1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

1.1.1 GENERAL INFORMATION

This Final Safety Analysis Report (FSAR) is submitted in support of the Carolina Power & Light Company's (CP&L) application for a Class 103 facility operating license for the Shearon Harris Nuclear Power Plant (SHNPP). This FSAR has been organized in accordance with the guidelines contained in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Revision 3 dated November 1978), and the regulations of the NRC set forth in 10 CFR 50.

On July 2, 2012, Duke Energy Corporation completed closure of a corporate merger with Progress Energy (i.e., Carolina Power & Light Company). Subsequently, on October 21, 2013, the NRC approved changing the licensee name from Carolina Power & Light Company to Duke Energy Progress, Inc. Until all sections of the FSAR are updated to incorporate this name change, where Carolina Power & Light Company (CP&L) or Progress Energy is used, except in an historical context, they shall mean Duke Energy Progress, Inc.

Lists of acronyms, abbreviations and names of major buildings and structures used throughout this FSAR are given in Tables 1.1.1-1, 1.1.1-2, and 1.1.1-3, respectively. Figures 1.1.1-1 and 1.1.1-2 provide flow diagram symbols while Figure 1.1.1-1a provides piping and instrumentation symbols used on engineering drawings throughout the FSAR.

Duke Energy Progress, Inc. owns the plant. Duke Energy Progress, Inc. has the overall responsibility to ensure that it is designed, constructed, and operated without undue risk to the health and safety of the public. Ebasco Services, Incorporated is the architect/engineer responsible for the design, engineering, and equipment and material procurement for SHNPP. This includes all plant structures, systems, and components except for those provided by Westinghouse Electric Corporation, the Nuclear Steam Supply System (NSSS) Supplier. Daniel Construction Company, Inc., as the constructor, performed the major part of the plant construction. Selected portions of the work, however, were performed by other contractors under direct supervision of CP&L.

1.1.2 STATION LOCATION

The SHNPP site is located in the extreme southwest corner of Wake County, North Carolina, and the southeast corner of Chatham County, North Carolina. The city of Raleigh, North Carolina, is approximately 16 miles northeast and the city of Sanford, North Carolina, is about 15 miles southwest.

1.1.3 NUCLEAR STEAM SUPPLIER

The Nuclear Steam Supply System (NSSS) for the Unit is a pressurized water reactor (PWR) consisting of three closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to the "hot" leg of one of the loops. The NSSS, along with the design and fabrication of the initial fuel core, is supplied by Westinghouse Electric Corporation.

1.1.4 CONTAINMENT

The Containment is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome and a flat base with a recess beneath the reactor vessel. The Containment is designed by Ebasco Services Incorporated, architect/engineer for SHNPP.

1.1.5 CORE THERMAL POWER

The Unit is licensed for a core thermal power output of 2948 megawatts thermal (Mwt). The total unit thermal output is approximately 2960.4 Mwt, which includes 12.4 Mwt from the reactor coolant pumps. The thermal output corresponds to an electrical output of approximately 930 megawatts electric (Mwe) net or 985 Mwe gross. All safety systems, including containment and engineered safety features, have been analyzed for operation at a core thermal power of up to 2958 MWt.

The originally licensed reactor thermal power was 2775 Mwt. A licensing request was granted in the year 2001 to increase reactor power to 2900 Mwt. This was accomplished in conjunction with replacing the steam generators in Refueling Outage number 10 (RFO-10), in the fall/winter of 2001. The effects of these major changes to the plant are seen throughout the FSAR and are sometimes noted with SGR for Steam Generator Replacement Project and PUR for Power Uprate Project.

In the spring of 2012, (RFO-17) implementation of a Measurement Uncertainty Recapture - Power Uprate (MUR-PU) was implemented by a licensing request allowing changes to the 10 CFR 50, Appendix K power measurement uncertainty requirements. This change reduced the required power uncertainty margin from 2% (measuring feedwater flow with the venturis) to the uncertainty associated with measuring feedwater flow using the Caldon Leading Edge Flow Meters (LEFMs) installed during RFO-16, in the fall of 2010. The more accurate LEFMs reduced the core power measurement uncertainty from 2.0% to 0.34%, allowing an increase in reactor core power of 1.66% from 2900 MWt to 2948 MWt. The effects of this change to the plant are seen throughout the FSAR and are sometimes noted with MUR-PU. Unless otherwise noted, the values and other information contained in the FSAR are based on the plant configuration after MUR-PU. The MUR-PU is defined as an increase in core thermal power from 2900 MWt (post SGR/PUR) to a (current) core power of 2948 MWt. These are nominal values without uncertainties.

1.1.6 SCHEDULE

The construction schedule for SHNPP is based on a commercial operation date in the fourth quarter of 1986. This schedule requires that an operating license be issued in time for fuel loading by June 1986.

1.2 GENERAL PLANT DESCRIPTION

The Shearon Harris Nuclear Power Plant (SHNPP) is designed to function as an electric generating station, and generate electric power utilizing a pressurized water reactor (PWR) and a closed regenerative cycle steam turbine-generator. The inherent design of the pressurized water, closed-cycle reactor minimizes the quantities of fission products released to the atmosphere. Four barriers exist between the fission product accumulation and the environment. These are the uranium dioxide fuel matrix, the fuel cladding, the reactor vessel and reactor

coolant loops, and the Containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through fuel cladding defects would be contained within the reactor pressure vessel, reactor coolant loops and associated auxiliary systems. Breach of these systems or equipment would release the fission products to the Containment where they would be contained. The Containment is designed to adequately contain these fission products under the most severe accident conditions, as analyzed in Chapter 15.

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a loss-of-coolant-accident (LOCA). These safety features include an Emergency Core Cooling System (ECCS). This system automatically delivers borated water to the reactor vessel for cooling the core under high and low reactor coolant pressure conditions. The ECCS also serves to effectively insert negative reactivity into the core in the form of borated water during Unit cooldown following a steam line break or an accidental steam release. Additional safety features include the Containment Spray System (CSS) which serves to remove thermal energy from the Containment in the event of a LOCA. The concentration of post LOCA airborne iodine fission products within the Containment will also be reduced by the CSS.

1.2.1 PRINCIPAL SITE CHARACTERISTICS

1.2.1.1 Location and Population

The site occupies approximately 10,723 acres of land in southwest Wake County and southeast Chatham County, North Carolina. The environment is rural and primarily devoted to farming and dairying. Local industrial activity is centered in an area west-southwest from the plant. Another major center of industrial and research activity is located to the north-northwest. The exclusion area is shown on Figure 2.1.2-1. The shortest distance to the exclusion boundary is 6640 ft. in the northwest direction; the longest distance is 7200 ft. in the south direction. The area within the exclusion boundary is directly controlled by DEP. The three mile low population zone was populated by 534 persons as of 1970. The nearest population center as defined in 10 CFR 100 is Cary at a distance of 10 miles to the city's nearest boundary. The fifty-mile radius population was approximately 1,300,000 people (1980 population). Population projections for the period of operation of the plant are located in Section 2.1.3. See Section 2.1 for a more detailed discussion of geography and demography.

1.2.1.2 Hydrology

The plant site is located at the confluence of Buckhorn and Whiteoak Creeks, just north of the Cape Fear River. The power block area is located between Tom Jack and Thomas Creeks. Figure 2.1.1-1 shows a plan of the site development.

The principal water source for the plant is the Main Reservoir which is formed by an impoundment of Buckhorn Creek just below its confluence with Whiteoak Creek. The project design also includes an adjoining and independent Auxiliary Reservoir for emergency cooling purposes. See Section 2.4 for a more detailed discussion of hydrology.

1.2.1.3 Meteorology

The meteorological conditions of the plant site are those typical of the transition zone delineating the Coastal Plain and Piedmont climatological classifications. Climatology of North Carolina is largely dependent upon elevation above sea level and distance from the Atlantic Ocean. The inland location of the plant site (115 miles from the Atlantic Ocean) modifies the effects of coastal storms, both tropical and extratropical, so that they are reduced in intensities to levels which are generally no greater than those produced by regional heavy thunderstorms. The severity of continental air mass systems approaching from the northwest is modified by the Appalachian Mountain range which acts as a protective barrier. The area is not on a usual path of either continental or coastal cyclonic storm centers, although frontal passages are frequent. The major weather influence in the region is the predominance of the subtropical belt of high pressure.

Influence of the Atlantic Ocean is reflected in generally high moisture content of the air masses usually over the region. The thermal effects of the ocean coupled with the barrier formed by the mountain range, result in a high frequency of occurrence of northeasterly winds in the fall and of southwesterly winds in the spring. The predominant annual wind direction at the plant site occurs from the southwesterly sectors; however, the bimodal wind direction characteristics of the region are evident from the onsite data. See Section 2.3 for a more detailed discussion of meteorology.

1.2.1.4 Geology and Seismology

The region surrounding the site is generally characterized by a gently rolling topography resulting from extensive weathering and erosion of the underlying bedrock. The site is located in the southeastern part of the Durham Basin, which is in the northern part of the Deep River Triassic Basin. Sediments that underlie much of the southeastern portion of the Durham Basin were placed as alluvial fans and stream channels and flood plain deposits. Below an occasional thin layer of alluvial sand and/or clay, there are from 0 to 15 ft. of residual soil. The depth of weathering below this to sound rock generally varies from about 0 to 15 ft. depending on the type of underlying rock. The foundations have been placed on sound rock.

A small fault was discovered during excavation for the Waste Processing Building. The studies performed showed that this fault is not a capable fault, as documented in the Shearon Harris Fault Investigation Report submitted to the NRC in 1975.

The nearest known fault outside the site is one lying just west of Merry Oaks about three miles to the southwest of the site.

Test borings showed nothing that would indicate the development of faults, joints, slickensides or other structural weakness since the late Triassic and early Jurassic time.

The site ground accelerations for the Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) are 0.150 and 0.075g, respectively. See Section 2.5 for a more detailed discussion of geology and seismology.

1.2.1.5 Nearby Industry and Commerce

Industrial activity in the region surrounding the plant site is not intensive and is concentrated to the north-northwest. See Section 2.2 for more information.

1.2.2 CONCISE PLANT DESCRIPTION

1.2.2.1 Principal Structures

Major plant structures include the Containment Building; Reactor Auxiliary Building, which contains the Control Room; Turbine Building; Waste Processing Building; Diesel Generator Building; a Service Building; Fuel Handling Building; Tank Building; and Cooling Tower. Figure 1.2.2-1 indicates the site plan for the Shearon Harris Nuclear Power Plant. Figures 1.2.2-3 through 1.2.2-87 indicate the general arrangements of plant structures and major equipment.

The seismic criteria used to design the structures and equipment in the plant are described in Sections 3.7, 3.8, and 3.10. The maximum horizontal ground acceleration for the operating condition is 0.075g (Operating Basis Earthquake). However, the design ensures that no undue risk to public health and safety results from a horizontal ground acceleration of 0.15g (Safe Shutdown Earthquake).

1.2.2.2 Nuclear Steam Supply System

The Nuclear Steam Supply System consists of a pressurized water reactor, a Reactor Coolant System (RCS), and associated auxiliary systems. The Reactor Coolant System is arranged as three closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and steam generator. An electrically heated pressurizer is connected to the hot leg of one reactor coolant loop.

The reactor core is composed of uranium dioxide pellets enclosed in pressurized Zirconium alloy 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 or Hafnium absorber rods and Zircaloy guide tubes located within the fuel assembly.

Fuel rod cladding is designed to maintain cladding integrity throughout fuel life. Fission gas released within the rods and other factors affecting design life are considered for the maximum expected exposure. The reactor and control systems are designed so that any xenon transients will be adequately damped. Assuming the chemical shim requirements are met (i.e., proper concentration of boric acid in the coolant), the rod cluster control assemblies (RCCA) are capable of holding the core subcritical at hot zero power condition with margin following a trip, even with the most reactive RCCA stuck in the fully withdrawn position.

The reactor, in conjunction with its protective systems, is designed to safely accommodate the anticipated operational occurrences. See Chapter 4 for further information. The reactor vessel and reactor internals contain and support the fuel and control rods. The reactor vessel is cylindrical with a hemispherical head and is clad internally with stainless steel. The pressurizer is a vertical cylindrical pressure vessel with a hemispherical head and is equipped with electrical heaters and spray nozzles for system pressure control.

The steam generators are vertical U-Tube type heat exchangers utilizing Inconel tubes. Internal moisture separating equipment reduces the moisture carryover of the steam at the outlet nozzle to 0.10 weight percent or less under maximum design load and ramp conditions defined in Section 5.4.2.

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

Auxiliary systems are provided to charge the RCS and to add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, remove residual heat when the reactor is shut down, provide for emergency safety injection, vent and drain the RCS, and provide sampling of reactor coolant water.

The RCS is designed and constructed to maintain its integrity throughout the plant life. Appropriate means for testing and inspection are provided. The reactor for the Shearon Harris Nuclear Power Plant is designed with provisions to permit plutonium recycle. The provisions included in the design consist of a control rod pattern which is based on current technology, and additional hardware such as guide tubes, vessel head adapters, control rod drive mechanisms, control rods, control rod position indicators, switches, required power supplies, and control room indications. See Chapter 5 for further information.

1.2.2.3 Engineered Safety Features

The Engineered Safety Features (ESF) provided have sufficient redundancy of components and power sources such that under the conditions of a loss-of-coolant-accident (LOCA) they can maintain the integrity of the Containment and ensure that the limits of 10 CFR 50.67 are not exceeded even when operating with partial effectiveness. The main engineered safety features are the Emergency Core Cooling System (ECCS), the Containment Building, the Containment Spray System (CSS), the Containment Cooling System (CCS), Combustible Gas Control System and Control Room Habitability System. The functions they serve are summarized below, and other ESF systems are listed:

a) The ECCS injects borated water into the RCS. The ECCS limits damage to the core and limits the energy and fission products released into the Containment following a loss-of-coolant-accident. All components necessary for the proper operation of the engineered safety features are operable from the Control Room.

b) The steel-lined reinforced concrete Containment Building, with its associated Containment Isolation System (CIS), provides a reliable barrier against the escape of fission products under various environmental conditions following a LOCA. Containment isolation is initiated by several process variables as discussed in Sections 6.2.4 and 7.3.

c) The Containment Spray System provides a spray of cool water containing sodium hydroxide to reduce containment pressure and remove the volatile iodine from the containment atmosphere following a LOCA.

d) The Containment Cooling System consists of four cooling units which provide a redundant means of reducing the containment pressure following a LOCA.

e) The Combustible Gas Control System maintains a safe post-LOCA hydrogen concentration within the Containment.

f) The habitability systems insure control room habitability following a LOCA.

See Chapter 6 for further information.

1.2.2.4 Instrumentation and Control Systems

Instrumentation and Control Systems provide the reactor operators with required information and control capability to operate in a safe and efficient manner. Where safety functions are involved, logic circuitry and actuators are provided to initiate equipment actions without operator assistance.

1.2.2.4.1 Controls

The Reactor Control System is used for startup and shutdown of the reactor and for adjustment of the reactor power in response to turbine load demand. The Nuclear Steam Supply System (NSSS) is capable of accommodating 10 percent of full power step changes in plant load and 5 percent of full power per minute ramp changes over the range from 15 percent up to 100 percent full power, without reactor trip.

Overall reactivity control is achieved by the 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 change is accomplished by moving the RCCA.

The reactor automatic control system is designed to maintain a programmed average temperature in the reactor coolant during steady state operation and to ensure that plant conditions do not reach trip settings as the result of a transient caused by a design load change.

The function of the Reactor Control System is to provide automatic control of the RCCA during power operation of the reactor. The system uses input signals including neutron flux, coolant temperature, and turbine load. The Chemical and Volume Control System (Chapter 9) supplements the Reactor Control System by the addition and removal of varying amounts of boric acid solution.

When the reactor is critical, the best indication of reactivity status in the core is the position of the rod control bank in relation to power and average coolant temperature. The direct relationship between control rod position and power is the relationship which establishes the lower insertion limit calculated by the rod insertion limit monitor. There are two alarm setpoints to alert the operator to take corrective action in the event a rod control bank approaches or reaches its lower insertion limit.

Any unexpected change in the position of the control banks under automatic control, or a change in reactor coolant temperature under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. In addition, periodic samples are taken for determination of the reactor coolant boron concentration. The variation in concentration during core life provides a further check on the reactivity status of the reactor

including core depletion. The provisions for monitoring the primary coolant boron concentration are discussed in Chapter 9.

The Reactor Control System is designed to enable the reactor to follow load changes automatically when the output is above approximately 15 percent of rated power. Control rod positioning may be performed automatically when plant output is above this value, and manually at any time.

Following a reactor and turbine trip, residual heat stored in the reactor coolant is removed without actuating the steam generator safety valves by venting steam to atmosphere or bypassing steam to the condenser with power operated valves and addition of main or auxiliary feedwater. Reactor Coolant System temperature is reduced in this way to the no load, hot standby, or cold shutdown condition.

1.2.2.4.2 Instrumentation

The primary function of nuclear instrumentation is to safeguard the reactor by monitoring the neutron flux and generating appropriate trips and alarms for various phases of reactor operating and shutdown conditions. It also provides a secondary control function by indicating reactor status during start-up and power operation. The Nuclear Instrumentation System (NIS) uses information from three separate types of instrumentation channels to provide three discrete protection levels. Each range of instrumentation ("source", "intermediate", and "power") provides the overpower reactor trip protection required during operation in that range. The overlap of instrument ranges provides reliable continuous protection beginning with source level through the intermediate and low power level. As the reactor power increases, the overpower protection level is increased according to plant procedures after satisfactory higher range instrumentation operation is obtained. Automatic reset to more restrictive trip protection is provided when reducing power.

Various types of neutron detectors, with appropriate solid-state electronic circuitry, are used to monitor the neutron flux from a completely shutdown condition to 120 percent of full power. The power range channels are capable of recording overpower excursions up to 200 percent of full power. The neutron flux covers a wide range between these extremes. Therefore, monitoring with several ranges of instrumentation is necessary. The lowest range ("source" range) covers six decades of neutron flux. The next range ("intermediate" range) covers eight decades. Detectors and instrumentation are chosen to provide overlap between the higher portion of the source range and the lower portion of the intermediate range. The highest range of instrumentation ("power" range) covers approximately 2.2 decades of the total instrumentation range. This is a linear range that overlaps with the higher portion of the intermediate range.

The system described above provides control room indication and recording of signals proportional to reactor neutron flux during core loading, shutdown, start-up, and power operation, as well as during subsequent refueling. Start-up rate indication for the source and intermediate range channels is provided at the main control board. Reactor trip and rod stop control and alarm signals are transmitted to the Reactor Control and Reactor Protection Systems (RPS) automatic plant control.

The engineered safety features instrumentation measures temperatures, pressures, flows and levels in the RCS, Steam and Power Conversion System, Containment, and auxiliary systems to actuate and monitor the operation of the engineered safety features equipment. Process

variables required on a continuous basis for the start-up, operation, and shutdown of the engineered safety features systems are indicated, recorded and controlled from the Control Room. The quantity and type of process instrumentation ensure safe and orderly operation of all systems and processes over the full operating range of the plant.

The Engineered Safety Features Instrumentation System actuates the Safety Injection System, Phase A and Phase B Containment Isolation, CSS, the diesel generators and vital auxiliary systems.

In general, the loss of instrumentation power to the sensors, instruments, or logic devices in the engineered safety features instrumentation places that channel in the trip mode. Exceptions which require instrument power for actuation are described in FSAR Section 7.3.2.

In order to preclude unsafe conditions for plant equipment or personnel, the Reactor Protection System (RPS) is provided. The RPS consists of sensors, calculators, logic and other equipment necessary to monitor selected nuclear steam supply system (NSSS) conditions and to affect reliable and rapid reactor shutdown (reactor trip) if any or a combination of the monitored conditions approach specified safety system settings. The RPS's functions are to protect the core fuel design limits and reactor coolant pressure boundary for anticipated operational occurrences and also to provide assistance in limiting conditions for certain accidents. The RPS is independent of the Reactor Control System, although the control system is dependent upon some signals derived from the protection system through isolation amplifiers. The Reactor Protection System may be used during a normal plant shutdown by inserting a manual reactor trip after the turbine is off line, if desired.

Protection and operational reliability are achieved by providing redundant instrumentation channels for each protective function. These redundant channels are electrically isolated and physically separated. The channel design incorporates separate sensors, separate power supplies, separate rack and panel mounted equipment, and separate logic devices for the actuation of the protective function. For protective functions where two-out-of-three or two-out-of-four coincident actuation is provided, a single channel failure will not impair the protective function.

The RPS is designed so that loss of voltage, the most probable mode of failure, in each channel or logic train results in a signal calling for a trip. The protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not violate reactor protection criteria.

The design philosophy for Reactor Protection and Reactor Control Systems is to make use, for both protection and control functions, of a variety of measurements. The protection and control systems are separate and identifiable. The RPS continuously monitors system variables by different means, demonstrating protection system diversity. The extent of RPS diversity has been evaluated for a wide variety of postulated accidents. Generally, two or more diverse protective functions would terminate an accident before unacceptable consequences could occur.

In the RPS two reactor trip breakers are actuated by two separate logic matrices which interrupt power to the control rod drive mechanism. The breaker main contacts are connected in series with the power supply so that opening either breaker interrupts power to all full length control rod drive mechanisms, permitting the rods to free fall into the core.

Further details on redundancy are provided through the description of the respective systems covered by the various subsections in this chapter. The power supply for the protection system is discussed in Chapter 8.

1.2.2.5 Electrical System

The SHNPP turbine generator provides power at nominal 22kV and is directly connected to a main transformer bank which steps up the voltage to the transmission lines, rated 230kV nominal.

Offsite power is provided to two half-capacity start-up transformers, at 230kV. These start-up transformers have the capability to supply the required safe shutdown and engineered safety features loads for the Unit. The plant auxiliary power is supplied at 6.9kV, 480V and 120V AC.

Redundant sources of onsite power are provided by two diesel generators, either of which is capable of supplying sufficient engineered safety features (ESF) loads to ensure safe shutdown and to maintain the Unit in a safe condition in the event of a complete loss of offsite power.

The ESF redundant systems have been electrically and physically designed and segregated so that a single electrical fault or a single credible event will not cause loss of power to both sets of redundant essential electrical components. See Chapter 8 for further details.

1.2.2.6 Steam and Power Conversion System

The Steam and Power Conversion Systems transform the thermal output of the reactor into electrical power via a turbine-generator. This is accomplished by the transfer of heat from the primary coolant loop to a secondary coolant loop through the steam generators. In the secondary loop, feedwater enters the steam generators and is heated to produce steam. This steam drives the turbine generator, is condensed by rejecting heat to the Circulating Water System, and is recirculated to the Feedwater System. The secondary coolant loop provides an additional barrier to the release of radioactivity to the environment.

The SHNPP has a Westinghouse (now Siemens Energy, Inc.) turbine-generator, rated at approximately 1039.4 MWe, which converts the potential energy of the steam into electrical energy. The steam supply path has the necessary flexibility, relief, and isolation valves to ensure integrity and safety.

The turbine is a three-element, tandem-compound, four flow-exhaust, 1800 rpm unit with moisture separation and single stage reheat between the high-pressure and low-pressure elements. The AC generator is directly connected to the turbine generator shaft.

The Steam Dump System provides the capability to sustain sudden large load decreases up to and including full load loss down to auxiliary loads, concurrent with the loss of external auxiliary power.

The Feedwater and Condensate System is a closed system that deaerates the condensate and pumps it from the condenser hotwell through the feedwater heaters to the steam generators. In the event of the loss of normal feedwater from any cause the safety related Auxiliary Feedwater System will provide water to the steam generators. The safety related Auxiliary Feedwater

System includes two 100 percent capacity motor driven pumps and one 200 percent capacity steam turbine driven pump. See Section 10.4 for additional information.

1.2.2.7 Nuclear Fuel Handling and Storage Systems

The Fuel Handling and Storage System provides for the safe handling of fuel assemblies and control element assemblies and for the required assembly, disassembly, and storage of the reactor vessel head and internals. The Nuclear Fuel Storage System is designed to store new fuel, and spent fuel produced at the SHNPP, H. B. Robinson Steam Electric Plant, and Brunswick Steam Electric Plant in the fuel pools.

The fuel storage system is designed such that the integrity of the fuel is maintained under normal and abnormal conditions. The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a fuel storage pool. The spent fuel is then placed in a cask for shipment from the site.

Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat.

The fuel handling structure may be generally divided in two areas: the refueling cavity which is flooded only during shutdown for refueling; and the spent fuel pools and fuel transfer canal system, which are kept full of water and are always accessible to operating personnel. The refueling cavity and the fuel transfer canal are connected by the fuel transfer tube through which an underwater conveyor transfers the fuel.

The manipulator crane, fuel transfer tube, manual tools and spent fuel bridge crane facilitate the transfer of fuel from the refueling cavity to the spent fuel racks.

The Spent Fuel Pool Cooling and Cleanup System removes decay heat and impurities from the fuel pools.

New fuel assemblies may be stored in any applicable fuel pool. New fuel is delivered to the reactor by lowering it into the appropriate spent fuel pool and taking it through the fuel transfer system. See Section 9.1 for further information.

1.2.2.8 Cooling Water and Other Auxiliary Systems

1.2.2.8.1 Circulating and Service Water System

The Circulating Water System provides the main condenser with a continuous supply of cooling water for removing the heat rejected by the main turbine. Three 33-1/3 percent capacity circulating water pumps, sized for the maximum heat rejection and the required system head, take suction from the cooling tower basin and deliver water to the condenser inlet waterboxes through two large reinforced concrete pipes. After leaving the condenser, the heated circulating water returns to the cooling tower. In addition to circulating water, service water for cooling of auxiliary equipment in the secondary portions is provided during normal operation from the cooling tower basin by means of service water pumps located in a separate intake pump structure. The service water is returned to the Circulating Water System at the outlet from the condenser, for cooling by the Cooling Tower. The Unit has a Service Water System designed to

provide redundant cooling water to those components necessary for safety either during normal operation or under accident conditions.

1.2.2.8.2 Component Cooling Water System

The Component Cooling Water System (CCWS) is an intermediate cooling water system serving components and systems important to the safety of the plant. The CCWS is designed to meet all assigned plant component cooling loads during normal operation, assuming the highest possible service water temperature (95°F). At that temperature, there are no limitations placed on normal plant operation.

Component cooling water transfers heat from the various components to the component cooling heat exchangers which are cooled by the Service Water System (SWS). Since the heat is transferred from the component cooling water to the service water, the CCWS serves as an intermediate system between various auxiliary systems and the SWS. This double barrier arrangement reduces the probability of leakage of potentially radioactive effluent into the Service Water System and insures that any leakage of the radioactive fluid from the components being cooled is contained within the plant.

Because the CCWS functions continuously during the life of the plant, contamination of this system by the SWS, which is treated with chlorine, is avoided by operating the CCWS at a higher pressure than the SWS.

The function of the CCWS of acting as a barrier against leakage of radioactive RCS coolant to the environment is assured by radiation monitors on the CCWS pump inlet lines, water level indication on the surge tank, and by the ability to valve off leaking components. Since the CCWS is one of the Engineered Safety Features and is vital during recovery from an accident, redundancy requirements are included in the system design. In consideration of single failure criteria, the CCWS consists of two separate flow paths for all engineered safeguard functions. See Section 9.2 for a discussion of the CCWS.

1.2.2.8.3 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) is designed to (1) adjust the concentration of chemical neutron absorber in the reactor coolant for reactivity control, (2) maintain the proper water inventory in the Reactor Coolant System (RCS), (3) provide the required seal water flow for the reactor coolant pump shaft seals, (4) provide high pressure flow to the Emergency Core Cooling System (ECCS) (this function is described in Section 6.2.), (5) maintain proper concentration of corrosion inhibiting chemicals in the reactor coolant, and (6) reduce the coolant inventory of corrosion products and fission products.

During normal operation, this system also provides for introduction of the following chemicals:

a) Hydrogen to the volume control tank,

b) Nitrogen as required for purging the volume control tank,

c) Hydrazine and lithium hydroxide as required via the chemical mixing tank to the suction of the charging pumps.

The Unit has one boric acid tank and auxiliary equipment.

In addition to the reactivity control achieved by the rod cluster control assemblies, reactivity control is provided by the CVCS which regulates the concentration of boric acid solution neutron absorber in the RCS. The system is designed to prevent, under system malfunction, uncontrolled or inadvertent reactivity changes which might cause operation in excess of design limits. For further information, see Chapter 9.

1.2.2.8.4 Boron Thermal Regeneration System

The Boron Thermal Regeneration System accepts borated water letdown from the Reactor Coolant System and returns it, with boron added or deleted as required to accomplish reactor coolant boron concentration changes for load follow. The boration and dilution rates made possible by the system are adequate to handle xenon transients resulting from the design load cycle. See Chapter 9 for further information.

1.2.2.8.5 Boron Recycle System

The Boron Recycle System receives and processes reactor coolant effluent for reclamation of the boron and purified water.

1.2.2.8.6 Residual Heat Removal System

The Residual Heat Removal System (RHRS) is designed to remove residual and sensible heat from the core and to reduce the temperature of the Reactor Coolant System during the second phase of Unit cooldown. The RHRS is placed in operation approximately four hours after reactor shutdown depending on reactor coolant temperature. As secondary functions, the RHRS is used for low head safety injection and for transfer of refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations. See Chapter 9 for further information.

1.2.2.8.7 Primary Sampling System (Nuclear)

The Primary Sampling System collects samples of the fluids in the Reactor Coolant System and auxiliary systems for analysis. The system consists of two sampling panels and they are operated from two sampling rooms. Chemical and radiochemical analyses are performed on the samples, and the results are used to regulate boron concentration adjustments, monitor fuel rod integrity, evaluate ion exchanger and filter performance, specify chemical additions to the various systems, and maintain the proper hydrogen overpressure on the volume control tank. For details, see Chapter 9.

1.2.2.8.8 Fire Protection System

The Fire Protection System provides fire prevention through the control, separation and guarding of sources of ignition; fire limitation by means of fire cutoffs and barriers; fire detection in areas containing safety related equipment or areas of high combustible loading; fire extinguishment by means of installed facilities commensurate with the fire hazard presented. The fire extinguishing function is performed by an automatic sprinkler system, with backup by manual suppression systems. See Chapter 9 for further information.

1.2.2.8.9 Compressed Air System

The Compressed Air System provides dry, oil-free compressed air to the instrument air and service air systems. This air is used to operate pneumatic instruments, controls, isolation valves and power relief valves. The system also supplies air for normal maintenance work. For details see Chapter 9.

1.2.2.8.10 Plant Ventilation System

The plant ventilation system provides suitable thermal environment and air quality for personnel comfort, health and safety, and proper equipment operation and integrity. The subsystems that make up the ventilation system are: 1) Control Room Ventilation System, 2) Reactor Auxiliary Building Ventilation System, 3) Containment Ventilation System, 4) Fuel Handling Building Ventilation System, 5) Essential Services Chilled Water System, 6) Waste Processing Building Ventilation System, 7) Turbine Building Ventilation System, and 8) Diesel Generator Building Ventilation System.

1.2.2.8.11 Demineralized Water System

The Demineralized Water System supplies deaerated, treated, demineralized water for various uses throughout the plant. Among these are reactor makeup storage tank, condensate storage tank, refueling water storage tank, radwaste equipment, radiation elements, and the fuel cask decontamination facility. Demineralized water is also used for all-purpose flushing and cleaning of tools, fixtures, etc. which may have become contaminated. (See Chapter 9 for further information.)

1.2.2.9 Waste Processing System

The Waste Processing System is designed to collect, monitor and process all liquid, gaseous and solid radioactive wastes originating from the operation of the plant. The principal design objective is to ensure that the release of radioactive material both in the plant and to the environs does not exceed the limits set forth in 10 CFR 20 and Appendix I of 10 CFR 50. In addition, the Waste Processing System is designed to provide for "as low as reasonably achievable (ALARA)," radioactive release to the environment. The Waste Processing System consists of three major subsystems: Liquid Waste Processing System, Gaseous Waste Processing System, and Solid Waste Processing System.

Liquid and gaseous waste handling at the plant site is based on achieving the lowest practicable radioactive release to the environment using "state of the art technology". Sources of radioactive gaseous waste are segregated into those which are either processed and discharged or processed and held within the plant. Liquid wastes are segregated into those which are 1) processed and recycled, 2) processed and either recycled or discharged, or 3) collected with the capability of processing before discharge should treatment be required. The concept of segregation is further carried out in waste handling such that each type of waste can be handled in specifically designed subsystems. This design philosophy results in minimizing both the quantity of contaminated effluents to be processed and subsequently released, as well as the residual radioactivity contained in these effluents. The solid waste is handled by the Solid Waste Processing System, which is designed to collect, package, store and ship offsite all the solid radioactive waste resulting from plant operation. See Chapter 11 for further information.

1.2.2.10 Radiation Monitoring System

The Radiation Monitoring System (RMS) is a group of independent radiation monitors used to measure the levels of radioactivity within various process streams, ventilation ducts, and plant general areas. The monitors are designed to provide information on the radiation levels to the plant Control Room as well as locally at the monitors. The monitors initiate alarm signals when predetermined setpoints are exceeded and initiate various control functions as required when selected alarms are initiated. The RMS continuously records the levels of radiation in the various streams and areas to indicate trends of increasing radiation so as to achieve ALARA radiation releases and personnel exposures.

1.3 COMPARISON TABLES

1.3.1 COMPARISON WITH SIMILAR FACILITY DESIGNS

Table 1.3.1-1 presents a design comparison of the Shearon Harris Nuclear Power Plant with the Beaver Valley Station and the North Anna Station.

These three facilities have the same Nuclear Steam System Supplier (Westinghouse), the same NSSS design (three steam generators), and approximately the same power output. All values presented in the table are characteristic of the first operating cycle for each plant.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

Table 1.3.2-1 presents a comparison of final and preliminary information for the Shearon Harris Nuclear Power Plant.

Table 1.3.2-1 is presented for historical information only.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 OWNERS

Carolina Power & Light Company

1.4.2 APPLICANTS

Carolina Power & Light Company and North Carolina Eastern Municipal Power Agency are the applicants for the operating licenses (the North Carolina Eastern Municipal Power Agency ownership was transferred to Duke Energy Progress, Inc. on July 31, 2015). Carolina Power & Light Company (now Duke Energy Progress, Inc.) is responsible for the design, construction, and operation of the Shearon Harris Nuclear Power Plant. Carolina Power & Light Company has been active in the nuclear power field since 1956, when the Company joined with three neighboring utilities to form the Carolinas-Virginia Nuclear Power Association to build and operate a nuclear steam generating plant at Parr, South Carolina. Since then, Carolina Power & Light Company has placed three nuclear units in commercial operation--the H. B. Robinson Plant at Hartsville, South Carolina, in 1971; the Brunswick Steam Electric Plant Unit 2 at Southport, North Carolina, in 1975; and the Brunswick Steam Electric Plant Unit 1 in 1977. Carolina Power & Light Company has engaged the contractors noted below to perform engineering and consultant services and provide equipment for Shearon Harris Nuclear Power

Plant. Carolina Power & Light Company has participated in the plant design, has reviewed the plant design, and has overall responsibility for SHNPP.

1.4.3 ARCHITECT/ENGINEER

Ebasco Services Incorporated (Ebasco) is the architect-engineer responsible to Carolina Power & Light Company for the design, engineering, and equipment and material procurement of SHNPP. This includes all plant structures, systems, and components other than those provided by the Nuclear Steam Supply System (NSSS) supplier, except for any promotional or other nonplant-oriented structures which will be done by outside contractors.

Ebasco has used the services of:

1.4.3.1 S. Seroho Associates (Waste Processing)

1.4.3.2 Franklin Research Institute (Applied Physics)

1.4.3.3 EDS Nuclear (Stress Analysis)

1.4.3.4 Dr. W. C. Pitman (Seismology)

1.4.3.5 Dr. A. L. Odom (Radiometric Age Determinations)

1.4.3.6 Dr. P. C. Ragland (Geochemistry)

1.4.3.7 Dr. S. B. Weed (Clay Mineralogy)

1.4.3.8 Dr. J. B. Butler (Petrography)

1.4.3.9 Dr. D. F. Schutz (Radiometric Age Determinations)

1.4.3.10 Dr. J. de Boer (Paleomagnetic Studies)

1.4.3.11 Dr. G. A. Kiersch (Geologic Report Reviewer)

1.4.3.12 Lehigh University (See Section 3.8)

1.4.4 CONTRACTORS

Carolina Power & Light Company has selected the services of Daniel Construction Company to construct the Shearon Harris Nuclear Power Plant. Carolina Power & Light Company is performing ASME nuclear code construction under Carolina Power & Light Company's quality assurance program which was evaluated and accepted by the ASME Survey team. The constructor, Daniel International Corporation, is working under direct supervision and technical control of Carolina Power & Light Company management personnel at the site. The responsibility for construction activities of this nuclear plant is that of Carolina Power & Light Company.

1.4.5 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) SUPPLIER

The NSSS, along with the design and fabrication of the initial fuel, is supplied by Westinghouse Electric Corporation.

1.4.6 CONSULTANTS

1.4.6.1 Dames & Moore

This firm has been retained to work in conjunction with Ebasco as consultants for seismic studies for the plant.

1.4.6.2 Dr. Jasper L. Stuckey

Dr. Stuckey has provided assistance in the area of geology.

1.4.6.3 Dr. B. J. Copeland

Dr. Copeland provided a report entitled "Ecological Report for CP&L on White Oak Creek Site" which briefly summarized the terrestrial and aquatic ecology of the White Oak basin.

1.4.6.4 Dr. Joffre L. Coe, Director, Research Laboratories of Anthropology, University of North Carolina

The Research Laboratories of Anthropology performed archaeological (historic and prehistoric) surveys of the site.

1.4.6.5 Aquatic Control, Inc.

Aquatic Control, Inc. provided consulting services in the fields of aquatic and terrestrial ecology by conducting and reporting on-site baseline surveys of the biota.

1.4.6.6 Mr. William Beck, Florida A&M University

Mr. Beck provided services for verification of benthic specimen identifications.

1.4.6.7 Dr. Samuel Mozley, North Carolina State University

Dr. Mozley provided services for verification of benthic specimen identifications.

1.4.6.8 Pickard, Lowe & Garrick, Inc.

This firm provided assistance as a nuclear consultant.

1.4.6.9 Research Triangle Institute

The Research Triangle Institute provided consulting services in the fields of meteorology and demography.

1.4.6.10 H.A.F.A. International, Inc.

This company provided consulting engineering services by reviewing the ASME Code Class 1, 2, and 3 boundary designations and reviewing preservice and inservice inspection requirements for each system, reviewing the pump and valve test program as well as the Type C testing program required by 10 CFR 50 Appendix J.

1.4.6.11 TERA Corporation

TERA Corporation provided consulting services involving the preparation and review of various subsections of the Radiation Protection section of the SHNPP FSAR.

1.4.6.12 Law Engineering Testing Co.

Law Engineering conducted geotechnical testing.

1.4.6.13 Nello L. Teer Co.

Nello L. Teer Co. performed excavation work at plant site.

1.4.6.14 Southwest Research Institute

This firm has provided review of the SHNPP for access to perform Preservice/Inservice Inspection.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

Deleted by Amendment No. 48

The remainder of Section 1.5 was deleted by Amendment No. 48

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 lists topical reports which provide information additional to that provided in this FSAR and have been filed separately with the Nuclear Regulatory Commission (NRC) in support of this and similar applications.

A legend to the review status code letters follows:

A - NRC review complete; NRC acceptance letter issued.

AE - NRC accepted as part of the Westinghouse emergency core cooling system (ECCS) evaluation model only; does not constitute acceptance for any purpose other than for ECCS analyses.

B - Submitted to NRC as background information; not undergoing formal NRC review.

O - On file with NRC; older generation report with current validity; not actively under formal NRC review.

U - Actively under formal NRC review.

N - Not applicable; i.e., open literature, etc.

Table 1.6-2 lists other reports which provide information additional to that provided in this FSAR and have been filed separately with Nuclear Regulatory Commission (NRC) in support of this application.

Table 1.6-3 provides a list of Design Documents which are incorporated by reference as part of the FSAR. These Design Documents are to be used in lieu of the cross-referenced FSAR Figures and Tables which have been removed from the FSAR.

Table 1.6-4 lists plant procedures, programs, or manuals which are incorporated by reference into the FSAR.

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

This section is not applicable to updated FSAR (Reference 1.7.0-1).

The remainder of Section 1.7 was deleted by Amendment No. 48

REFERENCES: SECTION 1.7

1.7.0-1 Generic Letter 81-06 issued by Darrell G. Eisenhut regarding Periodic Updating of Final Safety Analysis Reports (FSARs), December 15, 1980.

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

This section describes the extent to which the SHNPP project complies with all applicable NRC regulatory guides. All regulatory guides which by virtue of the implementation section of the guide itself are applicable to the project, or have been designated as Category 2, 3, or 4 by the NRC's Regulatory Requirements Review Committee have, as a minimum, been addressed.

Whenever the requirements of the technical specifications conflict with the requirements of regulatory guides and codes and standards, the requirements of the technical specifications shall govern.

Specific applicability of referenced standards (i.e., standards other than the primary one endorsed by the specific regulatory guide) are noted in that portion of this section where the regulatory guide has endorsed it as the primary standard. The extent to which a standard applies to a particular situation will be determined by responsible plant management as evidenced through concurrence/approval of governing procedures or other documents.

Regulatory Guide 1.1 NET POSITIVE SUCTION HEAD (NPSH) FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS (REV. 0)

The SHNPP Project complies with Regulatory Guide 1.1.

FSAR Reference: Sections 6.2.2, 6.3.2, and 6.5.2.

Regulatory Guide 1.2 THERMAL SHOCK TO REACTOR PRESSURE VESSELS (REV. 0)

This Regulatory Guide has been withdrawn by the NRC. The following is historical information. Refer to Section 5.3.1.8 for current information regarding Pressurized Thermal Shock.

The SHNPP project complies with this guide as described below:

Westinghouse follows all the recommendations of Regulatory Guide 1.2. Regulatory Position C.1 is followed by Westinghouse's own analytical and experimental programs as well as by participation in the heavy section steel technology (HSST) programs at Oak Ridge National Laboratory.

Analytical techniques have been developed by Westinghouse to perform fracture evaluations of reactor vessels under thermal shock loadings.

Under the HSST program a number of six in. thick 39 in. outside diameter steel pressure vessels containing carefully prepared and sharpened surface cracks are being tested. Test conditions include both hydraulic internal pressure loadings and thermal shock loadings. The objective of this program is to validate analytical fracture mechanics techniques and demonstrate quantitatively the margin of safety inherent in reactor pressure vessels.

A number of vessels have been tested under hydraulic pressure loadings, and results have confirmed the validity of fracture analysis techniques. The results and implications of the hydraulic pressure tests are summarized in Oak Ridge National Laboratory report ORNL-TM-5090.

Three thermal shock experiments have been completed and are now being evaluated. Preliminary information indicates that the analytical techniques do agree favorably with experimental results. Westinghouse is continuing to obtain fracture toughness data for reactor pressure vessel steels through internally funded programs as well as HSST sponsored work.

Fracture toughness testing of irradiated compact tension fracture toughness specimens has been completed. The complete post-irradiation data on 0.394 in., 2 in., and 4 in. thick specimens are now available from the HSST program. Both static and dynamic post-irradiation fracture toughness data have been obtained. Evaluation of the data obtained to date on material irradiated to fluences between 2.2 and 4.5 x 10^{19} n/cm^2 indicates that the reference toughness curve as contained in the American Society of Mechanical Engineers (ASME) Code, Section III, remains a conservative lower bound for toughness values for pressure vessel steels.

Regulatory Guide 1.3 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT FOR BOILING WATER REACTORS (REV 2)

This guide is not applicable to SHNPP.

Regulatory Guide 1.4 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOCA FOR PWR'S (REV. 2)

The LOCA dose analysis for the SHNPP project is not based on Regulatory Guide 1.4. The Alternate Source Term methodology is being used following SGR/PUR, therefore the guidance of Regulatory Guide 1.183 (Reference 1.8-16) was followed.

FSAR Reference: Sections 15.0A and 15.6.5.

Regulatory Guide 1.5 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS (REV 0)

This guide is not applicable to SHNPP.

Regulatory Guide 1.6 INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS (REV. 0)

The SHNPP project complies with Regulatory Guide 1.6 as described in FSAR Section 8.3.1.2.3.

Regulatory Guide 1.7 CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT (REV. 3)

The SHNPP project complies with Regulatory Guide 1.7 with the clarification that the hydrogen analyzers are powered via associated circuits from 1E power sources, classified per Reg Guide 1.97 Rev. 3 as Category 3, and environmentally qualified to function after beyond design-basis accidents using the methodology of IEEE-323-1974.

FSAR Reference: 6.2.5, 7.3.1.6.2.e.

Regulatory Guide 1.8 PERSONNEL SELECTION AND TRAINING (REVISION 2, FEBRUARY 1979 DRAFT)

SHNPP will comply with the requirements of ANSI/ANS 3.1, September 1979 Draft, with the alternatives listed herein. It is understood that the NRC has not endorsed this Standard, but when the SHNPP applied for its operating license, the September 1979 Draft was current. Because this standard was the existing guidance at the time of our operating license application, CP&L believes it is acceptable to use the draft Standard as the basis for selecting and training SHNPP personnel. The Company has received approval from NRC to follow the September 1979 Draft without further revisions.

1) Paragraph 2 defines the terms of the Standard. As stated in SHNPP FSAR Section 1.8, paragraph 1.74, CP&L has combined the definitions given in various ANSI standards, in order to provide an available reference source. The definitions in Section 1.8, paragraph 1.74 agree with ANSI/ANS 3.1, September 1979 Draft with the following exception:

   When the phrase "Bachelor's Degree or Equivalent" is used, except for the position of Shift Technical Advisor, the qualifications considered as minimal acceptable substitutes for a Bachelor's Degree are a high school diploma or its equivalent and one of the following:

   a) Four years of formal schooling in science or engineering;

b) Four years applied experience at a nuclear facility in the area for which qualification is sought;

c) Four years of operational or technical experience or education or training in nuclear power; or

d) Any combination of the above totaling four years.

2) Table 1.8-1 cross references the "Functional Level and Assignment of Responsibility" definitions found in Section 3 of the Standard with the positions/titles of the SHNPP organization and the "Qualifications" found in Section 4 of the Standard. The numbers enclosed in parentheses denote the specific exceptions or proposed alternatives to the Standard's requirements which are described in paragraph (3) below:
3) Exceptions or proposed alternatives:

a) Paragraph 4.2.2 describes the qualifications for the Operations Manager. Instead of requiring the Operations Manager to hold an SRO license, CP&L will require the Operations Manager to meet one of the following: (1) hold an SRO license or (2) have held an SRO license for a similar unit, or (3) have been certified for equivalent senior operator knowledge for a similar unit. If the Operations Manager does not hold an SRO license, an off-shift Operations Superintendent who reports directly to the Operations Manager and holds an SRO license will be designated to supervise shift work and licensed activities.

b) Paragraph 4.3.1 describes the qualifications for supervisors requiring NRC licenses. CP&L will prescribe qualification requirements for supervisors requiring NRC licenses, in accordance with FSAR Section 13.2.

c) Paragraph 4.3.2 describes the qualifications for supervisors who are not required to hold an NRC license, but who are associated with "systems, equipment, or procedures involved in meeting the Limiting Conditions for Operation, which are identified in Technical Specifications". CP&L does not feel plant safety will be enhanced by requiring these supervisors to perform their duties under direct on-site supervision for a minimum of six months. Instead, CP&L proposes to select qualified individuals for these positions based upon past performance and experience.

d) Paragraph 4.5.1.1 describes the requirements for non-licensed operators. CP&L does not feel plant safety will be enhanced by requiring non-licensed operators to have one year power plant experience. CP&L shall alternatively provide a training/qualification program commensurate to the functions and responsibilities these employees will perform.

e) Paragraph 4.5.1.2 describes the qualification requirements for licensed operators. CP&L will prescribe qualifications for licensed operators in accordance with FSAR Section 13.2.

f) Paragraphs 4.5.2 and 4.5.3 describe the qualifications for technicians and maintenance personnel. CP&L considers these technicians and maintenance employees to be "in training or apprentice positions", as described in paragraph

3.2.4. Therefore, CP&L shall comply with the requirements as stated in paragraph 3.2.4.

g) Personnel performing inspections will be trained and qualified in accordance with Regulatory Guide 1.58, which endorses ANSI N45.2.6. The SHNPP position on Regulatory Guide 1.58 addresses the SHNPP positions relative to ANSI N45.2.6.

h) Various CP&L positions are not addressed in the Standard. Therefore, CP&L lists these positions in Table 1.8-1 for reference, and CP&L will prescribe the training, responsibilities, and qualifications commensurate to the job requirements.

i) The ALARA Analyst/Engineer shall have a BS Degree or the equivalent and two years' experience, one of which shall be nuclear power plant experience, or the employee shall have an advanced degree and one year nuclear power plant experience.

j) The positions specified in Table 1.8-1 shall have a BS Degree in Engineering or the equivalent and two years' experience, one of which shall be nuclear power plant experience, or the employees shall have an advanced degree and one year nuclear power plant experience. These qualifications are required at initial core loading or at position appointment, whichever is later.

k) The Training Specialist shall have at least four years power plant experience, two of which shall be nuclear power plant experience. Individuals in this position shall demonstrate their competence by having held an SRO license or by having trained at the SRO level prior to teaching NSSS integrated response, transient analysis, or simulator courses. These qualifications are required at initial core loading or at position appointment, whichever is later.

l) The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

4) Paragraphs 4.7.1 and 4.7.2 describe the qualifications for independent review personnel. The FSAR also address the personnel requirements for individuals functioning in this capacity, and alternatively, CP&L shall comply with requirements of Section 17.3.4.1.3 of the FSAR.

5) Paragraph 5.2 outlines an acceptable training program for personnel to be licensed by the NRC. CP&L will provide a training program as described in FSAR Section 13.2 for licensed operators and senior operators.

   Paragraph 5.5.1 outlines the retraining program for licensed personnel. CP&L will provide a Licensed Operator Requalification training program in accordance with FSAR Section 13.2.

6) Paragraph 5.5.2.3 describes requirements to maintain certain documents. In order to provide consistency in the Document Control program, CP&L shall retain and maintain documents as required by ANSI N45.2.9-1974.

7) Paragraph 1, Scope, states in part, "this standard is further limited to personnel within the owner organization." However, paragraph 5.4 refers to temporary maintenance and service

personnel. CP&L will apply the requirements of ANS 3.1, September 1979 to only those personnel directly employed by CP&L, and only the training of paragraph 5.4 will be required to be given to temporary maintenance and service personnel.

8) Positions shown on the SHNPP organization chart that have not been described herein shall be filled by individuals, who by virtue of training and experience, have been deemed qualified to fill these positions.
9) Paragraph 5.5.2 outlines annual retraining of personnel not requiring NRC licenses. Harris will conduct retraining on a frequency as determined using processes derived from the Systematic Approach to Training, prescribed per the Institute of Nuclear Power Operations (INPO).

Regulatory Guide 1.9 SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES (REV. 2)

The SHNPP project complies with Regulatory Guide 1.9, as presented in FSAR Sections 8.3.1.2.4 and 14.2.12.1.16.

Regulatory Guide 1.10 MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURES (REV. 1)

The SHNPP project complies with Regulatory Guide 1.10 with the following clarification:

In regard to regulatory position C.3.(b), the design and as-built drawings for the SHNPP project indicate the relative locations of all cadweld splices that are made either to meet design requirements or as replacements for cadweld repairs or test splices. It is not necessary to dimensionally indicate the exact cadweld splice location on the drawings. The above documents will be maintained for the entire life of the plant.

FSAR Reference: Section 3.8.1.6, Appendix 3.8A.

Regulatory Guide 1.11 INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT (3-10-71) AND SUPPLEMENT TO SAFETY GUIDE II, 2-17-72

The SHNPP project meets the intent of Safety Guide 1.11 as follows:

There are four instrument lines which penetrate the Containment that are designed to Safety Guide 1.11. The lines are associated with pressure transmitters whose signals can initiate safety injection and containment isolation; and they are the only containment pressure-measuring signals available to initiate containment spray. They do not have automatic isolation valves since these instruments must be operable during a postulated accident to initiate containment spray. These instrument lines are connected to the containment atmosphere by a filled and sealed hydraulic transmission system similar to a sealed pressurizer water reference leg. This arrangement, together with the pressure sensors external to the Containment, forms a double barrier and is otherwise in agreement with Safety Guide 1.11.

Should a leak occur outside Containment, the sealed bellows inside Containment, which is designed to withstand full containment design pressure, will prevent the escape of containment atmosphere. Should a leak occur inside Containment, the diaphragm in the transmitter, which is designed to withstand full containment design pressure, will prevent any escape from

Containment. This arrangement provides automatic double barrier isolation without operator action and without sacrificing any reliability with regard to its safeguards functions (i.e., no valves to be inadvertently closed). Both the bellows and tubing inside Containment and the transmitter and tubing outside Containment are enclosed by protective shielding. Because of this sealed fluid filled bellows system, a postulated severance of the line during either normal operation or accident conditions will not result in any release from the Containment.

FSAR Reference: Section 6.2.4.

Regulatory Guide 1.12 INSTRUMENTATION FOR EARTHQUAKES (REV. 1).

The SHNPP project complies with Regulatory Guide 1.12. Interpretations and clarifications are described in FSAR Section 3.7.4.

FSAR Reference: 3.7.4.

Regulatory Guide 1.13 SPENT FUEL STORAGE FACILITY DESIGN BASIS (REV 1)

The SHNPP project complies with Regulatory Guide 1.13, based on the understanding that automatic actuation of the Fuel Handling Building HVAC System satisfies the requirement in regulatory position C.7 for automatic actuation of the filtration system and that automatic actuation of the fuel pool purification system is not required.

FSAR References: 9.1.3., 9.1.4.

Regulatory Guide 1.14 REACTOR COOLANT PUMP FLYWHEEL INTEGRITY (REV. 1)

The SHNPP project follows the recommendations of this guide with the following exceptions:

a) Post-spin inspection - Westinghouse has shown in Reference 1.8-1 that the flywheel would not fail at 290 percent of normal speed for a flywheel flaw of 1.15 inches or less in length. Results for a double-ended guillotine break at the pump discharge with full separation of pipe ends assumed, show the maximum overspeed to be less than 110 percent of normal speed. The maximum overspeed was calculated in Reference 1.8-1 to be about 280 percent of normal speed for the same postulated break, and an assumed instantaneous loss of power to the reactor coolant pump. In comparison with the overspeed presented above, the flywheel is tested at 125 percent of normal speed. Thus, the flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125 percent provided that no flaws greater than 1.15 inches are present. If the maximum speed were 125 percent of normal speed or less, the critical flaw size for failure would exceed 6 in. in length. Nondestructive tests and critical dimension examinations are all performed before the spin test. The inspection methods employed (described in Reference 1.8-1) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 in. for 290 percent of normal speed would be detected. Flaws in the flywheel will be recorded in the pre-spin inspection program (see Reference 1.8-1). Flaw growth attributable to the spin test (i.e., from a single reversal of stress, up to speed and back), under the most adverse conditions, is about three orders of magnitude smaller than that which nondestructive inspection techniques are capable of detecting. For

these reasons, Westinghouse performs no post-spin inspections and believes that pre-spin test inspections are adequate.

b) Interference for stress and excessive deformation - Much of Revision 1 to Regulatory Guide 1.14 deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because the Westinghouse design specifies a light interference fit between the flywheel and the shaft, at zero speed, the loop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and at normal speed, the flywheel is not in contact with the shaft in the sense intended by Revision 1. Hence, the definition of "Excessive Deformation," as defined in Revision 1 of Regulatory Guide 1.14, is not applicable to the Westinghouse design since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft does not cause unbalance of the flywheel. Extensive Westinghouse experience with reactor coolant pump flywheels installed in this fashion has verified the adequacy of the design.

Westinghouse's position is that combined primary stress levels, as defined in Revision 0 of Safety Guide 14 (Regulatory Positions C.2.a and C.2.c) are both conservative and proven and that no changes to these stress levels are necessary. Westinghouse designs to these stress limits and thus, does not have permanent distortion of the flywheel bore at normal or spin test conditions.

c) Discussion B, cross rolling ratio of 1 to 3 - Westinghouse's position is that specification of a cross rolling ratio is unnecessary since past evaluations have shown that ASME SA-533, Grade B, Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotropy in the flywheel material by means of cross rolling is unnecessary since adequate margins of safety are provided by both flywheel material selection (ASME SA-533, Grade B, Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material.

d) Regulatory position C.1.a, relative to vacuum-melting and degassing process or the electroslag process - The requirements for vacuum melting and degassing process or the electroslag process are not essential in meeting the balance of the regulatory position nor do they, in themselves, ensure compliance with the overall regulatory position. The initial Safety Guide 14 (10/27/71) stated that the "flywheel material should be produced by a process that minimized flaws in the material and improves its fracture toughness properties." This is accomplished by using ASME SA-533 material including vacuum treatment.

e) Regulatory position C.2.b - Westinghouse suggests that this paragraph be reworded as follows in order to remove the ambiguity of reference to an undefined overspeed transient.

"Design speed should be 125 percent of normal speed or the speed to which the pump motor might be electrically driven by station turbine generator during anticipated

transients, whichever is greater. Normal speed is defined as the synchronous speed of the alternating current drive motor at 60 hertz."

FSAR Reference: 5.4.1.

Regulatory Guide 1.15 TESTING OF REINFORCING BARS FOR CATEGORY I STRUCTURES (REV. 1)

The SHNPP project complies with this guide.

FSAR Reference: Section 3.8.1

Regulatory Guide 1.16 REPORTING OF OPERATING INFORMATION-APPENDIX A TECHNICAL SPECIFICATIONS (REV 4)

The SHNPP project complies with this Regulatory Guide except as noted below.

In lieu of positions C.1 and C.2 of the Regulatory Guide, those reports indicated in the guide shall be reported as per the plant Technical Specifications.

In lieu of position C.1.c of the Regulatory Guide, SHNPP will provide to the NRC the operating data described in Generic Letter 97-02 "Revised Contents of the Monthly Operating Report" via an industry database (e.g., the Consolidated Data Entry (CDE) program) by the end of the month following each calendar quarter.

Regulatory Guide 1.17 PROTECTION OF NUCLEAR POWER PLANTS FROM INDUSTRIAL SABOTAGE (REV. 1)

SHNPP does not commit to Regulatory Guide 1.17. The SHNPP Security Plan, which has been submitted separately, addresses protection from radiological sabotage per 10 CFR 73.

FSAR Reference: Section 13.6.

Regulatory Guide 1.18 STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENT (REV. 1)

The SHNPP project complies with this guide with the following exceptions:

The SHNPP Containment is a non-prototype reinforced concrete structure as discussed in FSAR Section 3.8.1.7.1. The regulatory positions pertaining to prototype containments are not applicable.

FSAR Reference: Section 3.8.1.7.1.

Regulatory Guide 1.19 NONDESTRUCTIVE TESTING OF PRIMARY CONTAINMENT LINER WELDS (REV. 1)

The SHNPP project complies with Regulatory Guide 1.19 with the following clarifications and exceptions:

a) Regulatory position C.1.b: The SHNPP project uses magnetic-particle or liquid penetrant testing for examining the liner seam welds where radiographic testing is not feasible or where the weld is located in areas which are not accessible after construction. It is felt that liquid penetrant testing, being a surface examination method, is more suited than ultrasonic testing for detection of possible leak prone discontinuities.

b) Regulatory position C.1.c: The SHNPP project does not require hourly checking of the soap solution for adequacy of bubble formation properties because this is not a requirement of the ASME Code, Section III, Division 2, Subsection CC.

c) Regulatory positions 7.a, 7.b, 8.a, and 8.b: The SHNPP project complies with the requirements of the ASME Code, Section III, Division 2, Subsection CC as further discussed in Section 3.8.1 and Appendix 3.8A. No spot radiography was performed prior to April 29, 1977.

d) Whenever there is a conflict between this regulatory guide and the ASME Code Section III Division 2, Winter 75 addenda the requirements of the ASME Code with the clarifications shown in Appendix 3.8A will prevail.

FSAR Reference: Section 3.8.1.

Regulatory Guide 1.20 COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING (REV 2)

The Westinghouse position regarding this guide is as described in the FSAR Section 3.9.2.4.

FSAR Reference: Section 3.9.2.

Regulatory Guide 1.21 MEASURING, EVALUATING AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT WATER COOLED NUCLEAR POWER PLANTS (REV. 1)

The SHNPP project complies with this guide.

FSAR Reference: Section 11.5.1.

Regulatory Guide 1.22 PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS (REV 0)

The SHNPP project complies with this guide as described in Section 7.1.2.5 and 7.3.2.

FSAR Reference: Sections 7.1.2 and 7.3.2.

Regulatory Guide 1.23 ONSITE METEOROLOGICAL PROGRAMS (REV. 0)

The SHNPP project complies with this guide.

FSAR Reference: Section 2.3.3.

Regulatory Guide 1.24 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE (REV 0)

The SHNPP project complies with the intent of Regulatory Guide 1.24. Methodology and assumptions differences are:

1) Dispersion factors are determined based on Regulatory Guide 1.145 methodology, as described in FSAR Section 2.3.4.

2) Instead of calculating "whole body doses," Total Effective Dose Equivalent (TEDE) doses were determined during the SGR/PUR effort, as described in Regulatory Guide 1.183 (Reference 1.8-16).

FSAR Reference: 15.7.1

Regulatory Guide 1.25 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS (REV 0)

The Fuel Handling Accident dose for the SHNPP project is not based on Regulatory Guide 1.25. The Alternate Source Term Methodology is being used following SGR/PUR, and therefore the guidance from Regulatory Guide 1.183 (Reference 1.8-16) was followed.

FSAR Reference: 15.7.4.

Regulatory Guide 1.26 QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE WASTE CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS (REV. 3)

Systems and components are classified Safety-Related in accordance with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", and ANSI N18.2a-1975, "Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

These categories are equivalent to the Quality Group categories of the Regulatory Guide.

FSAR Reference 3.2.2.

Regulatory Guide 1.27 ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS (REV 2)

The SHNPP project complies with this guide with the following clarifications:

Regulatory Position C.1.a-Section 2.4.11.1 indicates the assumptions used to calculate the maximum drawdown in the ultimate heat sink water source. Table 2.4.11-3 tabulates the

meteorological conditions utilized to compute the drawdown for the Auxiliary Reservoir and for the Main Reservoir. As indicated in Calc SW-0085, there is considerable additional water availability following the calculated periods of drawdown. Carolina Power & Light Company believes that utilization of the meteorological conditions specified in Regulatory Position C.1.a during the calculated periods of drawdown would not jeopardize ultimate heat sink availability.

Regulatory Position C.1.b-Section 2.4.11.7 indicates the assumptions used to calculate the maximum service water system (SWS) inlet temperature during the auxiliary reservoir ultimate heat sink operations. The Auxiliary Reservoir is the critical water source for determining this parameter during a critical period. The meteorological conditions that maximize the service water temperature are high solar heating, high ambient air temperature, high relative humidity, and low wind speed. The worst meteorology for one day occurred on June 27, 1952, and the worst month occurred between July 18 and August 15, 1949. The average values of these meteorological parameters for 1-day to 30-day periods were calculated based on these days to estimate the maximum service water system inlet temperature. The limiting analysis calculated a maximum pre-accident reservoir temperature of 94.2°F using a composite 10-day period including the worst 9-day consecutive meteorology (7/22/49 - 7/30/49) plus the worst 1-day (6/27/52). The auxiliary Reservoir initial water temperature of 82.2°F is assumed in the analysis based on the July reservoir equilibrium water temperature for the normal meteorological conditions. The analysis also found that a pre-accident reservoir temperature of 94°F would result in a final, 30-day, post-LOCA temperature of 95.33°F, which is just slightly above the ESW design-basis temperature of 95°F. This is acceptable because the reservoir analysis does not account for thermal stratification which would result in a pre-accident temperature below 94°F and a post-accident temperature below 95°F. (Reference Calc SW-0085.)

FSAR Reference: Section 2.4, 9.2.5.

Regulatory Guide 1.28 QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Conformance with Regulatory Guide 1.28 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3).

Regulatory Guide 1.29 SEISMIC DESIGN CLASSIFICATION (REV 3)

The SHNPP project complies with this guide with the following clarification: Westinghouse classifies each component important to safety as Safety Class 1, 2, or 3 and these classes are qualified to remain functional in the event of the safe shutdown earthquake, except where exempted by meeting all of the below requirements. Portions of systems required to perform the same safety function as required of a safety class component which is part of that system shall be likewise qualified or granted exemption. Conditions to be met for exemption are:

a) Failure would not directly cause an ANS Condition III or IV event (as defined in ANSI N18.2-1973),

b) There is no safety function to mitigate, nor could failure prevent mitigation of, the consequences of an ANS Condition III or IV event,

c) Failure during or following any ANS Condition II event would result in consequences no more severe than allowed for an ANS Condition III event, and

d) Routine post-seismic procedures would disclose loss of the safety function.

For items classified as Seismic Category I, only the pertinent requirements of 10 CFR 50 Appendix B apply, lineage traceability of materials is not required and certificates of compliance in lieu of certified material test reports are considered acceptable.

FSAR Reference Sections 3.2, 7.6.2.2.e, and 8.3.1.

Regulatory Guide 1.30 QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION AND TESTING OF INSTRUMENTATION AND ELECTRIC EQUIPMENT

Conformance with Regulatory Guide 1.30 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.31 CONTROL FERRITE CONTENT IN STAINLESS STEEL WELD MATERIAL (REV. 2)

The SHNPP meets the intent of this guide as follows:

NSSS - The Westinghouse position concerning the control of delta ferrite in stainless steel welding is discussed in Section 5.2.3. The Westinghouse production weld verification program, as described in Reference 1.8-2, was approved as a satisfactory substitute for conformance with the NRC Interim Position on Regulatory Guide 1.31 (April, 1974). The results of the verification program have been summarized and documented in Reference 1.8-3.

Balance of Plant - The extent of compliance with this guide described in Section 10.3.6.2 is applicable to all balance of plant austenitic stainless steel components.

Field Work - CP&L will comply with revision 3 in lieu of revision 2 of this guide.

FSAR Reference: Sections 5.2.3, and 10.3.6.

Regulatory Guide 1.32 CRITERIA FOR SAFETY-RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS (REV. 2)

The SHNPP project complies with the design requirements of this guide except as noted in Section 8.3.2.2.1.3 regarding the Class IE DC Power System. Engineered Safety Features System initiation applications and the instrumentation and control power supply system analysis are discussed in Sections 7.3.2.2 and 7.6.2.3, respectively.

FSAR Reference: Sections 7.3.2.2, 7.6.2.3, 8.3.1.2, and 8.3.2.2.

Regulatory Guide 1.33 QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Conformance with Regulatory Guide 1.33 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.34 CONTROL OF ELECTROSLAG WELD PROPERTIES (REV. 0)

The SHNPP project complies with this guide as described below:

NSSS - Where electroslag welding is used in fabricating nuclear plant components, the Westinghouse procurement practice requires vendors to follow the recommendations of Regulatory Guide 1.34.

Field Work and Balance of Plant items - This guide is not applicable since electroslag welding will not be performed on low alloy steels and austenitic stainless steels.

Regulatory Guide 1.35 INSERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES (REV 2)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.36 NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL (REV 0)

The SHNPP project complies with this guide as described below:

NSSS

The Westinghouse practice meets the recommendations of Regulatory Guide 1.36 and is more stringent in several respects as discussed below.

The tests for qualification specified by this Regulatory Guide (ASTM C692-71 or RDT M12-1T) allow use of the tested insulation materials if no more than one of the metallic test samples crack. Westinghouse rejects the tested insulation material if any of the test samples crack.

Fiberglass insulation procured after 1977 may be tested in accordance with ASTM C692-77 which endorses a more stringent 4 out of 4 acceptable sample criteria, a position that has been accepted by the NRC (Reference 1.8-17)

The Westinghouse procedure is more specific than the procedures suggested by this Regulatory Guide, in that the Westinghouse specification requires determination of leachable chloride and fluoride ions from a sample of the insulating material. The procedures in this Regulatory Guide, ASTM D512 and ASTM D1179, do not differentiate between leachable and unleachable halogen ions.

In addition, Westinghouse experience indicates that only one of the three methods allowed under ASTM D512 and ASTM D1179 for chloride and fluoride analysis is sufficiently accurate for reactor applications. This is the "referee" method, which is used by Westinghouse.

Balance of Plant

All balance of plant thermal insulation meets the recommendations of this guide.

Fiberglass insulation procured after 1977 may be tested in accordance with ASTM C692-77 which endorses a more stringent 4 out of 4 acceptable sample criteria, a position that has been accepted by the NRC (Reference 1.8-17).

FSAR Reference: Section 5.2.3.

Regulatory Guide 1.37 QUALITY ASSURANCE REQUIREMENTS FOR CLEANING FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.37 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.38 QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.38 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.39 HOUSEKEEPING REQUIREMENTS FOR WATER COOLED NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.39 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.40 QUALIFICATION TESTS FOR CONTINUOUS DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS (REV 0)

The SHNPP project will comply with this guide for Class 1 continuous duty motors located inside Containment.

FSAR Reference: Sections 3.10, 3.11, and 8.3.1.

Regulatory Guide 1.41 PREOPERATIONAL TESTING OF REDUNDANT ON-SITE ELECTRIC POWER SYSTEMS TO VERIFY PROPER LOAD GROUP ASSIGNMENTS (REV 0)

The SHNPP project will comply with this guide.

FSAR References: Sections 8.3.1, and 14.2.

Regulatory Guide 1.42 INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHT-WATER COOLED NUCLEAR POWER REACTORS (REV 1)

This guide was withdrawn by the NRC on March 18, 1976.

Regulatory Guide 1.43, CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS (REV. 0)

The SHNPP project complies with this guide as described below:

NSSS - Westinghouse practices achieve the same purpose as Regulatory Guide 1.43 by requiring qualification of any high heat input process, such as the submerged-arc wide-strip welding process and the submerged-arc 6-wire process used on ASME SA-508, Class 2, material, with a performance test as described in Regulatory Position 2 of the guide. No qualifications are required by the regulatory guide for ASME SA-533 material and equivalent chemistry for forging grade ASME SA-508, Class 3, material.

The fabricator monitors and records the weld parameters to verify agreement with the parameters established by the procedure qualification as stated in Regulatory Position C.3.

Field Work and Balance of Plant - Cladding of low alloy steels for safety-related components is not done.

Regulatory Guide 1.44 CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL (REV. 0)

NSSS - The Westinghouse position on Regulatory Guide 1.44 is discussed in part in Section 5.2.

Westinghouse compliance with the separate positions of this regulatory guide are as follows:

The use of processing, packaging and shipping controls, and preoperational cleaning to preclude adverse effects of exposure to contaminants on all stainless steel materials are in accordance with Regulatory Position C.1.

Austenitic stainless steel materials are utilized in the final heat treated conditions required by the respective ASME Code, Section II, material specification for the particular type or grade of alloy in accordance with Regulatory Position C.2.

The Westinghouse position concerning material inspection programs and Regulatory Position C.3 is discussed in Section 5.2.3.4.

Westinghouse meets the intent of Regulatory Position C.4 in the manner discussed in detailed in Section 5.2. Exception (b) to Regulatory Position C.4 is covered in the discussion of delta ferrite in Section 5.2.

Westinghouse practices are in agreement with Regulatory Position C.5 in the manner discussed in Section 5.2. Exception (a) to Regulatory Postion C.5 is covered in the discussion of delta ferrite in Section 5.2.

Westinghouse practices are in agreement with Regulatory Position C.6 in the manner discussed in Section 5.2.

Balance of Plant - The extent of compliance described in Section 10.3.6.2 is applicable to all balance of plant austenitic stainless steel components.

Field Work - CP&L will comply with this guide.

FSAR Reference: Sections 5.2, and 10.3.6.

Regulatory Guide 1.45 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS (REV 0)

The SHNPP project complies with the intent of this guide as described in Section 5.2.5, except as described below.

FSAR Reference: Section 5.2.5.

Regulatory Guide 1.45 requires that all leakage detector systems be able to respond to a 1 gpm, or its equivalent, leakage increase in 1 hour or less. It also states that when analyzing the sensitivity of leak detection systems using airborne particulate or gaseous radioactivity, a realistic primary coolant radioactivity assumption should be used. Per FSAR Section 5.2.5.3.2, the gaseous and particulate airborne radiation detectors can detect a postulated step increase from 0.1 to 1.0 gpm. This assumes 85% thermal power and RCS activity based on 0.1% failed fuel.

Actual RCS activity can be much less than the assumed RCS activity with 0.1% failed fuel. When Regulatory Guide 1.45 was written in 1973, actual RCS activity at most operating plants was high enough to detect a 1 gpm increase in RCS leakage in 1 hour using particulate or gaseous radioactivity detectors. Improvements in fuel performance have greatly decreased actual RCS activities. Actual RCS activities are normally less than the assumed activity in FSAR Section 5.2.5, therefore the radiation detectors may not be able to detect a 1 gpm increase in RCS leakage within 1 hour. The particulate and gaseous radioactivity detectors meet the suggested sensitivities listed in Regulatory Guide 1.45.

Regulatory Guide 1.46 PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT (REV 0)

The SHNPP project complies with Regulatory Guide 1.46 with exception for the postulation of break points for which the criteria of BTP MEB 3-1 has been adopted. Specific break points are contained in Appendix 3.6A.

Our clarification with reference to Regulatory Position are described in the following FSAR Sections:

a) Regulatory Position C.1.a through C.1.d - See Section 3.6.2.1.1.2

b) Regulatory Position C.2.a through C.2.d - See Section 3.6.2.1.1.3

c) Regulatory Position C.3.a - See Section 3.6.2.1.5(b)

d) Regulatory Position C.3.b - See Section 3.6.2.1.5(a)

e) Regulatory Position C.4.c - See Section 3.6.1.3(e)&(f)

f) Regulatory Position C.4.d - See Section 3.6.1.2

Regulatory Guide 1.47 BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEM (REV. 0)

The manner in which SHNPP project meets the intent of this guide is described in Sections 7.3.2.2.13 and 8.3.1.2.10.

FSAR Reference: Sections 7.3.2, 7.6.2.2.m, and 8.3.1.

Regulatory Guide 1.48 DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS (REV 0)

The SHNPP project meets the intent of this guide as described below:

NSSS - Westinghouse supplied components are designed using the stress limits and loading combinations presented in Sections 3.9.1 and 5.2 for ASME Code Class 1 components and in Section 3.9.3 for ASME Code Class 2 and 3 components. The conservatism in these limits and the associated ASME design requirements precludes any component structural failure.

The operability of active ASME Code Class 1, 2, and 3 valves and active ASME Code Class 2 and 3 pumps (there are no active ASME Code Class 1 pumps) will be verified by methods detailed in Sections 3.9.1 and 5.2 for ASME Code Class 1 components and in Section 3.9.3 for ASME Code Class 2 and 3 components.

The use of the above stated methods provides an acceptable alternate method to meeting the guidance of this Regulatory Guide.

Balance of Plant-Regulatory Positions C.6a and C.6b BOP systems will not utilize ASME Code Class 2 and 3 vessels designed to ASME Section VIII, Division 1 except for the nitrogen accumulators that supply the pressurizer PORV's. These accumulators are ASME Section VIII, Division 1, but designed to Section III. They comply with regulatory positions C.6a and C.6b and are discussed in Section 9.3.1-1 and Table 3.2.1-1.

Regulatory Position C.7 - This position is not applicable to SHNPP. BOP systems at the SHNPP will not utilize ASME Code Class 2 vessels assigned to Division 2 of Section VIII of the ASME Code.

Regulatory Position C.8.a - The allowable stress for ASME Code Class 2 and 3 piping is not exceeded although the loading combinations listed in Table 3.9.3-7 are greater than those required by Regulatory Position C.8.a(1).

The emergency loading of Regulatory Position C.8.a(2) is addressed in Table 3.9.3-11.

Regulatory Position C.8.b - For the faulted loading combination, Class 2 and 3 piping designed by Ebasco will meet the stress limits provided in Table 3.9.3-11.

Regulatory Position C.10.a - The allowable stress for ASME Code Class 2 and 3 pumps is not exceeded although the loading combinations listed in Table 3.9.3-7 are greater than those required by Regulatory Position C.10.a(1), except where the pump bending stresses are insignificant when compared to the membrane stresses. However, in no case will membrane stress exceed 0.75 yield stress under these conditions.

Table 3.9.3-8 specifies an allowable stress for the emergency plant condition as required by Regulatory Position C.10.a(2). See the response to Regulatory Position C.6.a above.

Regulatory position C.10.a(3) - The SHNPP Table 3.9.3-8 meets the guidance of the Regulatory Guide Note 11 (since pump operability will be demonstrated as discussed in FSAR Section 3.9.2), except where the pump bending stresses are insignificant when compared to the membrane stresses. Therefore, for those materials where the allowable stress is limited by yield stress rather than ultimate stress, the primary membrane stress could slightly exceed the yield stress. Under these conditions, the safety function of the pump would not be impaired.

Regulatory Positions C.11 and C.12 - Class 2 and 3 system pressure and temperature design conditions are determined for normal, upset, emergency and faulted plant conditions in conjunction with specified seismic events. A valve primary pressure rating is then specified to the manufacturer which is in excess of the limiting system pressure and temperature design conditions. Therefore, the requirements of Regulatory guide 1.48 are met.

In addition, allowable valve stress limits for specified plant conditions and seismic loadings are specified to the manufacturer as indicated in FSAR Table 3.9.3-8. These allowable stresses are generally more restrictive than those presently proposed by the ASME Task Group on valves.

FSAR References: 3.9.1, 3.9.2, 5.2.

Regulatory Guide 1.49 POWER LEVELS OF NUCLEAR POWER PLANTS (REV 1)

The SHNPP project complies with this guide.

Regulatory Guide 1.50 CONTROL OF PREHEAT TEMPERATURE FOR WELDING OF LOW ALLOY STEEL (REV 0)

NSSS - Westinghouse considers that this Regulatory Guide applies to ASME Code, Section III, Class 1 components.

The Westinghouse practice for ASME Code Class 1 components is in agreement with the recommendations of Regulatory Guide 1.50, except for Regulatory Positions C.1.b and C.2. For ASME Code Class 2 and 3 components, Westinghouse does not apply any of the recommendations of Regulatory Guide 1.50.

In the case of Regulatory Position C.1.b, the welding procedures are qualified within the preheat temperature ranges required by Section IX of the ASME Code. Westinghouse experience has shown excellent quality of welds using the ASME qualification procedures.

In the case of Regulatory Position C.2, the Westinghouse position documented in Reference 1.8-4 has been found acceptable by the NRC.

Balance of Plant - Control of preheat temperatures for welding carbon and low-alloy steels in safety-related components supplied by Ebasco complies with Regulatory Guide 1.50 as follows:

1) When used, low-alloy steels will be pre-heated as required by the Regulatory Guide, unless exceptions noted within the Regulatory Guide apply (e.g., Volumetric Examinations, etc.).

2) Vendor's welding procedure specifications for carbon steels specify the preheat and interpass temperatures to be in accordance with the recommendations for ASME Code, Section III, and Article D-1000.

3) All flux-bearing filler metals are specified to be low-hydrogen type.

4) Vendors are required to store all low-hydrogen electrodes in ovens at 200-300°F for eight hours following their removal from containers and prior to use.

Field Work - CP&L complies with the requirements of Regulatory Guide 1.50 as follows:

1) Welding Procedure Specifications (WPS's) utilized at HNP for welding low alloy steels in Code class 1, 2, or 3 applications have the minimum preheat and maximum interpass temperature specified (Position C.1.a).

2) Welding Procedure Specifications (WPS's) utilized at CP&L for welding low alloy steels in Code class 1, 2, or 3 applications were qualified utilizing the minimum specified temperature as allowed by ASME Section IX (Position C.1.b).

3) CP&L does not maintain the preheat on production welds until a post-weld heat treatment has been performed. However, CP&L does wrap the preheated weld joint after completion of the welding and allows the joint to slow cool. This practice has been satisfactory in the production welding accomplished to date since we have had no welds rejected because of under bead cracking (Position C.2). CP&L performs final inspections of welded joints after completion of any required post-weld heat treatment has been performed.

4) Preheat of production welding is monitored to verify that the limits on preheat and interpass temperatures are maintained (Position C.3).

5) Welding electrodes utilized for welding low alloy steel at CP&L are specified to be low hydrogen type.

Regulatory Guide 1.51 INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS (REV 0)

This guide was withdrawn by the NRC on July 15, 1975.

Regulatory Guide 1.52 DESIGN, TESTING AND MAINTENANCE CRITERIA FOR POST ACCIDENT ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (REV. 2)

The SHNPP project meets the intent of this guide as described in Sections 6.5.1.1 through 6.5.1.4 and Section 9.4.1 with the following exceptions:

Regulatory Position	Exceptions
c.3.d	This section requires HEPA filters to be in accordance with MIL-F-51068. MIL-F-51068 has been canceled and replaced by ASME AG-1; therefore, HEPA filter requirements will be allowed to either specification.
c.4.d	This section required that each ESF atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture on the absorbers and HEPA filters. The duration of the monthly operation of each ESF atmosphere cleanup train was changed from requiring 10 continuous hours to 15 continuous minutes with implementation of License Amendment 156, which was based on NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-522, Revision 0, “Revise Ventilation System Surveillance Requirements to Operate for 10 hours per Month.

Regulatory Guide 1.53 APPLICATION OF THE SINGLE FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS (REV 0)

The SHNPP project meets the intent of this guide as described below:

NSSS

Westinghouse furnished systems meet the recommendations of this Regulatory Guide as described in Section 7.1.2.7.

Balance of Plant (BOP)

BOP furnished systems meet the recommendation of this Regulatory Guide as described in Section 7.3.2.2.2.

FSAR Reference: Section 7.6.2.2.b.

Regulatory Guide 1.54 QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS (REV. 0)

Regulatory Guide 1.54 endorses ANSI N101.4-1972. The SHNPP project complies with the requirements of ANSI N101.4-1972, as it is endorsed by this guide for protective coatings for containment surfaces (steel and concrete) and exposed surfaces of large equipment and pipe.

FSAR Reference: Section 17.3

Regulatory Guide 1.55 CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (REV 0)

The SHNPP project complies with Regulatory Guide 1.55 with the following clarification:

Section 3.8.1.2 lists the codes and standards used to establish the standards of construction for the Seismic Category I Containment Building. With the exception of part UW, which pertains to pressure vessels, these same codes and standards are utilized in the construction of all Seismic Category I buildings. In addition to the applicable codes and standards listed in Section 3.8.1.2, the SHNPP specification for concrete incorporates portions of References 2, 4, 5, and 6 of Regulatory Guide 1.55.

Embedded piping will be pressure tested in accordance with ANSI B 31.1, NFPA 24, ASME Section III or Branch Technical Position ETSB No. 11-1 R-1, as applicable, rather than ACI-318 part b. The maximum limits set by ACI-318 will not be exceeded, as these standards have more stringent requirements.

Regulatory Guide 1.56 MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (REV 1)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.57 DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (REV 0)

Regulatory Guide 1.57 is applicable to the containment penetrations only. Type I penetrations conform to Regulatory Guide 1.57, except that a fatigue analysis will not be performed since all of the piping passing through the penetration is Safety Class 2 and does not require a fatigue analysis.

FSAR Reference: Section 3.8.2

Regulatory Guide 1.58 QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION AND TESTING PERSONNEL

Conformance with Regulatory Guide 1.58 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.59, DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS (REV. 2)

The SHNPP project complies with this guide.

FSAR Reference: Section 2.4.3.

Regulatory Guide 1.60 DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (REV. 1)

The design response spectra for Westinghouse supplied equipment complies with this guide with the clarification that the damping values recommended and approved by the NRC in Reference 1.8-5 are used in the dynamic analysis.

The design response spectra for all other structures, systems and components complies with the guide with the exceptions described in Section 3.7.1.1.

FSAR Reference: Section 3.7.1.

Regulatory Guide 1.61 DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (REV. 0)

The damping values listed in Regulatory Guide 1.61 are acceptable with the single exception of the large piping systems faulted condition value of 3 percent critical. Higher damping values when justified by documented test data have been provided for in Regulatory Position C.2. A conservative value of 4 percent critical has, therefore, been justified by testing for the Westinghouse reactor coolant loop configuration in Reference 1.8-7 and has been approved by the NRC.

FSAR Reference: Section 3.7.1.

Regulatory Guide 1.62 MANUAL INITIATION OF PROTECTIVE ACTIONS (REV 0)

The SHNPP project complies with this guide as described below:

NSSS - There are individual main steam isolation valve momentary control switches (one per loop) mounted on the main control board. Each switch when actuated, will isolate one of the main steam lines. In addition, there are two master switches. Each master switch actuates all three main steam line isolation and bypass valves.

Manual initiation of switchover of safety injection from injection to recirculation is in compliance with Section 4.17 of IEEE Standard 279-1971 with the following comment.

Manual initiation of either one of two redundant safety injection actuation main control board mounted switches provides for actuation of the components required for reactor protection and mitigation of adverse consequences of the postulated accident, including delayed actuation of sequenced started emergency electrical loads as well as components providing switchover from the safety injection mode to the cold leg recirculation mode following a loss of reactor coolant accident. Therefore, once safety injection is initiated, those components of the ECCS (see Section 6.3) which are realigned as part of the semiautomatic switchover, go to completion on low refueling storage tank water level without any manual action. Manual operation of other components or manual verification of proper position as part of emergency procedures is not precluded nor otherwise in conflict with the above described compliance to Section 4.17 of IEEE Standard 279-1971 of the semiautomatic switchover circuits.

No exception to the requirements of IEEE Standard 279-1971 has been taken in the manual initiation circuit of safety injection. Although Section 4.17 of IEEE Standard 279-1971 requires that a single failure within common portions of the protective system shall not defeat the protective action by manual or automatic means, the standard does not specifically preclude the sharing of initiated circuitry logic between automatic and manual functions. It is true that the

manual safety injection initiation functions associated with one actuation train (e.g., Train A) shares portions of the automatic initiation circuitry logic of the same logic train; however, a single failure in shared functions does not defeat the protective action of the redundant actuation train (e.g., Train B). A single failure in shared functions does not defeat the protective action of the safety function. The sharing of the logic by manual and automatic initiation is consistent with the system level action requirements of the IEEE Standard 279-1971, Section 4.17 and consistent with the minimization of complexity.

Balance of Plant - The recommendations of Regulatory Guide 1.62 are complied with by the following design:

a) Manual initiation of each protective action at the system level is provided.

b) Manual initiation of a system level protective action initiates all required supporting systems.

c) Manual initiation switches are located in the Control Room and are readily accessible by the operator.

d) No single failure within the manual, automatic or common portions of the protection system can prevent initiation of the protection action by manual or automatic means.

e) Manual initiation of protective action depends on the operation of a minimum of equipment.

f) Manual initiations at the system level are designed to go to completion once initiated.

FSAR References: Sections 7.2.2.2.3, 7.6.2.2.q, and 8.3.1.2.22.

Regulatory Guide 1.63 ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR WATER - COOLED NUCLEAR POWER PLANTS (REV. 2)

The SHNPP project complies with the intent of this guide as described in Section 8.3.1.2.11.

Regulatory Guide 1.64 QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.64 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.65 MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS (REV. 0)

Westinghouse follows the recommendations of Regulatory Guide 1.65 with the following exceptions:

a) The use of modified SA-540, Grade B-24, as specified in the ASME code (Code Case 1605) is permitted by Westinghouse, but is not listed in this Regulatory Guide. Code

Case 1605 has been found acceptable to the NRC for application in the construction of components for water-cooled nuclear power plants within the limitations discussed in Regulatory Guide 1.85 which are followed by the Westinghouse practice. The use of Code Case 1605 for reactor vessel closure stud materials is not precluded by this regulatory guide.

b) A maximum ultimate tensile strength of 170,000 psi is not specified by Westinghouse, as recommended by this regulatory guide. Westinghouse does not consider this exception to be a safety issue for the following reasons:

The ASME Code requirement for toughness for reactor vessel bolting has precluded the regulatory guide's additional recommendation for tensile strength limitation, since to obtain the required toughness levels, the tensile levels are reduced.

Westinghouse has specified both 45 ft.-lb. and 25 mils lateral expansion for control of fracture toughness determined by Charpy-V testing, required by the ASME Code, Section III, Summer 1973 Addenda and 10 CFR 50, Appendix G (Paragraph IV.A.4). These toughness requirements assure optimization of the stud bolt material tempering operation with the accompanying reduction of tensile strength level when compared with previous ASME Code requirements.

Prior to 1972, the ASME Code required a 35 ft.-lb. toughness level which provided maximum tensile strength levels ranging from approximately 155 to 178 kpsi (Westinghouse review of limited data - 25 heats).

After publication of the Summer 1973 Addenda to the ASME Code and 10 CFR 50, Appendix G, wherein the toughness requirements were modified to 45 ft.-lb. with 25 mils lateral expansion, all bolt material data reviewed on Westinghouse plants showed tensile strengths of less than 170 kpsi.

The specification of both impact and maximum tensile strength as stated in the Regulatory Guide results in unnecessary hardship in procurement of material without any additional improvement in quality.

The closure stud bolting material is procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. This strength level is compatible with the fracture toughness requirements of 10 CFR 50, Appendix G (Paragraph I.C) although higher strength level bolting materials are permitted by the ASME Code.

The primary concern of the regulatory position concerning a maximum tensile strength is to minimize the susceptibility of the bolting material to stress corrosion cracking. Stress corrosion has not been observed in reactor vessel closure stud bolting manufactured from material of this strength level. Accelerated stress corrosion test data do exist for materials of 170,000 psi minimum yield strength exposed to marine water environments stressed to 75 percent of the yield strength (given in Reference 2 of this Regulatory Guide). These data are not considered applicable to Westinghouse reactor vessel closure stud bolting because of the specified yield strength differences and a less severe environment; this has been demonstrated by years of satisfactory service experience.

Additional protection against the possibility of incurring corrosion effects is assured by:

a) Decrease in level of tensile strength compatible with the requirement of fracture toughness as described above.

b) Design of the reactor vessel studs, nuts, and washers, allowing them to be completely removed during each refueling, permitting visual and/or nondestructive inspection in parallel with refueling operations to assess protection against corrosion, as part of the in-service inspection described in Section 5.2.4. If a stud sticks in place and cannot be easily removed from the vessel flange, the exposed surface above the stud hole is still available for inspection. The risk of thread corrosion is expected to be less on the threads in the hole than above the stud hole due to:

1) The confined area down the stud hole,

2) The presence of thread lubricant on the stud and stud hole threads, and

3) The lack of a mechanism for further concentrating the boric acid in the stud hole during refueling.

c) Design of the reactor vessel studs, nuts, and washers, providing protection against corrosion by allowing them to be completely removed during each refueling and placed in storage racks on the containment operating deck, in accordance with Westinghouse refueling procedures. The stud holes in the reactor vessel flange are sealed with special plugs before removing the reactor closure. In the event a stud sticks in place and cannot be easily removed from the vessel flange, enclosing the stud in a cover or "can" is an alternative means of keeping the stud dry while the cavity is flooded. Thus, the bolting materials and stud holes are not normally exposed to the borated refueling cavity water.

FSAR Reference: Section 5.3.1
PCR-6575

Regulatory Guide 1.66 NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS (REV 0)

This guide was withdrawn by the NRC on September 28, 1977.

Regulatory Guide 1.67 INSTALLATION OF OVERPRESSURE PROTECTION DEVICES (10-73)

The SHNPP project complies with this guide as discussed in Section 3.9.3.3.

Regulatory Guide 1.68 INITIAL TEST PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS (REV 2)

The SHNPP project will comply with this guide as described in Section 14.2.

Regulatory Guide 1.68.1 PREOPERATIONAL AND INITIAL STARTUP TESTING OF FEEDWATER AND CONDENSATE SYSTEMS FOR BOILING WATER REACTOR POWER PLANTS (REV 1)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.68.2 INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS (REV 1)

The SHNPP project will comply with this guide as described in Section 14.2.

Regulatory Guide 1.68.3 PREOPERATIONAL TESTING OF INSTRUMENT AND CONTROL AIR SYSTEMS (REV 0)

The SHNPP project will comply with this guide as described in Section 14.2.12 except for Regulatory Position C.10 and C.11. Regulatory Position C.10 does not apply to SHNPP because the instrument air system does not contain any single large loads that would cause a significant perturbation on the normal instrument air pressure. Therefore, a test to verify the instrument air system's response to the conditions postulated in C.10 will not be conducted. Regulatory Position C.11 does not apply to SHNPP because of the design of the Instrument Air System. To overpressurize the system would require three failures; therefore, overpressurization is not a credible failure and applicable testing will not be done.

Regulatory Guide 1.69 CONCRETE RADIATION SHIELD FOR NUCLEAR POWER PLANTS (REV. 0)

The SHNPP project complies with this guide with the exceptions and clarifications described in Section 12.3.2.4.

Regulatory Guide 1.70 STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS (REV. 3)

The SHNPP project complies with this guide except where it conflicts with the guidance of General Letter 81-06 for conformance with 10 CFR 50.71(e). In this case, the guidance in Generic Letter 81-06 will be followed.

Regulatory Guide 1.71 WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY (REV 0)

NSSS - Shop Welding - Westinghouse practice does not require qualification or requalification of welders for areas of limited accessibility as described by Regulatory Guide 1.71. Experience shows that the current Westinghouse shop practice produces high quality welds. In addition, the performance of required nondestructive evaluations provides further assurance of acceptable weld quality.

Westinghouse believes that limited accessibility qualification or requalification, which is in excess of ASME Code, Section III and IX requirements, is an unduly restrictive requirement for component fabrication, where the welders' physical position relative to the welds is controlled and does not present any significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple productions of similar components, and such welding is closely supervised.

Balance of Plant - Shop Welding - The SHNPP project complies with Regulatory Guide 1.71, with the following clarifications and exceptions:

Carolina Power & Light Company does not plan to comply with the Regulatory Positions for procured components: "C.1. The performance qualification should require testing of the welder under simulated access conditions when physical conditions restrict the welder's access to a production weld to less than 30 to 35 cm (12 to 14 in.) in any direction from the point." "C.2 Requalification is required: (a) when significantly different restricted accessibility conditions occur, . . ." Limited accessibility qualification or requalification, which exceeds ASME Section III and IX requirements, is considered an unduly restrictive requirement for shop fabrication, where the welder's physical position relative to the welds is controlled and does not present any significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised. Also, critical shop welds are nondestructively tested in accordance with ASME Section III requirements which will reveal any defective welds. The adequacy of welds is determined by the nondestructive testing of welds which is in accordance with requirements of the ASME B&PV Code. If a welder is assigned to a job which he cannot perform satisfactorily, the test results of the weld joint will not be acceptable. There is no compromise in safety in such matters.

NSSS & BOP - Field Welding - The SHNPP project complies with Regulatory Guide 1.71, with the following clarifications and exceptions:

For field application, the type of qualification should be considered on a case-by-case basis due to the great variety of circumstances encountered.

When a full penetration field weld joint in a reactor coolant pressure boundary system, Code Class 1, 2, and 3 has the accessibility restrictions outlined by Regulatory Guide 1.71 and does not require final volumetric NDE, only welders previously qualified for manual welding with a restricted accessibility welding test shall perform the welding. For instrument lines this is only applicable to larger than 3/4 in. nominal pipe size including the weld joint of the first isolation valve inside the Containment. Automatic welding shall be considered exempt from Regulatory Guide 1.71 since limitations are inherent to the welding process for inaccessibility of locating the welding apparatus within the confinements of the weld joint. Special qualification for limited accessibility as defined in Regulatory Guide 1.71 and this paragraph shall be required for a welder who is physically limited in his ability to perform field welding of a joint because of barriers (walls, hangers, adjacent piping equipment, etc.) as close as 12 in. or less and encroaching into the envelope of space normally used by the welder when making a similar weld without such restrictions. The necessity for use of a mirror by the welder would be considered as a requirement for special welder qualification. Requalification by a different special welder qualification will be required when significantly different access condition occurs other than those originally required by the previous special welder qualification (example, addition of a mirror, significant physical changes in welders access, etc.).

Regulatory Guide 1.72 SPRAY POND PIPING MADE FROM FIBERGLASS-REINFORCED THERMOSETTING RESIN (REV 2)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.73 QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS (REV 0)

The SHNPP project complies with this guide as described below:

NSSS - For safety-related motor operated valves located inside Containment, environmental qualification is performed in accordance with IEEE Standard 382-1972. Auxiliary safety-related equipment (e.g., stem mounted limit switches) is qualified separately. Qualification conditions (temperature, pressure, radiation and chemistry) are those specified in Part III of IEEE Standard 382-1972 for pressurized water reactor applications. Since there are no exposed organic materials, consideration of beta radiation is not required.

BOP - Electric valve operators installed inside the Containment for the SHNPP project comply with this guide.

FSAR References: Sections 3.10, 3.11, 8.3.1.

Regulatory Guide 1.74 QUALITY ASSURANCE TERMS AND DEFINITIONS

Conformance with Regulatory Guide 1.74 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.76 DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (REV.0)

The SHNPP project complies with this guide (with the exception described below).

Regulatory Guide 1.76 Revision 1 was issued for use in March 2007. This regulatory guide provides licensees and applicants with new guidance that the staff of the NRC considers acceptable for use in selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear plant should be designed to withstand. This guidance divides the United States into three regions: the Harris Nuclear Plant is located in Region 1. The NRC staff accepts the methods described in Regulatory Guide 1.76 Revision 1 to evaluate submittals from operating reactor licensees after March 2007 who voluntarily propose to initiate system modifications that have a clear nexus with the guidance provided. No backfitting is intended or approved in conjunction with its issuance. The Harris Nuclear Plant adopts the guidance provided in Regulatory Guide 1.76 Revision 1 as an optional design basis for new system modifications occurring after March 2007.

FSAR Reference: Section 3.3.2.1.

Regulatory Guide 1.77 ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURE AND WATER REACTORS (REV. 0)

The SHNPP project complies with the guide with the exception described below:

Westinghouse methods and criteria are documented in Reference 1.8-8 which has been reviewed and accepted by the NRC.

The results of the Westinghouse analyses show agreement with Regulatory Positions C.1 and C.3. In addition, Westinghouse utilizes the assumptions given in Appendices A and B of the Regulatory Guide. However, Westinghouse takes exception to Regulatory Position C.2 which implies that the rod ejection accident should be considered as an emergency condition.

Westinghouse considers this a faulted condition as stated in ANSI N18.2. Faulted condition stress limits will be applied for this accident.

FSAR Reference: Section 15.4.8.

Regulatory Guide 1.78 EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (REV 1)

The SHNPP project complies with this guide.

FSAR Reference: Section 6.4 and Section 2.2.3.3.

Regulatory Guide 1.79 PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS (REV 1)

The SHNPP project will comply with this Regulatory Guide with the exception as noted in Section 14.2.7.

Regulatory Guide 1.80 PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS (REV 0)

Regulatory Guide 1.80 is superseded by Regulatory Guide 1.68.3.

Regulatory Guide 1.81 SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS (REV. 1)

Regulatory Guide 1.81 is not applicable to the SHNPP.

Regulatory Guide 1.82 WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT (REV. 3)

The SHNPP project satisfies the intent of this guide as described in the referenced FSAR Sections.

FSAR Reference: Sections 6.2.2.2, and 6.5.2.

Regulatory Guide 1.83 INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES (REV 1)

The SHNPP project complies with the recommendations of this guide. Westinghouse steam generators are designed to permit access to tubes for inspection and/or plugging. The inservice inspection program is discussed in the Technical Specifications.

Regulatory Guide 1.84 CODE CASE ACCEPTABILITY-ASME III DESIGN AND FABRICATION

The SHNPP project complies with this guide as described below:

<u>NSSS</u>

a) Westinghouse controls its suppliers to:

1) Limit the use of code cases to those listed in Regulatory Position C.1 of the Regulatory Guide 1.84 revision in effect at the time the equipment is ordered, except as allowed in item b) below.

2) Identify and request permission for use of any code cases as listed in Regulatory Position C.1 of the Regulatory Guide 1.84 revision in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.

3) Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

b) Westinghouse seeks NRC permission for the use of Class 1 code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the Regulatory Guide 1.84 revision in effect at the time the equipment is ordered and permits supplier use only if NRC permission is obtained or is otherwise assured (e.g., a later version of the Regulatory Guide includes endorsement).

Balance of Plant

The SHNPP project complies with this guide.

Regulatory Guide 1.85 CODE CASE ACCEPTABILITY-ASME III MATERIALS

The SHNPP project complies with this guide as described below:

NSSS

a) Westinghouse controls its suppliers to:

1) Limit the use of code cases to those listed in Regulatory Position C.1 of the Regulatory Guide 1.85 revision in effect at the time the equipment is ordered, except as allowed in item b) below.

2) Identify and request permission for use of any code cases as listed in Regulatory Position C.1 of the Regulatory Guide 1.85 revision in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.

3) Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

b) Westinghouse seeks NRC permission for the use of Class 1 code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the Regulatory Guide 1.85 revision in effect at the time the equipment is ordered and permits supplier use only if NRC permission is obtained or is otherwise assured (e.g., a later version of the Regulatory Guide includes endorsement).

Balance of Plant

The SHNPP project complies with this guide.

Regulatory Guide 1.86 TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS (REV 0)

The SHNPP project will comply with this guide.

Regulatory Guide 1.87 GUIDANCE FOR CONSTRUCTION OF CLASS 1 COMPONENTS IN ELEVATED-TEMPERATURE REACTORS (SUPPLEMENT TO ASME SECTION III CODE CASES 1592, 1593, 1594, 1595 AND 1596) (REV 1)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.88 COLLECTION, STORAGE AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS

Conformance with Regulatory Guide 1.88 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.89 QUALIFICATION OF CLASS 1E EQUIPMENT FOR NUCLEAR POWER PLANTS (REV. 0)

The SHNPP project meets the intent of the guide as described below:

NSSS - The Westinghouse approach to satisfying the guidelines of Regulatory Guide 1.89 and IEEE Standard 323-1974 is documented in Reference 1.8-9. The Westinghouse approach to satisfying the guidelines of IEEE Standard 323-1971 is documented in WCAP-7410-L, WCAP-7709-L, and Westinghouse Supplemental Environmental Qualification Testing Program (Re: Westinghouse letter NS-CE-692, Eicheldinger to Vassallo, July 10, 1975, and NRC letter Vassallo to Eicheldinger, November 9, 1975).

Balance of Plant - Class 1E electrical equipment supplied by Ebasco is qualified by the equipment manufacturer by utilizing analysis of type testing. Such qualifications incorporate the methods prescribed in IEEE-323-74, IEEE-334-71, and IEEE-317-76. Purchase specifications indicate that type testing is the preferred method of qualification. Ebasco supplied valve operators are qualified in accordance with IEEE-382-72 and Regulatory Guide 1.73.

The Emergency Diesel Generating System was qualified according to the requirements of Regulatory Guide 1.89 and IEEE Standard 323-1974. Replacement parts for the Emergency Diesel Generating System may be procured to the requirements of IEEE Standard 323-1983.

FSAR Reference: Sections 3.10 and 3.11.

Regulatory Guide 1.90 INSERVICE INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS (REV 1)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.91 EVALUATION OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANTS (REV 1)

The SHNPP project meets the intent of this guide as described in FSAR Section 2.2.3.

Regulatory Guide 1.92 COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS (REV 1)

The SHNPP project meets the intent of this guide as described below:

NSSS

The Westinghouse procedure for combining modal response is presented in Section 3.7.2.7B

Balance of Plant

The BOP design of the SHNPP meets the intent of this Regulatory Guide as described in FSAR Sections 3.7.2.1 and 3.7.3.7.

FSAR Reference: Appendix 3.9A

Regulatory Guide 1.93 AVAILABILITY OF ELECTRIC POWER SOURCES (REV 0)

The SHNPP project will comply with this guide with the following clarification: Section C of the guide permits electric power sources to be removed from service for corrective maintenance activities only. As permitted by Technical Specifications, electric power sources may be removed from service for other reasons under administrative controls provided these are restored to service within the allowed out of service time intervals.

FSAR References: Technical Specifications Subsection 3/4.8.

Regulatory Guide 1.94 QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.94 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR in Chapter 17 (see Section 17.3)

Regulatory Guide 1.97 INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT (REV 3)

The guidance of Regulatory Guide 1.97 (Revision 3) has been implemented along with that of NUREG 0737 Supplement 1, Section 6.2, in establishing the guidelines for variables to be monitored.

The compliance with these documents is detailed in a response to NRC ICSB Question 44 via CP&L letter LAP-83-405 dated September 6, 1983, CP&L letter NLS-85-109 dated June 3, 1985, and CP&L letter NLS-88-279 dated December 22, 1988.

Regulatory Guide 1.98 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFFGAS SYSTEM FAILURE IN A BOILING WATER REACTOR (REV 0)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.99 RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS (REV. 2)

The SHNPP project complies with the intent of Regulatory Guide 1.99 with the following clarification.

The results of the analysis of the third reactor vessel surveillance capsule indicated that the scatter in the shift of reference temperature - nil ductility transition (ΔRT_{NDT}) for two of the six plate surveillance specimens exceeded the criteria for "credibility" as defined by the regulatory guide. The weld surveillance data meets the credibility requirements of the regulatory guide. When surveillance data is "credible", i.e., without excessive scatter, the regulatory guide permits calculation of chemistry factors from the surveillance data, and halving of required margins applicable to reactor vessel pressure-temperature limits. Otherwise, chemistry factors are based on generic data from the regulatory guide, and full margins are to be used. SHNPP still uses the plate surveillance data to calculate chemistry factors, but has assumed the full margin terms in developing reactor vessel pressure-temperature limits.

FSAR Reference: Sections 5.3.1, 5.3.2

Regulatory Guide 1.100 SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS (REV 1)

The SHNPP project complies with this guide as described below:

The Westinghouse program for seismic qualification of safety-related electrical equipment to Regulatory Guide 1.100 is delineated in the latest revision of WCAP-8587 "Methodology for Qualifying Westinghouse PWR-SD Supplied NSSS Safety-Related Electrical Equipment," together with Supplement 1 to this report. In summary, seismic qualification for all Westinghouse supplied equipment will be demonstrated by the following methods:

a) For equipment not subject to high energy line break conditions which has been previously qualified, as identified in Supplement 1 to WCAP-8587, using the methods permitted by the 1971 version of IEEE Standard 344 (i.e., single axis sine-beat testing or analysis, after demonstration of no resonant frequency below 33Hz), no additional seismic qualification will be specified provided that:

    1) It can be shown, by separate component testing and/or analysis, that there are no aging mechanisms that could prejudice the previously completed seismic qualification.

2) Any design modifications made to the equipment do not significantly affect the seismic characteristics of the equipment.

3) The adequacy of the original seismic test levels can be demonstrated as conservative by plant specific verification.

b) For new equipment, or equipment that cannot meet the provisions of a) above, seismic qualification will be performed in accordance with IEEE Standard 344-75. The method to be employed (i.e., test and/or analysis) is indicated, for the safety-related equipment in the Westinghouse PWRSD Scope of Supply, in Supplement 1 to WCAP-8587. Where multifrequency biaxial inputs are employed for testing, the methodology described in WCAP-8695, "General Method of Developing Multifrequency Biaxial Test Inputs for Bistables," will be employed. When flexible equipment size and weight precludes biaxial testing, single axis testing with justification will be utilized to meet IEEE Standard 344-1975. For rigid equipment (i.e., no resonant frequency below 33Hz), qualification may be by analysis in accordance with IEEE Standard 344-1975.

All non-NSSS supplied equipment and supports are in compliance with the qualification and documentation of IEEE 344-75.

Regulatory Guide 1.101 EMERGENCY PLANNING FOR NUCLEAR POWER PLANTS (REV 1)

SHNPP does not comply with this Regulatory Guide.

Regulatory Guide 1.102 FLOOD PROTECTION FOR NUCLEAR POWER PLANTS (REV 1)

Shearon Harris Nuclear Power Plant complies with the Regulatory Guide 1.102 Rev. 1 with the following exceptions.

a) The plant grade at Elevation 260 ft. is higher than the maximum water level in the Auxiliary Reservoir or in the Main Reservoir as discussed in Sections 2.4.5 and 3.4.1. The main access road on the east side of the plant, which is constructed on an embankment through the Main Reservoir, is not classified as safety-related.

b) The east and south slopes of the plant site will be subject to water waves in the Main Reservoir. These slopes have been protected by sacrificial spoil fill as discussed in Section 2.4.3.

c) As discussed in Section 2.4.13, the groundwater level at the plant site will not exceed Elevation 251 ft. and the plant structures have been designed for the hydrostatic and the buoyant forces corresponding to this groundwater level.

   Ponding of storm water in the plant yard, as discussed in Section 2.4.2, due to the probable maximum precipitation (PMP) is not expected to change the groundwater table and is also not expected to cause any additional hydrostatic and buoyant forces on the structures because of the selected impervious backfill placed around the structures. For conservatism the stability of the structures against floatation, as discussed in Section 3.8.4, has been checked for water at the plant grade (Elevation 260 ft.).

d) The walls of safety-related structures, though not designed for the static and dynamic forces due to the ponded water in the yard, have adequate reserve strength to resist these forces. For the reason stated in c) above the walls of the plant structures have not been designed for static and dynamic forces due to the ponded water in the yard.

e) The roofs of safety-related structures have not been designed for standing water on the roofs up to the top of curbs; however the water will be drained off through roof drains.

Regulatory Guide 1.103 POST-TENSIONED PRESTRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS (REV 1)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.104 OVERHEAD CRANE HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS (REV 0)

This guide was withdrawn by the NRC on August 22, 1979.

Regulatory Guide 1.105 INSTRUMENT SETPOINTS (REV 1)

The SHNPP project meets the intent of this guide as described below:

NSSS

Technical Specifications provide the margin from the nominal setpoint to the technical specification limit to account for drift when measured at the rack during periodic testing. The allowances between the technical specification limit and the safety limit include a statistical combination of the following items: a) the inaccuracy of the instrument, 2) process measurement accuracy, 3) uncertainties in the calibration, 4) the potential transient overshoot determined in the accident analyses (this may include compensation for the dynamic effect), and 5) environmental effects on equipment accuracy caused by postulated or limiting postulated events (only for those systems required to mitigate consequences of an accident). Westinghouse designers choose setpoints such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis.

The range of instruments is chosen based on the span necessary for the instrument's function. Narrow range instruments are used where necessary. Instruments are selected based on expected environmental and accident conditions. The need for qualification testing is evaluated and justified on a case-by-case basis.

Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint securing device is required. Integral setpoint locking devices are supplied.

The assumptions used in selecting the setpoint values in Regulatory Position C.1 and the minimum margin with respect to the technical specification limit and calibration uncertainty is documented. Drift rates and their relationship to testing intervals is not documented.

Balance of Plant

Discussion of compliance is included below:

a) Setpoints are established with margins between technical specification limits for the process variable and the nominal trip setpoint which include allowance for instrument inaccuracy, calibration uncertainty, instrument drift anticipated between calibration intervals.

b) All instrument ranges are selected to ensure that the portion required for the setpoint is within the portion of the instrument ranges that yields the maximum accuracy and maintainability.

c) The ranges selected for the instrumentation fully encompass the expected operating range of the monitored variables and the selected range is always well within the saturation limits of the instrument.

d) The accuracy of all setpoints is equal to or better than the accuracy assumed in the safety analysis. Instrument intervals are chosen for the design conditions in which they are installed in order not to anneal, stress relieve, or work harden to the extent that they will not maintain the required accuracy. Design verification is included as part of the equipment qualification program as recommended in Regulatory Guide 1.89.

e) Instruments important to safety have securing devices on the setpoint adjustment mechanism and/or are under administrative control. The securing device is designed such that during securing or releasing it will not alter the setpoint.

f) Documentation of methodology and assumptions used in selecting setpoint values and minimum margins, drift rates and test intervals is contained in plant setpoint determination procedures/documents.

g) Safety related setpoints not covered by technical specification have sufficient documentation to support the setpoint value, tolerance, and margin to system process limits.

Regulatory Guide 1.106 THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES (REV. 1)

The SHNPP project meets the intent of this guide as described in Section 8.3.1.2.18.

Regulatory Guide 1.107 QUALIFICATIONS FOR CEMENT GROUTING FOR PRESTRESSING TENDONS IN CONTAINMENT STRUCTURES (REV 1)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.108 PERIODIC TESTING OF DIESEL GENERATORS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS (REV. 1)

Preoperational and periodic testing of the SHNPP diesel generators will comply with this guide as described in the referenced FSAR sections except as discussed below:

FSAR Reference: Section 8.3.1.1.2.14, 14.2.12.1.16, and 3.1.14.

RG 1.108, paragraph C.2.a(5) requires that design accident loading sequence be performed immediately after the 24-hour load run. The requirements of this paragraph will not be fulfilled immediately after the 24-hour load run. Instead, this test will be performed in conjunction with the Integrated Engineered Safety Features Actuation System Test. The Diesel Engine will be operated at full load conditions to reestablish full load temperature conditions. A loss of all A.C. voltage will then be simulated to demonstrate that the diesel generator unit can start automatically and attain required voltage and frequency. Also, proper operation for the design-accident-loading-sequence to design-load requirements while maintaining voltage and frequency within limits will be demonstrated as required. This will provide for accomplishment of 24-hour full load carrying capability demonstration as soon as the Emergency Diesel Generator Systems are ready.

RG 1.108, paragraph C.2.d defines the periodic test interval for the emergency diesel generators. The requirements of this section will be fulfilled, as required by FSAR Section 3.1.14, by performing the periodic tests in accordance with the schedule provided in Technical Specification Section 3/4.8.1.

Regulatory Guide 1.109 CALCULATIONS OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10 CFR 50, APPENDIX I (REV. 1)

The SHNPP project complies with this guide.

FSAR Reference: Section 11.2.3, 11.3.3.

Regulatory Guide 1.110 COST-BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS (REV 0)

The SHNPP project is not required to address this guide.

Regulatory Guide 1.111 METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASES FROM LIGHT-WATER-COOLED REACTORS (REV 1)

The SHNPP project complies with this guide.

FSAR Reference Section 2.3.4.

Regulatory Guide 1.112 CALCULATIONS OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT WATER COOLED REACTORS (REV. 0)

The SHNPP project complies with this guide.

FSAR Reference: Section 11.2.3, 11.3.3.

Regulatory Guide 1.113 ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I (REV. 1)

The SHNPP project complies with this guide and is discussed in the SHNPP Operating License Environmental Report, Section 5.2.2.1.

Regulatory Guide 1.114 GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT (REV. 1)

The SHNPP project complies with this guide.

FSAR Reference: Chapters 13 and 14.

Regulatory Guide 1.115 PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES (REV 1)

The SHNPP project complies with the intent of this guide in that due to the limited exposure of vital equipment and the high degree of barrier protection provided, and as stated in the NRC SER Supplement No. 3, dated July 1977, low trajectory turbine missiles will not be a significant threat to the plant.

FSAR Reference: Section 3.5.1

Regulatory Guide 1.116 QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS

Conformance with Regulatory Guide 1.116 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.117 TORNADO DESIGN CLASSIFICATION (REV. 1)

The SHNPP project complies with this guide.

FSAR Reference: Sections 3.3 and 3.5.1.4.

Regulatory Guide 1.118 PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS (REV 2)

Except as noted below, SHNPP complies with IEEE Standard 338-1977, Reg. Guide 1.118. With these exceptions, the project still meets IEEE 338-1971 as discussed in Section 7.1.2.18 and IEEE 338-1975 as discussed in Section 8.3.1.2.27.

<u>NSSS</u>

Westinghouse will make clear distinction between recommendations and requirements when addressing criteria. Detailed positions on the Regulatory Positions are presented below:

a) Regulatory Position C.1 - Westinghouse will provide a means to facilitate response time testing from the sensor input at the protection rack to and including the input to the actuation device. Examples of actuation devices are the protection system relay or bistable.

   Westinghouse defines "Protective Action Systems" to mean the electric, instrumentation and controls portions of those protection systems and equipment actuated and controlled by the protection system.

   Equipment performing control functions, but actuated from protection system sensors is not part of the safety system and will not be tested for time response.

   Status, annunciating, display, and monitoring functions, except for those related to the Post Accident Monitoring System (PAMS), are considered to be control functions. Reasonability checks, i.e., comparison between or among similar such display functions, will be made. Otherwise, the clarification note in Position C.1 is observed.

b) Regulatory Position C.5 - Response time testing for control functions operated from protection system sensors will not be performed. Nuclear instrumentation system detectors will not be tested for time response. The "expected environmental and mechanical configuration of the actual installation" will not be duplicated for the testing of sensors which must be removed to accomplish response time testing unless it can be shown that the duplication is practical and that the duplicated factors significantly influence the sensor time response.

   The Westinghouse scope protection system does not preclude the response time testing of process sensors by their removal at normal shutdown. The standard Westinghouse scope protection system does not include design provision which permit insitu testing of process or nuclear instrumentation system sensors. Nuclear instrumentation sensors are exempt from testing since "their worst case response time is not a significant fraction of the total overall system response (less than 5 percent)." This exemption is permitted by IEEE-338-1975.

c) Regulatory Position C.6 - Temporary jumper wires, temporary test instrumentation, the removal of fuses and other equipment not hard-wired into the protection system will be used where applicable.

d) IEEE Standard 338-1977, Section 5.7 states "Each test bypass condition utilized at a frequency of more than once a year shall be individually and automatically indicated to the operator in the main control room in such a manner that the bypassing of a protective function is immediately evident and continuously indicated."

   SHNPP takes exception to this requirement during performance of containment ventilation isolation from the containment ventilation isolation radiation monitors. During monthly testing of containment ventilation isolation, the opposite train of containment ventilation isolation is bypassed to ensure that both trains of containment vacuum relief are not made inoperable. This test is performed from the main control room and operators are made aware of the bypassed containment ventilation isolation from the radiation monitors.

Balance of Plant - The SHNPP project complies with this guide.

FSAR References: Sections 7.1.2.17, 7.3.2.2.10, 7.6.2.2.j, 8.3.1.2.20, and 13.5.1.3.e.

Regulatory Guide 1.119 SURVEILLANCE PROGRAM FOR NEW FUEL ASSEMBLY DESIGNS (REV 0)

This guide was withdrawn by the NRC on June 23, 1977.

Regulatory Guide 1.120 FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS (REV 1)

This guide is not applicable to the SHNPP project. For a discussion of fire protection requirements, see Section 9.5.1.

Regulatory Guide 1.121 BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES (REV 0)

The SHNPP project complies with this guide.

Regulatory Guide 1.122 DEVELOPMENT OF FLOOR DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF FLOOR-SUPPORTED EQUIPMENT OR COMPONENTS (REV 1)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.123 QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OR PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.123 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.124 SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS 1 LINEAR-TYPE COMPONENT SUPPORTS (REV 1)

The SHNPP project complies with the intent of this guide as described below:

<u>NSSS</u>

The Regulatory Guide states in paragraph B.1(b): "Allowable design limits for bolted connections are derived from tensile and shear stress limits and their non-linear interaction; they also change with the size of the bolt. For this reason, the increases permitted by NF-3231-1, XVII-2110(a), and F-1370(a) of Section III are not directly applicable to allowable shear stresses and allowable stresses for bolts and bolted connections.", and in paragraph C.4: "This increase of design limits does not apply to limits for bolted connections and shear stresses."

As noted above, the increase in bolt allowable stress under emergency and faulted conditions is not permitted because: 1) the interaction between the allowable tension and shear stress in

bolts is nonlinear, 2) the allowable tension and shear stress vary with the bolt size. Westinghouse believes that the present ASME Code rules are adequate since they do satisfy the two objectives raised in the above quoted paragraph and hence will use the present rules without further restrictions or justification. This position is based on the following:

It is well recognized after extensive experimental work by several researchers that the interaction curve between the shear and tension stress in bolts is more closely represented by an ellipse and not a line. This has been clearly recognized by the ASME. The latest revision of Code Case 1644 specifies stress limits for bolts and represents this tension/shear relationship as a non-linear interaction equation (ellipse). This interaction equation has a built-in safety factor that ranges between 2 and 3 (depending on whether the bolt load is predominantly tension or shear) based on the actual strength of the bolt as determined by test (Reference: "Guide to Design Criteria for Bolted and Riveted Joints," Fisher and Struik, copyright 1974, John Wiley and Sons, P. 54).

<u>Balance of Plant</u>

Regulatory Guide Position C is a supplement to Subsection NF of the ASME Code, Section III, service limits. The balance of plant design is not bound by Subsection NF; but is based on MSS-SP-58, "Pipe Hangers and Supports - Materials, Design and Manufacturer," 1975 Edition.

Regulatory Guide 1.125 PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR POWER PLANTS (REV 0)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.126 AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION (REV 1)

Westinghouse uses the fuel densification model presented in Reference 1.8-12 which has been approved by the NRC.

Siemens uses the fuel densification model presented in Reference 1.8-15 which has been approved by the NRC.

Regulatory Guide 1.127 INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS (REV 1)

The SHNPP project complies with this guide.

Regulatory Guide 1.128 INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (REV 1)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.129 MAINTENANCE, TESTING AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (REV 1)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.130 SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS 1 PLATE-AND-SHELL TYPE COMPONENT SUPPORTS (REV 1)

The SHNPP project meets the intent of the guide as described below:

NSSS

a) The Regulatory Guide states in Paragraph B.1: - "Allowable design limits for bolted connections are derived on a different basis that varies with the size of the bolt. For this reason, the increases permitted by NF-3224 and F-1323.1(a) of Section III are not directly applicable to bolts and bolted connections."

   It is the Westinghouse position that it is reasonable to allow an increase in the limits for bolted connections for emergency and faulted conditions. Further justification of this position can be found in the discussion of Regulatory Guide 1.124 on Class 1 linear type supports.

b) Paragraphs C.3, C.4(a), and C.6(a) of the Regulatory Guide state that the allowable buckling strength should be calculated using a design margin of 2 for flat plates and 3 for shells for normal, upset and emergency conditions.

   In the design of plate-type supports, member compressive axial loads shall be limited per the requirements of Paragraph C.3 for normal, upset and emergency conditions. There are no Class I shell-type supports in the Westinghouse NSSS.

c) In Paragraph C.7 of the Regulatory Guide, inclusion of the upset plant condition is inappropriate in the load combination under discussion. Westinghouse does not include the upset plant condition in this combination.

d) In Paragraphs C.7(a) and B.1 of the Regulatory Guide, the stress limits of F-1370(c) are discussed. The criterion stated in F-1370(c), "...loads should not exceed 0.67 times the critical buckling strength of the support...."

   In the design of plate-type component supports, member compressive axial loads shall be limited to 0.67 times the critical buckling strength. If, as a result of a more detailed evaluation of the supports, the member compressive axial loads is shown to safety exceed 0.67 times the critical buckling for the faulted condition, verification of the support function adequacy is documented and submitted to the NRC for review. The member compressive axial loads will not exceed 0.67 times the critical buckling strength without NRC acceptance.

e) In Paragraph C.7(b) of the Regulatory Guide, the limit based on the test load given in the Regulatory Guide, T.L. x 0.7 S'_u /$S_{u,}$ is overly conservative and is inconsistent with ASME Code requirements presented in Appendix F.

   Westinghouse uses the provisions of F-1370(d) to determine service level D allowable loads for supports designed by the load rating method.

Study of three interaction curves of allowable tension and shear stress based on the ASME Code (emergency condition allowables per XVII-2110 and faulted condition allowables per F-1370) and the ultimate tensile and shear strength of bolts (obtained from experimental work published by Chesson, Faustina, and Munse in "Proceedings of ASCE", October 1975) indicates that there is adequate safety margin between the emergency and faulted condition allowables and failure of the bolts.

From this study it is observed that:

1) For the emergency condition, the safety factor (ratio of ultimate strength to allowable stress) varies between a minimum of 1.63 and a maximum of 2.73 depending upon the actual tensile stress/shear stress (T/S) ratio on the bolt.

2) For the faulted condition, the safety factor varies between a minimum of 1.36 to a maximum of 2.29, again depending upon actual T/S ratio on the bolt.

   It is thus reasonable to allow an increase in these limits for the emergency and faulted conditions.

3) In Section III Subsection NA Table XVII-2461.1 the ASME Code provides the criterion for allowable stress on bolts. As per this table, the allowable stress depends upon the bolt size as well as the bolt material.

4) The structures designed to meet AISC Manual of Steel Construction have been proven to be adequately designed. It is also recognized that the ASME Code requirements for Class 1 linear type supports (Appendix XVII) have been derived from AISC Manual of Steel Construction. In Paragraph 1.5.6 of Manual of Steel Construction, AISC permits the increased allowables for "occasional loads such as wind and seismic". In view of this, restrictions by NRC on not permitting increased allowables under emergency and faulted conditions which also are "infrequent incidents and limiting faults" are not justified.

Based on item 1 through 4 above, for the emergency and faulted conditions, Westinghouse uses allowable bolt stresses specified in Code Case 1644, latest revision, as increased according to the provisions of XVII-2110(a) and F-1370(a), respectively. Westinghouse uses the latest revision of Code Case 1644 as opposed to Revision 4 specified in the Regulatory Guide, Paragraph B.I(a).

Concerning allowable shear stresses, a commonly used allowable shear stress for structural members is 0.55 F_y (F_y = yield stress). This limit is well documented in the literature and substantiated by numerous tests (see, for instance, the AISC Manual for Steel Construction, Part 2). This limit not only maintains the member load below the yield load, but keeps it well below the actual shear strength of the member. In addition, the moment carrying capacity of a structural member is not appreciably affected for shear loads corresponding to 0.55 F_y.

Westinghouse uses the following criteria to make the allowable shear stress compatible with other allowable emergency and faulted condition stresses, and at the same time, keep shear stresses within proven limits: For the emergency condition, shear stresses may be increased according to the provisions of XVII-2110(a). Faulted condition

allowable shear stresses should be limited to the lesser of 0.55 F_y or 0.45 S_u, where S_u is the material tensile strength.

f) In Paragraphs B.5 and C.8 of the Regulatory Guide, Westinghouse takes exception to the requirement that systems whose safety-related function occurs during emergency or faulted plant conditions must meet upset limits. The reduction of allowable stresses to no greater than upset limits (which in reality are only design limits since design, normal and upset limits are the same for linear supports) for support structures in those systems with normal safety-related functions occurring during emergency or faulted plant conditions is overly conservative for components which are not required to mechanically function (inactive components). In addition, Westinghouse believes that emergency and faulted condition criteria are acceptable for active components. However, when these criteria are invoked for active components, any significant deformation that might occur is considered in the evaluation of equipment operability.

g) Paragraph C.4 of the Regulatory Guide states: "However, all increases (i.e, those allowed by NF-3231.1(a), XVII-2110(a), and F-1370(a) should always be limited by XVII-2110(b) of Section III." Paragraph XVII-2110(b) specifies that member compressive axial loads shall be limited to 2/3 of critical buckling. Satisfaction of these criteria for the faulted condition is unnecessarily restrictive.

   The most significant faulted condition loads on equipment supports result from seismic disturbances and postulated loss-of-coolant accidents, both of which are dynamic events. The allowable faulted condition compressive load should not be limited to 2/3 of critical buckling because: (a) these faulted dynamic loads are of extremely short duration, and (b) support members can take impulsive loads that exceed static critical buckling load. Westinghouse will use a compressive axial load of 0.9 of critical buckling since the dynamic buckling capacity of the member is greater than the static buckling capacity.

h) Paragraph C.2 of the Regulatory Guide presents two methods of estimating the ultimate tensile strength, S_u, at elevated temperatures. It is believed that Method #2 is not conservative at elevated metal temperature (in excess of 800°F). In Westinghouse's judgment, values of S_u at these elevated temperatures should be determined by test rather than via the method given in C.2(b).

i) Paragraph C.6(a) of the Regulatory Guide appears to erroneously allow the use of faulted stress limits for the emergency condition. Westinghouse will interpret this paragraph as follows: "The stress limits of XVII-2000 of Section III and Regulatory Position 3, increased according to the provisions of XVII-2100(a) of Section III, should not be exceeded for component supports designed by the linear elastic analysis method."

j) Westinghouse uses the provisions of F-1370(d) to determine faulted condition allowable loads for supports designed by the load rating method. The method described in Paragraph C.7(b) of the Regulatory Guide is very conservative and inconsistent with the remainder of the faulted stress limit.

Balance of Plant

Regulatory Guide Position C is a supplement to Section 3, Subsection NF of the ASME Code, Section III. The balance of plant design is not bound by Subsection NF, but is based on MSS-SP-58, "Pipe Hangers and Supports - Material Design and Manufacturers," 1975 Edition.

Regulatory Guide 1.131 QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (REV 0)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.132 SITE INVESTIGATIONS FOR FOUNDATIONS OF NUCLEAR POWER PLANTS (REV 0)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.133 LOOSE-PART DETECTION PROGRAM FOR THE PRIMARY SYSTEM OF LIGHT-WATER-COOLED REACTORS

Carolina Power & Light Company concurs with the desirability of implementing a loose parts detection program in the SHNPP. Experience has shown the system's merits, functional adequacy and reliable performance in operating plants. Operating history has proven that there can be early warning benefits from experienced interpretation of system data. However, CP&L takes exception to requirements which go beyond the need for a reliable system which provides anything more than this basic reassurance. These exceptions are delineated as follows:

a) The functional performance requirements for this system serve solely to provide an alert, by impact monitoring, for circumstances which could result from loose metallic objects in the Reactor Coolant System. The system (a) does not maintain the reactor coolant pressure boundary (b) serves no automatic reactor protection function, and (c) does not classify as an IE system as defined in IEEE-308. Consequently, the application of Class IE criteria is unjustified.

b) Based on exception a) above, the requirements of Regulatory Guide 1.100 "Seismic Qualification of Electric Equipment for Nuclear Power Plants" are not applicable.

c) Based on exception a) above, the redundancy and separation requirements for the system are not applicable, except for in-containment hardware where good engineering practice prevails.

d) The use of system noise to provide functional tests is adequate. The need for calibrated simulated signals or in-containment calibration is unduly restrictive and unnecessary. Impact energy sensitivity is verifiable from system noise signatures.

e) Based on exception a) above, the limiting condition for operation imposed by Regulatory Position C.5.b is not applicable. The availability of a non-Class IE system is not essential for continued safe operation.

f) The implementation of the method described in Regulatory Guide 1.133 Section D is a backfit as defined by 10 CFR 50.109. Based on exception a) above, the backfit requirement cannot be justified.

g) Carolina Power & Light Company also takes exception to technical requirements not enumerated here but which include, as an example, the restriction of the background noise level of the primary system in a nuclear plant to 20 percent of an arbitrary system sensitivity level simultaneously established by the guide.

Regulatory Guide 1.134 MEDICAL CERTIFICATION AND MONITORING OF PERSONNEL REQUIRING OPERATOR LICENSES (REV 2)

The SHNPP project complies with this guide.

Regulatory Guide 1.135 NORMAL WATER LEVEL AND DISCHARGE AT NUCLEAR POWER PLANTS (REV 0)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.136 MATERIAL FOR CONCRETE CONTAINMENTS (REV 0)

This guide is not applicable to the SHNPP project.

Regulatory Guide 1.137 FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS (REV 1)

Carolina Power & Light Company ensures the quality and reliability of the diesel generator fuel oil by implementing the diesel generator fuel oil surveillance requirements described in the SHNPP Technical Specifications.

Carolina Power & Light Company uses the sampling and testing methods recommended in RG 1.137 with the following exceptions:

a) Instead of the standard recommended in Part C.2.a of RG 1.137 (ASTM D975-77), CP&L will use methods described in ASTM D975-81 to test the quality and reliability of oil stored in the fuel oil supply tank and the oil used to fill or refill the supply tank.

b) Instead of the two weeks recommended in Part C.2.b of RG 1.137, CP&L will complete analyses of fuel oil properties listed in applicable specifications (other than specific or API gravity, water and sediment and viscosity) within 31 days of the addition.

c) Instead of the standard recommended in Part C.2.c of RG 1.137 (ASTM D270-1975), CP&L will periodically sample the fuel oil in accordance with ASTM D2276-78.

d) Exceptions to Appendix B to ANSI N195-1976 are described in Section 9.5.4.1 and 9.5.4.5.

e) Fuel oil sampling may be performed at an offsite contracted lab that is on the Approved Suppliers List (ASL) or analyzed onsite to comply with the requirements of the Diesel Fuel Oil Testing Program as described in Technical Specification 6.8.4.q.

These surveillance requirements are partly based on a Technical Report entitled, "Surveillance Requirements for Emergency Diesel Fuel Oil Systems in Nuclear Power Plants" (Ref. 1.8-14).

This position endorses ASTM D975-81 and ANSI N195-1976 Appendix B with the exceptions noted above. Therefore, CP&L meets the intent of RG 1.137.

Regulatory Guide 1.138 LABORATORY INVESTIGATIONS OF SOILS FOR ENGINEERING ANALYSIS AND DESIGN OF NUCLEAR POWER PLANTS (REV 0)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.139 GUIDANCE FOR RESIDUAL HEAT REMOVAL (REV 0)

Section D, "Implementation", of Regulatory Guide 1.139 states that the method described will be used in the evaluation of submittals for construction permit applications docketed after January 1, 1978. On this basis, CP&L feels that compliance with this regulatory guide is not mandated. However, in recognition of the NRC's prerogative to review against this guide on a case-by-case basis, the following discussion of compliance is presented. The responses are organized per the appropriate subsection of Section C, "Regulatory Position".

1. Shearon Harris is a Class 2 plant, as defined by the implementation section of BTPRSB5-1. The safe shutdown design basis is hot standby. Thus, Shearon Harris does not fully comply with the functional requirements of Regulatory Guide 1.139.

    Four key functions are required to achieve and maintain cold shutdown. Means for performing these functions are described below:

    1. Circulation of the reactor coolant can be provided first by natural circulation (that is, using the reactor core as the heat source and the steam generators as the heat sink) and then by the residual heat removal pumps.

    2. Removal of residual heat can be accomplished first via the Auxiliary Feedwater System and then via the residual heat removal heat exchanger. Hot standby can be maintained by releasing steam via the safety grade steam generator safety valves. Cooldown to 350°F can be accomplished by releasing steam via operation of the steam generator power operated relief valves. Then cooldown to cold shutdown conditions can be achieved with the Residual Heat Removal System. A sufficient Seismic Category I supply of auxiliary feedwater to permit four hours' operation at hot standby plus cooldown to Residual Heat Removal System initiation conditions is provided by the condensate storage tank and the backup supply from the Emergency Service Water System.

    3. Boration can be accomplished using portions of the Chemical and Volume Control System. Boric acid from the boric acid tanks can be supplied to the suction of the centrifugal charging pumps by the boric acid transfer pumps. The centrifugal charging pumps can inject the boric acid into the reactor coolant system via the safety injection flow path or the normal charging and reactor coolant pump seal injection flow paths. Makeup in excess of that needed for boration can be provided from the refueling water storage tank.

    4. Depressurization can be accomplished using portions of the Chemical and Volume Control System. Either boric acid from the boric acid tanks or refueling

water can be used as desired for depressurization with the flow path being via the centrifugal charging pumps and auxiliary spray valve to the pressurizer.

Plant operation conditions which allow operation of the Residual Heat Removal System (approximately 350°F, 360 psig) can be achieved in approximately 36 hours or less following plant shutdown. Section 7.4.1 of the FSAR provides more descriptive information on the equipment used for shutdown and identifies other reference sections in the FSAR.

2a) With regard to isolation of the suction side of the RHRS, SHNPP meets the requirements of Regulatory Guide 1.139. Refer to FSAR Sections 5.4.7.1, 5.4.7.2.1, 5.4.7.2.4, and 5.4.7.2.6.

2b) With regard to isolation of the discharge side of the RHRS, SHNPP meets the requirements of Regulatory Guide 1.139. Refer to FSAR Section 5.4.7.1.

3a(1) The protection of the RHRS against overpressurization meets the requirements set forth in this part of Regulatory Guide 1.139. Refer to FSAR Sections 5.4.7.2.4, 5.4.7.2.5, and 3.2.

3.a(2) When the RHRS relief valves are stuck open, the fluid discharged inside containment goes to the pressurizer relief tank and the fluid discharged outside containment goes to the boron recycle holdup tank. Refer to FSAR Section 5.4.7.2.4.

3.b. The effect of a relief valve stuck in the open position can be mitigated by isolation of the associated RHRS loop. The minimum design capabilities of the ECCS require one RHRS loop to function. Refer to FSAR Section 6.3.2.

3c. See 3a(2) above.

4. SHNPP complies with Regulatory Guide 1.139. Refer to FSAR Section 5.4.7.2.

5. SHNPP meets the requirements of Regulatory Guides 1.68 and 1.139. Refer to FSAR Sections 14.2.12.28 and 14.2.12.29. The system can be tested during operation.

6. The RHRS meets the requirements of Regulatory Guide 1.139 due to the fact that the AFWS has the ability to remote manually take suction from the SWS to provide feedwater for much longer than four hours. Refer to FSAR Sections 10.4.9.2.2 and 10.4.9.3.

7. The operational procedures will be written to comply with the intent of Regulatory Guide 1.139.

Regulatory Guide 1.140 DESIGN, TESTING AND MAINTENANCE CRITERIA FOR NORMAL VENTILATION EXHAUST SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (REV 1)

The SHNPP project meets the intent of this guide as described in Sections 9.4.3, 9.4.4, and 9.4.7 with the following exceptions:

Regulatory Position	Exceptions
c.3.b	This section requires HEPA filters to be in accordance with MIL-F-51068. MIL-F-51068 has been canceled and replaced by ASME AG-1; therefore, HEPA filter requirements will be allowed to either specification.
3.g, 6.a, 6.b	Each original or replacement batch of impregnated activated carbon used in the absorber section should meet the qualification and batch test results summarized in Table 5.1 of ANSI/ASME N509-1980 with the additional exception that the 30°C/95% relative humidity methyl iodide test is performed per ASTM D3803-1989.
5.a, 5.b, 5.c, 5.d	In-Place Testing to be performed per ANSI/ASME N510-1980.
6.a, 6.b	Laboratory tests (radioiodine removal efficiency) of representative samples of used activated carbon to be performed per ASTM D3803-1989 at 30°C and 70% relative humidity.

Regulatory Guide 1.141 CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS (REV 0)

The SHNPP project complies with this guide as described below:

1) Design criteria for closed systems used as isolation barriers is described in Section 6.2.4.

2) Containment isolation valve leakage testing will be performed in accordance with Appendix J to 10 CFR 50 as described in Section 6.2.6.

Regulatory Guide 1.142 SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS (OTHER THAN REACTOR VESSELS AND CONTAINMENTS) (REV 0)

This guide does not have to be addressed by the SHNPP project.

Regulatory Guide 1.143 DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (REV 1)

The SHNPP project meets the intent of this guide. The Waste Processing Systems meet the requirements of Regulatory Guide 1.143. These guidelines are spelled out in SRP Section 11.2 and BTP ETSB 11.1 (Revision 1).

The QA program described in BTP ETSB 11-1, Revision 1, is being applied with the clarification that (i) Vendor QA programs based on ASME Section VIII or ANSI B31.1 for Boiler External Piping are considered acceptable to comply with ETSB 11-1 QA requirements; and (ii) the Radwaste QA Program applies only to the pressure boundary.

The hydrostatic testing of piping systems is being applied as described in BTP ETSB 11-1, Revision 1 except that, where such testing would damage equipment, the 75 psig minimum shall be waived. In this case, pneumatic testing should be performed.

The Waste Processing System meets the design and construction specifications with the exception to the recommendations, of the screwed connections providing the only seal for quick disconnects, small manual and pressure relief valves, which are not available in any flanged, socket weld or butt welded type.

The Waste Processing System meets the material specifications requirements of Regulatory Guide 1.143 with the exception of the flexible high pressure hoses utilized in the Modular Fluidized Transfer Demineralization System. Due to the Modular Design of this system it is necessary to use quick disconnects and high pressure hoses. It allows the operational flexibility needed for this system. These hoses were hydrostatically tested to 225 psig. They have a design operational pressure of 150 psig. The total dynamic head of the floor drain feed pumps are 117 psig. The hoses will be hydrostatically tested or replaced every four years.

Regulatory Guide 1.144 AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.144 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.146 QUALIFICATION OF QA PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.146 is addressed in the description of the Quality Assurance Program that is incorporated by reference into the FSAR Chapter 17 (see Section 17.3)

Regulatory Guide 1.147 INSERVICE INSPECTION CODE CASE ACCEPTABILITY, ASME SECTION XI DIVISION I

This Regulatory Guide lists those Section XI ASME Code Cases that are generally acceptable to the NRC Staff for implementation in the inservice inspection of components and supports at light-water cooled nuclear power plants. The Carolina Power & Light Company may make use of these code cases provided for in the Regulatory Guide.

Regulatory Guide 1.149 NUCLEAR POWER PLANT SIMULATION FACILITIES FOR USE IN OPERATOR TRAINING, LICENSE EXAMINATIONS, AND APPLICANT EXPERIENCE REQUIREMENTS (Rev. 4)

The SHNPP project complies with Regulatory Guide 1.149 which endorses ANSI/ANS-3.5-2009.

Regulatory Guide 1.150 ULTRASONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATIONS REVISION 1

The SHNPP project complies with the recommendation of this guide. The Inservice Inspection Program Plan identifies the Reactor Vessel Welds requiring the use of this Regulatory Guide.

Regulatory Guide 1.155 STATION BLACKOUT (August 1988)

The SHNPP project complies with Regulatory Guide 1.155.

FSAR Reference: Section 8.3.1.2.21

Regulatory Guide 1.183 ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (July, 2000)

The SHNPP project complies with the intent of Regulatory Guide 1.183 as described in the specific event analyses of FSAR Section 15.

FSAR References: Sections 6.5.2, 15.09, 15.15, 15.26, 15.3.3, 15.4.3 (by reference to 15.3.3), 15.4.7 (by reference to 15.4.3), 15.4.8, 15.6.2 (only for use of TEDE/10 CFR 50.67 dose limits; methods still remain described in Standard Review Plan), 15.6.3, 15.6.5, 15.7.1 (only for use of TEDE/10 CFR 50.67 dose evaluation method; assumptions and limits still remain described in HNP Tech Specs), 15.7.4.

Regulatory Guide 1.192 OPERATION AND MAINTENANCE CODE CASE ACCEPTABILITY, ASME OM CODE

This Regulatory Guide lists those ASME OM Code cases that are generally acceptable to the NRC staff for implementation in the Inservice Testing and Examination of components at light-water cooled nuclear power plants. The Carolina power and Light Company may make use of these Code cases provided for in the Regulatory Guide.

REFERENCES: SECTION 1.8

1.8-1 "Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973.

1.8-2 Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.

1.8-3 Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.

1.8-4 Caplan, J. S., "The Application of Preheat Temperatures After Welding Pressure Vessel Steels," WCAP-8577, February 1976.

1.8-5 "Damping Values of Nuclear Power Plant Components," WCAP-7921-AR, May 1974.

1.8-6 "Westinghouse Nuclear Energy Systems Divisions Quality Assurance Plan," WCAP-8370, Revision 7A, February 1975.

1.8-7 "Westinghouse Water Reactor Divisions Quality Assurance Plan," WCAP 8370, Revision 8A, September 1977.

1.8-8 Risher, D. H., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1A, January 1975.

1.8-9 "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587.

1.8-10 Deleted by Amendment No. 45.

1.8-11 Deleted by Amendment No. 45.

1.8-12 Hellman, J. M. (Ed.), Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary) and WCAP 8219-A (Non-Proprietary), March 1975.

1.8-13 "Westinghouse Water Reactor Divisions Quality Assurance Plan," WCAP-8370 Revision 9A, October 1979.

1.8-14 Strauss, K. H. "Surveillance Requirements for Emergency Diesel Fuel Oil Systems in Nuclear Power Plants" prepared for Standardized Nuclear Unit Power Plant System, September 23, 1983.

1.8-15 Siemens Power Corporation, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," XN-NF-81-58(P)(A), Supplements 1 and 2, Revision 2 (NRC Safety Evaluation Report Issued November 16, 1983).

1.8-16 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

1.8-17 NRC letter from Conrad McCracken, NRC, to Barry McCrudden, Lehigh Testing Laboratories, Inc. "Use of ASTM-C-692-77," dated June 21, 1989.

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

AAB	Accident Analysis Branch
AASHTO	American Association of State Highway and Transportation Officials
AB	auxiliary boiler
A/C	air conditioning
ABMA	American Boiler Manufacturers' Association
ACI	American Concrete Institute
ACP	auxiliary control panel
ACRS	Advisory Committee on Reactor Safeguards
A/E	architect/engineer
AEC	Atomic Energy Commission
AF	alternating field
AFI	Air Filter Institute
AFAS	Auxiliary Feedwater Actuation Signal
AFBMA	Anti-Friction Bearing Manufacturer's Association
AFS	Auxiliary Feedwater System
AFW	Auxiliary Feedwater
AGMA	American Gear Manufacturers Association
AGU	American Geophysical Union
AI	authorized inspector
AIA	authorized inspection agency
AISC	American Institute of Steel Construction
ALARA	as low as is reasonably achievable
AMCA	Air Moving & Conditioning Association
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOV	air-operated valve
APCSB	Auxiliary and Power Conversion Systems Branch
API	American Petroleum Institute
ARI	A/C and Refrigeration Institute
ARMS	Area Radiation Monitoring Subsystem
AS	auxiliary steam
ASB	Auxiliary Systems Branch
ASCE	American Society of Civil Engineers
ASHRAE	American Society of Heating, Refrigerating and Air Conditioning
ASME	American Society of Mechanical Engineers

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

ASNT	American Society of Nondestructive Testing
ASNT-TG-IA	American Society of Nondestructive Testing - Training Guide IA
ASTM	American Society for Testing and Materials
ATWS	anticipated transients without scram
ATWT	anticipated transients without trip
AVT	all volatile treatment
AWS	American Welding Society
AWWA	American Water Works Association
BAT	boric acid tank
BCMS	Boron Concentration Measurement System
BEF	Best Estimate Flow
BIR	Boron injection recirculation
BIST	Boron injection surge tank
BIT	boron injection tank
BOL	beginning of life
BOP	balance-of-plant
B&PV	boiler and pressure vessel
BRS	Boron Recycle System
BTP	Branch Technical Position
BTR	Boron Thermal Regeneration
BTRS	Boron Thermal Regeneration System
BWR	boiling water reactor
CAPES	Containment Atmosphere Purge Exhaust System
CAS	Compression Air System
CB	Control board
CC	Centrifugal charging
CEA	French Atomic Energy Commission
C of E	U. S. Army Corps of Engineers
CF	Correction factor
CIAS	containment isolation actuation signal
CR	Control Room
CCS	Containment Cooling System
CCW	component cooling water
CCWS	Component Cooling Water System
CDC	Computer design code
CFR	Code of Federal Regulation
CHRS	Containment Heat Removal System

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

CIS	Containment Isolation System
CIT	Corporate Investigation Team
CMAA	Crane Manufacturers Association of America, Incorporated
CMTR	Certified Material Test Reports
CNS	Corporate Nuclear Safety
COR	City of Raleigh
COV	center of vortex
CP	Construction Permit
CPB	Core Performance Branch
CPDS	Condensate Polishing Demineralizer System
CPER	Construction Permit Environmental Report
CPI	center pressure index
CPIS	containment purge isolation signal
CP&L	Carolina Power & Light Company
CPM	critical path method
CPP	Containment Pre-Entry Purge System
CPPMU	Containment Pre-Entry Purge Makeup
CPRW	condensate polishing regeneration waste
CPRWCT	condensate polishing regeneration waste collection tank
CQAA	Corporate Quality Assurance Audit
CRACS	Control Room Air Conditioning System
CRD	Control Rod Drive
CRDM	control rod drive mechanism
CRDS	control rod drive system
CREST	Committee on Reactor Safety Technology
CRI	Control Room Indicator
CRM	Chemical Remanent Magnitization
CRT	cathode ray tube
CS	carbon steel
CSAS	containment spray actuation signal
CSB	Containment Systems Branch
CSIP	Charging/Safety Injection Pump
CSS	Containment Spray System
CST	condensate storage tank/Central Standard Time
CSTB	Condensate Storage Tank Building
CT	Compactension
CVCS	Chemical and Volume Control System

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

CVPETS	Condensate Vacuum Pump Effluent Treatment System
CVIS	Containment Ventilation Isolation Signal
CWS	Circulating Water System
DAF	dynamic amplification factor
DBA	Design basis accident
DBE	Design basis earthquake
DCC	Daniel Construction Company
DCN	design change notice
DDNB	delayed departure from nucleate boiling
DDR	deficiency and disposition reports
DEB	double-ended break
DECL	double-ended cold leg
DECLG	double-ended cold leg guillotine
DEHL	double-ended hot leg
DEHLG	double-ended hot leg guillotine
DEIT	digital electro-hydraulic
DEMA	Diesel Engine Manufacturers Association
DEP	Duke Energy Progress, Inc.
DEPS	double-ended pump suction
DEPSG	double-ended pump suction guillotine
DER	Design electrical rating
DF	decontamination factor
DG	diesel generator
DGB	Diesel-Generator Building
DGCAIES	Diesel Generator Combustion Air Intake and Exhaust System
DGFOSTS	Diesel Generator Fuel Oil Storage and Transfer System
DLF	dynamic load factor
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DOP	dioctyl phthalate
DOT	Department of Transportation
DPE	discipline project engineer
DWR	Department of Water Resources
DWS	Demineralized Water System
DWST	demineralized water storage tank
DWT	Drop weight test

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

Ebasco	Electric Bond and Share Company
ECAR	East Central Area Reliability
ECCS	Emergency Core Cooling System
EDT	Eastern Standard Time
E&CQA	Engineering and Construction Quality Assurance/Quality Control
EES	Emergency Exhaust System
EFDS	Equipment and Floor Drain System
EAB	Exclusion area boundary
EFPD	effective full-power day(s)
EFPH	effective full-power hour(s)
EFPY	effective full-power year(s)
EH	electric hydraulic
EHC	Electrohydraulic Control (System)
EICSB	Electrical, Instrumentation, and Control Systems Branch
EOC	Emergency Operations Center
EOL	end of life
EPC	Engineering Planning Coordinator
EPM	engineering project manager
EPR	ethylene propylene rubber
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
EQDP	Environmental Qualifications Data Package
ER	environmental report
ERDA	Energy Research and Development Administration
ESCWS	Essential Services Chilled Water System
ESDR	engineered safeguards design rating
ESF	Engineered Safety Feature(s)
ESFAS	Engineered Safety Features Actuation System
ESSA	Environmental Science Services Administration
EST	Eastern Standard Time
ERTS	Earth Resources Technology Satellite
ESWS	Emergency Service Water System
ETSB	Effluent Treatment Systems Branch
EZB	exclusion zone boundary
FA	forced air
FAA	Federal Aviation Administration
FANP	Framatome Advanced Nuclear Power

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

FCC	Federal Communications Commission
FCV	flow control valve
FDS	Floor Drain System
FDT	floor drain tank
FERP	Fire Emergency Response Plan
FHB	Fuel Handling Building
FHS	Fuel Handling System
FMEA	failure modes and effects analysis
FMR	Factory Material Research
FOA	forced oil air
FOC	Fine Offices Committee of the British Standards Institution
FPC	Federal Power Commission
FPT	feed pump turbine
FR	friction ratio
FSAR	Final Safety Analysis Report
FTS	Fuel Transfer System
FTU	Formazine turbidity unit
FW	feedwater
FWIV	feedwater isolation valve
FWS	Feedwater System
GDC	general design criteria
GFFD	gross failed fuel detector
GM	geiger mueller
GWMS	Gaseous Waste Management System
GWPS	Gaseous Waste Processing System
H&V	heating and ventilating
HEPA	high-efficiency particulate air filters
HEI	Heat Exchanger Institute
HHSI	high-head safety injection
HI	Hydraulic Institute
HITC	Hydraulic Institute Test Codes
HNP	Harris Nuclear Plant
HSST	Heavy Section Steel Technology
HTP	High Thermal Performance
HVAC	heating, ventilating, and air conditioning
HX	heat exchanger
I&C	instrumentation and control

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

ICEA	Insulated Cable Engineers Association
ICRP	International Commission on Radiation Protection
ID	inside diameter
IEEE	Institute of Electrical and Electronic Engineers
IES	Illumination Engineering Society
IH	integrated head
ILRT	integrated leakage rate test
IPCEA	Insulated Power Cable Engineers Association
IRS	Iodine Removal System
ISI	inservice inspection
LBB	Leak Before Break
LED	light-emitting diode
LEFM	linear elastic fracture mechanics
LFDCP	local fire detection control panel
LFL	lower flammability limit
LHSI	low-head safety injection
LHST	laundry and hot shower tank
LL	liquid limit
LMTD	log mean temperature difference
LNG	liquified natural gas
LOCA	loss-of-coolant accident
LOPAR	Low Parasitic
LOR	lower oil reservoir
LP	low pressure
LPG	liquid petroleum gas
LPZ	low population zone
LRTS	Liquid Radwaste Treatment System
LSA	low specific activity
LSB	last stage blade
LTC	linear translation case
LTMD	less than minimum detectable (concentration)
LVDT	linear variable differential transducers
LWMS	Liquid Waste Management System
LWPS	Liquid Waste Processing System
LWR	Light Water Reactor
M&E	Mass and Energy
M&TE	Measuring and Test Equipment

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

MAAC	Mid-Atlantic Area Council
MCB	main control board
MCC	motor control center
MCES	Main Condenser Evacuation System
MDC	moderator density coefficient
MEB	Mechanical Engineering Branch
MFCS	Main Feedwater Control System
MFIS	main feedwater isolation signal
MFIV	main feedwater isolation valve
MFWLB	main feedwater line break
MG	motor generator
MIL	military standards
MIMS	metal impact monitoring system
MLW	mean low water
MOV	motor-operated valve
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MPCa	maximum permissible concentration in air
MPCw	maximum permissible concentration in water
MS	main steam
MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSL	mean sea level
MSLB	main steam line break
MSLI	main steam line isolation
MSR	moisture separator reheater
MSS	Manufacturers Standardization Society
MSSS	Main Steam Supply System
MTC	moderator temperature coefficient
MTEB	Materials Engineering Branch
MUR-PU	Measurement Uncertainty Recapture - Power Uprate
MWS	Makeup Water System
MWST	Makeup Water Storage Tank
NASA	National Aeronautics and Space Administration
NASTRAN	NASA Structural Analysis Computer Program
NBS	National Bureau of Standards
NCDWR	North Carolina Department of Water Resources

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

NCHP	North Carolina Highway Patrol
NCRERP	North Carolina Radiation Emergency Response Plan
NCSB	North Carolina State Building Code
NCSU	North Carolina State University
NDRC	National Defense Research Committee
NDE	nondestructive examination
NDT	nil-ductility transition or nondestructive testing
NDTT	nil-ductility transition temperature
NEC	National Electric Code
NELPIA	Nuclear Engineering Liability and Property Insurance Association
NEMA	National Electrical Manufacturer's Association
NEPIA	Nuclear Engineering Property Insurance Association
NESCWS	Non-Essential Service Chilled Water System
NFPA	National Fire Protection Association
NGS	National Geodetic Survey
NIOSH	National Institute of Occupational Safety and Health
NIS	Nuclear Instrumentation System
NM	nautical miles
NMC	Nuclear Mutual Limited
NNS	non-nuclear safety
NOAA	National Oceanic and Atmospheric Administration
NPS	nominal pipe size
NPSH	net positive suction head
NPT	national pipe thread
NRC	Nuclear Regulatory Commission
NRMCA	National Ready Mixed Concrete Association
NSF	National Sanitation Foundation/National Science Foundation
NSSS	Nuclear Steam Supply System
NUPPSCO	Nuclear Power Plant Standards Committee
NWL	normal water level
NWS	National Weather Service
OA	oil-to-air
OBE	Operating Basis Earthquake
OCD	Office of Civil Defense
OD	outside diameter
OQA	Operations Quality Assurance
OQAP	Operational Quality Assurance Plan

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

ORE	occupational radiation exposure
ORNL	Oak Ridge National Laboratory
OSC	Operational Support Center
OSGSF	Old Steam Generator Storage Facility
OSHA	Occupational Safety and Health Administration
PA	Public Address
PAG	Protective Action Guides
PAMS	Post-Accident Monitoring System
PABX	Private Automatic Brand Exchange
PAMI	Post Accident Monitoring Instrumentation
PAP	plant administrative procedure
PCI	Prestress Concrete Institute/Pellet-to-Clad Interaction
PCT	Peak clad temperature
PG	Particulate and noble gas monitor
PGDS	Pressurized Gas Distribution System
PI	plasticity index
PIG	Particulate, iodine, and noble gas monitor
P&ID	piping and instrument diagram
PL	plastic limit
PMF	probable maximum flood
PMH	probable maximum hurricane
PMP	probable maximum precipitation
PMS	Primary Makeup System
PMW	probable maximum wind
PNSC	Plant Nuclear Safety Committee
POQAP	plant operational quality assurance procedure
PORV	power-operated relief valve
PPC	pore pressure cell
PPDI	Plant process display instrumentation
PPM	Procedure Preparation Manual
PRT	pressurizer relief tank
PS	pressurizer surge
PSAR	Preliminary Safety Analysis Report
PSB	Power System Branch
PSS	Process Sampling System
PSWS	Potable and Sanitary Water System
PT	Peroidic Testing

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

PUR	Power Uprate
PVC	polyvinyl chloride
PWR	pressurized water reactor
QA	quality assurance
QAA	quality assurance audit
QAB	Quality Assurance Branch
QA/QC	quality assurance/quality control
QAP	quality assurance procedures
QC	quality control
QCP	quality control procedures
RAB	Reactor Auxiliary Building
RABNVS	Reactor Auxiliary Building Normal Ventilation System
RABSRVS	Reactor Auxiliary Building Switchgear Room Ventilation System
RAM	random access memory
RCA	reactor coolant activity
RCB	Reactor Containment Building
RCC	rod cluster control
RCCA	rod cluster control assembly
RCDT	reactor coolant drain tank
RCFC	reactor containment fan cooler
RCL	Reactor Coolant Loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	Reactor Coolant System
RC&T	Radiation Control and Test
RDT	Reactor Development Technology Division of the NRC
RG	Regulatory Guide
RH	relative humidity
RHR	Residual heat removal
RHRS	Residual Heat Removal System
RHT	recycle holdup tank
RIC	rotary inertia included case
RIS	Reservoir induced seismicity
RMS	Radiation Monitoring System
RMWS	Reactor Makeup Water System
RNDT	reference nil ductility temperature
RO	reactor operator

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

RPS	Reactor Protection System
RPT	radiation protection technician
RPV	reactor pressure vessel
RQD	rock quality designation
RRS	required response spectrum
RSB	Reactor Systems Branch
RSG	Replacement Steam Generator
RT	reference temperature
RTD	resistance temperature detector
RTP	rated thermal power
RTS	Reactor Trip System
RTT	reference transition temperature
RWP	radiation work permit
RWST	refueling water storage tank
SAB	Site Analysis Branch
SAMA	Scientific Apparatus Manufacturers Association
SAR	Safety Analysis Report
SAT	spray additive tank
SC	safety class
SCA	single channel analyzer
SCFM	standard cubic feet per minute
SCR	silicon control rectifier
SCS	Soil Conservation Service
SDD	system design description
SDF	spillway design flood
SDR	supplier deviation request
SDS	Steam Dump System
SEB	Structural Engineering Branch
SERC	Southeastern Electric Reliability Council
SFA	spent fuel assembly
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SG	steam generator
SGAS	Steam Generator Available Signal
SGBS	Steam Generator Blowdown System
SGFP	steam generator feedwater pump
SGR	Steam Generator Replacement
SGTP	Steam Generator Tube Plugging

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

SGTR	Steam Generator Tube Rupture
SHEEC	Shearon Harris Energy and Environmental Center
SHNPP	Shearon Harris Nuclear Power Plant
SI	safety injection
SIAS	safety injection actuation signal
SI/EB	safety injection/emergency boration signal
SIS	Safety Injection System
SIT	structural integrity test
SLB	status light boxes
SLF	seismic load factor
SMA	shielded metal arc
SMACNS	Sheet Metal and Air Conditioning Contractors National Association
SPC	Siemens Power Corporation
SPF	standard project flood
SPS	standard project storm
SPT	standard penetration test
SRO	senior reactor operator
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SRST	spent resin storage tank
SRV	safety relief valves
SS	stainless steel
SSE	Safe shutdown earthquake
SSPC	Steel Structure Painting Council
SSPS	Solid-State Protection System
SSS	Secondary Sampling System
STA	Shift Technical Advisor
STP	standard temperature and pressure
SWPS	Solid Waste Processing System
SWS	Service Water System
TB	Turbine Building
TC	thermocouple
TDC	thermal diffusion coefficient
TDH	Total Developed Head
TDS	total dissolved solids
TEFC	totally enclosed fan cooled
TEMA	Tubular (Exchanger) Manufacturer's Association

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

TG	turbine generator
TGB	Turbine-Generator Building
TLD	thermoluminescent dosimeter
TSC	Technical Support Center
UBC	Uniform Building Code
UFL	upper flammability limit
UHS	ultimate heat sink
UL	Underwriters' Laboratories
UNC	University of North Carolina
UOR	upper oil reservoir
UPS	uninterruptible power supply
USAEC	U. S. Atomic Energy Commission
USASI	U. S. American Standards Institution
USBR	U. S. Bureau of Reclamation
USDA	U. S. Department of Agriculture
USGS	U. S. Geological Survey
USNRC	U. S. Nuclear Regulatory Commission
USPHS	U. S. Public Health Service
UTM	universal transverse mercator
TGSS	Turbine Gland Sealing System
TGS	Turbine Generator & Associated Systems
VCT	volume control tank
VHF	very high frequency
VRS	Volume Reduction System
VSL0	valve stem leak-off
VWO	valve wide open
WAPD	Westinghouse Atomic Power Division
WCAP	Westinghouse Commercial Atomic Power
WECT	waste evaporator condensate tank
WHT	waste holdup tank
WMT	waste management tank
NES	Westinghouse Nuclear Energy System
WPB	Waste Processing Building
WPBCWS	Waste Processing Building Cooling Water System
WPCB	waste processing control board
WPS	Waste Processing System
Westinghouse	Westinghouse Electric Corporation

Table 1.1.1-1 ACRONYMS USED IN THE FSAR

TABLE 1.1.1-2 ABBREVIATIONS USED IN THE FSAR

Words or Term	Abbreviation
Acre	ac.
acre-foot	ac.-ft.
actual cubic feet per minute	acfm
alternating current	AC
ampere	a
ampere-hour	ah
angular velocity	l
atmosphere	atm
atomic mass unit	amu
atomic percent	at. %
average	avg
average temperature	Tavg
before present	bp
billion cubic feet	Mmcf
brake horsepower	bhp
British thermal unit	Btu
British thermal units per lbm	Btu/lbm
British thermal units per hour	Btu/hr.
calorie	cal
centigram	cg
centimeter	cm
Chi/Q	χ/Q
coefficient	Cr
counts per minute	counts/min or cpm
cubic centimeter	cm^3 or cc
cubic foot	$ft.^3$
cubic feet per hour	$ft.^3$/hr.
cubic feet per minute	$ft.^3$/min. or cfm
cubic feet per second	$ft.^3$/sec. or cfs
cubic feet per second-days	ft^3/sec-days
cubic inch	$in.^3$
cubic meter	m^3
cubic yard	cy
curie	Ci
cycles per second	cps or Hz
decibel	Db
degrees Baume'	B
Degrees Rankine	R
degrees Centigrade	C
degrees Fahrenheit per hour	F/hr

TABLE 1.1.1-2 ABBREVIATIONS USED IN THE FSAR

degrees Kelvin	K
direct current	DC
disintegrations per minute	dpm
dry bulb temperature	dbT
electromagnetic units	emu
electron volt	ev
exponent	E
feet (foot)	ft.
feet per minute	ft./min or fpm
feet per second	ft./sec. or fps
fluence, neutron	nvt
flux, neutron	nv
foot candle	ft.-cdl
foot-pound	ft.-lb.
gallon	gal.
gallons per day	gal./day or gpd
gallons per hour	gal./hr.
gallons per minute	gal./min. or gpm
gallons per second	gal./sec. or gps
gallons per year	gpy
giga volt ampere	gva
gram	g
Hertz	Hz
horsepower	HP
horsepower-hour	HP-hr.
hour	hr.
inch (inches)	in.
inch-pound	in.-lb.
inches per second	ips or in./sec.
inches of mercury absolute	in. Hg abs.
kiloelectron volt	kev
kilogram	kg
kilograms per cubic meter	kg/m^3
kilograms per second	kg/sec.
kilometer	km
kilopounds per hour	kpph
kilovolt	Kv
kilovolt-ampere	kVa
kilowatt	Kw
kilowatt-hour	Kwh
linear foot	LF
liter	l

TABLE 1.1.1-2 ABBREVIATIONS USED IN THE FSAR

logarithm, Napierian base 2.718	e
mean sea level	MSL
mercury (absolute)	Hg (abs)
megawatt	MW
megawatt day	Mwd
megawatt days per metric ton of uranium	MWD/MTU
megawatt (electric)	Mwe
megawatt hour	MwH
megawatt (thermal)	Mwt
megohm	Meg-ohm
meter	m
microcuries per cubic centimeter	$\mu Ci/cm^3$
microcuries per gram	µCi/gm
micron, micro (10^{-6})	µ
micro mho	µ mho
mile per hour	mph
milli	m
milliampere	ma
millicurie	mCi
milliliter	ml
millimeter	mm
millimicron	µm
million cubic feet	mcf
million electron volts	Mev
million electron volts per square centimeter	mev/cm^2
million gallons per day	MGD
million years	my
milliroentgen	Mr
milliroentgen per hour	Mr/hr.
milliroentgen equivalent man	mrem
mili second	msec
millivolt	mv
millivolt amperes	MVA
millivolt amperes reactive	mvar
milliwatt	mw
minute	min.
neutron multiplication factor, effective	K_{eff}
neutron multiplication factor, infinity	$K\infty$
neutrons per square centimeter	n/cm^2
neutrons per square centimeter-second	n/cm^2-sec.
ohms, million	Meg
ohm-1	mho

TABLE 1.1.1-2 ABBREVIATIONS USED IN THE FSAR

one thousands pounds	kip
parts per million	ppm
parts per billion	ppb
pcm	percent mille
phase	ph.
poly vinyl chloride	PVC
pound	lb.
pound mass	lbm
pound mass per hour	lbm/hr
pound mass per second	lbm/sec
pounds per cubic foot	lb./ft.^{3} or pcf
pounds per square foot	lb./ft.^{2} or psf
pounds per hour	lb./hr.
pounds per second	lb./sec
pounds per square inch	psi
pounds per square inch (absolute)	psia
pounds per square inch (differential)	psid
pounds per square inch (gage)	psig
radius	r
reactive kilovolt-ampere	kvar
reactive volt-ampere	var
reactivity	Δk/k
reference temperature	T_{ref}
revolutions per minute	rpm
revolutions per second	rps
roentgen	R
roentgen equivalent man	rem
roentgens per hour	R/hr.
root mean square	rms
running foot	RF
second	sec.
specific gravity	sp gr
square	$(\ )^2$ or sq.
square foot	ft.^{2} or sq. ft.
square inch	in.^{2} or sq. in.
square mile	mi.^{2} or sq. mi.
standard cubic feet	scf or std ft^3
standard cubic feet per minute	scfm or std ft.^{3}/min.
standard cubic feet per second	scfs
Thickness	T
thousand cubic feet	mcf
thousand pounds	kip

TABLE 1.1.1-2 ABBREVIATIONS USED IN THE FSAR

thousand pounds per linear foot	k/lf
thousand pounds per square inch	ksi
ton (short ton)	ton, st
tonne (metric ton, 2,204.62 lb.)	te, mt
volt	V
volt alternating	V AC
volt ampere	Va
temperature of the hot leg	T_{hot}
temperature of the cold leg	T_{cold}
average reactor coolant temperature	T_{avg}
vibrations per minute	Vpm
volt direct current	V DC
volts per phase per Hertz	V/ph./Hz
volume percent	vol. percent
inch water gage	in. wg or wg
watt	W
weight percent	wt. percent
wet bulb temperature	wb
yard	yd.
year	yr.

TABLE 1.1.1-3 MAJOR BUILDINGS AND STRUCTURES

Administration Building
Auxiliary Boiler Fuel Oil Storage Tank
Auxiliary Dam
Auxiliary Reservoir
Auxiliary Reservoir Channel
Auxiliary Reservoir Separating Dike
Auxiliary Transformer
Concrete Containment Structure
Containment Building: Containment
Control Room
Cooling Tower
Cooling Tower Makeup Water Intake Channel
Diesel Fuel Oil Storage Tank Building
Diesel Generator Building
Emergency Service Water and Cooling Tower Makeup Intake Structure
Emergency Service Water Discharge Channel
Emergency Service Water Discharge Structure
Emergency Service Water Intake Channel
Emergency Service Water Screening Structure
Fuel Handling Building
Fuel Handling Unloading Area
Main Dam and Spillway
Main Reservoir
Main Transformer
Makeup Water System Dikes
Meteorological Tower
Microwave Tower and Equipment House
Normal Service Water Intake Structure
Old Steam Generator Storage Facility
Reactor Auxiliary Building
Security Building
Service Building
Sewage Treatment Plant
Startup Transformer
Turbine Building
Tank Building
Warehouse
Waste Processing Building
Water Treatment Building
230 Kv Switchyard
Hot Shop

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Hydraulic and Thermal (1 & 2) Design Parameters				
Total Core Heat Output, MWt	4.1	2775	2652	2775
Total Core Heat Output, 10^6 Btu per hr.	4.1	9471	9051	9471.1
Heat Generated in Fuel, Percent	4.2	97.4	97.4	97.4
System Pressure, Nominal, psia	4.4	2250	2250	2250
System Pressure, Minimum Steady State, psia	4.4	2220	2220	2220
Minimum DNB Ratio at Nominal Initial	4.3			
Rating Conditions				
Typical Cell		1.98	2.27	2.15
Thimble Cell		1.68	1.86	1.77
Minimum DNBR for Design Transients	4.3	>1.30	>1.30	>1.30
DNB Correlation	4.3	“R” (W-3 with modified Spacer Factor)	“R” (W-3 with modified Spacer Factor)	“R” (W-3 with modified Spacer Factor)
Nuclear Enthalpy Rise Hot Channel Factor	4.3	1.55	1.55	1.55
Coolant Flow				
Total Flow Rate, 10^6 lb. per hr.	4.4	109.1	100.9	105.2
Effective Flow Rate for Heat Transfer, 10^6 lb. per hr.	4.4	102.5	96.3	100.5

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Effective Flow Area for Heat Transfer, $ft.^2$	4.4	41.6	41.5	41.5
Average Velocity Along Fuel Rods, ft. per sec	4.4	15.6	14.5	15.1
Average Mass Velocity, 10^6 lb. per hr.-ft^2	4.4	2.47	2.32	2.42
Coolant Temperatures F at 100 Percent Power	4.4			
Nominal Inlet		556.0	542.5	546.8
Average Rise in Vessel		62.9	76.4	67.0
Average Rise in Core		66.4	70.3	67.8
Average in Core (based on enthalpy)		592.6	579.3	583.4
Average in Vessel (based on enthalpy)		588.3	576.2	580.3
Heat Transfer at 100 Percent Power	4.4			
Active Heat Transfer Surface Area, ft^2		48,600	48,700	48,600
Average Heat Flux, Btu per hr.-ft^2		189,800	181,000	189.800
Maximum Heat Flux, Btu per hr.-ft^2		440,400	434,500	440,500
Average Thermal Output, kw per ft.		5.44 (4)	5.2	5.44 (3)
Maximum Thermal Output, kw per ft.		12.6	12.5	12.6
Fuel Central Temperature, F	4.4			
Maximum at 100 Percent Power		3250	3150	3250
Maximum at Overpower		4700	4400	4150
Core Mechanical Design Parameters	4.2			
Fuel Assemblies				
Number of Fuel Assemblies		157	157	157
Design		RCC Canless 17 x 17	RCC Canless 17 x 17	RCC Canless 17 x 17
Rod Pitch, in.		0.496	0.496	0.496
Overall Dimensions, in.		8.426 x 8.426	8.426 x 8.426	8.426 x 8.426

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Fuel Weight (as UO_2), lb.		181,190	181,205	181,205
Clad Weight, lb.		41,415	38,230	38,230
Number of Grids per Assembly(5)		8-Type R	8-Type R	8-Type R
Fuel Rods	4.2			
UO^2 Rods per Assembly		264	264	264
Number		41,448	41,448	41,448
Outside Diameter, in.		0.374	0.374	0.374
Diametrical Gap, in.		0.0065	0.0065	0.0065
Clad Thickness, in.		0.0225	0.0225	0.0225
Clad Material		Zircaloy-4 or M5	Zircaloy-4	Zircaloy-4
Fuel Pellets	4.2			
Material		UO2 Sintered	UO2 Sintered	UO2 Sintered
Density (percent of Theoretical)		95	95	95
Diameter, in.		0.3225	0.3225	0.3225
Length, in. (3)		0.530	0.60	0.530
Rod Cluster Control Assemblies	4.3			
Neutron Absorber Material		5% Cd-15% In-80% Ag or 100% Hf		5% Cd-15% In-80% Ag
Cladding Material		Type 304 SS-Cold Worked		Type 304 SS-Cold Worked
Clad Thickness, in.		0.0185		0.0185
Number of RCC Assemblies (Full/Part Length)		52/0		48/5
Number of Absorber Rods per RCC Assembly		24	24	24
Core Structure	4.4			

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Core Barrel I.D./ O.D., in.		133.9/137.9	133.9/137.9	133.9/137.9
Thermal Shield		Neutron Pads	Neutron Pads	Neutron Pads
Nuclear Design Data				
Structural Characteristics				
Core Diameter, in. (Equivalent)	4.2	119.7	119.7	119.7
Core Height, in. (Active Fuel)	4.2	144	144	144
Reflector Thickness and Composition	4.5			
Top - Water plus steel		~10 in.	~10 in.	~10 in.
Bottom - Water plus steel		~10 in.	~10 in.	~10 in.
Side - Water plus steel		~15 in.	~15 in.	~15 in.
H_2O/U, Molecular Ratio (lattice, cold)		2.4	2.4	2.4
Performance Characteristics				
Loading Technique	4.3	3 region, nonuniform	3 region, nonuniform	3 region, nonuniform
Power Density, kw per liter of core		104.5	99.9	104.5
Specific Power, kw per kg UO_2		38.4[(6)]	36.6	38.3
Feed Enrichments, w/o		2.10	2.10	2.10
Region 1		2.60	2.60	2.60
Region 2		3.10	3.10	3.10
Region 3				

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Reactor Coolant System-Code Requirements Component	5.2			
Reactor Vessel		ASME III Class 1 & 2	ASME III Class 1 & 2	ASME III Class 1 & 2
Steam Generator		ASME III Class 1	ASME III Class 1	ASME III Class 1
Tube Side Shell Side		ASME III Class 2 (7)	ASME Class 2	ASME Class 2
Pressurizer		ASME III Class 1	ASME III Class 1	ASME III Class 1
Pressurizer Relief Tank		ASME VIII	ASME III Class 3	ASME Class 3
Pressurizer Safety Valves	5.2	ASME III	ASME III	ASME III
Reactor Coolant Piping	5.2	ASME III Class 1	USAS B31.1	USAS B31.1
Principal Design Parameters of the Reactor Coolant System (100% Power)				
NSSS Heat Output MWt	5.1	2785	2660	2785
NSSS Heat Output, Btu per hr.	5.1	9505×10^6	9078×10^6	9503×10^6
Operating Pressure, psi gage	5.1	2235	2235	2235
Vessel Inlet Temperature	5.1	557.4	542.5	546.8

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Vessel Outlet Temperature	5.1	620.2	610.9	613.7
Number of Loops	5.1	3	3	3
Design Pressure, psi gage	5.1	2485	2485	2485
Design Temperature, F	5.1	650	650	650
Hydrostatic Test Pressure (Cold), psi gage	5.1	3107	3107	3107
Reactor Coolant System Volume, Including Total pressurizer, $ft.^3$	5.1	8963	9458	9438
Total Reactor Flow, gpm	5.1	292,000	265,500	278,400
Reactor Design Parameters of the Reactor Vessel				
Material (Vessels)	5.3	Shell & Head Plates: ASME Section II SA-533 Grade A, B, or C Class 1 or 2, Shell & Nozzle Forgings: SA 508 Class 2 or 3. Cladding: Type 304 stainless steel or equivalent and Inconel	ASME Section II SA 32 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	ASME Section II SA 302 Grade B, low alloy steel, internal clad with Type 304 austenitic stainless steel
Design Pressure, psi gage	5.3	2485	2485	2485
Design Temperature, F	5.3	650	650	650
Operating Pressure, psi gage	5.3	2235	2235	2235
Inside Diameter of Shell, in.	5.3	157	157	157
Overall Height of Vessel/Enclosure Head, ft.-in.	5.3	42-8	40-5	42-7-3/16

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Minimum Clad Thickness, in.	5.3	1/8	5/32	5/32
Principal Design Parameters of the Steam Generators				
Number of Units	5.4	3	3	3
Type	5.4	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator
Tube Material	5.4	Inconel	Inconel	Inconel
Shell Material	5.4	Carbon Steel	Carbon Steel	Carbon Steel
Tube Side Design Pressure, psi gage	5.4	2485	2485	2485
Tube Side Design Temperature, F	5.4	650	650	650
Tube Side Design Flow, lb. per hr.	5.4	36.4×10^6	33.6×10^6	35.03×10^6
Shell Side Design Pressure, psi gage	5.4	1185	1085	1085
Shell Side Design Temperature, F	5.4	600	600	600
Operating Pressure, Tube Side, Nominal, psi gage	5.4	2235	2235	2235
Operating Pressure, Shell Side, Maximum, psi gage	5.4	1091	1005	1100
Maximum Moisture at Outlet at Full Load, percent	5.4	0.25	0.25	0.25
Hydrostatic Test Pressure, Tube Side (cold), psi gage	5.4	3107	3107	3107
Principal Design Parameters of the Reactor Coolant Pumps				
Number of Units	5.4	3	3	3

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Type	5.4	Vertical, single mixed flow	Vertical, single stage mixed flow with bottom suction and horizontal discharge	Vertical, single stage mixed flow with bottom suction and horizontal discharge
Design Pressure, psi gage	5.4	2485	2485	2485
Design Temperature, F	5.4	650	650	650
Operating Pressure, Nominal, psi gage	5.4	2235	2235	2235
Suction Temperature, F	5.4	557.1	549	546.5
Design Capacity, gpm	5.4	96,600	88,500	92,800
Design Head, ft.	5.4	300	280	312
Hydrostatic Test Pressure (Cold), psi gage	5.4	3107	3107	3107
Motor Type	5.4	AC Induction single speed	AC Induction single speed	AC Induction single speed
Motor Rating	5.4	7000 HP	6000 HP	7000 HP
Principal Design Parameters of the Reactor Coolant Piping				
Material	5.4	Austenitic SS	Austenitic SS	Austenitic SS
Hot Leg - I.D., in.	5.4	29	29	29
Cold Leg - I.D., in.	5.4	27.5	27.5	27.5
Between Pump and Steam Generator I.D., in.	5.4	31	31	31
Design Pressure, psi gage	5.4	2485	2485	2485
Engineered Safety Features				
Number of High Head Safety Injection (Charging) Pumps	6.3	3	3	3
Number of Low Head Safety Injection Pumps	6.3	2	2	2
Containment Spray-Number of Pumps	6.5.2	2	2	2
Boron Injection Tanks - Number	9.3.5	1	1	1

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Refueling Water Storage Tanks - Number	6.3	1	1	1
Hydrogen Control Systems Recombiners	6.25	2	2	2
Containment System Parameters				
Type	3.8	Steel lined reinforced concrete	Subatmospheric	Subatmospheric
Inside Diameter, ft.	3.8	130	126	126
Height ft.	3.8	225	185	191
Free Volume, 10^6 x ft.^{3}	3.8	2.266	1.8	1.825
Design Pressure, psig	3.8	45	45	45
Concrete Thickness				
Vertical Wall ft.-in.	3.8	4-6	4-6	4-6
Dome ft.-in.	3.8	2-6	2-6	2-6
Containment Leak Rate Percent per day	3.8	0.1	0.1	0.1
Electrical Systems				
Transmission Lines	8.2	7-230 kV	4-345 kV 5-1358 kV	3-500 kV (for 2 units)
Main Transformer Number	8.2	3	3	3
Startup Transformers	8.2	2	2	3
Auxiliary Transformers	8.2	2	2	3
Emergency Diesel Generators	8.2	2	2	2
Unit Batteries (125V)	8.2	3	5	4
Radioactive Waste Management System				
Liquid Waste Processing System	11.2			
Reactor Coolant Waste Holdup Tank				

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Number		1	2-5,000 gal. 2-3,000 gal. 2-3,000 gal.	2-5,000 gal. 2-1,500 gal. 2-1,400 gal.
Capacity Each (gal.)		25,000		
Spent Resin Storage Tank				
Number		4		1
Capacity (ft.^{3})		500	-	1,800
Secondary Waste Concentrate Tank				
Number		2	2	2
Capacity (gal.)		4,000	7,500	5,000
Gaseous Waste Management System	11.3			
Waste Gas Decay Tank				
Number		10	3	2
Design Pressure (psig)		150	100	145
Volume Each (ft.^{3})		600	743	3,400 gal.
Waste Gas Compressors				
Number		2	-	-
Design Pressure		150	-	-
Catalytic Hydrogen Recombiners				
Number		2	5	1
Solid Waste Management System	11.4			

TABLE 1.3.1-1 DESIGN COMPARISON WITH SIMLAR FACILITIES (INITIAL FUEL CYCLE

Item	Reference Section	Shearon Harris	Beaver Valley	North Anna
Solidification Pretreatment Tank Pump				
Number		2	1	1
Capacity (gpm)		35	-	50
Solidification Pretreatment Tank				
Number		2	1	1-1,800 gal.
Capacity (gpm)		5,000	-	1-500 gal.
Instrumentation Systems*				
Reactor Protection System	7.2	7.2	7.2	7.2
Reactor and Reactor Coolant System	7.7	7.7	7.7	7.7
Steam and Feedwater Control System	7.7	7.7	7.7	7.7
Nuclear Instrumentation	7.2	7.2	7.2	7.2
Plant Process Display Instrumentation	7.5	7.5	7.5	7.5

*This section is not suited for tabular description. SAR section numbers have been included for the location of the detailed description of each system.

NOTES:

1) There is not single design value for Axial Offset. There is however, a target value with an allowable band width, within which the plant operates. Analyses have been performed and the results presented in Chapter 4, which demonstrate that operation within the bank width results in axial power distributions no more severe than that used in the design.

2) For all three plants, the design axial power shape for DNB evaluations is a chopped cosine with peak to average value of 1.55.

3) This limit is associated with the value of Fq=2.32.

4) This value is associated with a design value of Fq=2.1. Power monitoring at this value will be performed as stated in Section 7.6.

5) Reflects current approved as-built design which may not be changed in all references SARs.

6) The specific power for Shearon Harris was based upon 94.5 percent of theoretical density.

7) The shell side of the steam generator conforms to the requirements for ASME Class 1 vessels and is so stamped as permitted under the rules of Section III.

TABLE 1.3.2-1*

COMPARISON OF FINAL AND PRELIMINARY INFORMATION
SIGNIFICANT DESIGN CHANGES

Item	Reference FSAR	Description of Change
Erosion of Hydraulic Structures	2.4.5.5	The riprap protection on the east and south edges of the plant island has been replaced by sacrificial spoil fill. Also, the auxiliary reservoir spillway has been widened.
Main Dam Low Level Release System	2.4.11	Added the low level release system for maintaining the water quality of the reservoir discharge.
Auxiliary Dam Diversion Channel	2.5.6	There is no diversion conduit through the Auxiliary Dam.
Tunnel Conduit Turbine Building	3.2.1	The tunnel under the Turbine Building was upgraded to Seismic Category I, since this tunnel is housing safety-related components.
Containment Penetrations	3.8.2.1	Changes were made on load combinations for the steel containment penetrations. These changes are deviations from SRP 3.8.2.
Electrical Penetrations	3.8.2.1	A continuous supply of pressurized nitrogen and instrument air is provided to the electrical penetrations.
Seismic Qualification of Electrical Equipment	3.10	All BOP electrical equipment previously qualified in accordance with IEEE 344-1971, has been updated to the requirements of IEEE 344-1975.
Reactor Core	4	Part length rods are deleted.
Integrated Reactor Vessel Head	9.4.8 4.6.1	The new configuration incorporates integral CRDM cooling capability and lifting device.
Containment Design Temperature	6.2.1, 3.B.1.3.1.i	The containment liner design temperature was changed to 248 F, based on the latest containment pressure and temperature analyses. Also, the dome liner will not be used to resist seismic shear stresses, in accordance with ACI 359-74.
ECCS	6.3	The switchover from injection to cold leg recirculation phase is done semi-automatically.

Item	Reference FSAR	Description of Change
Fuel Handling System	9.1.4	The system has been updated to reflect the use of two overhead cranes. Also, the fuel storage capacity has been increased. Provisions to handle and store offsite spent fuel have been made.
Cooling Tower Blowdown Discharge System	9.2.1	Cooling tower blowdown now discharges into the lake at a point approximately 3.5 miles south of the plant at about one mile north of the dam. The discharge has been changed from a multipoint diffuser to a single point jet. This change reduces circulation of the cooling tower blowdown discharge into the plant makeup structure.
Tornado Protection	9.4	Added tornado protection dampers to selected outside air intakes.
Steam Generator Blowdown System	10.4.8	The changes in the Steam Generator Blowdown System were made to upgrade the system to Safety Class 3 requirements in order to minimize pipe rupture considerations.

* THIS TABLE IS FOR HISTORICAL INFORMATION ONLY.

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977, Summer 1978" WCAP-8768, Revision 1, October, 1978.	4.3	B
TP-04124 “Missile Probability Analysis for the Siemens 13 9m2 Retrofit Design of Low-Pressure Turbine by Siemens AG”, Submitted to the Nuclear Regulatory Commission as Topical Report TP-04124-NP-A, For Public Record, June 7, 2004, Siemens Westinghouse Power Corporation.	3.5	A
"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	A
"Damping Values of Nuclear Plant Components," WCAP-7921-AR, May 1974.	3.7	A
"Safety Analysis of 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, 4.2	A
"Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, Provision 1, May 1977.	3.9	U
"Benchmark Problem Solution Employed for Verification of the WECAN Computer Program,"WCAP-8929, June 1977.	3.9	U
Takeuchi, K., et al., "Multiflex - A Fortran-IV Computer Program for Analyzing Thermal-Hydraulic- Structure System Dynamics" WCAP-8709 (February 1976).	3.9	A
Witt, F. J., Bamford, W. H., Esselman, T. C., "Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," WCAP-9288, March 1978.	3.9	U
Bogard, W. T., Esselman, T. C., "Combination of Safe Shutdown Earthquake and Loss-Of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants," WCAP-9279, March 1978.	3.9	U

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Bloyd, C. N., Ciarametaro, W. and Singleton, N. R., "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8780, May 1976.	3.9	A
"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	A
Bloyd, C. N. and Singleton, N. R. "UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations," WCAP-8516-P (Proprietary) and WCAP-8517 (Non-Proprietary) April, 1975.	3.9	A
Cooper, F. W. Jr., "17 x 17 Drive Line Components Tests - Phase 1B 11, 111, D-Loop-Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), December, 1974.	3.9	A
Kraus, S., "Neutron Shielding Pads," WCAP-7870, May, 1972.	3.9	A
Butterworth, G. and Miller, R. B., Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 2, February 1979.	3.10	U
"Equipment Qualification Data Packages," Supplement 1 to WCAP-8587, November 1978.	3.10	U
Morrone, A, "Seismic Vibration Testing With Sine Beats," WCAP-7558, October 1971.	3.10	U
Jarecki, S. J., "General Method of Developing Multi-frequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary), September 1975 and WCAP-8695 (Non-Proprietary), August 1975.	3.10	U
Butterworth, G. and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 2, February 1979.	3.11	U
"Equipment Qualification Data Packages, Supplement 1 to WCAP-8587, November 1978.	3.11	U

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Locante, J., "Environmental Testing of Engineered Safety Features Related Equipment," Vols. 1 & 2, WCAP-7410-L, December, 1970 (Westinghouse Proprietary Class II). Non-proprietary versions WCAP-7744, Vol. 1, August 1971, Vol. 2, January 1972.	3.11	A
Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model and Reactor Application," WCAP-8218-P-A (Proprietary) and WCAP-8219-A (Non-Proprietary), March, 1975.	4.1, 4.2 4.3, 4.4	A
Skaritka, J. and Iorii, J. A., "Operational Experience with Westinghouse Cores," (Up to December 31, 1977)," WCAP-8183, Revision 8, August 1978.	4.2	B
Beaumont, M. D., et al., "Properites of Fuel and Core Component Materials," WCAP-9179, Revision 1 (Proprietary) and WCAP-9224 (Non Proprietary), July 1978.	4.2	U
Christensen, J. A., Allio, R. J. and Biancheria, A., "Melting Point of Irradiated UO_2," WCAP-6065, February 1965.	4.2, 4.4	O
Miller, J. V. (Ed.), "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary), October 1976.	4.2	A
George, R. A., Lee, Y. C. and Eng, G. H., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.	4.2	A
Risher, D., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963 (Proprietary), November 1976 and WCAP-8964 (Non-Proprietary), August 1977.	4.2	A
Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977 - Summer 1978," WCAP-8768, Revision 2, October 1978.	4.2	B
Demario, E. E., "Hydraulic Flow Test of the 17 x 17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974.	4.2, 4.4	U
Reavis, J. R., et al., "Fuel Rod Bowing," WCAP-8691 (Proprietary) and WCAP-8692 (Non Proprietary), December 1975.	4.2	B

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Nuclear Fuel Division Quality Assurance Program Plan," WCAP-7800, Revision 4-A, March 1975.	4.2	A
Westinghouse Anticipated Transients Without Reactor Trip Analysis," WCAP-8330, August, 1974.	4.3, 4.6, 15.1, 15.2 15.4, 15.8	O
Langford, F. L. and Nath, R. J., "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7308-L (Proprietary) and WCAP-7810 (Non Proprietary), December, 1971.	4.3	U
Hellman, J. M. and Yang, J. W., "Effects of Fuel Densification Power Spikes on Clad Thermal Transients," WCAP-8359, July, 1974.	4.3	A
Moore, J. S., "Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-7811, December, 1971.	4.3	O
Morita, T., et al. "Power Distribution Control and Load Follow Procedures," WCAP-8385 (Proprietary) and WCAP-8403 (Non-Proprietary), September, 1974.	4.3, 4.4	A
McFarlane, A. F., "Power Peaking Factors, WCAP-7912-P-A (Proprietary) and WCAP-7912-A (Non Proprietary), January, 1975.	4.3, 4.4	A
Meyer, C. E. and Stover, R. L., "Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors," WCAP-8498, July, 1975.	4.3	U
Barry, R. F. and Altomare, S., "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Non-Proprietary), January, 1975.	4.3, 15.4	A
Cermak, J. O., et al., "Pressurized Water Reactor pH - Reactivity Effect Final Report," WCAP-3696-8 (EURAEC-2074), October, 1968.	4.3	O
Dominick, I. E. and Orr, W. L., "Experimental Verification of Wet Fuel Storage Criticality Analyses," WCAP-8682 (Proprietary) and WCAP-8683 (Non-Proprietary), December, 1975.	4.3	B
Poncelet, C. G. and Christie, A. M., "Xenon-Induced Spatial Instabilities in Large PWRs," WCAP-3680-20 (EURAEC-1974), March, 1968.	4.3	O

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Skogen, F. B. and McFarlane, A. F., "Control Procedures for Xenon-Induced X-Y Instabilities in Large PWRs," WCAP-3680-21 (EURAEC-2111), February, 1969.	4.3	O
Skogen, F. B. and McFarlane, A. F., "Xenon-Induced Spatial Instabilities in Three-Dimensions," WCAP-3680-22 (EURAEC-2116), September 1969.	4.3	O
Lee, J. C., et at., Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor," WCAP-7964, June, 1971.	4.3	B
Barry, R. F., et al., "The PANDA Code," WCAP-7048-P-A (Proprietary) and WCAP-7757-A (Non Proprietary), January, 1975.	4.3	B
Barry, R. F., "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September, 1963.	4.3, 15.4	O
Poncelet, C. G., "LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," WCAP-6073, April, 1966.	4.3	O
Olhoeft, J. E., "The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements," WCAP-2048, July, 1962.	4.3	O
Nodvik, R. J., et al., "Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium Through Curium," WCAP-6086, August, 1969.	4.3	O
Moore, J. S., "Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods," WCAP-7806, December, 1971.	4.3	B
Nodvik, R. J., "Saxton Core II Fuel Performance Evaluation," WCAP-3385-56, Part II, "Evaluation of Mass Spectrometric and Radiochemical Analyses of Irradiated Saxton Plutonium Fuel," July, 1970.	4.3, 4.4	O
Leamer, R. D., et al., "PUO_2-UO_2 Fueled Critical Experiments," WCAP-3726-1, July, 1967.	4.3	O
Camden, T. M. et al., "PALADON - Westinghouse Nodal Computer Code," WCAP-9485A (Proprietary) and WCAP-9486A (Non-Proprietary), December, 1978.	4.3	U

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
"Augmented Startup and Cycle 1 Physics Program Supplement 1," WCAP-8575 Supplement 1, June 1976 (Westinghouse Proprietary) and WCAP-8576, June 1976 (Non-Proprietary).	4.3	U
Motley, F. E., Wenzel, A. H. and Cadek, F. F., "Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22-Inch Grid Spacing," WCAP-8536 (Proprietary), May, 1975 and WCAP-8537, May, 1975.	4.4	A
Hill, K. W., Motley, F. E. and Cadek, F. F., "Effect of Local Heat Flux Spikes on DNB in Non-Uniform Heated Rod Bundles," WCAP-8174, August, 1973 (Proprietary) and WCAP-8202, August, 1973 (Non-Proprietary).	4.4	A
F. E. Motley, F. F. Cadek, "Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB," WCAP-7988 (Westinghouse Proprietary), and WCAP-8030-A (Non-Proprietary), October, 1972.	4.4	A
Chelemer, H., Weisman, J. and Tong, L. S., "Sub-channel Thermal Analysis of Rod Bundle Cores," WCAP-7015, Revision 1, January, 1969.	4.4	O
Motley, F. E. and Cadek, F. F., "DNB Tests Results for New Mixing Vane Grids (R)," WCAP-7695-P-A (Proprietary), January, 1975 and WCAP-7958-A, January, 1975.	4.4	A
Motley, F. E. and Cadek, F. F., "DNB Test Results for R Grid Thimble Cold Wall Cells," WCAP-7695 Addendum 1-P-A (Proprietary), January, 1975 and WCAP-7958, Addendum 1-A, January, 1975.	4.4	A
Hill, K. W. Motley, F. E., Cadek, F. F. , Wenzel, A. H., "Effect of 17 x 17 Fuel Assembly Geometry on DNB," WCAP-8296-P-A (Westinghouse Proprietary) and WCAP-8297-A (Non-Proprietary), February, 1975.	4.4	A
Cadek, F. F., Motley, F. E. and Dominicis, D. P., "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid, WCAP-7941-P-A (Proprietary), January, 1975 and WCAP-7959-A January, 1975.	4.4	A
Motley, F. E., Wenzel, A. H., Cadek, F. F., "The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8298-P-A (Proprietary), January, 1975 and WCAP-8299-A, January, 1975.	4.4	A

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Cadek, F. F., "Interchannel Thermal Mixing with Mixing Vane Grids," WCAP-7667-P-A (Proprietary), January, 1975 and WCAP-7755-A, January, 1975.	4.4	A
Hochreiter, L. E., and Chelemer, H., "Application of the THINC IV Program to PWR Design," WCAP-8054 (Proprietary), October, 1973, and WCAP-8195 (Non Proprietary), September, 1973.	4.4	A
Hetsroni, G., "Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, June, 1964.	4.4	O
Balfour, M. G., Christensen, J. A. and Ferrari, H. M., "In-Pile Measurement of UO_2 Thermal Conductivity," WCAP-2923, March 1966.	4.4	O
Hochreiter, L. E., Chelemer, H. and Chu, P. T., "THINC-IV An Improved Program for Thermal- Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June, 1973.	4.4	A
Poncelet, C. G., "Burnup Physics of Heterogeneous Reactor Lattices," WCAP-6069, June, 1965.	4.4	B
Carter, F. D., "Inlet Orificing of Open PWR Cores," WCAP-9004, January, 1969 (Proprietary) and WCAP-7836, January, 1972 (Non-Proprietary).	4.4	B
Shefcheck, J. "Application of the THINC Program to PWR Design," WCAP-7359-L (Proprietary), August, 1969 and WCAP-7838, January, 1972.	4.4	B
Novendstern, E. H. and Sandberg, R. O., "Single Phase Local Boiling and Bulk Boiling Pressure Drop Correlations," WCAP-2850-L (Proprietary) and WCAP-7916 (Non-Proprietary), April, 1966.	4.4	O
Burke, T. M., Meyer, C. E. and Shefcheck J., "Analysis of Data from the Zion (Unit 1) THINC Verification Test," WCAP-8453, May 1976.	4.4	A
Shopsky, W. E., "Failure Mode and Effects Analysis (FMEA) of the Solid State Full Length Rod Control System," WCAP-8976, August, 1977.	4.6	U

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Gangloff, W. C. and Loftus, W. D., "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L (Proprietary) and WCAP-7706 (Non-Proprietary), July, 1971.	4.6, 7.1, 7.2	U
Eggleston, F. T., Rawlins, D. H. and Petrow, J. R., "Failure Mode and Effects Analysis (FMEA) of the Engineering Safeguard Features Actuation System," WCAP-8584 (Proprietary) and WCAP-8760 (Non-Proprietary), April, 1976.	4.6	U
Cooper, L., Miselis, V. and Starek, R. M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June, 1972 (also letter NS-CE-622, dated April 16, 1975, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), additional information on WCAP-7769, Revision 1).	5.2	U
Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April, 1984.	5.2, 15.1 15.2, 15.3 15.4, 15.6	A
"Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metals," WCAP-9292, March, 1978.	5.2	U
Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7735, August, 1971.	5.2	B
Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June, 1974.	5.2	A
Buchalet, C. and Mager, T. R., " A Summary Analysis of the April 30 Incident at the San Onofre Nuclear Generating Station Unit 1", WCAP-8099, April 1973.	5.3	B
"Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September, 1973.	5.4	U
Whyte, D. D. and Picone, L. F., "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss-of-Coolant Accident Environment," WCAP-7798 L (Proprietary), November, 1971 and WCAP-7803 (Non Proprietary), December, 1971.	6.1	B

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Picone, L. F., "Evaluation of Protective Coatings for Use in Reactor Containment," WCAP-7198-L (Proprietary), April, 1968 and WCAP-7825 (Non-Proprietary), December, 1971.	6.1	B
Geets, J. M., "MARVEL, A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909, June, 1972.	6.3	B
Marasco, F. W. and Siroky, R. M., "Westinghouse 7300 Series Process Control System Noise Tests," WCAP-8892-A, June, 1977.	7.1	A
Reid, J. B., "Process Instrumentation for Westinghouse Nuclear Steam Supply Systems," WCAP-7913, January, 1973.	7.2, 7.3	B
Lipchak, J. B., "Nuclear Instrumentation System," WCAP-8255, January, 1974.	7.2, 7.7	B
Katz, D. N., "Solid State Logic Protection System Description," WCAP-7488-L (Proprietary), March, 1971 and WCAP-7672 (Non-Proprietary), May, 1971.	7.2, 7.3	A
Swogger, J. W., "Testing of Engineered Safety Features Actuation System," WCAP-7705, Revision 2, January, 1976. (Information only; i.e., not a generic topical WCAP).	7.3	B
WCAP-9230, Report on the Consequences of a Postulated Main Feedline Rupture, January 1978.	15.2	U
Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972.	15.3, 15.4	U
Balwin, M. S., Merrian, M. M., Schenkel, H. S. and Van DeWalle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975.	15.3	U
Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-A (Proprietary) and WCAP-8028-7 (Non-Proprietary), January 1975.	15.4	A
Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.	15.4	A

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Bordelon, F. M., Massie, H. W. and Zordan T. A., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, July 1974.	15.6	A
Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), June 1974.	15.6	A
Kelly, R. D., et at., "Calculational Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), June 1974.	15.6	A
Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974.	15.6	A
Bordelon, F. M., et al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), June 1974.	15.6	A
Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471 (Proprietary) and WCAP-8472 (Non Proprietary), April 1975.	15.6	A
"Westinghouse ECCS Evaluation Model - October 1975 Version," WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November 1975.	15.6	A
Esposito, V. J., Kesavan, K. and Maul, B. A., "WFLASH, A FORTRAN-IV Computer Program for Simulation of Transients in Multi-Loop PWR," WCAP-8200, Revision 2 (Proprietary) and WCAP-8261, Revision 1 (Non-Proprietary), July 1974.	15.6	A
Skwarek, R., Johnson, W., Meyer, P., "Westinghouse Emergency Core Cooling System Small Break October 1975 Model", WCAP-8970 (Proprietary) and WCAP-8971 (Non-Proprietary), April, 1977.	15.6	A
Kelly, R. D., Thompson, C. M., et. al., "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation with One Loop Out of Service for Plants Without Loop Isolation Valves," WCAP-9166, February 1978.	15.6	U

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
Eicheldinger, C., "Westinghouse ECCS Evaluation Model, February 1978 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-P-A (Non-Proprietary Version), February 1978.	15.6	A
"Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-Proprietary), July, 1974.	15.6	A
Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July, 1974.	15.6	A
"Design Considerations for the Protection from the Effects of Pipe Rupture," ETR-1002, November, 1975.	3.6	U
"Design Considerations for the Protection from the Effects of Pipe Rupture," ETR-1002P (Proprietary), November, 1975.	3.6	U
Babcock and Wilcox, H. W. Behnre, et al., "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G", BAW-10046A, Revision 2, June 1986.	5.3	A
J.R. Worsham III, "Fluence and Uncertainity Methodologies," BAW-2241 P-A, Revision 1, Framatome Technologies, Inc., Lynchburg, Virginia, April 1999.	5.3	A
"SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WAP-10698-P-A (Proprietary), August 1987	15.6.3	A
"Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1 to WCAP-10698-P-A, March 1986 (Proprietary).	15.6.3	A
"Streamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment, Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup," WCAP-10961, Rev. 1, (Proprietary), October 1985	3.11	A
"Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), September 1976.	6.2	A

TABLE 1.6-1 TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference(s)	Review Status
"Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), September 1986.	6.2	A
"Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatomospheric Containment Designs," WCAP-8822-S2-P-A (Proprietary and WCAP-8860-S2-A (Nonproprietary), September, 1986.	6.2	A
"Westinghouse LOCA Mass & Energy Release Model for Containment Design - March 1979 Version," WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Nonproprietary), May 1983.	6.2	A
"Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, Rev. 1 (Proprietary), WCAP-8312-A (Nonproprietary), August 1975.	6.2	A

TABLE 1.6-2

OTHER REPORTS INCORPORATED BY REFERENCE

Report	Reference Sections
Ebasco Services, Inc., 1975, "Fault Investigation," Shearon Harris Nuclear Power Plant Units 1, 2, 3, 4, Vol.2 (Appendices)	2.5
Ebasco Services, Inc., 1981, "Final Geologic Report on Foundation Conditions, Power Plant, Dams, and Related Structures."	2.5E
Ebasco Services, Inc., 1981, "Final Embankment Report for the Main Dam, Auxiliary Dam, and Auxiliary Separating Dike."	2.5F

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
1.1.1-1A	FLOW DIAGRAM LEGEND-PIPING AND INSTRUMENTATION SYMBOLS	5-G-0041
1.2.2-1	SITE PLAN	5-G-0003
1.2.2-2	PLOT PLAN	5-G-0002
1.2.2-3	GENERAL ARRANGEMENT-CONTAINMENT BUILDING PLAN EL. 221.00' & 236.00' UNIT 1	5-G-0011
1.2.2-7	GENERAL ARRANGEMENT-CONTAINMENT BUILDING PLAN EL. 261.00' & 286.00' UNIT 1	5-G-0012
1.2.2-11	GENERAL ARRANGEMENT-CONTAINMENT BUILDING SECTIONS-SHEET 1	5-G-0013
1.2.2-15	GENERAL ARRANGEMENT-CONTAINMENT BUILDING SECTIONS-SHEET 2	5-G-0014
1.2.2-19	GENERAL ARRANGEMENT-CONTAINMENT AUXILIARY BUILDING PLAN EL. 190.00' & 216.00'	5-G-0015
1.2.2-23	GENERAL ARRANGEMENT-CONTAINMENT GENERAL ARRANGEMENT REACTOR EL. 236.00'	5-G-0016
1.2.2-27	GENERAL ARRANGEMENT-CONTAINMENT AUXILIARY BUILDING EL. 261.00'	5-G-0017
1.2.2-31	GENERAL ARRANGEMENT-REACTOR AUXILIARY BUILDING PLAN EL. 286.00'	5-G-0018
1.2.2-35	GENERAL ARRANGEMENT-REACTOR AUXILIARY PLAN EL. 305.00'	5-G-0019
1.2.2-39	GENERAL ARRANGEMENT-REACTOR AUXILIARY BUILDING SECTIONS-SHEET 1	5-G-0020
1.2.2-43	GENERAL ARRANGEMENT-REACTOR AUXILIARY BUILDING SECTIONS-SHEET 2	5-G-0021
1.2.2-47	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING PLAN AT EL. 211.00' & 216.00'	5-G-0910
1.2.2-48	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING PLAN AT EL. 236.00'	5-G-0911
1.2.2-49	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING PLAN AT EL. 261.00'	5-G-0912
1.2.2-50	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING PLAN AT EL. 276.00'	5-G-0913
1.2.2-51	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING PLAN AT EL. 286.00' & EL. 291.00'	5-G-0914
1.2.2-52	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING SECTIONS-SHEET 1	5-G-0916
1.2.2-53	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING SECTIONS-SHEET 2	5-G-0917
1.2.2-54	GENERAL ARRANGEMENT-WASTE PROCESSING BUILDING SECTIONS-SHEET 3	5-G-0918
1.2.2-55	GENERAL ARRANGEMENT-FUEL HANDLING BUILDING PLANS-SHEET 1	5-G-0022
1.2.2-56	GENERAL ARRANGEMENT-FUEL HANDLING BUILDING PLANS-SHEET 1	5-G-0023
1.2.2-57	GENERAL ARRANGEMENT-FUEL HANDLING- BUILDING SECTIONS	5-G-0024
1.2.2-58	GENERAL ARRANGEMENT-FUEL HANDLING BUILDING SECTIONS	5-G-0025
1.2.2-59	GENERAL ARRANGEMENT-FUEL HANDLING BUILDING SECTIONS	5-G-0026
1.2.2-59A	GENERAL ARRANGEMENT-FUEL HANDLING BUILDING MISCELLANEOUS PLANS	5-G-0038
1.2.2-60	GENERAL ARRANGEMENT-TURBINE BUILDING PLAN EL. 240.00'	5-G-0004
1.2.2-64	GENERAL ARRANGEMENT-TURBINE BUILDING GROUND FLOOR PLAN	5-G-0005
1.2.2-68	GENERAL ARRANGEMENT-TURBINE BUILDING MEZZANINE FLOOR PLAN	5-G-0006
1.2.2-72	GENERAL ARRANGEMENT-TURBINE BUILDING OPERATION FLOOR PLAN	5-G-0007
1.2.2-76	GENERAL ARRANGEMENT-TURBINE BUILDING SECTIONS-SHEET 1	5-G-0008

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
1.2.2-80	GENERAL ARRANGEMENT-TURBINE BUILDING SECTIONS-SHEET 2	5-G-0009
1.2.2-84	GENERAL ARRANGEMENT-TANK AREA PLANS AND SECTIONS	5-G-0033
1.2.2-86	GENERAL ARRANGEMENT-DIESEL GENERATOR BUILDING PLANS	5-G-0036 S01
1.2.2-87	GENERAL ARRANGEMENT-DIESEL GENERATOR BUILDING SECTIONS	5-G-0036 S02
2.5.6-1	MAIN DAM GENERAL PLAN	7-G-6240
2.5.6-2	MAIN DAM PROFILE & TYPICAL SECTION	7-G-6241
2.5.6-3	RESERVOIR-WEST AUXILIARY DAM GENERAL PLAN	7-G-6270
2.5.6-4	AUXILIARY DAM-PROFILE AND TYPICAL SECTIONS	7-G-6271
2.5.6-5	AUXILIARY SEPARATIONS DIKE GENERAL PLAN, PROFILE AND SECTIONS	7-G-6343
2.5.6-6	AUXILIARY RESERVOIR CHANNEL	7-G-6371
2.5.6-7	ESW-INTAKE CHANNEL PLAN, SECTIONS & DETAILS	7-G-2940
2.5.6-8	ESW-DISCHARGE CHANNEL PLAN, SECTIONS & DETAILS	7-G-2941
2.5.6-26	MAIN DAM-DIVERSION SYSTEM	7-G-6232
2.5.6-28	COOLING TOWER MAKEUP WATER INTAKE CHANNEL	7-G-2942
3.4.1-1	PLANT SITE PLAN	5-G-0003
3.6A-1	REACTOR AND REACTOR AUXILIARY BLDG. BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-MAIN STEAM PIPING PLANS	SK-2165-MNE-R-0067
3.6A-2	MAIN STEAM PIPING-BREAK & RESTRAINT LOCATIONS & JET IMPINGEMENT ENVELOPES PLAN SHEET 2	SK-2165-MNE-R-0068
3.6A-5	REACTOR AND REACTOR AUXILIARY BLDG. BREAK AND RESTRAINT LOCATIONS AND IMPINGEMENT ENVELOPES-FEEDWATER PIPING PLANS	SK-2165-MNE-R-0071
3.6A-6	TURBINE BUILDING-BREAK & RESTRAINT LOCATIONS FEEDWATER PIPING PLANS	SK-2165-MNE-R-0072
3.6A-7	REACTOR AND REACTOR AUXILIARY BLDG. BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-FEEDWATER PIPING SECTIONS	SK-2165-MNE-R-0073
3.6A-8	CONTAINMENT BUILDING AND TUNNEL AREA BREAK RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES- AUXILIARY FEEDWATER PIPING	SK-2165-MNE-R-0074
3.6A-9	CONTAINMENT BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-CHEMICAL AND VOLUME CONTROL PIPING PLAN	SK-2165-MNE-R-0137
3.6A-10	CONTAINMENT BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-CHEMICAL AND VOLUME CONTROL PIPING-PARTIAL	SK-2165-MNE-R-0138
3.6A-11	CONTAINMENT BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-CHEMICAL AND VOLUME CONTROL PIPING-SECTIONS AND DETAILS	SK-2165-MNE-R-0139
3.6A-12	REACTOR AUXILIARY BLDG.-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES- CHEMICAL AND VOLUME CONTROL PIPING-PLANS	SK-2165-MNE-R-0140
3.6A-13	REACTOR AUXILIARY BLDG.-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES- CHEMICAL AND VOLUME CONTROL PIPING SECTIONS	SK-2165-MNE-R-0141
3.6A-14	CONTAINMENT BUILDING-BREAK & RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-REACTOR COOLANT PIPING PLAN	SK-2165-MNE-R-0147
3.6A-17	REACTOR AUXILIARY BLDG.-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-RHR AND SAFETY INJECTION PIPING PLAN-EL. 190	SK-2165-MNE-R-0151
3.6A-18	REACTOR AUXILIARY BLDG.-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-RHR AND SAFETY INJECTION PIPING PLAN - EL. 236	SK-2165-MNE-R-0152
3.6A-19	REACTOR AUXILIARY BLDG.-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-RHR AND SAFETY INJECTION PIPING- PARTIAL PLANS AND SECTIONS	SK-2165-MNE-R-0153
3.6A-20	CONTAINMENT BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-RHR AND SAFETY INJECTION PIPING PLAN SHEET 1	SK-2165-MNE-R-0154

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
3.6A-21	CONTAINMENT BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-RHR AND SAFETY INJECTION PIPING PLAN- SHEET 2	SK-2165-MNE-R-0155
3.6A-22	CONTAINMENT BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-RHR AND SAFETY INJECTION PIPING-PARTIAL PLANS AND SECTIONS	SK-2165-MNE-R-0156
3.6A-23	BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES- REACTOR PRIMARY COOLANT LOOP PIPING AND CONNECTIONS	SK-2165-MNE-R-0163
3.6A-24	CONTAINMENT BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-STEAM GENERATOR BLOWDOWN AND SAMPLING PIPING	SK-2165-MNE-R-0176
3.6A-25	REACTOR AUXILIARY BUILDING-BREAK AND RESTRAINT LOCATIONS-STEAM GENERATOR BLOWDOWN PIPING	SK-2165-MNE-R-0177
3.6A-26	COMPOSITE PIPING-SHIELDED PIPE TUNNEL-REACTOR AUXILIARY BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES	SK-2165-MNE-R-0244
3.6A-27	COMPOSITE PIPING-SHIELDED PIPE TUNNEL-REACTOR AUXILIARY BUILDING-BREAK AND RESTRAINT LOCATIONS AND JET IMPINGEMENT ENVELOPES-SHEET 2	SK-2165-MNE-R-0245
3.7.3-7	BACKFILL ADJACENT TO SEISMIC CATEGORY 1 BUILDINGS- PLANT AREA	7-G-0488
3.8.3-7	CONCRETE CONTAINMENT INTERNAL STRUCTURES-STEAM GEN. & PRESSURIZER SHIELD WALLS	7-G-0840
3.8.4-22A	DIESEL FUEL OIL STORAGE TANK BUILDING SECTIONS & DETAILS	7-G-3171
3.8.4-23	PROTECTIVE MATS FOR CLASS 1 YARD DUCT RUNS AND MISCELLANEOUS	7-G-0509
3.8.4-24	CLASS 1 UNDERGROUND ELECTRIC MANHOLES MASONRY	7-G-0512
3.8.4-25	ESW SCREENING STRUCTURE MASONRY	7-G-2875
3.8.4-26	ESW-SCREENING STRUCTURE MASONRY	7-G-2876
3.8.4-27	ESW-SCREENING STRUCTURE MASONRY	7-G-2877
3.8.4-28	ESW AND CT MAKEUP INTAKE STRUCTURE MASONRY	7-G-2845
3.8.4-29	ESW AND CT MAKEUP INTAKE STRUCTURE MASONRY	7-G-2846
3.8.4-30	ESW AND CT MAKEUP INTAKE STRUCTURE MASONRY	7-G-2847
3.8.4-31	ESW AND CT MAKEUP INTAKE STRUCTURE MASONRY	7-G-2848
3.8.4-32	ESW-DISCHARGE STRUCTURE MASONRY	7-G-2895
3.8.4-34	MAIN DAM SPILLWAY PLAN AND PROFILE	7-G-6248
3.8.4-35	MAIN DAM SPILLWAY SECTIONS	7-G-6249
3.8.4-36	MAIN DAM SPILLWAY OGEE, PIER AND ABUTMENTS	7-G-6252
3.8.4-37	AUXILIARY DAM SPILLWAY PLAN AND PROFILE	7-G-6280
3.8.5-3	GENERAL LAYOUT OF THE CONTAINMENT BUILDING-FOUNDATION MAT	7-G-0610
3.8.5-4	GENERAL LAYOUT OF THE CONTAINMENT BUILDING-FOUNDATION MAT	7-G-0611
3.11.4-1	DBA TEMPERATURE PROFILE INSIDE CONTAINMENT (COMBINED LOCA/MSLB) FOR ENVIRONMENTAL QUALIFICATION	DBD-1000-V02 FIGURE CB-1 AND TABLE CB-1
3.11.4-2	DBA TEMPERATURE PROFILE INSIDE CONTAINMENT (LOCA) FOR ENVIRONMENTAL QUALIFICATION	DBD-1000-V02 FIGURE CB-3 AND TABLE CB-3
3.11.4-3	DBA TEMPERATURE PROFILE INSIDE CONTAINMENT (MSLB) FOR ENVIRONMENTAL QUALIFICATION	DBD-1000-V02 FIGURE CB-4 AND TABLE CB-4
3.11.4-4	DBA TEMPERATURE PROFILE INSIDE MAIN STEAM TUNNEL (MSLB) (1.4 FT^2 MSLB AT 102% POWER)	DBD-1000-V02, FIGURE AB-1
3.11.6-1	PRESSURE PROFILE INSIDE CONTAINMENT (LOCA/MSLB) FOR ENVIRONMENTAL QUALIFICATION	DBD-1000-V02 FIGURE CB-2 AND TABLE CB-2
3.11.6-2	PRESSURE PROFILE INSIDE CONTAINMENT (MSLB) FOR ENVIRONMENTAL QUALIFICATION	DBD-1000-V02 FIGURE CB-5 AND TABLE CB-5

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
3.11.6-3	PRESSURE PROFILE INSIDE MAIN STEAM TUNNEL (MSLB) (1.4 FT^2 MSLB AT 102% POWER)	DBD-1000-V02 FIGURE AB-2

Table	Table Title	Design Document
3.11B-1	EQ PLANT LOCATION/ZONE LIST	DBD-1000-V02 TABLE 3-1
3.11B-2	TEMPERATURE	DBD-1000-V02, TABLES FOR TEMPERATURE, PRESSURE, HUMIDITY, SPRAY, AND SUBMERGENCE FOR ZONES OUTSIDE CONTAINMENT
3.11B-3	VALVE CONTAINMENT TEMPERATURES FOLLOWING ACCIDENT	DBD-1000-V02, TABLE AB01-02

Figure	Figure Title	
3.11B-1	GENERAL ARRANGEMENT CONTAINMENT BLDG. PLAN EL. 221.00’ & 236.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES CB11-1 AND CB21-1 AND TABLES CB11-1 AND CB21-1
3.11B-2	GENERAL ARRANGEMENT CONTAINMENT BLDG. PLAN EL. 261.00’ AND 286.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES CB31-1, AND CB41-1 AND TABLES CB31-1 AND CB41-1
3.11B-3	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 190.00’ AND 216.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & AND POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES AB01-1, AND AB11-1 AND TABLES AB01-1, AND AB11-1
3.11B-4	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 236.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02 FIGURE AB21-1 AND TABLE AB21-1
3.11B-5	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 261.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES AB31-1, AND AB32-1 AND TABLES AB31-1 AND AB32-1
3.11B-6	ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE TA12-1 AND TABLE TA12-1
3.11B-11	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 286.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02 FIGURES AB32-1, AB41-1, AND AB43-1; TABLES AB32-1, AB41-1 AND AB43-1
3.11B-12	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 305.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES AB51-1, A1, AB52-1, AND FH61-1 AND TABLES AB51-1, AB51-2, AB52-1, AND FH61-1
3.11B-13	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 216.00’ AND 236.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE FH21-1, AND TABLE FH21-1
3.11B-14	GENERAL ARRANGEMENT FUEL HANDLING BLDG. - PLAN EL. 261.00’ & 286.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES FH31-1 AND FH41-1 AND TABLES FH31-1 AND FH41-1
3.11B-15	GENERAL ARRANGEMENT TANK AREA - PLAN EL. 236.00’ & 261.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE TA11-1, AND TABLE TA11-1
3.11B-16	GENERAL ARRANGEMENT WASTE PROCESSING BLDG. - PLAN EL. 236.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE SW21-1 AND TABLE SW21-1

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
3.11B-17	GENERAL ARRANGEMENT DIESEL GENERATOR BLDG. - PLAN EL. 261.00’, 280.00’, & 292.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES DG31-1, DG31-2, AND DG31-3 AND TABLES DG31-1, DG31-2 AND DG31-3
3.11B-18	GENERAL ARRANGEMENT DIESEL OIL STORAGE TANK AREA - PLAN EL. 242.25’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE DF31-1 AND TABLE DF31-1
3.11B-19	GENERAL ARRANGEMENT EMERGENCY SERVICE WATER INTAKE STRUCTURE - PLAN EL. 262.00’ ENVIRONMENTAL PARAMETERS DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE 1E31-1 AND TABLE 1E31-1
3.11B-20	GENERAL ARRANGEMENT CONTAINMENT BLDG. PLAN EL. 221.00’ & 236.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES CB11-2 AND CB21-2 AND TABLES CB11-2 AND CB21-2
3.11B-21	GENERAL ARRANGEMENT CONTAINMENT BLDG. PLAN EL. 261.00’ & 286.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES CB31-2, AND CB41-2 AND TABLES CB31-2 AND CB41-2
3.11B-22	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 190.00’ & 216.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES AB01-2 AND AB11-2 AND TABLES AB01-3 AND AB11-2
3.11B-23	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 236.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE AB21-2 AND TABLE AB21-2
3.11B-24	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 261.00’ UNIT 1 INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES AB31-2 AND AB32-2 AND TABLES AB31-2 AND AB32-2
3.11B-25	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 286.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES AB32-2, AB41-2 AND AB43-2, AND TABLES AB32-2, AB41-2 AND AB43-2
3.11B-26	GENERAL ARRANGEMENT REACTOR AUX. BLDG. - PLAN EL. 305.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES AB51-2, AB52-2, AND FH61-2 AND TABLES AB51-3 AND AB52-2, AND FH61-2
3.11B.27	GENERAL ARRANGEMENT FUEL HANDLING BLDG. - PLANS AT EL. 216.00’ & 236.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURES FH11-1, AND FH21-2 AND TABLES FH11-1, FH21-2 AND FH21-3
3.11B-28	GENERAL ARRANGEMENT FUEL HANDLING BLDG. - PLANS AT EL. 261.00’ & 286.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02 FIGURES FH31-2 AND FH41-2 AND TABLES FH31-2, FH31-3, AND FH41-2
3.11B-29	GENERAL ARRANGEMENT TANK AREA PLAN AT EL. 236.00’ INTEGRATED RADIATION DOSES TO EQUIP. DURING NORMAL & POST-ACCIDENT ENVIRONMENTS	DBD-1000-V02, FIGURE TA11-2, AND TABLE TA11-2
4.2.2-9A	AREVA ROD CLUSTER CONTROL ASSEMBLY OUTLINE	1364-042493
5.1.2-1	REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM	5-G-0800
5.1.2-2	REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM	5-G-0801
5.1.3-1	ELEVATION DRAWINGS OF REACTOR PRIMARY COOLANT LOOP PIPING AND CONNECTIONS	5-G-0163
5.4.7-1	RESIDUAL HEAT REMOVAL SYSTEM- FLOW DIAGRAM	5-G-0824
6.2.2-1	FLOW DIAGRAM CONTAINMENT SPRAY SYSTEM	5-G-0050
6.2.2-2	CONTAINMENT SPRAY PIPING CONTAINMENT BUILDING PLAN AND SECTIONS	5-G-0119
6.2.2-3	FLOW DIAGRAM CONTAINMENT COOLING SYSTEM PURGE AND VACUUM RELIEF	8-G-0517
6.2.2-10	HVAC CONTAINMENT BUILDING PLANS-SHEET 1	8-G-0519 S01
6.2.2-11	HVAC CONTAINMENT BUILDING PLANS-SHEET 2	8-G-0519 S02

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
6.2.2-12	HVAC CONTAINMENT BUILDING PLANS-SHEET 3	8-G-0519 S03
6.2.2-13	HVAC CONTAINMENT BUILDING PLANS-SHEET 4	8-G-0519 S04
6.2.2-14	HVAC CONTAINMENT BUILDING SECTIONS-SHEET 1	8-G-0520
6.2.2-15	HVAC CONTAINMENT BUILDING SECTIONS-SHEET 2	8-G-0521
6.2.2-16	HVAC CONTAINMENT BUILDING SECTIONS-SHEET 3	8-G-0521 S02
6.2.2-19	CONTAINMENT BUILDING RECIRCULATION SUMP SCREENS	1364-036778
6.3.2-1	SAFETY INJECTION SYSTEM FLOW DIAGRAM	5-G-0808
6.3.2-2	SAFETY INJECTION SYSTEM FLOW DIAGRAM	5-G-0809
6.3.2-3	SAFETY INJECTION SYSTEM FLOW DIAGRAM	5-G-0810
7.2.1-1 S01	SHEET 1-INDEX AND SYMBOLS	1364-000864
7.2.1-1 S02	SHEET 2-REACTOR TRIP SIGNALS	1364-000865
7.2.1-1 S03	SHEET 3-FUNCTIONAL DIAGRAM- NUCLEAR INSTR. AND MANUAL TRIP SIGNALS	1364-000866
7.2.1-1 S04	SHEET 4-FUNCTIONAL DIAGRAM- NUCLEAR INSTR. PERMISSIVES AND BLOCKS	1364-000867
7.2.1-1 S05	SHEET 5-FUNCTIONAL DIAGRAM- PRIMARY COOLANT SYSTEM TRIP SIGNALS	1364-000868
7.2.1-1 S06	SHEET 6-FUNCTIONAL DIAGRAM- PRESSURIZER TRIP SIGNALS	1364-000869
7.2.1-1 S07	SHEET 7-STEAM GENERATOR TRIP SIGNALS	1364-000870
7.2.1-1 S08	SHEET 8-FUNCTIONAL DIAGRAM- SAFEGUARDS ACTUATION SIGNALS	1364-000871
7.2.1-1 S09	SHEET 9-ROD CONTROLS AND ROD BLOCKS	1364-000872
7.2.1-1 S10	SHEET 10-FUNCTIONAL DIAGRAM- STEAM DUMP CONTROL	1364-000873
7.2.1-1 S11	SHEET 11-FUNCTIONAL DIAGRAM- PRESSURIZER PRESSURE LEVEL CONTROL	1364-000874
7.2.1-1 S12	SHEET 12-FUNCTIONAL DIAGRAM- PRESSURIZER HEATER CONTROL	1364-000875
7.2.1-1 S13	SHEET 13-FEEDWATER CONTROL AND ISOLATION	1364-000876
7.2.1-1 S14	SHEET 14-FUNCTIONAL DIAGRAM- AUXILIARY FEEDWATER PUMPS STARTUP	1364-000877
7.2.1-1 S15	SHEET 15-FUNCTIONAL DIAGRAM- TURBINE TRIPS RUNBACKS AND OTHER SIGNALS	1364-000878
7.3.1-1 S01	SHEET 1-SOLID STATE PROTECTION SYSTEM FUNCTIONAL DIAGRAMS	1364-000864
7.3.1-1 S02	SHEET 2-SOLID STATE PROTECTION SYSTEM FUNCTIONAL DIAGRAMS	1364-000871
7.3.1-1 S03	SHEET 3-SOLID STATE PROTECTION SYSTEM FUNCTIONAL DIAGRAMS	1364-000877
7.3.1-1 S06	SHEET 6-SOLID STATE PROTECTION SYSTEM FUNCTIONAL DIAGRAMS	1364-000876
7.3.1-1 S07	SHEET 7-SOLID STATE PROTECTION SYSTEM FUNCTIONAL DIAGRAMS	1364-000878
7.3.1-3	CONTAINMENT SPRAY SYSTEM - LOGIC & SCHEMATIC DIAGRAMS	6-G-0423
7.3.1-4 S01	SHEET 1-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.64
7.3.1-4 S02	SHEET 2-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.65
7.3.1-4 S03	SHEET 3-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.65A
7.3.1-4 S04	SHEET 4-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.66
7.3.1-4 S05	SHEET 5-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.67
7.3.1-4 S06	SHEET 6-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.67A

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
7.3.1-4 S07	SHEET 7-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.68
7.3.1-4 S08	SHEET 8-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.69
7.3.1-4 S09	SHEET 9-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.69A
7.3.1-4 S10	SHEET 10-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.70
7.3.1-4 S11	SHEET 11-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.71
7.3.1-4 S12	SHEET 12-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.71A
7.3.1-4 S13	SHEET 13-CONTAINMENT COOLING SYSTEM, SAFETY RELATED-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.72
7.3.1-5 S01	SHEET 1-CONTAINMENT ISOLATION VALVE, MOTOR OPERATED-TYPICAL LOGIC DIAGRAMS	6-B-430 30.1
7.3.1-6	CONTAINMENT ISOLATION VALVE, AIR OPERATED-TYPICAL LOGIC	6-B-430 30.3
7.3.1-7 S01	SHEET 1-LOGIC DIAGRAMS-MAIN STEAM DRAIN LINES	6-B-430 08.7
7.3.1-7 S02	SHEET 2-LOGIC DIAGRAMS-MAIN STEAM ISOLATION VALVES FOR STEAM GENERATOR-1A	6-B-430 08.8
7.3.1-7 S03	SHEET 3-LOGIC DIAGRAMS-MAIN STEAM ISOLATION VALVES FOR STEAM GENERATOR-1B	6-B-430 08.9
7.3.1-7 S04	SHEET 4-LOGIC DIAGRAMS-MAIN STEAM ISOLATION VALVES FOR STEAM GENERATOR-1C	6-B-430 08.10
7.3.1-7 S05	SHEET 5-LOGIC DIAGRAMS-MAIN STEAM ISOLATION VALVES	6-B-430 08.11
7.3.1-7 S06	SHEET 6-LOGIC DIAGRAMS-MAIN STEAM ISOLATION VALVE BYPASS VALVES	6-B-430 08.12
7.3.1-8 S01	SHEET 1-FEEDWATER TO STEAM GENERATOR 1A INSTRUMENT SCHEMATICS AND LOGIC DIAGRAMS	6-G-0424 S01
7.3.1-8 S02	SHEET 2-FEEDWATER TO STEAM GENERATOR 1B INSTRUMENT SCHEMATICS AND LOGIC DIAGRAMS	6-G-0424 S02
7.3.1-8 S03	SHEET 3-FEEDWATER TO STEAM GENERATOR 1C INSTRUMENT SCHEMATICS AND LOGIC DIAGRAMS	6-G-0424 S03
7.3.1-9	AUXILIARY FEEDWATER SYSTEM, MOTOR DRIVEN PUMP-LOGIC & SCHEMATIC DIAGRAM (1 SHEET)	6-G-0427
7.3.1-10	AUXILIARY FEEDWATER SYSTEM, TURBINE DRIVEN PUMP-LOGIC & SCHEMATIC DIAGRAM (1 SHEET)	6-G-0428
7.3.1-11 S01	SHEET 1-RAB EMERGENCY EXHAUST SYSTEMS LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.33
7.3.1-11 S02	SHEET 2-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.33A
7.3.1-11 S03	SHEET 3-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35BA
7.3.1-11 S04	SHEET 4-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35BB
7.3.1-11 S05	SHEET 5-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35DA
7.3.1-11 S06	SHEET 6-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35DB
7.3.1-11 S07	SHEET 7-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35E
7.3.1-11 S08	SHEET 8-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35EA
7.3.1-11 S09	SHEET 9-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35EB
7.3.1-11 S10	SHEET 10-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35CA
7.3.1-11 S11	SHEET 11-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35CB
7.3.1-11 S13	SHEET 13-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35M
7.3.1-11 S14	SHEET 14-RAB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.35N
7.3.1-12 S09	SHEET 9-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33J
7.3.1-12 S10	SHEET 10-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33K
7.3.1-12 S11	SHEET 11-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33L

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
7.3.1-12 S12	SHEET 12-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33M
7.3.1-12 S13	SHEET 13-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33N
7.3.1-12 S14	SHEET 14-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33O
7.3.1-12 S15	SHEET 15-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33P
7.3.1-12 S16	SHEET 16-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33Q
7.3.1-12 S17	SHEET 17-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33R
7.3.1-12 S18	SHEET 18-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33S
7.3.1-12 S19	SHEET 19-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33T
7.3.1-12 S20	SHEET 20-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33U
7.3.1-12 S21	SHEET 21-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33V
7.3.1-12 S22	SHEET 22-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33W
7.3.1-12 S23	SHEET 23-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33X
7.3.1-12 S24	SHEET 24-RAB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.33Y
7.3.1-13 S04	SHEET 4-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.60D
7.3.1-13 S05	SHEET 5-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.60E
7.3.1-13 S06	SHEET 6-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.60F
7.3.1-13 S07	SHEET 7-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.60H
7.3.1-13 S11	SHEET 11-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.61D
7.3.1-13 S12	SHEET 12-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.61E
7.3.1-13 S13	SHEET 13-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.61F
7.3.1-13 S14	SHEET 14-FHB EMERGENCY EXHAUST SYSTEMS, LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.61H
7.3.1-14 S01	SHEET 1-FHB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.56D
7.3.1-14 S02	SHEET 2-FHB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.56A
7.3.1-14 S03	SHEET 3-FHB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.56E
7.3.1-14 S04	SHEET 4-FHB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.56B
7.3.1-14 S05	SHEET 5-FHB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.56F
7.3.1-14 S06	SHEET 6-FHB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.56G
7.3.1-14 S07	SHEET 7-FHB ISOLATION DAMPERS- LOGIC DIAGRAMS	6-B-430 31.56H
7.3.1-15 S01	SHEET 1-STATION SERVICE WATER SYSTEM-LOGIC AND SCHEMATIC DIAGRAMS	6-G-0425 S01
7.3.1-15 S02	SHEET 2-STATION SERVICE WATER SYSTEM-LOGIC AND SCHEMATIC DIAGRAMS INSTRUMENTATION AND CONTROLS	6-G-0425 S02
7.3.1-15A	LOGIC AND SCHEMATIC SERVICE WATER TO AND FROM COMPONENT COOLING WATER HEAT EXCHANGERS	6-B-430 21.2
7.3.1-16 S01	SHEET 1-ESSENTIAL SERVICE CHILLED WATER SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.82
7.3.1-16 S02	SHEET 2-ESSENTIAL SERVICE CHILLED WATER SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.82A
7.3.1-16 S03	SHEET 3-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.82B
7.3.1-16 S04	SHEET 4-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.82C
7.3.1-16 S05	SHEET 5-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.82D

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
7.3.1-16 S07	SHEET 7-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.82F
7.3.1-16 S10	SHEET 10-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.82I
7.3.1-16 S11	SHEET 11-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.83
7.3.1-16 S12	SHEET 12-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.83A
7.3.1-16 S13	SHEET 13-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.83C
7.3.1-16 S15	SHEET 15-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.84
7.3.1-16 S16	SHEET 16-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.84A
7.3.1-16 S17	SHEET 17-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.84B
7.3.1-16 S18	SHEET 18-ESSENTIAL SERVICE CHILLED WATER SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.84C
7.3.1-17 S01	SHEET 1-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.120
7.3.1-17 S02	SHEET 2-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.121
7.3.1-17 S03	SHEET 3-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.122
7.3.1-17 S04	SHEET 4-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.123
7.3.1-17 S05	SHEET 5-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.124
7.3.1-17 S06	SHEET 6-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.130
7.3.1-17 S07	SHEET 7-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.131
7.3.1-17 S08	SHEET 8-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.132
7.3.1-17 S09	SHEET 9-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.134
7.3.1-17 S10	SHEET 10-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.138
7.3.1-17 S11	SHEET 11-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.140
7.3.1-17 S12	SHEET 12-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.141
7.3.1-17 S13	SHEET 13-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.142
7.3.1-17 S14	SHEET 14-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.143
7.3.1-17 S15	SHEET 15-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.144
7.3.1-17 S16	SHEET 16-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.145
7.3.1-17 S17	SHEET 17-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.146
7.3.1-17 S18	SHEET 18-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.147
7.3.1-17 S19	SHEET 19-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.148
7.3.1-17 S20	SHEET 20-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.149
7.3.1-17 S21	SHEET 21-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.150
7.3.1-17 S22	SHEET 22-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.151
7.3.1-17 S23	SHEET 23-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.153
7.3.1-17 S24	SHEET 24-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.152
7.3.1-17 S25	SHEET 25-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.154
7.3.1-17 S26	SHEET 26-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.155
7.3.1-17 S27	SHEET 27-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.156

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
7.3.1-17 S28	SHEET 28-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.157
7.3.1-17 S29	SHEET 29-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.158
7.3.1-17 S30	SHEET 30-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.159
7.3.1-17 S31	SHEET 31-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.160A
7.3.1-17 S32	SHEET 32-CONTROL ROOM HVAC POST-ACCIDENT OAI ALARMS	6-B-430 31.160B
7.3.1-17 S33	SHEET 33-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.161
7.3.1-17 S34	SHEET 34-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.163
7.3.1-17 S35	SHEET 35-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.165
7.3.1-17 S36	SHEET 36-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.167
7.3.1-17 S37	SHEET 37-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.169
7.3.1-17 S38	SHEET 38-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.170
7.3.1-17 S39	SHEET 39-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.173
7.3.1-17 S40	SHEET 40-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.173A
7.3.1-17 S41	SHEET 41-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.174
7.3.1-17 S42	SHEET 42-CONTROL ROOM VENTILATION SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.174A
7.3.1-18 S01	SHEET 1-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85A1
7.3.1-18 S02	SHEET 2-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85A2
7.3.1-18 S03	SHEET 3-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85A3
7.3.1-18 S04	SHEET 4-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85A4
7.3.1-18 S05	SHEET 5-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85A5
7.3.1-18 S06	SHEET 6-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85A6
7.3.1-18 S07	SHEET 7-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85A7
7.3.1-18 S08	SHEET 8-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85B1
7.3.1-18 S09	SHEET 9-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85B2
7.3.1-18 S10	SHEET 10-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85B3
7.3.1-18 S11	SHEET 11-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85B4
7.3.1-18 S12	SHEET 12-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85B5
7.3.1-18 S13	SHEET 1-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85B6
7.3.1-18 S14	SHEET 14-ESF EQUIPMENT COOLING SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.85B7
7.3.1-19 S01	SHEET 1-DIESEL GENERATOR BUILDING VENTILATION-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.180
7.3.1-19 S02	SHEET 2-DIESEL GENERATOR BUILDING VENTILATION-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.180A
7.3.1-19 S03	SHEET 3-DIESEL GENERATOR BUILDING VENTILATION-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.180B
7.3.1-19 S04	SHEET 4-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.180C
7.3.1-19 S05	SHEET 5-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.180D
7.3.1-19 S06	SHEET 6-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.181
7.3.1-19 S07	SHEET 7-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.181A

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
7.3.1-19 S08	SHEET 8-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.181B
7.3.1-19 S09	SHEET 9-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.182
7.3.1-19 S10	SHEET 10-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.182A
7.3.1-19 S11	SHEET 11-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.182B
7.3.1-19 S12	SHEET 12-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.182C
7.3.1-19 S13	SHEET 13-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.182D
7.3.1-19 S14	SHEET 14-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.182E
7.3.1-19 S15	SHEET 15-DIESEL GENERATOR BUILDING VENTILATION- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.182F
7.3.1-20 S01	SHEET 1-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.29
7.3.1-20 S02	SHEET 2-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.30
7.3.1-20 S03	SHEET 3-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.30A
7.3.1-20 S04	SHEET 4-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.30B
7.3.1-20 S07	SHEET 7-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31
7.3.1-20 S08	SHEET 8-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31A
7.3.1-20 S09	SHEET 9-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31B
7.3.1-20 S10	SHEET 10-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31C
7.3.1-20 S11	SHEET 11-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31D
7.3.1-20 S12	SHEET 12-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31E
7.3.1-20 S13	SHEET 13-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31F
7.3.1-20 S14	SHEET 14-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31G
7.3.1-20 S15	SHEET 15-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31H
7.3.1-20 S16	SHEET 16-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31K
7.3.1-20 S17	SHEET 17-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31L
7.3.1-20 S18	SHEET 18-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.31M
7.3.1-20 S20	SHEET 20-ELECTRIC EQUIPMENT PROTECTION ROOM HVAC SYSTEM- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.32
7.3.1-21 S02	SHEET 2-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.73B
7.3.1-21 S03	SHEET 3-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.74
7.3.1-21 S04	SHEET 4-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.74A
7.3.1-21 S05	SHEET 5-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.75
7.3.1-21 S06	SHEET 6-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.75A
7.3.1-21 S07	SHEET 7-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.75B
7.3.1-21 S08	SHEET 8-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.75C
7.3.1-21 S09	SHEET 9-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.75D
7.3.1-21 S10	SHEET 10-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.75E
7.3.1-21 S11	SHEET 11-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.76A
7.3.1-21 S12	SHEET 12-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.76B

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
7.3.1-21 S13	SHEET 13-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.76C
7.3.1-21 S14	SHEET 14-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.77
7.3.1-21 S15	SHEET 15-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.77A
7.3.1-21 S16	SHEET 16-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.78
7.3.1-21 S17	SHEET 17-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.78A
7.3.1-21 S18	SHEET 18-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.78B
7.3.1-21 S19	SHEET 19-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.78C
7.3.1-21 S20	SHEET 20-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.78D
7.3.1-21 S21	SHEET 21-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.78E
7.3.1-21 S22	SHEET 22-RAB SWITCHGEAR ROOMS- LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.78F
7.3.1-22 S01	SHEET 1-FHB SPENT FUEL POOL PUMP ROOM VENTILATION SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.62
7.3.1-22 S03	SHEET 3-FHB SPENT FUEL POOL PUMP ROOM VENTILATION SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.62B
7.3.1-22 S06	SHEET 6-FHB SPENT FUEL POOL PUMP ROOM VENTILATION SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.63
7.3.1-22 S08	SHEET 8-FHB SPENT FUEL POOL PUMP ROOM VENTILATION SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.63B
7.3.1-22 S09	SHEET 9-FHB SPENT FUEL POOL PUMP ROOM VENTILATION SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.63C
7.3.1-24 S01	SHEET 1-FUEL OIL TRANSFER BUILDING VENTILATION SYSTEM	6-B-430 31.183
7.3.1-24 S02	SHEET 2-FUEL OIL TRANSFER BUILDING VENTILATION SYSTEM	6-B-430 31.183A
7.3.1-24 S03	SHEET 3-FUEL OIL TRANSFER BUILDING VENTILATION SYSTEM	6-B-430 31.183B
7.3.1-25 S01	SHEET 1-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.184
7.3.1-25 S02	SHEET 2-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.184A
7.3.1-25 S03	SHEET 3-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.184B
7.3.1-25 S04	SHEET 4-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.184C
7.3.1-25 S05	SHEET 5-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.184D
7.3.1-25 S06	SHEET 6-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.184E
7.3.1-25 S07	SHEET 7-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.184F
7.3.1-25 S08	SHEET 8-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.185
7.3.1-25 S09	SHEET 9-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.185A
7.3.1-25 S10	SHEET 10-EMERGENCY SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM	6-B-430 31.185B
7.3.1-26 S01	SHEET 1-DIESEL FUEL OIL TRANSFER SYSTEM	6-B-430 19.1
7.3.1-26 S02	SHEET 2-DIESEL FUEL OIL TRANSFER SYSTEM	6-B-430 19.2
7.3.1-26 S03	SHEET 3-DIESEL FUEL OIL TRANSFER SYSTEM	6-B-430 19.3
7.3.1-26 S04	SHEET 4-DIESEL FUEL OIL TRANSFER SYSTEM	6-B-430 19.4
7.3.1-26 S05	SHEET 5-DIESEL FUEL OIL TRANSFER SYSTEM	6-B-430 19.5
7.3.1-26 S06	SHEET 6-DIESEL FUEL OIL TRANSFER SYSTEM	6-B-430 19.6
7.3.1-27 S01	SHEET 1-CONTAINMENT H2 PURGE SYSTEM-LOGIC & SCHEMATIC DIAGRAMS INSTRUMENTATION AND CONTROLS	6-B-430 31.135
7.3.1-27 S02	SHEET 2-CONTAINMENT H2 PURGE SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.136

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
7.3.1-27 S03	SHEET 3-CONTAINMENT H2 PURGE SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.137
7.3.1-27 S04	SHEET 4-CONTAINMENT H2 PURGE SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.186A
7.3.1-27 S05	SHEET 5-CONTAINMENT H2 PURGE SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.186B
7.3.1-27 S06	SHEET 6-CONTAINMENT H2 PURGE SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.186C
7.3.1-27 S07	SHEET 7-CONTAINMENT H2 PURGE SYSTEM-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 31.186D
7.4.1-4	MAIN STEAM FROM STEAM GENERATOR 1A	6-B-430 08.1
7.4.1-5	MAIN STEAM FROM STEAM GENERATOR 1B AND STEAM TO AUXILIARY FEEDWATER PUMP TURBINE	6-B-430 08.2
7.4.1-6	MAIN STEAM FROM STEAM GENERATOR 1C AND STEAM TO AUXILIARY FEEDWATER PUMP TURBINE	6-B-430 08.3
7.4.1-8	AUXILIARY CONTROL PANEL ARRANGEMENT	6-SK-E-0260
7.5.1-2	CONTROL WIRING DIAGRAM ESF SYSTEM A & B LIGHT BOX ENGRAVING	6-B-401 0602
7.5.1-4	CONTROL WIRING DIAGRAM SLB-5 FRONT VIEW ENGRAVING	6-B-401 0050E
7.5.1-5	CONTROL WIRING DIAGRAM SLB-6 FRONT VIEW ENGRAVING	6-B-401 0050F
7.5.1-6	CONTROL WIRING DIAGRAM SLB-9 FRONT VIEW ENGRAVING	6-B-401 0050J
7.5.1-7	CONTROL WIRING DIAGRAM SLB-8 FRONT VIEW ENGRAVING	6-B-401 0050H
7.5.1-14	CONTROL WIRING DIAGRAM AEP-1 STATUS LIGHT BOX SLB-10 (SA) ENGRAVING	6-B-401 0050K
7.5.1-15	CONTROL WIRING DIAGRAM AEP-1 STATUS LIGHT BOX SLB-11 (SA) ENGRAVING	6-B-401 0050L
7.6.1-10	FHB FUEL POOL B-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 04.3
7.6.1-11	FHB FUEL POOL A-LOGIC & SCHEMATIC DIAGRAMS	6-B-430 04.4
7.6.1-12	FHB FUEL POOL C&D LOGIC AND SCHEMATIC DIAGRAMS	6-B-430 05.1
7.6.1-13	FHB FUEL POOL C&D LOGIC AND SCHEMATIC DIAGRAMS	6-B-430 05.2
7.6.1-14	FHB FUEL POOL C LOGIC AND SCHEMATIC DIAGRAMS	6-B-430 05.3
7.6.1-15	FHB FUEL POOL D LOGIC AND SCHEMATIC DIAGRAMS	6-B-430 05.4
8.2.1-7	230 KV SWITCHYARD AUXILIARY ONE-LINE DIAGRAM	6-G-0025
8.2.1-8	TRANSFORMER YARD ARRANGEMENT PLAN	6-G-0172
8.3.1-1	DIESEL GENERATOR LOGIC DIAGRAM	6-G-0039
9.1.3-1	FLOW DIAGRAM FUEL POOLS COOLING SYSTEM SOUTH END	5-G-0305
9.1.3-2	FLOW DIAGRAM FUEL POOLS COOLING SYSTEM NORTH END	5-G-0307
9.1.3-3	FLOW DIAGRAM FUEL POOLS CLEAN-UP SYSTEM-SHEET 1	5-G-0061
9.1.3-4	FLOW DIAGRAM FUEL POOLS CLEAN-UP SYSTEM-SHEET 2	5-G-0062
9.1.4-5	FUEL TRANSFER SYSTEM	1364-002642
9.2.1-1	FLOW DIAGRAM CIRCULATING AND STATION SERVICE WATER SYSTEMS-SHEET 1	5-G-0047
9.2.1-2	FLOW DIAGRAM CIRCULATING AND STATION SERVICE WATER SYSTEMS-UNIT 1-SHEET 2	5-G-0048
9.2.2-1	FLOW DIAGRAM-COMPONENT COOLING WATER SYSTEM	5-G-0819
9.2.2-2	FLOW DIAGRAM-COMPONENT COOLING WATER SYSTEM	5-G-0820
9.2.2-3	FLOW DIAGRAM-COMPONENT COOLING WATER SYSTEM	5-G-0821
9.2.2-4	FLOW DIAGRAM-COMPONENT COOLING WATER SYSTEM	5-G-0822

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
9.2.2-5	FLOW DIAGRAM-COMPONENT COOLING WATER SYSTEM	5-G-0822 S01
9.2.3-1	FLOW DIAGRAM POTABLE AND DEMINERALIZED WATER SYSTEMS	5-G-0049 S02
9.2.3-2	FLOW DIAGRAM-REACTOR AUXILIARY BLDG. PRIMARY AND DEMINERALIZER WATER SYSTEMS	5-G-0299
9.2.3-3	FLOW DIAGRAM-MAKEUP WATER DEMINERALIZER	1364-020974
9.2.4-1	FLOW DIAGRAM-POTABLE AND DEMINERALIZED WATER SYSTEMS	5-G-0049 S01
9.2.8-3	FLOW DIAGRAM-HVAC ESSENTIAL SERVICES CHILLED WATER CONDENSER-SA	8-G-0498 S02
9.2.9-1	FLOW DIAGRAM-NONESSENTIAL SERVICES CHILLED WATER	8-G-0497 S02
9.2.9-2	NONESSENTIAL SERVICES CHILLED WATER SYSTEM-FLOW RATES &MISCELLANEOUS DETAILS	8-G-0497 S01
9.2.10-1	FLOW DIAGRAM-COOLING WATER SYSTEM FOR WASTE PROCESSING BUILDING-SHEET 1	5-G-0876
9.2.10-2	FLOW DIAGRAM-COOLING WATER SYSTEM FOR WASTE PROCESSING BUILDING-SHEET 2	5-G-0877
9.3.1-1	FLOW DIAGRAM-COMPRESSED AIR SYSTEM	5-G-0188
9.3.1-2	FLOW DIAGRAM-SERVICE AIR SYSTEM	5-G-0300
9.3.1-2A	FLOW DIAGRAM-SERVICE AIR SYSTEM	5-G-0300 S02
9.3.1-3	FLOW DIAGRAM-INSTRUMENT AIR SYSTEM	5-G-0301
9.3.1-3A	FLOW DIAGRAM-INSTRUMENT AIR SYSTEM	5-G-0301 S02
9.3.2-1	FLOW DIAGRAM-SAMPLING SYSTEM	5-G-0052
9.3.2-1A	PRIMARY SAMPLE PANEL 1A FLOW DIAGRAM	1364-006781 S01
9.3.2-1B	PRIMARY SAMPLE PANEL 1A FLOW DIAGRAM	1364-006781 S02
9.3.2-2	FLOW DIAGRAM-SAMPLING SYSTEM (NON-NUCLEAR) AND STEAM GENERATOR WET LAY-UP SYSTEM	5-G-0089
9.3.2-2A	FLOW DIAGRAM-SECONDARY SAMPLING SYSTEM-SAMPLING RECLAMATION SYSTEM	5-G-0089 S01
9.3.2-2B	FLOW DIAGRAM-SECONDARY SAMPLING SYSTEM-SAMPLING RECLAMATION SYSTEM	5-G-0089 S02
9.3.2-3	POST ACCIDENT REACTOR COOLANT SAMPLING SYSTEM	1364-052455
9.3.3-1	FLOW DIAGRAM-REACTOR AUX. BLDG. DRAINAGE SYSTEMS	5-G-0184
9.3.3-2	FLOW DIAGRAM-CONTAINMENT TURBINE BLDG. AND TANK AREA DRAINAGE SYSTEM	5-G-0185
9.3.3-3	FLOW DIAGRAM-FUEL HANDLING BLDG. DRAINAGE SYSTEMS	5-G-0187
9.3.3-4	FLOW DIAGRAM-WASTE PROCESSING BUILDING-DRAINAGE SYSTEMS- SHEET 1	5-G-0427
9.3.3-5	FLOW DIAGRAM-WASTE PROCESSING BUILDING-DRAINAGE SYSTEMS- SHEET 2	5-G-0428
9.3.3-6	FLOW DIAGRAM-WASTE PROCESSING BUILDING-DRAINAGE SYSTEM- SHEET 3	5-G-0430
9.3.4-1	CHEMICAL AND VOLUME CONTROL SYSTEM-FLOW DIAGRAM	5-G-0803
9.3.4-2	CHEMICAL AND VOLUME CONTROL SYSTEM-FLOW DIAGRAM	5-G-0804
9.3.4-3	CHEMICAL AND VOLUME CONTROL SYSTEM-FLOW DIAGRAM	5-G-0805
9.3.4-4	CHEMICAL AND VOLUME CONTROL SYSTEM-FLOW DIAGRAM	5-G-0806
9.3.4-6	BORON RECYCLE SYSTEM- FLOW DIAGRAM	5-G-0811
9.3.4-7	BORON RECYCLE SYSTEM- FLOW DIAGRAM	5-G-0812
9.4.0-1	SYMBOLS AND ABBREVIATIONS FOR HVAC SYSTEM	8-G-0528 S02
9.4.1-1	HVAC-AIR FLOW DIAGRAM CONTROL ROOM REACTOR AUXILIARY BUILDING	8-G-0517 S04

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
9.4.2-1	HVAC-AIR FLOW DIAGRAM FUEL HANDLING BUILDING	8-G-0533
9.4.3-1	HVAC-AIR FLOW DIAGRAM REACTOR AUXILIARY BUILDING-SHEET 1	8-G-0517 S02
9.4.3-2	HVAC-AIR FLOW DIAGRAM REACTOR AUXILIARY BUILDING-SHEET 2	8-G-0517 S03
9.4.3-3	HVAC-AIR FLOW DIAGRAM WASTE PROCESSING BUILDING-SHEET 1	8-G-0533 S02
9.4.3-4	HVAC-AIR FLOW DIAGRAM WASTE PROCESSING BUILDING-SHEET 2	8-G-0533 S03
9.4.3-5	HVAC-AIR FLOW DIAGRAM WASTE PROCESSING BUILDING-SHEET 3	8-G-0533 S04
9.4.3-6	HVAC-AIR FLOW DIAGRAM WASTE PROCESSING BUILDING-SHEET 4	8-G-0533 S05
9.4.3-7	HVAC-AIR FLOW DIAGRAM WASTE PROCESSING BUILDING-SHEET 5	8-G-0533 S06
9.4.3-8	HVAC-AIR FLOW DIAGRAM WASTE PROCESSING BUILDING-SHEET 6	8-G-0533 S07
9.4.4-1	HVAC-AIR FLOW DIAGRAM TURBINE BUILDING	8-G-0562
9.4.5-1	HVAC-AIR FLOW DIAGRAM REACTOR AUXILIARY BUILDING SWITCHGEAR ROOMS AND EQUIPMENT PROTECTION ROOMS	8-G-0517 S05
9.4.5-2	HVAC-AIR FLOW DIAGRAMS MISCELLANEOUS BUILDINGS	8-G-0548
9.4.9-1	HVAC-AIR FLOW DIAGRAM RAB COMPUTER, COMMUNICATION ROOM AND BATTERY ROOM	8-G-0532 S05
9.4.9-2	HVAC-COMPUTER & COMMUNICATION MECH EQUIP ROOM ROOF PLAN & REFRIG PIPING DIAGRAMS	8-G-0525 S05
9.5.4-1	FLOW DIAGRAM DIESEL FUEL OIL SYSTEM	5-G-0063
9.5.4-2	FUEL OIL PIPING SCHEMATIC	1364-007818
9.5.5-1	DIESEL GENERATOR COOLING WATER SYSTEM	1364-007812
9.5.5-2	FLOW DIAGRAM DIESEL GENERATOR SYSTEMS, UNIT 1	5-G-0133
9.5.6-1	DIESEL GENERATOR AIR STARTING SYSTEM	1364-007813
9.5.7-1	DIESEL GENERATOR LUBRICATION SYSTEM	1364-007817
10.1.0-1	FLOW DIAGRAM MAIN STEAM SYSTEM	5-G-0042
10.1.0-2	FLOW DIAGRAM EXTRACTION STEAM SYSTEM	5-G-0043
10.1.0-3	FLOW DIAGRAM FEEDWATER SYSTEM	5-G-0044
10.1.0-3A	FLOW DIAGRAM FEEDWATER SYSTEM	5-G-0044 S02
10.1.0-4	FLOW DIAGRAM CONDENSATE AND AIR EVACUATION SYSTEMS	5-G-0045
10.1.0-5	FLOW DIAGRAM HEATER DRAIN AND VENT SYSTEMS	5-G-0046
10.1.0-6	FLOW DIAGRAM STEAM GENERATOR BLOWDOWN SYSTEM	5-G-0051
10.1.0-6A	FLOW DIAGRAM STEAM GENERATOR BLOWDOWN HEAT RECOVERY & CLEAN UP SYSTEM	5-G-0051 S02
10.2.2-1	ASSEMBLY-LONGITUDINAL SECTION	1364-093034
10.2.2-2	TURBINE GENERATOR OUTLINE DRAWING	1364-000843
10.2.2-3	TURBINE GENERATOR OUTLINE DRAWING	1364-000844
10.2.2-5	FLOW DIAGRAM MISCELLANEOUS GAS SYSTEMS	5-G-0058
10.2.2-6	FLOW DIAGRAM MISCELLANEOUS SYSTEMS	5-G-0088
10.2.2-7	ELECTRO-HYDRAULIC FLUID SYSTEM AND LUBRICATION DIAGRAM SHEET 1 OF 3	1364-002795 S01
10.2.2-8	ELECTRO-HYDRAULIC FLUID SYSTEM AND LUBRICATION DIAGRAM SHEET 2 OF 3	1364-002795 S02
10.2.2-9	ELECTRO-HYDRAULIC FLUID SYSTEM AND LUBRICATION DIAGRAM SHEET 3 OF 3	1364-002795 S03

TABLE 1.6-3 DESIGN DOCUMENTS INCORPORATED BY REFERENCE

Figure	Figure Title	Design Document
10.2.2-10	TURBINE TRIP RUNBACK & OTHER SIGS W/REQUIREMENTS	1364-000878
10.4.5-1	CIRCULATING WATER SYSTEM COOLING TOWER INTAKE STRUCTURE AND CANAL	7-G-2770
10.4.5-2	CIRCULATING WATER SYSTEM COOLING TOWER INTAKE STRUCTURE AND CANAL	7-G-2771
10.4.6-1	FLOW DIAGRAM CONDENSATE DEMINERALIZER SYSTEM	1364-003628
10.4.6-2	FLOW DIAGRAM CONDENSATE DEMINERALIZER SYSTEM	1364-003629
10.4.6-3	FLOW DIAGRAM CONDENSATE DEMINERALIZER SYSTEM	1364-003630
10.4.7-1	FEEDWATER PIPING-PLANS	5-G-0071
10.4.7-2	FEEDWATER PIPING-PLANS	5-G-0072
10.4.7-3	FEEDWATER PIPING-SECTIONS	5-G-0073
10.4.7-4	AUXILIARY FEEDWATER PIPING CONTAINMENT BUILDING AND TUNNEL AREA	5-G-0074
11.2.2-1	FLOW DIAGRAM-CONTAINMENT BUILDING WASTE PROCESSING SYSTEM	5-G-0813
11.2.2-2	FLOW DIAGRAM-WASTE PROCESSING SYSTEM, WASTE HOLD-UP AND EVAPORATION	5-G-0814
11.2.2-3	FLOW DIAGRAM-WASTE PROCESSING SYSTEM-SPENT RESIN STORAGE	5-G-0815
11.2.2-4 S01	SHEET 1-FLOW DIAGRAM-WASTE PROCESSING SYSTEM FLOOR DRAIN STORAGE AND TREATMENT	5-G-0866
11.2.2-4 S02	SHEET 2-FLOW DIAGRAM-WASTE PROCESSING SYSTEM FLOOR DRAIN STORAGE AND TREATMENT	5-G-0816
11.2.2-5	FLOW DIAGRAM-WASTE PROCESSING SYSTEM LAUNDRY AND HOT SHOWER STORAGE AND TREATMENT-SHEET 1	5-G-0825 S01
11.2.2-6	FLOW DIAGRAM-WASTE PROCESSING SYSTEM LAUNDRY AND HOT SHOWER STORAGE AND TREATMENT-SHEET 2	5-G-0825 S02
11.2.2-7	FLOW DIAGRAM-WASTE PROCESSING SYSTEM LAUNDRY AND HOT SHOWER STORAGE AND TREATMENT-SHEET 3	5-G-0825 S03
11.2.2-8 S01	SHEET 1-FLOW DIAGRAM-SECONDARY WASTE TREATMENT SYSTEM	5-G-0090
11.2.2-8 S02	SHEET 2-FLOW DIAGRAM-SECONDARY WASTE TREATMENT SYSTEM	5-G-0480
11.2.2-9	FLOW DIAGRAM-MODULAR FLUIDIZED TRANSFER DEMINERALIZATION SYSTEM	1364-097532
11.3.2-1	FLOW DIAGRAM-WASTE PROCESSING SYSTEM-GAS DECAY STORAGE	5-G-0817
11.3.2-2	FLOW DIAGRAM-WASTE PROCESSING SYSTEM, WASTE GAS COMPR, AND RECOMBINER	5-G-0818
11.4.2-1	FLOW DIAGRAM-WASTE PROCESSING SYSTEM RADWASTE SOLIDIFICATION	5-G-0826
11.4.2-3	FLOW DIAGRAM-WASTE PROCESSING SYSTEM VOLUME REDUCTION	5-G-0846 S02
11.4.2-4	FLOW DIAGRAM-WASTE PROCESSING SYSTEM-CONCENTRATES STORAGE AND SPENT RESIN TRANSFER	5-G-0827
11.4.2-5	FLOW DIAGRAM-WASTE PROCESSING SYSTEM-FILTER BACKWASH SYSTEM-SHEET 1	5-G-0849 S01
11.4.2-6	FLOW DIAGRAM-WASTE PROCESSING SYSTEM-FILTER BACKWASH SYSTEM-SHEET 2	5-G-0849 S02
11.4.2-7	FLOW DIAGRAM-WASTE PROCESSING SYSTEM-SPENT RESIN TRANSFER	5-G-0828
11.4.2-8	FLOW DIAGRAM-WASTE PROCESSING BUILDING-FILTER BACKWASH SYSTEM	5-G-0873
11.4.2-9	FLOW DIAGRAM-REACTOR AUXILIARY BUILDING-FILTER BACKWASH SYSTEM UNIT 1	5-G-0829
12.3.2-18	CONTAINMENT BUILDING LINER PENETRATIONS	5-G-0066
12.3.3-1	HVAC-REACTOR AUXILIARY BUILDING NORMAL EXHAUST EQUIPMENT ROOM EL. 286.00'	8-G-0524 S03
12.3.3-2	HVAC-REACTOR AUXILIARY BUILDING NORMAL EXHAUST EQUIPMENT ROOM SECTIONS-EL.286.00'	8-G-0539 S03

TABLE 1.6-4

PROCEDURES, PROGRAMS, OR MANUALS INCORPORATED BY REFERENCE

Document	Document Title
PLP-106	TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM AND CORE OPERATING LIMITS REPORT
PLP-114	RELOCATED TECHNICAL SPECIFICATIONS AND DESIGN BASES REQUIREMENTS
EGR-NGGC-0153	ENGINEERING INSTRUMENT SETPOINTS
DUKE-QAPD-001-A	DUKE ENERGY CORPORATION TOPICAL REPORT QUALITY ASSURANCE PROGRAM DESCRIPTION OPERATING FLEET

TABLE 1.8-1 FUNCTIONAL LEVEL, ASSIGNMENT OF RESPONSIBILITY AND QUALIFICATION CROSS REFERENCE FOR SHNPP

ANS 3.1 Section	SHNPP Title
Managers	
4.2.1	Plant Manager - Harris Plant
4.2.3	Manager - Maintenance
4.2.2(a)	Manager - Operations
4.3.2	Manager - Support Services
4.4.7	Manager - Training
4.4.5	Manager - Nuclear Oversight Section
4.6.1 Professional - Technical	
4.6.1	Director - Design Engineering
4.6.1	Director - Engineering (Refer to FSAR Section 13.1.1)
4.6.2 (l)	Shift Technical Advisor
4.6.2 (i)	ALARA Analyst
4.6.2 (j)	Superintendent - Work Control/Coordination
4.6.2 (j)	Superintendent - Nuclear Operations Performance
4.6.2 (j)	Supervisor - Design Engineering
4.6.2 (j)	Supervisor - System Engineering
4.6.2 (j)	Supervisor - Technical Programs
4.4.1	Senior Engineer - Reactor
4.4.4 * or 4.3.2(c)	Superintendent - Radiation Protection
4.4.3	Superintendent - Environmental & Chemistry
4.4.2	Maintenance Superintendent
4.3.2(c)	Supervisor - Spent Fuel
Supervisors/Foreman	
4.3.1 (b)	Superintendent - Shift Operations
4.3.2	Administrative Supervisor
4.3.2	Superintendent - Nuclear Security
4.3.2(c)	Fire Protection Coordinator
4.3.2(c)	Maintenance Supervisor - Mechanical
4.3.2(c)	I&C Supervisor
4.3.2(c)	Electrical Supervisor
4.3.2(c)	Mechanical Supervisor
4.3.2(c)	Chemistry Supervisor
4.3.2(c)	Health Physics Supervisor
4.3.2	Emergency Preparedness - Supervisor
4.3.2 (g)	Nuclear Oversight Superintendent
4.3.2 (k)	Specialist - Training
Operators/Technicians/ Maintenance Personnel	
4.5.2	Technician I - Engineering
4.5.2	Technician I - Health Physics
4.5.2(f)	Technician II/III - Health Physics
4.5.2	Technician I - Chemistry
4.5.2(f)	Technician II/III - Chemistry
4.5.2(g)	Technician - QA/QC/NDE
4.5.2(h)	Technical Aide - Security
4.5.2(h)	Fire Protection Technician
4.5.2(h)	Technical Aide - Training
4.5.2	Technician I - Maintenance
4.5.2	Technician I - I&C
4.5.2(f)	Technician II - I&C
4.5.3	Electrician I
4.5.3	Planner Analyst
4.5.3	Senior Mechanic
4.5.3	Mechanic I
4.5.3(f)	Mechanic II
4.5.1.2(e)	Senior Control Operator
4.5.1.2(e)	Control Operator
4.5.1.1(d)	Auxiliary Operator

TABLE 1.8-1 FUNCTIONAL LEVEL, ASSIGNMENT OF RESPONSIBILITY AND QUALIFICATION CROSS REFERENCE FOR SHNPP

( ) denotes exceptions or alternatives proposed in paragraph 3 above.
* Required when individual is designated as site Radiation Protection Manager.

FIGURE 1.1.1-1

FLOW DIAGRAM LEGEND

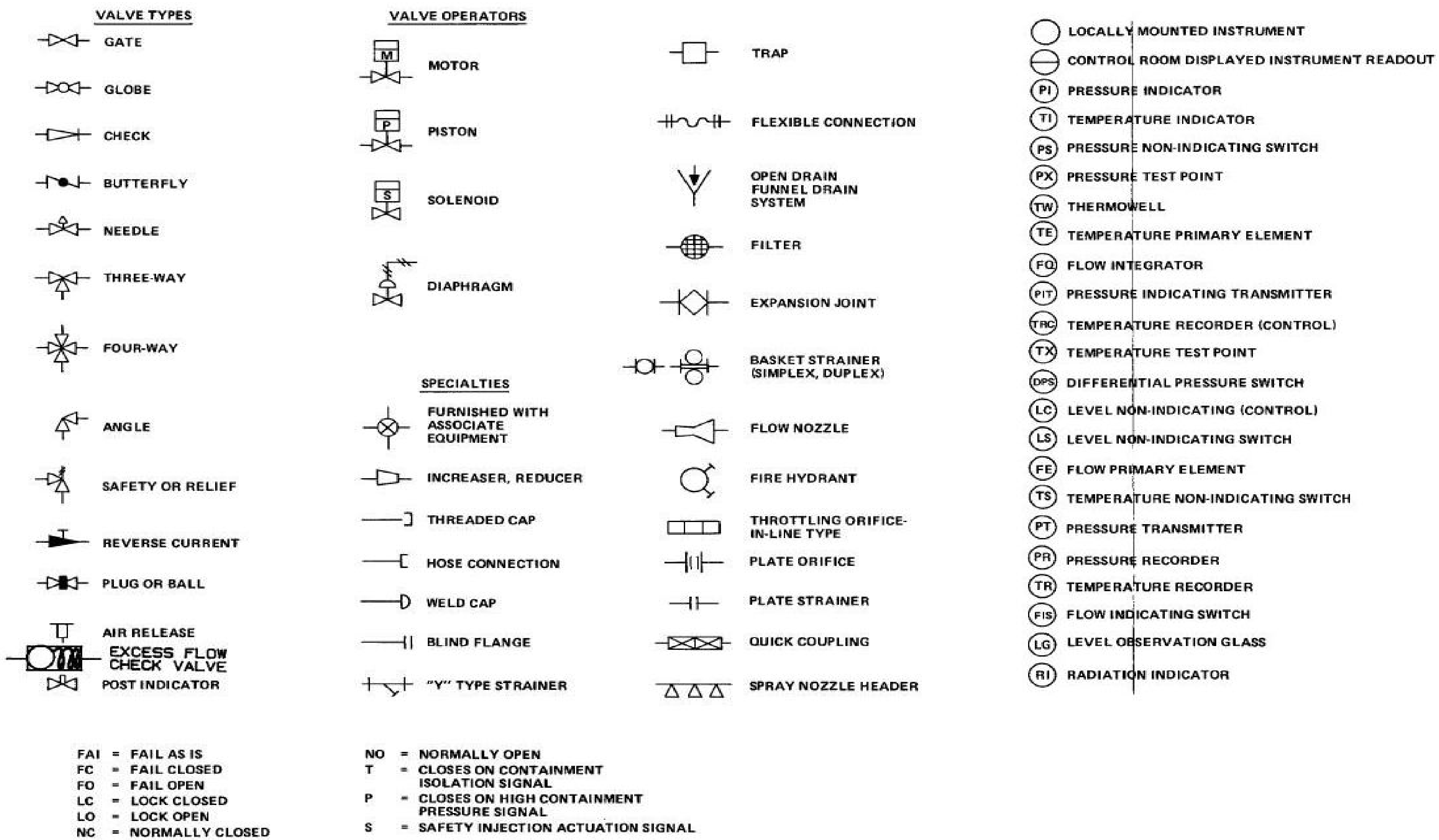

FIGURE 1.1.1-2

FLOW DIAGRAM LEGEND