicting Steady-State Boiling Vapor Fractions in Reactor Coolant Channels (June 1960)	4.4
WAPD-TM-283	Effects of High Burnup on Zircaloy-Clad, Bulk U02 Plate Fuel Element Samples (September 1962)	4.2
WAPD-TM-416	WIGLE - A Program for the Solution of the Two-Group Space-Time Diffusion Equations in Slab Geometry (1964)	4.3
WAPD-TM-629	Irradiation Behavior of Zircaloy-Clad Fuel Rods Containing Dished End U02 Pellets (July 1967)	4.2
C. Bechtel Corporation Reports		
BN-TOP-1	"Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants," Revision 1, November 1972	3.8.3, 3.8.1
BN-TOP-2	"Design for Pipe Rupture Effects," Revision 2, May 1974	3.6, 3.8
BN-TOP-4	"Subcompartment Pressure Analysis" Revision 0, July 1976, and Revision 1	6.2.1.2 Appendix 3E
BC-TOP-1	"Containment Building Liner Plate Design Report," December 1972, Rev. 1	3.8.1, 3.8.3

TABLE 1.6-1: REFERENCED REPORTS (CONTINUED)

Report Number	Title	Referenced in FSAR Section
BC-TOP-3A	"Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," Revision 3, August 1974	3.3, 3.8
BC-TOP-4	"Seismic Analysis of Structures and Equipment for Nuclear Power Plants," Revision 1, September 1972, including Addendum 1 dated April, 1973.	3.7, 3.8
BC-TOP-5A	"Prestressed Concrete Nuclear Reactor Containment Structures," Rev. 3, February 1975	3.8.1
BC-TOP-9A	"Design of Structures for Missile Impact," Rev. 2, September 1974	3.5.3

1.7 ELECTRICAL, INSTRUMENTATION AND CONTROL DRAWINGS

Table 1.7-1 contains a list of non-proprietary electrical, instrumentation, and control drawings which are incorporated in the FSAR by reference. This table lists those drawings which are considered to be necessary to evaluate the safety-related features in Chapter 7 and 8. These tables will be updated in future amendments as necessary.

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J0200 Sh 1	Logic Symbols
J0204 Sh 8	P64 Fire Protection Sys Aux Bldg Isln Valves Unit 1
J0240 Sh 0	Z51 Control Room HVAC Sys
J-0251 Sh 96	G17 Floor Drain Filter Outlet Valve
J-0251 Sh 96	G17 Equipment Drain Filter Outlet Valve
J-0251 Sh 98	G17 Equipment Drain Floor Drain Filter Bypass Valves
J0300 Sh 0	A21 Loop Diagram Legend Index
J0340 Sh 0	Z51 Control Room HVAC Sys Index
J0400	Control Room Panel Location
J0401	Upper Cable Spreading Room Panel Location
J0402	Lower Cable Spreading Room Panel Location
J0419	Control Room Vent VB SH13 P855
J1202 Sh 0	P21 Makeup Water Treatment Sys

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J1203 Sh 0	P66 Domestic Water Sys
J1216 Sh 7	P11 Cond & Refueling Water Transfer & Stg Sys Isln Valves
J1216 Sh 11	P11 Cond & Refueling Water Transfer Sys Aux Bldg Isln Valves
J1216 Sh 12	P11 Cond & Refueling Water Transfer & Stg Sys Isln Valves
J1221 Sh 0	P41 Standby Service Water Sys Index
J1222 Sh 6	P44 Plant Service Water Sys Aux Bldg Isln Valves
J1222 Sh 9	P44 Standby Service Water to Plant Service Water Crosstie Valves
J1222 Sh 16	P44 Plant Service Water Instr & Svc Air Cprsrs Cut-out Valve
J1223 Sh 11	SP43 Turbine Bldg Cool Wtr Service Air Compsr Iso Valves
J1224 Sh 0	P42 Component Cooling Water System
J1225 Sh 4	P71 Plant Chilled Water Isln Valves

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J1225 Sh 5	P71 Plant Chilled Water Sys Aux Bldg Isln Valves
J1226 Sh 3	P52 Service Air Isln Control Air-Operated Valves
J1226 Sh 4	P52 Service Air Sys Aux Bldg Isln Valves
J1226 Sh 5	P52 Service Air Isln Control Motor-Operated Valve
J1228 Sh 1	C11 CRD Pump Suction Aux Bldg Isln Valve
J1231 Sh 1	M41 Containment Cooling Sys Containment Isln Valves
J1231 Sh 2	M41 Containment Cooling Sys Drywell Isln Valves
J1231 Sh 14	M41 Containment Cooling Sys Aux Bldg Isln Valves
J1231 Sh 16	M41 Containment Cooling Sys Containment Isln Values
J1233 Sh 6	T41 Aux Bldg Vent Sys Isln Valves
J1234 Sh 1	T42 Fuel Handling Area Vent Sys Isln Valves
J1235 Sh 0	T51 Emergency Pump Room Vent Sys Index

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J1236 Sh 0	T48 Standby Gas Treatment Sys Index
J1237 Sh 0	E61 Combustible Gas Control Rooms Ventilation System Index
J1241 Sh 0	X77 Diesel Generator Rooms Ventilation System Index
J1250 Sh 4	E31 Leak Detection Trip Unit Fault Monitor Alarms
J1254 Sh 0	P75 Standby Diesel Generator System Index
J1255 Sh 6	P72 Drywell Chilled Water Sys Isln Valves
J1256 Sh 1	P45 Floor & Equipment Drain Sys Isln Valves
J1256 Sh 2	P45 Floor & Equipment drain Sys Aux Bldg Isln Valves
J1256 Sh 30	P45 Drywell Chemical Waste Isln Valves
J1256 Sh 44	P45 Containment Isolation Valves
J1258 Sh 0	Y47 Standby Service Water Pump House Vent System Index

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J1259 Sh 0	Z77 Safeguard Switchgear and Btry Rooms Vent System Index
J1260 Sh 0	M71 Containment Drywell & Aux Bldg Instm and Control
J1261 Sh 0	P81 HPCS Diesel Generator System Index
J1262 Sh 6	P53 Instrument Air Isln Control Air Operated Valve
J1262 Sh 8	P53 Instrument Air Sys Aux Bldg Isln Valves
J1262 Sh 11	P53 Instrument Air Isln Control Motor Operated Valve
J1267 Sh 0	T46 Engineered Safety Features Elec Switchgear Rooms Cooling System Index
J1271 Sh 0	E12 Residual Heat Removal System
J1272 Sh 2	G46 FPCC Filter-Demin Sys Backwash Aux Bldg Isln Valve
J1277 Sh 1	G36 RWCU Backwash Rcvg Tank Containment Isln Valves
J1277 Sh 2	G36 RWCU Backwash Rcvg Tank Aux Bldg Isln Valves

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J1279 Sh 0	E30 Suppression Pool Makeup System Index
J1281 Sh 0	B21 Nuclear Boiler System Index
J1284 Sh 1	G33 Reactor Water Cleanup Aux Bldg Isln Valves
J1293 Sh 3	E38 Block Flow/Ctmt Isln Valves A & B
J1297 Sh 2	P60 Suppression Pool Cleanup Sys Containment & Aux Bldg Isln Valves
J1298 Sh 0	E38 Feedwater Leakage Control System
J1321 Sh 0	P41 Standby Service Water System Index
J1322 Sh 0	P44 Plant Service Water System
J1324 Sh 0	P42 Component Cooling Water System Index
J1328 Sh 0	C11 CRD Hydraulic System Index
J1336 Sh 0	T48 Standby Gas Treatment System Index
J1337 Sh 0	E61 Combustible Gas Control System Index

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J1341 Sh 0	X77 Diesel Generator Bldg Vent System Index
J1350 Sh 0	E31 Leak Detection Sys Index
J1354 Sh 0	P75 Standby Diesel Generator System Index
J1358 Sh 0	Y47 Standby Service Water Pump House Vent System Index
J1359 Sh 0	Z77 Safeguard Switchgear and Battery Rooms Vent Sys Index
J1360 Sh 0	M71 Containment Drywell and Aux Bldg Instrumentation and Control Index
J1361 Sh 0	P81 HPCS Diesel Generator System Index
J1367 Sh 0	T46 Engineered Safety Features Electrical Switchgear Rooms Cooling System Index
J1368 Sh 0	C71 Reactor Protection System Index
J1369 Sh 0	C61 Remote Shutdown System Index
J1374 Sh 0	D21 Area Radiation Monitoring System
J1375 Sh 0	D23 Drywell Monitoring System

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J1379 Sh 0	E30 Suppression Pool Level
J1398 Sh 0	E38 Feedwater Leakage Control System
J0400	Control Room Panel Location
J0401	Upper Cable Spreading Room Panel Location
J0402	Lower Cable Spreading Room Panel Location
J1414	Diesel Generator BB 1H13-P864
J1416	Div IV Engineered Safety Features Logic VB 1H13-P878
J1417	Div I Engineered Safety Features Logic VB 1H13-P871
J1418	Div II Engineered Safety Features Logic VB 1H13-P872
J1431	Div III Engineered Safety Features Logic VB 1H13-P877
J1487A thru D	Remote Shutdown Control Panel 1H22-P150
J1488A & B	Remote Shutdown Control Panel 1H22-P151

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 1, Instrumentation/Control (Bechtel)

Drawing No. (FSAR Figure No.)	Title
J0501	Instrument Location - Control Bldg El 93-0, 113-0, 133-0, and 148-0
J0502	Instrument Location - Control Bldg El 166-0, 175-0, and 189-0
J1502	Instrument Location Turbine Bldg El 113-0
J1503	Instrument Location Turbine Bldg El 133-0
J1504	Instrument Location Turbine Bldg El 166-0
J1505	Instrument Location Aux and Cntmt Bldg El 93-0 and 100-9
J1506	Instrument Location Aux and Cntmt Bldg El 119-0 and 114-6
J1507	Instrument Location Aux and Cntmt Bldg El 135-4, 139-0 and 147-7
J1508	Instrument Location Aux and Cntmt Bldg El 161-10 and 166-0
J1509	Instrument Location Aux and Cntmt Bldg El 184-6 and 185-0

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E0001 (8.1-1)	Main One Line Diagram
E0002	Phasing Diagram
E0004	Phasing Diagram
E0010	Synchronizing Diagram Engineered Safety Features Buses 15AA, 16AB, 17AC, 25AA, 26AB, 27AC
E0013	One Line Meter and Relay Diag Aux Elec Dist Sys and Boiler Bus 19UD
E0014	One Line Meter and Relay Diag Aux Elec Dist Sys and Boiler Bus 29UE
E0021	Ground Detection Sch for 3 Wire Ungrounded DC Sys
E0022	Ground Detection Sch for 2 Wire Ungrounded DC Sys
E0028	Three Line Meter and Relay Diagram Engineered Safety Features Transformers
E0030-000 Shs A-71	General Symbols, Notes and Details
E0032 (8.3-7a)	One Line Meter and Relay Diagram 120/240V AC Uninterruptible Power Supplies

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E0111-000 Shs 0-2	4.16kV Switchgear Typical Circuit Breaker Internal Details
E0116-000 Shs 0-2	480V Switchgear Typical Circuit Breaker Internal Details
E0131-000 Shs 0-30	Control Room HVAC Sys
E0231 Sh 0	Fire Protection System Index
E0231 Sh A	Fire Protection System Relay Tabulation
E0231 Sh 14	Fire Protection System
E0231 Sh 19	Fire Protection System Aux Bldg Isln Valve F282A
E0231 Sh 22	Fire Protection System BOP Computer Points
E0232-000 Shs 0-9	Domestic Water System Index and Relay Tabulation
E0627	Lighting and Comm Plan Control Bldg El 148-0
E0628	Lighting and Comm Plan Control Bldg El 166-0
E0630	Lighting and Comm Plan Control Bldg El 177

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E0637	Lighting and Comm Plan Control Bldg El 111-0
E0638	Lighting and Comm Plan Control Bldg El 133-0
E0648	Control and Aux Bldgs Lighting System
E-0649A (9.5-009B)	Public Address System, Sound Power Telephone & Warning Light Diagram El 93'-0" & El 103'-0"
E-0649B (9.5-009C)	Public Address System, Sound Power Telephone & Warning Light Diagram El 133'-0", 113'-0", 118'-0" & Partial Plan 111'-0"
E-0649C (9.5-009D)	Public Address System, Sound Power Telephone & Warning Light Diagram El 133'-0", 136'-0", 139'-0" & Partial Plan El 148'-0"
E-0649D (9.5-009E)	Public Address System, Sound Power Telephone & Warning Light Diagram El 166'-0", 161'-0" & Partial Plan El 166'-0" & 177'-0"
E-0649E (9.5-009F)	Public Address System, Sound Power Telephone & Warning Light Diagram El 185'-0", 189'-0" & Partial Plan El 208'-10"

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E-0649F (9.5-009G)	Public Address System, Sound Power Telephone & Warning Light Diagram Site Plan
E-0649G (9.5-009H)	Public Address System Administration Building
E-0649H (9.5-009I)	Public Address System, M&E Building
E-0660	Site Raceway Plan
E-0663	Enlarged Site Raceway Plan
E-0672	Enlarged Site Raceway Plan
E-0674	Enlarged Site Raceway Plan
E-0688	Raceway Plan Control Bldg El 111-0 Area 25A
E-0689	Raceway Plan Control Bldg El 133-0 Area 25A
E-0690	Raceway Plan Control Bldg El 148-0 Area 25A
E-0691	Raceway Plan Control Bldg El 166-0 Area 25A
E-0692	Raceway Plan Control Bldg El 189-0 Area 25A
E-0693	Raceway Plan Control Bldg El 177-0 Area 25A

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E-0694	Raceway Plan Control Bldg Area 25A Ceiling El 93-0
E-0695	Raceway Sections and Details Control Bldg Area 25A
E-0700	Raceway Plan Control Bldg El 93 Ceiling Area 25B
E-0701	Raceway Plan Control Bldg El 111-0 Area 25B
E-0701A	Raceway Plan Control Bldg El 111-0 Area 25B
E-0702	Raceway Plan Control Bldg El 133-0 Area 25B
E-0703	Raceway Plan Control Bldg El 148-0 Area 25B
E-0703A	Raceway Plan Control Bldg El 148-0 Area 25B
E-0704	Raceway Plan Control Bldg El 166-0 Area 25B
E-0705	Raceway Plan Control Bldg El 189-0 Area 25B
E-0705A	Raceway Plan Control Bldg El 1890 Area 25B

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E-0706	Raceway Plan at Ceiling Control Bldg El 177-0 Area 25B
E-0716	Raceway Plan Control Bldg Sections and Details
E-0724	Raceway Sections and Details Control Bldg Area 25B
E-0725-000 Shs A-51 (except Shs 15, 16, 17, 36, 41, 42, 43, 45)	Raceway Notes, Symbols and Details
E-0950	Raceway Plan Control Bldg El 93-0, 111-0, 133-0, 148-0 Fire & Smoke Detection System Units 1 & 2
E-0951	Raceway Plan Control Bldg El 166-0, 177-0, 189-0 Fire & Smoke Detection System Units 1 & 2
E-0961	Raceway Plan Radwaste Bldg El 93-0 Fire & Smoke Detection System Units 1 & 2
E-0962	Raceway Plan Radwaste Bldg El 118-0 Fire & Smoke Detection System Units 1 & 2
E-0963	Raceway Plan Radwaste Bldg El 136-0 Fire & Smoke Detection System Units 1 & 2

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E-0964	Raceway Plan Misc. Bldgs Fire & Smoke Detection System Units 1 & 2
E-0965	Raceway Plan Water Treatment Bldg El 133-0 and Stdby Wtr Pump HS Basin A & B Fire & Smoke Detection System Units 1 & 2
E-1004	One Line Meter and Relay Diag 6.9kV BOP Buses 11 HD and 12 HE, Unit 1
E-1008 (8.3-1)	One Line Meter and Relay Diag 4.16kV ESF System
E-1009	One Line Meter and Relay Diag 4.16kV ESF System
E-1017	One Line Meter and Relay Diag 480V Buses 15BA1, 15BA2, 15BA3, 15BA4
E1018	One Line Meter and Relay Diag 480V Buses 16BB1, 16BB2, 16BB3, 16BB4
E1019	One Line Meter and Relay Diag 480V Buses 15BA5, 16BB5
E1020	One Line Meter and Relay Diag 480V Buses 15BA6 and 16BB6
E1022 (8.3-10B)	One Line Meter and Relay Diagram 125V Bus

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1023 (8.3-10)	One Line Meter and Relay Diag 125V DC Buses 11DA, 11DB, and 11DC
E1024 (8.3-7)	One Line Meter and Relay Diag 120/240V AC Uninterruptible Power Supplies
E1026 (8.3-7b)	One Line Meter and Relay Diag 120 V AC ESF Uninterruptible Power Supplies
E1032-000 Shs 0-15	208 - 120V AC Engineered Safety Features Power Panels
E1034-000 Shs 0-9	120V AC Power Supplies to Cont and Instr Panels
E1036-000 Shs 0-4	125V DC Power Supplies to Cont and Instr Panels
E1027 (8.3-10a)	One Line Meter and Relay Diag 125 V DC Buses 11DK and 11DL
E1039 (8.3-9)	Logic Diagram Load Shedding and Sequencing Panel
E1042 (8.3-8)	Diesel Logic Diagram Engineered Safety Features Div I
E1043	Diesel Logic Diagram Engineered Safety Features Div II

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1053	Three Line Meter and Relay Diagram ESF Div I
E1054	Three Line Meter and Relay Diagram ESF Div II
E1057 Shs 1,2	480V ESF MCC 15B41 Aux Bldg
E1058 Shs 1,2	480V ESF MCC 16B21 Aux Bldg
E1059	480V ESF MCC 17B11 Cont Bldg
E1081 Shs 1,2	480V ESF MCC 15B11 Aux Bldg
E1082 Shs 1-2	480V ESF MCC 15B31 Aux Bldg
E1083 Shs 1-2	480V ESF MCC 15B21 Aux Bldg
E1084 Sh 1	480V ESF MCC 15B61 Cntrl Bldg
E1085	480V ESF MCC 15B51 Standby Service Water Pump House
E1086 Shs 1-3	480V ESF MCC 16B11 Aux Bldg
E1087 Shs 1-2	480V ESF MCC 16B31 Aux Bldg
E1088 Shs 1-2	480V ESF MCC 26B41 Aux Bldg
E1089	480V ESF MCC 16B61 Containment Bldg

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1090	480V ESF MCC 16B51 Standby Service Water Pump House
E1091	480V ESF MCC 17B01 Cont Bldg
E1098 Shs 1-2	480V ESF MCC 16B42 Aux Bldg
E1099 Shs 1-2	480V ESF MCC 15B42 Aux Bldg
E1100 Shs 1-2	Motor Control Cabinet Tabulation Index
E1109-000 Shs 0-27 (except Shs 4, 9, 8, 10, 11, 13, 14, 15, 16, 19)	4.16kV Engineered Safety Features System
E1110-000 Shs 0-23 (except Shs 2, 5, 6, 7)	Standby Diesel Generator System Division I
E1111-000 Shs 0-23 (except Shs 2, 5, 6, 7)	Standby Diesel Generator System Division II
E1112-000 Shs 0-5 (except Sh 2)	HPCS Diesel Generator Fuel Oil Transfer
E1115-000 Shs 0-13	480V Load Center ESF Division I

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1116-000 Shs 0-13	480V Load Center ESF Division II
E1117-000 Shs 0-4	125V DC ESF Distribution System
E1118-000 Shs 0-1	125V Battery Chargers
E1120-000 Shs 0-7	Load Shedding and Sequencing Tables
E1155-000 Shs 0-4	Feedwater Leakage Control System
E1159-000 Shs 0-3 (except Sh 1)	Nuclear Boiler System
E1167-000 Shs 0-3	Control Rod Drive System Unit 1
E1174 (8.3-14)	Schematic Diagram C71 RPS MG Set Control System
E1180-000 Shs 0-5	Residual Heat Removal System
E1186-000 Shs 0-45 (except Shs 10, 21,26,28,35,36,39)	Combustible Gas Control System

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1203-000	Reactor Water Cleanup System Index and Relay Tabulation
E1205-000	Filter/Demineralizer Sys Index and Aux Rly Tab
E1205 Sh 1	Filter/Demineralizer Sys Backwash Rcvg Tk Containment Isln Valve F101
E1205 Sh 2	Filter/Demineralizer Sys Backwash Rcvg Tank Aux
E1205 Sh 6	Filter/Demineralizer Sys 120V AC Fuse Panel Power Supplies
E1205 Sh 8	Filter/Demineralizer Sys Backwash Rcvg Tank Containment Bldg Isln Valve F106
E1208-000	Fuel Pool Cooling and Cleanup Index
E1208 Sh A	Fuel Pool Cooling and Cleanup Filter/Demin Sys Relay Tabulation
E1208 Sh 1	Fuel Pool Cooling and Cleanup Filter/Demin Sys Backwash Aux Bldg Isln Valve F253

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1208 Sh 8	Fuel Pool Cooling and Cleanup Filter/Demin Sys 120V AC Power Supply
E1213 Sh 0	Containment Cooling Sys Index
E1213 Sh 1	Containment Cooling Sys Relay Tabulation
E1213 Sh 2	Containment Cooling Sys Containment Isolation Valve F011
E1213 Sh 3	Containment Cooling Sys Containment Isolation Valve F012
E1213 Sh 4	Containment Cooling Sys Drywell Isolation Valve F015
E1213 Sh 5	Containment Cooling Sys Drywell Isolation Valve F013
E1213 Sh 27	Containment Cooling Sys 125V DC and 120V AC Fuse Panel Power Supply
E1213 Sh 28	Containment Cooling Sys 120V AC Fuse Panel Power Supply
E1213 Sh 29	Containment Cooling Sys Control Room Ann
E1213 Sh 30	Containment Cooling Sys Control Room Ann

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1213 Sh 32	Containment Cooling Sys Computer Points
E1215 Sh 10	P72 Drywell Chilled Water System Isolation Control F121-A
E1215 Sh 9	P72 Drywell Chilled Water System Isolation MOV F123-B
E1219-000 Shs 0-24 (except Shs 15, 16)	Containment Drywell and Aux Bldg Instrumentation and Control
E1220-000 Shs 0-13 (except Shs 5, 6)	Suppression Pool Makeup Sys and Aux Relay Tabulation
E1221 Sh 0	Condensate and Refueling Water Storage and Transfer Index Unit 1
E1221 Sh 10	Condensate and Refueling Water Storage and Transfer Aux Bldg Isln Valve F062
E1221 Sh 11	Condensate and Refueling Water Storage and Transfer Aux Bldg Isln Valve F064
E1221 Sh 13	Condensate and Refueling Water Storage and Transfer 120V AC & 125V DC Fuse Panel Power Supplies

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1221 Sh 17	Condensate and Refueling Water Storage and Transfer
E1221 Sh 18	Condensate and Refueling Water Storage and Transfer
E1222 Sh 0	Makeup Water Treatment Sys Index & Relay Tabulation
E1222 Sh 1	Makeup Water Treatment Sys Isln Motor Operated Valve F018-B
E1222 Sh 2	Makeup Water Treatment Sys Isln Motor Operated Valve F017-A
E1222 Sh 3	Makeup Water Treatment Sys Aux Bldg Isln Valve F024
E1222 Sh 4	Makeup Water Treatment Sys 120V AC Power Supplies
E1222 Sh 5	Makeup Water Treatment Sys Computer Points
E1225-000 Shs 0-56 (except Shs 12, 38, 47)	Standby Service Water Sys
E1226 Sh 0	Component Cooling Water System Index
E1228 Sh 0	Plant Service Water Sys Index

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1228 Sh A	Plant Service Water Sys Relay Tabulation
E1228 Sh 7	Plant Service Water Sys SSW Crosstie Motor Operated Valve F067
E1228 Sh 8	Plant Service Water Sys SSW to Plant Service Water Crosstie Motor-Operated Valve F054
E1228 Sh 9	Standby Service Water System Aux Bldg Outboard Valve F068
E1228 Sh 17	Plant Service Water Sys Power Distribution
E1228 Sh 21	Plant Service Water Sys Aux Bldg Isln Valve F121
E1228 Sh 22	Plant Service Water Sys Computer Points
E1228 Sh 23	Plant Service Water Sys Computer Points
E1229 Sh 0	Instrument Air System Index
E1229 Sh A	Instrument Air System Relay Tabulation
E1229 Sh 6	Instrument Air System 120V AC Fuse Panel Power Supplies

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1229 Sh 9	Instrument Air System Aux Bldg Isln Valve F026A
E1229 Sh 10	Instrument Air System Isln Control Motor Operated Valve F003A
E1229 Sh 11	Instrument Air System Isln Control Motor Operated Valve F005B
E1229 Sh 13	Instrument Air System Computer Points
E1229 Sh 14	Instrument Air System Computer Points
E1234 Sh 0	Plant Chilled Water System Index
E1234 Sh A	Plant Chilled Water Sys Relay Tabulation
E1234 Sh 3	Plant Chilled Water Sys Isln Valve F148
E1234 Sh 4	Plant Chilled Water Sys Isln Valve F149
E1234 Sh 5	Plant Chilled Water Sys Aux Bldg Isln Valve F306

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1234 Sh 13	Plant Chilled Water Sys 120V AC & 125V DC Fuse Panel Power Supplies
E1234 Sh 14	Plant Chilled Water Sys 120V AC Fuse Panel Power Supplies
E1234 Sh 16	Plant Chilled Water Sys Computer Points
E1239 Sh 0	Service Air System Index
E1239 Sh A	Service Air System Relay Tabulation
E1239 Sh 3	Service Air System Isln Valve F105
E1239 Sh 4	Service Air System Isln Valve F221A
E1239 Sh 5	Service Air System Isln Valve F195B
E1239 Sh 7	Service Air System 120V AC Fuse Panel Power Supplies
E1239 Sh 9	Service Air System Computer Points
E1240 Sh 0	Suppression Pool Cleanup System

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1253 Sh 0	Aux Bldg Ventilation System Index
E1253 Sh 6	Aux Bldg Ventilation Sys Aux Bldg Vent Sys Isln Valve F007
E1253 Sh 11	Aux Bldg Ventilation Sys ESF 120V AC Fuse Panel Power Supplies
E1254 Sh 0	Fuel Handling Area Vent System Index
E1254 Sh A	Fuel Handling Area Vent Sys Relay Tabulation
E1254 Sh 1	Fuel Handling Area Vent Sys Isln Valve F004
E1254 Sh 2	Fuel Handling Area Vent Sys Isln Valve F011
E1254 Sh 3	Fuel Handling Area Vent Sys Isln Valve F019
E1254 Sh 26	Fuel Handling Area Vent Sys 120V AC & 125V DC Fuse Panel Power Supplies
E1257-000 Shs 0-25 (except Sh 9)	Standby Gas Treatment Sys

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1258-000 Shs 0-5	Emergency Pump Room Vent Sys Relay Tabulation
E1265-000 Shs 0-10 except Sh 4)	Diesel Generator Room Vent Sys
E1266-000 Shs 0-12 (except Sh 5)	Standby Service Water Pump House Vent Sys
E1267 Sh 12	Safeguard Switchgear and Battery Rooms Vent Sys and Relays
E1269-000 Shs 0-2	Engineered Safety Features Electrical Switchgear Room Cooling Sys
E1271 Sh 0	Floor and Equipment Drains Sys
E1271 Sh A	Floor and Equipment Drains Sys Relay Tabulation
E1271 Sh 13	Floor and Equipment Drains Sys Fl & Eqpt Dr Isln Valve F004
E1271 Sh 14	Floor and Equipment Drains Sys Fl & Eqpt Dr Isln Valve F099
E1271 Sh 15	Floor and Equipment Drains Aux Bldg Isln Valve F158

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1271 Sh 16	Floor and Equipment Drains Sys Drywell Chem Waste Isln Motor Operated Valve F096A
E1271 Sh 17	Floor and Equipment Drains Sys 120V AC Power Supplies
E1271 Sh 18	Floor and Equipment Drains Sys Computer Points
E1283	Control Room PGCC Isolators, Digital and Analog
E1284	Control Room PGCC Isolators, Digital and Analog
E1285	Control Room PGCC Isolators, Digital and Analog
E1286-000 Shs 0-6	Local Isolators
E-1358-1F	Appendix R Alternate Shutdown Engraving - 1H22-P295
E-1358-1G	Appendix R Alternate Shutdown Engraving - 1H22-P296
E-1358-1J	Appendix R Alternate Shutdown Engraving - 1H22-P298

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E-1358-1K	Appendix R Alternate Shutdown Engraving - 1H22-P299
E1625	Lighting and Communication Plan Aux & Containment Bldg El 119-0 and 120-10
E1626	Lighting and Communication Plan Aux and Containment Bldg El 139-0 & 145-4
E1627	Lighting and Communication Plan Aux and Containment Bldg El 161-10, 116-0 and 170-0
E1672	Raceway Plan Aux Bldg El 93-0 Area 7
E1673	Raceway Plan Aux Bldg El 93-0 Area 8
E1675	Raceway Plan Aux Bldg El 93-0 Area 10
E1676	Raceway Plan Aux Bldg El 119-0 Area 7
E1677	Raceway Plan Aux Bldg El 119-0 Area 8
E1678	Raceway Plan Aux Bldg El 119-0 Area 9

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1679	Raceway Plan Aux Bldg El 119-0 Area 10
E1680	Raceway Plan Aux Bldg El 139-0 Area 7
E1681	Raceway Plan Aux Bldg El 139-0 Area 8
E1682	Raceway Plan Aux Bldg El 139-0 Area 9
E1683	Raceway Plan Aux Bldg El 139-0 Area 10
E1684	Raceway Plan Aux Bldg El 166-0 and 170-0 Area 7
E1685	Raceway Plan Aux Bldg El 166-0 and 170-0 Area 8
E1686	Raceway Plan Aux Bldg El 166-0 and 167-6 Area 9
E1687	Raceway Plan Aux Bldg El 166-0 and 167-6 Area 10
E1688	Raceway Plan Aux Bldg El 185-0 Area 9
E1689	Raceway Plan Aux Bldg El 185-0 Area 10

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1690	Raceway Plan Aux Bldg El 208-10 Area 9
E1691	Raceway Plan Aux Bldg El 208-10 Area 10
E1692	Raceway Plan Aux Bldg El 245-0 and 228-0 Area 9
E1693	Aux Bldg Vertical Cable Tray Chase
E1694	Raceway Aux Bldg Misc Sect and Details
E1695	Raceway Plan Aux Bldg Misc Sect and Details
E1700	Raceway Plan Containment Bldg El 93-0 and 100-9 Area 11
E1701	Raceway Plan Containment Bldg El 114-6 and 120-10 Area 11
E1702A	Raceway Plan Containment Bldg El 135-4 Azimuth 0 to 90 Area 11
E1702B	Raceway Plan Containment Bldg El 135-4 AZ 90 to 180 Area 11
E1702C	Raceway Plan Containment Bldg El 135-4 AZ 180 to 270 Area 11

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1702D	Raceway Plan Containment Bldg El 135-4 AZ 270 to 0 Area 11
E1702F	Raceway Plan Hydrogen Igniter System
E1703	Raceway Plan Containment Bldg El 161-10 and 170-0 Area 11
E1704	Raceway Plan Containment Bldg El 184-6 Area 11
E1705	Raceway Plan Containment Bldg El 208-10 Area 11
E1706	Raceway Containment Bldg Misc Sect and Details
E1707	Raceway Containment Bldg Misc Sect and Details
E1708	Raceway Plan Containment Bldg Developed View of Drywell Wall Inside Drywell AZ 90 to 270
E1709	Raceway Plan Containment Bldg RPIS Channel Under RPV El 114-6 Area 11
E1710	Raceway Plan Containment Bldg RPS Channel Under RPV El 113-6 Area 11

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E1711	Raceway Details Containment Bldg Cable and Channel Under RPV Area 11
E-1712	Raceway Containment Bldg Developed View of Drywell Wall Inside Drywell AZ 270 to 90
E-1713	Raceway Containment Bldg Misc Sects and Details
E-1714	Raceway Plan Diesel Generator Bldg Area 12 El 133-0
E-1715	Raceway Plan Diesel Generator Bldg Area 12 El 158-0
E-1716	Raceway Plans Cooling Towers (SSW) No. 1 & 2
E-1719	Raceway Plan Diesel Generator Bldg Misc Sects & Details
E-1805	Raceway Plan Turbine Bldg El 93-0 Fire & Smoke Detection System Unit 1
E-1806	Raceway Plan Turbine Bldg El 113-0 Fire & Smoke Detection System Unit 1

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E-1807	Raceway Plan Turbine Bldg El 133-0 Fire & Smoke Detection System Unit 1
E-1808	Raceway Plan Turbine Bldg El 166-0 Fire & Smoke Detection System Unit 1
E-1809	Raceway Plan Aux Bldg & Cntmt El 93-0, 100-9 Fire & Smoke Detection System Unit 1
E-1800	Raceway Plan Aux Bldg & Cntmt El 119-0, 120-10, 114-6 Fire & Smoke Detection System Unit 1
E-1801	Raceway Plan Aux Bldg & Cntmt El 139-0, 135-4, 147-7 Fire & Smoke Detection System Unit 1
E-1802	Raceway Plan Aux Bldg & Cntmt El 161-10, 166-0 Fire & Smoke Detection System Unit 1
E-1803	Raceway Plan Aux Bldg & Cntmt El 184-6, 185-0 Fire & Smoke Detection System Unit 1
E-1804	Raceway Plan Aux Bldg & Cntmt El 208-10 Fire & Smoke Detection System Unit 1

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 2, Electrical (Bechtel)

Drawing No. (FSAR Figure No.)	Title
E-7177 SH 0	Plant Air System Index & Relay Tabulation
E-7177 SH 5	Plant Air System Heater Control and Power Distribution

Section 3, Electrical and Instrumentation/Control (General Electric)

828E234BA Shs 1-4	Standby Liquid Control System (SLC)
828E525BA Shs 1-5	Feedwater Control System
828E231BA Shs 1-4	Control Rod Drive - Hydraulic System
865E344BA Shs 1-4	Reactor Water Cleanup System (RWCS)
828E447 Shs 1-3	Jet Pump Instrumentation System
828E446 Shs 1-35	Reactor Recirculation System
828E549BA Shs 1-5	Nuclear Boiler Process Instrumentation System
828E534BA Shs 1-20	Residual Heat Removal System (RHR)

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 3, Electrical and Instrumentation/Control (General Electric)

Drawing No. (FSAR Figure No.)	Title
828E536BA Shs 1-7	High-Pressure Core Spray System (HPCS)
828E537BA Shs 1-15	HPCS - Power Supply
828E535BA Shs 1-7	Low-Pressure Core Spray System (LPCS)
828E539BA Shs 1-13	Reactor Core Isolation Cooling System (RCIC)
828E444BA Shs 1-12	Automatic Depressurization System (ADS)
828E445BA Shs 1-15	Nuclear Steam Supply Shutoff System
E-1187 Shs 1-16	Leak Detection System
E-1177 Shs 1-36	Process Radiation Monitoring System
E-1176 Shs 1-6	Neutron Monitoring System (NMS) - Startup Range Detector Drive Control
E-1171 Shs 1-22	NMS - Startup Range

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 3, Electrical and Instrumentation/Control (General Electric)

Drawing No. (FSAR Figure No.)	Title
E-1170 Shs 1-5	NMS - Traversing Incore Probe
E-1172 Shs 1-60 (except Shs 21, 22, 53, 55)	NMS - Power Range
E-1173 Shs 1-28 (7.2-1)	Reactor Protection System (RPS)
828E532BA Shs 1-3 Scheme	RPS Interconnection
E-1174 (8.3-11)	RPS MG Set Control Remote Shutdown System
E-1151 Shs 1-30	Offgas System
E-1210 Shs 1-3 Shs 6-21	Main Steamline Isolation Valve Leakage Control System (MSIV-LCS)
E-1165 Shs 1-22	Rod Control and Information System
E-1206 Shs 1-24	RWCU - F/D Control
E-1207 Shs 1-16	Fuel Pool Cooling and Cleanup System

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 3, Electrical and Instrumentation/Control (General Electric)

Drawing No. (FSAR Figure No.)	Title
E-1209 Shs 1-24	Fuel Pool Filter Demineralizer System
762E401 Shs 1-4	Nuclear Boiler System FCD
105D4920 Shs 1-5	Reactor Recirculation System FCD
762E429 Shs 1-7	Control Rod Drive Hydraulic System FCD
762E434	Standby Liquid Control System FCD
762E459 Shs 1-7	Neutron Monitoring System FCD
944E453 Shs 1-5	Residual Heat Removal System FCD
762E294BA Shs 1-2	Low-Pressure Core Spray System FCD
851E892BA Shs 1-3	High-Pressure Core Spray System FCD
105D5046 Shs 1-5	HPCS Power Supply FCD
105D5116 Shs 1-3	Leak Detection System FCD

TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

Section 3, Electrical and Instrumentation/Control (General Electric)

Drawing No. (FSAR Figure No.)	Title
865E343 Shs 1, 4	MSIV Leakage Control System FCD
762E297BA Shs 1-5	Reactor Core Isolation Cooling System FCD
762E298BA, Rev. 5 Shs 1-2 (8.3 - 12a, b)	Div 3 HPCS Power System
762E298BA (8.3 - 13)	Div 3 HPCS ESF - DC System
762E407	Reactor Core Cleanup System FCD
762E414BA Shs 1-2	Fuel Pool Cooling & Cleanup System FCD
807E523BA Shs 1-5	Offgas System FCD
944E990 Shs 1-3 (7.2-1a,b,c)	Reactor Protection System IED
762E293WJ Shs 1-4 (7.6-2a,b,c,d)	Leak Detection System IED
3636-120-001 (9.5-15)	HPCS Jacket Cooling Water System

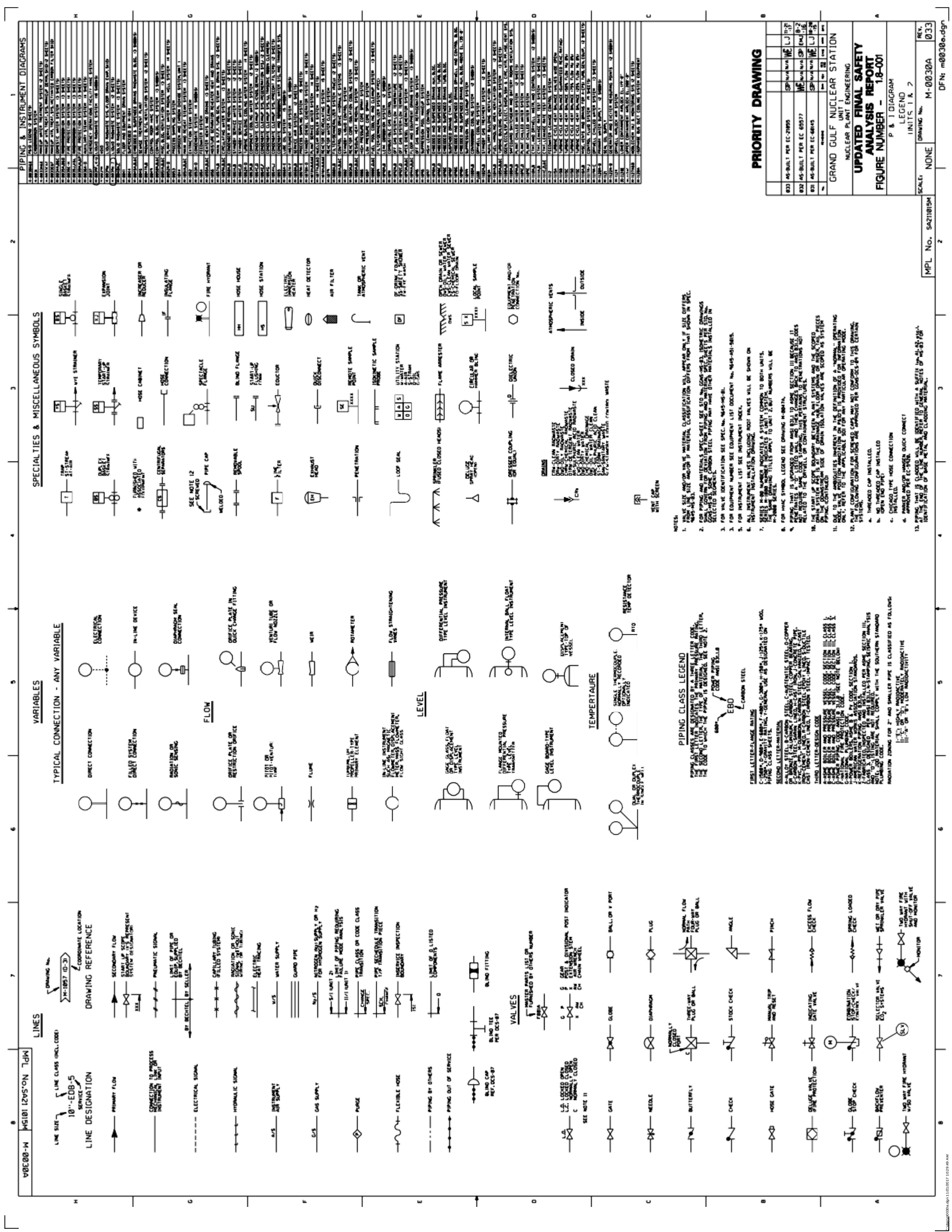
TABLE 1.7-1: NONPROPRIETARY ELECTRICAL AND INSTRUMENTATION/ CONTROL DRAWINGS INCORPORATED BY REFERENCE (CONTINUED)

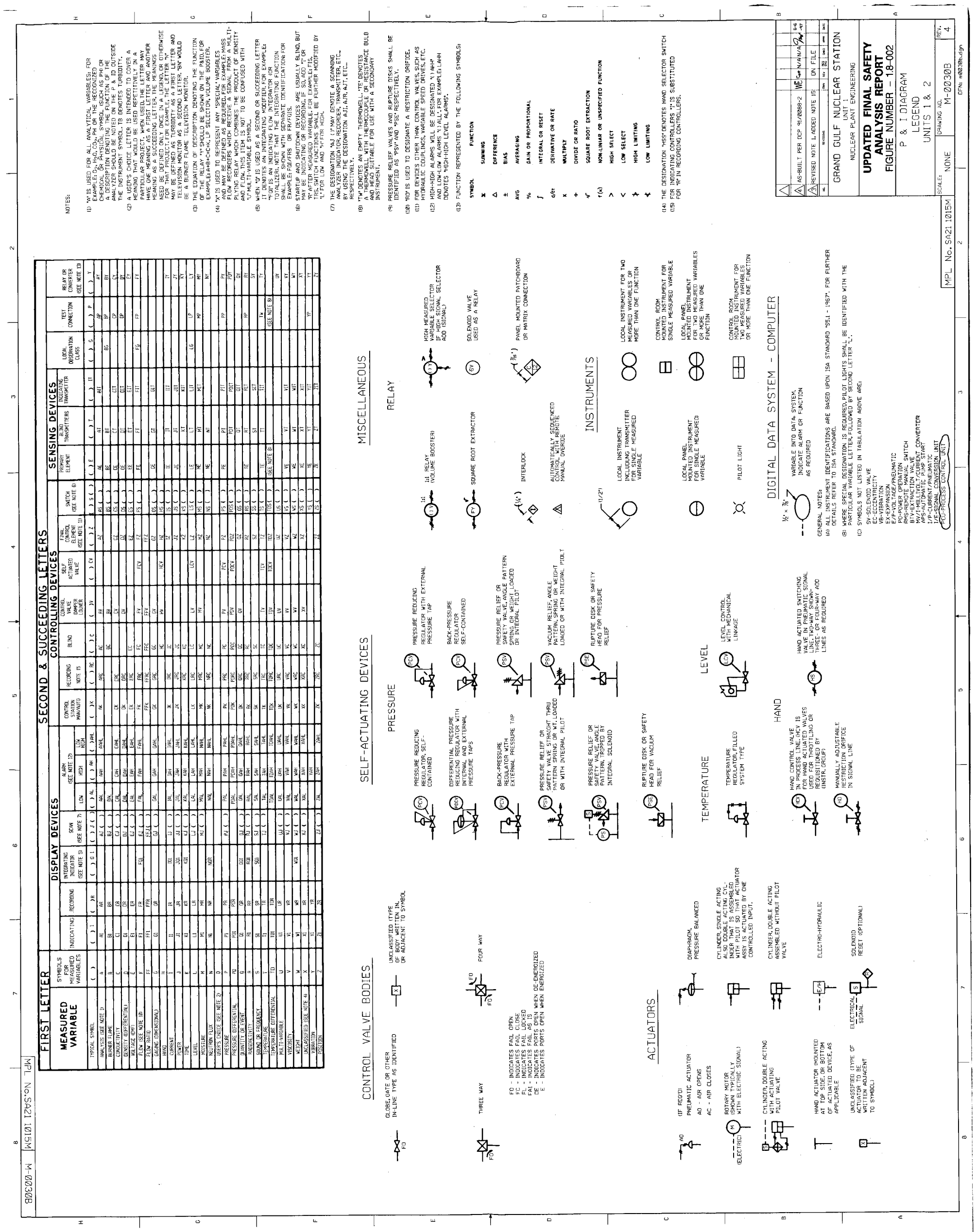
Section 3, Electrical and Instrumentation/Control (General Electric)

Drawing No. (FSAR Figure No.)	Title
3636-130-001 (9.5-16)	HPCS Air Start System
3636-119-1 (9.5-18)	HPCS Lubrication System
3636-119-2 (9.5-18)	

1.8 SYMBOLS USED IN ENGINEERING DRAWINGS

The symbols applicable to piping and instrumentation diagrams (P&IDs) used throughout this report are shown in Figures 1.8-1 through 1.8-3.


PRIORITY DRAWING
GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER – 1.8-001
P & I DIAGRAM
LEGEND
UNITS 1 & 2
M-0030A
PIPING & INSTRUMENT DIAGRAMS
SPECIALTIES & MISCELLANEOUS SYMBOLS
VARIABLES
TYPICAL CONNECTION - ANY VARIABLE
FLOW
LEVEL
TEMPERATURE
PIPING CLASS LEGEND
LINES
LINE DESIGNATION
DRAWING REFERENCE
VALVES
NOTES: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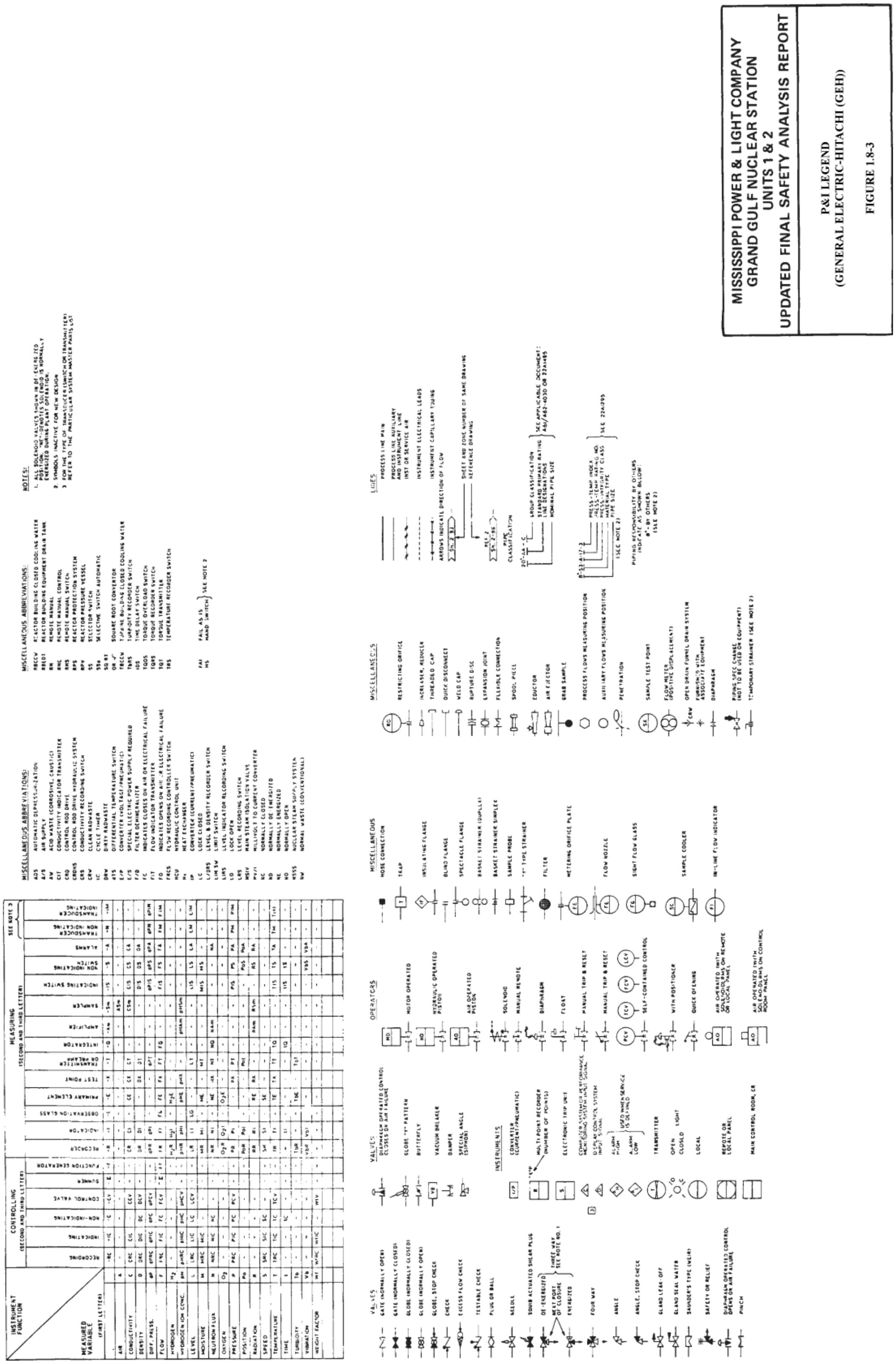
SECOND & SUCCEEDING LETTERS
DISPLAY DEVICES
SENSING DEVICES
CONTROLLING DEVICES
CONTROL VALVE BODIES
ACTUATORS
SELF-ACTUATING DEVICES
PRESSURE
TEMPERATURE
LEVEL
HAND
MISCELLANEOUS
RELAY
INSTRUMENTS
DIGITAL DATA SYSTEM - COMPUTER
NOTES:
GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
P & I DIAGRAM
LEGEND
UNITS 1 & 2
M-0230B


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
P&I LEGEND
(GENERAL ELECTRIC-HITACHI (GEH))
FIGURE 1.8-3

1.9 ABBREVIATIONS

Table 1.9-1 is a list of the abbreviations used in this Updated Final Safety Analysis Report.

TABLE 1.9-1: ABBREVIATIONS

AASHO	American Association of State Highway Officials
A-CC	Allis-Chalmers Corporation
A-CPSI	Allis-Chalmers Power Systems Inc.
ACI	American Concrete Institute
ACRS	Advisory Committee for Reactor Safeguards
ADHRS	Alternate Decay Heat Removal Subsystem
ADS	Automatic Depressurization System
AE	Architect Engineer
AISC	American Institute of Steel Constructors
ALARA	"as low as reasonably achievable"
ANI	American Nuclear Insurers
ANSI	American National Standards Institute
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARM	Area Radiation Monitor
ARSS	Alternate Reactor Scram System
ASCE	American Society of Civil Engineers
ASCS	Agricultural Stabilization Conservation Service
ASDC	Alternate Shutdown Cooling
ASTM	American Society for Testing Materials
ATWS	Anticipated Transients Without Scram
AWS	American Welding Society
BEA	Bureau of Economic Analysis
BOF	Bottom of Active Fuel
BOP	Balance of Plant
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CAMS	Continuous Air Monitors
CAV	Cumulative Average Velocity
CCW	Component Cooling Water
CFFF	Condensate Full Flow Filtration System
CFR	Code of Federal Regulations
CGCS	Combustible Gas Control System
CHF	Critical Heat Flux
CM	Center of Mass
CMAA	Crane Manufacturing Association of America
CP	Construction Permit
CPR	Critical Power Ratio

TABLE 1.9-1: ABBREVIATIONS (Continued)

CR	Center of Rigidity
CRACIS	Control Room Atmospheric Control and Isolation System
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRPI	Control Rod Position Indication
CRVICS	Containment and Reactor Vessel Isolation Control System
CRWE	Control Rod Withdrawal Error
CRWST	Condensate and Refueling Water Storage and Transfer
CST	Condensate Storage Tank
CTO	Checkout and Turnover Organization
DBA	Design-Basis Accident
DCS	Distributed Control System
DELS	Diesel Engine Lubrication System
DG	Diesel Engine-Generator
DGCAIES	Diesel Generator Combustion Air Intake and Exhaust System
DGCWS	Diesel Generator Cooling Water System
DGSS	Diesel Generator Starting System
DOP	Dioctyl Pathalate
DPA	Displacements Per Atom
DPF	Design Project Flood
ECA	Engineering Change Authorization
ECCS	Emergency Core Cooling System
ECN	Engineering Change Notice
ECP	Electrochemical Corrosion Potential
EDS	Engineering Data Systems
EFCV	Excess Flow Check Valve
EHC	Electrohydraulic Control
EIC	Eberline Instrument Corporation
EOC	End of Cycle
EPU	Extended Power Uprate
ER	Environmental Report
ERTS	Earth Resources Technology Satellite
ESF	Engineered Safety Feature
FA	Full Arc (Mode of TCV Operation)
FANP	Framatome-ANP
FAP	Fatigue Analysis Program
FCD	Functional Control Diagram

TABLE 1.9-1: ABBREVIATIONS (Continued)

FCV	Flow Control Valve
FEDS	Floor and Equipment Drainage System
FDDR	Field Deviation Disposition Request
FHA	Fuel Handling Accident
FLECHT	Full-Length Emergency Cooling Heat Transfer
FM&IS	Flow Monitoring and Isokinetic Sampling
FMD	Forced Helium Dehydration
FMEA	Failure Modes and Effects Analysis
FPCC	Fuel Pool Cooling and Cleanup
FWCF	Feedwater Controller Failure
FWLCS	Feedwater Leakage Control System
FSAR	Final Safety Analysis Report
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy Americas LLC
GESSAR	General Electric Standard Safety Analysis Report
GETAB	General Electric Thermal Analysis Basis
GGNS	Grand Gulf Nuclear Station
GNP	Gross National Product
HCU	Hydraulic Control Unit
HEPA	High-Efficiency Particulate Air/Absolute (referring to filters)
HMI	Human Machine Interface
HPCS	High Pressure Core Spray
HPU	Hydraulic Power Unit
HTGR	High-Temperature Gas-Cooled Reactor
H&V	Heating and Ventilating
HVAC	Heating, Ventilating, and Air-conditioning
HWC	Hydrogen Water Chemistry
HX	Heat Exchanger
IAC	Interim Acceptance Criteria (NRC)
IBA	Intermediate Break Accident
IDS	Instrument Data Sheet
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	lntergranular Stress Corrosion Cracking
IRM	Intermediate Range Monitor
ISI	Inservice Inspection
IST	Inservice Testing
KWU	Kraftwerk Union
LCD	Local Climatological Data
LCO	Limiting Condition of Operation
LCR	Logarithm of Count Rate

TABLE 1.9-1: ABBREVIATIONS (Continued)

LCS	Leakage Control System
LDS	Leak-Detection System
LFMGS	Low Frequency Motor-Generator Set
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LOFWH	Loss of Feedwater Heating
LOOP	Loss-of-Offsite Power
LPCI	Low-Pressure Coolant Injection
LPCS	Low-Pressure Core Spray
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPZ	Low Population Zone
LRNB	Generator Load Reject W/0 Bypass
LSSS	Limiting Safety System Setting
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MELLLA	Maximum Extended Load Line Limit Analysis
MEOD	Maximum Extended Operating Domain
MG	Motor-Generator Set
MLD	Mean Low Water Datum
MP&L	Mississippi Power & light
MPM	MP Machinery and Testing, LLC
MPC	Maximum Permissable Concentration
MSL	Mean Sea Level
MSL	Main Steam Line
MSIV	Main Steam Isolation Valve
MSIV-LCS	Main Steam Isolation Valve Leakage Control System
NAPSIC	North American Power Systems Interconnection Committee
NB	Nuclear Boiler
NBR	Nuclear Boiler Rated (power)
NCC	Network Control Center
NCIG	Nuclear Construction Issues Group
NCMA	National Concrete Masonry Association
NDT	Nil-Ductility Transition/Nondestructive Testing
NED	Nuclear Energy Division
NFPA	National Fire Protection Association
NMS	Neutron-Monitoring System
NPDES	National Pollutant Discharge Elimination System

TABLE 1.9-1: ABBREVIATIONS (Continued)

NPSH	Net Positive Suction Head
NPSHA	Net Positive Suction Head Available
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NSSSS	Nuclear Steam Supply Shutoff System
NSOA	Nuclear Safety Operational Analysis
ODCM	Offsite Dose Calculation Manual
OSRC	On-Site Safety Review Committee
OBE	Operating Basis Earthquake
OFS	Orificed Fuel Support
OL	Operating License
OPRM	Oscillation Power Range Monitor
ORE	Occupational Radiation Exposures
OSHA	Occupational Safety & Health Administration
PA	Public Address
PAM	Post Accident Monitoring
PASS	Post Accident Sampling System
PBS	Power Systems Branch
PCIOMR	Preconditioning Interim Operational Management Recommendation
PCS	Process Computer System
PCT	Peak Cladding Temperature
P&ID	Piping and Instrumentation Diagram
PMF	Probable Maximum Flood
PMF	Probable Maximum Flood
PPA	Peak to Peak Pressure Amplitude
PQL	Product Quality Checklist
PRA	Peak Recording Accelerometers
PRFDS	Pressure Regulator Failure- Down Scale
PRM	Power Range Monitor
PRM	Process Radiation Monitoring
PSAR	Preliminary Safety Analysis Report
PSRC	Plant Safety Review Committee
PSS	Process Sampling System
PSTF	Pressure Suppression Test Facility
PSW	Plant Service Water
PSWRW	Plant Service Water Radial Well
PTLR	Pressure-Temperature Limits Report
PUSAR	Power Uprate Safety Analysis Report
PVS	Plant Vent Stack
PWR	Pressurized Water Reactor

TABLE 1.9-1: ABBREVIATIONS (Continued)

QA/QC	Quality Assurance/Quality Control
QGC	Quality Group Classification
RBM	Rod Block Monitor
RCA	Radiological Controlled Area
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RFP	Reactor Feed Pump
RFWT	Reduced Feedwater Temperature
R.G.	Regulatory Guide
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RO	Reactor Operator
RPCS	Rod Pattern Control System
RPIS	Rod Position Information System
RPTS	Recirculation Pump Trip System
RPIS	Rod Position Information System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRS	Required Response Spectra
RSO	Reactor System Outline
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
RWM	Rod Worth Minimizer
RWST	Refueling Water Storage Tank
SACF	Single Active Component Failure
SAF	Single Active Failure
SAR	Safety Analysis Report
SBA	Small Break Accident
SBO	Station Blackout
SCC	Stress Corrosion Cracking
SDIV	Scram Discharge Instrument Volume
SDV	Scram Discharge Volume
SER	Safety Evaluation Report
SERI	System Energy Resources, Inc
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SLCS	Standby Liquid Control System

TABLE 1.9-1: ABBREVIATIONS (Continued)

SLO	Single Loop Operation
SMA	Strong Motion Accelerometers
SMEPA	South Mississippi Electric Power Association
SOE	Single Operator Error
SPC	Siemens Power Corporation
SPCU	Suppression Pool Cleanup
SPMU	Suppression Pool Makeup
SQCF	Seismic Qualification File
SQRT	Seismic Qualification Review Team
SRDI	Safety-Related Display Information
SRFI	Slow Recirculation Flow Increase
SRLR	Supplemental Reload Licensing Report
SRM	Source Range Monitor
SRO	Senior Reactor Operator
SRP	Standard Review Procedure
SRSS	Square Root of the Sum of the Squares
SRV	Safety/Relief Valve
SRVDLs	Safety Relief Valve Discharge Line
SS	Safe Shutdown
SSE	Safe Shutdown Earthquake
SSW	Standby Service Water
TAF	Top of Active Fuel
TBCWS	Turbine Building Cooling Water System
TCV	Turbine Control Valve
TG	Turbine-Generator
TIP	Traversing Incore Probe
TRM	Technical Requirements Manual
TRS	Test Response Spectra
TSVC	Turbine Stop Valve Closure Load
TTNB	Turbine Trip W/O Bypass
UBC	Uniform Building Code
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
URC	Ultrasonic Cleaning
USGS	U.S. Geological Survey
UT	Ultrasonic Testing
Vac	Volts-Alternating Current
VBWR	Vallecitos Boiling Water Reactor
VCT	Vertical Cask Transport
VRF	Velocity Range Factor

1.10 DRAWING NUMBER-FSAR FIGURE NUMBER CROSS-REFERENCE

Table 1.10-1 is a list of Bechtel drawing numbers cross-referenced with their corresponding FSAR figure numbers. This information has been provided to supplement FSAR figures containing P&IDs or SFDS which have been flagged to indicate that specific process lines are continued on other drawings.

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS

Drawing No.	System Identification	FSAR Figure No.
C-0012	Orientation of Principal Structures	1.2-1
M-0001	General Arrangement Plan at El. 93'-0" and 100'-9"	1.2-2
M-0002	General Arrangement Plan at El. 113'-0", 111'-0", 119'-0", 112'-0", and 114'-6"	1.2-3
M-0003	General Arrangement Plan at El. 133'-0", 148'-0", 139'-0", 135'-4", and 147'-7"	1.2-4
M-0004	General Arrangement Plan at El. 166'-0", 161'-10", and 170'-0"	1.2-5
M-0005	General Arrangement Plan at El. 184'-6", 184'-0", and 189'-0"	1.2-6
M-0006	General Arrangement Plan at El. 208'-10"	1.2-7
M-0007	General Arrangement Sections "AA" and "BB"	1.2-8
M-1001	Turbine Building Plan at EL. 93'-0"	1.2-9c
M-1006	Turbine Building General Arrangement Sections "A-A" and "B-B"	1.2-9A
M-1007	Turbine Building General Arrangement Sections "C-C," "D-D," and "E-E"	1.2-9B
M-0017	Radwaste Building Plan at El. 93'-0"	1.2-10
M-0018	Radwaste Building Plan at El. 118'-0"	12.3-6
M-0019	Radwaste Building Plan at El. 136'-0"	12.3-7
M-0020	Radwaste Building Plan Sections "A-A" and "B-B"	1.2-13 12.3-8
M-0021	Radwaste Building Sections "C-C" and "D-D"	1.2-14
M-015.0-N1W20W001N-1.1-001	Natural Draft Cooling Tower	1.2-15
M-7005	Auxiliary Cooling Tower	1.2-16

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-0030 A	P&ID Legend	1.8-1
M-0030 B	P&ID Legend	1.8-2
M-0033 A	Makeup Water Treatment System	9.2-11
M-0033 B	Makeup Water Treatment System	9.2-12
M-0034 A	Domestic Water System	9.2-13
M-0034 B	Domestic Water System	9.2-14
M-0035 A	Fire Protection System	9.5-1
M-0035 B	Fire Protection System	9.5-2
M-0035 D	Fire Protection System	9.5-4
M-0035 E	Fire Protection System	9.5-5
M-0035 F	Fire Protection System	9.5-6
M-0035 G	Fire Protection System	9.5-7
M-0035 H	Fire Protection System	9.5-8
M-0035 J	Fire Protection System	9.5-8a
M-0035 K	Fire Protection System	9.5-8b
M-0035 L	Fire Protection System	9.5-8c
M-0035 R	Fire Protection System	9.5-8e
M-0036 B	Auxiliary Steam System	9.5-20 (Sh. 1)
M-0036 C	Auxiliary Steam System	9.5-20a
M-0036 D	Auxiliary Steam System	9.5-20 (Sh. 2)
M-0039 K	Liquid Radwaste System	11.2-1
M-0039 L	Liquid Radwaste System	11.2-2
M-0039 M	Liquid Radwaste System	11.2-3
M-0039 N	Liquid Radwaste System	11.2-4
M-0039 P	Liquid Radwaste System	11.2-5
M-0039 Q	Liquid Radwaste System	11.2-6
M-0039 R	Liquid Radwaste System	11.2-7
M-0039 S	Liquid Radwaste System	11.2-8
M-0039 T	Liquid Radwaste System	11.2-9
M-0039 U	Liquid Radwaste System	11.2-10
M-0039 V	Liquid Radwaste System	11.2-11
M-0039 W	Liquid Radwaste System	11.2-12
M-0039 X	Liquid Radwaste System	11.2-12a
M-0039 Y	Liquid Radwaste System	11.2-12b
M-0040 A	Solid Radwaste System	11.4-1
M-0040 B	Solid Radwaste System	11.4-1a
M-0040 C	Solid Radwaste System	11.4-1b

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-0040 D	Solid Radwaste System	11.4-1c
M-0041	Floor and Equipment Drain System	9.3-12
M-0045 A	Embedded and Suspended Drains Radwaste Bldg.	9.3-13
M-0045 B	Embedded and Suspended Drains Radwaste Bldg.	9.3-14
M-0045 C	Embedded and Suspended Drains Pipe Tunnel & Waste Treatment Building Units 1 & 2	9.3-14a
M-0046	Sewage Treatment Plant	9.2-15
M-0047 A	Radwaste Building Ventilation System	9.4-4
M-0047 B	Radwaste Building Ventilation System	9.4-5
M-0047 C	Radwaste Building Ventilation System	9.4-005B
M-0049	Control Room HVAC System	9.4-1
M-0050 A	Control Building HVAC System	9.4-16a
M-0050 B	Control Building HVAC System	9.4-16b
M-0050 C	Hot Machine Shop/Decontamination Facility HVAC	9.4-16c
M-0051 A	Miscellaneous Building Ventilation System	9.4-14
M-0051 B	Miscellaneous Building Ventilation System	9.4-15
M-0051 C	Miscellaneous Building Ventilation System	9.4-15a
M-0052 A	Plant Service Water Radial Well System	9.2-27 (Sh. 1)
M-0052 B	Plant Service Water Radial Well System	9.2-27 (Sh. 2)
M-0052 C	Plant Service Water Radial Well System	9.2-27 (Sh. 3)
M-0053	Process Sampling System, Part 4	9.3-8a
M-1044 A	Hydrogen and Carbon Dioxide System	10.2-001-1
M-1044 B	Hydrogen and Carbon Dioxide System	10.2-001-2
M-1051 A	Main and Reheat Steam System	10.3-1 (Sh. 1)

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1051 B	Main and Reheat Steam System	10.3-2
M-1051 C	Main and Reheat Steam System	10.3-3
M-1051 D	Main and Reheat Steam System	10.3-1 (Sh. 2)
M-1052	Extraction Steam System	10.3-4
M-1053 A	Condensate System	10.4-10 (Sh. 1)
M-1053 B	Condensate System	10.4-11
M-1053 C	Condensate System	10.4-12
M-1053 E	Condensate System	10.4-10 (Sh. 2)
M-1054	Feedwater System	10.4-13
M-1054 B	RFPT EHC System	10.4-13B
M-1055 A	Heater Vents and Drains	10.3-5 (Sh. 1)
M-1055 B	Heater Vents and Drains	10.3-6
M-1055 C	Heater Vents and Drains	10.3-7
M-1055 D	Heater Vents and Drains	10.3-5 (Sh. 2)
M-1055 E	Heater Vents and Drains	10.3-5 (Sh. 3)
M-1056 A	Moisture Separator-Reheater Vents and Drains	10.3-8 (Sh. 1)
M-1056 B	Moisture Separator-Reheater Vents and Drains	10.3-8 (Sh. 2)
M-1057 A	Main and RFP Turbine Steam Seal and Drain System	10.4-3
M-1057 B	Main and RFP Turbine Steam Seal and Drain System	10.4-4
M-1059 A	Circulating Water System	10.4-5
M-1059 B	Circulating Water System	10.4-6
M-1059 C	Circulating Water System	10.4-7
M-1059 D	Circulating Water System	10.4-007a
M-1059 E	Circulating Water System	10.4-005-02
M-1060 A	Condenser Air Removal System	10.4-1
M-1060 B	Condenser Air Removal System	10.4-2
M-1061 A	Standby Service Water System	9.2-1
M-1061 B	Standby Service Water System	9.2-2
M-1061 C	Standby Service Water System	9.2-3
M-1061 D	Standby Service Water System	9.2-4
M-1062 A	Turbine Bldg. Cooling Water System	9.2-24
M-1062 B	Turbine Bldg. Cooling Water System	9.2-25
M-1062 C	Turbine Bldg. Cooling Water System	9.2-26

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1062 D	Turbine Bldg. Cooling Water System	9.2-26a
M-1063 A	Component Cooling Water System	9.2-9
M-1063 B	Component Cooling Water System	9.2-10
M-1064 C	Condensate Cleanup System	10.4-9a
M-1064 D	Condensate Cleanup System	10.4-9b
M-1064 E	Condensate Cleanup System	10.4-9c
M-1064 F	Condensate Cleanup System	10.4-9d
M-1064 G	Condensate Cleanup System	10.4-9e
M-1064 H	Condensate Cleanup System	10.4-9f
M-1064 J	Condensate Cleanup System	10.4-9g
M-1065	Condensate and Refueling Water Storage and Transfer System	9.2-16
M-1067 A	Instrument Air System	9.3-1
M-1067 B	Instrument Air System	9.3-2
M-1067 C	Instrument Air System	9.3-2b
M-1067 D	Instrument Air System Auxiliary Building	9.3-2c
M-1067 E	Instrument Air System	9.3-2d
M-1067 F	Instrument Air System	9.3-2e
M-1067 G	Instrument Air System	9.3-2f
M-1067 H	Instrument Air System Containment	9.3-002j
M-1067 M	Instrument Air	9.3-001b
M-1068 A	Service Air System	9.3-3 (Sh. 1)
M-1068 B	Service Air System	9.3-4
M-1068 C	Service Air System	9.3-3 (Sh. 3)
M-1068 D	Service Air System	9.3-3 (Sh. 2)
M-1069 A	Process Sampling System	9.3-5
M-1069 B	Process Sampling System	9.3-6
M-1069 C	Process Sampling System	9.3-7
M-1069 D	Process Sampling System	9.3-7a
M-1070 A	Standby Diesel Generator System	9.5-11
M-1070 B	Standby Diesel Generator System	9.5-12
M-1070 C	Standby Diesel Generator System	9.5-11a
M-1070 D	Standby Diesel Generator System	9.5-12a
M-1072 A	Plant Service Water System	9.2-22
M-1072 B	Plant Service Water System	9.2-23 (Sh. 1)
M-1072 D	Plant Service Water System	9.2-23b

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1072 E	Plant Service Water System	9.2-23c
M-1072 F	Plant Service Water System	9.2-23 (Sh. 2)
M-1072 H	Plant Service Water System	9.2-23a
M-1077 A	Nuclear Boiler System	5.2-6 (Sh. 1)
M-1077 B	Nuclear Boiler System	5.2-7
M-1077 C	Nuclear Boiler System	5.2-8 (Sh. 1&2)
M-1077 D	Nuclear Boiler System	5.2-6 (Sh. 2)
M-1078 A	Reactor Recirculation System	5.4-2 (Sh. 1)
M-1078 B	Reactor Recirculation System	5.4-3
M-1078 E	Reactor Recirculation System	5.4-2 (Sh. 2)
M-1079	Reactor Water Cleanup System	5.4-21
M-1080 A	Filter/Demineralizer System (RWCU)	5.4-25
M-1080 B	Filter/Demineralizer System (RWCU)	5.4-26
M-1081 A	Control Rod Drive Hydraulic System	4.6-7
M-1081 B	Control Rod Drive Hydraulic System	4.6-8 (Sh. 1)
M-1081 C	Control Rod Drive Hydraulic System	4.6-8 (Sh. 2)
M-1082	Standby Liquid Control System	9.3-24
M-1083 A	Reactor Core Isolation Cooling System	5.4-10
M-1083 B	Reactor Core Isolation Cooling System	5.4-11
M-1085 A	Residual Heat Removal System	5.4-16 (Sh. 1)
M-1085 B	Residual Heat Removal System	5.4-17
M-1085 C	Residual Heat Removal System	5.4-16 (Sh. 2)
M-1085 D	Residual Heat Removal System	5.4-17a
M-1086	High Pressure Core Spray System	6.3-1
M-1087	Low Pressure Core Spray System	6.3-4
M-1088 C	Fuel Pool Cooling and Cleanup System	9.1-26 (Sh. 1)
M-1088 D	Fuel Pool Cooling and Cleanup System	9.1-27
M-1088 E	Fuel Pool Cooling and Cleanup System	9.1-26 (Sh. 2)
M-1089	Filter Demineralizer System (FPCC)	9.1-28
M-1090 A	Leak Detection System	7.6-16
M-1090 B	Leak Detection System	7.6-17
M-1091	Combustible Gas Control System	6.2-81

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1092 A	Offgas System - Low Temperature	11.3-5
M-1092 B	Offgas System - Low Temperature	11.3-6
M-1092 C	Offgas System - Low Temperature	11.3-7
M-1092 D	Offgas System - Low Temperature	11.3-8
M-1092 E	Offgas System - Low Temperature	11.3-10
M-1093 A	HPCS Diesel Generator System	9.5-13
M-1093 B	HPCS Diesel Generator System	9.5-13a
M-1093 C	HPCS Diesel Generator System	9.5-13b
M-1094 A	Floor and Equipment Drains System	9.3-9 (Sh. 1)
M-1094 B	Floor and Equipment Drains System	9.3-10
M-1094 C	Floor and Equipment Drains System	9.3-11
M-1094 E	Floor and Equipment Drains System	9.3-9 (Sh. 2)
M-1095	Offgas Vault Refrigeration System	11.3-9
M-1096	Suppression Pool Makeup System	6.2-82
M-1097	MSIV Leakage Control System	6.7-1
M-1098 A	Embedded and Suspended Floor Drains, Aux. Bldg.	9.3-15
M-1098 B	Embedded and Suspended Floor Drains, Aux. Bldg.	9.3-16
M-1098 C	Embedded and Suspended Floor Drains, Turbine Bldg.	9.3-17
M-1098 D	Embedded and Suspended Floor Drains, Turbine Bldg.	9.3-18
M-1098 E	Embedded and Suspended Floor Drains, Turbine and Ctmt Bldg	9.3-19
M-1098 F	Embedded and Suspended Floor Drains, Turbine and Ctmt Bldg	9.3-20
M-1098 G	Embedded and Suspended Floor Drains, Ctmt Bldg and Drywell	9.3-21
M-1098 H	Embedded and Suspended Floor Drains, Ctmt Bldg and Drywell	9.3-22
M-1099	Suppression Pool Cleanup System	9.3-23
M-1100 A	Containment Cooling System	9.4-11
M-1100 B	Containment Cooling System	9.4-12
M-1101	Drywell Cooling System	9.4-13
M-1102 A	Standby Gas Treatment System	6.5-2
M-1102 B	Standby Gas Treatment System	6.5-3

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1103 A	Auxiliary Bldg. Ventilation System	9.4-10
M-1104 A	Fuel Handling Area Ventilation System	9.4-2
M-1104 B	Fuel Handling Area Ventilation System	9.4-3
M-1105 A	Turbine Building Ventilation System, Unit 1	9.4-6
M-1105 B	Turbine Bldg. Ventilation System	9.4-7
M-1105 C	Turbine Bldg. Ventilation System, Unit 1	9.4-7a
M-1106 A	Diesel Gen Rm, ESF, Electrical SWGR, SSW, and Circ. Water Pumphouse	9.4-9a
M-1106 B	Diesel Gen Rm, ESF, Electrical SWGR, SSW, and Circ. Water Pumphouse	9.4-9b
M-1107 A	Process Radiation Monitoring System	11.5-2
M-1107 B	Process Radiation Monitoring System	11.5-4
M-1107 C	Process Radiation Monitoring System	11.5-5
M-1107 D	Process Radiation Monitoring System	11.5-3
M-1107 E	Process Radiation Monitoring System	11.5-6
M-1107 F	Process Radiation Monitoring System	11.5-7
M-1107 G	Process Radiation Monitoring System	11.5-1
M-1107 H	Process Radiation Monitoring System	11.5-8
M-1108 A	Safeguard Switchgear and Battery Room Ventilation System	9.4-8a
M-1108 B	Safeguard Switchgear and Battery Room Ventilation System	9.4-8b
M-1109 A	Plant Chilled Water System	9.2-17
M-1109 B	Plant Chilled Water System	9.2-18
M-1109 C	Plant Chilled Water System	9.2-19
M-1109 D	Plant Chilled Water System	9.2-20
M-1109 E	Plant Chilled Water System	9.2-21
M-1109 F	Plant Chilled Water System	9.2-21a
M-1110 A	Containment and Drywell Instrument and Control System	7.5-5

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1110 B	Ctmt and Drywell Inst and Control System	7.5-6
M-1111 A	Containment Leak Rate Test System	6.2-76
M-1111 B	Containment Leak Rate Test System	6.2-77
M-1111 C	Containment Leak Rate Test System	6.2-78
M-1111 D	Containment Leak Rate Test System	6.2-79
M-1111 E	Containment Leak Rate Test System	6.2-79a
M-1112	Feedwater Leakage Control System	6.7-5
M-1115 B	Turbine Cycle Heat Balance	10.1-2
M-1118	Seismic Instrumentation System, Unit 1 & Common	3.7-82
M-1119 A	Drywell Chilled Water System	9.2-48
M-1119 B	Drywell Chilled Water System	9.2-49
M-1119 C	Drywell Chilled Water System	9.2-50
M-1126	Plant Air System	9.3-31
M-1126 B	Plant Air System	9.3-32
M-1126 C	Plant Air System	9.3-33
M-1500	Internally Generated Missiles Ctmt. El. 93'-0" & 100'-9" Area 11 - Unit 1	3.5-6
M-1501	Internally Generated Missiles Ctmt. El. 114'-6" & 120'-10" Area 11 - Unit 1	3.5-13
M-1502	Internally Generated Missiles Ctmt. El. 135'-4", 140'-0" & 147'-7" Area 11 - Unit 1	3.5-7
M-1503	Internally Generated Missiles Ctmt. El. 161'-10" & 170'-0" Area 11 - Unit 1	3.5-15
M-1504	Internally Generated Missiles Ctmt. - Unit 1 Section "A-A"	3.5-16
M-1505	Internally Generated Missiles Ctmt. - Unit 1 Section "B-B"	3.5-17
M-1506	Internally Generated Missiles Ctmt. - Unit 1 Misc. Sections & Details	3.5-5

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1507	Internally Generated Missiles Ctmt. - Unit 1 Misc. Sections & Details	3.5-19
M-1508	Internally Generated Missiles Aux. Bldg. El. 93'-0" Area 10 Unit 1	3.5-10
M-1509	Internally Generated Missiles Aux. Bldg. El. 119'-0" Area 8 Unit 1	3.5-12
M-1510	Internally Generated Missiles Aux. Bldg. El. 139'-0" Area 7 Unit 1	3.5-8
M-1511	Internally Generated Missiles Aux. Bldg. El. 139'-0" Area 8 Unit 1	3.5-11
M-1512	Internally Generated Missiles Aux. Bldg. Sections Unit 1	3.5-14
M-1513	Internally Generated Missiles Aux. Bldg. Sections Unit 1	3.5-18
M-1514	Internally Generated Missiles Aux. Bldg. Sections Unit 1	3.5-20
M-1515	Internally Generated Missiles Aux. Bldg. Partial Plans Unit 1	3.5-9
M-1516	Internally Generated Missiles Control Bldg. El. 148'-0" Area 25A Unit 1	3.5-21
M1550 A	High Energy Pipe Break Main Steam "A" & "B" Inside Ctmt. Unit 1	3.6A-1A
M1550 B	High Energy Pipe Break Main Steam "C" & "D" Inside Ctmt. Unit 1	3.6A-1B
M1551	High Energy Pipe Break Main Steam System Outside Ctmt. Unit 1	3.6A-2
M-1552	High Energy Pipe Break Stm. Supply to RCIC & RHR from Main Stm. "A" Inside Ctmt. Unit 1	3.6A-3
M-1553	High Energy Pipe Break Stm. Supply to RCIC & RHR Outside Ctmt. Unit 1	3.6A-4

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1554 A	High Energy Pipe Break Feedwater System "A" Inside Ctmt. Unit 1	3.6A-5A
M-1554 B	High Energy Pipe Break Feedwater System "B" Inside Ctmt. Unit 1	3.6A-5B
M-1555	High Energy Pipe Break FW System Including RHR & RWCU Inj. Piping Outside Ctmt. Unit 1	3.6A-6
M-1556 A	High Energy Pipe Break Reactor Water to DCB-3	3.6A-7(Sh. 1)
M-1556B	High Energy Pipe Break RWCU System Inside Ctmt. Unit 1	3.6A-7(Sh. 2)
M-1556C	High Energy Pipe Break Reactor Water from Reactor to DBA-11	3.6A-7A
M-1557	High Energy Pipe Break RWCU System Outside Ctmt. Unit 1	3.6A-8
M-1558	High Energy Pipe Break RHR Suction Off of Recirc. Loop "B" Inside Ctmt. Unit 1	3.6A-9
M-1559	High Energy Pipe Break HPCS Piping Inside Ctmt. Unit 1	3.6A-10
M-1560	High Energy Pipe Break LPCS Piping Inside Ctmt. Unit 1	3.6A-11
M-1561	High Energy Pipe Break RHR-LPCI Piping & RPV Head Spray Inside Ctmt. Unit 1	3.6A-12
M-1562A	High Energy Pipe Break Main Steam Drains Inside Ctmt. Unit 1	3.6A-13A
M-1562B	High Energy Pipe Break Main Steam Drains Outside Ctmt. Unit 1	3.6A-13B
M-1563	High Energy Pipe Break Reactor Steam Unit 1	3.6A-13C
M-1564	High Energy Pipe Break DRW Vents & Drains Unit 1	3.6A-13D
M-1565A	High Energy Pipe Break Main Steam Line Drain Outside Ctmt. from Isolation Valves Unit 1	3.6A-13E (Sh. 1)

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
M-1565B	High Energy Pipe Break Main Steam Line Drain Outside Ctmt. from Isolation Valves Unit 1	3.6A-13E (Sh. 2)
M-1565C	High Energy Pipe Break Main Steam Line Drains Outside Ctmt. from Isolation Valves Unit 1	3.6A-13E (Sh. 3)
M-1565D	High Energy Pipe Break Main Steam Line Drain Outside Ctmt. from Isolation Valves Unit 1	3.6A-13E (Sh. 4)
M-1565E	High Energy Pipe Break Main Steam Line Drain Outside Ctmt. from Isolation Valves Unit 1	3.6A-13E (Sh. 5)
M-1566A	High Energy Pipe Break Sodium Pentaborate to RPV Unit 1	3.6A-13F (Sh. 1)
M-1566B	High Energy Pipe Break Sodium Pentaborate to RPV Unit 1	3.6A-13F (Sh. 2)
M-1556D	High Energy Pipe Break PWCU System Inside Containment	3.6A-7 (Sh. 3)
M-1556E	High Energy Pipe Break PWCU System Inside Containment	3.6A-7 (Sh. 4)
M-1556F	High Energy Pipe Break PWCU System Inside Containment	3.6A-7 (Sh. 5)
M-1556G	High Energy Pipe Break PWCU System Inside Containment	3.6A-7 (Sh. 6)
M-1568	High Energy Pipe Break Condensate to H.P. Condensers Unit 1	3.6A-13G
M-1569A	High Energy Pipe Break Steam to MSIV Leakage Control System Unit 1	3.6A-13H (Sh. 1)
M-1569B	High Energy Pipe Break Steam to MSIV Leakage Control System Unit 1	3.6A-13H (Sh. 2)
M-1570	High Energy Pipe Break Feedwater Leakage Control System Unit 1	3.6A-13I
M-1571	High Energy Pipe Break Reactor Recirculation Loops A & B Unit 1	3.6B-4a
SFD-0039 A	Liquid Radwaste System	11.2-13
SFD-0039 B	Liquid Radwaste System	11.2-14
SFD-0039 C	Liquid Radwaste System	11.2-15
SFD-0039 D	Liquid Radwaste System	11.2-16

TABLE 1.10-1: CROSS-REFERENCE LIST OF DRAWING NUMBERS AND FSAR FIGURE NUMBERS (Continued)

Drawing No.	System Identification	FSAR Figure No.
SFD-0039 E	Liquid Radwaste System	11.2-17
SFD-0039 F	Liquid Radwaste System	11.2-18
SFD-0040	Solid Radwaste System	11.4-2
SFD-0049	Control Room HVAC System	6.5-1
SFD-1077	Nuclear Boiler System, Unit 1	5.2-9
SFD-1079	Reactor Water Cleanup System	5.4-22
SFD-1081	Control Rod Drive Hydraulic System	4.6-10
SFD-1082	Standby Liquid Control System	9.3-25
SFD-1083 A	Reactor Core Isolation Cooling System, Unit 1	5.4-12
SFD-1083 B	Reactor Core Isolation Cooling System, Unit 1	5.4-13
SFD-1085	Residual Heat Removal System	5.4-18
762E445	High Pressure Core Spray System	6.3-2
SFD-1087	Low Pressure Core Spray System	6.3-5
SFD-1088	Fuel Pool Cooling and Cleanup System	9.1-30
SFD-1089	Filter Demineralizer System (FPCC)	9.1-29
SFD-1092 A	Offgas System - Low Temperature	11.3-5
SFD-1092 C	Offgas System - Low Temperature	11.3-7
SFD-1094 Sh 1	Floor and Equipment Drains System	9.3-28
SFD-1094 Sh 2	Floor and Equipment Drains System	9.3-29
SFD-1097	MSIV Leakage Control System	6.7-2
SFD-1102	Standby Gas Treatment System	6.5-4
SFD-1111	Containment Leak Rate Test System	6.2-80