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FSAR NOTE

With the exception of the description of 10CFR50.2 design basis parameters, numerical values throughout this FSAR should be considered nominal in nature. They are intended to provide a sense for the value of the parameter and should NOT be viewed as an actual value observable in the plant. Plant operation at values other than those presented herein is acceptable. The actual values are documented within established Technical Specification, Design Basis Documentation, or other controlled documents. 02-01

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

NOTE 1.1

Section 1.1 is being retained for historical purposes only. 99-01

1.1 INTRODUCTION

This Final Safety Analysis Report is submitted in support of an application by South Carolina Electric and Gas Company (SCE&G) for a class 103 license to operate a nuclear power station designated as the Virgil C. Summer Nuclear Station. This report presents descriptions and analyses of the final station design.

1.1.1 STATION LOCATION

The plant site is located north (2.5 miles) of Parr, South Carolina. Parr is the site of existing fossil and hydro power stations operated by SCE&G and the decommissioned, experimental Carolinas Virginia Tube Reactor (CVTR). The nuclear plant site is adjacent to a manmade reservoir created by placing a series of dams across Frees Creek, a tributary of the Broad River in western Fairfield County, South Carolina. The resulting Monticello Reservoir provides water requirements for the nuclear station and a pumped storage facility. The reservoir is located east of the Broad River and west of South Carolina State Highway 215, about 26 miles north of Columbia, South Carolina. The pumped storage facility raises and lowers the reservoir level approximately 4.5 feet when in operation.

1.1.2 NUCLEAR STEAM SUPPLY SYSTEM

The station includes a pressurized water reactor nuclear steam supply system (NSSS), designed and furnished by Westinghouse Electric Corporation and a turbine generator, designed and furnished by General Electric Company. This equipment is similar in design to several projects licensed or currently under review by the Nuclear Regulatory Commission (see Section 1.3). The balance of the station is designed and constructed by SCE&G, with the assistance of its agent, Gilbert Associates, Inc.

1.1.3 CONTAINMENT

Containment is provided by the Reactor Building, a reinforced concrete structure designed by Gilbert Associates, Inc. The Reactor Building is comprised of a flat foundation mat, cylindrical wall and shallow dome roof. The foundation mat and cylindrical wall are reinforced with conventional mild steel reinforcing. The cylindrical wall is prestressed in the vertical and horizontal directions by a post-tensioning system. The shallow dome roof is prestressed by a three-way post-tensioning system. The inside surface of the Reactor Building is lined with a carbon steel liner to ensure a high degree of leak tightness under operating and accident conditions.

1.1.4 CORE THERMAL POWER LEVELS

The NSSS is designed for a rated power output of 2912 MWt, which is the license application rating, with an equivalent station net electrical output of approximately 950 MWe. Containment and engineered safety features are designed and evaluated for operation based upon an Engineered Safety Design Rating (ESDR) and Licensed Power Level of 2900 MWt in the core. Postulated accidents having offsite dose consequences are evaluated at 2900 MWt. Analyses of all accident consequences, to show conformance to 10CFR50.46, have been performed at 2900 MWt. Conformance to 10CFR50.46 for Large Break LOCA is evaluated at 2900 MWt, the rated core power level following Refueling 9 in the Spring of 1996.

1.1.5 SCHEDULE

The project schedule is based upon plant completion (to fuel loading) in February, 1982 and commencement of commercial operation approximately six months after issuance of the operating license.

NOTE 1.2 Section 1.2 is being retained for historical purposes only.

99-01

1.2 GENERAL PLANT DESCRIPTION

1.2.1 SITE

1.2.1.1 Location

The Virgil C. Summer Nuclear Station site is located in Fairfield County, South Carolina, approximately 15 airline miles west of Winnsboro, the county seat. Newberry, the county seat of Newberry County, is about 17 airline miles away in a westerly direction. Therefore, the site is situated approximately midway between 2 county seats. All other communities within 15 miles of the site are small villages. Columbia, the state capital, is located about 26 miles to the southeast, as shown by the Regional Location Map, Figure 2.1-1. The Site Location Map, Figure 2.1-2 shows the site with respect to nearby roads, highways and villages in the vicinity.

1.2.1.2 Population and Land Use

The station site is located within a sparsely populated rural area in which forestry is the principal land use. In 1967, about 86% of the region within 50 miles of the site was forest or agricultural land. Columbia, the state capital, represents the major concentration of population and industry in the region. The sector south-southeastward from the site probably will continue to be the dominant urban and industrial center in the future. The migration of rural population to urban centers is believed to have maximized and should diminish with the establishment of industry, increased development of forestry and consolidation of small farms.

Within 10 miles of the site, approximately 20% of the land was being used for agriculture in 1967. This usage is decreasing and should diminish during the next decade, to be replaced largely by increases in forest land and, in some areas, by urbanization. In the next 40 years, land use within 10 miles of the site is expected to be predominantly forest land with lesser amounts devoted to urban and agricultural use. These figures are based upon factors and land use trends during the period 1958 to 1967, population growth forecasts and the expected influence of local circumstances, such as available public transportation facilities and utilities, existing soils and topography.

1.2.1.3 Meteorology

The monthly average temperature at the station site varies from about 45°F in winter to 80°F in summer. The annual precipitation averages about 45 inches. Monthly precipitation values vary from about 2-1/2 inches in November to about 5-1/2 inches in July.

The types of severe weather which may affect the site include tornadoes (with a return period of 1389 years), hurricanes (about 1 every 2 years passes within 250 miles of the site) and thunderstorms (about 54 days per year). The fastest wind speed for a 100-year return period is 100 miles per hour, sustained for about a 30 to 40-second duration.

The prevailing wind direction is from the southwest and the average wind speed is 6.1 miles per hour. This prevailing wind direction is fairly typical for all seasons except in the fall and early winter, during which time northeast to northwest winds prevail.

The occurrence of stable conditions as measured at the plant site is about 51% annually. During these conditions, the average wind speed is 4.9 miles per hour. There is no evidence that unusual topographic influences on wind direction frequencies, such as a valley drainage wind, occur during stable conditions.

The diffusion characteristics at the site are adequate to support the sitting of the Virgil C. Summer Nuclear Station. Development of Monticello Reservoir and the Fairfield Pumped Storage Facility is likely to enhance the diffusion qualities.

1.2.1.4 Hydrology

The site is situated on an irregularly shaped ridge about 1 mile east of the Broad River, the principal stream in the site area. The Broad River watershed originates in North Carolina and traverses South Carolina. Monticello Reservoir occupies approximately 6800 acres just north of the site and was created by constructing dams across Frees Creek, a tributary of the Broad River.

The climate in the area is moderately humid, and rainfall averages about 45 inches per year. Available data indicates that runoff plus small groundwater contributions to streams is approximately 17 inches annually, and evapotranspiration is about 70% (31 inches) of precipitation. Evaporation from water bodies averages 42 inches annually.

Groundwater at the site occurs in fractured crystalline bedrock and overburden soils. Water tables in the area are generally subdued replicas of topography and are deeper at higher elevations. The quality of groundwater is usually acceptable for most purposes, but large yields are seldom attained. Wells in the surrounding area are concentrated in the Jenkinsville community, approximately 3 miles southeast of the site.

At the site, impoundment of Monticello Reservoir to elevation 425’ may cause the direction of groundwater flow to be reversed from the northeast toward the west and south. Seepage will ultimately enter tributaries of the Broad River. Local domestic wells or springs or municipal and industrial wells are not located down gradient from the site and will not be affected by operation of the plant.

1.2.1.5 Geology

The Virgil C. Summer Nuclear Station site lies within the Piedmont Physiographic Province. The surface of the Piedmont Province consists of elevated, gently rolling hills which are separated on the northwest from the intensely folded and faulted Appalachian Mountains by intervening hills of the Blue Ridge Province, and overlapped on the southeast by sediments of the Coastal Plain Province. The Piedmont Province is essentially a dissected peneplain and is characterized by northeast-southwest trending belts of crystalline metamorphic and plutonic rocks. Piedmont rocks in general were found under many different combinations of geothermal and pressure conditions, and represent a complex succession of geologic events.

The Virgil C. Summer Nuclear Station site overlies complex zones of crystalline rocks, including migmatites in transitional areas between metamorphic rocks and injected igneous bodies. There are numerous joints in the rocks and small displacements have occurred along shears. Detailed geologic studies at the site and radiometric dating of rock samples from site excavations show latest movements along the shears occurred no later than 45 million years ago, and probably occurred 150 to 300 million years ago.

The Piedmont is generally covered by a deep mantle of residual soils derived by the inplace weathering of the underlying rock. The soil profile is typically characterized by an upper silty and clayey horizon overlying saprolite, which grades with depth to decomposed rock and unweathered rock. Soil strengths typically increase with depth. Transported soils are restricted to surficial veneers of alluvium near present day streams and isolated deposits of colluvium on lower slopes of some hills.

Detailed geologic information is provided in Section 2.5.

1.2.1.6 Seismology

Earthquake epicenters are scattered in the Piedmont and no causal faults have been identified.

The historic earthquake closest to the site occurred in 1945 about 5 miles west-southwest of the site. An epicentral intensity of Modified Mercalli (MM) VI has been assigned to this event due to the large area over which the effects were felt. The largest Piedmont earthquake within 200 miles of the site was the Union County shock of January 1913. Its epicenter was approximately 35 miles northwest of the site and its intensity is now regarded as VI-VII MM, recently downgraded from VII MM. The Safe Shutdown Earthquake (SSE) for the Virgil C. Summer Nuclear Station site is based upon this shock.

The largest earthquake in historic time within 250 miles of the site was the 1886 Charleston Earthquake, rated at an intensity of X MM. Its epicenter was about 125 miles southeast of the site in the Coastal Plain, within an area of recurring earthquake activity in the vicinity of Charleston, South Carolina.

Foundation conditions at the site are satisfactory from the standpoint of earthquake resistant design. Seismic Category I structures at the Virgil C. Summer Nuclear Station site are supported upon either rock or compact soils which are not adversely affected by earthquake motions.

Detailed seismic information, including reservoir induced seismicity, is included in Section 2.5.

1.2.1.7 Environmental Radiation Monitoring

Prior to operation of the Virgil C. Summer Nuclear Station, an environmental radiological surveillance will be initiated. Selected biota, air, water, soil and selected foods ingested by man and animals will be monitored. The survey will provide background data on the amount, type and source of radioactivity in the vicinity of the station. The data will be used for the following purposes:

1. Estimation of population exposure.

2. Verification of the adequacy of control of radioactivity sources.

3. Demonstration of the quality of the monitoring program with regard to its capability for providing useful data and the sensitivity and suitability of the detection equipment.

4. As a source of data for public information.

The surveillance program will continue subsequent to commencement of plant operation.

1.2.2 PRINCIPLE DESIGN CRITERIA

The criteria followed in the design of Virgil C. Summer Nuclear Station have been developed as performance criteria which define or describe safety objectives and procedures. They provide a guide to the type of design information included in this FSAR. These criteria are addressed in those FSAR sections to which they pertain. Criteria which are generally applicable to the design of Virgil C. Summer Nuclear Station are presented in Sections 1.2.2.1 through 1.2.2.7.

Virgil C. Summer Nuclear Station is designed to comply with the intent of "General Design Criteria for Nuclear Power Plants," Appendix A to 10 CFR 50. The specific applications of the General Design Criteria to Virgil C. Summer Nuclear Station are discussed in Section 3.1. Appendix 3A addresses NRC Regulatory Guides.

Compliance with NRC Regulations 10 CFR Parts 20, 50 and 100 is documented in a letter to the NRC dated November 14, 1980.

1.2.2.1 Quality Standards

Safety-related features of the plant are essential for the prevention of accidents which could affect the public health and safety and the mitigation of the consequences of such accidents. These features are designed, fabricated and erected to quality standards that reflect the importance of the safety function to be performed.

The Quality Assurance Program, described in Chapter 17, provides a planned program to ensure that safety-related activities are performed and documented to the appropriate degree. The Quality Assurance Program applies to safety-related systems, components and structures. This program continues through initial design, procurement, fabrication, site construction, erection, construction modifications, preoperational and startup testing and operation. Records are kept and audited to ensure proper program functioning.

Safety-related features of the plant essential to accident prevention and mitigation are the fuel cladding, Reactor Coolant System and containment barriers; the controls and emergency cooling systems, whose functions are to maintain the integrity of these barriers; systems which depressurize and reduce the contamination level of the reactor building; emergency power supplies and essential services to the above features; and the components employed to safely convey and store radioactive wastes and spent reactor fuel. Quality standards for material selection, design, fabrication and inspection governing the above features conform to the applicable provisions of recognized codes and standards.

Reviews are performed by other affected design disciplines (Structural, Electrical, Instrumentation and Control) to ensure that interface requirements are satisfied and that the design requirements are consistent with overall plant design.

1.2.2.2 Performance Standards

Certain features of the plant are essential to the prevention of accidents which could affect the public health and safety and the mitigation of the consequences of such accidents. The plant is designed to withstand, without loss of the capability to protect the public, the additional loadings imposed by the most severe earthquakes, flooding conditions, winds, ice or other natural phenomena which could conservatively be estimated to occur in the vicinity of the site.

The Nuclear Steam Supply System (NSSS) and other safety-related systems, including their support structures, are designed to withstand seismic disturbances conservatively predicted for the site, including the dynamic response of the structures to ground acceleration, based upon characteristics of the site foundation and damping of the foundation and structures.

Materials for structures, equipment and piping in Seismic Category I buildings are selected for compatibility with expected normal and accident environments in addition to selection based upon design and operational suitability.

1.2.2.3 Fire Protection

Primary emphasis is directed at minimizing the risk of fire by use of thermal insulation and adhesives which do not support combustion, flame retardant wiring, adequate electrical overload and short circuit protection, separation and elimination of combustible trim and furnishings. The plant is equipped with Fire Protection Systems for controlling and limiting the effects of fires. See Section 9.5.1 for a description of the Fire Protection System. Structures, systems and components important to safety, the loss of which would affect other systems, components or structures, are located to minimize the effects of fires and explosions.

1.2.2.4 Missile Protection

The Virgil C. Summer Nuclear Station is designed so that missiles from external or internal sources do not prevent safe shutdown of the plant or adversely affect the health and safety of the public.

Section 3.5 presents a discussion of missile protection measures.

1.2.2.5 Safety Classification

Classification of structures, systems, and components, in accordance with their importance to safety, is discussed in Section 3.2.

1.2.2.6 Vital Component Separation

To avoid common mode failures of redundant vital components due to occurrences, such as fire or missile impact, such components are physically separated in accordance with the following criteria:

1. Major Mechanical Equipment

   The physical separation criteria are applied to a number of major mechanical systems, including but not limited to the following safety-related systems and components:

   a. Reactor Building Cooling Units.

   b. Emergency Diesel Generators.

   c. Emergency Feedwater System components.

   d. Service Water System components.

   e. Main steam lines and valves to the outermost Main Steam Isolation Valves.

   f. Component Cooling Water System components.

   g. Reactor Building Spray System components.

   h. Post Accident Hydrogen Removal System components.

2. Piping

   Redundant portions of safety-related piping are protected where necessary by concrete barrier walls or by separation as discussed in Section 3.5.

3. Instruments and Controls

   Separation of redundant protection channels originates at the process sensors and continues through the wiring route and containment penetrations to the protection racks. Physical separation is used to the maximum practical extent to achieve separation of redundant transmitters. Separation of redundant wiring is achieved using adequately separated wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Redundant equipment is separated by locating modules in different protection rack sets. Each redundant protection channel set is energized from a separate instrument bus.

   The possibility of fires is reduced by using fire retardant materials, metal wireways and metal cabinets. Cables within trays are suitably derated. Loading is designed to minimize heat buildup. Trays are physically separated for fire protection. Safety-related cables are clearly identified to provide assurance that replacement, testing or maintenance activities are properly performed.

1.2.2.7 Common Mode Failure Consideration

Consideration is given to reduction of the potential for common mode failures in vital structures, systems and components during design.

Where possible, previously approved designs and designs which have been tested or proven in actual operations are utilized. These designs are updated where practical to include desirable modifications to improve system reliability.

Redundant components, piping, cabling and electrical equipment are protected through physical separation or by barriers. Pipe restraints are used to minimize pipe whip. Missile barriers are installed in areas where missiles from adjacent equipment may cause damage. Means are provided to ensure that temperatures suitable for equipment and personnel are maintained. Adequate instrumentation is provided to monitor system and component operation.

Designs are reviewed to ensure that assumptions, calculations and system layouts satisfy the requirements of the individual system and that the design does not adversely affect a redundant system or component (i.e., design of one system does not void the redundancy of another).

In addition to the normal reviews and tests, formal design reviews are conducted for safety-related designs by engineers not associated with the original design. Quality assurance personnel audit design review activities in accordance with the approved Quality Assurance Program.

Field installations are checked by Quality Control personnel to ensure conformity with installation drawings. Quality assurance personnel perform audit and surveillance functions in accordance with the approved Quality Assurance Program. Nonconformances in the installation of safety-related equipment are documented. The situation is reviewed and an appropriate disposition is made. If the field installation affects several design interfaces, the engineering disciplines affected review the situation and make a combined decision. Plans (drawings) are updated to show any field modifications.

Modifications to safety-related equipment or systems are reviewed and calculations and/or tests are performed as necessary to ensure that the modification does not prevent the system or equipment from performing its design function. After installation, system checkout in accordance with approved written procedures ensures proper system and component operation.

Design of the NSSS entails the following specific considerations:

1. The primary objective of Westinghouse design and quality assurance activities is to design and procure or manufacture equipment which is safe and reliable. An essential portion of this objective is the combination of policies and procedures which lead to prevention of common mode failures.

2. The cognizant System Engineer has the basic responsibility for translating license application and operational parameters into functional equipment requirements. The work of the Systems Engineer is verified through independent calculations or system testing. Functional requirements are provided to Equipment Design Engineers who develop equipment specifications or drawings. These design documents are reviewed by Licensing, Materials, Quality Assurance and interfacing disciplines as appropriate, to assure that the parts and fabrication processes specified are adequate for the production of equipment which will perform its intended function. Supplier detail designs are also reviewed by the cognizant engineer and other engineering groups to assure the adequacy of parts and processes. In addition, formal and comprehensive design reviews are selectively performed on equipment and system designs. These design reviews bring together experts from many disciplines to critically analyze the safety and reliability of the equipment or system. Design control processes are performed in accordance with written procedures which provide for documented results.

3. The design control measures of items 1 and 2 minimize the potential for common mode failures by:

    a. Assuring the use of previously qualified standard parts and processes. Where nonstandard parts or processes are required, their use is preceded by qualification tests to assure against misapplication.

    b. The multiple reviews by parties not directly involved in the design assure that adequate consideration is given to the effect of any potential malfunctions, including common mode failures. Typically, the effect of a malfunction is controlled by having the equipment fail in a safe mode, providing redundant equipment to assure continuing operation and including sufficient separation to prevent single external events from causing common mode failures of equipment performing parallel functions.

4. The effectiveness of these design techniques is confirmed by both system testing and operation. Component failure data are analyzed to determine the potential or actual effect on system function.

1.2.3 PLANT DESCRIPTION

1.2.3.1 <u>General</u>

Virgil C. Summer Nuclear Station incorporates a closed-cycle, pressurized water NSSS, a tandem compound turbine generator and necessary auxiliaries. Additional auxiliary systems, structures and other onsite facilities required for a complete, operational plant are also provided. The general arrangement of Virgil C. Summer Nuclear Station is shown by the plot plan and layout drawings, Figures 1.2-1 through 1.2-26.

As required by NUREG-0737, items I.D.1 and II.D.3, a Preliminary Control Room Design Review was performed by the Essex Corp. to identify significant human factors problems followed by a Human Factors Design Review/Audit conducted by the NRC Staff. A November 10, 1980 letter from R. L. Tedesco to SCE&G transmitted the NRC Audit Report. A copy of the Essex Preliminary Control Room Design Review Report is attached to the program plan for the human factors engineering evaluation and improvement of the Virgil C. Summer Nuclear Station Control Room, transmitted by a November 12, 1980 letter from SCE&G to the NRC. RN 98-171

This plan presents a program for resolving the problems identified in both the NRC Audit and the Essex Preliminary Review for completing a comprehensive evaluation of the control room after the NUREG-0700 Guidelines are issued. The Audit Reports and a draft of this plan were discussed with the NRC Human Factors Engineering Branch in an October 22, 1980 meeting.

Supplemental reports dated January 15, 1981 and November 25, 1981 describe specific methods of correcting each discrepancy identified in the NRC Audit and the Essex Preliminary Evaluation.

As required by NUREG-0578 Section 2.2.2.b, the Onsite Technical Support Center as shown on Figure 1.2-15 is located on the 463' elevation of the Control Building adjacent to the main control room. The center has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The habitability system for the center is the same as the one provided for the main control room. The records that pertain to the as-built conditions and layout of structures, systems and components are located in the center. Communications are provided between the Technical Support Center, and the Control Room, Operational Support Center, Emergency Operations Facility, Nuclear Regulatory Commission, state and local agencies, and key contractor organizations. More detailed information for the Onsite Technical Support Center can be found in Section 7.7.3.

As required by NUREG-0578 Section 2.2.2.c, the Operational Support Center, is located in the Control Building. In the event of an emergency, operations personnel not assigned to the Control Room and designated maintenance personnel will secure the operation in which they are performing and proceed to the Operational Support Center for emergency instructions and accountability.

The Emergency Operations Facility, required by NUREG-0737 and NUREG-0696, is located offsite within the SCE&G Joint Information Center (JIC) at 113 Ballentine Crossing Lane, Ballentine, SC (near the junction of SC Highway 176 and I-26). In the event of an emergency, the offsite emergency organization personnel will occupy this area in order to provide external support of the onsite emergency organization. RN 10-029

Communications are provided between the Technical Support Center and the Emergency Operations Facility. Additional communications are available with the Nuclear Regulatory Commission, state and local agencies, and key contractor organizations. More detailed information concerning the Emergency Operations Facility can be found in Section 13A.

The news media area will be established in the general location of the Emergency Operations Facility. Equipment and facilities in the news media area will allow the various media representatives to receive correct information concerning emergency conditions.

System diagrams are included with the appropriate system descriptions in various chapters of this FSAR. Symbols and abbreviations used on these system diagrams are illustrated by Figures 1.2-27 and 1.2-28. Figure 1.2-28a is an equipment list. RN 98-171

1.2.3.2 Structures

The major structures are the Reactor Building, Auxiliary Building, Fuel Handling Building, Intermediate Building, Control Building, Turbine Building, Diesel Generator Building, Service Water Intake Structure, Circulating Water Intake Structure, Water Treatment Building, Independent Spent Fuel Storage Installation and Service Building. The general arrangement of buildings is shown by Figures 1.2-1 through 1.2-26. RN 98-171 13-018

The reinforced concrete Reactor Building consists of a flat foundation mat, cylindrical wall and shallow dome roof. The foundation mat is reinforced with conventional mild steel reinforcing. The cylindrical wall is prestressed in the vertical and horizontal directions by a post tensioning system. The shallow dome roof is prestressed by a three-way post tensioning system. The inside surface of the Reactor Building is lined with a carbon steel liner to ensure a high degree of leak tightness under operating and accident conditions. The Reactor Building is designed to withstand postulated accidents as discussed in Chapter 15, severe environmental phenomena and to satisfy biological shielding requirements.

Seismic criteria used in the design of structures are discussed in Section 3.7.

1.2.3.3 Nuclear Steam Supply System

The NSSS (Figure 1.2-29) consists of a Pressurized Water Reactor, Reactor Coolant System and associated auxiliary fluid systems. The Reactor Coolant System is arranged as 3 closed reactor coolant loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump and steam generator. An electrically heated pressurizer is connected to the hot leg of 1 reactor coolant loop.

The initial reactor core is of the three-region cycled type. The core is composed of uranium dioxide pellets enclosed in zircaloy tubes with welded end plugs. The tubes are supported in assemblies by a spring clip grid structure. The mechanical control rods consist of clusters of stainless steel clad silver-indium-cadmium absorber rods and zircaloy guide tubes located within the fuel assembly. The fuel is loaded in 3 core regions. New fuel is introduced into the outer region, moved inward into a checkerboard pattern during successive refuelings and removed from the inner region to spent fuel storage. The typical reactor core fuel assembly arrangement is discussed in FSAR Section 4.3.2.1. RN 01-022 RN 01-022

The Reactor Coolant System is designed so that static and dynamic loads imposed upon reactor coolant pressure boundary components as a result of an inadvertent and sudden pressure change does not cause rupture of the pressure boundary. To continually ensure system integrity, the components containing reactor coolant pressure have provisions for inspection and testing to assess the structural and leaktight integrity of the boundary components during their service lifetime.

The reactor coolant pumps are vertical, single stage, centrifugal pumps, equipped with controlled leakage shaft seals.

The steam generators are vertical, U-tube type heat exchangers with Alloy 690 tubes. Integral moisture separating equipment is designed to reduce the moisture content of the steam to 0.1% or less.

The pressurizer is a vertical, cylindrical, pressure vessel with hemispherical heads, equipped with electric heaters and spray nozzles for system pressure control.

High pressure water circulates through the reactor core to remove the heat generated by the controlled nuclear chain reaction. The heated water exits the reactor vessel and passes through the reactor coolant loops to the steam generators. In the steam generators, heat from the reactor coolant is transferred to the feedwater, generating steam. The cycle is completed when the reactor coolant is pumped from the steam generators to the reactor vessel. The entire Reactor Coolant System is composed of leaktight or controlled leakage components, thus ensuring that reactor coolant is confined within the system or associated auxiliary systems.

Auxiliary system components are provided to charge the Reactor Coolant System and add makeup water, collect leakage, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control, cool system components, remove residual heat when the reactor is shut down and for emergency core cooling.

1.2.3.4 Engineered Safety Features

Engineered Safety Features (ESF) are provided to mitigate the consequences of postulated accidents up to and including a design basis accident as discussed in Chapter 15. The ESF systems provided for Virgil C. Summer Nuclear Station have sufficient redundancy and independence of components and power sources that, under postulated accident conditions, the following are accomplished:

1. Core cooling limits the core thermal transient, thereby preventing excessive metal-water reactions and maintaining coolable core geometry.

2. Reactor Building structural integrity is maintained.

3. Radiation dose to the public is maintained within the limits of 10 CFR 100.11 and 10 CFR 50.67. RN 12-034

The ESF systems provided are summarized in Sections 1.2.3.4.1 through 1.2.3.4.5. Further details of the ESF system are available in Section 7.3. RN 98-171

1.2.3.4.1 Containment System

The steel lined, prestressed, reinforced concrete Reactor Building provides a barrier against the escape of fission products should a Loss of Coolant Accident (LOCA) occur. The general containment design parameters are described in Section 1.2.3.2.

1.2.3.4.2 Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) injects borated water into the Reactor Coolant System under high and low reactor coolant pressure conditions. This limits any potential damage to the core and limits the release of energy and fission products into the reactor building following a LOCA. The ECCS also inserts negative reactivity into the core through injection of borated water following a steam line break accident or after an accidental steam release.

Following a LOCA, the reactor core is cooled by automatic injection of water from the passive accumulators into the reactor coolant loops or by automatically pumping water from the Refueling Water Storage Tank (RWST) using the charging pumps and Residual Heat Removal (RHR) pumps. The charging pumps discharge to the reactor coolant loops. The RHR pumps discharge to the reactor coolant loop cold legs.

Long term emergency core cooling is accomplished by operation of the charging pumps and RHR pumps to circulate spilled coolant and previously injected water from the reactor building recirculation sumps.

1.2.3.4.3 Reactor Building Heat Removal Systems

The Reactor Building heat removal systems cool the Reactor Building atmosphere following a LOCA or steam line break. Thus, Reactor Building pressure and energy content are reduced. The Reactor Building heat removal systems include the Reactor Building cooling units and the Reactor Building Spray System.

The Reactor Building cooling units contain fans and heat exchangers to condense the steam and transfer heat to the Service Water System. These units also provide cooling under normal conditions; however, the cooling fluid is supplied by the Industrial Cooling Water System rather than the Service Water System.

The Reactor Building Spray System reduces the energy content of the Reactor Building by spraying the interior with a relatively cool, alkaline, borated water solution. The spray solution also reduces radioactivity levels since it absorbs radioactive iodine from the Reactor Building atmosphere. During the injection phase, a mixture of water from the RWST and sodium hydroxide tank is sprayed into the Reactor Building interior.

1.2.3.4.4 Combustible Gas Control

Two (2) hydrogen recombiners of the Post Accident Hydrogen Removal System remove hydrogen from the Reactor Building atmosphere following a LOCA by combining the hydrogen with oxygen to form water. Thus, the Reactor Building hydrogen concentration is maintained within allowable limits. An alternate purge system is provided as a backup to the Post Accident Hydrogen Removal System.

1.2.3.4.5 Containment Isolation System

The Containment Isolation System isolates pipe lines not essential to operation of other ESF systems which penetrate the containment boundary. Thus, under severe accident conditions, radioactivity inside the Reactor Building is not released to the environment.

1.2.3.5 <u>Instrumentation and Control</u>

Plant instrumentation and control systems are provided to monitor significant variables in the reactor core, Reactor Coolant System and Reactor Building over their anticipated ranges. The installed instrumentation provides capability for continuous monitoring, maintenance and initiation of required safety functions. More detailed information is provided in FSAR Section 7.0. 00-01

Overall reactivity control is achieved by a combination of chemical shim and Rod Cluster Control Assemblies (RCCA). Long term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short term reactivity control for power level changes is accomplished by moving RCCAs.

The control system allows the plant to accept a step load increase or decrease of 10% and a ramp load increase or decrease of 5% per minute within the load range of 15% to 100%, subject to possible xenon limitations. The steam dump system permits 100% load rejection without turbine or reactor trip.

A control room and technical support center, designed for safe occupancy under both normal and accident conditions, are provided. Adequate shielding is included to maintain reasonable radiation levels. In the unlikely event that the control room is uninhabitable, total controls are provided to effect hot shutdown from outside the control room and to maintain the plant in a hot shutdown condition for an extended period of time. It is also possible to achieve cold shutdown under such conditions (refer to SCE&G letters to the NRC dated October 16, 1981, November 9, 1981 and November 11, 1981). More detailed information is provided in FSAR Sections 7.4.1.3 and 7.4.1.4. 00-01

1.2.3.6 Electrical Systems

The main generator is an 1800 rpm, 24 kV, 3 phase, 60 hertz, conductor cooled unit. The main step-up transformer delivers power to the substation at 230 kV. Transmission lines interconnect with the SCE&G and SCPSA transmission systems from this substation at 230 kV.

The station service system consists of auxiliary transformers; 7200 volt and 480 volt switchgear and buses; 480 volt motor control centers; 115 volt A-C, vital instrument buses; 125 volt D-C batteries, chargers, inverters and equipment.

The normal source of Balance Of Plant (BOP) station service power is the unit auxiliary transformer which is connected to the generator isolated phase bus duct. The preferred power supplies for the engineered safety features buses are from a 115 kV line which is part of the SCE&G transmission system and a second power supply from the local 230 kV substation.

Two (2) independent diesel generators are provided in the event of complete loss of system power. Each diesel generator has sufficient capacity for operation of engineered safety features equipment which must be operated in the event of a LOCA or other emergency conditions.

1.2.3.7 Fuel Handling and Storage Systems

The reactor is refueled by equipment designed to handle spent fuel underwater from the time the spent fuel is removed from the reactor vessel until it is placed in a cask for removal from the Fuel Handling Building. Underwater transfer of spent fuel provides an effective, economic and transparent radiation shield and a reliable cooling medium for removal of decay heat. RN 13-018

The Fuel Handling System includes 2 areas: the refueling cavity and refueling canal which are flooded only during plant shutdown for refueling; and the spent fuel pool and transfer canal which are kept full of water and are accessible to operating personnel. The refueling canal and transfer canal are connected by the fuel Transfer System.

Spent Fuel is removed from the reactor vessel by a refueling machine and is placed in the fuel Transfer System carriage. The Fuel Transfer System moves the spent fuel from the refueling canal to the transfer canal. The spent fuel is removed from the Fuel Transfer System carriage in the transfer canal and placed in the spent fuel storage racks in the spent fuel pool. After a decay period the spent fuel is removed from the spent fuel racks and loaded into a cask for removal from the Fuel Handling Building. RN 13-018

Handling and storage of both new and spent fuel is performed using equipment and procedures which ensure that the fuel remains subcritical.

1.2.3.8 Auxiliary Systems

1.2.3.8.1 Service Water System

The Service Water System is vital to plant safety and is therefore equipped with redundant pumps, heat exchangers, flow circuits, etc. System design is such that no single failure will result in loss of cooling to necessary equipment. The service water pumps take suction from the service water pond and supply cooling water to the component cooling heat exchangers, diesel generators, HVAC (Heating, Ventilating and Air Conditioning) mechanical water chiller condensers, service water pump house ventilation cooling coils and, under emergency conditions, to the reactor building cooling units. The normal supply of cooling to the reactor building cooling units is from the industrial cooling water system. 98-01

1.2.3.8.2 Cooling Systems

The cooling systems include the Component Cooling Water System, Spent Fuel Cooling System, Residual Heat Removal System, and Chilled Water System. These systems are closed loop systems which can receive makeup water from primary grade water sources or the Demineralized Water System. For a more detailed description of the make-up sources, consult the 302 drawings in the pertinent FSAR section for these systems. 02-01

1.2.3.8.3 Ultimate Heat Sink

The ultimate heat sink, the safety class service water pond, provides sufficient cooling water for normal and postulated accident conditions.

1.2.3.8.4 Other Water Systems

The Demineralized Water Makeup System pretreats and demineralizes water from Monticello Reservoir and provides for storage of demineralized water for primary and secondary plant makeup. Potable water is obtained from the system following pretreatment but prior to demineralization.

The Potable Water System serves sanitary plumbing fixtures, showers, laundry and emergency eye wash units. Sanitary drainage from plumbing fixtures served by the Potable Water System and toilet room floor drains is collected by a drainage system which terminates at an onsite sanitary disposal facility.

1.2.3.8.5 Condensate Storage Facilities

Condensate storage is provided in the form of a 500,000 gallon tank, of which approximately 160,000 gallons is reserved for the Emergency Feedwater System.

1.2.3.8.6 Compressed Air Systems

Two (2) nonsafety-related compressed air systems are provided. One system supplies Reactor Building instrument air; the other supplies instrument and service air to the balance of the plant. These systems provide compressed air of suitable quality and at the pressures required for use by various components. Certain air operated devices are required to operate after an accident. These devices utilize air volume tanks to perform these functions. The devices and justification for their uses are provided in the appropriate section which discusses associated fluid system operation.

1.2.3.8.7 Process Sampling System

The sampling systems collect primary and secondary plant gaseous and liquid samples as required for radiochemical and chemical analyses.

1.2.3.8.8 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) adjusts boron concentration in the reactor coolant, maintains proper Reactor Coolant System water inventory, provides required seal water flow to the reactor coolant pump shaft seals, provides high pressure flow to the ECCS, maintains proper concentrations of corrosion inhibiting chemicals in the reactor coolant and reduces the inventory of corrosion and fission products in the reactor coolant.

The Boron Thermal Regeneration System (BTRS) accepts borated water letdown from the Reactor Coolant System and returns it with boron removed as necessary based upon operating requirements. RN 99-075

1.2.3.8.9 Boron Recycle System

The Boron Recycle System processes reactor coolant effluent to obtain reusable boron and purified water.

1.2.3.8.10 Air Conditioning, Heating, Cooling and Ventilation Systems

The following ventilation systems are provided:

1. Control Building, including control room, relay room, cable spreading areas, computer room and controlled access area.

2. Fuel Handling Building, including spent fuel pool area.

3. Auxiliary Building.

4. Turbine Building.

5. Intermediate Building.

6. Reactor Building.

7. Diesel Generator Building.

8. Service Water Intake Structure.

9. Circulating Water Intake Structure.

10. Service Building.

Each ventilation system is provided to maintain a suitable environment for equipment and personnel and to filter effluents as necessary.

1.2.3.8.11 Fire Protection System

The Fire Protection System is designed to detect, extinguish and mitigate the effects of fires. Details are presented in Section 9.5.1.

1.2.3.8.12 Communications Systems

Reliable systems are provided for both onsite and offsite communications. Details are presented in Section 9.5.2.

1.2.3.8.13 Lighting Systems

The lighting system provides adequate lighting under normal operating and postulated accident conditions. Details are presented in Section 9.5.3.

1.2.3.8.14 Diesel Generator Auxiliaries

The diesel generator auxiliaries consist of the following:

1. Bulk Fuel Oil Storage and Transfer System.
2. Fuel oil day tank.
3. Jacket Water and Air Intercooler Cooling System.
4. Starting System.
5. Lubricating Oil System.
6. Combustion air intake and exhaust equipment.
7. Generator exciter/regulator.
8. Engine-generator control panel.

The auxiliaries for each diesel generator are designed to ensure independence from the other diesel generator.

1.2.3.9 Steam and Power Conversion System

1.2.3.9.1 Turbine Generator

The turbine is a General Electric tandem compound, 4 flow exhaust, 1800 rpm unit. The overspeed protection system operates through the turbine electrohydraulic control system. The overspeed protection system is designed so that no single failure can lead to destructive turbine overspeed. Provision is made for inservice testing of the turbine protection system. 99-01

1.2.3.9.2 Main Steam System

The Main Steam System delivers steam from the steam generators to the turbine generator, main feedwater pump turbines, steam driven emergency feedwater pump turbine, reheaters and auxiliary steam system.

Connections from the Main Steam System to the Main Steam Dump System are provided. Overpressure protection is provided by a total of 15 safety valves, 5 in each steam header, outside the Reactor Building upstream of the main steam isolation valves. One (1) power operated relief valve is also provided for each steam header. One (1) quick closing main steam isolation valve, capable of closing tightly regardless of steam flow direction, is provided in each steam header, just outside of the Reactor Building.

1.2.3.9.3 Main Condenser

The main condenser is a twin shell, dual pressure unit with circulating water arranged to pass through the two (2) sections in series. Condensate is collected in the hotwell section of each shell, which are interconnected. The hotwell storage level is controlled to provide sufficient condensate retention to accommodate sudden increases in load demand with makeup provided by the Condensate Storage Tank. A duplex, closed feedwater heater is horizontally mounted in each condenser neck. 99-01

1.2.3.9.4 Condenser Air Removal System

Evacuation of each main condenser shell is accomplished by a mechanical vacuum pump. The auxiliary condenser shells are piped and normally aligned to vent to the low pressure shell of the main condenser. Mechanical vacuum pumps are provided for backup air removal of the auxiliary condensers. Non-condensible gases removed from the condensate are monitored to detect radioactivity. Normally such gases are discharged through the Auxiliary Building charcoal filters. Provision is made for discharge of noncondensibles to the environment if no radioactivity is present. 99-01

1.2.3.9.5 Turbine Gland Seal System

The Turbine Gland Seal System prevents steam leakage from the high pressure turbine shaft seals and air leakage into the condenser through the low pressure turbine shaft seals.

1.2.3.9.6 Turbine Bypass System (Main Steam Dump System)

Connections to the Main Steam System are provided for the Turbine Bypass System. The Turbine Bypass System contains 8 bypass control valves for steam dump to the main condenser, 3 bypass control valves for steam dump to atmosphere and 3 power relief valves. The maximum flow capacity of each individual valve is limited to prevent uncontrolled plant cooldown should a single valve fail in the open position.

1.2.3.9.7 Circulating Water System

The Circulating Water System provides cooling water to the main and auxiliary condensers. Cooling water is supplied from Monticello Reservoir through an intake structure. Circulating water pumps, traveling water screens and associated screen wash pumps, a vacuum priming system and a condenser tube cleaning system are provided.

1.2.3.9.8 Condensate and Feedwater System

Condensate is withdrawn from the high pressure condenser hotwell by 3 motor driven condensate pumps. The pumps discharge into a common header which carries condensate to the steam packing condenser. Downstream of the steam packing condenser, the common header divides into 2 lines which carry condensate through 3 stages of low pressure feedwater heaters. Effluent from each of these 2 lines enters the deaerator. 99-01

Feedwater booster pumps take suction from the deaerator storage tank and discharge to the suction of the feedwater pumps which discharge through 2 stages of half capacity, high pressure feedwater heaters, arranged in parallel, to a discharge header. Feedwater from the discharge header is distributed to each steam generator through individual feedwater flow control valves. 99-01

During startup and shutdown and as required during condenser leaks, the condensate cleanup system will be used. The cleanup system is located between the condensate pump discharge and steam packing condenser. 02-01

1.2.3.9.9 Steam Generator Blowdown

The Steam Generator Blowdown and Nuclear Blowdown Processing Systems control the chemistry of the secondary side by continuous blowdown of the steam generator. Normally, blowdown from each steam generator is cooled and directed to the Nuclear Blowdown Processing System or, alternatively, discharged into the circulating water discharge.

An abnormally high level of blowdown radioactivity activates an alarm and automatically diverts the fluid to the Nuclear Blowdown Processing System. Effluent from the Nuclear Blowdown Processing System is normally returned to the Condensate System via the main condenser. Effluent can also be discharged to the penstocks. 02-01

1.2.3.9.10 Emergency Feedwater System

One (1) full capacity, steam turbine driven and 2 100% capacity, motor driven emergency feedwater pumps are provided to supply feedwater to the steam generators during loss of offsite power, under postulated accident conditions or during plant startup and shutdown. Steam is supplied to the steam turbine driven emergency feedwater pump from the Main Steam System; electric power is supplied to the motor driven emergency feedwater pumps from the Class 1E power system. Feedwater flow to individual steam generators is controlled by a remotely operated control valve in each supply line. RN 98-012 RN 98-012

1.2.3.9.11 Auxiliary Steam System

The Auxiliary Steam System, consisting of the auxiliary steam supply header and auxiliary boiler, supplies steam throughout the plant for various auxiliary services. During normal operation, auxiliary steam is supplied by extraction steam from the main turbine. During shutdown periods, the auxiliary boiler supplies auxiliary steam. Under low load or startup conditions, auxiliary steam is available from the Main Steam System.

1.2.3.9.12 Turbine Building Closed Cycle Cooling Water System

The Turbine Building Closed Cycle Cooling Water System supplies cooling water to steam and power conversion equipment and utilizes a wet surface industrial cooling tower for the heat sink.

1.2.3.10 Radioactive Waste Management

Radioactive waste management is accomplished through use of 3 interrelated systems:

1. Liquid Waste Processing System.
2. Gaseous Waste Processing System.
3. Solid Waste Disposal System.

The Liquid Waste Processing System collects and processes potentially radioactive waste liquids generated during plant operation and refueling, for either recycling within the plant or for discharge. Radioactive constituents are removed or concentrated and processed until suitable for discharge or shipment offsite.

The Gaseous Waste Processing System removes fission product gases from the reactor coolant and allows for retention or controlled release.

The Solid Waste Disposal System provides the capability to process and package solid waste in Department of Transportation approved containers for shipment offsite or for storage onsite (refer to Chapter 11).

1.2.3.11 Radiation Monitoring System

The Radiation Monitoring System detects, indicates, annunciates and records radiation levels at selected locations throughout the plant. Three (3) subsystems are provided: area gamma monitoring, atmospheric monitoring, and liquid monitoring. Fixed monitors display their measured variable and any alarm condition in the control room except for monitors associated with the radwaste solidification equipment (refer to Chapter 11).

NOTE 1.3
Section 1.3 is being retained for historical purposes only. 99-01

1.3 COMPARISON TABLES

1.3.1 COMPARISON WITH SIMILAR FACILITY DESIGNS

Table 1.3-1 provides significant similarities and differences between the Virgil C. Summer Nuclear Station and other designs.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

Table 1.3-2 provides significant design changes since submittal of the Preliminary Safety Analysis Report.

TABLE 1.3-1

DESIGN COMPARISON WITH SIMILAR FACILITIES [(1)]

Chapter Number	Chapter Title System/Component	Virgil C. Summer Nuclear Station FSAR References	Significant Similarities	Significant Differences	
4.0	Reactor	Section 4.0	Similar to North Anna Units 1 and 2 total 1 NSSS heat output of 2785 MWt.	Beaver Valley Unit 1 has a total NSSS heat output of 2660 MWt.	02-01
	Fuel	Section 4.2.1	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2.	Small variations in design parameters based on nuclear and thermal-hydraulic design.	
	Reactor Vessel Internals	Section 4.2.2	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2.	Beaver Valley Unit 1 and North Anna Units 1 and 2 have thermal shields; Virgil C. Summer Nuclear Station has neutron pads.	
	Reactivity Control Systems	Section 4.2.3	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2.	None.	02-01
	Nuclear Design	Section 4.3	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2	Small variations in nuclear parameters.	
	Thermal-Hydraulic Design	Section 4.4	Similar to Beaver Valley Unit 1 and North Anna Unit 1 and 2	Small variations in thermal-hydraulic and heat transfer parameters.	

(1) Comparisons identified relative to thermal power are compared to the pre-uprate power level of 2775 MWt RTP.

TABLE 1.3-1 (Continued)

DESIGN COMPARISON WITH SIMILAR FACILITIES

Chapter Number	Chapter Title System/Component	Virgil C. Summer Nuclear Station FSAR References	Significant Similarities	Significant Differences
5.0	Reactor Coolant System	Sections 5.1, 5.2	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2.	Virgil C. Summer Nuclear Station does not have loop stop valves.
	Reactor Vessel	Section 5.4	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2.	No significant differences.
	Reactor Coolant Pumps	Section 5.5.1	The hydraulics are similar to Beaver Valley Unit 1 and North Anna Units 1 and 2	Virgil C. Summer Nuclear Station motors are of modular construction.
	Steam Generators	Section 5.5.2	Similar to Wolfe Creek & Millstone Unit 3.	Alloy 690 tube material, triangular tube pitch, larger heat transfer area.
	Piping	Section 5.5.3	Similar to Surry Units 1 and 2	Virgil C. Summer Nuclear Station has no electroslag welding.
	Residual Heat Removal System	Section 5.5.7	The piping and fittings are similar to Joseph M. Farley Units 1 and 2	No significant differences.
	Pressurizer	Section 5.5.10	Similar to North Anna Units 1 and 2	No significant differences.
6.0	Engineered Safety Features Electric Hydrogen Recombiner	Section 6.2.5	Similar to Surry Units 1 and 2	No significant differences. 02-01
	Emergency Core Cooling System	Section 6.3	Similar to Joseph M. Farley Units 1 and 2	Automatic sump valve opening logic is different.

TABLE 1.3-1 (Continued)

DESIGN COMPARISON WITH SIMILAR FACILITIES

Chapter Number	Chapter Title System/Component	Virgil C. Summer Nuclear Station FSAR References	Significant Similarities	Significant Differences	
7.0	Instrumentation and Controls Reactor Trip System	Section 7.2	Similar to Joseph M. Farley, Unit 1	No significant differences.	02-01
	Engineered Safety Features Actuation System	Section 7.3	Similar to Joseph M. Farley, Unit 1	No significant differences.	
	System Required for Safe Shutdown	Section 7.4	Similar to Joseph M. Farley, Unit 1	No significant differences.	
	Safety-Related Display Instrumentation	Section 7.5	Similar to Joseph M. Farley, Unit 1	No significant differences but physical display may differ.	
	Other Safety System	Section 7.6	Similar to Joseph M. Farley, Unit 1	No significant differences.	02-01
	Control Systems	Section 7.7	Similar to Joseph M. Farley, Unit 1	Joseph H. Farley, Unit 1 has a 50% load rejection capability. Virgil C. Summer Nuclear Station has an 85% load rejection capability.	

TABLE 1.3-1 (Continued)

DESIGN COMPARISON WITH SIMILAR FACILITIES

Chapter Number	Chapter Title System/Component	Virgil C. Summer Nuclear Station FSAR References	Significant Similarities	Significant Differences	
8.0	Electric Power				
	Offsite Power	Section 8.2		Three Mile Island Nuclear Station Unit 1 has two offsite sources, 230 kV/6.9 kV. Virgil C. Summer Nuclear Station has one 115 kV/7.2 kV and one 230 kV/7.2 kV offsite source.	02-01
	Onsite Power	Section 8.3	Joseph M. Farley Units 1 and 2 have Colt Industries 12PC2V diesel engines	Three Mile Island Nuclear Station Unit 1 has four 125 volt batteries for supply of vital d-c power. Virgil C. Summer has two 125 volt batteries for supply of vital d-c power.	
	120 Volt Vital a-c Bus System		Three Mile Island Nuclear Station Unit 1 normal source of power is 480 volt a-c ESF bus; alternate source is 125 volt d-c system.	Three Mile Island Nuclear Station Unit 1 has four 15kVA single phase inverters. Virgil C. Summer Nuclear Station has six 7.5kVA single phase inverters, arranged in four independent channels.	
9.0	Auxiliary System Fuel Handling System	Section 9.1.4	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2	Each project has difference capacities for spent fuel storage.	02-01
	Chemical and Volume Control System	Section 9.3.4	Similar to Joseph M. Farley Units 1 and 2	No significant differences.	

TABLE 1.3-1 (Continued)

DESIGN COMPARISON WITH SIMILAR FACILITIES

Chapter Number	Chapter Title System/Component	Virgil C. Summer Nuclear Station FSAR References	Significant Similarities	Significant Differences
10.0	Radioactive Waste Management Source Terms	Section 11.1	Similar to Joseph M. Farley Units 1 and 2	Differences in source term analysis.
	Liquid Waste Processing	Section 11.2	Performance characteristics similar to Joseph M. Farley Units 1 and 2	Differences in source term analysis.
	Gaseous Waste Processing	Section 11.3	Similar to Joseph M. Farley Units 1 and 2	Differences in source term analysis.
14.0	Initial Tests and Inspections	Chapter 14	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2.	No significant differences.
15.0	Accident Analysis	Chapter 15	Similar to Beaver Valley Unit 1 and North Anna Units 1 and 2	The accident analysis sections have been updated. New sections have been added, e.g., single RCCA withdrawal, accidental depressurization of the RCS, code descriptions, etc.

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TABLE 1.3-2

COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Item	FSAR Reference	Change
Service Water Pond Bottom and North Dam Foundation	2.5.6	The following changes were made to the service water pond as conservative measures to reduce the potential for seepage losses from the pond during a postulated loss of the Monticello Reservoir: a. Installation of a grout curtain along the centerline of the North Dam foundation. b. Placement of a clay blanket on the portion of the pond bottom adjacent to the South Dam.
Auxiliary Building Charcoal Exhaust System	Table 3.2-1 and 9.4.2.2.3	The auxiliary building charcoal exhaust system was declassified from SC-3 to non-nuclear safety class. Release calculations indicated that this system was not necessary to maintain offsite dose rates within acceptable limits. However, areas exhausted were increased and HEPA filters were added. 99-01
Change in Seismic Instrumentation	3.7.4	Seismic instrumentation program was changed to meet the required sensitivity of instrumentation to measure the seismic response of nuclear power plant features important to safety.
Increase of Reactor Building Internal Design Pressure	3.8.1.3.1.2 and 6.2.1	The reactor building internal design pressure was increased from 55 psig to 57 psig to maintain the required margin above the calculated pressure.

TABLE 1.3-2 (Continued)

COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Item	FSAR Reference	Change
Compressive Strength of 5000 psi Concrete was Changed from 28 day Test to 90 day Test	3.8.1.6.1.2.1	The heat generated during the curing period of mass concrete calculated from results obtained from actual field tests, utilizing the actual dimensions of the reactor building mat, was higher than acceptable. The reduction in concrete mix cement content and consequent lower temperature in the concrete was acceptable by using the 90 day strength as a design requirement.
Control Building Foundation	3.8.5.1.4	The control building foundation was changed from caissons to mat on fill concrete.
Fuel	4.2.1	The reactor will be fueled with 17 x 17 fuel assemblies in lieu of 15 x 15 fuel assemblies.
Reactor Internals	4.2.2	The reactor internals have been modified to accept 17 x 17 fuel assemblies.
		The thermal shield has been replaced by neutron pads. This change simplifies core support design and reduces flow pressure drop and velocity.
Fuel Pellet Enrichments and Density	4.3	The fuel pellet enrichments as well as other core parameters change between the PSAR and FSAR to reflect the evolution of the design as core performance and safety requirements are met. Fuel density has been increased from 94, 92, and 91 percent of theoretical to 95 percent of theoretical.

TABLE 1.3-2 (Continued)

COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Item	FSAR Reference	Change
Burnable Poison Loading Pattern	4.3	The burnable poison pattern shown in the FSAR reflects more detailed calculations than that of the PSAR.
Reactor Top Head Penetrations	5.4	Reactor vessel top head penetrations and CRDM were redesigned to meet requirements for inservice inspection.
Reactor Vessel Nozzle Insulation	5.4	Non crushable insulation is used.
Reactor Building Ventilation System	6.2.2 and 9.4.8	HEPA filters and bypass dampers were added to the reactor building ventilation system to reduce offsite releases following a LOCA.
Post Accident Hydrogen Recombiner	6.2.5	Change of post accident hydrogen recombiners from externally mounted to internally mounted units.
Semi-Automatic Emergency Core Cooling System (ECCS) Switchover	6.3	System was revised to automatically change residual heat removal (RHR) pump suction from the refueling water storage tank (RWST) to the recirculation sump of the reactor building during changeover from the injection phase to the recirculation phase during post accident core cooling. This revision ensures continual water supply to the suction of the RHR pump.
Automatic Reactor Building Spray Switchover	6.3.2	System was revised to automatically change spray pump suction from the RWST to the recirculation sump of the Reactor Building on lo-lo level in the RWST. This revision ensures continuous water supply to the suction of the pumps. 99-01

TABLE 1.3-2 (Continued)

COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Item	FSAR Reference	Change
Elimination of Redundancy of Instrument Air Supply for Ventilation Control in the Control Room	6.4 and 9.4.1	Since the control air system is not nuclear safety-related and all dampers/valves fail to the post accident mode, redundancy is not required.
Steam Dump	7.2	Power operated steam relief valves were automated to increase dump capacity.
Undervoltage Reactor Trip	7.2	The undervoltage sensors and underfrequency sensors were moved to the motor side of the reactor coolant pump breaker. Trip on pump breaker open was removed.
Safety Injection	7.2	Injection signal on pressurizer low pressure in lieu of coincident low level and low pressure.
Feedwater Control	7.2	The programmed setpoint for the feedwater control valve uses steam flow/feed flow mismatch and S/G level. The bypass valves use S/G level and neutron flux to control valve position. Also, the feedwater pump speed is programmed using line pressures and steam flow. 02-01
Main Steam Isolation Signal	7.2	Changed the main steam isolation from activating on high steam line flow coincident with low steam line pressure or lo-lo T_{avg} to low steam line pressure or high steam line flow coincident with lo-lo T_{avg}.
Blowdown and Sample Line Isolation	7.2	Automatic initiation of emergency feedwater causes isolation. Manual isolation was removed.

TABLE 1.3-2 (Continued)

COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Item	FSAR Reference	Change
Changes to Manual Actuation of the following Three Systems: Main Steam Isolation, Containment Isolation Reactor Building Spray and Automatic Initiation of Reactor Building Spray	7.2, Figure 7.2-1 and Table 7.3-2	Changes have been made to these systems to provide the required redundancy for system initiation.
Post Accident Monitoring	7.5	A post accident monitoring system has been added.
Analog to Digital	7.7	Analog rod position indication has been replaced by digital rod position indication to improve reactor protection instrumentation.
Incorporation of a Circuit Breaker between the Generator Terminals and the Unit Auxiliary Transformer	8.2.1.1	The addition of this circuit breaker improves operational flexibility by permitting the use of the unit auxiliary transformer during startup and eliminating the need of station service transfer during turbine trip.
Physical Arrangement and Separation Criteria Requirements of Installed Electrical Systems and Equipment	8.3.1.1 and 8.3.2.1	Electrical equipment such as motor control centers, batteries, battery chargers, distribution panels, diesel generators, etc., have been located to provide protection and/or separation from postulated accidents (i.e. missiles, seismic event, fire hazards, etc.).
Removal of Bus Tie Breaker between Class 1E d-c Buses 1A and 1B	8.3.2.1.2 and 8.3.2.2.1	The manually operated bus tie breaker between Class 1E d-c buses 1A and 1B was removed to improve system reliability and safety.

TABLE 1.3-2 (Continued)

COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Item	FSAR Reference	Change
Diesel Generator Fuel Oil System	8.3.1.1.2.1 and 9.5.4	The following changes have been made to the diesel generator fuel oil system: a. Incorporation of two independent fuel oil storage systems, including independent buried storage tanks, in lieu of a common storage system. b. Incorporation of two a-c motor driven transfer pumps per system in lieu of one a-c and one d-c motor driven pump. c. Reduction in operating time for fuel available in day tanks to a nominal 1.5 hours to be consistent with the maximum size of day tank permitted by National Fire Protection Association Standard (NFPA) 37.
Spent Fuel Racks	9.1	The storage lattice spacing has been reduced to 14 inches from 21 inches on center to increase storage from approximately 1-1/3 core to 4-1/3 core.
Reactor Building Instrument Air System	9.3.1	The reactor building instrument air system was revised to: a. Delete one air compressor. b. Relocate remaining two compressors outside the reactor building.

TABLE 1.3-2 (Continued)

COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Item	FSAR Reference	Change
Nuclear Sampling System	9.3.2	Recent regulatory guidelines permit declassification of nuclear sampling system piping outside reactor building without reduction in system reliability or performance.
Fuel Handling Building Charcoal Exhaust System	9.4.3.2.1	Additional areas were added to the exhaust system to reduce normal plant releases. Redundant plenums and additional HEPA filters were added. 99-01
Steam Generator Blowdown System	10.4.8	The steam generator blowdown system was redesigned to accommodate the changes in the secondary cycle chemistry control from phosphate to all volatile treatment (AVT).
Redesign of Radiation Monitoring	11.4, 12.1.4 and 12.2.4	Various changes to the radiation monitoring system to increase surveillance and detection capabilities of plant effluents.
Automatic Emergency Feedwater Switchover from Condensate Tank to Service Water	10.4.9	System was revised to automatically change emergency feedwater pump suction from the condensate storage tank to the service water system on lo suction pressure to the EFW pumps. This revision ensures continuous water supply to the suction of the pumps.
ATWS Mitigation System Actuation Circuitry (AMSAC)	7.8	Added instrumentation and controls necessary to respond to 10CFR50.62 for mitigation of ATWS events.

NOTE 1.4 Section 1.4 is being retained for historical purposes only.

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1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

South Carolina Electric and Gas Company (SCE&G) is the sole applicant for the construction permit and operating license for the Virgil C. Summer Nuclear Station. SCE&G is two-thirds owner of the Virgil C. Summer Nuclear Station. The remaining one-third is owned by the South Carolina Public Service Authority (SCPSA) which together with SCE&G owns the Virgil C. Summer Nuclear Station as tenants in common. It is specifically noted that SCE&G is the agent for the SCPSA with regard to the Virgil C. Summer Nuclear Station, and that SCE&G retains sole responsibility for the overall technical direction in the licensing, design, construction, operation, management, maintenance and decommissioning. Likewise, it is specifically noted that the ownership change reflected in the amended application does not effect a transfer of control from SCE&G to the SCPSA, nor does the ownership change involve a physical or design change in the facility or site.

Gilbert Associates, Inc. (Gilbert), has been retained by SCE&G as architect engineer for the entire project, including plant layouts and system arrangements, and design of balance of plant equipment. A brief summary of Gilbert technical qualifications is included in Section 1.4.2.

SCE&G has contracted with the Westinghouse Electric Corporation (Westinghouse) for the design and manufacture of the complete Nuclear Steam Supply System. In addition, Westinghouse supplies competent technical consultation in areas such as initial fuel loading, testing, and initial startup. Westinghouse is involved in the training of SCE&G personnel. A brief summary of Westinghouse technical qualifications is included in Section 1.4.3.

Dames and Moore, Inc. (D&M), is retained as environmental consultant and has assisted SCE&G in the preparation of the environmental portions of this report. In addition, D&M has assisted SCE&G in the preparation of the required Environmental Report-Operating License Stage. A brief summary of D&M technical qualifications is included in Section 1.4.4.

1.4.1 SOUTH CAROLINA ELECTRIC AND GAS COMPANY QUALIFICATIONS AND EXPERIENCE

SCE&G owns and operates an integrated electric generation, transmission, and distribution system which serves approximately 484,000 customers in a 15,000 square mile service area. This area, stretching from the central region to the coastal plains, includes Columbia, the state capital, and Charleston, South Carolina’s principal seaport. SCE&G’s transmission system is part of the interconnected grid extending over a large part of the central and eastern portion of the nation.

SCE&G, acting as its own general contractor, has constructed generating facilities which provide most of SCE&G’s present generating capacity. For the construction of the Virgil C. Summer Nuclear Station, SCE&G has engaged Daniel Construction Company (Daniel) of Greenville, S.C. to act as the general contractor. A summary of the qualifications of Daniel is given in Section 1.4.5.

SCE&G is a partner with Carolina Power and Light Company, Duke Power Company and Virginia Electric and Power Company in Carolinas Virginia Nuclear Power Associates, Inc. (CVNPA). CVNPA was formed in 1956 to build and operate a 17,000 kWe nuclear steam generating plant at Parr, S.C., for research, operating and engineering experience. This plant, the Carolinas Virginia Tube Reactor (CVTR), was constructed and operated under license granted by the Atomic Energy Commission. The CVTR was decommissioned in 1967 after completion of a successful operating and research program. SCE&G actively participated in the CVTR project in the planning, management, training, research, technical and operational levels.

SCE&G has maintained active participation in the fast breeder programs of both Westinghouse and Atomics International as well as participating in the Savannah River Nuclear Study Group which examined the feasibility of developing power from a production reactor.

Organization charts for SCE&G relevant to this project, are Figures 13.1-1 through 13.1-5. The engineering and construction of this project is the responsibility of the Vice-President, Group Executive, Engineering and Construction.

1.4.2 GILBERT ASSOCIATES, INC. QUALIFICATIONS AND EXPERIENCE

Gilbert Associates, Inc. (Gilbert) engineers and consultants, is the architect engineer for the Virgil C. Summer Nuclear Station. The company, with its main office in Reading, Pennsylvania, was originally known as W. S. Barstow and Company which was organized in 1906. The corporate name was changed to E. M. Gilbert Engineering Corporation in 1933. In 1942, the corporate structure was revised to provide for complete employee ownership and the name became Gilbert Associates, Inc. In 1973, Gilbert Associates acquired Commonwealth Associates, Inc., a large engineering and consulting firm of similar capabilities.

Throughout the past 70 years, Gilbert has progressively grown in size and in scope of activity. The collective experience and capabilities of the firm offer complete consulting and engineering services to both investor-owner utilities and general industry in such diverse fields as: nuclear and conventional power generation; transmission, substation, and distribution systems; economic engineering and management consulting service; steel making and processing; cement and minerals processing; chemical and general industrial facilities; water desalination plants; institutional and commercial installations; environmental and solid waste treatment; and water production projects. Projects undertaken have ranged from large electric power generating plants and production facilities to small industrial boiler plants and allied service facilities.

Since 1942, Gilbert has been responsible for the design of well over 100 thermal generating units, both fossil and nuclear power, representing approximately 30,000,000 kW of new generating capacity. The company's experience includes one of the first reheat units, one of the first once through boiler units and one of the first supercritical steam pressure units. Individual unit designs have ranged in ratings up to 1,200,000 kW and stations have varied complexity - nuclear, mine-mouth, closed cycle cooling tower, base-load, peaking and others. At present, Gilbert has over 17,000,000 kW of generation under design, of which 10,600,000 kW is nuclear.

Since 1950, Gilbert has played an active and important role in the development of nuclear energy for private utilities, industry and governmental agencies. Gilbert projects include complete programs of nuclear power development involving analysis of sites, complete evaluations of proposals, contract and fuel program assistance, preparation of license applications, containment vessel design concepts, complete plant design and procurement. More than a score of studies, cost estimates, evaluations, concept developments and preliminary plant designs have been prepared since 1953 for various utility customers and other clients such as the Nuclear Regulatory Commission (formerly the Atomic Energy Commission), Westinghouse Electric Corporation, Atomics International and the Air Force. Development of reactor containment design concepts for plants near large population centers was accomplished in 1963. The first complete station design was the Saxton Experimental Power Reactor (20 MWe) in 1959. Other completed designs include the 490 MWe Robert Emmett Ginna Station for the Rochester Gas and Electric Corporation; the 320 MWe Mihama plant for the Kansai Electric Power Company, Inc.; the 781 MWe Takahama Station for the Kansai Electric Power Company, Inc. as subcontractor to Westinghouse Electric International Company; and the 870 MWe Three Mile Island Nuclear Station, Unit 1 for the Metropolitan Edison Company. Stations for which Gilbert now has overall plant design and engineering responsibility are the 855 MWe Crystal River Plant, Unit 3, for Florida Power Corporation; the 2 unit, 1200 MWe each, Perry Nuclear Power Plant for the Cleveland Electric Illuminating Company; the 2 unit, 1122 MWe each, Ohi Station (ice condenser), as subcontractor to Westinghouse for the Kansai Electric Power Company, Inc.; the 564 MWe Ko-Ri Station, Unit 1, as subcontractor to Westinghouse for the Korea Electric Power Company; and the 615 MWe Nuclear Power Plant Krsko for Savske Elektrarne, Ljubljana, Slovenia; Elektroprivreda, Zagreb, Croatia, as subcontractor to Westinghouse.

The Gilbert organization includes nearly 3700 employees with a complete staff of engineers, draftsmen and many technical specialists. Included in the total staff are over 2200 engineers, technical specialists and draftsmen. This includes members of management, professional personnel and individuals in other specialized fields.

Responsibility for engineering and design of nuclear power plants is centered in the Utilities Division of the Company. Every nuclear project is assigned to a Project Manager, selected from a staff of engineers having an average of about 15 years with Gilbert. Through this divisional control, the production function of the project is carried through to completion.

The engineering disciplines, such as nuclear, mechanical, electrical, civil, structural, architectural, environmental engineering, etc., are grouped into departments and provide technical resources to the project. Each department is managed by a chief engineer who assumes technical and administrative responsibility for the personnel assigned to the various projects. Figures 1.4-1 and 1.4-2 depict the Gilbert organization.

Additional support necessary to the project is provided from other service departments, including drafting, estimating, specifications, legal, accounting, purchasing, expediting, etc.

1.4.3 WESTINGHOUSE ELECTRIC CORPORATION QUALIFICATIONS AND EXPERIENCE

Westinghouse Electric Corporation’s (Westinghouse) experience in nuclear plants for the electric utility industry is demonstrated by the Pressurized Water Reactor (PWR) plants that Westinghouse has designed, developed, and manufactured. Table 1.4-1 lists all Westinghouse PWR plants to date, including those plants currently under construction or on order.

Westinghouse has long held a position of leadership in the electrical manufacturing industry. Traditionally this leadership has been based on technological development of both standard and new products, reliability and product quality. Nowhere is this traditional leadership displayed more vividly than in nuclear power. Through early participation in basic research and basic engineering development, Westinghouse has established a broad technological foundation in nuclear power application. This has been followed by a continuing program of sound technological development which enables Westinghouse to offer to the electric utility industry a reliable and safe source of power.

The experience of Westinghouse in nuclear activity is evident in numerous nuclear power projects completed, soon to go into operation, or under construction. The following paragraphs describe Westinghouse designed PWR plants which are presently in operation.

1.4.3.1 Plants in Operation

Westinghouse PWR plants in operation are as follows:

1. 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 Nuclear Regulatory Commission (NRC) contract. Shippingport's PWR has produced power for Duquesne Light Company since December 1957.

2. 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.

3. 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 261 MWe.

4. 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 Societe 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.

5. 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 430 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 operation began in January 1968.

6. 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.

7. Jose Cabrera-Zorita

The Jose Cabrera-Zorita 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-Zorita is owned and operated by the Union Electrica, S.A., a Spanish utility.

8. 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 four years after award of the plant contract. Beznau No. 2 achieved criticality in October 1971 and began commercial operation in early 1972.

9. Robert Emmett Ginna

The Robert Emmett Ginna Plant, owned and operated by the 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.

10. 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 PWR's 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, 781 MWe unit. Initial criticality was achieved in March 1974. 02-01

11. H. B. Robinson No. 2

This plant is a three-loop, 700 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 September 1970 and first power to the system in October 1970.

12. 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.

13. Surry No. 1 and No. 2

The Surry Power Station, two three-loop 788 MWe units, is owned by the Virginia Electric and Power Company. (The James River Station is about 30 miles from Norfolk, Virginia). First criticality on Surry No. 1 was achieved July 1972. Commercial operation began in September 1972. Initial criticality on Surry No. 2 was achieved in March 1973.

14. Turkey Point No. 3 and No. 4

Florida Power and Light Company is the owner of a four-unit station on Biscayne Bay, Florida. Turkey Point Nos. 3 and 4 of the station are three-loop, 666 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.

15. Indian Point No. 2 and 3

Consolidated Edison Company of New York operates three nuclear units located in Buchanan, New York; two are (units 1 and 2) owned by the company and one (unit No. 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.

16. 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.

17. Zion No. 1 and No. 2

    Commonwealth Edison Company is the owner of these four-loop, 1050 MWe units 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.

18. 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.

19. 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.

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

    Indiana and Michigan Electric Company is the owner of these four-loop, 1060 MWe plants located in Bridgman, Michigan. These plants are the first to use the Westinghouse ice condenser containment design. Unit 1 initial criticality was achieved in January 1975, and Unit 2 initial criticality was achieved in 1978.

21. 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.

22. 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 April 1977.

23. Salem No. 1

Salem No. 1, 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 in late 1976.

1.4.3.2 Westinghouse Facilities

Westinghouse Electric Corporation was contracted to design, fabricate, and deliver the Nuclear Steam Supply System and Nuclear Fuel for the Virgil C. Summer Nuclear Station. Westinghouse provided technical assistance for the installation and startup of their supplied equipment.

1.4.4 DAMES AND MOORE, INC. QUALIFICATIONS AND EXPERIENCE

The partnership of Dames and Moore, Inc. (D&M) was founded in 1938 in Los Angeles, California. Since then the firm has grown to more than 1400 employees in 25 offices in the United States and 17 offices in foreign countries.

The professional staff of D&M has a diversified background in the fields of land use planning, socioeconomics, demography, meteorology, air and water quality, biology, ecology, geology, soil and rock mechanics and dynamics, foundation engineering, geophysics, seismology, hydrology, marine engineering and systems management.

D&M has served more than 8500 clients in over 100 countries. The firm has performed more than 30,000 investigations of various types. D&M has been involved in site and environmental studies for close to half of the nuclear power plants under construction or planned in the United States. Within the Southeast, these projects include:

Arkansas Power & Light Company	Arkansas Nuclear One
Alabama Power Company	Joseph M. Farley
Carolina Power & Light Company	H. B. Robinson
Carolina Power & Light Company	Shearon Harris
Duke Power Company	Oconee
Florida Power & Light Company	Turkey Point
Florida Power & Light Company	South Dade
Florida Power & Light Company	St. Lucie
Georgia Power Company	Edwin I. Hatch
Virginia Electric & Power Company	North Anna
Virginia Electric & Power Company	Surry

1.4.5 DANIEL CONSTRUCTION COMPANY QUALIFICATIONS AND EXPERIENCE

Daniel Construction Company (Daniel), a division of Daniel International Corporation, Greenville, S.C., has a wide variety of engineering and construction assignments. The parent company, Daniel International Corporation has in excess of $10,000,000 worth of projects currently in engineering or construction in many parts of the world. A recent survey of the nation's 400 largest contractor rates Daniel 3rd in contract awards, and 32nd in design/construction awards. Daniel has acquired extensive construction and project management experience in major industrial complexes for the chemical, paper, rubber, textile, aluminum and power generation industries. These construction services involve the ability to meet precise tolerances and specifications on erection, fabrication and equipment installation and have required a thorough knowledge of heavy construction, mechanical, electrical and instrumentation techniques and methods. This experience and the developed capabilities are applicable to the construction of nuclear power facilities. 02-01

The Daniel quality assurance program for ASME nuclear code construction has been evaluated and accepted by an ASME survey team and an interim certificate of authorization issued.

The certificate of authorization to perform code construction ("N" Stamp) will be obtained following the successful completion of an ASME survey team field implementation and enforcement audit at the Virgil C. Summer Nuclear Station.

Daniel's experience, past and present, include construction of nuclear and fossil fueled power plants. First project of this nature was construction of the nuclear power CVTR at Parr, S.C. This facility operated several years as a prototype plant. Power plants completed or currently under construction are specified below:

1. Alabama Power Co.
   J. M. Farley Project Unit 1 & 2
   847 MW (each)

2. Carolina Power & Light Co.
   Shearon Harris Unit 1, 2, 3, & 4
   900 MW (each)
   Darlington Elec. Plant (Turbines)
   660 MW

3. Detroit Edison Company
   Fermi II Unit 2
   1150 MW

4. Duke Power Company
   Spencer Facility
   170 MW
   Cliffside Facility
   575 MW

5. Georgia Power Company
   Vogtle Plant (Turbines)
   300 MW
   McManus Plant (Turbines)
   360 MW
   Plant Wansley Unit 1 & 2
   880 MW (each)

6. Kansas City Power & Light Company
   Kansas Gas & Electric
   La Cygne Unit 2
   600 MW
   Wolf Creek Unit 1
   1150 MW

7. Kansas City Power & Light Company
   St. Joseph Light & Power
   Iatan Station
   630 MW

8. South Carolina Electric & Gas Company
   Arthur M. Williams (Turbines)
   60 MW
   Arthur M. Williams Unit 1
   600 MW
   Virgil C. Summer Unit 1
   918 MW
   Fairfield Pumped Storage (8 Units)
   480 MW

9. Union Electric Company
   Callaway Generating Plant Unit 1 & 2
   1150 (each)

10. Virginia Electric & Power Company
    Bath County Pumped Storage (6 units)
    2100 MW

In addition to constructing numerous gas turbine power stations, Daniel is currently building a nuclear fuel reprocessing plant for Allied Gulf Nuclear Services.

TABLE 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Shippingport	Duquesne Light Company; Research & Development Administration	Pennsylvania	1957	90	4
Yankee-Rowe	Yankee Atomic Electric Company	Massachusetts	1961	175	4
Trino Vercellese (Enrico Fermi)	Ente Nazionale per L'Energia Elettrica (ENEL)	Italy	1965	261	4
Chooz (Ardennes)	Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA)	France	1967	309	4
San Onofre No. 1	Southern California Edison Co.; San Diego Gas and Electric Co.	California	1968	430	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

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
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	700	3
Beznau No. 2	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1972	350	2
Point Beach No. 2	Wisconsin Electric Power Co.; Wisconsin Michigan Power Co.	Wisconsin	1973	497	2

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Surry No. 1	Virginia Electric and Power Co.	Virginia	1972	788	3
Turkey Point No. 3	Florida Power and Light Co.	Florida	1972	666	3
Indian Point No. 2	Consolidated Edison Company of New York, Inc.	New York	1973	873	4
Prairie Island No. 1	Northern States Power Company	Minnesota	1973	530	2
Turkey Point No. 4	Florida Power and Light Co.	Florida	1973	666	3
Surry No. 2	Virginia Electric and Power Co.	Virginia	1973	788	3
Zion No. 1	Commonwealth Edison Company	Illinois	1973	1050	4

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Kewaunee	Wisconsin Public Service Corp.; Wisconsin Power and Light Co.; Madison Gas and Electric Co.	Wisconsin	1974	541	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	Commomwealth Edison Company	Illinois	1974	1050	4
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	1060	4
Ringhals No. 2	Statens Vattenfallsverk (SSPB)	Sweden	1975	822	3

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Almaraz No. 1	Union Electrica, S. A.,; Compania Sevillana de Electricidad, S. A.; Hidroelectrica Espanola, S. A.	Spain	1981	902	3
Beaver Valley No. 1	Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Company	Pennsylvania	1976	852	3
Diablo Canyon No. 1	Pacific Gas and Electric Co.	California	1980	1084	4
Indian Point No. 3	Consolidated Edison Company of New York, Inc.	New York	1976	965	4
Lemoniz No. 1	Iberduero, S. A.	Spain	1982	902	3
Salem No. 1	Public Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic City Electric Co.; Delmarva Power and Light Co.	New Jersey	1977	1090	4

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Trojan	Portland General Electric Co.; Eugene Water and Electric Board; Pacific Power and Light Company	Oregon	1976	1130	4
Almaraz No. 2	Union Electrica, S. A.; Compania Sevillana de Electricidad, S. A.; Hidroelectrica Espanola, S. A.	Spain	1984	902	3
Asco No. 1	Fuerzas Electricas de Cataluna, S. A. (FESCA)	Spain	1982	902	3
Diablo Canyon No. 2	Pacific Gas and Electric Co.	California	1981	1106	4
Joseph M. Farley No. 1	Alabama Power Company	Alabama	1977	829	3
Ko-Ri No. 1	Korea Electric Power Co., Ltd.	Korea	1978	564	2
North Anna No. 1	Virginia Electric and Power Co.	Virginia	1978	898	3

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
North Anna No. 2	Virginia Electric and Power Co.	Virginia	1980	898	3
Ohi No. 1	The Kansai Electric Power Co., Inc.	Japan	1979	1122	4
Ohi No. 2	The Kansai Electric Power Co., Inc.	Japan	1979	1122	4
Ringhals No. 3	Statens Vattenfallsvert (SSPB)	Sweden	1981	900	3
Sequoyah No. 1	Tennessee Valley Authority	Tennessee	1980	1148	4
Angra dos Reis No. 1	Furnas-Centrais Electricas, S.A.	Brazil	1981	626	2
Asco No. 2	Fuerzas Electricas de Cataluna, S. A. (FESCA); Empresa Nacional Hidroelectrica del Ribagorzana, S. A. (ENHER); Fuerzas Hidroelectricas	Spain	1983	902	3

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Donald C. Cook No. 2	Indiana and Michigan Electric Company (AEP)	Michigan	1978	1060	4
Lemoniz No. 2	Iberduero, S. A.	Spain	1984	902	3
Sequoyah No. 2	Tennessee Valley Authority	Tennessee	1981	1148	4
Watts Bar No. 1	Tennessee Valley Authority	Tennessee	1981	1177	4
William B. McGuire No. 1	Duke Power Company	North Carolina	1981	1180	4
Joseph M. Farley No. 2	Alabama Power Company	Alabama	1980	829	3
Krsko	Savske Elektrarne, Ljubljana, Slovenia, Elektroprivreda Zagreb, Croatia	Yugoslavia	1981	615	2
Ringhals No. 4	Statens Vattenfallsvert (SSPB)	Sweden	1982	900	3

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Salem No. 2	Public Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic City Electric Co.; Delmarva Power and Light Co.	New Jersey	1980	1115	4
Virgil C. Summer	South Carolina Electric and Gas Company;	South Carolina	1981	900	3
Watts Bar No. 2	Tennessee Valley Authority	Tennessee	1982	1177	4
William B. McGuire No. 2	Duke Power Company	North Carolina	1982	1180	4
Comanche Peak No. 1	Texas Utilities Generating	Texas	1983	1150	4
Bryon No. 1	Commonwealth Edison Co.	Illinois	1984	1120	4
Seabrook No. 1	Public Service Company of New Hampshire; United Illuminating Company	New Hampshire	1984	1200	4

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
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	1983	1250	4
Beaver Valley No. 2	Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Co.; Cleveland Electric Illuminating Company; Toledo Edison Company	Pennsylvania	1986	852	3
Braidwood No. 1	Commonwealth Edison Company	Illinois	1985	1120	4
Callaway No. 1	SNUPPS - Union Electric Co.	Missouri	1983	1150	4
Catawba No. 1	Duke Power Company	South Carolina	1985	1153	4
Ko-Ri No. 2	Korea Electric Power Co., Ltd.	Korea	1983	605	2
Braidwood No. 2	Commonwealth Edison Company	Illinois	1986	1120	4

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Byron No. 2	Commonwealth Edison Company	Illinois	1985	1120	4
Catawba No. 2	Duke Power Company	South Carolina	1987	1153	4
Comanche Peak No. 2	Texas Utilities Generating Co.	Texas	1985	1150	4
Marble Hill No. 1	Public Service Company of Indiana, Inc.; Northern Indiana Public Service Company	Indiana	1984	1150	4
Millstone No. 3	Northeast Nuclear Energy Co.	Connecticut	1987	1156	4
Seabrook No. 2	Public Service Company of New Hampshire; United Illuminating Company	New Hampshire	1989	1200	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	1986	1250	4

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Taiwan Unit No. 5	Taiwan Power Company	Taiwan	1984	950	3
Wolf Creek Unit No. 1	SNUPPS - Kansas Gas and Electric Company; Kansas City Power and Light Company	Kansas	1983	1150	4
Alvin W. Vogtle No. 1	Georgia Power Company	Georgia	1986	1113	4
Taiwan Unit No. 6	Taiwan Power Company	Taiwan	1985	950	3
Alvin W. Vogtle No. 2	Georgia Power Company	Georgia	1991	1113	4
Marble Hill No. 2	Public Service Company of Indiana, Inc.; Northern Indiana Public Service Company	Indiana	1985	1150	4
Shearon Harris No. 1	Carolina Power and Light Co.	North Carolina	1987	900	3

TABLE 1.4-1 (Continued)

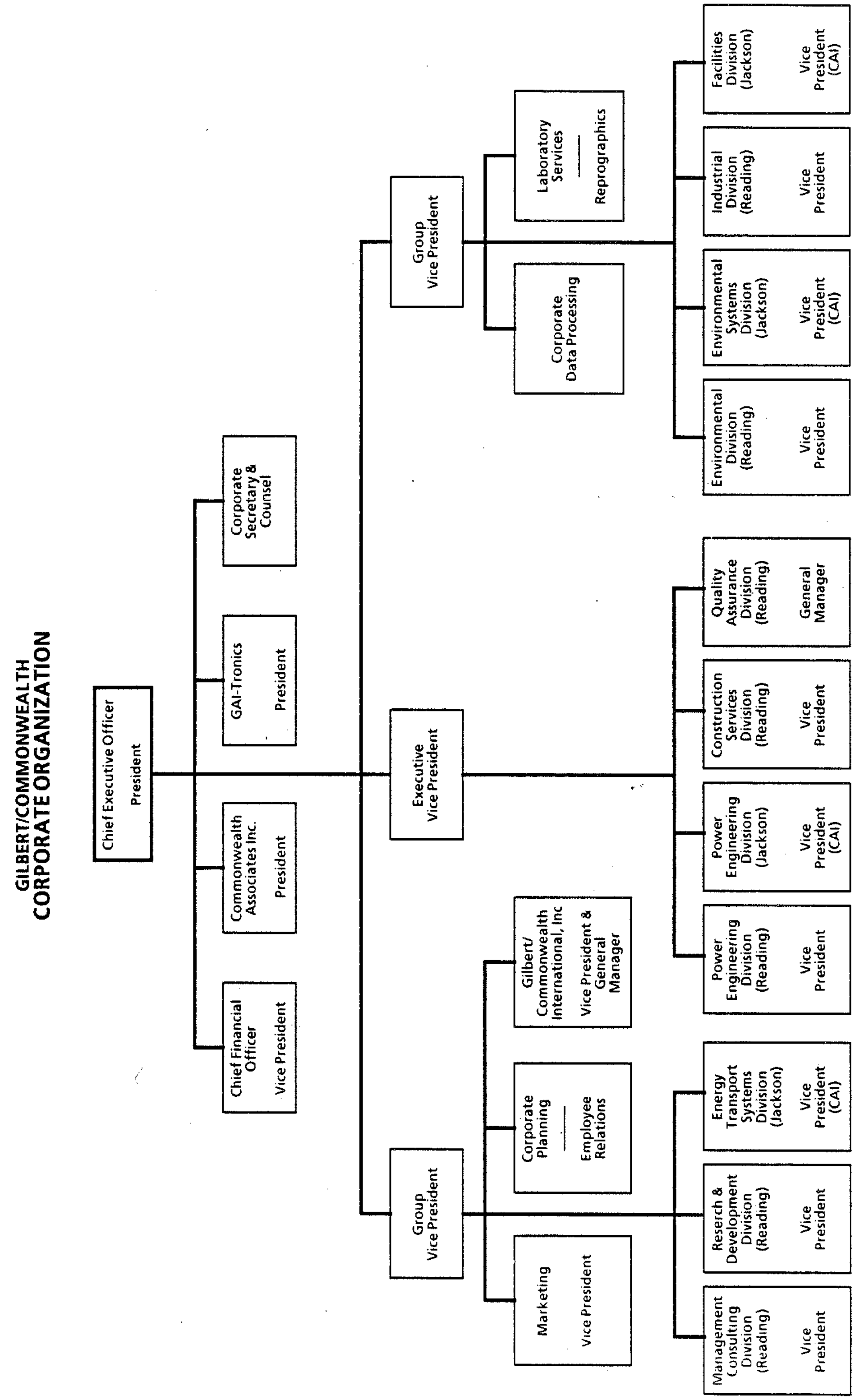
WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Shearon Harris No. 2	Carolina Power and Light Co.	North Carolina	1988	900	3
Atlantic No. (O.P.S.)	Public Service Electric and Gas Company; Atlantic City Electric Co.; Jersey Central Power and Light Company	New Jersey	1987	1150	4
Shearon Harris No. 4	Carolina Power and Light Co.	North Carolina	1988	900	3
Sundesert No. 2	San Diego Gas and Electric Co.	California	1988	950	3
Sayago No. 1	Iberduero, S. A.	Spain	1980's	1000	3
Unit No. 4	Iberduero, S. A.	Spain	1980's	1000	3
Shearon Harris No. 3	Carolina Power and Light Co.	North Carolina	1990	900	3

TABLE 1.4-1 (Continued)

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

Plant	Owner Utility	Location	Scheduled Commercial Operation	Mwe Net	Number of Loops
Unassigned No. 1 (O.P.S.)	Public Service Electric and Gas Company; Atlantic City Electric Company	New Jersey	1990	1150	4
Unassigned No. 2 (O.P.S.)	Public Service Electric and Gas Company; Atlantic City Electric Company	New Jersey	1992	1150	4

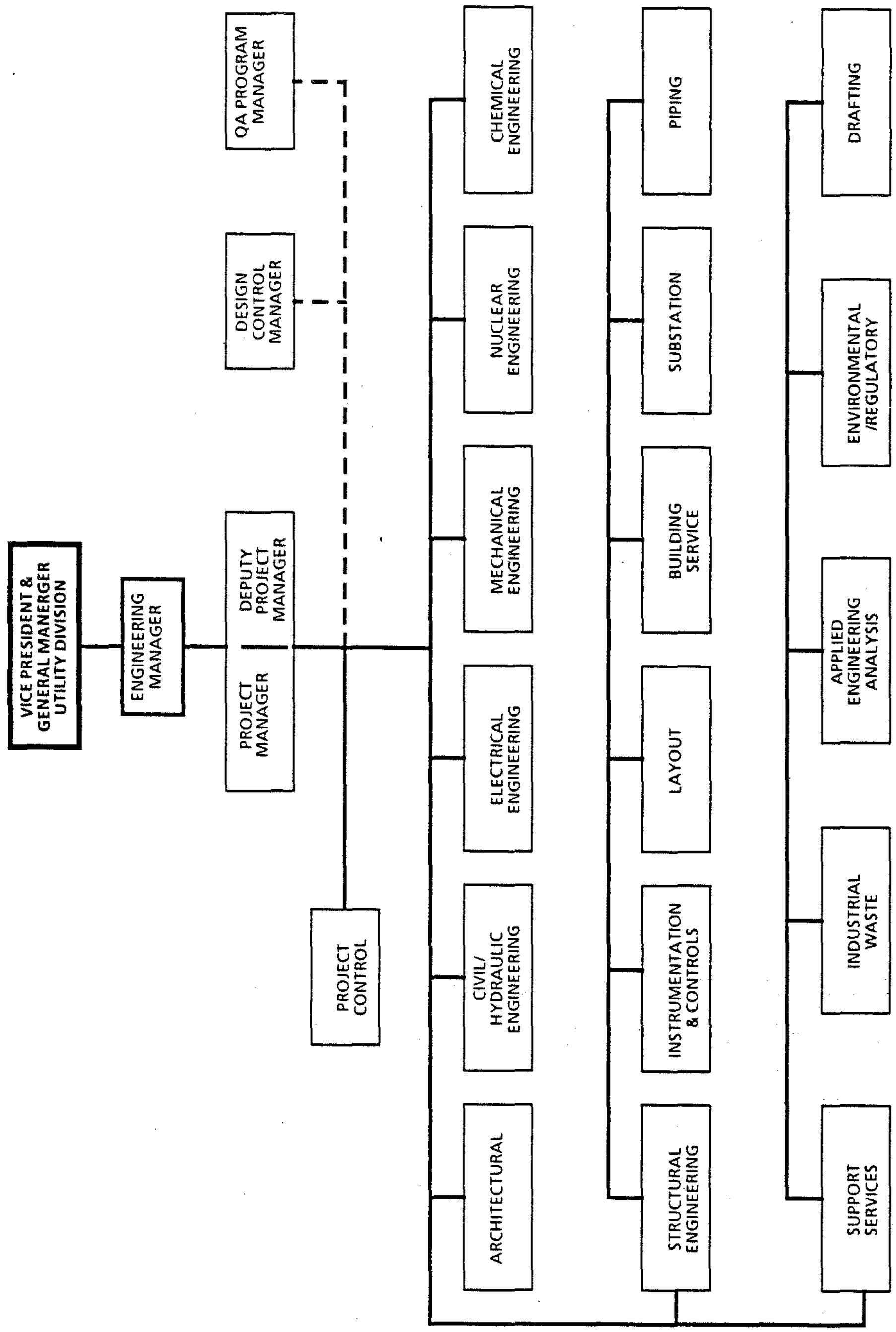


Amendment 0
August 1984

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

Organization Chart
Gilbert/Commonwealth Companies

Figure 1.4-1



Amendment 0
August 1984

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

Organization Chart
Gilbert Project Engineering

Figure 1.4-2

NOTE 1.5
Section 1.5 is being retained for historical purposes only. 99-01

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

Reference [1] presents descriptions of the safety-related research and development programs which are being carried out for, or by, or in conjunction with, Westinghouse Nuclear Energy Systems, and which are applicable to Westinghouse Pressurized Water Reactors.

For each program described in this section and still in progress, the safety-related program is first introduced, followed, where appropriate, by background information. There is, then, a description of the program which relates the program objectives to the problem and presents pertinent recent results. Finally, an alternate position may be given for programs (generally experimental rather than analytical) which have not yet reached a stage where it is reasonably certain that the results confirm the expectation. The alternate position is one that might be used if the results are unfavorable; it is not necessarily the only course that might be taken.

The term "research and development", as used in this section, is the same as that used by the Commission in Section 50.2 of its regulations, that is:

(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 technical information generated by these research and development programs will be used either to demonstrate the safety of the design and more sharply define margins of conservatism, or will lead to design improvements.

Progress in these development programs will be reported on a timely basis. New safety-related research and development programs, which include existing programs which become safety-related, will also be described.

Included in the overall research and development effort are the programs below which are applicable to the 17 x 17 fuel assembly. The test programs were completed during 1975 in order to support the initial loading of the first 17 x 17 fuel in late 1975.

1.5.1 VERIFICATION TESTS (17 x 17)

The design of the reactor uses a 17 x 17 square array of fuel rods and thimbles in a fuel assembly and is conceptually similar to, but geometrically different from, the 15 x 15 array used in previous designs. The 17 x 17 design is considered to be a relatively small extrapolation of the 15 x 15 design. Comprehensive testing has been performed to verify that the extrapolation is sufficiently conservative.

Westinghouse maintains that no plant need be designated a prototype and instrumented to verify the 17 x 17 fuel design. The change in flow induced vibration response of the internals from a 15 x 15 to a 17 x 17 fuel design will be minimal for the following reasons:

1. The only structural changes in the internals, other than the fuel assemblies, resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and control rod drive line.

2. The guide tube is rigidly attached at the upper core support plate only. The upper core plate serves only to align the guide tubes. Because of this type of support arrangement, the guide tube has a minimal contribution to the vibrational response of the core barrel and other internal components.

3. The effective flow area of the 17 x 17 guide tube is essentially the same as that of the 15 x 15 guide tube and therefore, there are no significant differences in the flow distribution in the upper plenum.

4. The fuel assembly lateral spring rate was found by applying loads at various grid locations with both nozzles restrained. This test assures that the effects of the fuel on the vibrational response of the reactor internals will remain essentially unchanged between the 15 x 15 and the 17 x 17 fuel assembly design because of the small variance of mass and spring rate between the tow designs. The preoperational hot functional flow testing presented in Chapter 14 is considered the most conservative test condition since higher flowrates exist.

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[2].
	Lateral Impact	Additional grid shares impact load [2].
Prototype Assembly Test	Pressure Drop	The margin between 7 grid design ΔP and D-Loop results[3] is adequate to cover the additional ΔP resulting from the additional grid (< 5% increase in ΔP).
	Lift Force	The margin between 7 grid design lift force and D-Loop results[3] 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)	DNB Correlation	Addition of a grid increases mixing which increases DNB margin. The effect of additional grid testing is presented in Reference [4] .
Incore Flow Mixing	Thermal Diffusion Coefficient (TDC)	TDC increases as grid spacing decreases[5].

The above tabulation shows that:

1. Additional design changes are not required (e.g., number of new fuel assembly holddown springs) due to the addition of a grid, and

2. Seven grid test information can be used to assess the adequacy of the eight grid design.

Additional testing to specifically investigate the eight grid assembly is not required.

1.5.1.1 Rod Cluster Control Spider Tests

1.5.1.1.1 Test Purpose and Parameters

The 17 x 17 rod cluster control spider 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 the 15 x 15 design).

The rod cluster control spider tests verified the structural adequacy of the design. The spider vane to hub joint was tested for structural adequacy by:

1. Vertical static load test to failure, and

2. Vertical fatigue test to approximately 3.0×10^6 steps.

The static load test was performed by applying tensile and compressive loads to the spider. The load was applied parallel to the spider hub and reacted between the spider hub and fingers. The spider fingers shared the load equally. The number of cycles for the fatigue test was determined from the expected number of steps a control rod drive mechanism would experience during 20 years in a load follow reactor (1.5×10^6 steps). The test met the recommended cyclic test requirements of the ASME Code, Section III, Appendix II, paragraph 1520.

The spring pack within the spider hub was tested to determine the spring load deflection characteristic as a function of the loading cycles seen by the spring. The test was terminated after 1000 cycles compared to a 400 cycle (rod drop) design value. The test loads were equal to or greater than that predicted to result in spring material yielding. These loads were in excess of the design values. The test acceptance criterion was for the spring to retain adequate preload after the repeated cycling.

1.5.1.1.2 Facility

The 17 x 17 spider tests were performed at the Westinghouse Engineering Mechanics Laboratory (see Section 1.5.3.2.15).

1.5.1.1.3 Status

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 2.8×10^6 steps without failure. The spider loading was 110 percent of the design value for 1.8×10^6 cycles and 220 percent of the design loading for 1×10^6 cycles. Design load is 3600 pounds compression and 1800 pounds tension. The spring test resulted in negligible preload loss.

1.5.1.2 Grid Tests

1.5.1.2.1 Test Purpose and Parameters

The 17 x 17 grid is conceptually similar to, 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.

Load deflection tests have been made on the grid spring and dimple. Grid spring radial (normal) stiffness and the grid dimple radial and tangential stiffness were obtained. This information was used to verify that the fuel rod clad wear evaluation has been based on conservative values of these parameters. The fuel rod wear evaluation is conservative as shown by the flow test results presented in Reference [3].

The grid buckling strength has been determined from tests using both static and dynamic loads. The loads were applied uniformly to the face of the outside strap, transmitted directly through the grid and reacted at the grid face opposite the input. A description of the grid impact test along with a description of the analytical use of the test parameters is given in Reference [2]. These tests are used to verify that grid buckling during a postulated seismic occurrence does not interfere with control rod insertion.

1.5.1.2.2 Facility

The grid tests were conducted at the Westinghouse Engineering Mechanics Laboratory (see Section 1.5.3.2.15).

1.5.1.2.3 Status

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 [2].

1.5.1.3 Fuel Assembly Structural Tests

1.5.1.3.1 Test Purpose and Parameters

The 17 x 17 fuel assembly tests were performed to determine mechanical strength and properties. The fuel assembly parameters obtained were as follows:

1. Lateral and axial stiffness.

2. Impact and internal structural damping coefficients.

3. Vibrational characteristics.

4. 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 loss of coolant (blowdown) accident analysis. The remaining tests are primarily to demonstrate that the assembly has sufficient mechanical strength to preclude damage during shipment, normal handling, and normal operation.

The fuel assembly is subjected to both lateral and axial loads to obtain the respective static axial and lateral stiffnesses. The information obtained from these tests is used to establish parameters primarily for accident analysis since these conditions appear limiting. The axially applied loads, which were well in excess of shipment, normal handling and normal operational design loads, did not result in any fuel assembly permanent deformation or damage.

Static loads were incrementally applied to the fuel assembly to determine its lateral stiffness. The tests were accomplished with both nozzles fixed in place and forces applied to various grids.

The fuel assembly was dynamically tested in a vertical position using core pins to simulate reactor support conditions. An electrodynamic shaker was attached to the center (4th) grid to provide excitation. The fuel assembly mode shapes and corresponding natural frequencies were obtained from displacement transducers. A comparison of analytical and experimental results is given in Reference [2]. Experimental vibrational studies of individual fuel rods were also performed. The rods were tested under simulated fuel assembly support conditions and as assembled in a prototype fuel assembly. The information obtained from these tests included the fundamental frequencies and mode shapes. A general test description and a summary of the results is presented in Reference [3].

The fuel assembly axial stiffness was found by incrementally increasing the static load (compressive) and then incrementally decreasing the static load.

Lateral impact tests were performed by displacing the center of the assembly with the nozzles fixed in place. The assembly was released and allowed to impact on lateral restraints at each of the five center grid locations.

The axial impact response and damping were found by dropping the fuel assembly from various heights. The axial impact test was performed with the fuel assembly in the upright position.

The relevant parameters measured during the lateral and axial impact tests are as follows:

1. Impact duration versus impact load.

2. Impact force versus drop height or initial displacement.

3. Impact damping or restitution as a function of impact force.

A general description of the test procedure, including a description of use of the parameters as related to accident analysis is presented in Reference [2].

There is a general axial test buckling criterion which does not allow local buckling of components which could preclude control rod insertion during an accident. The fuel assembly overall buckling and component local buckling is checked during the axial static and dynamic tests. The lateral displacement associated with the fuel assembly overall (beam type) buckling is limited by the reactor internals clearances and therefore buckling does not reduce the fuel assembly ultimate strength. Local component buckling was not experienced during either the static or dynamic tests for loads well in excess of the design values. The general acceptance criteria was not violated.

1.5.1.3.2 Facility

These tests were conducted at the Westinghouse Engineering Mechanics Laboratory (see Section 1.5.3.2.15).

1.5.1.3.3 Status

The fuel assembly structural tests have been completed. The fuel assembly structural test results are factored into the seismic and blowdown analyses [2].

1.5.1.4 Guide Tube Tests

A new guide tube was designed to accommodate the 24 rodlet pattern adopted for 17 x 17 cores. This guide tube is sufficiently strong to provide increased margins of safety over present guide tubes. The main features of the new design are full length enclosures and cylindrical upper enclosures. The 17 x 17 (24 rodlet) pattern reduced the central area available for driveline passage significantly, thus necessitating a generally tighter design of the rod guidance elements.

The following guide tube tests are considered as engineering tests.

These tests are used as design tools and are not specifically required for demonstration of plant safety.

1. Engineering Prototype Assembly Tests.

2. Guide Tube Drop and Deflection Test.

1.5.1.5 Engineering Prototype Assembly Tests

1.5.1.5.1 Test Purpose and Parameters

The purpose of these tests was to demonstrate that the 17 x 17 fuel assembly and driveline hardware designs perform as predicted. These tests were run prior to the required plant functional tests and are used as an engineering information test to obtain experimental data. A single set of driveline hardware, including control rods, was used in the tests. The fuel assemblies and driveline 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.

These tests are used to verify, from an engineering confidence standpoint, the integrated fuel assembly and rod cluster control performance in several areas. Data obtained included pressures and pressure drops throughout the system, hydraulic loadings on the fuel assembly and driveline, 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. None of this information is considered to be safety-related.

Specifically, two full size 17 x 17 fuel assemblies (one for Phase I and one for Phase II and III testing), one control rod, drive shaft and control rod drive mechanism were installed and tested in the 24 inch inside diameter x 40 foot high D-Loop at the Westinghouse Test Engineering Laboratory Facility.

1.5.1.5.1.1 Fuel Assembly Life Test (Phase I)

The first fuel assembly was subjected to the maximum expected control rod travel during one fuel assembly lifetime. The nominal test conditions were a flow velocity based on the design flowrate, a temperature of 585°F and a pressure of 2000 psig. These conditions represent an extreme set of conditions.

Using an instrumented 17 x 17 prototype fuel assembly, guide tube and rod cluster control drive assembly, the tests conducted in the D-Loop obtained information on the following:

1. Mechanical integrity and performance

2. Drop time

3. Fuel rod vibration

4. Control rod velocity

5. Hydraulic lift force

6. Guide thimble dashpot pressure

Following this, the prototype fuel assembly underwent a complete post test evaluation and the guide tubes and driveline were inspected for any abnormal wear conditions. The purpose of this test was basically to determine the effect of the 17 x 17 fuel assembly and control rod configuration of Items 1 through 6 in Phase I and Items 1 and 2 in Phases II and III. The effect on control rod drop due to a seismic disturbance is evaluated analytically.

1.5.1.5.1.2 Guide Tube and Rod Cluster Control Life Test (Phase II and Phase III)

The second fuel assembly was then installed to continue the test at the same flow and temperature until 3,000,000 total steps of the driveline were accumulated. For Phases II and III, testing was run at temperatures between 100°F and 585°F and at flowrates from 50 percent to 140 percent of the design flowrate.

The test included a program of control rod drops and mechanism stepping that approximates the driveline duty for the design lifetime of an operating plant (Phase II). Specifically, approximately 1,275,000 mechanism steps and approximately 270 control rod drops were accumulated. The components were then inspected. Following inspections, testing was continued until a total of 3,000,000 mechanism steps and approximately 500 control rod drops were accumulated (Phase III testing).

These tests were directed toward:

1. Life wear evaluation

2. Drop time

3. Rod stall characteristics

These tests are not safety-related tests; they are used as engineering tools. Final verification is demonstrated during the precritical rod drop tests.

At the completion of Phase II tests, the test assembly was inspected to determine guide tube and driveline wear characteristics. This inspection was repeated at the end of the test (Phase III).

1.5.1.5.2 Facility

The above testing is conducted in the Westinghouse Test Engineering Laboratory Facility (see Section 1.5.3).

1.5.1.5.3 Status

The D-Loop testing has been completed. The results of the testing are given in References [3] and [6].

1.5.2 LOCA HEAT TRANSFER TESTS (17 x 17)

1.5.2.1 17 x 17 Reflood Heat Transfer Tests

Extensive experimental programs have recently been performed with a simulated 17 x 17 assembly to determine its behavior under Loss of Coolant Accident (LOCA) conditions. The 17 x 17 tests were conducted in the G-Loop facility at the Westinghouse Forest Hills Laboratory.

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 (see Reference [7]).

1.5.2.2 Facility Description

The 17 x 17 test facility provides experimental measurements on the reflooding behavior of a 17 x 17 rod array following a LOCA. The test assembly consists of an array of 336 electrically heated rods and 25 guide tube thimbles arranged in a 17 x 17 array. The heater rod diameter, the active heated length, and pitch spacing is identical to that used in the 17 x 17 fuel. There were eight Westinghouse production mixing vane grids in the bundle.

1.5.2.3 Delayed Departure From Nucleate Boiling Testing

1.5.2.3.1 Introduction

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 the ECCS. 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 specialist’s 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 has initiated a program to experimentally simulate the blowdown phase of a LOCA. This testing is scheduled for the middle of 1976 as part of an Electric Power

Research Institute (EPRI) sponsored Blowdown Heat Transfer Program. This program is scheduled for completion at the end of 1976. Development of a transient DNB correlation for use in the Westinghouse ECCS analysis is planned for completion in 1977.

1.5.2.3.2 Objective

The objective of the delayed departure from nucleate boiling (DDNB) test is to determine the time that DNB occurs under LOCA conditions. This information will be used to confirm the existing, or develop a new, Westinghouse transient DNB correlation. The steady-state DNB data obtained from 15 x 15 and 17 x 17 test programs can be used to assure that the minimal geometric differences between the two fuel arrays can be correctly treated in the transient correlations.

1.5.2.3.3 Program

The program is divided into two phases. Phase I simulates PWR behavior during a LOCA to permit definition of the time delay associated with onset of DNB. Tests in this phase will cover the large double ended guillotine cold leg break. All tests in Phase I are started after establishment of typical steady-state operating conditions. The fluid transient is then initiated, and the rod power decay is programmed in such a manner as to simulate the actual heat input of fuel rods. The test is terminated when the heater rod temperatures reach a predetermined limit (dependent on power level).

Typical parameters which can be studied under Phase I testing are shown in Table 1.5-1.

Phase II will provide separate effects data to permit heat transfer correlation development.

The Phase II tests will also start from steady-state conditions, with sufficient power to maintain nucleate boiling throughout the bundle. Controlled ramps of decreasing test section pressure or flow will initiate DNB. By applying a series of controlled conditions, investigation of the DNB will be studied over a range of qualities and flows and at pressures relevant to a PWR blowdown.

Typical parameters which can be studied under Phase II testing are shown in Table 1.5-2.

1.5.2.3.4 Test Description

The experimental program is being 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 to be used in this program will be assembled using internally heated rods and Westinghouse grids. The heater rods are designed for high reliability, long life, and high power density. The maximum power is 18.8 kW/ft, and the total power is 135 kW for extended periods over the 12 foot heated length of the rod. Heat is generated internally by means of a varying cross-section, rugged, tubular resistor which approximates a U^2 cos U power distribution, skewed to the bottom. Each rod is adequately instrumented with a total of 20 thermocouples (8 resistors and 12 sheath thermocouples).

1.5.2.3.5 Results

The experiments in the DDNB facility will result in cladding temperature and fluid properties measured as a function of time throughout the blowdown range from 0 to 20 seconds.

Facility modifications and installation of the initial test bundle have been completed. A series of shakedown tests in the J-Loop have been performed. These tests provided data for instrumentation calibration and check-out, and provided information regarding facility control and performance. Initial program tests were performed during the first half of 1975.

1.5.2.4 Single Rod Burst Test

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

Previously, single rod and multi-rod burst tests have been completed on the 15 x 15 fuel assembly rods under conditions which exist during the LOCA. The conclusion of these tests was that fuel rods burst in a staggered manner so that maximum average flow area blockage for the assembly is 55 percent during blowdown and 65 percent during reflood, based on the characteristics of the 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 bursting specimens at the various internal pressures and heating rates in a steam atmosphere.

In addition, tests were run on 15 x 15 fuel assembly rods to assure reproducibility of the 1972 single rod burst test results. Results from the program are documented in References [8] and [9].

The single rod burst tests and evaluation have been completed and are reported in References [8] and [9]. Results of the tests showed that the LOCA behavior of the 17 x 17 clad in comparison to that of the 15 x 15 clad exhibited no significant differences in failure ductility. Because of the results and the geometric scaling, the flow blockage (%) as determined by 15 x 15 multi-rod burst test simulation can be used for the 17 x 17 geometry.

1.5.2.5 Power-Flow Mismatch

Design emphasis has been placed on reliable and effective control and protection systems for the reactor core and engineered safety features to ensure adequate margins within incidents can be terminated before the onset of fuel failure. For this reason, investigations into the mechanisms of fuel failure and its propagation, and phenomena of molten fuel-coolant interaction have been limited.

Obtaining complete answers to the question related to fuel rod failure will require extensive testing because of the multiplicity of parameters involved. Such an extensive inpile test program has been proposed by the Aerojet Nuclear Company under NRC (AEC) sponsorship for the Power Burst Facility (PBF).

The proposed PBF program is expected to simulate conditions appropriate to major accidents postulated for reactor systems, i.e., loss of flow, loss of coolant, and reactivity initiated accidents. Phase I of the proposed PBF program is expected to be completed approximately two years after facility checkout or by mid 1977. Phase I is expected to concentrate upon establishing the thresholds for and the consequences of fuel failure and utilizes fuel rod clusters of various sizes to determine the extent of failure propagation. This program would broaden the experimental basis for evaluating reactor safety, but should not be considered essential for the design and safe operation.

Westinghouse will closely follow any such experimental program or analytical studies that may become available. Until such information is available, clearly demonstrating that local fuel melting is an acceptable condition, the emphasis in design will continue to be placed on providing adequate margins to minimize the probability of fuel melting. The margins incorporated in the design within which incidents can be terminated, before the onset of fuel failure, provide a sound basis for safe operation.

1.5.3 WESTINGHOUSE TEST ENGINEERING LABORATORY FACILITY

1.5.3.1 Introduction

The Test Engineering Laboratory at Forest Hills, Pennsylvania, has long been the major Westinghouse center for nuclear research and development. The Test Engineering Laboratory is totally involved with the design and implementation of facilities and programs to prove the reliability of Westinghouse PWR concepts and components.

The Test Engineering Laboratory has full in-house capabilities to design and construct PWR loops for both hydraulic and heat transfer testing programs.

Historically the Test Engineering Laboratory has been in a state of transition, depending upon the current need for its services. Today's great need is for ECCS data and the verification of many new PWR system components. Past needs and accomplishments have included the development of supercritical heat transfer once through loops; rod cluster control drive mechanisms; fuel assemblies; underwater handling tools; and fuel assembly grid design, among many other earlier projects. Testing has included air filter tests; water chemistry tests; inpile testing for fuel rods; single fuel rod burst tests; hydraulic studies on fuel assemblies; and corrosion testing of Zircaloy and other PWR components and materials, with and without heat transfer.

The Test Engineering Laboratory is a very flexible installation, one which will continue to expand and develop as future needs for its services arise. Its staff, too, varies according to requirements.

There are currently more than 100 persons involved in laboratory projects, including 12 electrical and mechanical engineers, more than 75 highly skilled technicians, and some 30 specialists from other divisions of Westinghouse. The Test Engineering Laboratory has the option of obtaining personnel from the entire Corporation, depending upon the need for specific skills, knowledge and experience.

Ongoing research performed at the Test Engineering Laboratory continues to demonstrate the reliability of Westinghouse PWR plant components and greatly facilitates the development of improved reactor system components. As the test center for Westinghouse Nuclear Energy Systems, the Test Engineering Laboratory is totally committed to the advancement of the nuclear energy industry.

1.5.3.2 Tests, Test Loops and Equipment

This section contains a brief description of the major tests, test loops and test equipment at the Westinghouse Test Engineering Laboratory Facility.

1.5.3.2.1 A and B-Loops, Low Flow/High Pressure Hydraulic Facilities

These loops are small, high pressure, stainless steel facilities, used for testing small components and individual parts of larger components under normal working conditions. A canned motor pump circulates water in both the A-Loop and the B-Loop at 150 gpm. Operating temperatures are obtained from the conversion of the pumping power into heat, as well as from external heaters. Typical tests run in these loops are:

1. Full scale gate and check valves;
2. Material corrosion-erosion, with variable water chemistry; and
3. Corrosion product release and transport properties of crud.

Characteristics of A and B Loops

Maximum Flowrate	150 gpm at 300 feet
Maximum Pump Head	335 feet at 60 gpm
Maximum Allowable Temperature	650°F
Normal Working Pressure	2000 psi
Normal Working Temperature	600°F

1.5.3.2.2 D-Loop, Medium Flow/High Pressure Hydraulic Facility

The D-Loop is a flexible test facility used for demonstrating the interplay of reactor subsystems and evaluating component design concepts. It contains a canned motor pump, which produces a 290 foot head at 3000 gpm. All piping (10 inch Schedule 160) in contact with the primary water is stainless steel. Loop pressure is established and maintained by an air driven charging pump operating in conjunction with a gas loaded back pressure valve. Most of the power required to establish and maintain loop temperature is derived from the circulating pump operation, and 75 kW of heat is available from electric strip heaters.

The D-Loop services a 24 inch inside diameter x 40 foot long test vessel, which accommodates full scale models of large PWR core components for operational studies.

Characteristics of D Loop

Maximum Flowrate	4400 gpm
Maximum Allowable Pressure	2400 psi
Maximum Allowable Temperature	650°F
Normal Working Pressure	2000 psi
Normal Working Temperature	600°F
Pump Head at 3000 gpm	290 feet
Maximum Pump Head	340 feet (at 1500 gpm)
Main Loop Flow Measurement	10 inch venturi
Auxiliary Flow Measurement	6 inch venturis (2 inch branch lines)

1.5.3.2.3 E-Loop, Low Flow/Low Pressure Hydraulic Facility

The E-Loop is a low pressure, six inch, stainless steel loop, with 2 circulating pumps. These pumps may be connected in parallel, giving 2000 gpm at a 130 foot head, or in series, giving 1000 gpm at a 260 foot head. Flow and vibration studies are conducted with this loop, and, because of its low pressure, plastic models for visual observation or photography may be used. In addition, a four inch Rockwell water meter in a branch line permits the calibration of flow meters up to 800 gpm.

Characteristics of E Loop

Maximum Flowrate	2000 gpm at 130 feet 1000 gpm at 260 feet
Maximum Working Pressure	Pump Head

1.5.3.2.4 G-Loop, Emergency Core Cooling System Facility

The G-Loop is a high pressure, ECCS test facility designed and fabricated to ASME Section I for 2000 psi and 650°F. It consists of a main test section and vessel, exhaust system, piping, separators and muffler, flash chamber steam supply system, and high pressure/low pressure cooling systems.

This loop is basically designed to obtain test data for analysis of LOCAs, for breaks up to and including double ended pipe breaks for PWRs. Tests are initiated at simulated conditions existing 8 seconds after the start of a LOCA. A typical run consists of constant power and pressure, followed by pressure blowdown, power decay and

emergency core cooling. The G-Loop is capable of performing the following methods of emergency core cooling: Current, Upper Head Injection (UHI), UHI w/Current and other Core Spray Systems. It may also be used for constant temperature/pressure small leg break tests (core uncovering tests). These consist of boiling off water at a constant bundle power input until the rods can no longer be cooled.

The G-Loop test bundle consists of 480 electrically heated rods, 16 grid support thimbles, and 33 spray thimbles bounded by an octagonal stainless steel baffle and arranged as per a four-loop 15 x 15 rod bundle configuration. The loop is controlled (fully automated during transients) through a PDP-II-DEC-16K computer with a 600 point Computer Products A-D Converter operating at a sweep rate of 40,000 points per second for data acquisition. Figure 1.5-1 is a schematic of the G-Loop test facility.

G-Loop System Components and Characteristics

		Rated		Typical Operating		
Component	Material	Pressure (psi)	Temperature (°F)	Pressure (psi)	Temperature (°F)	02-01
Test Vessel	Carbon Steel	2000	650	1000	545	
Downcomer Side Tank	Carbon Steel	2000	650	1000	545	
In-Line Mixer	Carbon Steel	2000	650	1000	545	
Mixer Accumulator	Stainless Steel	2500	650	1800	100	
Flash Chamber	Carbon Steel	3000	700	2800	660	
Separators Nos. 1&2	Carbon Steel	2000	650	1000	545	
Spray Accumulators Nos. 1&2	Carbon Steel	2000	650	1800	150	
Spray Accumulator No. 3	Stainless Steel	2500	650	1800	150	
Reflood Tank	Stainless Steel	Atmosphere	212	Atmosphere	150	
Primary Piping	Carbon Steel	2000	650	1000	545	

1.5.3.2.5 H-Loop, High Flow Hydraulic Facility

The H-Loop constitutes a versatile hydraulic facility, capable of supplying 14,000 gpm of water at a developed head of 600 feet and at temperatures as high as 200°F. This four-loop system can simultaneously handle either full scale prototype test assemblies, or one large scale reactor model. The major purpose of the H-Loop is to permit the use of 1/7 scale reactor models and full scale fuel assemblies for conducting mixing studies, flow distribution studies, and similar low temperature/low pressure hydraulic tests.

Characteristics of H Loop

Maximum Flowrate	14,000 gpm
Pressure Drop Across Vessel Model	120 psi
Minimum Vessel Outlet Pressure	10 psig
Flow Accuracy	1/2%
Water Temperature Range	7-200°F
Maximum Loop-to-Loop Temperature Variation	2°F
Maximum Loop-to-Loop Flowrate Variation	3%

02-01

1.5.3.2.6 J-Loop, Delayed Departure from Nucleate Boiling Heat Transfer Facility

The J-Loop is a completely instrumented pressurized water test facility for verifying DDNB phenomena during a LOCA, and for conducting steady-state heat transfer studies. This test loop is a full size, single-loop simulation of a typical four-loop reactor system; it will accept a full length 5 x 5 bundle of internally heated fuel rods. The J-Loop is designed to operate at 2500 psia at 650°F, and at variable flowrates of up to 450 gpm. During LOCA tests, fluid input to the reactor vessel is closely controlled by 2 servo-controlled mixers, which inject a two-phase water/steam mixture into the test vessel, to simulate flow from the unbroken loops. Figure 1.5-2 is a schematic of the J-Loop test facility.

Characteristics of J-Loop

Test Fluid	Demineralized Water
Design Pressure	2500 psia
Design Temperature	650°F
Maximum Flowrate (hot)	450 gpm
Power Input to Test Vessel	3,500,000 watts (maximum)
Primary Test Heat Exchanger Rating	11,400,000 BTU/hour

1.5.3.2.7 K-Loop, Boron Thermal Regeneration Test

The K-Loop, Boron Thermal Regeneration System (BTRS) test facility is used to study the performance and to verify the component sizing of both the currently available THERM I and the improved THERM II BTRS. The function of this system is to process boron-containing effluents from the Reactor Coolant System (RCS) to yield a high boron concentration fraction, which can be used to borate the RCS. A relatively boron free fraction is also processed, which can be used to dilute the RCS, such as that required in load follow operations.

Characteristics of K-Loop

Total Tank Capacity	30,000 gallons	02-01
Chiller Capacity	48 ice-tons	
Maximum Ion Exchange Resin Test Volume	75 ft^3	
Maximum Test Process Rate Capability	10 gpm/ft^2 bed area	
Maximum Flow Test Capability	200 gpm	
Minimum Boron Storage Mode Fluid Temperature	50°F	
Maximum Boron Release Mode Fluid Temperature	160°F	

1.5.3.2.8 FLECHT-SET, Emergency Core Cooling System Facility

The FLECHT-SET is a low pressure facility, designed to provide experimental data on the influence of system effects on ECCS during the reflood phase of a LOCA.

The facility consists of a once through system, including an electrically heated test section (fuel rods and housing), accumulator, steam generator simulators, pressurizer, catch vessels, instrumentation, and the necessary piping to simulate the reactor primary coolant loop. Data acquisition is accomplished through a PDP-II-DEC-16K Computer with a 256 point Computer Products A-D Converter, operating at a sweep rate of 1200 points per second.

Characteristics of FLECHT-SET

100 Rod Bundle Maximum Power	100 kW
Maximum Bundle Flooding Rate	86 gpm
Water Temperature Range	100-200°F
System Pressure	0-60 psia

1.5.3.2.9 Single Rod Loop, Heater Rod Development Facility

The Single Rod Loop is used to evaluate prototype heater rods and for in-depth study of existing rods in pressurized water systems. The test section of the loop is easily replaced to facilitate the installation of various length and diameter heater rods. The Single Rod Loop is electrically controlled and operated by one person. Steady-state and blowdown at various conditions can be simulated in the loop. The main test section can be replaced with a quartz tube, and DNB phenomenon can be observed on a single rod with a remotely operated camera.

Characteristics of Single Rod Loop

Maximum Operating Pressure	2250 psia
Maximum Operating Temperature	650°F
Maximum Flowrate	10 gpm
System Capacity	5 gallons
Maximum Power Available	200 kW
Piping Size	1 and 3 inch

1.5.3.2.10 Hydraulic Model Testing

Miscellaneous hydraulic tests on mock-ups of reactor system parts and components are routinely performed at the Test Engineering Laboratory. Typical of this type of testing are the two discussed below, which were recently completed:

1. Emergency Core Cooling Flow Distribution

   A 10 x 10 rod bundle was installed in a plastic housing with a water supply at the top. A grid collection unit at the bottom of the bundle collected the water as it flowed through the model and diverted it to the measuring tubes at the base. Knowledge of the flow distribution in the bundle was obtained in this manner.

2. Sample System Mixing Test

   This test used one thermocouple to measure the temperature of water from four locations in a reactor. The purpose of the procedure was to determine whether the indication from the single thermocouple was representative of the average temperature of the four water supplies. A mock-up of the mixing chamber was constructed so that hot or cold water (at closely controlled pressure) could be supplied to any of the four inlets. By running combinations of hot and cold inlets and making simultaneous recordings of the various temperatures, highly useful information was obtained.

1.5.3.2.11 Autoclave Testing

The Test Engineering Laboratory is equipped with autoclaves ranging in size from 1/2 gallon to 100 gallons. These devices are in constant use to determine the effect of various water chemistries on core components, as well as to perform corrosion tests. The units have also been used as boilers to provide steam for miscellaneous development tests, including acoustic leak detection.

1.5.3.2.12 Mechanical Component and Vibration Test

Full scale mechanical and vibration tests are performed at the Test Engineering Laboratory on plant and reactor components to prove the reliability of equipment design. Vibration testing of reactor components is also performed in the laboratory, using electronically excited shaker heads. Three sizes are available (2 lbs, 50 lbs, and 150 lbs) for regular scale model testing for frequencies from 5 Hz to 50 Hz.

1.5.3.2.13 Electrical Component Assembly

Highly skilled technicians are available at the Test Engineering Laboratory for constructing complex control and instrumentation systems. Work is initiated with engineering ideas and sketches, and includes mounting of process controllers, recorders, meters, relay logic, protection circuits, switches and indicators.

Point-to-point wiring is used, as required. Final as-built drawings are prepared, inspection and thorough electrical checkout is performed before installation in a facility.

1.5.3.2.14 Surveillance System Development

Surveillance systems provide on-line monitoring of pressure vessels for flaws. Electronic components are being developed at the Test Engineering Laboratory for an acoustic emission monitoring system for inservice inspection of operating plant vessels and piping. This system is designed to detect and locate initiation and propagation of cracks at various locations, such as welds and stress risers. Vessel flaw growth and rupture data have been obtained through joint programs at the Idaho National Engineering Laboratories and at the Oak Ridge National Laboratories. Pipe rupture data has been obtained from NRC (AEC) sponsored tests, and hydrostatic test data, operational noise and attenuation characteristics have been measured at various Westinghouse operating plants.

1.5.3.2.15 Engineering Mechanics Laboratory

Bench tests are performed in fixtures designed for the particular test using standard test equipment and techniques.

1.5.4 REFERENCES

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

7. Burnett, A. J. and Kopelic, S. D., "Westinghouse ECCS Evaluation Model-October 1975 Version," WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November,1975.

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9. "A Temperature Sensitivity Study of Single Rod Burst Tests," WCAP-8289, Addendum 1 (Proprietary) and WCAP-8290, Addendum 1 (Non-Proprietary), December, 1975.

TABLE 1.5-1

DELAYED DEPARTURE FROM NUCLEATE BOILING

PHASE I TEST PARAMETERS

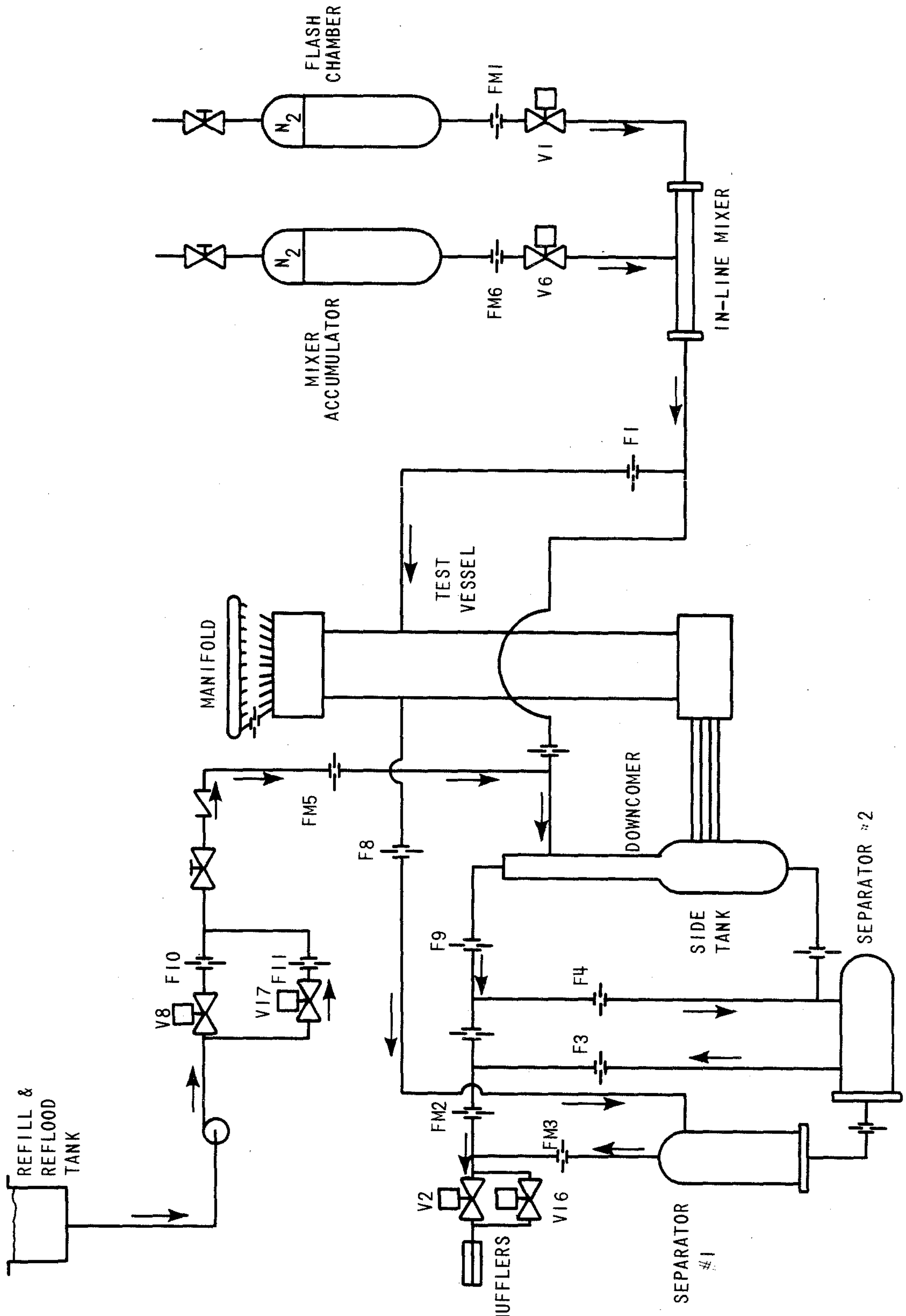
Parameters	Nominal Value
INITIAL STEADY-STATE CONDITIONS	
Pressure	2250 psia
Test section mass velocity	2.5×10^6 lb/hr-ft^2
Inlet coolant temperature	560°F
Maximum heat flux	531,000 BTU/hr-ft^2
TRANSIENT CONDITIONS	
Simulated break	Double ended cold leg guillotine breaks

TABLE 1.5-2

DELAYED DEPARTURE FROM NUCLEATE BOILING

PHASE II TEST PARAMETERS

Parameters	Nominal Value
INITIAL STEADY-STATE CONDITIONS	
Pressure	1750 to 1900 psia
Test section mass velocity	2.0 to 3.0×10^6 lb/hr-ft^2
Core inlet temperature	530 to 560°F
Maximum heat flux	440,000 to 560,000 BTU/hr-ft^2
TRANSIENT RAMP CONDITIONS	
Pressure decrease	0 to 350 psi/sec (subcooled depressurization)
Flow decrease	0 to 100% sec
Inlet enthalpy	Constant



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August 1984

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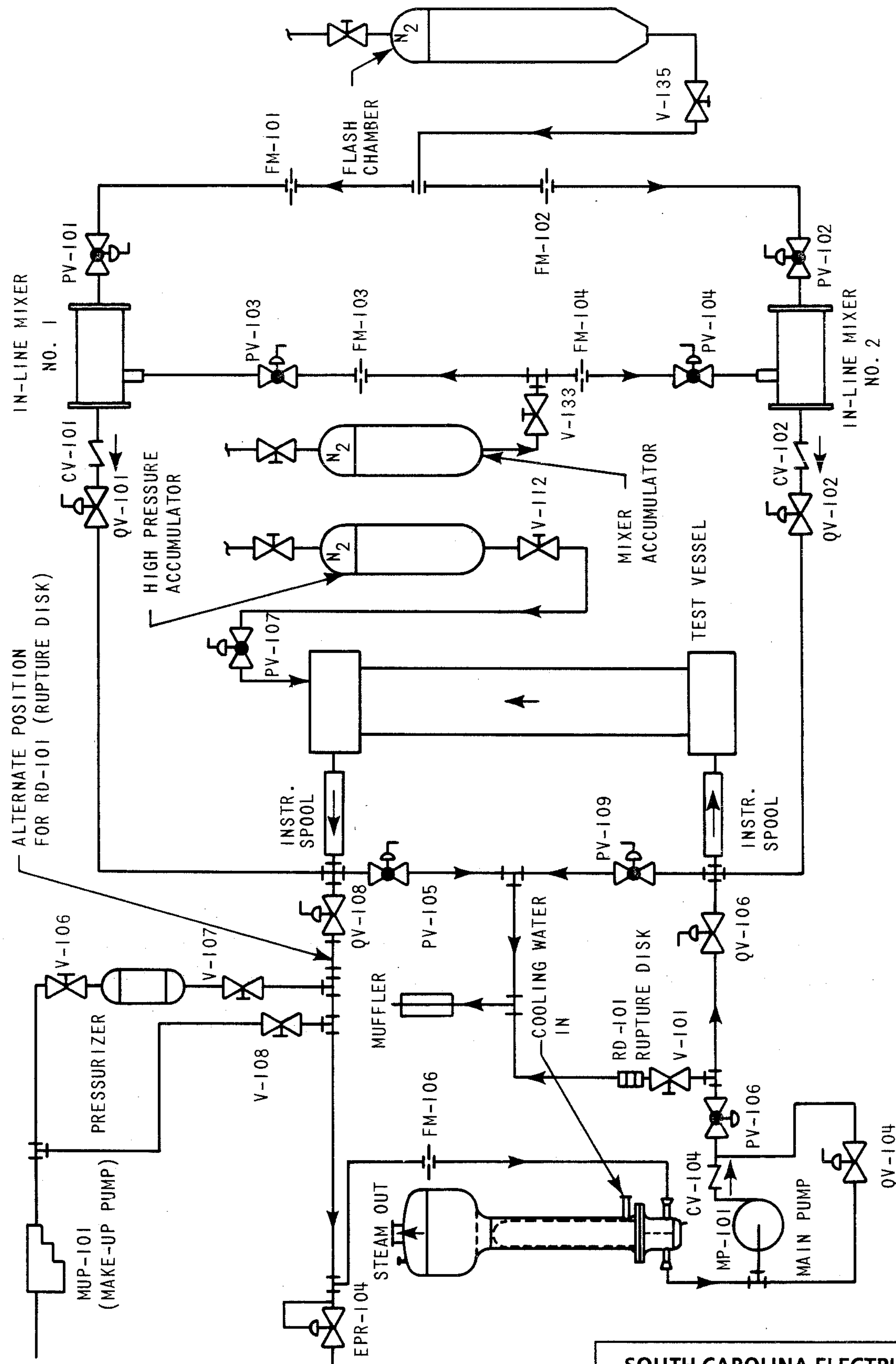
Schematic of G - Loop Test Facility Figure 1.5-1



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VIRGIL C. SUMMER NUCLEAR STATION

Schematic of J - Loop
Test Facility

Figure 1.5-2

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NOTE 1.6
Section 1.6 is being retained for historical purposes only, 99-01

1.6 MATERIAL INCORPORATED BY REFERENCE

This section lists topical reports, referenced throughout this FSAR, which provide information additional to that provided in this FSAR, and have been filed separately with the NRC in support of this and similar applications.

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"Single Phase Local Boiling and Bulk Boiling Pressure Drop Correlations," WCAP-2850, (Proprietary), April 1966 and WCAP-7916 (Non-Proprietary), June 1972	4.4	99-01
"In-Pile Measurement of UO_2 Thermal Conductivity," WCAP-2923, 1966	4.4	
"Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, June 1964	4.4	
"LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM - 7094," WCAP-3269-26, September 1963	4.3, 15.1, 15.3	99-01
"Saxton Core II Fuel Performance Evaluation," WCAP-3385-56, Part II, "Evaluation of Mass Spectrometric and Radio-chemical Analyses of Irradiated Saxton Plutonium Fuel," July 1970	4.3, 4.4	
"Xenon-Induced Spatial Instabilities in Large PWRs," WCAP-3680-20, (EURAEC-1974) March 1968	4.3	
"Control Procedures for Xenon-Induced X-Y Instabilities in Large PWR’s," WCAP-3680-21, (EURAEC-2111) February 1969	4.3	
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"Pressurized Water Reactor pH - Reactivity Effect Final Report," WCAP-3696-8, (EURAEC-2074) October 1968	4.3	

Report	Reference Section(s)	
"PUO_2 - UO_2 Fueled Critical Experiments," WCAP-3726-1, July 1967	4.3	
"Melting Point of Irradiated UO_2," WCAP-6065, February 1965	4.2, 4.4	
"Burnup Physics of Heterogeneous Reactor Lattices," WCAP-6069, June 1965	4.4	
"LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," WCAP-6073, April 1966	4.3	
"Supplementary Report on Evaluation of Mass Spectrometric and Radio-chemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium Through Curium," WCAP-6086, August 1969	4.3	
"Subchannel Thermal Analysis of Rod Bundle Cores," WCAP-7015, Revision 1, January 1969	4.4	
"The PANDA Code," WCAP-7048-P-A (Proprietary) and WCAP-7757-A (Non-Proprietary), January 1975	4.3	
"The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Non-Proprietary), January 1975	4.3, 15.1, 15.3	99-01
"Core Power Capability in Westinghouse PWRs," WCAP-7267-L (Proprietary), October 1969 and WCAP-7809 (Non-Proprietary), December 1971	4.3	
"Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," WCAP-7306, April 1969	15.4	
"Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7308-L (Proprietary) and WCAP-7810 (Non-Proprietary), December 1971	4.3	
"Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation," WCAP-7332-L-AR (Proprietary) and WCAP-7822-AR (Non-Proprietary), November 1973	3.9	
"Application of the THINC Program to PWR Design," WCAP-7359-L (Proprietary), August 1969 and WCAP-7838 (Non-Proprietary), January 1972	4.4, 15.2, 15.3, 15.4	99-01

Report	Reference Section(s)	
"Seismic Testing of Electrical and Control Equipment," WCAP-7397-L (Proprietary), February 1970 and WCAP-7818 (Non-Proprietary), December 1971	3.10	
"Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)," WCAP-7397-L, Supplement 1 (Proprietary), January 1971 and WCAP-7817, Supplement 1 (Non-Proprietary), December 1971	3.10	
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"An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486-L (Proprietary), March 1971 and WCAP-7486 (Non-Proprietary), May 1971	15.2	
"Solid-State Logic Protection System Description," WCAP-7488-L (Proprietary), March 1971 and WCAP-7672 (Non-Proprietary), May 1971	7.1, 7.2, 7.3	
"Radiological Consequences of a Fuel Handling Accident," WCAP-7518-L (Proprietary) and WCAP-7828 (Non-Proprietary), June 1970	15.4	
"Process Instrumentation for Westinghouse Nuclear Steam Supply Systems," WCAP-7547-L (Proprietary), March 1971 and WCAP-7671 (Non-Proprietary), May 1974	5.2	

Report	Reference Section(s)	
"Seismic Vibration Testing with Sine Beats," WCAP-7558, October 1971	3.10	
"An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975	4.4, 15.4	
WCAP-7918, Revision 2 (Non-Proprietary), July 1975	3.9	99-01
"Dynamic Fracture Toughness Properties of Heavy Section A533 Grade B Class 1 Steel Plate," WCAP-7623, December 1970	5.2	
"Interchannel Thermal Mixing with Mixing Vane Grids," WCAP-7667-P-A (Proprietary) and WCAP-7755-A (Non-Proprietary), January 1975	4.4	
"DNB Tests Results for New Mixing Vane Grids (R)," WCAP-7695-P-A (Proprietary) and WCAP-7958-A (Non-Proprietary), January 1975	4.4	
"DNB Test Results for R Grid Thimble Cold Wall Cells," Addendum 1-A (Non-Proprietary), January 1975	4.4	
"Testing of Engineered Safety Features Actuation System," WCAP-7705, Revision 2, January 1976	7.3	
"An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L (Proprietary) and WCAP-7706 (Non-Proprietary), February 1973	7.1, 7.2	
"Electric Hydrogen Recombiner for Water Reactor Containments," WCAP-7709-L (Proprietary), July 1971 and WCAP-7820 (Non-Proprietary), December 1971	7.3	99-01
"Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 1 (Proprietary), May 1974 and WCAP-7820, Supplement 1 (Non-Proprietary), May 1972	7.3	99-01
"Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 2 (Proprietary) and WCAP-7820, Supplement 2 (Non-Proprietary), November 1973	7.3	99-01
"Electric Hydrogen Recombiner for PWR Containments-Long Term Tests," WCAP-7709-L, Supplement 3 (Proprietary) and WCAP-7820, Supplement 3 (Non-Proprietary), January 1974	7. 3	99-01

Report	Reference Section(s)	
"Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 4 (Proprietary) and WCAP-7820, Supplement 4 (Non-Proprietary), April 1974	7.3	99-01
"Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 5 (Proprietary) and WCAP-7820, Supplement 5 (Non-Proprietary), December 1975	7.3	99-01
"Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 6 (Proprietary) and WCAP-7820, Supplement 6 (Non-Proprietary), October 1976	7.3	99-01
"Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplement 7 (Proprietary) and WCAP-7820, Supplement 7 (Non-Proprietary), August 1977	3.10, 7.3, 8.3	99-01
"A Comprehensive Space-Time Dependent Analysis of Loss of Coolant (SATAN-IV Digital Code)," WCAP-7750, August 1971	3.6	
"Westinghouse PWR Internals Vibration Summary 3-Loop Internals Assurance," WCAP-7765-L-AR (Proprietary) and WCAP-7765-AR (Non-Proprietary), December 1973	3.9	
"Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971	15.2	
"Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972	5.2	
"Nuclear Fuel Division Quality Assurance Program Plan," WCAP-7800, Revision 4-A, April 1975	4.2, 17.1	
"Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)," WCAP-7817, Supplement 2, December 1971	3.10	
"Seismic Testing of Electrical and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)," WCAP-7817, Supplement 3, December 1971	3.10	
"Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants)," WCAP-7817, Supplement 4, November 1972	3.10	

Report	Reference Section(s)	
"Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel) (Low Seismic Plants)," WCAP-7817, Supplement 5, March 1975	3.10	
"Seismic Testing of Electrical and Control Equipment, Static Inverter and Instrument Bus Distribution Panel," WCAP-7821, Supplement 2, Addendum 1, October 1975	3.10	
"Evaluation of Steam Generator Tube, Tube Sheet and Divider Plate Under Combined LOCA Plus SSE Conditions," WCAP-7832, December 1973	3.9	99-01
"Inlet Orificing of Open PWR Cores," WCAP-7836, January 1972	4.4	
"Plant Startup Test Report, H. B. Robinson Unit No. 2" WCAP-7844, January 1972	4.3	
"Neutron Shielding Pads," WCAP-7870, May 1972	3.9, 4.2	99-01
"LOFTRAN Code Description", WCAP-7907-P-A, April 1984	15.1, 15.2, 15.3, 15.4	
""FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972 WCAP-7908-A, December, 1989	15.1, 15.2, 15.3, 15.4	99-01
"Power Peaking Factors," WCAP-7912-P-A (Proprietary) and WCAP-7912-A (Non -Proprietary), January 1975	4.3, 4.4	99-01
"Process Instrumentation for Westinghouse Nuclear Steam Supply Systems," WCAP-7913, January 1973	7.2, 7.3	
"Indian Point No. 2 Primary Loop Vibration Test Program", WCAP-7920, September 1972	3.9	
"Damping Values of Nuclear Power Plant Components," WCAP-7921-AR, May 1974	3.7	
"Basis for Heatup and Cooldown Limit Curves," WCAP-7924-A, April 1975	5.2, 5.5	99-01
"Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-P-A (Proprietary) and WCAP-7959-A (Non-Proprietary), January 1975	1.5, 4.4	
"Fuel Assembly Safety Analysis for Combined Seismic and Loss of Coolant Accident," WCAP-7950, July 1972	3.7, 3.9	

Report	Reference Section(s)	
"THINC-IV An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973	4.4	
"Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor," WCAP-7964, June 1971	4.3	
"Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP-7973, September 1972	15.3	99-01
"TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975	15.1, 15.2, 15.4	99-01
"Application of the THINC-IV Program to PWR Design," WCAP-8054 (Proprietary) and WCAP-8195 (Non-Proprietary), October 1973	4.4	99-01
"Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," WCAP-8082-P-A (Proprietary) and WCAP-8172-A (Non-Proprietary), January 1975	3.6	
"Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973	5.5	
"Calculational Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), June 1974	15.4	
"Operational Experience with Westinghouse Cores," WCAP-8183, latest revision Reference	4.2	
"Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary) and WCAP-8219-A (Non-Proprietary), March 1975	4.1, 4.2, 4.3, 4.4, 15.3	99-01
"Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236 (Proprietary), December 1973 and WCAP-8288 (Non-Proprietary), January 1974	1.5, 3.7, 4.2	99-01
"Safety Analysis for the 8-Grid 17 x 17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236, Addendum 1 (Proprietary), March 1974 and WCAP-8288, Addendum 1 (Non-Proprietary), April 1974	3.7	

Report	Reference Section(s)	
"Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, April 1974	3.6, 5.2	
"Source Term Data for Westinghouse Pressurized Water Reactors," WCAP-8253, Revision 1, July 1975	11.1, 12A	99-01
"Nuclear Instrumentation System," WCAP-8255, January 1974	7.2, 7.7	
"Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A (Proprietary) and WCAP-8312-A (Non-Proprietary), August 1975	6.2	
"Hydraulic Flow Test of the 17 x 17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974	1.5, 4.2, 4.4	
"17 x 17 Design Fuel Rod Behavior During Simulated LOCA Conditions," WCAP-8289 (Proprietary) and WCAP-8290 (Non-Proprietary), November 1974	1.5	
"A Temperature Sensitivity Study of Single Rod Burst Tests," WCAP-8289, Addendum 1 (Proprietary) and WCAP-8290, Addendum 1 (Non-Proprietary), December 1975	1.5	
"Effect of 17 x 17 Fuel Assembly Geometry on DNB," WCAP-8296, March 1974, WCAP-8296-P-A, February, 1975 (Proprietary) and WCAP-8297, (Non-Proprietary)	4.4.6	99-01
"The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8298-P-A (Proprietary) and WCAP-8299-A (Non-Proprietary), January 1975	4.4	
"LOCTA-IV Program: Loss of Control Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), June 1974	15.3	99-01
"SATAN-IV Program: Comprehensive Space-Time Dependent Analysis of Loss of Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (non-Proprietary), June 1974	15.4	
"Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," WCAP-8303-P-A (Proprietary) and WCAP-8317-A (Non-Proprietary), July 1975	3.9	

Report	Reference Section(s)	
"Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975	3A, 5.2	99-01
"Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974	15.4	
"Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974	4.3, 15.2	
"Westinghouse Nuclear Energy Systems Division Quality Assurance Plan," WCAP-8370	17.1	99-01
"Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974," WCAP-8373, August 1974	3.10	
"Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974	4.2	
"Power Distribution Control and Load Follow Procedures," WCAP-8385 (Proprietary) and WCAP-8403 (Non-Proprietary), September 1974	4.3	99-01
"An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975	15.3	
"17 x 17 Drive Line Components Tests - Phase IB, II, III, D-Loop Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), December 1974	1.5, 3.9	99-01
"Analysis of Data from the Zion (Unit 1) THINC Verification Test," WCAP-8453-A, May 1976	4.4	
"Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471-P-A (Proprietary) and WCAP-8472-A (Non-Proprietary), April 1975	15.4	
"Methods for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients," WCAP-8510, June 1976	5.2	
"Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22 Inch Grid Spacing," WCAP-8536 (Proprietary) and WCAP-8537 (Non-Proprietary), May 1975	1.5, 4.4	
"Improved Thermal Design Procedure," WCAP-8567, July 1975.	4.4	99-01

Report	Reference Section(s)	
"Westinghouse ECCS - Three Loop Plant (17 x 17) Sensitivity Studies," WCAP-8572-P (Proprietary) and WCAP-8573 (Non-Proprietary), July 1975	15.4	
"Westinghouse ECCS Evaluation Model - October 1975 Version," WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November 1975	1.5	99-01
"General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary) and WCAP-8695 (Non-Proprietary), September 1975	3.10	
"Experimental Verification of Wet Fuel Storage Criticality Annuluses," WCAP-8682 (Proprietary) and WCAP-8683 (Non-Proprietary), December 1975	4.3	
"Fuel Rod Bowing," WCAP-8691 (Proprietary) Revision 1, 1979	4.2, 4.4	99-01
"Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976	3A, 5.2	99-01
"MULTIFLEX - A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708 (Proprietary) and WCAP-8709 (Non-Proprietary), February 1977	3.9, 5.2	99-01
"New Westinghouse Correlations WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A, July 1976.	4.4, 15.1	99-01
"Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary), October 1976	4.2	
"Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, Spring 1976," WCAP-8768, June 1976	1.5, 4.2, 4.3	
"Seismic Operability Demonstration Testing of the WISD 7300 Series Process Instrumentation System Bistables," WCAP-8828 (Proprietary) and WCAP-8829 (Non-Proprietary), October 1976	3.10	
"Seismic Operability Demonstration Testing of the Nuclear Instrumentation System Bistables," WCAP-8830 (Proprietary) and WCAP-8831 (Non-Proprietary), October 1976	3.10	

Report	Reference Section(s)	
"7300 Series Process Control System Noise Tests," WCAP-8892-A, June 1977	7.2	99-01
Gilbert/Commonwealth Companies, "Computation of Radiological Consequences Using INHEC Computer Program," GAI-TR-101P-A (Proprietary), GAI-TR-101-A (Non-Proprietary), March 1976	15.4	
Gilbert/Commonwealth Companies, "Hydraulic Analysis of Piping Networks Using PIPF Computer Program," GAI-TR-105P-A (Proprietary), GAI-TR-105-A (Non-Proprietary), June 1978	6.2	
"Setpoint Study SCE&G V. C. Summer Nuclear Plant," WCAP-9399, December 1978.	15.1, 15.4	99-01
"The 1981 Version of the Westinghouse ECCS Evaluation Model Using the Bash Code," WCAP-10266-P-A Rev. 2, 1987.	15.4	
"Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP-10298-A (Non-Proprietary), June 1983	15.2	99-01
"Application of the THINC Program to PWR Design," WCAP-7838, January 1972 (Non-Proprietary)	15.3, 15.4	99-01
"Westinghouse Emergency Core Cooling System Plant-Sensitivity Studies," WCAP-08340, July 1974 (Proprietary) WCAP-8356 (Non-Proprietary)	15.3, 15.4	
"Westinghouse Emergency Core Cooling System Evaluation Model-Sensitivity Studies," WCAP-08341, July 1974	15.4	
"Westinghouse Emergency Core Cooling System Evaluation Model-Sensitivity Studies," WCAP-08342, July 1974.	15.4	
"Westinghouse Emergency Core Cooling System Evaluation Model-Summary," WCAP-08339, June 1974.	6.2, 15.4	99-01
"Structural Analysis of Reactor Coolant Loop for the V. C. Summer Nuclear Station Vol. 1 Piping Analysis of the Reactor Coolant Loop," WCAP-09119, August 1977.	5.2	
"Westinghouse ECCS Evaluation Model-February 1981 Version," WCAP-09220-P-A, February 1982.	6.2, 15.4	99-01

Report	Reference Section(s)	
"Westinghouse ECCS Evaluation Model-February 1981 Version," WCAP-09221, February 1981.	6.2, 15.4	99-01
"Reactor Core Response To Excessive Secondary Steam Releases", WCAP-09226-R1, September 1978.	4.4, 15.2, 15.4	99-01
"Report on Small Break Accidents for Westinghouse Nuclear Steam Supply System," WCAP-09601, June 1979.	6.3, 15.3	99-01
"Westinghouse Delta 75 Steam Generator Design and Fabrication Information For The V. C, Summer Nuclear Station," WCAP-13480 Rev. 1, October 1993.	5.5	
"Technical Justification For Eliminating Large Primary Loop Pipe Rupture As The Structural Design Basis For The V. C. Summer Nuclear Power Plant," WCAP-13206, April 1992.	3.6, 5.5	99-01
"Qualification Of The Phoenix-P/ANC Nuclear Design System For Pressurized Water Reactor Cores," WCAP-11596-P-A Original Version: November 1987 Approved Version: June 1988.	4.3, 15.3	99-01 02-01
"Methodology For The Analysis Of The Dropped Rod Event," WCAP-11394-P-A, January 1990.	15.2	
"Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code," WCAP-11145-P-A, 1986	15.3	99-01
"Revised Thermal Design Procedure," WCAP-11397-P-A Original Version: February 1987 Approved Version: April 1989.	4.4, 15.1, 15.2, 15.4	
"Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A Original Version: December 1983 Approved Version: September 1985.	4.1, 15.4	
"Methodology For The Analysis Of The Dropped Rod Event," WCAP-11395-A, January 1990.	15.2	02-01
"Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.	4.2, 4.4	99-01

Report	Reference Section(s)	
"DNB Test Results for R-Grid Thimble Cold Wall Cells," WCAP-7695-L Addendum 1, October 1972.	4.4	
"Improved THINC IV Modeling for PWR Core Design," WCAP-12330-P-A, September 1991.	4.4	
"Reactor Pressure Vessel and Internals System Evaluations for the Virgil C. Summer Steam Generator Replacement/Uprating Program," WCAP-13712, April 1993.	4.4, 5.2	
"Addendum to Final (Summary) Stress Report 157" PWR Vessel Virgil C. Summer Nuclear Station Revision 2 (Uprating/Steam Generator Replacement Evaluation)," WCAP-13777, June 1993.	5.2	
"Westinghouse LOCA Mass and Energy Release Model for Containment Design," WCAP-10325-P-A March 1979 (Proprietary) WCAP-10326-A, May 1983(Non-Proprietary).	6.2	
"NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.	15.3	
"Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.	15.3	
LOCBART, Version 14 - "Computer Code for Hot Assembly Thermal Hydraulic Transients"	15.4	99-01
"Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-NP-A, Revision 2, January 1996.	5.2, 5.5	
"Analysis of Capsule W From the South Carolina Electric & Gas Company V. C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-15101, Revision 0, September 1998.	5.2	
"V. C. Summer Unit 1 Heatup and Cooldown Limit Curves For Normal Operation," WCAP-15102, Revision 2, October 1999.	5.2	02-01
"Evaluation of Pressurizer thermal Shock for V. C. Summer Unit 1," WCAP-15103, Revision 0, September 1998.	5.2	
"Beacon Core Monitoring and Operations Support System," WCAP-12472-P-A, August, 1994.	4.3, 4.4, 7.7, 15.3	
"Control Rod Insertion Following a Cold Leg LOCA Generic Analysis for 3-Loop and 4-Loop Plants," WCAP-15704 (Proprietary), October 2001.	15.4	RN 02-052

NOTE 1.7
Section 1.7 is being retained for historical purposes only. 99-01

1.7 ELECTRICAL, INSTRUMENTATION AND CONTROL DRAWINGS

Electrical, instrumentation and control drawings have been provided under separate cover in the Wiring Schematic Package. Included in this package is a listing of drawings with drawing number, title, and system sheet number. This list is presented herein as Table 1.7-1. 98-01

TABLE 1.7-1

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-004	AH024	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	02-01
B-208-004	AH025	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	
B-208-004	AH102	Control Room Emer Filt Sys Fan A (XFN30A) & Dmprs (XDP23A & XDP24A)	
B-208-004	AH103	Control Room Emer Filt Sys Fan A (XFN30A) & Dmprs (XDP23B & XDP24B)	02-01
B-208-004	AH104	Control Room Normal Supply Fan A (XFN32A) & Dmpr (XDP105A)	
B-208-004	AH105	Control Room Normal Supply Fan B (XFN32B) & Dmpr (XDP105B)	
B-208-004	AH107	Relay Room Cooling System Fan A (XFN36A) (Unit 1)	
B-208-004	AH108	Relay Room Cooling System Fan B (XFN36B) (Unit 1)	
B-208-004	AH112	Computer Room Supply Fan B (XFN41B)	
B-208-004	AH121	Control Room Supp Inlet DPRS “A” (XDP22A, 239A) & Butterfly Isol Valves “A” (XVB0003A, 0004A)	
B-208-004	AH122	Control Room Supp Inlet DPRS “B” (XDP22B, 239B) & Butterfly Isol Valves “B” (XVB0003B, 0004B)	
B-208-004	AH123	HVAC Control Board Status Light Inputs	
B-208-004	AH124	HVAC Control Board Status Light Inputs	
B-208-004	AH125	HVAC Control Board Status Light Inputs	
B-208-004	AH126	HVAC Control Board Status Light Inputs	
B-208-004	AH129	HVAC Control Board Graphic Display Annun Alarm Inputs	
B-208-004	AH130	HVAC Control Board Graphic Display Annun Alarm Inputs	
B-208-004	AH131	HVAC Control Board Graphic Display Annun Alarm Inputs	
B-208-004	AH132	HVAC Control Board Graphic Display Annun Alarm Inputs	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-004	AH141	Relay, Computer Room & Instrument Repair Room Cooling System	
B-208-004	AH142	Relay, Computer Room & Instrument Repair Room Cooling System	
B-208-004	AH143	MCR Ventilation Isolation Normal and Emerg. Loop A	
B-208-004	AH144	MCR Ventilation Isolation Normal and Emerg. Loop B	
B-208-004	AH145	MCR Ventilation Isolation Normal and Emerg. Loop B	
B-208-004	AH146	MCR Ventilation Isolation Normal and Emerg. Loop B	
B-208-004	AH147	Controlled Access Damper Control XDP-45	
B-208-004	AH148	Controlled Access Damper Control XDP-106	
B-208-004	AH149	MCR Toilet Exhaust Relief Dampers XDP245A&B	
B-208-004	AH164	Diesel Generator Room B Supply Fan A (XFN45A)	
B-208-004	AH165	Diesel Generator Room B Supply Fan B (XFN45B)	
B-208-004	AH166	Diesel Generator Room A Supply Fan A (XFN75A)	RN 01-108
B-208-004	AH167	Diesel Generator Room A Supply Fan B (XFN75B)	
B-208-004	AH170	HVAC Control Board Graphic Display Annun Alarm Inputs	
B-208-004	AH174	Fuel Handling Building Exhaust Fan A (XFN23A) & Dmprs (XDP12A & XDP13A)	02-01
B-208-004	AH175	Fuel Handling Building Exhaust Fan B (XFN23B) & Dmprs (XDP12B & XDP13B)	
B-208-004	AH177	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	
B-208-004	AH178	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	
B-208-004	AH179	Vibration Switches YS9652, YS9684A, YS9684B	
B-208-004	AH182	Battery Room ‘A’ Dampers (XDP152) & (XDP155)	
B-208-004	AH183	Battery Room ‘B’ Dampers (XDP153) & (XDP156)	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-004	AH194	Battery Room Supply Fan A (XFN38A) & Damper (XDP89A)	
B-208-004	AH195	Battery Room Supply Fan B (XFN38B) & Damper (XDP89B)	
B-208-004	AH196	Battery Room Exhaust Fan A (XFN39A) & Damper (XDP88A)	
B-208-004	AH197	Battery Room Exhaust Fan B (XFN39B) & Damper (XDP88B)	
B-208-004	AH202	Vibration Switch YS9886A Battery Room IB	02-01
B-208-004	AH204	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	
B-208-004	AH205	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	
B-208-004	AH206	Vibration Switch YS9886B Battery Room IB	
B-208-004	AH208	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	02-01
B-208-004	AH209	HVAC Control Board Graphic Display Annun Alarm and Status Light Inputs	
B-208-004	AH236	Reactor Building Vibration Switch Reset Pushbutton	
B-208-004	AH245	Reactor Building Purge Sup Isol XVB-1A	
B-208-004	AH246	Reactor Building Purge Sup Isol XVB-1B	
B-208-004	AH247	Reactor Building Purge Exh Isol XVB-2A	
B-208-004	AH248	Reactor Building Purge Exh Isol XVB-2B	
B-208-004	AH249	Misc. Alarms - Reactor Building Air Handling System	
B-208-004	AH250	Misc. Alarms - Reactor Building Air Handling System	
B-208-004	AH251	Misc. Alarms - Reactor Building Air Handling System	
B-208-004	AH252	Misc. Alarms - Reactor Building Air Handling System	
B-208-004	AH273	Reactor Building Cooling Unit Fan A (XFN64A) Motor (MFN97A)	
B-208-004	AH275	Reactor Building Cooling Unit Fan B (XFN64B) Motor (MFN97B)	02-01

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-004	AH277	Reactor Building Cooling Unit Fan A (XFN65A) Motor (MFN97C)	
B-208-004	AH279	Reactor Building Cooling Unit Fan B (XFN65B) Motor (MFN97D)	
B-208-004	AH281	RBCU HEPA Filter Bypass Damper XDP110A	
B-208-004	AH282	RBCU HEPA Filter Bypass Damper XDP110B	
B-208-004	AH283	RBCU HEPA Filter Bypass Damper XDP111A	
B-208-004	AH284	RBCU HEPA Filter Bypass Damper XDP111B	
B-208-004	AH285	Reactor Building - Smoke Detectors XA9299A & XA9299B	
B-208-004	AH286	Reactor Building - Smoke Detectors XA9299C & XA9299D	
B-208-004	AH287	Reactor Building - Vibration Switches IYE40000, IYE40001, IYE40002, & IYE40003	RN 09-023
B-208-004	AH326	Service Water Pumphouse Supply Fan ‘A’ (XFN80A) & Damper (XDPR74A)	
B-208-004	AH327	Service Water Pumphouse Supply Fan ‘B’ (XFN80B) & Damper (XDPR74B)	
B-208-004	AH328	Dampers XDP-71A, 72A, 73A Mode Control	02-01
B-208-004	AH329	Dampers XDP-71B, 72B, 73B Mode Control	
B-208-004	AH330	HVAC Control Board Graphic Display Annun Alarm & Status Light Inputs	
B-208-004	AH331	HVAC Control Board Graphic Display Annun Alarm & Status Light Inputs	
B-208-004	AH334	Vibration Switches for XFN - 80A, 80B	
B-208-004	AH399	HVAC Control Board Graphic Display Annun Alarm & Status Light Inputs	
B-208-004	AH400	HVAC Control Board Graphic Display Annun Alarm & Status Light Inputs	
B-208-008	AS50	Aux Steam to Aux Bldg Isolation XVG-265	02-01
B-208-008	AS51	Aux Steam to Aux Bldg Isolation XVG-273	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-009	BD07	S/G A Blowdown Isolation Valve XVG-503A
B-208-009	BD08	S/G B Blowdown Isolation Valve XVG-503B
B-208-009	BD09	S/G C Blowdown Isolation Valve XVG-503C
B-208-009	BD10	Steam Gen. Blowdown Isolation
B-208-009	BD12	Miscellaneous Alarms - Blowdown System
B-208-011	CC01	Component Cooling A Pump (XPP1A)
B-208-011	CC02	Component Cooling B Pump (XPP1B)
B-208-011	CC03	Component Cooling C Pump (XPP1C) (Channel A)
B-208-011	CC04	Component Cooling C Pump (XPP1C) (Channel B)
B-208-011	CC10	RHR Heat Exchanger Inlet Valve XVB9503A
B-208-011	CC11	RHR Heat Exchanger Inlet Valve XVB9503B
B-208-011	CC12	Non-Essential Equip. Isolation Valve XVB9524A
B-208-011	CC13	Non Essential Equip. Isolation Valve XVB9524B
B-208-011	CC14	Non-Essential Equip. Isolation Valve XVB9525A
B-208-011	CC15	Non-Essential Equip. Isolation Valve XVB9525B
B-208-011	CC16	Non-Essential Equip. Isolation Valve XVB9526A
B-208-011	CC17	Non-Essential Equip. Isolation Valve XVB9526B
B-208-011	CC18	Reactor Coolant & Containment Isolation Valve XVG9568
B-208-011	CC24	R.C. Pump Thermal Barrier Isolation Valve XVG9600
B-208-011	CC25	R.C. Pump Thermal Barrier Isolation Valve XVG9605
B-208-011	CC26	R.C. Pump Thermal Barrier Isolation Valve XVB9606

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-011	CC27	Non-Essential Equip. Isolation Valve XVB9687A
B-208-011	CC28	Non-Essential Equip. Isolation Valve XVB9687B
B-208-011	CC29	Non-Essential Loop to Comp. Clg. Bstr. Pp. Isol. Valve XVG9625
B-208-011	CC30	Non-Essential Loop to Comp. Clg. Bstr. Pp. Isol. Valve XVG9626
B-208-011	CC31	Component Cooling Pump 1A Speed Switch XES2001A
B-208-011	CC32	Component Cooling Pump 1A Speed Switch XES2001B
B-208-011	CC33A	Component Cooling Pump 1C Speed Switch XES2001C
B-208-011	CC33B	Component Cooling Pump 1C Speed Switch XES2001C
B-208-011	CC52	S.W. Supply to Component Cooling XVG9627A
B-208-011	CC53	S.W. Supply to Component Cooling XVG9627B
B-208-011	CC60	Component Cooling System Alarms
B-208-011	CC61	Component Cooling System Alarms
B-208-011	CC62	Miscellaneous Alarms - Component Cooling System
B-208-011	CC63	Miscellaneous Alarms - Component Cooling System
B-208-020	CR01	Reactor Trip Switchgear XSW0001A
B-208-020	CR01A	Reactor Trip Switchgear XSW0001A
B-208-020	CR02	Reactor Trip Switchgear XSW0001B
B-208-020	CR02A	Reactor Trip Switchgear XSW0001B
B-208-020	CR55	Full Length Bank Selector Switch
B-208-021	CS01	Boric Acid Transfer Pump A (XPP13A)
B-208-021	CS02	Boric Acid Transfer Pump B (XPP13B)

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-021	CS04	Charging/Safety Injection Pump C Transfer SW XET21002C Elec. Connections	
B-208-021	CS05	Charging/Safety Injection Pump A (XPP43A)	
B-208-021	CS06	Charging/Safety Injection Pump B (XPP43B)	02-01
B-208-021	CS07	Charging/Safety Injection Pump C (XPP43C) (Channel A)	
B-208-021	CS08	Charging/Safety Injection Pump C (XPP43C) (Channel B)	02-01
B-208-021	CS09	Bat to Charging Injection Pump Suction Valve 8104 (XVT8104)	
B-208-021	CS10	Seal Water Injection Filter Isol. Valve 8102A (XVT8102A)	02-01
B-208-021	CS11	Seal Water Injection Filter Isol. Valve 8102B (XVT8102B)	
B-208-021	CS12	Seal Water Injection Filter Isol. Valve 8102C (XVT8102C)	
B-208-021	CS13	Charging Pump 1A Miniflow Valve 8109A (XVT8109A)	02-01
B-208-021	CS14	Charging Pump 1B Miniflow Valve 8109B (XVT8109B)	
B-208-021	CS15	Charging Pump 1C Miniflow Valve 8109C (XVT8109C)	
B-208-021	CS16	Charging Pump Miniflow Valve 8106 (XVG8106)	
B-208-021	CS17	Seal Water Return Isol. Valve 8100 (XVT8100)	
B-208-021	CS18	Seal Water Return Isol. Valve 8112 (XVT8112)	
B-208-021	CS19	RCS Charging Line Valve 8107 (XVG8107)	
B-208-021	CS20	RCS Charging Line Valve 8108 (XVG8108)	
B-208-021	CS21	Seal Water Injection Valve 8105 (XVT8105)	
B-208-021	CS22	Charging Pump Suction Header Isol. Valve 8130A (XVG8130A)	
B-208-021	CS23	Charging Pump Suction Header Isol. Valve 8130B (XVG8130B)	02-01

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-021	CS24	Charging Pump Suction Header Isol. Valve 8131A (XVG8131A)	
B-208-021	CS25	Charging Pump Suction Header Isol. Valve 8131B (XVG8131B)	
B-208-021	CS26	Charging Pump Discharge Header Isol. Valve 8132A (XVG8132A)	02-01
B-208-021	CS27	Charging Pump Discharge Header Isol. Valve 8132B (XVG8132B)	
B-208-021	CS28	Charging Pump Discharge Header Isol. Valve 8133A (XVG8133A)	
B-208-021	CS29	Charging Pump Discharge Header Isol. Valve 8133B (XVG8133B)	
B-208-021	CS30	Charging Pump A Aux. Lube Oil Pump	
B-208-021	CS31	Charging Pump B Aux. Lube Oil Pump	
B-208-021	CS32	Charging Pump C Aux. Lube Oil Pump	
B-208-021	CS33	Refueling Water Supply Line Stop Valve LCV-115B (XVG0115B)	
B-208-021	CS34	Volume Control Tank Outlet Line Stop Valve LCV-115C (XVG0115C)	
B-208-021	CS35	Refueling Water Supply Line Stop Valve LCV-115D (XVG115D)	
B-208-021	CS36	Volume Control Tank Outlet Line Stop Valve LCV-115E (XVG0115E)	
B-208-021	CS40	XET2002C Transfer Scheme Channel “A”	
B-208-021	CS41	XET2002C Transfer Scheme Channel “B”	
B-208-021	CS54	Boric Acid Injection to Blender FCV-113A	99-01
B-208-021	CS55	Boric Acid Injection to Charging Pump Header FCV-113B	
B-208-021	CS56	Boric Acid Dilution Injection to VCT FCV-168A	
B-208-021	CS57	Reactor Makeup Water Injection to Boric Acid Blender FCV-168B(CS)	
B-208-021	CS58	Letdown to VCT or Holdup Tank LCV-115A	
B-208-021	CS59A	Letdown Line Isolation Valve LCV-459	
B-208-021	CS59B	Letdown Line Isolation Valve LCV-459	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-021	CS60A	Letdown Line Isolation Valve LCV-460	
B-208-021	CS60B	Letdown Line Isolation Valve LCV-460	
B-208-021	CS79	Excess Letdown to VCT XVM-8143	
B-208-021	CS81	Norm. Charging to Reactor Cooling System Isolation XVT-8146	
B-208-021	CS82	Aux. Charging to Reactor Cooling System Isolation XVT-8147	02-01
B-208-021	CS83A	Letdown Orifice Isolation XVT-8149A	
B-208-021	CS83B	Letdown Orifice Isolation XVT-8149A	
B-208-021	CS83C	Letdown Orifice Isolation XVT-8149A	
B-208-021	CS84A	Letdown Orifice Isolation XVT-8149B	
B-208-021	CS84B	Letdown Orifice Isolation XVT-8149B	
B-208-021	CS84C	Letdown Orifice Isolation XVT-8149B	
B-208-021	CS85A	Letdown Orifice Isolation XVT-8149C	
B-208-021	CS85B	Letdown Orifice Isolation XVT-8149C	
B-208-021	CS85C	Letdown Orifice Isolation XVT-8149C	
B-208-021	CS86	Letdown Line Isolation XVT-8152	
B-208-021	CS87	Excess Letdown Line XVT-8153	
B-208-021	CS88	Excess Letdown Line XVT-8154	
B-208-021	CS92	VCT to Vent Hdr. Isolation XVG-8101	
B-208-021	CS94	Boron Meter Isolation Valve 8455	
B-208-021	CS95	Boron Meter Isolation Valve 8461	
B-208-021	CS100	Miscellaneous Alarms - Chemical & Volume Control System	
B-208-021	CS101	Miscellaneous Alarms - Chemical & Volume Control System	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-024	DG01	Diesel Generator 1A Breaker	
B-208-024	DG01A	Diesel Generator 1A Breaker	
B-208-024	DG02	Diesel Generator 1B Breaker	
B-208-024	DG02A	Diesel Generator 1B Breaker	
B-208-024	DG03	Diesel Generator Fuel Transfer “A” Pump (XPP0004A)	
B-208-024	DG04	Diesel Generator Fuel Transfer “B” Pump (XPP0004B)	
B-208-024	DG05	Diesel Generator Fuel Oil Transfer “A” Pump (XPP141A)	
B-208-024	DG06	Diesel Generator Fuel Oil Transfer “B” Pump (XPP141B)	
B-208-024	DG10	Diesel Generator System Misc. Alarms	
B-208-024	DG11	Diesel Generator System Misc. Alarms	
B-208-024	DG12	Diesel Generator System Misc. Alarms	
B-208-024	DG13	Diesel Generator System Misc. Alarms	
B-208-024	DG14	Diesel Generator Status Lights	
B-208-024	DG15	Diesel Generator A Exciter XEX4201	
B-208-024	DG16	Diesel Generator B Exciter XEX4202	
B-208-028	EI01	Earthquake Instrumentation System Alarms	02-01
B-208-032	EF01	Motor Driven Emergency Feedwater Pump A (XPP21A)	
B-208-032	EF02	Motor Driven Emergency Feedwater Pump B (XPP21B)	
B-208-032	EF03	Motor Driven Emerg. F.W. Pump A Suct. Isol. Valve XVG1001A	
B-208-032	EF04	Motor Driven Emerg. F.W. Pump B Suct. Isol. Valve XVG1001B	
B-208-032	EF05	Turb. Driven Emerg. F.W. Pump S.W. Loop B Isol. Valve XVG1002	
B-208-032	EF06	Turb. Driven Emerg. F.W. Pump S.W. Loop A Isol. Valve XVG1008	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-032	EF07	Service Water Loop A Isol. Valve XVG1037A	
B-208-032	EF08	Service Water Loop B Isol. Valve XVG1037B	
B-208-032	EF30A	Diesel Buses Undervoltage Signal	
B-208-032	EF31	Steam Generator A Isolation Valve XVC-1009A	
B-208-032	EF32	Steam Generator B Isolation Valve XVC-1009B	
B-208-032	EF33	Steam Generator C Isolation Valve XVC-1009C	
B-208-032	EF34A	EF Isol. From Motor Driven Pumps to Steam Gen “A” IFV-3531	
B-208-032	EF34B	EF Isol. From Motor Driven Pumps to Steam Gen “A” IFV-3531	
B-208-032	EF35A	EF Isol. From Motor Driven Pumps to Steam Gen “B” IFV-3541	
B-208-032	EF35B	EF Isol. From Motor Driven Pumps to Steam Gen “B” IFV-3541	
B-208-032	EF36A	EF Isol. From Motor Driven Pumps to Steam Gen “C” IFV-3551	
B-208-032	EF36B	EF Isol. From Motor Driven Pumps to Steam Gen “C” IFV-3551	
B-208-032	EF37A	EF Isol. From Turb. Driven Pumps to Steam Gen “A” IFV-3536	
B-208-032	EF37B	EF Isol. From Turb. Driven Pumps to Steam Gen “A” IFV-3536	
B-208-032	EF38A	EF Isol. From Turb. Driven Pumps to Steam Gen “B” IFV-3546	02-01
B-208-032	EF38B	EF Isol. From Turb. Driven Pumps to Steam Gen “B” IFV-3546	
B-208-032	EF39A	EF Isol. From Turb. Driven Pumps to Steam Gen “C” IFV-3556	
B-208-032	EF39B	EF Isol. From Turb. Driven Pumps to Steam Gen “C” IFV-3556	
B-208-032	EF50	Miscellaneous Alarms Emergency Feedwater System	
B-208-032	EF51	Miscellaneous Alarms Emergency Feedwater System	
B-208-032	EF52	Miscellaneous Alarms Emergency Feedwater System	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-037	ES07	7.2 kv Bus 1DB Normal Incoming Breaker
B-208-037	ES08	7.2 kv Bus 1DB Emergency Incoming Breaker
B-208-037	ES09	7.2 kv Bus 1DA Normal Incoming Breaker
B-208-037	ES10	7.2 kv Bus 1DA Emergency Incoming Breaker
B-208-037	ES19	Transformer 1DB1 & 1DB2 Feeder Breaker
B-208-037	ES20	Transformer 1DA1 & 1DA2 Feeder Breaker
B-208-037	ES21	Transformer 1EB1 Feeder Breaker 1EB1
B-208-037	ES22	Transformer 1EA1 Feeder Breaker 1EA1
B-208-037	ES23	Service Water Pump House Bus 1EB Feeder Breaker
B-208-037	ES24	Service Water Pump House Bus 1EA Feeder Breaker
B-208-037	ES36	480V Bus 1DB1 Feeder Breaker
B-208-037	ES37	480V Bus 1DB2 Feeder Breaker
B-208-037	ES38	480V Bus 1DA1 Feeder Breaker
B-208-037	ES39	480V Bus 1DA2 Feeder Breaker
B-208-037	ES40	480V Bus 1EB1 Feeder Breaker
B-208-037	ES41	480V Bus 1EA1 Feeder Breaker
B-208-037	ES47	480V Bus 1B1-1DA1 Tie Breaker
B-208-037	ES48	480V Bus 1B3-1DB1 Tie Breaker
B-208-037	ES53	7.2 kv Switchgear Bus 1DA Control Circuits
B-208-037	ES54	7.2 kv Switchgear Bus 1DB Control Circuits
B-208-037	ES65	Reactor Protection Under Freq. and Under Voltage Relays 7.2 kv Bus 1A, 1B, & 1C

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-037	ES66A	7.2 kv Bus 1DA Undervoltage Relaying
B-208-037	ES66B	7.2 kv Bus 1DA Undervoltage Relaying
B-208-037	ES67A	7.2 kv Bus 1DB Undervoltage Relaying
B-208-037	ES67B	7.2 kv Bus 1DB Undervoltage Relaying
B-208-037	ES80	480 V Switchgear Bus 1DA1 & 1DA2 Control Circuits
B-208-037	ES81	480 V Switchgear Bus 1DB1 & 1DB2 Control Circuits
B-208-037	ES82	480 V Switchgear Bus 1EA1 & 1EB2 Control Circuits
B-208-037	ES100	Miscellaneous Alarms Electrical System
B-208-037	ES102	Miscellaneous Alarms Electrical System
B-208-037	ES107	7 kv Motor Overcurrent Indication Light
B-208-039	EV01	Misc. Alarms Vital Bus 120V Distribution
B-208-039	EV02	Misc. Alarms Vital Bus 120V Distribution
B-208-044	FS01	Fire Service Containment Isolation
B-208-045	FW41	Chemical Feed Containment Isolation Valve XVK1633A
B-208-045	FW42	Chemical Feed Containment Isolation Valve XVK1633B
B-208-045	FW43	Chemical Feed Containment Isolation Valve XVK1633C
B-208-045	FW50	FW Pumps A, B & C Turbines Tripped
B-208-045	FW51	W ESF-FW Isolation Trip
B-208-045	FW61A	FW Isolation Loop A XVG1611A
B-208-045	FW61B	FW Isolation Loop A XVG1611A
B-208-045	FW62A	FW Isolation Loop B XVG1611B
B-208-045	FW62B	FW Isolation Loop B XVG1611B

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-045	FW63A	FW Isolation Loop C XVG1611C
B-208-045	FW63B	FW Isolation Loop C XVG1611C
B-208-045	FW64A	Feedwater Isolation
B-208-045	FW64B	Feedwater Isolation
B-208-045	FW64C	Feedwater Isolation
B-208-045	FW65	Stm. Gen. A FW Control FCV478
B-208-045	FW66	Stm. Gen. B FW Control FCV488
B-208-045	FW67	Stm. Gen. C FW Control FCV498
B-208-045	FW68	Stm. Gen. FW. Control
B-208-045	FW70	Loop #1 Feedwater Flow Select FS/478Y
B-208-045	FW71	Loop #2 Feedwater Flow Select FS/488Y
B-208-045	FW72	Loop #3 Feedwater Flow Select FS/498Y
B-208-045	FW73	Loop #1 Steam Line Flow Select FS/478Z
B-208-045	FW74	Loop #2 Steam Line Flow Select FS/488Z
B-208-045	FW75	Loop #3 Steam Line Flow Select FS/498Z
B-208-045	FW81A	FW Bypass Control Valves IFV-3321, 3331, 3341
B-208-045	FW81B	FW Bypass Control Valves IFV-3321, 3331, 3341
B-208-045	FW88	FW Isol. Reset Switch (Low T_{avg} Coincid with Reactor Trip)
B-208-045	FW89	FW Isol. Reset Switch (Low T_{avg} Coincid with Reactor Trip) 02-01
B-208-045	FW90	Miscellaneous Relays - Feedwater System
B-208-045	FW94	S/G Level - RX Trip Setpoint Indicator LI-474A Input Signal Select SW LS-4742

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-045	FW100	FW Reverse Flush Valve XVG-1678A
B-208-045	FW101	FW Reverse Flush Valve XVG-1678B
B-208-045	FW102	FW Reverse Flush Valve XVG-1678C
B-208-045	FW103	FW Reverse Flush Valve XVG-1678A, B, C
B-208-054	HR01	Post Accident H_2 Removal Exh Sys Contain Isol Valve (XVG6056)
B-208-054	HR02	Post Accident H_2 Removal Exh Sys Contain Isol Valve (XVG6057)
B-208-054	HR21	Post Accident H_2 Analyzer A
B-208-054	HR22	Post Accident H_2 Analyzer B
B-208-054	HR23	6" Continuous Purge Line to HR Sys. Isol. Valve XVG-6067
B-208-054	HR24	6" Continuous Purge Line to HR Sys. Isol. Valve XVG-6066
B-208-054	HR25	Post Accident H_2 Removal Loop A (IRB) XVX6050A
B-208-054	HR26	Post Accident H_2 Removal Loop B (IRB) XVX6050B & 6051B
B-208-054	HR27	Post Accident H_2 Removal Loop A Suction A Suction Valve XVX6051A
B-208-054	HR28	Post Accident H_2 Removal Loop A Dom Suct. Valve XVX6051C
B-208-054	HR29	Post Accident H_2 Removal Loop A (ORB) XVX6052A & 6053A
B-208-054	HR30	Post Accident H_2 Removal Loop B (ORB) XVX6052B & 6054B
B-208-054	HR31	Containment Isolation for PT8254 XVX6054
B-208-054	HR32	HVAC Annunciator and Status Light Inputs - H_2 Removal System
B-208-057	IA50	R.B. Air Service Isolation Valve XVT2662A
B-208-057	IA51	R.B. Air Service Isolation Valve XVT2662B
B-208-057	IA52	R.B. Air Service Isolation Valve XVT2660

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-057	IA55	Instrument Air System Alarms
B-208-057	IA56	Instrument Air System Alarms
B-208-060	LD01	Leak Detection System Miscellaneous Alarms
B-208-060	LD02	Leak Detection System Miscellaneous Alarms
B-208-060	LD03	Leak Detection System Miscellaneous Alarms
B-208-060	LD04	Leak Detection System Miscellaneous Alarms
B-208-060	LD05	Leak Detection System Miscellaneous Alarms
B-208-060	LD06	Leak Detection System Miscellaneous Alarms
B-208-063	MB02	Steam Dump Interlock Select
B-208-063	MB03	Steam Dump Interlock Select
B-208-063	MB05A	Cooldown Condenser Steam Dump Valve IFV2116
B-208-063	MB06A	Cooldown Condenser Steam Dump Valve IFV2096
B-208-063	MB07A	Condenser Steam Dump Valve IFV2097
B-208-063	MB08A	Condenser Steam Dump Valve IFV2106
B-208-063	MB09A	Condenser Steam Dump Valve IFV2107
B-208-063	MB10A	Condenser Steam Dump Valve IFV2117
B-208-063	MB11A	Condenser Steam Dump Valve IFV2126
B-208-063	MB12A	Condenser Steam Dump Valve IFV2127
B-208-063	MB13A	Atmospheric Steam Dump Valve IFV2006
B-208-063	MB14A	Atmospheric Steam Dump Valve IFV2016
B-208-063	MB15A	Atmospheric Steam Dump Valve IFV2026

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-064	MD30	Intermediate Bldg. Flood Protection
B-208-064	MD31	Intermediate Bldg. Flood Protection
B-208-066	MI01	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI02	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI03	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI04	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI05	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI06	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI07	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI08	Containment Isolation A and Safety Injection ESF Monitor Lights
B-208-066	MI09	Cold Leg - Hot Leg Recirculation ESF Monitor Lights
B-208-066	MI10	Cold Leg - Hot Leg Recirculation ESF Monitor Lights
B-208-066	MI11	Cold Leg - Hot Leg Recirculation ESF Monitor Lights
B-208-066	MI12	Safety Injection ESF Monitor Lights
B-208-066	MI13	Safety Injection ESF Monitor Lights
B-208-066	MI14	Safety Injection (BOP) ESF Monitor Lights
B-208-066	MI15	Safety Injection (BOP) ESF Monitor Lights
B-208-066	MI16	Safety Injection (BOP) ESF Monitor Lights
B-206-066	MI17	Safety Injection (BOP) ESF Monitor Lights
B-206-066	MI18	Safety Injection (BOP) ESF Monitor Lights
B-206-066	MI19	Containment Isolation Phase B and Reactor Building Spray ESF Monitor Lights
B-206-066	MI20	Containment Isolation Phase B and Reactor Building Spray ESF Monitor Lights

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-206-066	MI21	Safety Inj. West. Group 1 ESF Monitor Lights (Channel XA)
B-206-066	MI22	Safety Inj. West. Group 1 & 2 ESF Monitor Lights (Channel XA)
B-206-066	MI28	Safety Inj. West. Group 1 & 3 ESF Monitor Lights (Channel XB)
B-206-066	MI29	Safety Inj. West. Group 1 & 3 ESF Monitor Lights (Channel XB)
B-206-067	MS01	Main Stm. Loop 2 to Turb. Drvn. Emerg. Feed Pump XVG2802A
B-206-067	MS02	Main Stm. Loop 3 to Turb. Drvn. Emerg. Feed Pump XVG2802B
B-206-067	MS04	Emerg. Feedwater Turbine Steam Line Drain Valve XVT2813
B-206-067	MS20A	Power Relief Valve IPV2000
B-206-067	MS20B	Power Relief Valve IPV2000
B-206-067	MS21A	Power Relief Valve IPV2010
B-206-067	MS21B	Power Relief Valve IPV2010
B-208-067	MS22A	Power Relief Valve IPV2020
B-208-067	MS22B	Power Relief Valve IPV2020
B-208-067	MS29	Main Steam Line A Drain Valve XVT2843A
B-208-067	MS30	Main Steam Line A Drain Valve XVT2877A
B-208-067	MS31	Main Steam Line B Drain Valve XVT2843B
B-208-067	MS32	Main Steam Line C Drain Valve XVT2843C
B-208-067	MS33	Main Steam Line C Drain Valve XVT2877B
B-208-067	MS34	Main Steam Lines Drain Valves (Train B)
B-208-067	MS35	Main Steam Isolation Valve XVM2801A
B-208-067	MS36	Main Steam Isolation Valve XVM2801B
B-208-067	MS37	Main Steam Isolation Valve XVM2801C

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-067	MS38A	Main Steam Isolation Valves XVM2801A, 2801B & 2801C - Train B
B-208-067	MS38B	Main Steam Isolation Valves XVM2801A, 2801B & 2801C - Train B
B-208-067	MS40	Main Steam Line A Bypass Isolation XVM2869A
B-208-067	MS41	Main Steam Line B Bypass Isolation XVM2869B
B-208-067	MS42	Main Steam Line C Bypass Isolation XVM2869C
B-208-067	MS43	Main Steam Lines Bypass Isolation (MS)
B-208-067	MS46	Main Steam to EF Turbine Isolation IFV2030
B-208-067	MS47A	Main Steam to EF Turbine Isolation IFV2030
B-208-067	MS47B	Main Steam to EF Turbine Isolation IFV2030
B-208-067	MS48	Miscellaneous Alarms - Main Steam System
B-208-067	MS51	Miscellaneous Alarms - Main Steam System
B-208-068	MU01	Reactor Makeup Water Pump A (XPP40A)
B-208-068	MU02	Reactor Makeup Water Pump B (XPP40B)
B-208-068	MU04	Reactor Coolant Makeup Control
B-208-068	MU60	Reactor Makeup Discharge Valves Auto Close
B-208-068	MU61	Reactor Makeup Flow to Non-Essential Comp. Isol. XVD1920A
B-208-068	MU62	Reactor Makeup Flow to Non-Essential Comp. Isol. XVD1920B
B-208-068	MU63	Reactor Makeup Demineralized Water Supply XVD1921
B-208-068	MU64	Reactor Makeup System Alarms
B-208-068	MU65	Reactor Makeup System Alarms
B-208-072	ND50	Reactor Building Sump Isolation XVD-6242A
B-208-072	ND51	Reactor Building Sump Isolation XVD-6242B

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-074	NI01A	Source Range Reset Block Logic
B-208-074	NI01B	Source Range Reset Block Logic
B-208-074	NI02A	Intermediate Range Block Logic
B--208-074	NI02B	Intermediate Range Block Logic
B-208-074	NI03A	Power Range Block Logic
B-208-074	NI03B	Power Range Block Logic
B-208-074	NI04	Nuclear Flux Recorder Select Pen 1
B-208-074	NI05	Nuclear Flux Recorder Select Pen 2
B-208-076	NN01	PORV (PCV-445A) N2 Supply Isolation Valve XVX-6309A
B-208-076	NN02	PORV (PCV-445B) N2 Supply Isolation Valve XVX-6309B
B-208-082	RC01	Reactor Coolant Pump “A” (XPP30A)
B-208-082	RC02	Reactor Coolant Pump “B” (XPP30B)
B-208-082	RC03	Reactor Coolant Pump “C” (XPP30C)
B-208-082	RC10	Pressurizer Relief Isol. Valve 8000A (XVG8000A)
B-208-082	RC11	Pressurizer Relief Isol. Valve 8000B (XVG8000B)
B-208-082	RC12	Pressurizer Relief Isol. Valve 8000C (XVG8000C)
B-208-082	RC14	Reactor Head Vent VV to Press. Relief Tank (XVG8095A)
B-208-082	RC15	Reactor Head Vent VV to Press. Relief Tank (XVG8095B)
B-208-082	RC16	Reactor Head Vent VV to Press. Relief Tank (XVG8096B)
B-208-082	RC17	Reactor Head Vent VV to Press. Relief Tank (XVG8096A)
B-208-082	RC50	Pressurizer Power Relief Valve PCV444B

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-082	RC51	Pressurizer Power Relief Valve PCV445A
B-208-082	RC52	Pressurizer Power Relief Valve PCV445B
B-208-082	RC53	Pressurizer Relief Tank Primary Water Valve XVD8028
B-208-082	RC57	Pressurizer Relief Tank to Gas Decay XVD8033
B-208-082	RC58	Pressurizer Relief Tank to Gas Decay XVD8047
B-208-082	RC60	Pressurizer SI Block Reset Channel A
B-208-082	RC61	Pressurizer SI Block Reset Channel B
B-208-082	RC62	Pressurizer Level Control Switch LS/459Z
B-208-082	RC63	Pressurizer Level Record Switch LS/459Y
B-208-082	RC64	DT-T_{avg} Protect Switch TS/412Z
B-208-082	RC65	HI-T_{avg} Control Select Switch TS/408
B-208-082	RC66	HI-DT Control Select Switch TS/409
B-208-082	RC67	Turbine 1st stage Pressure Select Switch PS/446Z
B-208-082	RC68	Pressurizer Pressure and Level
B-208-082	RC69	Pressurizer Pressure and Level
B-208-082	RC70	Pressurizer Pressure and Level
B-208-082	RC72	RCS Cold Overpressurization Relays
B-208-082	RC74	Pressurizer Safety Valve Indication
B-208-082	RC77	Pressurizer Steam to Pressurizer Relief Tank PCV-444C
B-208-082	RC78	Pressurizer Steam to Pressurizer Relief Tank PCV-444D
B-208-084	RH01	Residual Heat Removal Pump A (XPP31A)
B-208-084	RH02	Residual Heat Removal Pump B (XPP31B)

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-084	RH03	RHR Loop 1 Inlet Isol. Valve 8701A (XVG8701A)	
B-208-084	RH04	RHR Loop 3 Inlet Isol. Valve 8701B (XVG8701B)	
B-208-084	RH05	RHR Loop 1 Inlet Isol. Valve 8702A (XVG8702A)	02-01
B-208-084	RH06	RHR Loop 3 Inlet Isol. Valve 8702B (XVG8702B)	
B-208-084	RH07	RHR Pump Miniflow Valve FCV-602A (XVT602A)	
B-208-084	RH08	RHR Pump Miniflow Valve FVC-602B (XVT602B)	
B-208-084	RH09	RHRS to Charging Pump Valve 8706A (XVG8706A)	02-01
B-208-084	RH10	RHRS to Charging Pump Valve 8706B (XVG8706B)	
B-208-084	RH20	Miscellaneous Alarms - Residual Heat Removal	
B-208-093	SF01	Spent Fuel Pool Cooling “A” Pump XPP32A	
B-208-093	SF02	Spent Fuel Pool Cool & Trnsf Pump B XPP32B	
B-208-094	SG01A	R.B. Spray Actuation, Containment Isolation ϕA & B & Containment Ventilation Isolation	02-01
B-208-094	SG01B	R.B. Spray Actuation, Containment Isolation ϕA & B & Containment Ventilation Isolation	
B-208-094	SG02	Containment Isolation ϕA, Containment Ventilation Isolation	
B-208-094	SG03	R.B. Spray Reset, Containment Isolation ϕA & B Reset & Containment Ventilation Isolation Reset	
B-208-094	SG04	ESF Load Sequence Train A Status Lights	
B-208-094	SG05	ESF Load Sequence Train B Status Lights	
B-208-094	SG06	Miscellaneous Alarms System SG	
B-208-094	SG07	ESF Loading Sequence A Reset Control & Safety Injection Test Circuit	02-01
B-208-094	SG08	ESF Loading Sequence B Reset Control & Safety Injection Test Circuit	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-095	SI09	High Head to Cold Leg Injection Valve 8801A (XVG8801A)
B-208-095	SI10	High Head to Cold Leg Injection Valve 8801B (XVG8801B)
B-208-095	SI13	High Head to Hot Leg Injection Valve 8884 (XVG8884)
B-208-095	SI14	High Head to Cold Leg Injection Valve 8885 (XVG8885)
B-208-095	SI15	High Head to Hot Leg Injection Valve 8886 (XVG8886)
B-208-095	SI16	Accumulator Isolation Valve 8808A (XVG8808A)
B-208-095	SI17	Accumulator Isolation Valve 8808B (XVG8808B)
B-208-095	SI18	Accumulator Isolation Valve 8808C (XVG8808C)
B-208-095	SI19	RWST to RHR Pump A Isolation Valve 8809A (XVG8809A)
B-208-095	SI20	RWST to RHR Pump B Isolation Valve 8809B (XVG8809B)
B-208-095	SI21	Recirc. Sump to RHR Pump A Isol. Valve 8811A (XVG8811A)
B-208-095	SI22	Recirc. Sump to RHR Pump B Isol. Valve 8811B (XVG8811B)
B-208-095	SI23	Containment Sump Isolation Valve 8812A (XVG8812A)
B-208-095	SI24	Containment Sump Isolation Valve 8812B (XVG8812B)
B-208-095	SI25	Low Head to Hot Leg Cross Tie Valve 8887A (XVG8887A)
B-208-095	SI26	Low Head to Hot Leg Cross Tie Valve 8887B (XVG8887B)
B-208-095	SI27	Low Head to Cold Leg Cross Tie Valve 8888A (XVG8888A)
B-208-095	SI28	Low Head to Cold Leg Cross Tie Valve 8888B (XVG8888B)
B-208-095	SI29	Low Head to Hot Leg Cross Tie Valve 8889 (XVG8889)
B-208-095	SI30	Valve 8889, 8888A & 8884 Power Lockout
B-208-095	SI31	Valve 8886 & 8888B Power Lockout

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-095	SI52	Safety Injection Reset Switches
B-208-095	SI54	Safety Injection Reset Switches
B-208-095	SI58	Accumulator Fill Line 8860
B-208-095	SI59	Accumulator Holding Tank Isolation Valve 8871
B-208-095	SI72	Accumulator N_2 Supply Isolation Valve 8870
B-208-095	SI76	Accumulator Holding Tank Isolation Valve 8961
B-208-095	SI77A	Critical Function Valve Alarm
B-208-095	SI77B	Critical Function Valve Alarm
B-208-095	SI86	Latched Safety Injection Reset Switch
B-208-095	SI87	Latched Safety Injection Reset Switch
B-208-095	SI88	Redundant Position Indication for XVG-8884, 8888A, 8889
B-208-095	SI89	Redundant Position Indication for XVG-8886, 8888B
B-208-095	SI90	P11-Isolation Relay
B-208-097	SP01	Reactor Bldg. Spray Pump A (XPP38A)
B-208-097	SP02	Reactor Bldg. Spray Pump B (XPP38B)
B-208-097	SP03	RWST to R.B. Spray Pump “A” Suction Valve XVG3001A
B-208-097	SP04	RWST to R.B. Spray Pump “B” Suction Valve XVG3001B
B-208-097	SP05	NaOH Tank to R.B. Spray Pump “A” Suction Valve XVG3002A
B-208-097	SP06	NaOH Tank to R.B. Spray Pump “B” Suction Valve XVG3002B
B-208-097	SP07	Spray Header Isolation Valve “A” XVG3003A
B-208-097	SP08	Spray Header Isolation Valve “C” XVG3003B

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-097	SP09	Reactor Bldg. Sump Isolation Valve “A” XVG3004A
B-208-097	SP10	Reactor Bldg. Sump Isolation Valve “B” XVG3004B
B-208-097	SP11	Reactor Bldg. Sump Isolation Valve “A” XVG3005A
B-208-097	SP12	Reactor Bldg. Sump Isolation Valve “B” XVG3005B
B-208-097	SP20	Reactor Building Spray System Alarms
B-208-097	SP21	Reactor Building Spray System Alarms
B-208-099	SS01	Reactor Bldg. Air Sample XVA9311A
B-208-099	SS02	Reactor Bldg. Air Sample XVA9311B
B-208-099	SS03	Reactor Bldg. Air Sample XVA9312A
B-208-099	SS04	Reactor Bldg. Air Sample XVA9312B
B-208-099	SS07	Pressurizer Steam Sample XVX9356A
B-208-099	SS08	Pressurizer Liquid Sample XVX9356B
B-208-099	SS09	Pressurizer Sample XVX9357
B-208-099	SS10	Reactor Coolant Hot Leg Sample XVX9364B
B-208-099	SS11	Reactor Coolant Hot Leg (Loop 3) XVX9364C
B-208-099	SS12	Reactor Coolant Hot Leg (Loop 2) XVX9365B
B-208-099	SS13	Reactor Coolant Hot Leg (Loop 3) XVX9365C
B-208-099	SS23	Accumulator Sample XVX9387
B-208-099	SS24	Steam Generator A Blowdown Sample XVX9398A
B-208-099	SS25	Steam Generator B Blowdown Sample XVX9398B
B-208-099	SS26	Steam Generator C Blowdown Sample XVX9398C
B-208-099	SS27	EF Pump Start Signal

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-099	SS44	Post Accident Sample Part Isolation XVX9339
B-208-099	SS45	Post Accident Sample Part Isolation XVX9341
B-208-101	SW01	Service Water Pump A (XPP39A)
B-208-101	SW02	Service Water Pump B (XPP39B)
B-208-101	SW03	Service Water Pump C (XPP39C) (Channel A)
B-208-101	SW04	Service Water Pump C (XPP39C) (Channel B)
B-208-101	SW05	Service Water Screens Isolating Relays
B-208-101	SW06	Service Water Booster “A” Pump (XPP45A)
B-208-101	SW07	Service Water Booster “B” Pump (XPP45B)
B-208-101	SW08	Screen Wash “A” Pump (XPP44A)
B-208-101	SW09	Screen Wash “B” Pump (XPP44B)
B-208-101	SW10	Screen Wash “C” Pump (XPP44C)
B-208-101	SW11	(XMC1EC1X) Transfer Scheme Channel A
B-208-101	SW14A	Service Water Pump C Speed Switch (XES2003C)
B-208-101	SW14B	Service Water Pump C Speed Switch (XES2003C)
B-208-101	SW15	(XMC1EC1X) Transfer Scheme Channel B
B-208-101	SW16	Traveling Screen A (XRS0002A)
B-208-101	SW17	Traveling Screen B (XRS0002B)
B-208-101	SW18	Traveling Screen C (XRS0002C)
B-208-101	SW20	Service Water Pump A Discharge Valve (XVB3116A)
B-208-101	SW21	Service Water Pump B Discharge Valve (XVB3116B)
B-208-101	SW22	Service Water Pump C Discharge Valve (XVB3116C) (Channel C)

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-101	SW23	Recirc. Unit A Containment Isolation Valve (XVG3103A)
B-208-101	SW24	Recirc. Unit B Containment Isolation Valve (XVG3103B)
B-208-101	SW25	Reactor Building Inlet A Isolation Valve (XVB3106A)
B-208-101	SW26	Reactor Building Inlet B Isolation Valve (XVB3106B)
B-208-101	SW27	Reactor Building Outlet A Isolation Valve (XVB3107A)
B-208-101	SW28	Reactor Building Outlet B Isolation Valve (XVB3107B)
B-208-101	SW29	Reactor Building Recirc. Unit A Isolation Valve (XVG3108A)
B-208-101	SW30	Reactor Building Recirc. Unit B Isolation Valve (XVG3108B)
B-208-101	SW31	Reactor Building Recirc. Unit C Isolation Valve (XVG3108C)
B-208-101	SW32	Reactor Building Recirc. Unit D Isolation Valve (XVG3108D)
B-208-101	SW33	Reactor Building Recirc. Unit A Isolation Valve (XVG3109A)
B-208-101	SW34	Reactor Building Recirc. Unit B Isolation Valve (XVG3109B)
B-208-101	SW35	Reactor Building Recirc. Unit C Isolation Valve (XVG3109C)
B-208-101	SW36	Reactor Building Recirc. Unit D Isolation Valve (XVG3109D)
B-208-101	SW37	Building Service Inlet A Isolation Valve (XVB3110A)
B-208-101	SW38	Building Service Inlet B Isolation Valve (XVB3110B)
B-208-101	SW39	Building Service Outlet A Isolation Valve (XVG3111A)
B-208-101	SW40	Building Service Outlet B Isolation Valve (XVG3111B)
B-208-101	SW41	Building Service Outlet A Isolation Valve (XVK3112A)
B-208-101	SW42	Building Service Outlet B Isolation Valve (XVK3112B)

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TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-101	SW43	Screen Wash Pump Discharge Valve (XVG3113A)
B-208-101	SW44	Screen Wash Pump Discharge Valve (XVG3113B)
B-208-101	SW45	Screen Wash Pump Discharge Valve (XVG3113C)
B-208-101	SW50	Emergency Water Supply to DG “A” Cooler XVT3105A
B-208-101	SW51	Emergency Water Supply to DG “B” Cooler XVT3105B
B-208-101	SW52	DG Cooler A Isolation XVB3104A & 3121A
B-208-101	SW53	DG Cooler B Isolation XVB3104B & 3121B
B-208-101	SW54	SW Pumps A & C Cross Connect XVB3118A & 3118C
B-208-101	SW55	SW Pumps B & C Cross Connect XVB3118B & 3118D
B-208-101	SW57	CC Heat Exchanger A Isolation XVB3122A & 3123A
B-208-101	SW58	CC Heat Exchanger B Isolation XVB3122B & 3123B
B-208-101	SW59	HVAC Chiller Cond. A Isolation XVB3126A & 3127A
B-208-101	SW60	HVAC Chiller Cond. B Isolation XVB3126B & 3127B
B-208-101	SW65	SW Booster Pump A Suction XVB3134A
B-208-101	SW66	SW Booster Pump B Suction XVB3134B
B-208-101	SW70	Service Water System Alarms
B-208-101	SW71	Service Water System Alarms
B-208-101	SW72	Service Water System Alarms
B-208-101	SW73	Service Water System Alarms
B-208-101	SW74	Service Water System Alarms
B-208-101	SW75	Service Water System Alarms

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-101	SW78	DRPI Cooling Unit Coil Isolation Valve XVT-3164	
B-208-101	SW79	DRPI Cooling Unit Coil Isolation Valve XVT-3165 & 3169	
B-208-101	SW86	SW Inlet to HVAC Chiller “A” (XVB3126A)	
B-208-101	SW87	SW Inlet to HVAC Chiller “B” (XVB3126B)	RN 11-018
B-208-101	SW88	“A” Train SW Inlet to HVAC Chiller “C” (XVB3128A)	
B-208-101	SW89	“B” Train SW Inlet to HVAC Chiller “C” (XVB3128C)	
B-208-103	TB02	Redundant Turbine Trips - Train A	
B-208-103	TB03	Redundant Turbine Trips - Train B	
B-208-103	TB04	Turbine Tripped Relaying	
B-208-103	TB07	Turbine Stop Valves and Control Valves Limit Switches	
B-208-108	VL01	Aux. Bldg. MCC - Switchgear AHU Fan (XFN132)	
B-208-108	VL02	Aux. Bldg. MCC - Switchgear AHU Fan (XFN133)	
B-208-108	VL05	Charging/SI Pump Room 1 Cooling Unit Fan A (XFN46A)	
B-208-108	VL06	Charging/SI Pump Room 3 Cooling Unit Fan B (XFN46B)	02-01
B-208-108	VL07	Charging/SI Pump Room 2 Cooling Unit Fan (XFN47)	
B-208-108	VL08	RHR/Spray Pump Room 1 Cooling Unit Fan A (XFN49A)	
B-208-108	VL09	RHR/Spray Pump Room 2 Cooling Unit Fan B (XFN49B)	
B-208-108	VL10	Ventilation System Annunciator & Alarm Inputs - Aux Bldg	
B-208-108	VL11	Ventilation System Annunciator & Alarm Inputs - Aux Bldg	
B-208-108	VL12	Ventilation System Annunciator & Alarm Inputs - Aux Bldg	
B-208-108	VL18	Speed Switch Room Air Handling Fan (XFN-106A-VL)	
B-208-108	VL19	Speed Switch Room Air Handling Fan (XFN-106B-VL)	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE	
B-208-108	VL22	ESF Switchgear Room 1DA Cooling Unit Fan (XFN50)	
B-208-108	VL24	ESF Switchgear Room 1DB Cooling Unit Fan (XFN76)	
B-208-108	VL26	Service Water Booster Pump Area Cooling Unit Fan A (XFN81A)	
B-208-108	VL27	Service Water Booster Pump Area Cooling Unit Fan B (XFN81B)	
B-208-108	VL30	Emergency Feedwater Pump Area Cooling Unit Fan A (XFN83A)	
B-208-108	VL31	Emergency Feedwater Pump Area Cooling Unit Fan B (XFN83B)	
B-208-108	VL34	Ventilation System Annunciator & Alarm Inputs - Int. Bldg	
B-208-108	VL35	Ventilation System Annunciator & Alarm Inputs - Int. Bldg	
B-208-108	VL36	Ventilation System Annunciator & Alarm Inputs - Int. Bldg	
B-208-108	VL37	Ventilation System Annunciator & Alarm Inputs - Int. Bldg	
B-208-109	VU01	HVAC Mechanical Water Chiller XHX0001A	
B-208-109	VU02	HVAC Mechanical Water Chiller XHX0001B	02-01
B-208-109	VU03	HVAC Mechanical Water Chiller XHX0001C (“A” Channel)	
B-208-109	VU04	HVAC Mechanical Water Chiller XHX0001C (“B” Channel)	
B-208-109	VU05	HVAC System Chilled Water Pump XPP0048A	
B-208-109	VU06	HVAC System Chilled Water Pump XPP0048B	
B-208-109	VU07	HVAC System Chilled Water Pump XPP0048C (“A” Channel)	
B-208-109	VU08	HVAC System Chilled Water Pump XPP0048C (“B” Channel)	
B-208-109	VU09	HVAC System Water Chiller “A” Control Scheme	
B-208-109	VU10	HVAC System Water Chiller “B” Control Scheme	
B-208-109	VU11	HVAC System Water Chiller “C” Control Scheme	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SAFETY RELATED ELEMENTARIES OF SAFETY RELATED SYSTEMS

DRAWING NO.	SYSTEM SHEET	TITLE
B-208-109	VU16	Comp Cooling Pump Motor Isol Valve XVG6516
B-208-109	VU17	Comp Cooling Pump Motor Isol Valve XVG6519
B-208-109	VU18	Comp Cooling Pump Motor Isol Valve XVG6517
B-208-109	VU19	Comp Cooling Pump Motor Isol Valve XVG6518
B-208-109	VU25	Chilled Water Nonessential Load Isolation Valve XVT-6412A
B-208-109	VU26	Chilled Water Nonessential Load Isolation Valve XVT-6412B
B-208-109	VU27	Chilled Water Nonessential Load Isolation Valve XVT-6384A
B-208-109	VU28	Chilled Water Nonessential Load Isolation Valve XVT-6384B
B-208-109	VU29	Chilled Water Nonessential Load Isolation Valve XVT-6385A
B-208-109	VU30	Chilled Water Nonessential Load Isolation Valve XVT-6385B
B-208-109	VU31	Chilled Water Nonessential Load Isolation Valve XVT-6490A
B-208-109	VU32	Chilled Water Nonessential Load Isolation Valve XVT-6490B
B-208-109	VU34	HVAC Control Graphic Display Annunciator Alarm Inputs
B-208-109	VU35	HVAC Control Graphic Display Annunciator Alarm Inputs
B-208-109	VU36	HVAC Control Graphic Display Annunciator Alarm Inputs
B-208-120	WP50	Reactor Coolant Drain Tank Vent XVD-7126
B-208-120	WP51	Reactor Coolant Drain Tank Pump XVD-7136
B-208-120	WP52	Reactor Coolant Drain Tank Vent XVD-7150
B-208-120	WP53	Reactor Coolant Drain Tank Pump XVD-7170

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
ONE-LINE AND RELAY DIAGRAMS

DRAWING NO.	TITLE	
E-206-001	Notes, Legend & References	
E-206-011	Balance of Plant Power System	
E-206-012	Engineered Safety Features Power System	
E-206-021	7200 V Swgr. Busses 1A, 1B, 1C & 1DX	
E-206-022	7200 V Swgr. Busses 1DA, 1DB, 1EA & 1EB	
E-206-031	480/277 V Swgr. Busses 1A1, 1A3, 1B1, 1B3, 1C1 & 1C3	
E-206-032	480/277 V Swgr. Busses 1A2, 1A4, 1B2, 1B4, 1C2 & 1C4	02-01
E-206-033	480/277 V Switchgear 1A5	
E-206-034	480/277 V Swgr. Busses 1DA1, 1DA2, 1DB2, 1EA1 & 1EB1	
E-206-042 Sheet 1	240/120 V Miscellaneous AC Distribution	
E-206-042 Sheet 2	240/120 V Miscellaneous AC Distribution	
E-206-043	480/277 V Miscellaneous AC Distribution	
E-206-047	480 V Engineered Safeguard MCC Power Feeds	
E-206-061 Sheet 1	Balance of Plant - Vital A.C. - D.C. System	
E-206-061 Sheet 2	Balance of Plant - Vital A.C. - D.C. System	
E-206-062 Sheet 1	Engineered Safety Features Vital A.C. - D.C. System	
E-206-062 Sheet 2	Engineered Safety Features Vital A.C. - D.C. System	
E-206-062 Sheet 3	Engineered Safety Features Vital A.C. - D.C. System	
E-206-071	480 V Switchgear - Pressurizer Heaters	

TABLE 1.7-1 (Continued)

VIRGIL C. SUMMER NUCLEAR STATION
WIRING SCHEMATIC PACKAGE
SCHEMATIC DIAGRAMS

DRAWING NO.	GAI MFRS. DWG. NO.	TITLE
D-2544-1013 Sheet 1	IMS-28-096-1-2	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 2	IMS-28-096-2-3	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 3	IMS-28-096-3-4	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 4	IMS-28-096-4-3	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 5	IMS-28-096-5-3	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 6	IMS-28-096-6-5	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 7	IMS-28-096-7-2	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 8	IMS-28-096-8-6	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 9	IMS-28-096-9-2	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 10	IMS-28-096-10-4	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 11	IMS-28-096-11-4	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 12	IMS-28-096-12-4	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 13	IMS-28-096-13-4	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 14	IMS-28-096-14-4	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 15	IMS-28-096-15-6	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 16	IMS-28-096-16-1	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 17	IMS-28-096-17-2	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 18	IMS-28-096-18-2	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 19	IMS-28-096-19-2	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 20	IMS-28-096-20-4	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 21	IMS-28-096-21-3	(AII/VLD) ESF Loading Sequence Electrical Schematics
D-2544-1013 Sheet 22	IMS-28-096-22-2	(AII/VLD) ESF Loading Sequence Electrical Schematics

NOTE 1.8

Section 1.8 is being retained for historical purposes only.

99-01

1.8 TMI ACTION PLAN REQUIREMENTS

A cross reference between various TMI Action Plan Requirements (NUREG-0737, November 1980) addressed in the FSAR and the appropriate FSAR sections is presented by Table 1.8-1.

TABLE 1.8-1

CROSS REFERENCE

TMI ACTION PLAN REQUIREMENTS TO FSAR SECTIONS

ACTION PLAN REQUIREMENT	FSAR SECTION/SCE&G LETTERS	
I.A.1.1	13.1.2.1, 13.1.2.2.2, 13.1.3.1.12, 13.2.1, SCE&G letter to NRC dated 1/2/81	99-01
I.A.1.2	13.1.2.2, 13.5.1.3.1	
I.A.1.3	13.1.2.3, 13.5.1.3, SCE&G Letter to NRC dated 1/22/82	
I.A.2.1	13.2.1, SCE&G letters to NRC dated 10/28/80, 10/31/80, 5/22/84, 7/19/84, 11/28/84	99-01
I.A.2.3	13.2, SCE&G letter to NRC dated 10/28/80	99-01
I.A.3.1	13.2.1, 13.2.2, SCE&G letter to NRC dated 10/28/80	99-01
I.B.1.2	13.1.1.4, SCE&G letter to NRC dated 1/2/81	
I.C.1	6.3.3.3.1, 13.5.2, SCE&G letters to NRC dated 11/14/80, 12/2/80, 3/17/82, and Westinghouse Owners Group Letter OG-47 dated 12/15/80	
I.C.2	13.5.1.3	
I.C.3	13.5.1.3.1, Tech Spec (6.1.2)	
I.C.4	13.5.1.3	
I.C.5	13.5.1.13	
I.C.6	13.5.1.6, SCE&G letter to NRC dated 12/11/80	
I.C.7	13.5.1.3.2, SCE&G letter to NRC dated 12/2/80 and 12/22/80	
I.C.8	13.5.1.3.3, SCE&G letter to NRC dated 11/14/80	
I.D.1	1.2.3.1, SCE&G letters to NRC dated 11/12/80, 1/15/81, 11/25/81, 2/23/82, 3/26/82, 6/11/82, 4/15/83, 7/21/83, 10/28/83, 4/4/84, 4/15/85	
I.D.2	7.7.3, SCE&G letters to NRC dated 3/31/83, 4/15/83, 12/28/83, 4/18/85, 12/23/85	
I.G.1	14.1.4.4, SCE&G letters to NRC dated 10/31/80, 12/2/80, 12/22/80, 3/31/82 and 7/29/82	
II.B.1	5.5.15, SCE&G letters to NRC dated 2/19/81, 12/30/81, 1/18/84	
II.B.2	12.1.2.3, App. 12A, SCE&G letters to NRC dated 8/27/80 and 11/21/80	
II.B.3	9.3.2, App. 12A, SCE&G letter to NRC dated 3/30/82	
II.B.4	13.2.1, 13.2.2, SCE&G letter to NRC dated 10/28/80	99-01

TABLE 1.8-1

CROSS REFERENCE

TMI ACTION PLAN REQUIREMENTS TO FSAR SECTIONS

ACTION PLAN REQUIREMENT	FSAR SECTION/SCE&G LETTERS	
II.D.1	5.5.13.4, SCE&G letters to NRC dated 3/25/81, 7/29/81, 3/26/82, 4/1/82, 6/29/82 and 7/30/82	
II.D.3	1.2.3.1, 1.7, 5.5.10.2.2.4, 5.6, 7.7.4, SCE&G letters to NRC dated 1/13/81, 2/26/82, 3/12/82 and 8/26/82	
II.E.1.1	SCE&G letters to NRC dated 8/15/80, 11/5/80 and 12/2/80	
II.E.1.2	7.3.1.1.1, 7.3.2.2, 7.3.2.3, 7.5.1, 10.4.9.1, 10.4.9.2, 10.4.9.3, 10.4.9.5.3	
II.E.3.1	8.3.1.1.1.a, 8.3.1.1.3, SCE&G letter to NRC dated 2/23/82	99-01
II.E.4.1	6.2.5.2.1	
II.E.4.2	6.2.4.3; 9.4.8.2.2, Items 8.l and 8.m; 9.4.8.2.3, Items 8.h and 8.i	99-01
II.F.1	6.2.5.1.3, 6.2.5.2.3, 6.2.5.3.3, 6.2.5.4.3, 6.2.5.5.3, 6.2.5.5.4, 7.7.3.1, 11.4, 11.4.2, Figure 11.4-2, 12.2.5, Figure 12.2-2, 12.3.2.2, SCE&G letters to NRC dated 8/28/80, 12/22/80, 6/30/82, 1/18/84, 3/22/84	99-01
II.F.2	1.2.3.1, 5.6, SCE&G letters to NRC dated 12/4/80, 12/15/80, 12/30/80, 2/19/81, 6/8/81, 7/30/81, 1/18/82, 3/16/82, 4/30/82, 7/20/82, 3/8/83, 3/10/83, 4/22/83, 8/26/83, 2/17/84, 4/13/84, 4/16/84, 4/30/84 and 6/6/84	99-01
II.G.1	7.4.1.2.1, 8.3.1.1.3	
II.K.1	7.3.1.1, 13.5.1.6, and Technical Specifications	99-01
II.K.2	SCE&G letters to NRC dated 12/11/80, 1/6/81, and 12/31/81	
II.K.3	5.2.2.3; 7.2.1.1.2, Item 6; Figure 7.7-4; Technical Specifications, SCE&G letters to NRC dated 9/9/80, 1/6/81, 1/12/81, 2/19/81, 3/10/81, 3/23/81, 12/31/81, and 4/22/83, 5.5.1.3.13, 13.5.1.3.4, 13.5.1.14, 15.3.1.2.4, Westinghouse letters to NRC (NS-EPR-2581) dated 3/26/82 and (NS-EPR-2581) dated 6/28/82	99-01
III.A.1.1	Radiation Emergency Plan, SCE&G letters to NRC dated 5/12/81 and 6/16/81	
III.A.1.2	1.2.3.1, 6.4, 7.7.3, Appendix 12A, SCE&G letters to NRC dated 3/16/82, 7/23/82, 8/6/82, 11/24/82, 12/3/82, 3/31/83, 4/15/83, 5/16/83, 9/14/84	99-01
III.A.2	2.3.3.2, Radiation Emergency Plan, SCE&G letters to NRC dated 5/12/81, 6/16/81, 3/31/83	
III.D.1.1	6.3.2.11.2, Technical Specifications, SCE&G letter to NRC dated 2/23/82	
III.D.3.3	6.2.5.1.3, 6.2.5.2.3, 6.2.5.3.3, 6.2.5.4.3, 6.2.5.5.4, 12.1.4.2, 12.3.2.2.4	
III.D.3.4	2.2.1, 2.2.2, 2.2.3, 6.4, 15.4, SCE&G letters to NRC dated 11/15/80, 12/15/80, 1/18/84	