1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) complies with the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 1) issued by the Nuclear Regulatory Commission in October 1972. For a discussion of the format of this report refer to subsection 1.1.7, Organization of Contents.

1.1.1 LICENSE REQUESTED

The Final Safety Analysis Report (FSAR) was submitted in support of the application of the Alabama Power Company (APC) for the original operating license for a nuclear power plant, designated as Joseph M. Farley Nuclear Plant (FNP) Units 1 and 2, located on a site near Dothan, Alabama. The updated Final Safety Analysis Report is maintained by Southern Nuclear Operating Company, which became a joint licensee and exclusive operator of Plant Farley in 1991.

1.1.2 PLANT UNITS

The original application was for two reactor units, each at a rated power level of 2652 MWt under Section 103 of the Atomic Energy Act of 1954, as amended, and the regulations of the Nuclear Regulatory Commission set forth in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50).

A license amendment request was submitted to the Nuclear Regulatory Commission in February, 1997, to increase the rated power level of each unit. The amendment request was approved by the Nuclear Regulatory Commission on April 29, 1998. An additional license amendment was submitted to the Nuclear Regulatory Commission in October, 2019 to increase rated power level to the current level of 2821 MWt. This increase was approved by the Nuclear Regulatory Commission on October 9, 2020.

The two units are essentially the same, and the descriptions of one unit are interpreted as applying to both units. Differences between the two units, and particularly structures, systems, and components which are shared between the two units, are specifically pointed out.

1.1.3 PLANT LOCATION

This plant site is located in Houston County, about 16.5 miles east of Dothan, Alabama. The nearest public road is State Highway 95, which forms the western boundary of the site property as shown on figure 2.1-3.

1.1.4 CONTAINMENT TYPE

The containment for each of the FNP units (designed by Bechtel Power Corporation) consists of a steel lined prestressed concrete structure.

1.1.5 NUCLEAR STEAM SUPPLY SYSTEM

1.1.5.1 Reactor Type and Supplier

The nuclear steam supply system (NSSS) for each of the two FNP units is a pressurized water reactor (PWR). The Westinghouse Electric Corporation has designed and supplied units for the FNP.

1.1.5.2 Power Output

Each NSSS was originally designed for a warranted power output of 2660 MWt, including 8 MWt of heat from nonreactor sources (primarily pump heat), with a corresponding gross electrical output of approximately 861 MWe. Each NSSS was originally expected to be capable of an output of approximately 2774 MWt, and all steam and power conversion equipment, including the turbine generator, was expected to have the capability of generating a maximum calculated gross output of approximately 898 MWe. Although the original license application was for 2652 MWt, all safety systems, including the containment and engineered safeguards, were designed and evaluated for operation at the higher power level. The power rating of 2774 MWt was originally used in the analysis of all postulated accidents bearing significantly on the acceptability of the site. The thermal hydraulic and nuclear aspects of the core were originally evaluated on the basis of a core thermal output of 2652 MWt.

The rated thermal power level (i.e., core thermal power) of each NSSS was subsequently increased to 2775 MWt. The corresponding NSSS thermal power level was 2785 MWt, including 10 MWt of heat from nonreactor sources (primarily pump heat). Analyses and evaluations at these increased thermal power levels and a conservative electrical power level of 943.4 MWe. The uprated gross electrical output was approximately 933.4 MWe.

The rated thermal power level (i.e., core thermal power) of each NSSS was increased again as part of the measurement uncertainty recapture (MUR) power uprate. The rated thermal power for each unit is currently 2821 MWt, with a corresponding NSSS thermal power level of 2831 MWt which includes 10 MWt of heat from nonreactor sources (primarily pump heat). The uprated gross electrical output is approximately 944.7 MWe (Unit 1) and 953.3 MWe (Unit 2).

1.1.6 SCHEDULE FOR COMPLETION AND COMMERCIAL OPERATION

The two FNP units were completed and began commercial operation as tabulated below:

Unit	Completion of Construction (Fuel Loading)	Commercial Power Operation
1	7-4-77	12-1-77
2	3-8-81	7-30-81

1.1.7 ORGANIZATION OF CONTENTS

1.1.7.1 Subdivisions

This FSAR is organized into 17 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 also referred to as subsections.

1.1.7.2 Standard Format

This FSAR has been written to comply with the Standard Format and Content Safety Analysis Reports for Nuclear Power Plants (Revision 1) as issued by the Nuclear Regulatory Commission in October 1972. This FSAR uses the same chapter, section, subsection, and paragraph headings as those used in the standard format. Where appropriate, the 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, subsection, or paragraph), this information is presented under the appropriate general heading as a subdivision following all subdivisions containing information specifically requested by the standard format. (For example, subsection 1.1.7 is not requested in the standard format. Since it apparently belonged in section 1.1, it was placed after the six subsections containing information requested by the standard format.)

1.1.7.3 References

References to another location in the FSAR are made by chapter section, or subsection number.

1.1.7.4 Tables and Figures

Tabulations of data are designated "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 (e.g., figure 3.4-2 is the second figure of section 3.4). Figures are located at the end of the applicable section. Some plant project drawings are included in the FSAR by reference to the drawing identification number (e.g., D-177024) in lieu of inclusion in the FSAR as a figure.

1.1.7.5 Numbering of Pages

Pages are numbered sequentially within each section. For example, 1.1-4 is the fourth page of section 1.1. When it becomes necessary during revision of this 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 order would appear: 3.2-4, 3.2-4a, 3.2-4b, 3.2-5.)

1.1.7.6 Revising the FSAR

When it becomes necessary to submit additional information or revise information presently contained in the FSAR as required by 10 CFR 50.71(e), the following procedures will be followed:

1. When a change is made to the FSAR text, those pages affected will be marked with the date of change or revision number or both in the lower righthand corner and a vertical line in the righthand margin next to the material affected. Where it is necessary to insert additional pages between existing pages within a section, letters will be used. (To insert a page between pages 3.2-4 and 3.2-5, the insert will be numbered 3.2-4a.)

2 Figures will be revised by indicating the date of change or revision number or both in the lower righthand corner.

Such revisions shall reflect all changes up to a maximum of 6 months prior to the date of filing.

TABLE 1.1-1

FSAR REVISION DATES

FSAR REVISION	DATE
0	7/82
1	7/83
2	7/84
3	7/85
4	7/86
5	7/87
6	7/88
7	7/89
8	7/90
9	7/91
10	6/92
11	6/93
12	10/94
12A	6/95
13	4/96
14	12/97
15	6/99
16	11/00
17	5/02
18	10/03
19	5/05
20	11/06
21	5/08

1.2 GENERAL PLANT DESCRIPTION

1.2.1 SITE CHARACTERISTICS

1.2.1.1 Location

The site is located in southeast Alabama on the west side of the Chattahoochee River, about 6 miles north of the intersection of U. S. Highway 84 and State Highway 95, as shown on figures 2.1-1 and 2.1-2. It is in the northeastern section of Houston County, Alabama, just across the river from Early County, Georgia. The site is about 100 miles southeast of Montgomery, Alabama, and about 180 miles south-southwest of Atlanta, Georgia. The Universal Transverse Mercator Grid Coordinates for the center of the Unit 1 containment are Zone 16-R, central meridian 87 degrees, east 678,872.5 meters, north 3,455,620.1 meters; for the center of the Unit 2 containment they are Zone 16-R, central meridian 87 degrees, east 679,871.6 meters, north 3,455,705.8 meters.

1.2.1.2 Site Ownership

Alabama Power Company (APC) owns the 1850-acre site. Boundaries are shown on figure 2.1-3.

1.2.1.3 Access to the Site

All activities on the site are under the control of Southern Nuclear Operating Company. Access to the plant proper is controlled by a security fence and Security Force Member (SFM).

1.2.1.4 Site Environs

There are no people living on the site. Approximately 45 percent of the land area around the site is wooded, with the remainder being used for various agricultural purposes. The nearest industry is located about 4 miles to the south in Early County, Georgia. The nearest developed community is Columbia, Alabama, 5 miles to the north. The nearest major city is Dothan, Alabama, about 16.5 miles to the west, having an estimated 1980 population of 48,700.

1.2.1.5 Geology

The site is located in the extreme southeastern portion of the East Gulf Coastal Plain physiographic province, which covers about 65 percent of the State of Alabama. This province is underlain by Mesozoic and Cenozoic sedimentary rocks which dip southward at 10 to 40 ft per mile. These deposits consist of marine and nonmarine gravels, sands, silts, clays, marls, and their consolidated equivalents such as sandstone and limestone. In southeastern Alabama and southwestern Georgia, the gentle south-dipping Paleocene through Oligocene sequence is influenced by only minor structural features. These structures have been inactive since

Miocene time, and do not affect materials underlying the site. The structure nearest the site is the east-west trending Gordon anticline, 10 miles south of the site.

No major or active faults were found nor are believed to exist within the 50-mile radius studied. No evidence of surface displacement was observed during the field investigation.

The site is characterized by two topographic features:

1. The gently undulating upland, which ranges from about el 130 to 240 ft mean sea level (msl).

2. The Chattahoochee River valley, which includes the floodplain and the river channel itself, all lying between el 130 and 70 ft msl.

The Lisbon formation (Eocene), which is a soft to moderately hard sedimentary rock and dense sand formation approximately 130 ft thick, is the principal load-bearing material for plant structures. The upper formational contact of the Lisbon is slightly undulating and ranges from el 90 to 103 ft msl.

1.2.1.6 Seismology

The southern Chattahoochee River Valley is located within a broad region of infrequent seismic activity encompassing southern Alabama, southern Georgia, and adjacent Florida. This region is one of the least seismically active regions in the United States and is characterized by a few low-magnitude and low-intensity shocks. Historic records show that earthquakes have never been felt at the site with an intensity greater than Modified Mercalli V and that no earthquake of epicentral intensity greater than Modified Mercalli IV has occurred within 150 miles of the plant site. In addition, the seismicity of the area was evaluated on the basis of the interim revised Environmental Science Survey Administration (ESSA) Seismic Risk Map of the United States (Algermissen, 1969). This map shows that the site is located on the Zone 0 - Zone 1 border. The closest distance from the site to a Zone 2 boundary is 85 miles and to a Zone 3 boundary is 255 miles.

The maximum intensity postulated as having been experienced at the site, as a result of any historical earthquake, is low to moderate V. This intensity corresponds to a surface acceleration of 0.03g on Hershberger's (1956) curve. Therefore, it is considered conservative to select 0.10g surface acceleration as the safe shutdown earthquake (SSE). The SSE is that earthquake which produces the vibratory ground motion for which Category I structures, systems, and components are designed to remain functional. A surface acceleration of 0.05g is referred to as the 1/2 SSE. Category I structures, systems, and components will also be designed to withstand the effects of vibratory motion at the 1/2 SSE in combination with other appropriate loads. Vertical acceleration is taken as two-thirds of the above horizontal accelerations. For detailed discussions, refer to sections 2.5 and 3.7.

1.2.1.7 Hydrology

The dominant hydrological features of the site region are the Chattahoochee River and some small tributary streams. The Chattahoochee River joins the Flint River about 44 river miles downstream of the plant site and forms the Apalachicola River, which empties into the Gulf of Mexico 144 river miles downstream.

The area of the drainage basin affecting the Chattahoochee River at the site is about 8246 square miles. The average flow in the river during 33 years of record at Columbia, Alabama (about 6 river miles upstream) has been 10,600 ft^3/s, and the minimum flow was 1210 ft^3/s. The maximum historical flow, based on 60 years of record, was 207,000 ft^3/s during the flood of 1929. This corresponds to an estimated maximum flood stage at the site of about 124 ft msl.

The probable maximum flood for the Chattahoochee River at the Farley site has been calculated to be 642,000 ft^3/s, which corresponds to an estimated maximum flood stage of 144.2 ft msl. This compares to the plant site grade of 154.5 ft msl.

1.2.1.8 Meteorology

The climate at the site is typical of that in the Southern Gulf Coastal Plain, being hot and humid in the summer and mild in the winter. Maximum rainfall in the 30-year period of record at Blakely, Georgia, 15 miles northeast of the site, was about 11 in. in 24 hours, and the maximum average monthly rainfall was about 7 in. Napier Field, near Dothan, Alabama, about 22 miles west-northwest of the site, is the closest offsite point for which wind direction and velocity data are available. Wind directions are well distributed, with a slight predominance from the south-southwest.

During the period from 1900 to 1966, there were 44 hurricanes or post-hurricane paths which passed within 100 miles of the site. Since the site is approximately 80 miles from the southern coast of Florida, the hurricane wind speeds are, in general, lower than those farther to the south.

During the 50-year period from 1918 to 1967, for which 44 years of record are available, there have been 39 tornadoes within a 25-mile radius of the site. During the period from 1958 to 1967, there were 13 tornadoes reported within this radius. The high frequency in the latter period probably is due to improved reporting. The effects of severe weather were taken into account in the design of plant buildings.

[HISTORICAL] *In March 1971, an onsite meteorological program began operation. Data for the program are gathered from instruments placed at selected positions on the 240-ft microwave tower located north of the plant site. The program included measurements of wind speed and direction at 50 and 150 ft above the tower base and temperature measurements at 35 and 200 ft above the tower base.*

The instruments are installed on a 60-m (197-ft) tower located near the microwave tower at various heights. FSAR subsection 2.3.3 and table 2.3.10 reflect the current environmental monitoring equipment and locations.

1.2.1.9 Radiological Environmental Monitoring Program

The purpose of the Radiological Environmental Monitoring Program (REMP) is to measure radioactive material in the environment which may be released from the plant. The type of environmental measurement to be made is selected so as to evaluate possible significant modes of human exposure. Background radiation reference measurements to assess radiation sources in the environment not attributable to the plant are made concurrently with measurements of radiation in the environment due to plant operation. To accomplish this, two types of sampling stations are established--one type (indicator stations) is placed where maximum doses attributable to the plant are expected, the other (control stations) is placed where radioactive levels are not expected to be significantly influenced by radioactive releases from the plant. Appropriate samples are collected and analyzed to ensure that the significant pathways of radioactive material to man are evaluated.

1.2.2 GENERAL ARRANGEMENT

The site plot plan, which indicates the arrangement and orientation of the plant buildings, storage pond, river water intake, and plant service water intake structures is shown on drawing D-170084.

Each containment houses a nuclear steam supply system (NSSS) consisting of a reactor, steam generators, reactor coolant pumps, pressurizer, and some of the reactor auxiliaries.

The auxiliary buildings house the waste treatment facilities, engineered safeguards system components, heating and ventilation system components, switchgear, laboratories, offices, laundry, and the control room. The spent-fuel pool and the new fuel storage facilities are located in the auxiliary building of the respective units. These facilities are under controlled ventilation whenever spent fuel is being moved or stored in that section. The transfer of fuel to the spent-fuel pool of each unit from its associated containment is through a fuel transfer tube.

The emergency diesel generators are housed in the diesel generator building, which is located south of the Unit 1 auxiliary building. The turbine building houses the turbine generator, condensers, feedwater heater, condensate and feedwater pumps, turbine auxiliaries, and certain nonsafety-related switchgear for both units.

The solidification and dewatering facility located east of the auxiliary buildings is provided to solidify or dewater spent radioactive resins and to solidify chemical drains and evaporator concentrates. The facility will also be used for decontamination activities.

The service building on the south end of the turbine building provides office, shop, and warehouse space for the plant. The computer/office building west of the service building provides office, shop, and computer facilities. The outage support building, located northeast of the Unit 2 containment building, provides office and storage space, and a craft break area.

The following structures, systems, and components are shared by the two units.

Structures
Control room
Access control area
Diesel generator building
Auxiliary building HVAC penthouse
Switchyards
Hot machine shop and decontamination room
Hot instrument shop
Ultimate heat sink storage pond, river intake structure, and service water intake structure
Water treatment plant
Sewage treatment plant
Liquid Radwaste Processing Facility (LRWPF)
Technical support center
Training center
Respirator storage
Solidification and dewatering facility
Low-level radwaste storage buildings
Secondary chemistry laboratory
Outage support building
Old steam generator storage facility
Security Diesel Generator (SDG) building

Systems
Control room HVAC system
Diesel generator fuel storage/transfer system
Diesel generator HVAC systems for shared diesel generators and switchgear rooms
Carbon dioxide supply system
Nitrogen supply system
Hydrogen supply system
Oxygen supply system
Demineralized water system
River water system to storage pond
Potable and sanitary water system
Well water system
Auxiliary steam supply system
Reactor makeup water system
Service water intake structure compressed air system and backup nitrogen system
Service water intake structure HVAC
Cathodic protection system
Communications system
Liquid Radwaste Processing Facility (LRWPF)

Components
One pressurizer heater station service transformer

Three of five diesel generators and auxiliary support equipment
Diesel generator 600-volt switchgear and MCCs
Service water intake structure 600-volt switchgear, batteries and battery chargers
Fire protection system water supply
Fire pumps
Spent-fuel cask crane
Load centers H, J, N, K, L, R, and S
Service water discharge pipes to UHS pond and common discharge pipe to river
Spent-resin storage tanks

Unit 1 and Unit 2 plant layout drawings are shown in figures 1.2-1 through 1.2-9. The Unit 2 plant layout is generally a mirror image of the Unit 1 layout except for the shared spaces and equipment listed above. Figure 12.1-1 shows a layout of the combined control room for Units 1 and 2.

The criterion that is followed in the design of Units 1 and 2 is that each unit operates independently of the other.

1.2.3 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system (NSSS) for each unit consists of a pressurized water reactor and a three-loop reactor coolant system. The mechanical, thermal hydraulic, and nuclear design of the reactor core is similar to the design of other Westinghouse units under construction. The maximum expected thermal output of each unit is 2774 MWt. A heat balance, showing the major parameters of the plant for the rated power condition, is shown in figure 10.1-1 and for the maximum calculated power condition is shown in figure 10.1-2.

1.2.3.1 Reactor Core

The reactor core is a three-region core. The fuel rods are pressurized, cold worked zircaloy tubes containing slightly enriched uranium dioxide fuel. For the initial core, fuel assemblies with the highest enrichments are placed in the core periphery, or outer region, and the two groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In reload cores, approximately 72 assemblies are discharged and the fresh fuel is loaded in the center of the core with once burned fuel predominately on the periphery.

The fuel assembly is a canless type with the basic assembly consisting of the control rod guide thimbles attached to the grids and the top and bottom nozzles. The fuel rods are held by the spring clip grids in this assembly which provide a very stiff support for the fuel rods.

Full-length control rod assemblies and burnable absorber rods are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-indium-cadmium sealed in stainless steel tubes. The absorber material in the fixed burnable absorber rods is either in the form of borosilicate glass sealed in stainless steel tubes or in the form of B_4C in an alumina matrix sealed in zircaloy tubes.

Above the core, each cluster of control rods is attached to a spider connector and drive shaft, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. The control rod drive mechanisms for the full-length control rod assemblies are of the magnetic-latch type. The latches are controlled by three magnetic coils. They are so designed that, upon loss of power to the coils, the rod cluster control assembly is released and falls by gravity to shut down the reactor.

1.2.3.2 Reactor Coolant System

The reactor coolant system consists of three similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control and protection.

The reactor coolant system transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Borated demineralized light water is circulated at a flow rate and temperature consistent with achieving the desired reactor core thermal hydraulic performance. The water also acts as a neutron moderator, reflector, and solvent for the neutron absorber.

The reactor coolant pumps are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type. The power supply systems for the pumps are designed so that adequate coolant flow is maintained to cool the reactor core under all required conditions.

The steam generators are Westinghouse vertical U-tube units which contain inconel tubes. Integral moisture separators reduce the moisture content of the steam to 0.10% or less.

The reactor coolant piping and all of the pressure containing and heat transfer surfaces in contact with the reactor coolant are stainless steel, stainless steel clad. The steam generator tubes and fuel tubes are inconel and zircaloy, respectively. The reactor core internals, including the control rod drive shafts, are stainless steel.

An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions.

1.2.4 STEAM AND POWER CONVERSION SYSTEM

The turbine generator is furnished by the Westinghouse Electric Corporation (modified by Siemens) and is an 1800-rpm tandem-compound, four-flow exhaust, indoor unit designed for saturated steam conditions. The unit utilizes two parallel strings of six feedwater heaters each. The main steam generator feedwater pumps are driven by steam turbines supplied normally with reheated steam and exhausting to the condenser.

The auxiliary feedwater system is designed to provide emergency heat removal, and two motor-driven pumps and one turbine-driven pump are provided.

1.2.5 CONTAINMENT

The containment uses a prestressed concrete design and is a vertical right cylindrical structure with a dome and flat base. The containment interior is lined with carbon steel plate for leaktightness.

Inside the containment, the reactor and other NSSS components are shielded with concrete. A vent stack is attached to the outside of the containment and extends to an elevation 10 ft above the top of the containment dome. Access to portions of the containment during power operation is permissible.

The containment, in conjunction with engineered safety features, is designed to withstand the internal pressure and coincident temperature resulting from the energy release of the loss-of-coolant accident (LOCA) associated with 2774 MWt. The containment design conditions are for an internal pressure of 54 psig and coincident temperature of 280°F.

1.2.6 SAFETY FEATURES

The engineered safety features system, especially the containment and cooling water systems, provides protection for the public and plant personnel against the accidental release of radioactivity from the reactor system, particularly as the result of a LOCA. These safety features function to localize, control, mitigate, and terminate such accidents so as to maintain the exposure to the public within the requirements of 10 CFR 50.67.

The engineered safety features include, but are not limited to, the following systems:

- The emergency core cooling system (ECCS).
- The containment spray system.
- The containment cooling system.
- The penetration room filtration system.

Subsection 6.1.1 provides a detailed list of engineering safety features systems.

The ECCS injects borated water into the reactor coolant system following a LOCA. This provides cooling to limit core damage and fission product release and assures an adequate shutdown margin regardless of temperature. The ECCS also provides continuous long-term post-accident cooling of the core by recirculating borated water between the containment sump and the reactor core.

The containment for each unit is equipped, as follows, with two independent, full-capacity systems for cooling the containment atmosphere after the postulated LOCA:

A. The containment spray system supplies borated water to cool the containment atmosphere. The spray system, in combination with at least one of the containment air coolers, is sized to provide adequate cooling with one of the two containment spray pumps in service on emergency power. These pumps take suction from the refueling water storage tank. When this supply is depleted, suction of these pumps is aligned to pump water from the containment sump directly into the containment during the recirculation mode of operation. Trisodium phosphate is added to the sump to enhance iodine removal from the containment atmosphere in the postaccident condition.

B. The containment cooling system is designed to provide containment atmosphere mixing and cooling. The system design basis is to provide adequate containment cooling from the operation of one train of containment spray and one containment cooler.

The penetration room filtration system for each unit collects and processes potential ECCS recirculation leakage to limit environmental activity levels following a LOCA. This system is also used to mitigate the consequences of a fuel handling accident in the spent-fuel pool.

1.2.7 UNIT CONTROL

The reactor is controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. During steady-state operation, the reactor control system maintains a programmed average reactor coolant temperature that rises in proportion to the load.

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

1.2.8 PLANT ELECTRICAL POWER

The electrical systems are designed to provide reliable power sources for electrical equipment for startup, normal operation, safe shutdown, and emergency situations in both Units 1 and 2.

The Unit 1 output is fed through a step-up transformer to the 230-kV switchyard, and Unit 2 output is fed through a step-up transformer to the 500-kV switchyard. The 230-kV switchyard of Unit 1 is connected to the high voltage transmission system through three 230-kV transmission lines which approach the site from different directions. The 500-kV switchyard is connected to the high-voltage transmission system by two 500-kV lines, and the 230-kV and 500-kV substations are connected through two auto transformers. A switchable 230-kV shunt reactor is provided to help control switchyard voltages and improve stability during transmission system light load conditions. In addition to carrying the electrical output of the plant, the high voltage transmission system provides a means for power to be supplied to the plant from external sources. Startup power for normal loads and auxiliary power for emergency loads is taken from the 230-kV switchyard through startup auxiliary transformers. Since the emergency buses are normally connected to and fed from offsite sources through the startup auxiliary transformer, in

case of emergency, power transfer to offsite sources will not be necessary. Normal plant auxiliary power is taken from the generator main leads through the unit auxiliary transformer(s) or may be taken directly from the startup auxiliary transformers. Offsite power for the engineered safeguards for each unit is provided from a startup auxiliary transformer. Onsite emergency power is provided by four diesel generators. Each unit has one diesel generator dedicated to it specifically and two of the diesel generators are shared between the two units. An additional diesel generator is dedicated as the alternate power source during station blackout (SBO) events. Auxiliary power for each unit is utilized at 4160, 600, 480, and 208/120 volts. Direct current systems are also provided for emergency power, engineered safety features control, essential nuclear instrumentation, and switchyard control and relaying.

The normal and emergency sources of electrical power are each adequate to permit prompt shutdown and maintain safe conditions under all credible circumstances. The capacity of the power sources is sufficient for the required safety function under postulated abnormal conditions.

1.2.9 PLANT INSTRUMENTATION AND CONTROL SYSTEM

Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain nuclear power, primary coolant pressure, temperature, and control rod positions within prescribed operating ranges.

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

The reactor control system provides for startup and shutdown of the reactor and for adjustment of the reactor power in response to turbine load demand. The reactor is controlled by control rod cluster motion, which is required for load follow transients and for startup and shutdown; and by a soluble neutron absorber (boron in the form of boric acid) which is inserted during cold shutdown, partially removed at startup, and adjusted in concentration during core lifetime to compensate for such effects as fuel consumption and accumulation of fission products which tend to slow the nuclear chain reaction. The control system permits the unit to accept step load increases or reductions of 10 percent and ramp load increases of 5 percent per minute over the load range from 5 percent to, but not exceeding, 100-percent power under normal operating conditions, subject to xenon limitations.

Control of both the reactor and turbine generator is accomplished from the control room by NRC licensed personnel.

1.2.10 AUXILIARY SYSTEMS

Nuclear auxiliary systems are provided to perform the following functions:

A. Supply reactor coolant system requirements.

B. Purify reactor coolant.

C. Introduce chemicals for corrosion inhibition.

D. Introduce and remove chemicals for reactivity control.

E. Cool system components.

F. Remove residual heat during a portion of the reactor cooldown period and during shutdown of the reactor.

G. Cool the spent-fuel pool water.

H. Permit sampling of reactor coolant water.

I. Provide for safety.

J. Vent and drain the reactor coolant system and the auxiliary systems.

K. Provide containment ventilation and cooling.

L. Dispose of liquid and gaseous wastes and provide for disposal of solid wastes.

These functions are performed by the following systems.

1.2.10.1 Chemical and Volume Control System

The purity level in the reactor coolant system is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the reactor coolant system is cooled in the regenerative heat exchanger. From there, the coolant flows to a letdown heat exchanger and then through a demineralizer where corrosion and fission products are removed. It then passes through a filter and is sprayed into the volume control tank.

The chemical and volume control system automatically adjusts the amount of reactor coolant to compensate for changes in specific volume, due to coolant temperature changes and reactor coolant pump shaft seal leakage, in order to maintain a constant level in the pressurizer.

1.2.10.2 Residual Heat Removal System

The residual heat removal system is used to reduce the temperature of the reactor coolant at a controlled rate from 350°F to a refueling temperature of approximately 140°F and to maintain the proper reactor coolant temperature during refueling.

The residual heat removal pumps are used to circulate the reactor coolant through two residual heat removal heat exchangers, returning it to the reactor coolant system through the low pressure injection header.

1.2.10.3 Component Cooling System

The component cooling system provides an intermediate barrier between the reactor coolant system and the service water system.

The component cooling system consists of three pumps and three heat exchangers to remove heat from the various auxiliary systems handling the reactor coolant. Corrosion inhibited demineralized water is circulated by the system through the letdown heat exchanger, the reactor coolant pump coolers, sample heat exchangers, spent-fuel pool heat exchangers, recycle and waste evaporators, waste gas compressors, etc. Component cooling water provides cooling for the residual heat exchangers.

1.2.10.4 Fuel Handling and Storage System

The reactor is refueled by the use of equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Transfer of spent fuel under water permits the economic use of an optically transparent radiation shield as well as providing a reliable source of coolant for removal of residual heat.

The fuel handling system provides for the safe handling of rod cluster control assemblies and for the required assembly and disassembly of reactor internals. The system is divided into two pool regions: the refueling cavity, which is flooded for refueling, and the spent-fuel pool, which is external to the reactor containment and is always accessible to plant personnel. The two pools are connected by the fuel transfer system, which transports the fuel from the refueling cavity to the transfer canal by the use of:

A. Manipulator crane, which removes spent fuel from the reactor, located inside the containment above the refueling pool.

B. Fuel transfer carriage.

C. Upending devices.

D. Fuel transfer tube.

E. Fuel handling crane in the spent-fuel pool area.

F. Various devices used for handling the reactor vessel head and internals.

New fuel is stored dry in vertical racks in a storage area in the auxiliary building. Space is provided for over one-third of a core and fuel assembly, and spacing is such as to preclude criticality.

Each unit has a stainless steel lined, reinforced concrete spent fuel pool containing borated water which provides storage for approximately 9 cores. Spent-fuel assemblies are stored in

vertical racks so spaced as to preclude criticality with administrative controls on placement and no credit taken for the borated pool water. Additional credit is taken for the presence of soluble boron in the pool water to maintain $K_{eff} \leq 0.95$.
A shared independent spent-fuel storage installation (ISFSI) is provided for temporary storage of spent fuel pending offsite transportation. The ISFSI is located inside the protected area south of the diesel generator building and consists of five storage pads, each designed to store 12 spent-fuel casks. Following a suitable decay period, the spent fuel will be transported offsite for disposal. The ISFSI is operated in accordance with the general license provisions of 10 CFR Part 72 as described in the Joseph M. Farley 10 CFR 72.212 Report.

Purification and redundant cooling equipment is provided for the spent-fuel pool water. This equipment may also be used for cleanup of refueling water after each fuel change in the reactor.

1.2.10.5 Sampling Systems

Two sampling systems are provided--one for the reactor coolant and its auxiliary systems and one for the turbine steam and feedwater system. These systems are used for determining both chemical and radiochemical conditions of the various fluids used in the plant.

1.2.10.6 Cooling Water Systems

The turbine generator condenser is cooled by the circulating water system, which is a closed system, rejecting heat to cooling towers.

The service water requirements for the nuclear components, turbine building equipment, and diesel generators are supplied by vertical centrifugal pumps taking suction from the storage pond. Water makeup to the storage pond is supplied by vertical centrifugal pumps taking suction from the river.

1.2.10.7 Plant Ventilation Systems

Separate ventilation systems are provided for the containment, auxiliary building fuel handling facility, control room and TSC, turbine building, and emergency diesel generator building. In addition, a purge system and containment preaccess filtration system are provided for the containment atmosphere.

The auxiliary building penetration rooms are ventilated by the penetration room filtration system, which includes filters for control of any leakage through the containment penetrations during accident conditions.

1.2.10.8 Plant Fire Protection System

The major fire protection system contains both diesel and electrical powered fire pumps which supply the various hydrants, hose stations, sprinklers, and deluge systems. Supplementary to

these facilities, chemical fire extinguishing equipment is provided to accommodate special requirements for various classes of hazards.

Consideration is given to the use of noncombustible and fire resistant materials throughout the facility, particularly in areas containing critical portions of the plant such as the containment, control room, and components of the engineered safeguards system.

Hydrants and hose stations are manually operated, while the sprinkler and deluge systems can be both automatic and manually actuated systems. A sufficient number of portable extinguishers are placed at key locations for use in extinguishing fires. A complete description of the fire protection system is presented in appendix 9B.

1.2.10.9 Compressed Air Systems

Nonlubricated air compressors, in combination with aftercoolers, discharge compressed air to air receivers which supply compressed air to a common header. This header furnishes compressed air for the service air system and instrument air system. Instrument air is dried and filtered downstream of this common supply header.

The plant air system provides compressed air for normal maintenance service at various stations throughout the plant, while the instrument air system provides compressed air for the operation of all air operated instruments and valves.

1.2.11 WASTE DISPOSAL SYSTEM

The waste disposal system provides controlled handling and disposal of liquid, gaseous, and solid wastes. The waste processing system provides all equipment necessary for controlled treatment and preparation for retention or disposal of all liquid, gaseous, and solid wastes produced as a result of reactor operation.

The liquid waste processing system collects, processes, and recycles reactor grade water, removes or concentrates radioactive constituents, and processes them until suitable for release or shipment off site. Liquid wastes are sampled and activity levels verified and recorded prior to release. Processed liquid effluent from the reactor coolant system will have been subjected to the chemical and volume control system purification ion exchanger and the components of the waste processing system and will be within the limits established by technical specifications.

The gaseous waste processing system functions to remove fission product gases from the reactor coolant and to contain these gases during normal plant operation. The system also collects the gases generated from the boron recycle evaporator. Waste gases are collected in the vent header. These gases are withdrawn from the vent header by one of two compressors and discharged to a waste gas decay tank. The tank contents will be released to the environment in accordance with technical specifications. Three waste gas decay tanks are provided; each tank has a 45-day storage capacity. Gaseous wastes are discharged through an absolute particle filter to the vent stack.

Solid wastes, when required, are stored in suitable containers for offsite disposal.

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Figure Withheld Under 10 CFR 2.390

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNITS 1 AND 2 PLANT GENERAL ARRANGEMENT PLAN AT EL. 155 FT AND 189 FT

FIGURE 1.2-1

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNITS 1 AND 2 PLANT GENERAL ARRANGEMENT PLAN AT EL. 139 FT

FIGURE 1.2-2

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNITS 1 AND 2 PLANT GENERAL ARRANGEMENT PLAN AT EL 121 FT AND 129 FT

FIGURE 1.2-3

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NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNITS 1 AND 2 PLANT GENERAL ARRANGEMENT PLAN AT EL 105 FT 6 IN. AND BELOW

FIGURE 1.2-4

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UNIT 1 AND UNIT 2

UNITS 1 PLANT GENERAL ARRANGEMENT SECTION “A-A”

FIGURE 1.2-5

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UNIT 1 AND UNIT 2

UNIT 1 PLANT GENERAL ARRANGEMENT SECTION “B-B”

FIGURE 1.2-6

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UNIT 1 AND UNIT 2

PLANT GENERAL ARRANGEMENT SECTION “C-C”

FIGURE 1.2-7

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SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PLANT GENERAL ARRANGEMENT SECTION “D-D” FIGURE 1.2-8

Security Related Information
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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNITS 1 AND 2 PLANT GENERAL ARRANGEMENT
ROOF PLAN AT EL 175 FT

FIGURE 1.2-9

1.3 COMPARISON TABLES

1.3.1 *[HISTORICAL] [COMPARISONS WITH SIMILAR FACILITY DESIGNS*

Table 1.3-1 presents a design comparison of the nuclear steam supply system design for the Joseph M. Farley plant with the North Anna Power Station and the Surry Power Station.

Table 1.3-2 presents a design comparison of the containment system parameters and engineering safety features design for the Farley plant with the Calvert Cliffs Power Station, the Oconee Power Station, the Palisades Power Station, and the Turkey Point Power Station.]

1.3.2 COMPARISON OF FINAL AND PRELIMINARY DESIGNS

All of the significant changes that have been made in the facility design since submittal of the Preliminary Safety Analysis Report (PSAR) are listed in table 1.3-3. Each item in table 1.3-3 is cross referenced to the appropriate section in the FSAR which describes the changes and the bases for them.

Significant changes in NSSS design since the PSAR are listed, with reasons for change, in table 1.3-4.

[HISTORICAL] [TABLE 1.3-1 (SHEET 1 OF 3)

DESIGN COMPARISON

Nuclear Steam Supply System - Comparison with North Anna Power Station (Units 1 and 2) and Surry Power Station (Units 1 and 2)

Chapter Number	Chapter Title System/Component	References (FSAR)	Significant Similarities	Significant Differences
4.0	Reactor			
	Fuel	Subsection 4.2.1	Similar to North Anna and Surry	Differences exist in design parameters based on nuclear design and thermal-hydraulic design parameters (Surry has 15-x-15 fuel assembly)
	Reactor vessel internals	Subsection 4.2.2	Similar to North Anna and Surry except as noted	Surry has a diffuser plate. North Anna and Surry have a thermal shield (Farley has neutron pads)
	Reactivity control systems	Subsection 4.2.3	Similar to North Anna and Surry except as noted	Surry has 12 glass rods in a burnable poison assembly
	Nuclear design	Section 4.3	Similar to North Anna and Surry except as noted	Differences exist in fuel burnup rates, fuel enrichments, k_{eff}, and core kinetic characteristics
	Thermal-hydraulic design	Section 4.4	Similar to North Anna and Surry except as noted	Surry has a core thermal output of 2441 MW. Differences exist in thermal and hydraulic and heat transfer parameters
5.0	Reactor Coolant System	Section 5.1, 5.2	Similar to North Anna and Surry except as noted	Differences exist because North Anna and Surry employ loop stop valves
	Reactor vessel*	Section 5.4	Similar to North Anna and Surry	Farley's steam generator is similar to the initial configuration of the Surry steam generator. Surry steam generators have since been modified
	Reactor coolant pumps*	Subsection 5.5.1	Similar to North Anna and Surry	None
	Steam generators*	Subsection 5.5.2	Similar to North Anna and Surry	None
	Piping	Subsection 5.5.3	Similar to North Anna and Surry	None

TABLE 1.3-1 (SHEET 2 OF 3)

Chapter Number	Chapter Title System/Component	References (FSAR)	Significant Similarities	Significant Differences
	Residual heat removal system (RHRS)	Subsection 5.5.7	Functionally similar to North Anna and Surry	The RHRS has no safety function for North Anna or Surry
	Pressurizer*	Subsection 5.5.10	Similar to North Anna and Surry	None
	Loop stop valves		None	- -
6.0	Engineered Safety Features			
	Emergency core cooling system	Section 6.3	Similar to North Anna and Surry except as noted	Farley uses the RHR heat exchangers for long term cooling. North Anna and Surry each have a separate low head recirculation system
7.0	Instrumentation and Controls			
	Reactor trip system	Section 7.2	System functions are similar to North Anna and Surry	None
	Engineered safety features systems	Section 7.3	System functions are similar to North Anna and Surry	None
	Systems required for safety shutdown	Section 7.4	System functions are similar to North Anna and Surry	None
	Safety-related display instrumentation	Section 7.5	Parametric display is similar to that of North Anna and Surry	Actual physical configuration may differ due to customer design philosophy
	Other safety systems	Section 7.6	Operational functions are similar to North Anna and Surry	Some valve interlock designs differ slightly on Surry
	Control systems	Section 7.7	System functions are similar to North Anna and Surry	None
9.0	Auxiliary Systems			
	Chemical and volume control system	Subsection 9.3.4	Similar to North Anna and Surry except as noted	Farley uses a 4 wt/% boric acid solution for make up. North Anna and Surry each have a separate boron recovery system. Farley has a boron recovery subsystem

TABLE 1.3-1 (SHEET 3 OF 3)

Chapter Number	Chapter Title System/Component	References (FSAR)	Significant Similarities	Significant Differences
11.0	*Radioactive Waste Management*			
	Source terms	*Section 11.1*	*Similar to North Anna and Surry except as noted*	*Differences are based on plant operational influences*
	Liquid waste processing	*Section 11.2*	*Functionally similar to North Anna and Surry*	*None*
	Gaseous waste processing	*Section 11.3*	*Functionally similar to North Anna and Surry except as noted*	*North Anna and Surry have provisions for periodic release of the gaseous wastes*
	Process radiation monitoring	*Section 11.4*	*Functionally similar to North Anna and Surry*	*Differences which exist are due to differences in customer requirements*
14.0	*Initial Tests and Inspection*	*Chapter 14*	*Similar to North Anna and Surry*	*None*
15.0	*Accident Analysis*	*Chapter 15*	*Similar to North Anna and Surry*	*Additional analysis presented for North Anna and Farley*

**Designed and manufactured to code in effect]*

[HISTORICAL][TABLE 1.3-2 (SHEET 1 OF 6)

COMPARISON OF PLANT CHARACTERISTICS

Containment System Parameters	*Farley Units 1 and 2 FSAR*	*Calvert Cliffs Units 1 and 2 FSAR*	*Oconee Units 1, 2 and 3 FSAR*	*Palisades Unit 1 FSAR*
Type	*Steel-lined, prestressed posttensioned concrete cylinder, curved dome roof*	*Steel-lined, prestressed posttensioned concrete cylinder, curved dome roof*	*Steel-lined, prestressed posttensioned concrete cylinder, curved dome roof*	*Steel-lined prestressed posttensioned concrete cylinder, curved dome roof*
Design parameters				
Inside diameter (ft)	*130*	*130*	*116*	*116*
Inside height (ft)	*183*	*182*	*208*	*190*
Free volume (ft)	*2,024,900*	*2,000,000*	*1,900,000*	*1,600,000*
Design pressure (psig)	*54*	*50*	*59*	*55*
Concrete thickness (ft)				
Vertical wall	*3-3/4*	*3-3/4*	*3-3/4*	*3*
Dome	*3-1/4*	*3-1/4*	*3-1/4*	*2-1/2*
Containment leak prevention and mitigation systems	*Leaktight penetration and continuous steel liner. Automatic isola-tion where required. The exhaust from pene-tration rooms to vent.*	*Leaktight penetration and continuous steel liner. Automatic isola-tion where required. The exhaust from pene-tration rooms to vent.*	*Leaktight penetration and continuous steel liner. Automatic isola-tion where required. The exhaust from pene-tration rooms to vent.*	*Leaktight penetration and continuous steel liner. Automatic isola-tion where required. The exhaust from pene-tration rooms to vent.*
Gaseous effluent purge	*Discharge through stack*	*Discharge through stack*	*Discharge through stack*	*Discharge through stack*
ENGINEERED SAFETY FEATURES				
Safety injection system				
No. of high head (pumps)	*3*	*3*	*3*	*3*
No. of low head (pumps)	*2*	*2*	*3*	*2*
Containment fan coolers				
No. of units	*4*	*4*	*3*	*4*
Air flow capacity, each at emergency conditions (ft /min)	*40,000*	*40,000*	*58,000*	*60,000]*

TABLE 1.3-2 (SHEET 2 OF 6)

Containment System Parameters	*Farley Units 1 and 2 FSAR*	*Calvert Cliffs Units 1 and 2 FSAR*	*Oconee Units 1, 2 and 3 FSAR*	*Palisades Unit 1 FSAR*
Postaccident filters				
No. of units	*None*	*3*	*None*	*None*
(ft /min)	*None*	*20,000*	*None*	*None*
Containment spray				
No. of pumps	*2*	*2*	*2*	*3*
Emergency power				
Diesel-generator units	*5 total for both units*	*3 total for both units*	*Hydro*	*1*
Safety injection tanks, number	*3*	*4*	*2*	*4*

TABLE 1.3-2 (SHEET 3 OF 6)

Turkey Point Units 3 and 4 FSAR	*Significant Similarities*	*Significant Differences*	*References By Sections*
Steel-lined, prestressed posttensioned concrete cylinder, curved dome roof	*Containment types are the same for all units*		
116 *169* *1,550,000* *59* *3-1/2* *3*	*Design parameters are the same for Farley and Calvert Cliffs*	*Containment design parameters for Farley and Calvert Cliffs, and the rest of the listed units, differ because of differences in dome height*	*6.2*
Leaktight penetration and continuous steel liner. Automatic isolation where required. The exhaust from penetration rooms to vent.	*Containment Leak Prevention and Mitigation Systems are of the same design bases for Farley, Calvert Cliffs, and Oconee*	*Containment leak prevention and mitigation systems for Palisades and Turkey Point differ from those of Farley, Calvert Cliffs, and Oconee. The former type has no exhaust from penetration rooms to vent*	*6.2.4* *6.2.2*
Through particulate filter and monitors part of main exhaust systems.	*All units discharge through the stack except Turkey Point*		
2 *2*	*The Safety Injection System is the same basic design for all the listed plants except Oconee*	*Oconee has 3 low head injection pumps*	*6.2.2*
3 *25,000*	*The number of containment fan coolers is the same for Farley, Calvert Cliffs, and Palisades. They have 4 each. Oconee and Turkey Point have 3 each*	*Calvert Cliffs and Palisades have emergency-condition flowrates of 60,000 ft /min. Oconee, Farley, and Turkey Point have emergency flowrates of 58,000, 40,000, and 25,000 ft /min, respectively*	*6.2.2.2*
3 *37,500*		*Calvert Cliffs and Turkey Point both have postaccident filters. The other units do not*	
2	*Farley, Calvert Cliffs, Oconee, and Turkey Point have 2 containment spray pumps each*	*Palisades has 3 containment spray pumps. The others have 2 each*	*6.2.3.1.1*

TABLE 1.3-2 (SHEET 4 OF 6)

Turkey Point Units 3 and 4 FSAR	*Significant Similarities*	*Significant Differences*	*References by Sections*
2 total for both units		*Oconee derives its emergency power from hydropower. Palisades utilizes 1 diesel generator; Turkey Point has 2 diesel generators for both plants; and Farley has a total of 5 diesels for both its plants.*	*8.3.1.1*
3		*Oconee utilizes 2 safety injection tanks, while Farley has 3, Calvert Cliffs has 4, Palisades has 4, and Turkey Point has 3*	*6.2*

TABLE 1.3-2 (SHEET 5 OF 6)

Containment System Parameters Electrical Components	*Farley Units 1 and 2 FSAR*	*Calvert Cliffs Units 1 and 2 FSAR*	*Oconee Units 1, 2 and 3 FSAR*
Standby power system	*Total of 5 diesels of which 3 are shared between Units 1 and 2. Diesels are connected to 4160-V buses*	*Three diesels connected to 4-kV buses and shared between Units 1 and 2*	*Two hydro units. 230-kV network and startup transformers*
Engineered safety feature buses	*Six 4160-V buses/unit divided into two separate and redundant systems.*	*Two 4-kV buses/unit divided into separate and redundant systems*	*Three 4160-V redundant buses per unit*
dc system	*Separate and redundant 125-V dc systems for ESF loads. Separate dc systems for loads in auxiliary building, turbine building, cooling tower area, diesel generator building, and switchyard*	*Four batteries between 2 units divided to give two separate and redundant 125-V dc systems. Separate dc systems are provided for turbine building and the switchyard*	*Separate and redundant 125-V dc systems for instrumentation and control power system. Separate dc systems for large loads, switching station control, and control of Keowee hydro station*
Vital instrumentation systems	*System comprised of 4 inverters arranged to give 4 separate and redundant channels*	*Four inverters provided between two units to give 4 separate and redundant channels per unit*	*Four inverters arranged to give 4 separate and redundant buses*
Offsite power system	*Unit 1 – 230-kV switchyard. Unit 2 – 500-kV switchyard. Each unit is comprised of two startup transformers and two unit auxiliary transformers with the ESF buses supplied from startup transformers*	*500-kV switchyard. Two startup transformers shared between two units*	*Units 1 and 2 connected to the 230-kV switchyard and Unit 3 to 500-kV switchyard. Each unit is provided with one unit auxiliary and one startup transformer*

TABLE 1.3-2 (SHEET 6 OF 6)

Palisades Unit 1 FSAR	Turkey Point Units 3 and 4 FSAR	Significant Similarities	Significant Differences	Reference by Sections
Total of 2 diesels. Diesels are connected to 2400-V buses	*Two diesels connected to 4160-V buses and shared between the two units*	*Farley is similar to Calvert Cliffs, Palisades, and Turkey Point, except as noted*	*Oconee has hydro units. Farley has one diesel permanently aligned to an ESF bus per unit, as against that for Calvert Cliffs and Turkey Point, which have shared diesels only*	*8.3.1*
Two 2400-V separate and redundant buses	*Two 4160-V buses/unit divided into separate and redundant systems*	*Farley is similar to all plants it is being compared with*	*None*	*8.3.1*
Separate and redundant 125-V dc systems supplying ESF loads and non-ESF loads	*Separate, redundant 125-V dc system supplying ESF loads and non-ESF loads. Separate dc system for switchyard*	*Farley is similar to Calvert Cliffs and Oconee*	*A separate dc system is provided for non-ESF loads for Farley, in contrast to that for Palisades and Turkey Point*	*8.3.2*
Four inverters arranged to give 4 separate and redundant channels	*Four inverters arranged to give 4 separate redundant channels*	*Farley is similar to all plants being considered*	*None*	*8.3.1*
Switchyard - 345 kV. Two unit auxiliary and two startup transformers provided. ESF buses normally supplied from unit auxiliary transformers	*Switchyard - 240 kV. Each unit is provided with one unit auxiliary and one startup transformer*	*Farley is similar to Palisades*	*Farley has two unit auxiliary and two startup transformers compared to one of each for Oconee and Turkey Point. The startup transformers for Farley are not shared between the two units, as is the case for Calvert Cliffs*	*8.2]*

TABLE 1.3-3 (SHEET 1 OF 3)

SIGNIFICANT DESIGN CHANGES

	Item	Change Discussed in FSAR Subsection	Reason for Change
1.	Steam jet air ejector filtration	9.4.4	Charcoal filters were added in the steam jet air ejector discharge. Change initiated to reduce normal plant releases and/or plant personnel exposures
2.	Radwaste ventilation area	9.4.3	Charcoal filters were added in the radwaste ventilation area. Change initiated to reduce normal plant releases and/or plant personnel exposures
3.	Steam generator blowdown system	10.4.8	The blowdown heat exchanger was installed in place of the blowdown flash tank. Change initiated to reduce normal plant releases and/or personnel exposures
4.	Containment volume	3.8.1.1	The containment volume was increased. Containment volume increased to hold peak containment post-LOCA pressures within acceptable limits when considering the effect of post-reflood energy addition to the containment
5.	Cable routing	8.3.1.4.3	Deleted reference to cable tray as Category I equipment. Cable tray supports are designed to meet Category I seismic requirements. No safety significance
6.	Emergency lighting	9.5.3.3	Emergency lighting supplied either from plant batteries or individual battery packs and not from 600-V emergency buses. Change results in increased reliability
7.	Startup auxiliary transformers Two startup auxiliary transformers provided in each unit instead of three for both	8.2.1-3	Load growth required an increase in capacity. Size of transformers could not be increased because interrupting capacity of switchgear would not be adequate for larger transformers. Four transformers decreased the number of ties between units
8.	Service water to diesel engine heat exchanger Each diesel engine HX was supplied by only one service water header until Unit 2 went into service, instead of two service water headers	8.5.1c (11)	Adequate separation of water pipes from Train A and Train B was impossible to maintain and still supply each engine from both headers; alternate safeguards such as barriers or pipes within pipes were not considered feasible. Each engine will be supplied water from Unit 1 and Unit 2

TABLE 1.3-3 (SHEET 2 OF 3)

	Item	Change Discussed in FSAR Subsection		Reason for Change
9.	Main steam isolation valves Each main steam line contains two redundant swing disc trip valves instead of one swing disc trip valve and one check valve.	10.3.9		The development of refined steam line blowdown mass and energy flows following postulated steam line breaks dictated the deletion of the swing check valves. It was judged im-practical to design a swing check valve for the new loadings. The planned swing check valves were changed to swing disc trip valves to provide redundancy in stopping forward flow. It was determined that stopping backflow from the intact steam generators and piping would be assured by the redundant forward flow trip valves, obviating the previous need for swing check valves
10.	Fuel handling area ventilation system	9.4.2.2.2		Charcoal filters were added in the fuel handling area ventilation system. Change initiated to reduce normal plant releases and/or personnel exposure
11.	Containment purge system	6.2.3.2.3		Charcoal filters were added in the containment purge system. Change initiated to reduce normal plant releases and/or personnel exposure
12.	Containment pre-access filtration system	6.2.3.2.3		This system was added in the containment ventilation system. Change initiated to reduce normal plant releases and/or personnel exposure
13.	Containment isolation barrier	6.2.4(1)	(1)	Deletion of valves in the containment pressure instrument lines; refer to table 6.2-19, sheet 6 of 6, Note 5. Also refer to table 1.3-4, item 20. No safety significance
		(2)	(2)	Deletion of the vents from the airlock doors to the penetra-tion room. The airlock doors contain a double seal arrange-ment. After a review of our PSAR design, it was determined that total plant releases following an assumed single failure of the innermost seal would be minimized by removing the vent line, since the contribution to total site boundary dose due to leakage past the second seal would be less than the con-tribution associated with free venting of gross leakage past the inner seal to the penetration room, even when the effects of the penetration room filtration system are considered
14.	Features for mitigating high-energy line rupture outside containment	Appendix 3K		Added a venting penthouse to avoid overpressurizing the main steam and feedwater valve room. Relocated hot shutdown panel due to environmental considerations. Motor-driven auxiliary feedwater pump rooms are now watertight and the turbine-driven auxiliary feedwater pump room is no longer watertight

TABLE 1.3-3 (SHEET 3 OF 3)

	Item	Change Discussed in FSAR Subsection	Reason for Change
15.	Moved the emergency pond spillway structure from the northeast abutment of the main dam to the north leg of the pond	2.4.8	To remove spillway from the dike so as to eliminate potential effects on the downstream dike slope due to discharge over the spillway. Either design would be safe, but there is less risk of damage with the current design. (Reference AEC letter of 3/1/73 for acceptance of the relocated spillway.)

TABLE 1.3-4 (SHEET 1 OF 3)

SIGNIFICANT DESIGN CHANGES SINCE PSAR

Item	Change in Design	FSAR Reference	Reason for Change
1.	Core internals change - redesign of thermal shield pads to neutron pads	4.2.2.2	In order to simplify the core support structure, a design study was conducted to investigate the possibility of replacing the thermal shield with locally positioned plate members called neutron shielding pads. The plot of the circumferential fluence distribution of the vessel, which is very localized, showed this to be a practical approach. The design study established the feasibility of replacing the thermal shield with a series of plate members of equivalent thickness attached to the core barrel by pins and bolts. Both analytical and experimental methods were utilized to determine design adequacy. The above design study is described in WCAP-7870, which has been submitted to the NRC
2.	Core internals change - redesign of lower core support plate	4.2.2.2	A standard flat plate forging was developed to replace the original casting design
3.	Core internals change - redesign of upper internals support system	4.2.2.2	Allows a flat upper support plate to be used without affecting the hydraulic flow characteristics, and results in short support columns and guide tubes providing increased margins from possible flow-induced vibrations
4.	Reactor vessel top and bottom head penetration and CRDM were redesigned to meet ISI requirements	5.4.1.4, 5.4.4.4	To meet requirements of Section XI of the ASME Boiler and Pressure Vessel Code
5.	Addition of removable insulation on the closure and lower reactor vessel heads	5.4.2	To provide access to these areas for inspection purposes
6.	Rod withdrawal stop from rod drop signal and automatic turbine load cutback initiated by rod drop have been replaced by the power range neutron flux positive rate trip	7.2	The positive neutron flux rate trip ensures that the criteria appropriate for an ANS Condition IV event are met even for rod ejections from partial power.
7.	Analog RPI change to digital RPI	7.7.1.3	Improved performance with reduced setup and calibration time. An advancement in the "state of the art"
8.	Steam generator blowdown demineralizers added to blowdown system	11.2	Added system flexibility to the blowdown system
9.	Changed all clean radioactive waste treatment system valves to diaphragm valves except where system function or fluid conditions dictate otherwise	11.2	To reduce system leakage sources

TABLE 1.3-4 (SHEET 2 OF 3)

Item	Change in Design	FSAR Reference	Reason for Change
10.	Fuel pellet density	4.3	The pellet densities cited in the PSAR show a variation by fuel region of 91-, 92-, and 94-percent TD; whereas this FSAR cites 94-percent TD for all regions. The PSAR reflects the former design philosophy which provided lower density pellets for higher burnup regions to accommodate fuel swelling; however, experience has shown that swelling is not a strong function of burnup as previously believed, so that a uniform core pellet density can be employed. As discussed in section 1.5 and paragraph 4.2.1.3.1, the pellet density specification is subject to considerations of irradiation-induced densification
11.	Revised method and accuracy of determining water-soluble nitrates in tendon sheathing filler material from ASTM to Hach Chemical Co. procedure	3.8.2	The Hach procedure is simpler, faster, and offers the same degree of accuracy
12.	Two startup auxiliary transformers are provided for each unit instead of three for two units	8.1.2	Load growth requires an increase in capacity. Size of transformers could not be increased because interrupting capacity of switchgear would not be adequate for larger transformers. Four transformers decreased the number of ties between units
13.	Each diesel engine heat exchanger was supplied by only one service water header until Unit 2 went into service, instead of two service water headers	8.3.1.1.7 (11)	Adequate separation of water pipes from Train A and Train B was impossible to do and still supply each engine from both headers, and alternate safeguards such as barriers of pipes within pipes were not considered feasible. Each engine will be supplied water from Unit 1 and Unit 2
14.	Changed the stainless steel in the spent fuel pool and refueling cavity from Type 304L to Type 304	3.8.1.1.3	Type 304L was originally specified for better welding characteristics. Subsequent experience has shown Type 304 to be welded as easily as Type 304 L
15.	The material of the main steam lines up the check valves was changed to A-155 KCF70 Class I	Figure 3.2-8	To assure satisfactory results in impact tests as per ASME Section III.
16.	Deleted the 12-in. minimum vertical spacing between cable trays containing different classes of circuits in the same safeguard train as specified in paragraphs 8.7.2.5.c and 8.7.2.5.d	8.3.1.4.3	The design meets NRC requirements on 5-ft vertical and 3-ft horizontal separation between cable trays of redundant safeguard trains. In a few instances, it was not possible to meet the 12-in. separation between cable trays containing different classes of circuits of the same safeguard train. These instances have been examined by the designer on a case-by-case basis to ensure that the spacings are adequate
17.	Fuel assembly change- 15-x-15 fuel assemblies to 17-x-17 fuel assemblies	4.1, 4.4.1	The average linear power is reduced with the 17-x-17 fuel assembly
18.	RCS pipe design code change from ANSI B31.7 to ASME III	5.5.3	The ASME III code went into effect subsequent to the PSAR. The code effectivity date covered Farley

TABLE 1.3-4 (SHEET 3 OF 3)

Item	Change in Design	FSAR Reference	Reason for Change
19.	Steam flow restrictors installed in the steam generator outlet nozzle	5.5.4, 10.3.6	Use of the integral steam flow restrictor for steam flow measurements precludes the need for an additional flow measurement device. See section 3.2 for design criteria
20.	Deletion of valves in containment pressure instrument lines	6.2.4	The present sealed sensing lines without valve arrangement provides automatic double barrier isolation without operator action and without sacrificing any reliability with regard to its safeguards function
21.	Method of LOCA analysis	15.3, 15.4	The results of LOCA analysis are in conformance with 10 CFR 50.46 and appendix K
22.	Specification of fuel design limits	4.4, 15.2	Acceptable fuel design limits for the 17-x-17 fuel assembly for anticipated operational occurrences are presented in the references sections

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 APPLICANT - OWNER AND OPERATOR

Alabama Power Company (APC) is the owner of the Farley Nuclear Plant. Alabama Power Company and Southern Nuclear Operating Company (SNC) are the licensees. SNC has exclusive responsibility for and control over the physical construction, operation, and maintenance of the facility.

1.4.1.1 Description of Business

Alabama Power Company is a public utility incorporated under the laws of the State of Alabama and is engaged in the generation, distribution, and sale of electricity at retail in 639 cities and towns and at wholesale to 15 municipal electric systems and 11 rural electric cooperatives. Alabama Power Company's electric generating facilities consist of 8 steam electric generating plants, 14 hydroelectric generating plants, and 1 combustion turbine generating plant, with an installed capacity of 10,285,475 kW as of December 31, 1991.

Alabama Power Company, acting as its own general contractor, has constructed 27 steam electric units and at present has none under construction. Within the last 30 years, these include the following:

In Operation

Unit	Mwe
Greene County 1	250
Greene County 2	250
Barry 4	350
Barry 5	700
Gorgas 10	700
Gaston 5	880
Miller 1	660
Miller 2	660
Miller 3	660
Miller 4	660
Joseph M. Farley Nuclear Plant No. 1	829
Joseph M. Farley Nuclear Plant No. 2	829

1.4.1.2 Description of Corporate Organization

Alabama Power Company (APC) is a public utility incorporated under the laws of the State of Alabama with its principal offices located at 600 North 18th Street, Birmingham, Alabama. Southern Nuclear Operating Company (SNC) is responsible for operational support functions for the six nuclear units within the Southern electric system. APC and SNC are wholly owned subsidiaries of The Southern Company.

1.4.1.3 Technical Qualifications

The licensees and their affiliated companies have participated in the development of nuclear power for about 30 years as members of Atomic Power Development Associates, Inc., and Power Reactor Development Company, the designers and operators of the Enrico Fermi Atomic Power Plant. The participation has consisted of both financial contributions and assignment of personnel.

A course, Introduction of Nuclear Power, conducted by the University of Alabama, was begun in January 1970. The course requirements, consisting of 144 classroom hours, were completed by a group of 24 engineers in the Engineering, Power Supply, and Construction Departments. A second group of 11 personnel completed the course on an audit basis.

The technical qualifications of APC and SNC are further delineated in section 13.1, Organization Structure of APC.

1.4.2 NUCLEAR OPERATIONS ORGANIZATION

1.4.2.1 Southern Nuclear Operating Company, Inc.

Southern Nuclear Operating Company, Inc. (SNC) is a wholly owned subsidiary of The Southern Company.

SNC was formed from support organizations of Southern Company Services, Inc. (SCS), Georgia Power Company, and APC, whose individuals collectively have many years of experience in the area of nuclear operations. SNC is responsible for the operational and maintenance support of FNP. SNC is a licensee, along with APC, of FNP, as well as the sole operator of the plant.

1.4.3 ARCHITECT - ENGINEER

SNC is responsible for the engineering and design of the Farley Nuclear Plant. Bechtel Power Corporation has been retained by SNC as a subcontractor for the major portion of the plant design.

1.4.3.1 Bechtel Power Corporation

Bechtel Power Corporation is retained by SNC to assist in the engineering and design of the Farley Nuclear Plant. Bechtel has extensive experience in the design and construction of over 182 thermal generating units representing more than 82,900,000 kW of new generating capacity, of which 54 units are nuclear with more than 43,700,000 kW.
Among the numerous nuclear projects for which Bechtel is presently acting or has acted as engineer-constructor are the following pressurized water reactor (PWR) units:

- Robert E. Ginna Nuclear Station (450,000 kW) for Rochester Gas & Electric Corporation (construction only).
- Turkey Point Units 3 and 4 (760,000 kW each) for Florida Power & Light Company.
- Palisades Unit 1 (770,000 kW) for Consumers Power Company.
- Point Beach Units 1 and 2 (450,000 kW each) for Wisconsin-Michigan Power Company.
- Oconee Units 1, 2, and 3 (840,000 kW each) for Duke Power Company (engineering only).
- Calvert Cliffs Units 1 and 2 (800,000 kW each) for Baltimore Gas and Electric Company.
- Millstone Unit 2 (830,000 kW) for Northeast Utilities Company (The Millstone Point Company).
- Davis-Besse Nuclear Power Station (872,000 kW) for the Toledo Edison Company and the Cleveland Electric Illuminating Company.
- Arkansas Nuclear 1, Units 1 and 2 (900,000 kW each) for Arkansas Power and Light Company.
- Trojan Unit 1 (1,100,000 kW) for Portland General Electric Company.

1.4.4 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) SUPPLIER

1.4.4.1 Westinghouse's Qualifications and Experience

Westinghouse Electric Corporation (Westinghouse) is responsible for supplying the nuclear steam supply system (NSSS) and fuel for Farley Units 1 & 2.

Westinghouse has designed, developed, and manufactured nuclear power facilities since the 1950s, beginning with the world's first large central station nuclear power plant (Shippingport), which produced power from 1957 to 1982. Completed or presently contracted commercial nuclear capacity totals in excess of 97,000 MW. Westinghouse pioneered new nuclear design concepts, such as chemical shim control of reactivity and the rod cluster control concept, throughout the last two decades. Westinghouse manufacturing facilities include the largest commercial nuclear fuel fabrication facility in the world, and the world's most modern heat transfer equipment production facility, as well as other facilities producing NSSS components. Table 1.4-1 lists all Westinghouse pressurized water reactor (PWR) plants to date, including those plants currently under construction.

The US Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute have contracted with Westinghouse for research into NSSS related activities. Westinghouse experience was also utilized by the NRC and Metropolitan Edison immediately following the Three Mile Island Unit 2 accident and remains as a heavy participant with the Westinghouse Owner's Group of utilities in addressing the NRC's action plan for corrective actions.

1.4.4.2 Plants in Operation

Westinghouse PWR plants in operation are as follows:

A. Shippingport

Shippingport was the world's first large central station nuclear power plant. The reactor plant was designed by the Bettis Atomic Power Laboratory, which is operated by Westinghouse under a NRC contract. Shippingport's PWR has produced power for Duquesne Light Company since December 1957.

B. Yankee-Rowe

Singled out by the NRC as a "Nuclear Success Story," Yankee-Rowe went online in November 1960. Owned and operated by the Yankee Atomic Electric Company, Yankee-Rowe has progressed from an initial rating of 120 MWe to its present 175 MWe rating. Westinghouse supplied the NSSS and the turbine generator.

C. Trino Vercellese (Enrico Fermi)

The Trino Vercellese nuclear plant was one of the first Westinghouse designed plants to incorporate chemical shim control of reactivity. Chemical shim has since become a standard feature of Westinghouse PWR control. Trino Vercellese achieved initial criticality in June 1964 and began power operation in October 1964. The plant is rated at 260 MWe.

D. Chooz (Ardennes)

The Chooz plant is unique in that the Westinghouse PWR and its auxiliaries are housed in man-made caverns. Ardennes, a joint Franco-Belgian undertaking, owned and operated by the Societa d 'Energie Nucleaire Franco-Belge des Ardennes (SENA), is located in France near the French-Belgian border. Chooz achieved initial criticality in October 1966 and began power operation in 1967.

E. San Onofre No. 1

San Onofre No. 1 employs the Westinghouse developed rod cluster control concept which has since become a standard feature on the Westinghouse PWR. Owned by the Southern California Edison Company and the San Diego Gas and Electric Company, the 450 MWe plant is located near San Clemente, California. Westinghouse supplied the NSSS and the turbine generator. Initial criticality was achieved in June 1967, and power operating began in January 1968.

F. Haddam Neck (Connecticut Yankee)

Owned and operated by the Connecticut Yankee Atomic Power Company, this plant went critical in July 1967 and attained full power operation in December 1967. Like San Onofre No. 1, the plant employs rod cluster control in conjunction with chemical shim control. Westinghouse supplied the NSSS and the turbine generator. The plant has been uprated to 575 MWe.

G. Jose Cabrera - (Zorita)

The Jose Cabrera station is located near Zorita, Spain. The 153 MWe plant employs rod cluster control, chemical shim control and a zircaloy-clad core. Construction began in mid-1965, and power operation began in 1968. Jose Cabrera is owned and operated by the Union Electra, S.A., a Spanish utility.

H. Beznau No. 1 and No. 2

Beznau No. 1, Switzerland's first commercial nuclear power plant, achieved initial criticality in June 1969 and supplied power to the system in July 1969. The 350 MWe plant was designed and constructed by the Westinghouse-Brown Boveri Consortium for the owner/operator utility, Nordostschweizerische Kraftwerke A.G. The plant started producing power less than 4 years after award of the plant contract. Beznau No. 2 achieved criticality in October 1971 and began commercial operation in early 1972.

I. Robert Emmett Ginna

The Robert Emmett Ginna Plant, owned and operated by Rochester Gas and Electric Corporation, is located in New York on the south shore of Lake Ontario. Westinghouse supplied the 490 MWe plant on a turnkey basis. Construction began in April 1966 with initial criticality being achieved in November 1969 (just 42 months after start of construction). Power was supplied to the system in December 1969.

J. Mihama No. 1 and Takahama No. 1

These plants are owned by the Kansai Electric Power Company, Inc. Mihama No. 1 is a two-loop, 320 MWe unit and marks the beginning of a line of Westinghouse PWRs supplying the generation needs of the Far East. Westinghouse International Company was the prime contractor for the Mihama project, supplying the NSSS engineering, nuclear fuel, and some major system components. Mihama No. 1 required only 44 months from the start of site construction to first power production in August 1970. Takahama No. 1 is a three-loop, 780 MWe unit. Initial criticality was achieved in March 1974.

K. H. B. Robinson No. 2

This plant is a three-loop, 707 MWe unit which was built on a turnkey basis for the Carolina Power and Light Company. The plant is located at a site near Hartsville, South Carolina, on a man-made cooling lake. The construction permit was granted in April 1967. The plant achieved criticality in August 1970 and first power to system in October 1970.

L. Point Beach No. 1 and No. 2

The Point Beach Project consists of two 497 MWe units, which were built on a turnkey basis for the Wisconsin Michigan Power Company and the Wisconsin Electric Power Company. The plants are located near Two Creeks, Wisconsin, 90 miles north of Milwaukee on Lake Michigan. This was the first two-unit station to utilize many common facilities and shared auxiliary systems. The construction permit for Point Beach No. 1 was granted in July 1967 with initial criticality and first power to the system in November 1970. Point Beach No. 2 went critical in May 1972 and was available for commercial operation in October 1972.

M. Surry No. 1 and No. 2

The Surry Power Station, two three-loop 822 MWe units, is owned by the Virginia Electric and Power Company. The James River Station is about 30 miles from Norfork, Virginia. First criticality on Surry No. 1 was achieved in July 1972. Commercial operation began in September 1972. Initial criticality on Surry No. 2 was achieved in March 1973.

N. Turkey Point No. 3 and No. 4

Florida Power and Light Company is the owner of a four-unit station in Biscayne Bay, Florida. Turkey Point No. 3 and 4 of the station are three-loop, 745 MWe plants. Commercial status for Turkey Point No.3 was achieved in December 1972. Initial criticality for Turkey Point No. 4 was achieved in June 1973.

O. Indian Point No. 2 and No. 3

Consolidated Edison Company of New York operates three nuclear units located in Buchanan, New York. Units 1 and 2 are owned by the Company and Unit 3 is owned by the Power Authority of the State of New York. Units 2 and 3 are Westinghouse PWRs rated at 873 and 965 MWe, respectively. Indian Point No. 2 achieved initial criticality in May 1973 and Indian Point No. 3 achieved initial criticality in April 1976.

P. Prairie Island No. 1 and No. 2

Northern States Power Company is the owner of these two-loop, 530 MWe units located in Welch, Minnesota. Initial criticality was achieved in December 1973 for Prairie Island No. 1, and in December 1974 for Prairie Island No. 2.

Q. Zion No. 1 and No. 2

Commonwealth Edison Company is the owner of these two four-loop, 1050 MWe units. The units are located on Lake Michigan near Zion, Illinois. Initial criticality was achieved in June 1973 for Zion No. 1 and in December 1973 for Zion No. 2.

R. Kewaunee

Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company are the owners of this two-loop, 541 MWe plant located in Kewaunee, Wisconsin. Initial criticality was achieved in March 1974.

S. Ringhals No. 2

Statens Vattenfallsverk (SSPB) is the owner of this three-loop, 822 MWe unit located in Sweden. Initial criticality was achieved in June 1974.

T. Donald C. Cook No. 1 and No. 2

Indiana and Michigan Electric Company is the owner of these four-loop, 1090 MWe plants located in Bridgman, Michigan. These plants are the first to use the Westinghouse Ice Condenser Containment design. Initial criticality was achieved in January 1975 for Unit 1 and March 1978 for Unit 2.

U. Trojan

This four-loop, 1130 MWe plant is jointly owned by Portland General Electric Company, Eugene Water and Electric Board, and Pacific Power and Light Company. In addition to being the first commercial nuclear plant to operate in the Pacific Northwest (located on the Oregon shore of the Columbia River near Rainier, Oregon), Trojan is the first 17 x 17 fuel-rod-per-assembly plant to achieve criticality. Initial criticality was achieved in December 1975.

V. Beaver Valley No. 1

This three-loop, 852 MWe plant is jointly owned by Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company. Beaver Valley No. 1 is located on the Ohio River, 22 miles northwest of Pittsburgh, Pennsylvania. Commercial operation began in early 1976.

W. Salem No. 1 and No. 2

Salem No. 1 & 2, owned jointly by the Public Service Electric and Gas Company, Philadelphia Electric Company, Atlantic Electric Company, and Delmarva Power and Light Company, is located on Artificial Island, a man-made peninsula in Salem County, New Jersey. The 1090 MWe, four-loop plant achieved initial criticality for Unit 1 in late 1976 and Unit 2 achieved criticality in August 1980.

X. North Anna No. 1 and No. 2

Virginia Electric and Power Company owns the two approximately 907 MWe (net) plants located 40 miles north of Richmond, Virginia, on Lake Anna. Unit 1 achieved criticality in May 1978 and Unit 2 achieved criticality in June 1980.

Y. Joseph M. Farley No. 1 and No. 2

The two Alabama Power Company units are located at Dothan, Alabama (approximately 180 miles south - southwest of Atlanta, Georgia). Unit 1 achieved criticality in August 1977 and Unit 2 achieved criticality in February 1981. The two units were originally designed with a gross electrical output of approximately 861 MWe. Modifications to the HP turbine in the 1980's improved the turbine efficiency to approximately 885 MWe (gross). Subsequent modifications performed in support of power uprate in the late 1990's increased predicted gross output to approximately 910 MWe (approximately 883 MWe net). Replacement of the Unit 1 and Unit 2 low pressure turbines in 2010 and 2011, respectively, increased predicted gross output to approximately 933.4 MWe. Implementation of the measurement uncertainty recapture (MUR) power uprate and the replacement of the Unit 1 and Unit 2 high pressure turbines in 2021 and 2020, respectively, increased predicted gross output to approximately 954.4 MWe (Unit 1) and 952.8 MWe (Unit 2).

Z. Sequoyah No. 1 and No. 2

The two units approximately 1148 MWe (net) are located on the Tennessee River near Chattanooga, Tennessee. These two units are owned by Tennessee Valley Authority. Sequoyah No. 1 received a full power license in September 1980.

1.4.4.3 Westinghouse Facilities

Westinghouse, in its effort to plan for the future, has developed a broad range of facilities to satisfy the needs of the nuclear industry. The following paragraphs briefly describe these facilities.

A. Columbia Plant, Nuclear Fuel Division

The Columbia Plant is capable of performing all operations necessary to manufacture finished nuclear fuel assemblies. These operations include conversion of uranium hexafluoride to uranium dioxide powder, fabrication of fuel assembly grids, complete pellet loading, and final fabrication of assemblies. The plant, located at Columbia, South Carolina, began full production in early 1970. The Columbia Plant is the largest commercial nuclear fuel fabrication facility in the world.

B. Westinghouse Pensacola Plant

The Westinghouse Pensacola Plant, located on Escambia Bay on the northwest coast of Florida, produces precision reactor vessel internals, steam generators, and pressurizers. Contributing to the precision manufacturing capability is an environmental control system which minimizes year round temperature changes throughout the shop area. Transportation facilities of the plant include a railroad spur which permits loading and unloading inside the shop, and access to barge loading facilities on Escambia Bay.

C. Cheswick Plant, Electro-Mechanical Division

The Electro-Mechanical Division was established in Cheswick, Pennsylvania, in 1953 to manufacture canned motor primary coolant pumps for nuclear reactors. Today, the product line has expanded to include shaft seal pumps (reactor coolant pumps), valves from 4 inches to 31 inches, and control rod drive mechanisms, essential components of the Westinghouse PWR. The facility occupies 250,000 square feet and now contains the most modern facilities available for the production and testing of nuclear plant components.

D. Specialty Metals Division

The Specialty Metals Division located in Blairsville, Pennsylvania, was completed in late 1967. Several essential PWR component processes are accomplished at Blairsville, including: the precision manufacture of inconel tubing for steam generators, and the complete processing of zircaloy seamless tubing for nuclear fuel cladding. At Blairsville, complete quality control facilities are utilized for the evaluation and analysis of all specialty metal products used in Westinghouse nuclear systems.

E. Westinghouse Nuclear Center

The headquarters of Westinghouse Nuclear Energy Systems is located just east of Pittsburgh, in Monroeville, Pennsylvania. Operating primarily as a headquarters and engineering facility, the complex houses many of the divisions which encompass Westinghouse's nuclear activities associated with the electric utility industry.

F. Zion Nuclear Training Center

The Westinghouse Electric Corporation and the Commonwealth Edison Company of Chicago have built and are operating a nuclear training center at Zion, Illinois. The 28,000 square foot training center contains classrooms, a training reactor, training material center, video recording facilities, and multi-plant nuclear power plant simulators .Westinghouse staffs and operates the center, supplies all the equipment required, and is responsible for the development and presentation of all training programs. Commonwealth Edison provided the building, access to the Zion nuclear units for conducting in-plant observation training, and advises and assists Westinghouse in developing training programs.

G. Instrumentation, Technology, and Training Center - Strategic Operations Division

The Strategic Operations Division was formed to address the man/machine interface including instrumentation, control, and training. Man/machine interface activities include the technical support center, plant computer development, instrumentation development, simulator, and software development related to all of these activities. The Strategic Operations Division also provides comprehensive training for nuclear power plant personnel.

1.4.5 PLANT CONSTRUCTION

Southern Nuclear is responsible for operation of Farley Nuclear Plant and utilizes contractors as appropriate to support operational, construction, and outage activities. Contractors are required to conform to technical qualification and quality assurance requirements associated with assigned work activities.

1.4.6 DIVISION OF RESPONSIBILITY

1.4.6.1 Alabama Power Company (APC)

Alabama Power Company and Southern Nuclear Operating Company (SNC) are the holders of the facility license. In fulfilling its responsibility, APC has delegated certain activities to other organizations as explained in the paragraphs covering SNC, Southern Company Services, Inc. (SCS), Bechtel Power Corporation, and Westinghouse Electric Corporation, which follow.

1.4.6.2 Southern Nuclear Operating Company, Inc.

SNC is a licensee, along with APC, and is the sole operator of Farley Nuclear Plant. SNC provides technical and general operations and maintenance services for nuclear operations, including (i) work relating to the licensing, operation, surveillance, testing, maintenance, quality assurance, outage planning, health physics, plant production, security, retirement, decommissioning, training, emergency planning and responses, waste management, engineering and environmental studies, nuclear fuel supply studies, and core design engineering; (ii) performance or procurement of engineering and design work associated with SNC nuclear operations; and (iii) making available to APC the services of qualified technicians or specialists, inspectors, and supervisory personnel for many phases of nuclear operations.

Administrative, technical, training, operations, maintenance, performance and planning, and plant modifications personnel make up the plant operating staff of SNC.

In addition, SNC also provides administrative services in support of SNC nuclear operations. These services include procurement services, accounting and statistical services, employee relation services, and information management services.

Southern Company Services, Inc. had a traditional relationship with Alabama Power Company based on many years of association together as member companies in The Southern Company. During this association, SCS acted as the architect-engineer for APC in the design and engineering of new fossil-fired steam generating units. For the Farley Nuclear Plant (FNP), SCS had the original responsibility for developing and implementing the design for the turbine generator building and the balance of the plant not assigned to Bechtel, including the review or audit, approval, and documentation of the basic design concepts, detail designs, specifications, and drawings.

As a result of the consolidation of SCS and SNC nuclear expertise and in addition to being the licensee, SNC also serves as its own Architect/Engineer and performs the functions previously performed by SCS.

1.4.6.3 Bechtel Power Corporation

Bechtel Power Corporation has been retained by SNC to act as its consultant on the nuclear project portion of the plant. In this capacity, Bechtel is responsible for the review or audit, approval, and documentation of the basic design concepts, detail designs, specifications, and drawings for the reactor building, auxiliary building, and other structures and facilities of the FNP. Bechtel also acts in an advisory capacity to SNC on other matters that may be assigned to them from time to time.

1.4.6.4 Westinghouse Electric Corporation

Westinghouse Electric Corporation has been awarded a contract by APC to design and fabricate the nuclear steam supply system. The contract covers the standard 3-loop plant of Westinghouse.

TABLE 1.4-1 (SHEET 1 OF 6)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Commercial Operation	MWe Net	Number of Loops
Yankee-Rowe	Yankee Atomic Electric Company	Massachusetts	1961(a)	175	4
Trino Versellese (Enrico Fermi)	Ente Nazionale per l'Energia Elettrica (ENEL)	Italy	1965	260	4
Chooz (Ardennes)	Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA)	France	1967	305	4
San Onofre No. 1	Southern California Edison Co.; San Diego Gas and Electric Co.	California	1968	450	3
Haddam Neck (Connecticut Yankee)	Connecticut Yankee Atomic Power Company	Connecticut	1968	575	4
Jose Cabrera-Zorita	Union Electrica, S. A.	Spain	1969	153	1
Beznau No. 1	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1969	350	2
Robert Emmett Ginna	Rochester Gas and Electric Corporation	New York	1970	490	2
Mihama No. 1	The Kansai Electric Power Company, Inc.	Japan	1970	320	2
Point Beach No. 1	Wisconsin Electric Power Co.; Wisconsin Michigan Power Co.	Wisconsin	1970	497	2
H. B. Robinson No. 2	Carolina Power and Light Co.	South Carolina	1971	707	3
Beznau No. 2	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1971	350	2
Point Beach No. 2	Wisconsin Electric Power Co.; Wisconsin Michigan Power Co.	Wisconsin	1972	497	2
Surry No. 1	Virginia Electric and Power Co.	Virginia	1972	822	3
Turkey Point No. 3	Florida Power and Light Co.	Florida	1972	745	3

TABLE 1.4-1 (SHEET 2 OF 6)

Plant	Owner Utility	Location	Commercial Operation	MWe Net	Number of Loops
Indian Point No. 2	Consolidated Edison Company of New York	New York	1974	970	4
Prairie Island No.1	Northern States Power Company	Minnesota	1973	530	2
Turkey Point No. 4	Florida Power and Light Co.	Florida	1973	745	3
Surry No. 2	Virginia Electric and Power Co.	Virginia	1973	822	3
Zion No. 1	Commonwealth Edison Company	Illinois	1973	1,050	4
Kewaunee	Wisconsin Public Service Corp.; Wisconsin Power and Light Co.; Madison Gas and Electric Co.	Wisconsin	1974	503	2
Prairie Island No. 2	Northern States Power Company	Minnesota	1974	530	2
Takahama No. 1	The Kansai Electric Power Company, Inc.	Japan	1974	781	3
Zion No. 2	Commonwealth Edison Company	Illinois	1974	1,050	4
Beaver Valley No. 1	Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Company	Pennsylvania	1976	852	3
Doel No. 1	Indivision Doel	Belgium	1975	390	2
Doel No. 2	Indivision Doel	Belgium	1975	390	2
Donald C. Cook No. 1	Indiana and Michigan Electric Company (AEP)	Michigan	1975	1,090	4
Donald C. Cook No. 2	Indiana and Michigan Electric Company (AEP)	Michigan	1978	,090	4
Indiana Point No. 3	Power Authority of the State of New York (PASNY)	New York	1976	965	4
Ko-Ri No. 1	Korea Electric Co.	Korea	1978	564	2
Ringhals No. 2	Statens Vattenfallsverk (SSPB)	Sweden	1975	822	3

TABLE 1.4-1 (SHEET 3 OF 6)

Plant	Owner Utility	Location	Commercial Operation	MWe Net	Number of Loops
Trojan	Portland General Electric Co.; Eugene Water and Electric Board; Pacific Power and Light Company	Oregon	1976	1,130	4
Almaraz No. 1	Union Electrica, S. A.; Compania Sevillana de Electricidad, S. A.; Hidroelectrica Espanola, S.A.	Spain	1981	902	3
Diablo Canyon No. 1	Pacific Gas and Electric Co.	California	1985	1,084	4
Joseph M. Farley No. 1	Alabama Power Company	Alabama	1977	829	3
Lemoniz No. 1	Iberduero, S.A.	Spain	--(c)	902	3
Salem No. 1	Pacific Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic Electric Co.; Delmarva Power and Light Co.	New Jersey	1977	1,090	4
Sequoyah No. 1	Tennessee Valley Authority	Tennessee	1981	1,148	4
Almaraz No. 2	Union Electrica, S. A.; Compania Sevillana de Electricidad, S. A.; Hidroelectrica Espanola, S. A.	Spain	1984	902	3
Angra 1 dos Reis	Furnas-Centrais Electricas, S. A.	Brazil	1984	626	2
Asco No. 1	Fuerzas Electricas de Cataluna, S. A. (FECSA)	Spain	1983	902	3
Diablo Canyon No. 2	Pacific Gas and Electric Co.	California	1986	1,160	4
Joseph M. Farley No. 2	Alabama Power Company	Alabama	1981	829	3
North Anna No. 1	Virginia Electric and Power Co.	Virginia	1978	898	3
North Anna No. 2	Virginia Electric and Power Co.	Virginia	1980	898	3

TABLE 1.4-1 (SHEET 4 OF 6)

Plant	Owner Utility	Location	Commercial Operation	MWe Net	Number of Loops
Ohi No. 1	The Kansai Electric Power Company, Inc.	Japan	1979	1,122	4
Ohi No. 2	The Kansai Electric Power Company, Inc.	Japan	1979	1,122	4
Ringhals No. 3	Statens Vattenfallsverk (SSPB)	Sweden	1981	900	3
Sequoyah No. 2	Tennessee Valley Authority	Tennessee	1982	1,148	4
Asco No. 2	Fuerzas Electricas de Cataluna, S. A. (FESCA); Empresa Nacional Hidroelectrica del Ribagorzana, S. A. (ENHER); Fuerzas Hidroelectricas de Cataluna, S.A.; Hidroelectrica del Segre, S. A.	Spain	1986	902	3
Krsko	Savske Elektrane Ljubljana; Slovenia; Electroprivreda Zagreb, Croatia	Yugoslavia	1983	615	2
Lemoniz No. 2	Iberduero, S. A.	Spain	--(c)	902	3
Watts Bar No. 1	Tennessee Valley Authority	Tennessee	--(b)	1,177	4
William B. McGuire No. 1	Duke Power Company	North Carolina	1981	1,180	4
Millstone No. 3	Northeast Nuclear Energy Co.	Connecticut	1986	1,156	4
Ringhals No. 4	Statens Vattenfallsverk (SSPB)	Sweden	1983	900	3
Salem No. 2	Public Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic Electric Co.; Delmarva Power and Light Co.	New Jersey	1981	1,115	4
Virgil C. Summer	South Carolina Electric and Gas Company	South Carolina	1984	900	3
Watts Bar No. 2	Tennessee Valley Authority	Tennessee	--(c)	1,177	4

TABLE 1.4-1 (SHEET 5 OF 6)

Plant	Owner Utility	Location	Commercial Operation	MWe Net	Number of Loops
William B. McGuire No. 2	Duke Power Company	North Carolina	1984	1,180	4
Byron No. 1	Commonwealth Edison Co.	Illinois	1985	1,120	4
Catawba No. 1	Duke Power Company	South Carolina	1985	1,153	4
Comanche Peak No. 1	Texas Utilities Generating Co.	Texas	--(b)	1,150	4
Ko-Ri No. 2	Korea Electric Co.	Korea	1983	605	2
Seabrook	New Hampshire Yankee	New Hampshire	1990	1,200	4
South Texas Project Unit No. 1	Houston Lighting and Power Co.; Central Power and Light Co.; City Public Service of San Antonio; City of Austin, Texas	Texas	1988	1,250	4
Sayago No. 1	Iberduero, S. A.	Spain	--(c)	1,000	3
Vandellos No. 2	Asociacion Nuclear Vandellos	Spain	1988	930	3
Beaver Valley No. 2	Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Co.; Cleveland Electric Illuminating Company; Toledo Edison Company	Pennsylvania	1987	852	3
Braidwood No. 1	Commonwealth Edison Company	Illinois	1988	1,120	4
Callaway No. 1	SNUPPS - Union Electric Co.	Missouri	1984	1,150	4
Braidwood No. 2	Commonwealth Edison Company	Illinois	1988	1,120	4
Byron No. 2	Commonwealth Edison Company	Illinois	1987	1,120	4
Catawba No. 2	Duke Power Company	South Carolina	1986	1,153	4
Comanche Peak No. 2	Texas Utility Generating Co.	Texas	--(b)	1,150	4

TABLE 1.4-1 (SHEET 6 OF 6)

Plant	Owner Utility	Location	Commercial Operation	MWe Net	Number of Loops
Comanche Peak No. 2	Texas Utility Generating Co.	Texas	--(b)	1,150	4
South Texas Project Unit No. 2	Houston Lighting and Power Co.; Central Power and Light Co.; City Public Service of San Antonio; City of Austin, Texas	Texas	1989	1,250	4
Maanshan No. 1	Taiwan Power Company	Taiwan	1984	890	3
Wolf Creek	SNUPPS - Wolf Creek Nuclear Operating Corporation	Kansas	1983	1,188	4
Alvin W. Vogtle No. 1	Georgia Power Company; Oglethorpe Electric Membership Corp., Municipal Authority of Georgia; City of Dalton, Georgia	Georgia	1987	1,113	4
Alvin W. Vogtle No. 2	Georgia Power Company; Oglethorpe Electric Membership Corp., Municipal Authority of Georgia; City of Dalton, Georgia	Georgia	1989	1,113	4
Maanshan No. 2	Taiwan Power Company	Taiwan	1985	890	3
Shearon Harris No. 1	Carolina Power and Light Co.	North Carolina	1987	900	3
Napot Point No. 1	National Power Corp.	Phillippines	1985	620	2

a. In the decommissioning phase.
b. Uncompleted.
c. Indefinitely postponed.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The design of the Farley Nuclear Plant Units 1 and 2 is based upon proven concepts which have been developed and successfully applied to the design of numerous other pressurized water reactor systems.

The term "research and development" (R&D) as used in this section is the same as that used by the Commission in Section 50.2 of its 10 CFR 50 as follows:

> "(n) 'Research and development' means (1) theoretical analysis, exploration or experimentation; or (2) the extension of investigative findings and theories of a scientific nature into practical application for experimental and demonstration purposes including the experimental production and testing of models, devices, equipment, materials and processes."

The research and development discussed in the FSAR is to confirm the engineering and design values normally used to complete equipment and system designs. It does not involve the creation of new concepts or ideas.

The technical information generated is used either to demonstrate the safety of the design and more sharply define margins of conservatism or to lead to design improvements.

Each research and development program is briefly summarized for identification and its relationship to the Farley Nuclear Plant Units 1 and 2 is discussed. Detailed discussions of each R&D program are available in a more expanded summary form in reference 1 and other references as noted throughout this section.

1.5.1 PROGRAMS REQUIRED FOR PLANT OPERATION

In the Farley PSAR, three programs were identified as required for plant design and operation.

A. Core Stability Evaluation.

B. Fuel Rod Burst Program.

C. In-Pile Fuel Densification Program.

A. Core Stability Evaluation (Item 1 in reference 1)

The purpose of this program is to establish means for the detection and control of potential xenon oscillations and for the shaping of the axial power distribution for improved core performance. This program has been completed. Refer to reference 1 for a further discussion of these tests.

B. Fuel Rod Burst Program (Item 2 in reference 1)

The original rod burst program, a study of the performance of zircaloy cladding under simulated loss-of-coolant accident (LOCA) conditions, has been completed.

It has supplied empirical data from which the effect of geometry distortion on the ability of the emergency core cooling system (ECCS) to meet the LOCA design criteria has been determined using present analytical design techniques.

The program included burst and quench tests on single rods and burst tests on rod bundles. As a result of single rod tests, specific design limits have been established on peak clad temperature and allowable maximum metal water reaction to assure effective core cooling. The multirod burst tests demonstrated that even when rod to rod contact does occur after burst, the remaining flow area is always sufficient to ensure adequate core cooling.

The single rod burst test program for the 17 x 17 fuel pin array is discussed in subsection 1.5.4.3.

C. In-Pile Fuel Densification (Item 22 in reference 1)

Operating experience with uranium dioxide fuel has indicated that the fuel may densify under irradiation, to a density higher than that to which it was manufactured. This densification can lead to shorter active fuel length stacks, increased initial pellet to clad radial gaps, and pellet to pellet axial gaps. The shorter fuel stack length gives rise to a small increase in overall, average linear power density (kW/ft). Increased radial gaps reduce gap conductance and lead to higher pellet temperatures. Axial gaps give rise to local power peaking due to decreased neutron absorption.

Westinghouse fuel densification research was directed toward producing fuel with a structure that minimizes in-pile densification (hereafter called stable fuel). The objective of the program was to define material characteristics and manufacturing processes that lead to stable fuel. Stable fuel is defined as fuel with a small densification. Residual effects of densification were evaluated on a model developed by this program. A more detailed description of the program and results is presented in reference 1.

1.5.2 OTHER PROGRAMS REQUIRED FOR PLANT OPERATION

Other areas of research and development, as outlined below, are those which give added confirmation that the design is conservative.

A. Burnable Absorber Program (Item 7 in reference 1)

Burnable absorber rod development is complete. The burnable absorber rods for the first core are borosilicate glass encased in stainless steel tubes. The fixed rods are used to reduce the concentration of boric acid absorber in the moderator, thereby ensuring that the moderator temperature coefficient of reactivity is always within its limit specified in the Core Operating Limits Report, as required by the Technical Specifications. Refer to reference 1 for a further discussion of this program.

B. Fuel Development Program for Operation at High Power Densities (Item 8 in reference 1)

To demonstrate satisfactory operation of fuel at high burnup and power densities, and to define design margins, a program was designed to test fuel in both the Saxton and Zorita reactors. The Saxton loose-lattice irradiation program was designed to demonstrate fuel performance at conditions significantly in excess of PWR design limits, and to establish power burnup limits for the fuel. The Zorita reactor is the first PWR with a zircaloy core to operate at similar core conditions as the current design units. Because of the timely manner in which fuel can be irradiated in Zorita, four fuel assemblies are being tested there to demonstrate satisfactory operation of the fuel in a commercial PWR environment.

Sustained successful operation of special Zorita fuel rods at peak design power levels, in excess of those planned for the Farley Nuclear Units, will increase assurance that the fuel had adequate performance margins to accommodate transient overpower operation.

The Saxton loose-lattice irradiation and Saxton parametric irradiation subprograms have been completed. It is concluded that the loose lattice program has satisfactorily completed the test objective. The work of the loose-lattice assemblies was partly performed under USAEC Contract AT (11-1) - 3044 and has been reported on a quarterly basis (reference 10); a fuel materials performance report has been published. (See reference 11.)

C. FLECHT (Full Length ECCS Heat Transfer Test) (Item 12 in reference 1)

The objective of the FLECHT program was to obtain experimental reflooding heat transfer data under simulated loss-of-coolant accident conditions for use in evaluating the heat transfer capabilities of pressurized water reactor emergency core cooling systems.

The current test results verified the ability of a bottom flooding ECCS design to terminate the temperature increase during a LOCA. The LOCA evaluation presented in this application utilized the results of the FLECHT program for the analysis of the reflooding phase of the accident.

D. Loss-of-Coolant Analysis Program (Item 14 in reference 1)

This program has been completed with the results of the Flashing Heat Transfer Program (item 13 in reference 1) being incorporated in the core thermal design code used in the LOCA analysis presented in this application.

The loss-of-coolant analysis program was established to integrate, as appropriate, the more realistic heat transfer models obtained from experimental and analytical development programs into the core thermal design codes used to evaluate the loss-of-coolant accident.

E. Reactor Vessel Thermal Shock (Item 16 in reference 1)

The effects of safety injection water on the integrity of the reactor vessel, following a postulated loss-of-coolant accident, have been analyzed using data on fracture toughness of heavy section steel at the beginning of plant life and after irradiation corresponding to approximately 40 years of equivalent plant life.[a] The results show that under the postulated accident conditions, the integrity of the reactor vessel is maintained.

Fracture toughness data are obtained from a Westinghouse experimental program which is associated with the Heavy Section Steel Technology (HSST) Program at ORNL and EURATOM programs. Since results of the analyses are dependent on the fracture toughness of irradiated steel, efforts are continuing to obtain additional confirmatory data. Data on 2-in.-thick specimens became available in 1970 from the HSST program. These data indicated a strong temperature dependence with a rapid increase in toughness at approximately NDT. For results obtained in the HSST program, the HSST Semiannual Progress Report, issued by the Oak Ridge National Laboratory (quarterly beginning in 1974), should be consulted.

F. Blowdown Forces Program (Item 15 in reference 1)

The objective of the Blowdown Forces program was to develop a digital computer program for the calculation of pressure, velocity, and force transients in the reactor coolant system during a loss-of-coolant accident, and to utilize this code in the calculation of blowdown forces on the fuel assemblies and reactor internals to ensure that the stress and deflection criteria used in the design of these components were met.

Westinghouse has completed the development of BLODWN-2, an improved digital computer program for the calculation of local fluid pressure, flow, and density transients in the reactor coolant system during a LOCA.

Extensive comparisons have been made between BLODWN-2 and test data. Agreement between code predictions and data has been good.

Analysis using the BLODWN-2 program to evaluate the effects of blowdown is presented elsewhere in this application. It was concluded from the analysis that the design of this reactor meets the established design criteria.

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a. The operating licenses for both FNP units have been renewed and the original licensed operating terms have been extended by 20 years. Reactor vessel neutron embrittlement was evaluated as a time-limited aging analysis (TLAA) for license renewal in accordance with 10 CFR Part 54. The results of this analysis are provided in chapter 18, subsection 18.4.1.

G. ESADA DNB Program (Item 11 in reference 1)

The ESADA DNB program has been completed. The experimental program was conducted with rod bundles to determine the effect on DNB from:

1. Axially uniform and nonuniform heat flux distributions.

2. Radially uniform and nonuniform heat flux distributions.

3. Mixing vane grids.

Data obtained covered ranges for such parameters as pressure, temperature, mass flow, and heat flux pertinent to present PWR design. Good data agreement was obtained with the nonuniform heat flux W-3 DNB correlation which had been developed on single channel data. The scatter of rod bundle data was decidedly less than the scatter of single channel data used to develop the correlation.

The effects of the new 17 x 17 fuel assembly geometry on the DNB heat flux is discussed in subsection 1.5.3.6.

1.5.3 17 x 17 FUEL ASSEMBLY VERIFICATION TESTS

A comprehensive test program for the 17 x 17 assembly has successfully been completed by Westinghouse. Reference 1 contains a summary discussion on the program.

Some of the verification work described herein was conducted using 17 x 17 assemblies of seven-grid design, whereas the selected 17 x 17 assembly design has eight grids. Tabulated below are those 17 x 17 tests which utilized a seven-grid geometry, and the effect of adding an eighth grid.

Test	**Parameter**	**Effect**
Fuel Assembly Structural test	Axial Stiffness	Negligible effect at blowdown impact forces (9)
	Lateral Impact	Additional grid shares impact load (9)
Prototype Assembly Test	Pressure Drop	The margin between 7-grid design ΔP and D-loop results(10) is adequate to cover the additional ΔP resulting from the additional grip (<5 percent increase in ΔP)

Test	Parameter	Effect
	Lift Force	The margin between 7-grid design lift force and D-loop results[10] is adequate to cover the additional lift force resulting from the additional grid
	Rod Vibration	Decreased span length results in improved vibration characteristics and reduced rod wear
Departure from Nucleate Boiling	DNB Correlation	Addition on a grid increases mixing which increases DNB margin
Incore Flow Mixing	TDC	TDC increases as grid spacing decreases[4]

The above tabulation shows that additional design changes are not required (e.g. no new fuel assembly hold-down spring) due to the addition of a grid, and seven-grid test information can be used to assess the adequacy of the eight-grid design. Additional testing to investigate the eight-grid assembly specifically is not required.

1.5.3.1 Rod Cluster Control (RCC) Spider Tests

The 17 x 17 rod cluster control (RCC) spider (subsection 4.2.3.2) is conceptually similar to, but geometrically different from, the 15 x 15 spider. The 17 x 17 spider supports 24 rodlets (the 15 x 15 design supports 20); with no vane supporting more than two rodlets (same as 15 x 15 design). The RCC spider tests verified the structural adequacy of the design.

Spider tests have been completed. A vertical static load test approximately seven times the design dynamic load did not result in spider vane to hub joint failure. A spider was tested to 2.8 x 10^6 steps without failure. The spider loading was 110 percent of the design value for 1.8 x 10^6 cycles and 220 percent of the design loading for 1 x 10^6 cycles. Design load is 3600 lb compression and 1800 lb tension. The spring test resulted in negligible preload loss.

1.5.3.2 Grid Tests

The 17 x 17 grid (subsection 4.2.1.2) is conceptually similar but geometrically different from the 15 x 15 "R" grid. The purpose of the grid tests is to verify the structural adequacy of the grid design.

The grid tests have been completed. Test results are in agreement with pretest design values. The test results, along with fuel assembly structural test results, were factored into the seismic analysis. (See reference 9.)

1.5.3.3 Fuel Assembly Structural Tests

The 17 x 17 fuel assembly (subsection 4.2.1.2) tests were performed to determine mechanical strength and properties. The fuel assembly parameters obtained were as follows: lateral and axial stiffness, impact and internal structural damping coefficients, vibrational characteristics, and the lateral and axial impact response for postulated accident loads. The parameters obtained from the lateral dynamic tests are used for seismic analysis, while those obtained from the axial tests are incorporated in the LOCA (blowdown) accident analysis.

The fuel assembly structural tests have been completed. The fuel assembly structural test results are factored into the seismic and blowdown analyses.[9]

1.5.3.4 Guide Tube Tests

To verify the structural adequacy of the guide tubes, an extensive series of tests was conducted to determine guide tube deflection with simulated blowdown forces comparable to those expected during a LOCA and to determine the maximum acceptable deflection which ensures insertion of a control rod by free fall. Additional tests were conducted to determine fatigue strength, displacement as a function of strain, and the natural frequencies of the guide tubes for use in dynamic analyses. Refer to references 4 and 15 for a discussion of these items.

1.5.3.5 Prototype Assembly Tests

The purpose of these tests was to demonstrate that the 17 x 17 fuel assembly and control rod hardware designs perform as predicted. Two prototype assemblies were sequentially tested in order to obtain the required experimental data. A single set of control rod hardware, including driveline, was used in the tests. The fuel assemblies were subjected to flow and system conditions covering those most likely to occur in a plant during normal operation, as well as during a pump overspeed transient. Seismic testing was not included in the test sequence.

These tests were used to verify the integrated fuel assembly and RCC performance in several areas. Data obtained includes pressures and pressure drops throughout the system, hydraulic loadings on the fuel assembly and drive line, control rod drop time and stall velocity, fuel rod vibration and control rod, driveline, guide tube, and guide thimble wear during a lifetime of operation.

1.5.3.6 Departure from Nucleate Boiling (DNB)

The effect of the 17 x 17 fuel assembly geometry on the departure from nucleate boiling (DNB) heat flux has been determined experimentally and has been incorporated in a modified spacer

factor for use with the W-3 correlation. The effect of cold-wall thimble cells in the 17-x-17 geometry has also been quantified.

A similar program was conducted to quantify the DNB performance of the R-type mixing vane grid as developed for the 15-x-15 fuel assembly design.[2] [3] The results of that program were used to develop a modified spacer factor which quantifies the power capability associated with the use of the R mixing vane grid as well as the change in power capability due to the axial spacing of the grids. The modified spacer factor, along with the W-3 correlation with the cold wall factor, was shown to be applicable to cold wall thimble cells in the 15-x-15 geometry.[3]

1.5.3.7 Incore Flow Mixing

In the thermal-hydraulic design of a reactor core, the effect of mixing or turbulent energy transfer within the hot assembly is evaluated using the THINC code. The rate of turbulent energy transfer is formulated in the THINC analysis in terms of a thermal diffusion coefficient (TDC).

A program[4] to determine the proper value of TDC for the R grid vane, as used in the 15-x-15 fuel assembly design, has been completed and showed that a design value of 0.038 (for 26-in. spacing) can be used for TDC. These results also showed that TDC was independent of Reynold's number, mass velocity, pressure, and quality over the ranges tested.

A similar TDC experimental program employed a geometry typical of the 17 x 17 fuel assembly to determine the effects of the geometry on mixing and an appropriate value for TDC. A uniform axial heat flux was used. There is no analytical reason to expect that the mixing coefficient would be affected by a nonuniform axial heat flux. The THINC computer code considers the mixing in each increment along the heated length; within that increment, the heat flux is considered uniform. The tests reported by Cadek[8] indicate that there was no difference, within experimental accuracy, between a test section with a uniform flux (Pitt) and 1/2 a cosine flux (Columbia). The heat flux varied between the simulated fuel rods in the test section to create a thermal gradient in the radial direction. Using different flow rates and inlet temperatures, the TDC for the 17 x 17 geometry was determined.

Status

The TDC tests are completed and the results are reported in reference 12 and summarized in subsection 4.4.3.3.

1.5.4 LOCA HEAT TRANSFER TESTS (17 x 17)

Extensive experimental programs have been completed to determine the thermal hydraulic characteristics of 15 x 15 fuel assemblies, and to obtain experimental reflooding heat transfer data under simulated loss-of-coolant accident conditions.

Complementary experimental programs were completed with a simulated 17 x 17 assembly to determine its behavior under similar LOCA conditions. Results from the 17 x 17 programs were

compared with data from the 15 x 15 assembly test programs and were used to confirm predictions made by correlations and codes based on the 15 x 15 test results.

1.5.4.1 Blowdown Heat Transfer Testing (Formerly Titled Delayed Departure From Nucleate Boiling)

The NRC Acceptance Criteria for emergency core cooling systems (ECCS) for light-water power reactors was issued in Section 50.46 of 10 CFR 50 on December 28, 1973. It defines the basis and conservative assumptions to be used in the evaluation of the performance of emergency core cooling systems. Westinghouse believes that some of the conservatism of the criteria is associated with the manner in which transient DNB phenomena are treated in the evaluation models. Transient critical heat flux data presented at the 1972 specialists meeting of the Committee on Reactor Safety Technology (CREST) indicated that the time to DNB can be delayed under transient conditions. To demonstrate the conservatism of the ECCS evaluation models, Westinghouse initiated a program to experimentally simulate the blowdown phase of a LOCA. This testing is part of the Electric Power Research Institute (EPRI) sponsored Blowdown Heat Transfer Program, which was started early in 1976. Testing was completed in 1979. A DNB correlation will be developed by Westinghouse from these test results for use in the ECCS analyses.

Program

The program was divided into two phases. The Phase I tests started from steady state conditions, with sufficient power to maintain nucleate boiling throughout the bundle, controlled ramps of decreasing test section pressure or flow initiated DNB. By applying a series of controlled conditions, investigation of the DNB was studied over a range of qualities and flows, and at pressures relevant to a PWR blowdown.

Phase I provided separate-effects data for heat transfer correlation development.

Typical parameters used for Phase I testing are as shown:

Initial Steady State Conditions	**Nominal Valve**
Pressure	1250 to 2250 psia
Test section mass velocity	1.12 to 2.5 x 10^6lb/h-ft^2
Core inlet temperature	550 to 600°F
Maximum heat flux	306,000 to 531,000 Btu/h-ft^2
Transient Ramp Conditions	
Pressure decrease	0 to 350 psi/s and subcooled depressurization from 2250 psia
Flow decrease	0 to 100 percent/s
Inlet enthalpy	Constant

Phase II simulated PWR behavior during a LOCA to permit definition of the time delay associated with onset of DNB tests in this phase covered the large double-ended guillotine cold leg break. All tests in Phase II were also started after establishment of typical steady state operating conditions. The fluid transient was then initiated, and the rod power decay was programmed in such a manner as to simulate the actual heat input of fuel rods. The test was terminated when the heater rod temperatures reach a predetermined limit.

Typical parameters used for Phase II testing are as shown:

Initial Steady State Conditions	**Nominal Valve**
Pressure	2250 psia
Test section mass velocity	2.5×10^6 lb/h-ft^2
Inlet coolant temperature	545°F
Maximum heat flux	531,000 Btu/h-ft^2
Transient Conditions	
Simulated break	Double-ended cold leg guillotine breaks

Test Description

The experimental program was conducted in the J-loop at the Westinghouse Forest Hills Facility with a full length 5 x 5 rod bundle simulating a section of a 15 x 15 assembly to determine DNB occurrence under LOCA conditions.

The heater rod bundles used in this program were internally-heated rods, capable of a maximum power of 18.3 kW/ft, with a total power of 135 kW (for extended periods) over the 12-foot heated length of the rod. Heat was generated internally by means of a varying cross-sectional resistor which approximates a chopped cosine power distribution. Each rod was adequately instrumented with a total of 12 clad thermocouples.

1.5.4.2 Single Rod Burst Test (SRBT)

The single rod burst test (SRBT) results were used to quantify the maximum assembly flow blockage which is assumed in LOCA analyses.

Previously, single rod and multi-rod burst test (MRBT) have been completed on 15 x 15 fuel assembly rods under conditions which exist during the loss-of-coolant accident. The conclusion of these tests were that fuel rods burst in a staggered manner, so that maximum average assembly-wise flow area blockage is 55 percent during blowdown and 65 percent during reflood, based on the characteristics of the pressurized PWR fuel rod and the conservative peak clad temperature predicted during the LOCA transient.

The single rod burst test program for the 17 x 17 fuel assembly rods consisted of testing specimens at the two internal pressures and the three heating rates listed below in a steam atmosphere.

Heading Rate	Internal Pressure
(725°F to 1940°F)	psi
5°F/s	1200, 1800
25°F/s	1200, 1800
100°F/s	1200, 1800

All specimens are then heated 5°F/s from 1940°F to about 2300°F, held for a short time, and then cooled 5°F/s to 1200°F.

Metallography is done on specimens to determine the degree of wall thinning and the extent of oxygen embrittlement.

In addition, tests were run on 15 x 15 fuel assembly rods to insure reproducibility of the 1972 single rod burst test results.

Facility

The SRB tests are conducted in the Westinghouse Engineering Mechanics Laboratory in an electrically heated furnace.

Status

The single rod burst tests are complete and results are reported in reference 13. Results of initial tests showed that the LOCA behavior of 17 x 17 clad in comparison to that of 15 x 15 clad exhibited no significant differences in failure ductility. Because of the result and the geometry sealing, the flow blockage (90 percent) as determined by 15 x 15 MRBT simulation can be used for 17 x 17 fuel geometry.

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2. Motley, F. E. and Cadek, F. F., "DNB Results for New Mixing Vane Grid (R), "WCAP-7695-L, July 1972, (Westinghouse Proprietary), and WCAP-7958, October 1972.

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12. Motley, F. E., Wenzel, A. H., and Cadek, F. F., "The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP 8299, March 1974.

13. "Irradiation of 17x17 Demonstration Assemblies in Surry Units No. 1 and 2, Cycle 2," WCAP-8362, July 1974.

14. Kochirka, P., "17x17 Design Fuel Rod Behavior during Simulated Loss-of-Coolant Accident," WCAP-8289, (W Proprietary), WCAP-8290, November 1974.

15. Cooper, F.W., Jr., "17 x 17 Driveline Component Tests - Phase IB, II, III, D-Loop Drop and Deflection" WCAP-8446 (W Proprietary) and WCAP-8449 (Non-proprietary), December 1974.

1.6 MATERIAL INCORPORATED BY REFERENCE

1.6.1 WESTINGHOUSE TOPICAL REPORTS

1.6.1.1 Reports Referenced in FSAR

This section lists Westinghouse Topical Reports referenced throughout the FSAR, which provide bases for the design information presented.

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
1. PWR Staff, "Westinghouse Technical Position on Discrete Break Locations and Types for LOCA Analysis of the Primary Coolant Loop," Westinghouse Topical Report WCAP-8082, May 1973.	3.6	June 1973
2. L. T. Gesinski, "Fuel Assembly Safety Analysis for Combined Seismic and Loss-of-Coolant Accident," WCAP-7950, July 1972.	3.7, 3.9	July 1972
3. E. L. Vogeding, "Seismic Testing of Electrical and Control Equipment," WCAP-7817 and Supplements, December 1971, and WCAP-7897-L (W Proprietary).	3.7, 3.10, 7.6	January 1982
4. B.E. Olson, G. J. Bohn, "Indian Point No. 2 Primary Loop Vibration Test Program," WCAP-7662 (W Proprietary) and WCAP-7920, September 1972.	3.9	August 1973
5. G. J. Bohn, "Indian Point No. 2 Internals Mechanical Analysis for Blowdown Excitation," WCAP-7822, December 1971, and WCAP-7332-L (W Proprietary).	3.9	December 1971
6. S. Kraus, "Neutron Shielding Pads," WCAP-7870, May 1972.	1.3, 3.9, 4.2	July 1972
7. A. J. Kuenzel, "Westinghouse PWR Internals Vibration Summary 3-Loop Internals Assurance," WCAP-7765, September 1971, and WCAP-7765-L (W Proprietary).	3.9	October 1971

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
8. J. M. Hellman (Ed.), "Fuel Densification Experimental Results and Model For Reactor Operation," WCAP-8219, October 1973, and WCAP-8218, October 1973 (W Proprietary).	4.1, 4.2, 4.3, 4.4, 15.3	October 1973
9. L. Gesinski, K. Chiang, S. Nakazato, "Safety Analysis of the 17 x 17 Fuel Assembly For Combined Seismic and Loss-of-Coolant Accident," WCAP-8288, December 1973, and WCAP-8238.	4.2, 1.5	March 1973
10. W. J. Dollard, "Nuclear Fuel Division Quality Assurance Program Plan," WCAP-7800, Revision 3, November 1979.	4.2, 17C.1	October 1973
11. E. E. Demario and S. Nakazato, "Hydraulic Flow Test of the 17 x 17 Fuel Assembly," WCAP-8279, February 1974, and WCAP-8278.	1.5, 4.2, 4.4	March 1973
12. F. B. Skogen and A. F. McFarlane, "Control Procedures for Xenon-Induced X-Y Instabilities in Large PWRs" WCAP 3680-21, (EURACE-2111), February 1969.	4.3	September 1969
13. R. F. Barry, et al., "The PANDA Code," WCAP-7757, April 1967.	4.3	December 1973
14. W.C. Gangloff and W.D. Loftus, "An Evaluation of Solid State Reactor Protection in Anticipated Transients," WCAP-7706 July 1971, and WCAP-7706-L (W Proprietary) September 1971.	4.3, 7.1, 7.2, 7.3	September 1971
15. J. S. Moore, "Nuclear Design of Westinghouse PWRs with Burnable Poison Rods," WCAP-7806, December 1971.	4.3	December 1971
16. S. Altomare and R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758, September 1971, and WCAP-7213, June 1968 (W Proprietary).	4.3, 15.1, 15.2, 15.3	December 1973
17. F.E. Motley and F.F. Cakek, "DNB Test Results for New Mixing Vane Grids (R)," WCAP-7695-L, July 1972 (W Proprietary), and WCAP-7958, October 1972.	4.4, 1.5	July 1972

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
18. F.F. Cadek, "Interchannel Thermal Mixing with Mixing Vane Grids," WCAP-7667-L, May 1971 (W Proprietary), and WCAP-7755, September 1971.	4.4, 1.5	May 1971
19. F.F. Cadek, F.E. Motley, and D. P. Dominicis, "Effects of Axial Spacing on Interchannel Thermal Mixing with the (R) Mixing Vane Grid," WCAP-7941-L, June 1972 (W Proprietary) and WCAP-7755, September 1971.	4.4, 1.5	July 1972
20. L.E. Hochreiter, "Application of the THINC-IV Program to PWR Design," WCAP-8054, October 1973, (W Proprietary), and WCAP-8195, October 1974	4.4	December 1973
21. F.D. Carter, "Inlet Orificing of Open PWR Cores," WCAP-9004, January 1969 (W Proprietary), and WCAP 7836, January 1972.	4.4	March 1969
22. K.W. Hill, F.E. Motley, and F.F. Cadek, "Effect of Local Heat Flux Spikes on DNB in Non Uniform Heated Rod Bundles," WCAP-8174, August 1973 (W Proprietary), and WCAP 8202, August 1973.	4.4	December 1973
23. S. Nakazato and E.E. DeMario, "Hydraulic Flow Test of the Fuel Assembly," WCAP-8279, February 1974.	4.4, 1.5	March 1974
24. F.E. Motley and F.F. Cadek, "DNB Test Results for R Grid Thimble Cold Cells," WCAP-7958, and WCAP-7958, Addendum J.	1.5, 4.4	March 1973
25. D.H. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1, December 1971.	15.4	January 1972
26. L.E. Hochreiter, H. Chelemer, and P.T. Chu, "THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.	4.4	October 1973

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
27. F.F. Cadek, K.W. Hill, F.E. Motley, and A.H. Wenzel, "Effects of 17 x 17 Fuel Assembly Geometry on DNB," WCAP-8297, March 1974.	4.4, 1.5	April 1974
28. F.E. Motley, A.H. Wenzel, and F.F. Cadek, "The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8299, March 1974.	4.4, 1.5	March 1974
29. F.E. Motley and F.F. Cadek, "Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB," WCAP-8030, December 1972.	4.4	January 1973
30. W.S. Hazelton, "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7735, August 1971, and WCAP-7477-L (W Proprietary)	5.4	August 1971
31. K. Cooper, R.M. Starek and V. Miselis, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972.	5.2	June 1972
32. W.S. Hazelton, S.L. Anderson and E.E. Yanichko, "Basis for Heatup and Cooldown Limit Curves," WCAP-7924, August 1972.	5.2	August 1972
33. F. Bordelon and A. Nahavandi, "A Space Dependent Loss-of-Coolant Accident and Transient Analysis for PWR Systems (SATAN Digital Computer Code)," WCAP-7854, January 1972.	5.2	-
34. J. Locante and E.G. Igne, "Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope)," WCAP-7744, Volume I, August 1971 and WCAP-7410-L Volume I, (W Proprietary).	3.11, 6.3, 7.3	September 1971
35. M.J. Bell, J.E. Bulkowski. L.F. Picone, "Investigation of Chemical Additives for Reactor Containment Sprays," WCAP-7154, March 1968 (W Proprietary).	6A	February 1968

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
36. R. Bartholomew and J. Lipchak, "Test Report, Nuclear Instrumentation System Isolation Amplifier," WCAP-7819, Revision 1, January 1972.	7.2	January 1972
37. I. Garber, "Isolation Tests Process Instrumentation Isolation Amplifier Westinghouse Computer and Instrumentation Division Nucana 7300 Series," WCAP-7862, August 1972.	7.2	September 1972
38. D.N. Katz, "Solid State Logic Protection System Description," WCAP-7672, June 1971.	7.1, 7.2, 7.3	May 1971
39. J.T. Haller, "Engineered Safeguards Final Device or Actuator Testing," WCAP-7705, May 1972.	7.3	March 1974
40. A.E. Blanchard and D.N. Katz, "Solid State Rod Control System, Full Length," WCAP-7778, December 1971, and WCAP-9012-L (W Proprietary).	7.7	December 1971
41. J.B. Reid, "Process Instrumentation for Westinghouse Nuclear Steam Supply System," WCAP-7913, January 1973.	7.1, 7.2, 7.3	March 1973
42. A.E. Blanchard, "Rod Position Monitoring," WCAP-7571, March 1971.	7.7	April 1971
43. A.E. Blanchard, "Part Length Rod Control System," WCAP-7406, March 1971.	7.7	April 1971
44. "Source Term Data for W PWRs," Westinghouse Electric Corporation, Pittsburgh, PA, WCAP-8253, April 1970.	11A.3	June 1974
45. D.H. Risher Jr., and R.F. Barry, "TWINKLE-A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-8028 January 1973 (W Proprietary), WCAP-7979 November 1972.	15.1, 15.4, 15.2, 15.3	January 1970
46. F.M. Bordelon, "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP-7973, January 1973.	15.1	January 1973

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
47. D.B. Fairbrother and H.G. Hargrove, "WIT-6 Reactor Transient Analysis Computer Program Description," WCAP-7980, November 1972.	15.1, 15.2	September 1973
48. J.M. Geets and R. Salvatori, "Long Term Transient Analysis Program for PWRs (BLKOUT Code)," WCAP-7898, June 1972 and WCAP-7501.	15.1, 15.2	September 1972
49. T.W.T. Burnett, C.J. McIntyre, J.C. Buker, R.P. Rose, "LOFTRAN Code Description," WCAP-7907, June 1972 and WCAP-7877, October 1972.	15.2, 15.3	October 1972
50. C. Hunin, "FACTRAN, a Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972 and WCAP-7337.	15.2, 15.3	September 1972
51. J.M. Geets, "MARVEL- A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909, June 1972 and WCAP-7635.	15.1, 15.2, 15.4	October 1972
52. M.A. Mangan, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971.	15.2	June 1972
53. V.J. Esposito, K. Kesavan, B.A. Maul, "WFLASH-A FORTRAN-IV Computer Program for Simulation of Transients in a Multiloop PWR," WCAP-8261 Rev. 1, July 1974 and WCAP-8200, Rev. 2.	15.3	July 1974
54. F.M. Bordelon, et al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8305, June 1974.	15.3, 15.4	July 1974
55. R.D. Kelly, et al., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8171, June 1974 and WCAP-8170.	15.4	July 1974
56. F.M. Bordelon and E.T. Murphy, "Containment Pressure Analysis Code (COCO)," WCAP-8326, June 1974 and WCAP-8327.	15.4	July 1974

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
57. T.L. Buterbaugh, W.J. Johnson and S.D. Kopelic, "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8356, July 1974 and WCAP-8340.	15.3, 15.4	August 1974
58. F.M. Bordelon, W. Massie and T.A. Zordan, "Westinghouse ECCS Evaluation Model - Summary" WCAP-8339, July 1974.	15.3, 15.4	July 1974
59. F.M. Bordelon, et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, June 1974.	15.4	July 1974
60. K.D. Shepard, S. Cerni, J.R. Reavis, "An Evaluation of Fuel Rod Bowing," WCAP-8346, May 1974.	4.4	May 1974
61. K.W. Hill, F.E. Motley, F.F. Cadek, "Effect of Bowed Rods on DNB," WCAP-8176 (W Proprietary) and WCAP-8323.	4.4	May 1974
62. H. Chelemer, J. Weisman, and L.S. Tong, "Subchannel Thermal Analysis of Rod Bundle Cores," WCAP-7015 Rev. 1, January 1969.	4.4	February 1969
63. Wilson, J.F., "Electric Hydrogen Recombiner IEEE 323-1974 Qualification," WCAP-7709-6 Supplement 6 (Proprietary) October 1976, WCAP-7820 Supplement 6 (Nonproprietary), October 1976.	3.11, 6.2	October 1976
64. Lee, H. "Prediction of the of the Flow Induced Vibration Reactor Internals by Scale Model Tests," WCAP-8303 (W Proprietary) and WCAP-8317, May 1974.	3.9	May 1974
65. P. Kochirka "17 x 17 Design Fuel Rod Behavior During Simulated Loss-of-Coolant Accident," WCAP-8289 (W Proprietary) WCAP-8290 November 1974.	1.5	November 1974
66. R.A. George, Y.C. Lee, G.H. Eng, "Revised Clad Flattening Model," WCAP-8377 (Westinghouse Proprietary) and WCAP-8381 (Nonproprietary), July 1974	4.2	August 1974

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
67. T.M. Burke, C.E. Meyer, and J. Shefcheck, "Analysis of Data from the Zion (Unit 1) THINC Verification," WCAP-8453 (Westinghouse Proprietary) and WCAP-8454 (Nonproprietary), December 1974.	4.4	January 1975
68. K.M. Vashi, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, April 1974.	Appendix 3L, 3M.6	June 1975
69. "Westinghouse ECCS Evaluation Model - October 1975 Version," WCAP-8623, November 1975 (Nonproprietary).	15.4	November 1975
70. Julian, H.V., Tabone, C.J., Thompson, C.M., "Westinghouse ECCS-Three Loop Plant (17x17) Sensitivity Studies," WCAP-8853, October 1976, (Non-proprietary).	15.4	October 1976
71. Vogeding, E.L., "Seismic Testing of Electrical and Control Equipment for Low Seismic Plants," WCAP-7817 Supplement 7, September 1976.	3.10, 7.6	October 1976
72. Figenbaum, E., "Seismic Testing of Electrical and Control Equipment for High Seismic Plants," WCAP-7821 Supplement 2, Addendum 1, November 1975.		October 1976
73. Jarecki, S.J., Coslow, B.J., Croasdaile, T.R., Lipchak, J.B., "Seismic Operability Demonstration Testing of the Nuclear Instrumentation System Bistable Amplifier," WCAP-8830 (Proprietary) October 1976.	Ch. 3	November 1976
74. Land, R.E., "Mass and Energy Release Following Main Steam Ruptures," WCAP-8822 (Proprietary) September 1976 and WCAP-8860 (Nonproprietary) September 1976.	6.2	September 1976
75. F. F. Cadek, F. E. Motley, and D. P. Dominicis, "Effect of Axial Spacing on Inter-channel Thermal Mixing with R Mixing Vane Grid," WCAP-7941-L, June 1972, (Westing-house Proprietary), and WCAP-7959, October 1972.	1.5, 4.4	July 1972

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
76. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Nonuniform Heat Flux Distribution," J. Nucl. Energy, 21 pp. 241-248 (1967).	1.5	July 1972
77. R. H. Wilson, L. J. Stanek, J. S. Gellerstedt, and R. A. Lee, "Critical Heat Flux in a Nonuniformly Heated Rod Bundle," in Two-Phase Flow and Heat Transfer in Rod Bundles, pp. 56-7, ASME, New York, November 1969.	1.5	July 1972
78. E. R. Rosal, et al., "Rod Bundle Flux Tests and Data," WCAP-7411, December 1969 (W Proprietary), and WCAP-7813, December 1971.	1.5	July 1970
79. F. F. Cadek, "Interchannel Thermal Mixing with Mixing Vane Grids," WCAP-7667-L, May 1971 (W Proprietary), and WCAP-7755, September 1971.	1.5, 4.4	July 1970
80. L. Gesinski, D. Chiang, and S. Nakazato, "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8288, December 1973.	1.5, 4.2	-
81. "Irradiation of 17 x 17 Demonstration Assemblies in Surry Units No. 1 and 2, Cycle 2," WCAP-8362, July 1974.	1.5	-
82. F. W. Cooper, Jr, "17 x 17 Driveline Component Tests-Phase IB, II, III, D-Loop Drop and Deflection," WCAP-8446 (W Proprietary) and WCAP-8449 (Non-proprietary), December 1974.	1.5	January 1975
83. O. Linsuain, et al., “Westinghouse Performance Analysis and Design Model (PAD5),” WCAP-17642-P-A, Revision 1, November 2017.	4.2, 4.3, 4.5	October 2013

1.6.2 BECHTEL POWER CORPORATION TOPICAL REPORTS

Report No.	Title
1. B-TOP-3	"Design Criteria for Nuclear Power Plants Against Tornadoes," March 1970

Report No.	Title
2. BP-TOP-1	"Seismic Analysis of Piping System," Revision 1, February 1974
3. BC-TOP-4	"Seismic Analysis of Structures and Equipment for Nuclear Power Plants," Revision 1, September 1972
4. BC-TOP-5	"Prestressed Concrete Nuclear Reactor Containment Structures," Revision 1, December 1972
5. BC-TOP-7	"Full Scale Buttress Test for Prestressed Nuclear Containment Structures," Revision 0, September 1972
6. BC-TOP-8	"Tendon End Anchor Reinforcement Test," Revision 0, September 1972
7. BC-TOP-1	"Containment Building, Liner Plate Design Report," Revision 1, December 1972
8. BN-TOP-1	"Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants," Revision 1, November 1972.
9. BC-TOP-9A	"Design of Structures for Missile Protection," Revision 2, September 1974

1.6.3 GENERAL REPORTS

Material used for initial design and background information and not required for evaluation of the Farley Nuclear Plant are as follows:

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
1. J.J. Szyslowski and R. Salvatori, "Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System," WCAP-7503, Revision 1, February 1972.	3.6, 5.2	February 1972
2. J.S. Moore, "Westinghouse PWR Core Behavior Following Loss-of-Coolant Accident," WCAP-7422-L, January 1970, (W Proprietary), and WCAP-7422 August 1971.	3.9	January 1970
3. S. Fabic, "Loss-of-Coolant Analysis: Comparison Between BLODWN-2 Code Results and Test Data," WCAP-7401, November 1969.	3.9	February 1970

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
4. V.J. Placido, R.E. Schreiber, J. Skaritka, "Operational Experience – Westinghouse Cores," WCAP-8183, October 1973.	4.2	July 1974
5. J.E. Outzs, "Plant Startup Test Report, H.B. Robinson Unit No.2," WCAP-7844, January 1972.	4.3	January 1972
6. J.S. Moore, "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7810, December 1971.	4.3, 15.4	December 1971
7. J.S. Moore, "Power Distribution Control of Westinghouse PWRs," WCAP-7208, September 1968 (W Proprietary), and WCAP-7811, December 1971.	4.3	October 1968
8. G.C. Poncelet, "LASER-A Depletion Program for Lattice Calculations Based on MUFT and THERMUS," WCAP-2048, July 1962.	4.3	July 1972
9. J.E. Olhoeft, "The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements," WCAP-2048, July 1962.	4.3	July 1962
10. R.J. Nodvik, et al., "Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium through Curium," WCAP-6086, August 1969.	4.3	August 1969
11. F.B. Skogen and A.F. McFarlane, "Xenon-Induced Spatial Instabilities in Three Dimensions," WCAP-3680-22 (EURAEC-1976), March 1968.	4.3	September 1969
12. R.F. Barry, "LEOPARD, a Spectrum-Dependent, Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September 1963.	4.3, 15.1, 15.3	September 1963
13. A.F. McFarlane, "Core Power Capability in Westinghouse PWRs WCAP-7267-L, October 1964 (W Proprietary), and WCAP-7809, December 1971.	4.3	October 1969

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
14. A.F. McFarlane, "Power Peaking Factors" WCAP-7912-L, March 1972 (W Proprietary) and WCAP-7912, March 1972.	4.3, 4.4	March 1973
15. J.O. Cermak, "Pressurized Water Reactor pH-Reactivity Effect," Final Report, WCAP-3696-8 (EURAEC-2074), October 1968.	4.3	October 1968
16. G.C. Poncelet and A.M. Christie, "Xenon-Induced Spatial Instabilities in Large PWRs," WCAP-3680-20, (EURAEC-1974), March 1968.	4.3	March 1968
17. J.C. Lee, "Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor," WCAP-7964, June 1971.	4.3	June 1971
18. C.J. Kubit, "Safety Related Research and Development for Westinghouse PWRs, Program Summaries, Fall 1972," WCAP-8004, December 1972.	4.3, 4.2	January 1973
19. J.A. Christensen, R.J. Allio, and A. Biancheria, "Melting Point of Irradiated UO_2," WCAP-6065.	4.2, 4.4	February 1965
20. C.G. Poncelet, "Burnup Physics of Heterogeneous Reactor Lattices," WCAP-6069, June 1965.	4.4	June 1965
21. J. Shefcheck, "Application of the THINC Program to PWR Design," WCAP-7359-L, August 1969 (W Proprietary), and WCAP-7838, January 1972.	4.4	September 1969
22. E.H. Novendstern and R.O. Sandberg, "Single Phase Local Boiling and Bulk Boiling Pressure Drop Correlations," WCAP-2850, April 1966 (W Proprietary), and WCAP-7916, June 1972.	4.4	April 1966
23. G. Hetsroni, "Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, June 1964.	4.4	June 1964
24. R.L. Rosenthal, "An Experimental Investigation of the Effect of Open Channel Flow on Thermal-Hydrodynamic Flow Instability," WCAP-7966, December 1972.	4.4	December 1972

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
25. M.G. Balfour, J.A. Christensen, and H.M. Farrari, "In-Pile Measurement of UO_2, Thermal Conductivity," WCAP-2923, March 1966.	4.4	March 1966
26. J.A. Nay, "Process Instrumentation for Westinghouse Nuclear Steam Supply Systems," WCAP-7671, April 1971, and WCAP-7547-L, March 1971 (W Proprietary).	5.2	May 1971
27. W.O. Shabbits, "Dynamics Fracture Toughness Properties of Heavy Section A533 Grade B, Class 1, Steel Plate," WCAP-7623, December 1970.	4.2	January 1971
28. W.S. Hazelton, "Sensitized Stainless Steel in Westinghouse Heavy Section A533 Grade B Class 1 Steel Plate," WCAP-7623, December 1970.	5.2	December 1970
29. L.F. Picone, "Evaluation of Protective Coatings for Use in Reactor Containment," WCAP-7198, April 1968 (W Proprietary).	6A.8, 6A.9	April 1969
30. T.W.T. Burnett, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," WCAP-7306, April 1969.	7.1, 7.2, 15.4	April 1969
31. J.B. Lipchak and R.A. Stokes, "Nuclear Instrumentation System," WCAP-7669, April 1971. (Replaced by WCAP-8255).	7.1, 7.2, 7.7	May 1971
32. J.J. Loving, "In-Core Instrumentation (Flux-Mapping System and Thermocouples)," WCAP 7607, July 1971.	7.7	July 1971
33. A.E. Blanchard and Calpin, J.E. "Digital Rod Position Indication," WCAP-8014, December 1974.	7.7	March 1974
34. W.C. Gangloff, "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486, May 1971, and WCAP-7486-L.	15.2	May 1971
35. D.D. Malinowski, et al., "Radiological Consequences of a Fuel Handling Accident," WCAP-7878, December 1971, and WCAP-7518-L. (W Proprietary).	15.4	December 1971

REPORT TITLE	SECTION REFERENCE	NRC SUBMITTAL
36. F.F. Cadek, et al., PWR FLECT (Full Length Emergency Cooling Heat Transfer), Final Report," WCAP-7665, April 1971.	15.4, 15A	July 1971
37. R.J. French, "Indian Point Unit No. 2 Rod Ejection Analysis," WCAP-2940 May 1966.	15.4	June 1966
38. J.S. Moore, "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7810, December 1971.	15.4	December 1971
39. "Safety Related Research and Development for Westinghouse Pressurized Water Reactors," Program Summaries, Spring 1974, WCAP-8353, September 1974.	1.5	September 1974
40. "Anticipated Transients Without Reactor Trip in Westinghouse Pressurized Water Reactors," WCAP-8096, April 1973.	Appendix 3A	April 1973
41. F.M. Bordelon, "Small Loss of Coolant Accident Analysis for PWR Systems (SLAP Digital Computer Code)," WCAP-7983, November 1971.	15.3	September 1974

1.6.4 CROSS REFERENCE OF ENGINEERING DRAWINGS

This table lists FSAR project drawing numbers and their affiliated titles that were associated with FSAR figure numbers which were removed in Revision 13. The project drawings contained the same information, but typically provided much more detail than the original FSAR figures. Any project drawings added to the FSAR will be added to this table with their appropriate drawing title.

TABLE 1.6-1 (SHEET 1 OF 28)

Drawing Number	Drawing Title
A-177048	Unit 1 Relay Setting Sheets
A-207048	Unit 2 Relay Setting Sheets
A-508650	Unit 1 Fire Zone Data Sheets (Auxiliary Building, Turbine Building, Containment)
A-508651	Unit 1 & Unit 2 Fire Zone Data Sheets (Diesel Gen. Bldg., SWIS, HVAC Rm. El. 265, RWIS, Cable Tunnels A/B, Control Rm. Area, Low Voltage Switchyard)
A-509018	Unit 2 Fire Zone Data Sheets (Auxiliary Building, Turbine Building, Containment)
B-356750	Unit 2, 2RE-29B, Low, Mid and High Range Gas Radiation Monitor
B-356751	Unit 2, 2RE-29C, Particulate, Iodine and Low Range Gas Radiation Monitor
B-507156	Unit 1, 1RE-29B, Low, Mid, and High Range Gas Radiation Monitor
B-507157	Unit 1, 1RE-29C, Particulate, Iodine, and Low Range Gas Radiation Monitor
C-177012	Single-Line Protection and Metering 600-V Load Center 1F
C-177043	Single-Line Protection and Metering 4160-V Bus 1K
C-177044	Single-Line Protection and Metering 4160-V Bus 1L
C-177118	Interlock Schematic Station Service Transformer 1F
C-177119	Interlock Schematic Component Cooling Water Pump 1B
C-177120	Interlock Schematic HHSI Pump 1B
C-177121	Interlock Schematic S.W. Pump 1C
C-177133	Interlock Schematic Battery Charger 1C

TABLE 1.6-1 (SHEET 2 OF 28)

Drawing Number	Drawing Title
D-170060	Diesel Generator Fuel Oil System
D-170064	Condenser Vacuum System
D-170084	Site Plot Plan
D-170110	Well Water Storage and Supply System
D-170114, Sh. 1	Main Steam System
D-170114, Sh. 2	Main Steam System
D-170117, Sh. 1	Condensate and Feedwater System
D-170117, Sh. 2	Condensate and Feedwater System
D-170117, Sh. 3	Condensate and Feedwater System
D-170117, Sh. 4	Condensate and Feedwater System
D-170119, Sh. 1	Service Water System
D-170119, Sh. 2	Service Water System
D-170119, Sh. 3	Diesel Generator Cooling Water System Unit 1
D-170119, Sh. 6	River Water System
D-170119, Sh. 7	River Water System
D-170119, Sh. 9	Circulating Water System
D-170119, Sh. 10	Circulating Water System
D-170127	Potable and Sanitary Water System
D-170131, Sh. 1	Compressed Air System; Service Air Unit 1
D-170131, Sh. 2	Compressed Air System; Service Air Unit 1
D-170180, Sh. 1	Piping - Waste Water to River
D-170210, Sh. 1	Diesel Generator Fuel Oil System Physical Layout

TABLE 1.6-1 (SHEET 3 OF 28)

Drawing Number	Drawing Title
D-170211	Diesel Generator Fuel Oil System Physical Layout
D-170330	River Water Intake Structure Heating and Ventilation System
D-170331	River Water Intake Structure Heating and Ventilation System
D-170332	Service Water Intake Structure Heating and Ventilation System
D-170333	Service Water Intake Structure Heating and Ventilation System
D-170336	Diesel Generator Building Heating and Ventilation System Floor Plan
D-170337	Diesel Generator Building Heating and Ventilation System Roof Plan
D-170338, Sh. 1	Diesel Generator Building Heating and Ventilation System Sections
D-170338, Sh. 2	Diesel Generator Building Heating and Ventilation System Sections
D-170339	Diesel Generator Building Heating and Ventilation System Details and Design Data
D-170353	Suppression and Detection Annunciator List - Unit 1
D-170357, Sh. 1	Diesel Generator Fuel Oil System Physical Layout
D-170357, Sh. 2	Diesel Generator Fuel Oil System Physical Layout
D-170806	Diesel Generator Air System (1-2A, 1B, 2B)
D-170807	Diesel Generator Air Start System (1C, 2C)
D-171417	Outdoor Concrete Pond Fill Discharge Structure
D-171419	Outdoor Concrete Recirculating Water Discharge Structure
D-171426	Storm Drainage

TABLE 1.6-1 (SHEET 4 OF 28)

Drawing Number	Drawing Title
D-171488	Site Grades
D-172545	Single Line, Cable & Conn. Diag.- MCC 1X & 1Y 120/208 VAC Dist. Cabinets
D-172708, Sh. 1	Single Line, Cable & Conn. Diag.- D.C. Distribution - Train B - Service Water
D-172711, Sh. 1	Single Line, Cable & Conn. Diag.- D.C. Distribution - Turb. Bldg. Battery
D-173000	Transformer Physical Arrangement
D-175000, Sh. 1	Chemical Injection System
D-175000, Sh. 2	Chemical Injection System (Hydrazine and Amine P&ID)
D-175001	Access Control Areas
D-175002, Sh. 1	Unit 1 Component Cooling Water System
D-175002, Sh. 2	Unit 1 Component Cooling Water System
D-175002, Sh. 3	Unit 1 Component Cooling Water System
D-175003, Sh. 1	Unit 1 Service Water System Inside Containment and Auxiliary Building
D-175003, Sh. 2	Unit 1 Service Water System Inside Containment and Auxiliary Building
D-175003, Sh. 3	Unit 1 Service Water System Inside Containment and Auxiliary Building
D-175003, Sh. 4	Unit 1 Service Water System Inside Containment and Auxiliary Building
D-175004, Sh. 1	Unit 1 Containment and Auxiliary Building
D-175004, Sh. 2	Unit 1 Containment and Auxiliary Building
D-175005	Unit 1 Auxiliary Building Nonradioactive Drains

TABLE 1.6-1 (SHEET 5 OF 28)

Drawing Number	Drawing Title
D-175007	Unit 1 Auxiliary Feedwater System
D-175009, Sh. 1	Sampling System
D-175009, Sh. 2	Sampling System
D-175009, Sh. 3	Sampling System
D-175010, Sh. 1	Containment Cooling and Purge System - Unit 1
D-175010, Sh. 2	Containment Cooling and Purge System - Unit 1
D-175011, Sh. 1	Radwaste Area HVAC System
D-175011, Sh. 2	Radwaste Area HVAC System
D-175011, Sh. 3	Radwaste Area HVAC System
D-175012	Control Room and Computer Room HVAC and Filtration System
D-175014, Sh. 1	Nonradioactive Area and Electrical Equipment Rooms HVAC System
D-175014, Sh. 2	Nonradioactive Area and Electrical Equipment Rooms HVAC System
D-175016, Sh. 1	Fluid System Symbols
D-175016, Sh. 2	Fluid System Symbols
D-175016, Sh. 3	Standard P&ID Legend
D-175019	Postaccident Containment Combustible Gas Control System - Unit 1
D-175022	Penetration Filtration System
D-175027	Turbine Building Ventilating, Air Conditioning, and Filtration System
D-175031, Sh. 1	Turbine Building Chilled Water System
D-175031, Sh. 2	Turbine Building Chilled Water System

TABLE 1.6-1 (SHEET 6 OF 28)

Drawing Number	Drawing Title
D-175031, Sh. 3	Turbine Building Chilled Water System
D-175033, Sh. 1	Main Steam System
D-175033, Sh. 2	Main Steam System
D-175034, Sh. 1	Compressed Air System; Service Air Unit 1
D-175034, Sh. 2	Compressed Air System; Service Air Unit 1
D-175034, Sh. 3	Compressed Air System; Service Air Unit 1
D-175035, Sh. 1	Service Air
D-175035, Sh. 2	Service Air
D-175036	Reactor Makeup Water System
D-175037, Sh. 1	Unit 1 Reactor Coolant System
D-175037, Sh. 2	Reactor Coolant System - Unit 1
D-175037, Sh. 3	Reactor Coolant System - Unit 1
D-175038, Sh. 1	Safety Injection System - Unit 1
D-175038, Sh. 2	Safety Injection System
D-175038, Sh. 3	Safety Injection System (Containment Spray) Unit 1
D-175039, Sh. 1	Chemical and Volume Control System
D-175039, Sh. 2	Chemical and Volume Control System
D-175039, Sh. 3	Chemical and Volume Control System
D-175039, Sh. 4	Chemical and Volume Control System
D-175039, Sh. 5	Chemical and Volume Control System
D-175039, Sh. 6	Chemical and Volume Control System
D-175039, Sh. 7	Chemical and Volume Control System
D-175040	Boron Thermal Regeneration System

TABLE 1.6-1 (SHEET 7 OF 28)

Drawing Number	Drawing Title
D-175041	Unit 1 Residual Heat Removal System
D-175042, Sh. 1	Waste Processing System Flow Diagram
D-175042, Sh. 2	Waste Processing System Flow Diagram
D-175042, Sh. 3	Waste Processing System Flow Diagram
D-175042, Sh. 4	Waste Processing System Flow Diagram
D-175042, Sh. 5	Waste Processing System Flow Diagram
D-175042, Sh. 6	Waste Processing System Flow Diagram
D-175042, Sh. 11	Waste Processing System Flow Diagram
D-175042, Sh. 12	Waste Processing System Flow Diagram
D-175045	Spent-Fuel Pool Ventilation System
D-175047, Sh. 1	Demineralized Water System
D-175047, Sh. 2	Demineralized Water System
D-175071, Sh. 1	Steam Generator Blowdown Processing System
D-175071, Sh. 2	Steam Generator Blowdown Processing System
D-175071, Sh. 3	Steam Generator Blowdown Processing System
D-175073	Condensate and Feedwater System
D-175200	Containment Floor Plan at El 105 ft-6 in.
D-175378	Containment Spray Headers - Unit 1
D-175379	Containment Spray Headers - Unit 1
D-175380	Containment Spray Headers - Sections and Details - Unit 1
D-175381	Containment Spray Headers - Sections and Details - Unit 1

TABLE 1.6-1 (SHEET 8 OF 28)

Drawing Number	Drawing Title
D-175491	Containment Spray Headers -Details - Unit 1
D-175511	Safe Load Path, Reactor Head – Unit 1
D-175512	Safe Load Path, Upper Internals –Unit 1
D-175513	Safe Load Path, Reactor Coolant Pumps –Unit 1
D-175515, Sh. 1	Safe Load Path, Polar Crane Load Block – Unit 1
D-175515, Sh. 2	Safe Load Path, Polar Crane Load Block – Unit 1
D-176002	Auxiliary Building Floor Plan El 83 ft-0 in. and 77 ft-0 in. - Unit 1
D-176003	Containment and Auxiliary Building Floor Plan El 100 ft-0 in. and 105 ft-6 in. - Unit 1
D-176004	Containment and Auxiliary Building Floor Plan El 121 ft-0 in. and 129 ft-0 in. - Unit 1
D-176005	Containment and Auxiliary Building Floor Plan El 139 ft-0 in. - Unit 1
D-176006	Containment and Auxiliary Building Floor Plan El 155 ft-0 in. and 165 ft-0 in. - Unit 1
D-176007	Containment and Auxiliary Building Roof Plan - Unit 1
D-176035	Unit 1 Radiation Zones and Controlled Access Floor Plan - El 83 ft-0 in. and 77 ft-0 in.
D-176036	Unit 1 Radiation Zones and Controlled Access Floor Plan - El 100 ft-0 in. and 105 ft-6 in.
D-176037	Unit 1 Radiation Zones and Controlled Access Floor Plan - El 121 ft-0 in. and 129 ft-0 in.
D-176038	Unit 1 Radiation Zones and Controlled Access Floor Plan - El 139 ft-0 in.
D-176039	Unit 1 Radiation Zones and Controlled Access Floor Plan - El 155 ft-0 in.
D-176040	Unit 1 Radiation Zones and Controlled Access Floor

TABLE 1.6-1 (SHEET 9 OF 28)

Drawing Number	Drawing Title
	Plan
D-176041	Unit 1 Tabulation of Radiation Zones
D-176042	Unit 1 Tabulation of Radiation Zones
D-176043	Unit 1 Tabulation of Radiation Zones
D-176075	Unit 1 Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 83 ft-0 in. and 77 ft-0 in.
D-176076	Unit 1 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 100 ft-0 in. and 105 ft-0 in.
D-176077	Unit 1 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 121 ft-0 in. and 129 ft-0 in.
D-176078	Unit 1 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 139 ft-0 in.
D-176079	Unit 1 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 155 ft-0 in.
D-176107	Incore Instrumentation Tube Supports Structural Steel Containment - Unit 1
D-176145	Containment Prestressing Requirements Typical Details Unit 1
D-176151	Containment Penetrations
D-176238	Cross-Over Support Details Structural Steel - Containment - Unit 1
D-176275	Reactor Primary Shielding Penetrations Plan and Sections Containment - Unit 1
D-176277	Reactor Cavity Liner Plate - Unit 1
D-176278	Restriction Ring Bulkhead at Reactor Cavity - Thermal Movement Diagram - Unit 1

TABLE 1.6-1 (SHEET 10 OF 28)

Drawing Number	Drawing Title
D-176279	Reactor Cavity Liner Plate - Unit 1
D-176900	Site Boring Plan
D-176901	Plant Boring Plan
D-176902	Storage Pond Boring plan
D-176920	Generalized Soil Profiles
D-176921	Generalized Soil Profiles
D-176922	Generalized Soil Profiles
D-176923	Generalized Soil Profiles
D-176924	Generalized Soil Profiles
D-176925	Generalized Soil Profiles
D-176926	Generalized Soil Profiles
D-176927	Generalized Soil Profiles
D-176928	Generalized Soil Profiles
D-176929	General1zed Soil Profiles
D-176930	Generalized Soil Profiles
D-176931	Generalized Soil Properties
D-176932	Generalized Soil Properties
D-176933	Generalized Soil Properties
D-176934	Generalized Soil Properties
D-176939	Storage Pond Dam and Dike Instrumentation Schedule
D-176940	Boring Logs (L-2-2-1, 599)
D-176941	Boring Logs

TABLE 1.6-1 (SHEET 11 OF 28)

Drawing Number	Drawing Title
D-176942	Boring Logs
D-176943	Boring Logs
D-176944	Boring Logs
D-176945	Boring Logs
D-176946	Boring Logs
D-176947	Boring Logs
D-176948	Boring Logs
D-176949	Boring Logs
D-176950	Boring Logs
D-176951	Boring Logs
D-176952	Boring Logs
D-176953	Boring Logs
D-176954	Boring Logs
D-176955	Boring Logs
D-176956	Boring Logs
D-176957	Boring Logs
D-176958	Boring Logs
D-176959	Boring Logs
D-176960	Boring Logs
D-176961	Boring Logs
D-176962	Boring Logs
D-176963	Boring Logs

TABLE 1.6-1 (SHEET 12 OF 28)

Drawing Number	Drawing Title
D-176964	Boring Logs
D-176965	Boring Logs
D-176966	Boring Logs
D-176967	Boring Logs
D-176968	Boring Logs
D-176969	Boring Logs
D-176970	Boring Logs
D-176971	Boring Logs
D-176972	Boring Logs
D-176973	Boring Logs
D-176974	Boring Logs
D-176975	Boring Logs
D-176976	Boring Logs
D-176977	Boring Logs
D-176978	Boring Logs
D-176979	Boring Logs
D-176980	Storage Pond General Arrangement and Stripping Plan
D-176981	Storage Pond Drainage Plan, Sections and Details
D-176982	Storage Pond Dam and Dike Instrumentation Details
D-176983	Storage Pond Dam Excavation Plan
D-176984	Storage Pond Dam And Dike Sections
D-176985	Storage Pond Dam And Dike Sections

TABLE 1.6-1 (SHEET 13 OF 28)

Drawing Number	Drawing Title
D-176986	Storage Pond Dam And Dike Sections
D-176987	Storage Pond Dam And Dike Sections
D-176988	Storage Pond Dam And Dike Sections
D-176989	Storage Pond Dam And Dike Sections
D-176990	Refractive Geophysical Lines
D-176991	Downhole Geophysical Survey
D-176994	Storage Pond Dam And Dike Centerline Profile
D-176995	Storage Pond Dam And Dike Centerline Profile
D-176996	Boring Logs
D-176997	Storage Pond Dam Excavation Profile and Cross-Sections
D-177000	Unit 1 Single-Line Electrical Auxiliary System (Normal)
D-177001	Unit 1 Single-Line Electrical Auxiliary System (Emergency)
D-177005	Unit 1 Single-Line Protection and Metering 4160-V Bus 1F
D-177006	Unit 1 Single-Line Protection and Metering 4160-V Bus 1G
D-177007	Unit 1 Single-Line Protection and Metering 600-V Load Center 1A
D-177009	Unit 1 Single-Line Protection and Metering 600-V Load Center 1C
D-177010	Unit 1 Single-Line Protection and Metering 600-V Load Center 1D
D-177011	Unit 1 Single-Line Protection and Metering 600-V Load Center 1E

TABLE 1.6-1 (SHEET 14 OF 28)

Drawing Number	Drawing Title
D-177014	Unit 1 Single-Line Protection and Metering 600-V Load Center 1H
D-177015	Unit 1 Single-Line Protection and Metering 600-V Load Center 1J
D-177024	Unit 1 Single-Line 120 V-ac Vital and Regulated System A
D-177025	Unit 1 Single-Line 120 V-ac Vital and Regulated System B
D-177027	Unit 1 Single-Line Protection and Metering 4160-V Bus 1J
D-177032	Unit 1 Logic Diagram -Diesel 1B Auto Start and Loading
D-177033	Unit 1 Logic Diagram -Diesel 1-2A Auto Start and Loading
D-177036	Unit 1 Logic Diagram -Diesel 1C Auto Start and Loading
D-177037	Unit 1 Logic Diagram -Diesel 2C Auto Start and Loading
D-177038	Single Line Miscellaneous Panel Boards for Emergency Power
D-177045	Unit 1 Single-Line Protection and Metering 600-V Load Center 1K
D-177046	Unit 1 Single-Line Protection and Metering 600-V Load Center 1L
D-177082	Unit 1 Single-Line dc Distribution System 1A
D-177083	Unit 1 Single-Line dc Distribution System 1B
D-177122	Interlock Schematic 600-V Bus 1A
D-177155	Elem. Diagram 4160-V Bus 1F Incoming Startup Transformer 1A

TABLE 1.6-1 (SHEET 15 OF 28)

Drawing Number	Drawing Title
D-177161	Elem. Diagram 4160-V Bus 1F Incoming Startup Transformer 1B
D-177168	Elem. Diagram 4160-V Bus 1G Incoming Startup Transformer 1A
D-177169	Elem. Diagram 4160-V Bus 1G Incoming Startup Transformer 1B
D-177185	Unit 1 Elementary Diagram Component Cooling Water Pump 1B - Train “A”
D-177187	Unit 1 Elementary Diagram Component Cooling Water Pump 1B - Train “B”
D-177331	Unit 1 Communication Layout Auxiliary Building and Containment
D-177334, Sh. 1	Unit 1 Communication Layout Auxiliary Building
D-177334, Sh. 2	Unit 1 Communication Layout Auxiliary Building
D-177334, Sh. 3	Unit 1 Communication Layout Auxiliary Building
D-177335	Communication Layout River Water, Diesel Building, and Service Water
D-177336	Unit 1 Communication Layout Containment
D-177337, Sh. 1	Communication Layout Auxiliary Building
D-177337, Sh. 2	Unit 1 Communication Layout Auxiliary Building
D-177337, Sh. 3	Unit 1 Communication Layout Auxiliary Building
D-177338	Communication Layout River Water, Diesel Building, and Service Water
D-177339	Unit 1 Communication Layout Containment
D-177645	Elem. Diagram Loading B1F ESS SEQ.
D-177646	Elem. Diagram Loading Sequencer B1G ESS SEQ.
D-177647	Elem. Diag. ESS Loading SEQ. B1F Breaker Close

TABLE 1.6-1 (SHEET 16 OF 28)

Drawing Number	Drawing Title
	Failure Ind.
D-177648	Elem. Diag ESS Loading SEQ. B1G Breaker Close Failure Ind.
D-177649	Elem. Diagram Loading SEQ. B1F LOSP Sequencer BUS 1F
D-177650	Elem. Diagram Loading SEQ. B1G LOSP Sequencer BUS 1G
D-177653	Elem. Diagram Sequencer B1F Load Shedding CKT
D-177654	Elem. Diagram Sequencer B1G Load Shedding CKT
D-177659	Elem. Diagram Sequencer B1H BUS 1H Load Shedding CKT
D-177660, Sh. 1	Elem. Diagram Sequencer B1J BUS 1J Load Shedding CKT
D-177660, Sh. 2	Elem. Diagram Sequencer B1J BUS 1J Load Shedding CKT
D-177677	Single-Line Protection and Metering 600-V Load Center 1R
D-177678	Single-Line Protection and Metering 600-V Load Center 1S
D-177754	Tray and Conduit Layout, Cable Spreading Room
D-177944, Sh. 1	Single Line TDAFW Pumps UPS
D-181620	Conn. Diagram 208/120V Space Heater Panel N1R19L001C-N, 1D-N
D-181889, Sh. 2	Conn. Diagram Computer Power Distribution Equipment
D-200007	Main Steam System
D-200009	Extraction Steam System
D-200011, Sh. 1	Condensate and Feedwater System

TABLE 1.6-1 (SHEET 17 OF 28)

Drawing Number	Drawing Title
D-200011, Sh. 2	Condensate and Feedwater System
D-200011, Sh. 3	Condensate and Feedwater System
D-200013, Sh. 3	Diesel Generator Cooling Water System Unit 2
D-200013, Sh. 6	Circulating Water System
D-200013, Sh. 8	River Water, Service Water and Circulating Water Systems
D-200019, Sh. 1	Compressed Air System; Unit 2; Instrument Air
D-200019, Sh. 2	Compressed Air System; Unit 2; Instrument Air
D-200189	Suppression and Detection Annunciator List - Unit 2
D-202711, Sh. 1	Single Line & Cable Diagram D-C Distribution - Turb. Bldg. Battery
D-204620	Conn. Diagram 208/120V Space Heater Dist. Panel N2R19L001C-N, 1D-N
D-204889, Sh. 2	Conn. Diagram Computer Power Distribution Equipment
D-205002, Sh. 1	Unit 2 Component Cooling Water System
D-205002, Sh. 2	Unit 2 Component Cooling Water System
D-205002, Sh. 3	Unit 2 Component Cooling Water System
D-205003, Sh. 1	Unit 2 Service Water System Inside Containment and Auxiliary Building
D-205003, Sh. 2	Unit 2 Service Water System Inside Containment and Auxiliary Building
D-205003, Sh. 3	Unit 2 Service Water System Inside Containment and Auxiliary Building
D-205003, Sh. 4	Unit 2 Service Water System Inside Containment and Auxiliary Building

TABLE 1.6-1 (SHEET 18 OF 28)

Drawing Number	Drawing Title
D-205004, Sh. 1	Unit 2 Containment and Auxiliary Bldg
D-205004, Sh. 2	Unit 2 Containment and Auxiliary Bldg
D-205005	Unit 2 Auxiliary Building Drains Non- Rad RCP Oil Collection System
D-205007	Unit 2 Auxiliary Feedwater System
D-205009, Sh. 1	Sampling System
D-205009, Sh. 2	Sampling System
D-205009, Sh. 3	Sampling System
D-205010, Sh. 1	Containment Cooling and Purge System - Unit 2
D-205010, Sh. 2	Containment Cooling and Purge System - Unit 2
D-205011, Sh. 1	Radwaste Area HVAC System
D-205011, Sh. 2	Radwaste Area HVAC System
D-205011, Sh. 3	Radwaste Area HVAC System
D-205011, Sh. 4	Radwaste Area HVAC System
D-205012	Control Room and Computer Room HVAC and Filtration System
D-205014, Sh. 1	Nonradioactive Area and Electrical Equipment Rooms HVAC System
D-205014, Sh. 2	Nonradioactive Area and Electrical Equipment Rooms HVAC System
D-205019	Postaccident Containment Combustible Gas Control System - Unit 2
D-205022	Penetration Filtration System - Unit 2
D-205031, Sh. 1	Turbine Building Chilled Water System
D-205031, Sh. 2	Turbine Building Chilled Water System

TABLE 1.6-1 (SHEET 19 OF 28)

Drawing Number	Drawing Title
D-205031, Sh. 3	Turbine Building Chilled Water System
D-205033, Sh. 1	Main Steam System
D-205033, Sh. 2	Main Steam System
D-205034, Sh. 1	Compressed Air System; Unit 2; Instrument Air
D-205034, Sh. 2	Compressed Air System; Unit 2; Instrument Air
D-205034, Sh. 3	Compressed Air System; Unit 2; Instrument Air
D-205034, Sh. 4	Compressed Air System; Unit 2; Instrument Air
D-205035	Compressed Air System; Unit 2; Instrument Air
D-205036	Reactor Makeup Water System
D-205037, Sh. 1	Unit 2 Reactor Coolant System
D-205037, Sh. 2	Reactor Coolant System - Unit 2
D-205037, Sh. 3	Reactor Coolant System - Unit 2
D-205038, Sh. 1	Safety Injection System - Unit 2
D-205038, Sh. 2	Safety Injection System - Unit 2
D-205038, Sh. 3	Safety Injection System (Containment Spray) Unit 2
D-205039, Sh. 1	Chemical and Volume Control System
D-205039, Sh. 2	Chemical and Volume Control System
D-205039, Sh. 3	Chemical and Volume Control System
D-205039, Sh. 4	Chemical and Volume Control System
D-205039, Sh. 5	Chemical and Volume Control System
D-205039, Sh. 6	Chemical and Volume Control System
D-205039, Sh. 7	Chemical and Volume Control System

TABLE 1.6-1 (SHEET 20 OF 28)

Drawing Number	Drawing Title
D-205040	Boron Thermal Regeneration System
D-205041	Unit 2 Residual Heat Removal System
D-205042, Sh. 1	Waste Processing System Flow Diagram
D-205042, Sh. 2	Waste Processing System Flow Diagram
D-205042, Sh. 3	Waste Processing System Flow Diagram
D-205042, Sh. 4	Waste Processing System Flow Diagram
D-205042, Sh. 5	Waste Processing System Flow Diagram
D-205042, Sh. 6	Waste Processing System Flow Diagram
D-205042, Sh. 7	Waste Processing System Flow Diagram
D-205042, Sh. 9	Waste Processing System Flow Diagram
D-205042, Sh. 10	Waste Processing System Flow Diagram
D-205043	Spent Fuel Pool Cooling System
D-205045	Spent Fuel Pool Ventilation System
D-205047	Demineralized Water System
D-205071, Sh. 1	Steam Generator Blowdown Processing System
D-205071, Sh. 2	Steam Generator Blowdown Processing System
D-205071, Sh. 3	Steam Generator Blowdown Processing System
D-205073	Condensate and Feedwater System
D-205205	Auxiliary Building Floor and EquipmentDrains at El 121 ft-0 in.
D-205206	Auxiliary Building Floor and Equipment Drains at El 139 ft-0 in.
D-205207	Auxiliary Building Floor and Equipment Drains at El 155 ft-0 in.

TABLE 1.6-1 (SHEET 21 OF 28)

Drawing Number	Drawing Title
D-205378	Containment Spray Headers - Unit 2
D-205379	Containment Spray Headers - Unit 2
D-205380	Containment Spray Headers - Sections and Details - Unit 2
D-205381	Containment Spray Headers - Sections and Details - Unit 2
D-205491	Containment Spray Headers - Unit 2
D-205512	Safe Load Path, Reactor Head – Unit 2
D-205513	Safe Load Path, Upper Internals –Unit 2
D-205514	Safe Load Path, Reactor Coolant Pumps – Unit 2
D-205516, Sh. 1	Safe Load Path, Polar Crane Load Block – Unit 2
D-205516, Sh. 2	Safe Load Path, Polar Crane Load Block – Unit 2
D-206002	Auxiliary Building Floor Plan El 83 ft-0 in. and 77 ft-0 in. - Unit 2
D-206003	Containment and Auxiliary Building Floor Plan El 100 ft-0 in. and 105 ft-6 in. - Unit 2
D-206004	Containment and Auxiliary Building Floor Plan El 121 ft-0 in. and 129 ft-0 in. - Unit 2
D-206005	Containment and Auxiliary Building Floor Plan El 139 ft-0 in. - Unit 2
D-206006	Containment and Auxiliary Building Floor Plan El 155 ft-0 in. and 165 ft-0 in. - Unit 2
D-206007	Containment and Auxiliary Building Roof Plan - Unit 2
D-206035	Unit 2 Radiation Zones and Controlled Access Floor Plan - El 83 ft-0 in. and 77 ft-0 in.
D-206036	Unit 2 Radiation Zones and Controlled Access Floor

TABLE 1.6-1 (SHEET 22 OF 28)

Drawing Number	Drawing Title
	Plan - El 100 ft-0 in.
D-206037	Unit 2 Radiation Zones and Controlled Access Floor Plan - El 121 ft-0 in. and 129 ft-0 in.
D-206038	Unit 2 Radiation Zones and Controlled Access Floor Plan - El 139 ft-0 in.
D-206039	Unit 2 Radiation Zones and Controlled Access Floor Plan - El 155 ft-0 in.
D-206040	Unit 2 Radiation Zones and Controlled Access Roof Plan
D-206041	Unit 2 Tabulation of Radiation Zones
D-206042	Unit 2 Tabulation of Radiation Zones
D-206043	Unit 2 Tabulation of Radiation Zones
D-206075	Unit 2 Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 83 ft-0 in. and 77 ft-0 in.
D-206076	Unit 2 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 100 ft-0 in. and 105 ft-0 in.
D-206077	Unit 2 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 121 ft-0 in. and 129 ft-0 in.
D-206078	Unit 2 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 139 ft-0 in.
D-206079	Unit 2 Containment and Auxiliary Building Floor Plan Post-LOCA Radiation Zone Maps - El 155 ft-0 in.
D-206107	Incore Instrumentation Tube Supports Structural Steel - Containment - Unit 2
D-206238	Cross-Over Support Details Structural Steel - Containment - Unit 2
D-206275	Reactor Primary Shielding Penetration, Plan, and Containment - Unit 2

TABLE 1.6-1 (SHEET 23 OF 28)

Drawing Number	Drawing Title
D-206277	Reactor Cavity Liner Plate - Unit 2
D-206278	Restriction Ring and Bulkhead at Reactor Cavity - Thermal Movement Diagram - Unit 2
D-206279	Reactor Cavity Liner Plate - Unit 2
D-207000	Unit 2 Single-Line - Electrical Auxiliary System (Normal 4160-V and 600-V)
D-207001	Unit 2 Single-Line - Electrical Auxiliary System (Normal 4160-V and 600-V)
D-207005	Unit 2 Single-Line Protection and Metering 4160-V Bus 2F
D-207006	Unit 2 Single-Line Protection and Metering 4160-V Bus 2G
D-207007	Unit 2 Single-Line Protection and Metering 600-V Load Center 2A
D-207009	Unit 2 Single-Line Protection and Metering 600-V Load Center 2C
D-207010	Unit 2 Single-Line Protection and Metering 600-V Load Center 2D
D-207011	Unit 2 Single-Line Protection and Metering 600-V Load Center 2E
D-207014	Unit 2 Single-Line Protection and Metering 600-V Load Center 2H
D-207015	Unit 2 Single-Line Protection and Metering 600-V Load Center 2J
D-207018	Unit 2 Single-Line Protection and Metering 4160-V Bus 2H
D-207024	Unit 2 Single-Line 120 V-ac Vital and Regulated System A
D-207025	Unit 2 Single-Line 120 V-ac Vital and Regulated

TABLE 1.6-1 (SHEET 24 OF 28)

Drawing Number	Drawing Title
	System B
D-207027	Unit 2 Single-Line Protection and Metering 4160-V Bus 2J
D-207032	Unit 2 Logic Diagram -Diesel 2B Auto Start and Loading
D-207033	Unit 2 Logic Diagram -Diesel 1-2A Auto Start and Loading
D-207036	Unit 2 Logic Diagram -Diesel 1C Auto Start and Loading
D-207037	Unit 2 Logic Diagram -Diesel 2C Auto Start and Loading
D-207038	Single Line - Miscellaneous Panel Boards for Emergency Power
D-207045	Unit 2 Single-Line Protection and Metering 600-V Center 2K
D-207046	Unit 2 Single-Line Protection and Metering 600-V Center 2L
D-207082	Unit 2 Single-Line Dc Distribution System 2A
D-207083	Unit 2 Single-Line dc Distribution System 2B
D-207155	Elem. Diagram 4160-V Bus 2F Incoming Startup Transformer 2A
D-207161	Elem. Diagram 4160-V Bus 2F Incoming Startup Transformer 2B
D-207185	Unit 2 Elementary Diagram Component Cooling Water Pump 2B - Train “A”
D-207187	Unit 2 Elementary Diagram Component Cooling Water Pump 2B - Train “B”
D-207331	Unit 2 Communication Auxiliary Building and Containment

TABLE 1.6-1 (SHEET 25 OF 28)

Drawing Number	Drawing Title
D-207334, Sh. 1	Unit 2 Communication Layout Auxiliary Building
D-207334, Sh. 2	Unit 2 Communication Layout Auxiliary Building
D-207336	Unit 2 Communication Layout Containment
D-207337, Sh. 1	Unit 2 Communication Layout Auxiliary Building
D-207337, Sh. 2	Unit 2 Communication Layout Auxiliary Building
D-207339	Unit 2 Communication Layout Containment
D-207650	Elem. Diagram Loading SEQ. B2G LOSP Sequencer BUS 2G
D-207659	Elem. Diagram Sequencer B2H BUS 2H Load Shedding CKT
D-207660, Sh. 1	Elem. Diagram Sequencer B2J BUS 2J Load Shedding CKT
D-207660, Sh. 2	Elem. Diagram Sequencer B2J BUS 2J Load Shedding CKT
D-207944	Single Line TDAFW Pumps UPS
D-508475	Arrangement of Fire Main Around Farley Nuclear Plant Site, Units 1 and 2
D-508476	Arrangement of Fire Main Around Farley Nuclear Plant Site, Units 1 and 2
D-508507	Suppression and Detection Annunciator List Legend - Unit 1
D-356597, Sh. 1	Unit 2 Auxiliary Building Floor Plan El. 83’-0” FNP-0-AP-35.3 Exempt Areas
D-356597, Sh. 2	Unit 2 Auxiliary Building Floor Plan El. 100’-0” FNP-0-AP-35.3 Exempt Areas
D-356597, Sh. 3	Unit 2 Auxiliary Building Floor Plan El. 121’-0” FNP-0-AP-35.3 Exempt Areas
D-356597, Sh. 4	Unit 2 Auxiliary Building Floor Plan El. 139’-0”

TABLE 1.6-1 (SHEET 26 OF 28)

Drawing Number	Drawing Title
	FNP-0-AP-35.3 Exempt Areas
D-356597, Sh. 5	Unit 2 Auxiliary Building Floor Plan El. 159'-0" FNP-0-AP-35.3 Exempt Areas
D-506531, Sh. 1	Unit 1 Auxiliary Building Floor Plan El. 83'-0" FNP-0-AP-35.3 Exempt Areas
D-506531, Sh. 2	Unit 1 Auxiliary Building Floor Plan El. 100'-0" FNP-0-AP-35.3 Exempt Areas
D-506531, Sh. 3	Unit 1 Auxiliary Building Floor Plan El. 121'-0" FNP-0-AP-35.3 Exempt Areas
D-506531, Sh. 4	Unit 1 Auxiliary Building Floor Plan El. 139'-0" FNP-0-AP-35.3 Exempt Areas
D-506531, Sh. 5	Unit 1 Auxiliary Building Floor Plan El. 155'-0" FNP-0-AP-35.3 Exempt Areas
U-165665	Reactor Pressure Vessel Schematic
U-166231	Functional Diagrams Index and Symbols
U-166232	Reactor Trip Symbols
U-166233	Nuclear Instrumentation and Manual Trip Signals
U-166234	Nuclear Instrumentation Permissives and Blocks
U-166235	Primary Coolant Systems Trip Signals
U-166236	**Pressurizer Trip Signals**
U-166237	Steam Generator Trip Signals
U-166238	Safeguards Actuation Signals
U-166239	Rod Controls and Rod Blocks
U-166240	Steam Dump Control
U-166241	Pressurizer Pressure and Level Control
U-166242	Pressurizer Heater Control

TABLE 1.6-1 (SHEET 27 OF 28)

Drawing Number	Drawing Title
U-166243	Feedwater Control and Isolation
U-166244	Auxiliary Feedwater Pumps Startup
U-166245	Turbine Trips, Runbacks, and Other Signals
U-167647	Radiation Monitoring System Functional Block Diagram
U-167650	ALA/APR Radiation Monitoring System Functional Block Diagram
U-167651	Radiation Monitoring System Area Range R-17A, R-17B, and R-18
U-167652	ALA/APR Radiation Monitoring System Functional Block Diagram
U-170148	Solid State Protection System Interconnection Diagram
U-202225	RCS Equipment Support S.G. Inlet Restraint
U-261455	RCS Equipment Support S.G. Inlet Restraint
U-264612	Flow Diagram of the Reactor Vessel Head Vent System
U-419610	Cable Drive Installation Assembly Arrangement
U-419916	Ex-Vessel Neutron Dosimetry Housing, Clamp & Chain Stop
U-419917	Ex-Vessel Neutron Dosimetry Support Bar
U-419918	Ex-Vessel Neutron Dosimetry Chain Assembly
U-419920	Ex-Vessel Neutron Dosimetry Installation – Unit 1
U-419289	Replacement Reactor Vessel Closure Head Outline Drawing – Unit 1

TABLE 1.6-1 (SHEET 28 OF 28)

Drawing Number	Drawing Title
U-611138	Replacement Reactor Vessel Closure Head Outline Drawing – Unit 2
U-611432	Ex-Vessel Neutron Dosimetry Support Frame Installation – Unit 2
U-611433	Ex-Vessel Neutron Dosimetry Support Frame Assembly – Unit 2

1.7 GLOSSARY OF TERMS

This section presents abbreviations, symbols, indices, legends, and other aids to facilitate review of this Final Safety Analysis Report. Information was compiled from applicable specifications, drawings, FSAR sections, and related publications developed throughout the design, construction, and documentation of the Farley Nuclear Plant.

1.7.1 ABBREVIATIONS

The technical abbreviations in table 1.7-1 are used where appropriate throughout the FSAR.

1.7.2 DRAWING INDEX AND SYMBOLS

The piping and instrumentation diagrams (P&IDs) are listed in table 3.2-3. Drawings D-175016, sheet 1, D-175016, sheet 2, D-175016, sheet 3, figures 1.7-1 and 1.7-2 are provided to facilitate the understanding of the figures and referenced drawings throughout the FSAR.

1.7.3 HISTORICAL DESCRIPTIONS

Historical descriptions in the FSAR fall into one of the following categories:

1. Initial condition data, that may have been provided as part of original licensing activities, but that is not required for continued plant operation.

2. Initial test or analyses provided to document equipment acceptability.

3. References to initial construction related procedures or activities.

Material that falls into one of the above categories is not necessary to support the current operations of the plant but is being retained in order to provide additional information concerning the licensing history of FNP. The designation of material as historical is not to be used to justify the removal and/or abandonment of any structure, system or component.

Historical sentences, paragraphs, sections and/or tables have been annotated with the word "HISTORICAL" set off by brackets and marked with dollar signs ($). The annotation is placed at the beginning of the historical material and the material itself is also clearly marked to insure clarity. The following is an illustration of the annotation method; the historical material is represented by the words "historical material" in the section set of brackets:

[HISTORICAL] *[If historical material continues from one FSAR page to subsequent pages, each subsequent page is annotated exactly like the first page.]*

TABLE 1.7-1 (SHEET 1 OF 15)

TECHNICAL ABBREVIATIONS

Word	Abbreviation
absolute	abs
absolute ampere	abamp
actual cubic feet per minute	aft^3/min
alternating current	ac
altitude	alt
ampere(s)	A
ampere-hour(s)	Ah
ampere per square centimeter	A/cm^2
anno domini	A.D.
angstroms	Å
ante meridian	a.m.
antilogarithm	log^{-1}, antilog
approximately	≈ or approx
asymmetrical	asym
atmosphere (standard)	atm
atomic mass unit (unified)	u
atomic number	at no.
atomic percent	at %
atomic weight	at wt
atomic weight unit	awu
audio-frequency	af
average	avg

TABLE 1.7-1 (SHEET 2 OF 15)

Word	Abbreviation
bar(s)	bar
barn(s)	b
barrel(s)	bbl
Baume	Be
billion electronvolts	GeV
biot(s)	Bi
body centered cubic	bcc
boiling point	bp
brake horsepower	bhp
Brinell hardness number	Bhn
British thermal unit	Btu
British thermal unit per hour	Btu/h
British thermal unit per hour per degree Fahrenheit per foot (thermal conductivity)	Btu/h-ft-°F
calculated	calc
calorie(s)	cal
candela(s)	cd
candlepower	cp
Celsius (centigrade)	°C
cent(s)	¢
center line	cl
centigram	cg
centimeter(s)	cm

TABLE 1.7-1 (SHEET 3 OF 15)

Word	Abbreviation
centimeter-gram-second	cgs
centimeters per second	cm/s
centipoise	cP
chemically pure	cp
coefficient	coef
cologarithm	colog
concentrated	conc
constant	const
cosecant	csc
cosine	cos
cotangent	cot
coulomb(s)	C
counts per minute	cpm
cubic	cu
cubic centimeter(s)	cc or cm^3
cubic feet per minute	ft^3/min
cubic feet per second	ft^3/s
cubic foot (feet)	ft^3
cubic inch(es)	$in.^3$
cubic meter	m^3
cubic micron(s)	cu μ or μ^3
cubic millimeter(s)	cu mm, mm^3
cubic yard	yd^3

TABLE 1.7-1 (SHEET 4 OF 15)

Word	Abbreviation
curies	Ci
curies per minute	Ci/min
curies per second	Ci/s
cycles per second (hertz electronics)	Hz
cylinder	cyl
day	day
debye(s)	D
decibel(s)	dB
degree(s)	deg
degree Baume	°B
degree Celsius (centigrade)	°C
degree Fahrenheit	°F
degree Kelvin (absolute)	K
decimeter(s)	dm
diameter	diam, dia.
diamond pyramid hardness	DPH
direct current	dc
disintegration(s)	dis
disintegrations per minute	dpm
disintegrations per second	dps
dollar(s)	$
dyne(s)	dyn
east	E

TABLE 1.7-1 (SHEET 5 OF 15)

Word	Abbreviation
electromagnetic force	emf
electromagnetic unit	emu
electron volt(s)	eV
electrostatic units	esu
entropy units	eu
equation(s)	Eq, Eqs
equivalent	equiv
erg(s)	erg
exponential	exp
exponential integral	Ei
Fahrenheit	°F
farad(s)	F
feet (foot)	ft
feet per minute	ft/min
feet per second	ft/s
fermi (=10^{-13}cm)	F
figure	fig.
footcandle	fc
footlambert	fL
foot-pound	ft-lb
franklin(s)	Fr
frequency modulation	fm
gallon(s)	gal

TABLE 1.7-1 (SHEET 6 OF 15)

Word	Abbreviation
gallons per minute	gal/min
gallons per second	gal/s
gallons per hour	gal/h
gauss	G
gilbert(s)	Gb
gram(s)	g
gram-calorie	g-cal
gram-molecular volume	gmv
grams per liter	g/liter
henry(-ies)	H
hertz (cycle per second)	Hz
high frequency	hf
high voltage	hv
horsepower	hp
hour(s)	h
hydrogen ion concentration, negative logarithm of	pH
hyperbolic cosecant	csch
hyperbolic cosine	cosh
hyperbolic cotangent	coth
hyperbolic sine	sinh
inch(es)	in.
inches of mercury	in. Hg
inches of water	in. H_2O

TABLE 1.7-1 (SHEET 7 OF 15)

Word	Abbreviation
inch-pound	in.-lb
inside diameter	id
integrated neutron flux	nvt
intermediate frequency	if
intramuscular(ly)	im
intraperitoneal(y)	ip
intraveneous(ly)	iv
international angstrom	IA
joule(s)	J
kelvin	K
kilocalorie(s)	kcal
kilocurie	kCi
kilocycles per second	kHz
kiloelectron volt(s)	keV
kilogauss	kG
kilogram(s)	kg
kilogram meter	kg-M
kilogram-weight	kg-wt
kilohm(s)	KΩ
kilojoule(s)	kJ
kiloliter(s)	kliter
kilometer(s)	km
kilo-oersted	kOe

TABLE 1.7-1 (SHEET 8 OF 15)

Word	Abbreviation
kilovolt(s)	kV
kilovolt-ampere(s)	kVA
kilowatt(s)	kW
kilowatt-hour(s)	kWh
kinetic energy	KE or T
Knopp Hardness Number (microhardness)	KHN
laboratory	lab
lambert	L
limit	lim
liter(s)	liters
logarithm (common)	log
logarithm (natural)	ln
lumen	lm
lumens per watt	lm/W
lux	lx
magnetomotive force	mmf
magnified 50 times	50X
maximum	max
maxwell(s)	Mx
megacycles(s)	Mc
megacycles per second	Mc/s
megacycles per second (electronics)	MHz
megacycles per second (mechanics)	Mc/s

TABLE 1.7-1 (SHEET 9 OF 15)

Word	Abbreviation
megavolts	MV
megawatts	MW
megawatt-day(s)	MWday
megawatt-electric	MWe
megawatt-hour(s)	MWh
megawatt-second(s)	MWs
megawatt-year(s)	MWyear
megawatt-thermal	MWt
megohm(s)	MΩ
melting point	mp
meter(s)	m
meter-kilogram second	mks
metric ton	MT(tonne)
mho	mho
microampere(s)	μA
microangstrom	μÅ
microbar	μbar
microbarn(s)	μb
microcoulomb(s)	μC
microcuries	μCi
microgram	μg
microfarad	μF
microhenry	μH

TABLE 1.7-1 (SHEET 10 OF 15)

Word	Abbreviation
microinch	μin.
micromicrofarad	μμF
micromicrons	μμ
micromole	μM
micron(s)	μ
microsecond	μs
microvolt(s)	μV
microwatt(s)	μW
mile	mi
miles per hour	mph
milliampere(s)	mA
millicurie(s)	mCi
milligauss	mG
milligram	mg
milligrams per decimeter per day	mdd
millihenry	mH
milliliter(s)	mliter
milli-mass-unit	mmu
millimeter	mm
millimicron(s)	mμ
millimicrosecond(s) [nanosecond(s) perferred]	mμs
millimole(s)	mM
million electron volts	MeV

TABLE 1.7-1 (SHEET 11 OF 15)

Word	Abbreviation
million volts	MV
milliroentgen per hour	mR/h
millisecond(s)	ms
millivolt(s)	MV
minimum	min.
minute(s)	min
molal	molal
molar	M
mole	mole
mole percent	mole percent
molecular weight	mol wt
month	mo
nanocuries	nCi
nanosecond(s)	ns
neper(s)	Np
neutron flux	nv
neutrons per volume time	nvt
neutrons per square centimeter per second	n/cm^2-s
newton(s)	N
north	N
normal	N
nuclear magnetron	nm
number	No.

TABLE 1.7-1 (SHEET 12 OF 15)

Word	Abbreviation
oersted	Oe
ohm(s)	Ω
ounce(s)	oz
outside diameter	od
page	p
pages	pp
parts per billion	ppb
parts per million	ppm
percent	% (graphics)
percent milli-k	pcm
picofarad(s)	pF
poise	P
post meridian	p.m.
potential difference	PD
potential energy	PE or V
pound	lb
pounds per cubic foot	lb/ft^3
pounds per square foot	lb/ft^2
pounds per square inch	psi
pounds per square inch absolute	psia
pounds per square inch differential	psid
pounds per square inch gauge	psig
pressure (millimeter of mercury)	mm Hg

TABLE 1.7-1 (SHEET 13 OF 15)

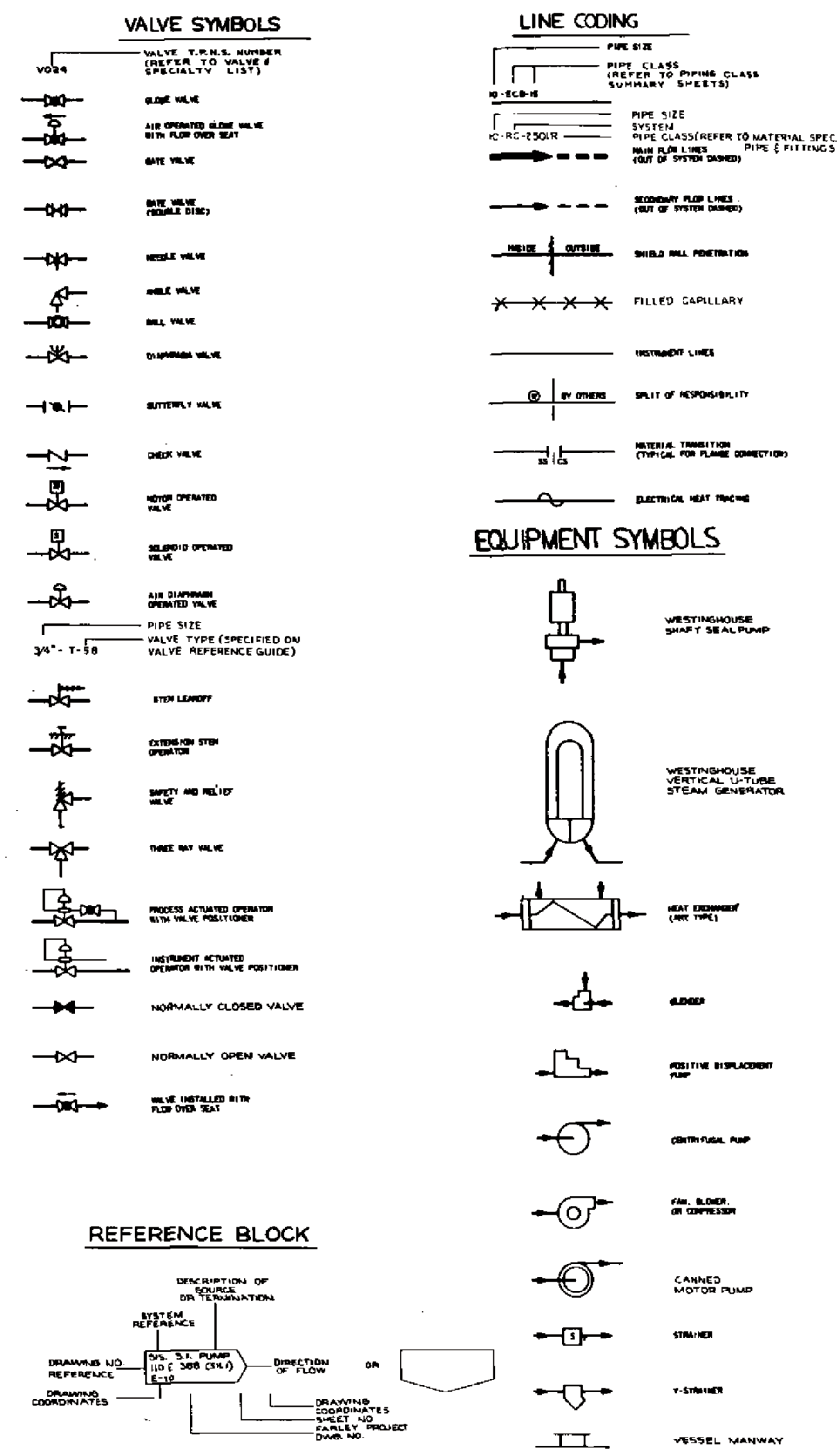
Word	Abbreviation
probable error	pe
radian	rad
radioactivity (measure of)	rad
rad equivalent man	rem
Radiation Protection Guide	RPG
Radioactivity Concentration Guide	RCG
Rankins (degree)	°R
revolutions per minute	rpm
revolutions per second	rps
roentgen(s)	R
root mean square	rms
secant	sec
second(s)	s
Section	sec.
sine	sin
south	S
specific gravity	sp gr, s.g.
square	sq
square centimeters	cm^2
square foot	ft^2
square inch(es)	$in.^2$
square kilometer(s)	km^2
square meter(s)	m^2

TABLE 1.7-1 (SHEET 14 OF 15)

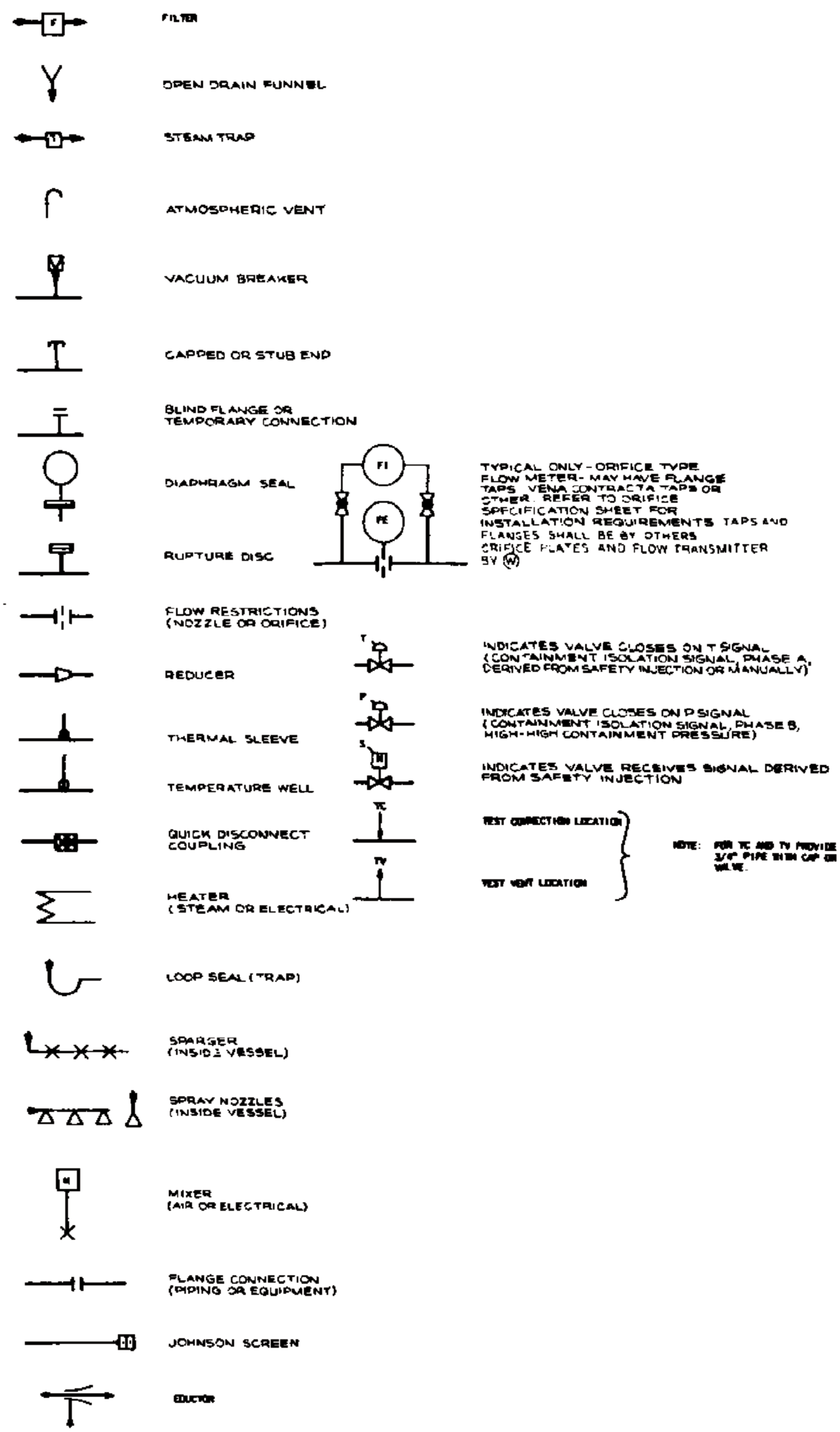
Word	Abbreviation
square micron(s)	μ^2
square millimeter(s)	mm^2
stainless steel	ss
standard	Std.
standard temperature and pressure	STP
steradian	sr
tangent	tan
temperature	temp
tensile yield strength	tys
tesla (Wb/m^2)	T
thousand circular mills	kcmil
thousand electron volts	keV
ton(s)	ton
trace	Tr
transpose	tr
ultimate tensile strength	uts
ultraviolet	uv
velocity	v
versus	vs
volt(s)	v
volume	vol
volume parts per million	vpm
water gauge	wg

TABLE 1.7-1 (SHEET 15 OF 15)

Word	Abbreviation
watt(s)	W
weber	Wb
weight	wt
weight percent	wt/%
west	W
x units	xu
yard(s)	yd
year(s)	year


VALVE SYMBOLS
VALVE T.P.N.S. NUMBER (REFER TO VALVE & SPECIALTY LIST)
V024
GLOBE VALVE
AIR OPERATED GLOBE VALVE WITH FLOW OVER SEAT
GATE VALVE
GATE VALVE (DOUBLE DISC)
NEEDLE VALVE
ANGLE VALVE
BALL VALVE
DIAPHRAGM VALVE
BUTTERFLY VALVE
CHECK VALVE
MOTOR OPERATED VALVE
SOLENOID OPERATED VALVE
AIR DIAPHRAGM OPERATED VALVE
PIPE SIZE
VALVE TYPE (SPECIFIED ON VALVE REFERENCE GUIDE)
3/4"- T-58
STEM LEAKOFF
EXTENSION STEM OPERATOR
SAFETY AND RELIEF VALVE
THREE WAY VALVE
PROCESS ACTUATED OPERATOR WITH VALVE POSITIONER
INSTRUMENT ACTUATED OPERATOR WITH VALVE POSITIONER
NORMALLY CLOSED VALVE
NORMALLY OPEN VALVE
VALVE INSTALLED WITH FLOW OVER SEAT
LINE CODING
PIPE SIZE
PIPE CLASS (REFER TO PIPING CLASS SUMMARY SHEETS)
PIPE SIZE
SYSTEM
PIPE CLASS (REFER TO MATERIAL SPEC. PIPE & FITTINGS
MAIN FLOW LINES (OUT OF SYSTEM DASHED)
SECONDARY FLOW LINES (OUT OF SYSTEM DASHED)
INSIDE
OUTSIDE
SHIELD WALL PENETRATION
FILLED CAPILLARY
INSTRUMENT LINES
BY OTHERS
SPLIT OF RESPONSIBILITY
MATERIAL TRANSITION (TYPICAL FOR FLANGE CONNECTION)
SS
CS
ELECTRICAL HEAT TRACING
EQUIPMENT SYMBOLS
WESTINGHOUSE SHAFT SEAL PUMP
WESTINGHOUSE VERTICAL U-TUBE STEAM GENERATOR
HEAT EXCHANGER (ANY TYPE)
BLENDER
POSITIVE DISPLACEMENT PUMP
CENTRIFUGAL PUMP
FAN, BLOWER, OR COMPRESSOR
CANNED MOTOR PUMP
STRAINER
Y-STRAINER
VESSEL MANWAY
REFERENCE BLOCK
DESCRIPTION OF SOURCE OR TERMINATION
SYSTEM REFERENCE
DRAWING NO. REFERENCE
DRAWING COORDINATES
DIRECTION OF FLOW
OR
DRAWING COORDINATES
SHEET NO.
FARLEY PROJECT DWG. NO.

REV 21 5/08



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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FLUID SYSTEM SYMBOLS

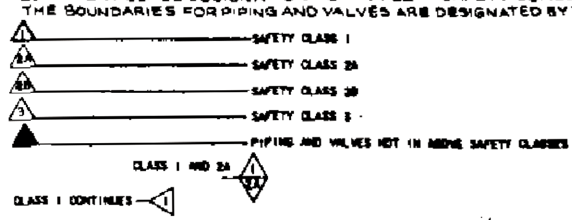
FIGURE 1.7-1 (SHEET 2 OF 3)

LEGEND:

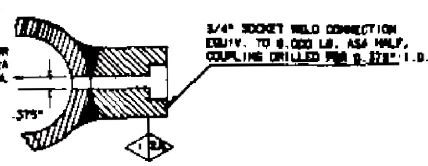
AUX.R.R.	-	AUXILIARY RELAY RACK
LPP	-	LIQUID PROCESSING PANEL
GPP	-	GAS PROCESSING PANEL
BRP	-	BORON RECYCLE PANEL
DRP		DRUMMING ROOM PANEL
TC	-	TEST CONNECTION
TV	-	TEST VENT CONNECTION
RWST	-	REFUELING WATER STORAGE TANK
SRST	-	SPENT RESIN STORAGE TANK
PRT	-	PRESSURIZER RELIEF TANK
WHT	-	WASTE HOLDUP TANK
ATM	-	ATMOSPHERE
RMW	-	REACTOR MAKEUP WATER
CCW	-	COMPONENT COOLING WATER
F.A.I.	-	FAIL AS IS
F.C.	-	FAIL CLOSED
F.O.	-	FAIL OPEN
L.C.	-	LOCKED CLOSED
L.O.	-	LOCKED OPEN
DH	-	DRAIN HEADER
GFFD		GROSS FAILED FUEL DETECTOR
RHT	-	RECYCLE HOLDUP TANK
N.O.	-	NORMALLY OPEN
N.C.	-	NORMALLY CLOSED
VH	-	VENT HEADER
BIT	-	BORON INJECTION TANK
ORC	-	OUTSIDE REACTOR CONTAINMENT
IRC	-	INSIDE REACTOR CONTAINMENT
DW	-	DEMINERALIZED WATER
D	-	LOCAL DRAIN TO FLOOR OR PORTABLE CONTAINER
V	-	VENT TO ATMOSPHERE
N_2	-	NITROGEN FROM WPS NITROGEN MANIFOLD
H_2	-	HYDROGEN FROM WPS HYDROGEN MANIFOLD
T	-	CONTAINMENT ISOLATION TRIP SIGNAL PHASE A (S SIGNAL OR MANUAL)
RCDT	-	REACTOR COOLANT DRAIN TANK
S	-	SAFETY INJECTION SIGNAL
CCWS	-	COMPONENT COOLING WATER SYSTEM
SFPCS	-	SPENT FUEL POOL COOLING SYSTEM
WPS	-	WASTE PROCESSING SYSTEM
SS	-	SAMPLING SYSTEM
IMB	-	INSIDE MISSILE BARRIER
OMB	-	OUTSIDE MISSILE BARRIER
RR	-	REACH ROD
RMWS	-	REACTOR MAKEUP WATER SYSTEM
RD	-	THERMAL REGENERATION SIGNAL
RCPS	-	REACTOR CONTROL & PROTECTION SYSTEM
A/E	-	ARCHITECT—/—ENGINEER
CS	-	CONTAINMENT SUMP
ECC	-	EMERGENCY COOLING CONNECTION
P	-	CONTAINMENT ISOLATION TRIP SIGNAL PHASE B (HI-HI CONTAINMENT PRESS.)
CCDT	-	COMPONENT COOLING DRAIN TANK
FDT	-	FLOOR DRAIN TANK
SW	-	SERVICE WATER
SC		SAFETY CLASS

GENERAL NOTES:

1. ADDITIONAL VENTS AND DRAINS WILL BE REQUIRED BY THE PIPING LAYOUT. ALL VENT AND DRAIN VALVES WILL BE 3/4 INCH OF THE APPROPRIATE PRESSURE CLASS.
2. ALL VALVES SHOWN TO HAVE LEAKOFF CONNECTIONS SHALL BE PERMANENTLY PIPED TO THE APPROPRIATE DRAIN POINT. ALL OTHER VALVES WITH LEAKOFFS WILL HAVE LEAKOFF CONNECTIONS CAPPED
3. ANS SAFETY CLASSIFICATIONS
   EQUIPMENT CLASS DESIGNATION - EXAMPLE: SAFETY CLASS 2A
   THE BOUNDARIES FOR PIPING AND VALVES ARE DESIGNATED BY THE FOLLOWING FLAGS



4. ON VENT AND DRAINS WHERE A DOUBLE BARRIER IS REQUIRED (HIGH PRESSURE PIPING; CLASS 601 AND ABOVE) THE SECOND BARRIER CAN BE 3/8" CLASS 2500 TUBING WITH SWAGELOK CAPS.
5. FLOW RESTRICTION FOR SAFETY CLASS 1 TO 2A TRANSITION. TYPICAL FLOW RESTRICTOR.



6. TRANSITION PIECES MUST BE PROVIDED IF A SCHEDULE DIFFERENCE EXISTS BETWEEN SYSTEM PIPING AND COMPONENTS. IT IS NOT ACCEPTABLE TO USE COMPONENTS AS THE TRANSITION.
7. FOR ALL INSTRUMENTATION CONNECTIONS WITH A 3/4" VALVE INSTALLED, THE PIPE SIZE IS 3/4" AND THE PIPING PRESSURE-TEMPERATURE CLASS IS IDENTICAL TO THE CLASS OF PIPE TO WHICH IT IS CONNECTED.

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SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	FLUID SYSTEM SYMBOLS FIGURE 1.7-1 (SHEET 3 OF 3)

PIPING CLASS CODE DEFINITIONS

Piping Classes are designated by a three letter code. The first letter indicates the primary valve and flange rating; the second letter indicates the type of material; the third letter indicates the code to which the piping is designed.

The designations are as follows:

First Letter - Rating

C - 1500#
D - 900#
E - 600#
G - 300#
H - 150#
J - For general use as designated on piping class sheet
K - For general use as designated on piping class sheet
L - For general use as designated on piping class sheet
M - For general use as designated on piping class sheet

Second Letter - Material

B - Carbon Steel
C - Stainless Steel
D - Cast Iron
E - For general use as designated on piping class sheet
F - For general use as designated on piping class sheet
N - For general use as designated on piping class sheet

Third Letter - Design Code

A - ASME Code, 1971, Section III, Nuclear Power Plant Components
Class 1
B - Class 2
C - Class 3
D - Code for Power Piping, ANSI B31.1.0
G - Portion of ANSI B31.1.0 applicable to roof drains

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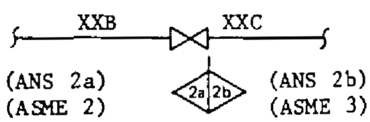
SAFETY/NUCLEAR CLASS

Flag Nomenclature

ANS Safety Class	ASME Section III Nuclear Class
1	1
2a	2
2b	3
3	3
NNS (Non-nuclear safety)	-

Example



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