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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

1.1.1 Introduction

This Final Safety Analysis Report (FSAR) is submitted in support of the application of the Tennessee Valley Authority (TVA) for Class 103 facility operating licenses for a two-unit nuclear power plant located approximately 50 miles northeast of Chattanooga at the Watts Bar site in Rhea County, Tennessee. Unit 1 received the low power operating license (NPF-20) on November 9, 1995, and the full power operating license (NPF-90) on February 7, 1996. TVA declared commercial operation on May 25, 1996. This FSAR reflects the Unit 2 plant. Unit 2 received a construction permit on January 1, 1973, and was placed in deferred status by letter to the NRC dated July 14, 2000. By letter dated August 3, 2007, TVA notified the NRC of its plans to resume unrestricted construction activities, under the existing construction permit and its plans to request an operating license prior to April 1, 2012. TVA expects to place Unit 2 in commercial operation by October 1, 2012.

This facility has been designated the Watts Bar Nuclear Plant. The plant is designed, built, and will be operated by TVA. The Unit employs a four-loop Pressurized Water Reactor Nuclear Steam Supply System (NSSS) furnished by Westinghouse Electric Corporation. The Unit is similar to Unit 1 and those of the Sequoyah Nuclear Plant and other similar Westinghouse plants licensed by the U. S. Nuclear Regulatory Commission (NRC).

The Unit 2 reactor core is rated at 3,411 MWt and, at this core power, the NSSS will operate at 3,425 MWt. The additional 16 MWt is due to the contribution of heat to the primary coolant system from nonreactor sources, primarily reactor coolant pump heat. The reactor core has an Engineered Safeguards design rating of 3,582 MWt, and each NSSS has a design rating of 3,596 MWt. The net electrical output is 1,160 MWe, and the gross electrical output is 1,218 MWe for the rated core power. Plant safety systems, including containment and engineered safety features, are designed and evaluated at the higher power level. The higher power rating is used in the analysis of postulated accidents which have as a consequence the release of fission product activity to the environment.

The containment for the reactor consists of a free standing steel vessel with an ice condenser and separate reinforced concrete Shield Building. The free standing steel vessel and the concrete Shield Building were designed by TVA, and the ice condenser was designed and furnished by the Westinghouse Electric Corporation.

1.1.2 Licensing Basis Documents

The following documents are typical documents submitted periodically to NRC following receipt of operating license. Implementation of changes to these documents without NRC approval may be controlled by regulation or the plant operating license.

The following list provides references on the review and approval requirements for the listed documents.

DOCUMENT	REGULATION OR REQUIREMENT	INCORPORATED BY REFERENCE IN FSAR
Updated Final Safety Analysis Report	10 CFR 50.59 10 CFR 50.71(e)	N/A
Technical Requirements Manual	Technical Requirement 5.1 10 CFR 50.59 10 CFR 50.36(c)(2)(ii)	Yes
Technical Specification Bases	Technical Specification 5.6 10 CFR 50.59	No
Organizational Topical Report	10 CFR 50.54(a)(3)	No
Quality Assurance Plan	10 CFR 50.54(a)(3)	No
Fire Protection Report		No
Offsite Dose Calculation Manual	Technical Specification 5.7.2.3	No
Physical Security Plan	10 CFR 50.54(p)	No
Radiological Emergency Plan	10 CFR 50.54(q)	No
Core Operating Limits Report	Technical Specification 5.9.5	No
Pressure and Temperature Limits Report	Technical Specification 5.9.6	No

1.1.3 NRC Commitments

The following programmatic commitments are incorporated to ensure control under the licensing basis process.

(1) Generic Letter 88-05 - Boric Acid Corrosion Of Carbon Steel Reactor Pressure Boundary Components In PWR Plants. TVA letter to NRC dated June 1, 1988 (L44880601805).

TVA has implemented a program to address the potential for boric acid corrosion of the reactor coolant pressure boundary in accordance with Generic Letter 88-05 (NCO880119002).

(2) Generic Letter 88-14 - Instrument Air Supply System Problems Affecting Safety-Related Equipment. TVA letters to NRC dated February 23, 1989 (L44890223805), July 12, 1990 (L44900712802), and July 14, 1995 (T04950714164).

Plant procedures require air quality sampling at remote locations in each of the three ACAS and SCSAS air headers at each unit on a six-month basis (NCO890050031).

Preventive maintenance procedures address prefilters, afterfilters, desiccant, valves, and diaphragms (NCO890050034).

Procedures require internal inspection of components suspected of contamination following indication of SCSAS or ACAS contamination due to the presence of water, particulates, or oil in system headers (NCO890050035).

The procedures, instructions, and physical plant drawings provide actions with respect to loss of air incident, incident recovery, plant response, manual actions, and unexpected component positioning (NCO890050036).

Test procedures and design control ensure that 1) accumulator check valves properly reseat upon both a gradual and rapid loss of upstream pressure, 2) check valves are properly designed for air service, 3) low accumulator tank pressure is properly annunciated, and 4) accumulator tank design is properly documented by calculations (NCO890050037).

Procedures govern the use of the lubricated condensate demineralizer air compressor (NCO890050041).

(3) Generic Letter 89-08 - Erosion/Corrosion-Induced Pipe Wall Thinning. TVA letter to NRC dated July 19, 1989 (L44890719803).

Procedures define single phase design program details such as grid locations, frequency of inspection, responsibilities, inspection performance, and acceptance criteria (NCO890173002).

Inspections are performed at scheduled refueling outages to establish the rate of wall loss and whether revision to the inspection intervals, material replacement, or design changes are warranted (NCO890173004 and NCO890173010).

Procedures address dual-phase erosion/corrosion (NCO890173008).

(4) Generic Letter 89-13 - Service Water System Problems Affecting Safety-Related Equipment. TVA letter to NRC dated January 26, 1990 (L44900126804).

Visual inspections, using divers, of the intake structure for Asiatic clams, sediment, and corrosion are initially conducted every 18 months, or each refueling outage. [After a trend has been developed (minimum of three operating cycles), the inspection frequency may be modified, based on the results of the evaluation of trend data.] Fouling accumulations are evaluated and removed as necessary (NCO900022007).

WBN has a program for heat exchanger performance testing (NCO900022008) which includes:

(1) Inspection of ERCW strainers on a periodic basis to verify strainer media is intact and to inspect for biofouling, silt, and corrosion products (NCO900022011).

(2) Inspection of ERCW pump motor-thrust bearing cooling coils on a periodic basis for biofouling, silt, and corrosion products (NCO900022012).

(3) A procedure for eddy current testing of the diesel generator and component cooling water heat exchangers (NCO900022013).

(4) A procedure to perform an evaluation whenever the ERCW system is breached and biofouling agents, corrosion products, silt and mortar fragments are found (NCO900022015).

(5) Generic Letter 81-07 and NUREG-0612 - Control Of Heavy Loads At Nuclear Power Plants. TVA letter to NRC dated July 28, 1993 (T04930728943).

Procedures are provided to implement TVA's NUREG-0612 response (NCO930238006).

(6) Generic Letter 93-04 - Rod Control System Failure And Withdrawal Of Rod Cluster Control Assemblies. TVA letter to NRC dated September 20, 1993 (L39930920800).

WBN has current order tests (current order traces from each group following each refueling outage) to ensure detectability of abnormalities and modify the rod control system current order timing to prevent any uncontrolled asymmetric rod withdrawal in the event of the failure identified at Salem. This provides a high degree of confidence that none of the rods will move if corrupted current orders are present (NCO930239005).

(7) IE Bulletin 84-03 - Refueling Cavity Water Seal. TVA letter to NRC dated December 6, 1984 (L44841206801).

A preventive maintenance program ensures that the properties of the seal are not degraded during storage, handling, and use (NCO920047116).

A maintenance instruction for removal and replacement of reactor pressure vessel head and attachments requires visual inspection and durometer readings of the seal before use (NCO920047112).

References

None

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Site Characteristics

1.2.1.1 Location

The plant site, consisting of approximately 1,770 acres, is located in southeastern Tennessee on the west shore of Chickamauga Lake approximately 50 miles northeast of Chattanooga and 31 miles northeast of the Sequoyah Nuclear Plant site.

1.2.1.2 Demography

The population density of the area surrounding the site is relatively low with only two cities within 60 miles of the plant having populations exceeding 100,000 people. The minimum exclusion and low population distances are 1,200 meters and 3 miles, respectively.

1.2.1.3 Meteorology

No known long-term meteorological measurements other than rainfall have been recorded in the immediate vicinity of the Watts Bar site. Therefore, the climatological appraisal of the site has been developed from meteorological data collected at stations within 50 miles. Based on the onsite data, categories of atmospheric stability conditions, by Pasquill classification, have been developed and atmospheric diffusion characteristics have been predicted for the site for use in accident analyses presented in Chapter 15 of this report. A permanent onsite meteorological facility has been in operation since May 1973 to meet the Nuclear Regulatory Commission requirements for the existence and operational use of such a facility at any nuclear plant site. There are no limiting meteorological factors. The details of the site area meteorology are discussed in Section 2.3. Despite the low probability of tornado occurrence at the site, the design of plant Seismic Category I structures includes consideration of the effects of a tornado having winds of 300 mph rotational velocity plus 60 mph translational velocity and a 3 psi pressure differential in 3 seconds.

1.2.1.4 Hydrology

Plant grade is Elevation 728 and the plant is designed for safe shutdown for floods exceeding plant grade level. The probable maximum flood could reach Elevation 738.8. Capability to maintain the plant in the safe shutdown condition is provided for the design basis flood elevations given in Section 2.4.14.1.1. The probability of this combination in any given year is near zero and its recurrence interval is near infinity.

Because of the contours of the land and strata there is little likelihood of abnormal releases of liquid wastes at the plant contaminating industrial or drinking water supplies derived from ground water sources.

1.2.1.5 Geology

The Watts Bar Nuclear Plant (WBN) is located in the Valley and Ridge Province of the Appalachian Highlands. This province is made up of a series of folded and faulted

mountains and valleys which are underlain by Paleozoic sedimentary formations totaling 40,000 feet in thickness. The plant site is situated in a bend of the Tennessee River that has been covered by alluvial terrace deposits. Beneath these deposits lies the Middle Cambrian Conasauga Formation, an interbedded shale and limestone unit upon which the Category I structures are founded.

The controlling feature of the geologic structure at the site is the Kingston thrust fault, which developed some 250 million years ago. The fault has been inactive for many millions of years, and recurrence of movement is not expected. The fault lies to the northwest of the site area and was not involved in the foundation for any of the major plant structures.

1.2.1.6 Seismology

WBN was designed based on the largest historic earthquake to occur in the Southern Appalachian Tectonic Province - the 1897 Giles County, Virginia earthquake. This earthquake is estimated to have had a body wave magnitude (m_b) of 5.8. The Safe Shutdown Earthquake (SSE) for the plant has been established as having a maximum horizontal acceleration of 0.18g and a simultaneous maximum vertical acceleration of 0.12g.

1.2.2 Facility Description

1.2.2.1 Design Criteria

The design criteria for the WBN are discussed in Section 3.1.

1.2.2.2 Nuclear Steam Supply System (NSSS)

The NSSS consists of a reactor and four closed reactor coolant loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump, a steam generator, loop piping, and instrumentation. The NSSS also contains an electrically heated pressurizer and certain auxiliary systems.

High pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated water exits the reactor vessel and passes via the coolant loop piping to the steam generators. Here it gives its heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel.

The inherent design of the pressurized water, closed-cycle reactor minimizes the quantities of fission products released to the atmosphere. Three barriers exist between the fission product accumulation and the environment. These are the fuel cladding, the reactor vessel and coolant loops, and the reactor 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 pressure vessel, loops, and auxiliary systems. Breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor

containment is designed to adequately retain these fission products under the most severe accident conditions, as analyzed in Chapters 6 and 15.

The license application NSSS power level is 3,425 MWt which includes 16 MWt from the reactor coolant pumps. Operation at the core design rating of 3,411 MWt yields a steady state core average linear power of 5.45 kW/ft and a corresponding peak power of 13.1 kW/ft. Reactivity coefficients and other design parameters, which are supported by analysis and experience with other similar plants, provide the basis for concluding that this reactor can be operated safely at the power levels of the application rating. The initial core load has a negative moderator temperature coefficient of reactivity at operating temperature at all times throughout core life.

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding the acceptable fuel damage limits. The core design, together with process and residual heat removal (RHR) systems, provides for this capability under expected conditions of normal operations with appropriate margins for uncertainties and anticipated transient situations, including, as examples, the effects of the loss of reactor coolant flow, turbine trips due to steam and power conversion system malfunctions, and loss of external electrical load. Acceptable fuel damage limits can be found in Section 4.2.

The reactor core is a multi-region cycled core. The fuel rods are cold worked ZIRLO® tubes containing slightly enriched uranium oxide fuel. The fuel assembly is a canless type with the basic assembly consisting of the guide thimbles mechanically fastened to the grids, top, and bottom nozzles. The fuel rods are held in the grids by spring clips. The internals, consisting of the upper and lower core support structures, are designed to support, align, and guide the core components, direct the coolant flow and guide the in-core instrumentation. Dissolved boric acid is used as a reactivity control device to minimize the use of burnable absorbers.

Rod cluster control assemblies (RCCAs) and burnable absorber rods are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the RCCAs are fabricated of silver-indium-cadmium alloy slugs sealed in stainless steel tubes. The absorber material in the burnable absorber rods is in the form of borosilicate glass sealed in stainless steel tubes. The control rod drive mechanisms for the RCCAs are of the magnetic jack type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the RCCA is released and falls into the core by gravity to shut down the reactor.

Pressure in the system is controlled by the pressurizer, where system pressure is maintained through the use of electrical heaters and water sprays. Steam can either be formed by the heaters, or condensed by the spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the reactor coolant system is described in Chapter 7. Spring-loaded steam safety valves and power-operated relief valves for overpressure protection are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

The reactor coolant pumps are Westinghouse vertical, single stage, mixed flow pumps of the shaft-seal type. The power supply system to the pumps is designed so that adequate coolant flow is maintained to cool the reactor core under credible circumstances.

The steam generators are Westinghouse vertical U-tube units which contain Inconel tubes. Integral moisture separation equipment reduces the moisture content of the steam to one-quarter of one percent or less.

The reactor coolant piping and the pressure-containing and heat transfer surfaces in contact with reactor water are stainless steel clad except the steam generator tubes and fuel tubes, which are Inconel and ZIRLO® respectively. Reactor core internals, including control rod drive shafts, are stainless steel.

Auxiliary system components are provided to charge the RCS and add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove decay heat when the reactor is shutdown, and provide for emergency safety injection.

1.2.2.3 Control and Instrumentation

Instrumentation and controls are provided to monitor and maintain essential reactor facility operating variables such as neutron flux, primary coolant pressure, temperature, and control rod positions within prescribed ranges.

The non-neutronic process and containment instrumentation measures temperatures, pressure, flows, and levels in the RCS, steam systems, containment, and auxiliary systems. The quantity and types of process instrumentation provided are adequate for safe and orderly operation of systems and processes over the full operating range of the plant.

Reactor protection is achieved by defining a region of reactor power and coolant conditions allowed by the principal tripping functions: the overpower ΔT trip, the overtemperature ΔT trip, and the nuclear overpower trip. The allowable operating region within these trip settings is designed to prevent any combination of power, temperatures, and pressure which would result in reducing Departure from Nucleate Boiling below the minimum Departure from Nucleate Boiling Ratio (DNBR) (Chapter 4, Table 4.1-1) . Additional tripping functions such as a high-pressurizer pressure trip, low-pressurizer pressure trip, high-pressurizer water-level trip, low reactor coolant flow trip, reactor coolant pump undervoltage and under frequency trips, steam generator low-low water-level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, neutron flux rate trips, and manual trip are provided to support the principal tripping functions for specific accident conditions and mechanical failures. Independent and redundant channels are combined in logic circuits which improve tripping reliability and minimize trips from spurious causes. Protection interlocks, initiation signals to the Safety Injection System, containment isolation signals, and turbine runback signals further assist in plant protection during operation.

The control system enables the nuclear plant to accept a step-load increase of 10% and a ramp increase of 5% per minute within the load range of 15% to 100% of nominal power. The control system is designed for a 50% load reduction with steam bypass without tripping the reactor.

1.2.2.4 Fuel Handling System

The fuel handling system is divided into two areas; the reactor cavity, which is flooded for refueling; and the Auxiliary Building which is external to the reactor containment and is always accessible to plant personnel. The two areas are connected by a fuel transfer system which carries the fuel through an opening in the reactor containment.The fuel handling equipment is designed to handle the new and spent fuel from the time it enters the site.

New fuel assemblies are removed one at a time from the shipping cask and stored dry in fuel storage racks or placed directly into the spent fuel pool. New fuel is delivered to the reactor vessel by placing a fuel assembly into the new fuel elevator, lowering it into the transfer canal, taking it through the fuel transfer system and placing it in the core by the use of the refueling machine. Spent fuel is removed from the reactor vessel by the refueling machine and placed in the fuel transfer system. In the spent fuel pool, the fuel is removed from the transfer system and placed in the spent fuel storage racks.

Spent fuel is handled entirely underwater. Underwater transfer of spent fuel provides an effective, economic and transparent shield, as well as a reliable cooling medium for removal of decay heat.

1.2.2.5 Waste Processing System

The Waste Processing System provides equipment necessary for controlled treatment, and preparation for retention or disposal of liquid, gaseous, and solid wastes produced as a result of reactor operation. The Liquid Waste System collects, processes, and recycles reactor grade water, removes or concentrates radioactive constituents and processes them until suitable for release or shipment offsite.

The gaseous waste processing system functions to remove fission product gases from the reactor coolant. The system also collects the gases generated from the boron recycle evaporator. The waste processing systems, including both liquid and gas, are designed to ensure that the quantities of radioactive releases from the total plant to the surrounding environment will not exceed the 10 CFR 20 limits and are as low as reasonably achievable (ALARA).

1.2.2.6 Steam and Power Conversion System

The steam and power conversion system consists of a turbine-generator, main condenser, vacuum pumps, turbine seal system, turbine bypass system, hot well pumps, condensate booster pumps, main feed pumps, main feed pump turbines (MFPT), condenser feedwater heater, feedwater heaters, heater drain tank pumps, and condensate storage system. The system is designed to convert the heat produced in the reactor to electrical energy through conversion of a portion of the energy

contained in the steam supplied from the steam generators, to condense the turbine exhaust steam into water, and to return the water to the steam generator as feedwater.

The turbine generator unit consists of a tandem arrangement of one double-flow high-pressure turbine and three double-flow low-pressure turbines driving a direct-coupled generator at 1800 RPM. The generator has a nameplate rating of 1,411,000 kVA at 0.9 PF with 75 psi hydrogen pressure. The unit employs a single pass surface condenser of a horizontal, triple pressure, single shell type. Return to the steam generator is through three stages of feedwater pumping and seven stages of feedwater heating. Safety relief valves and power operated relief valves, as well as a turbine bypass to the condenser are provided in the steam lines.

1.2.2.7 Plant Electrical System

The plant electric power system consists of the main generators, the unit station service transformers, the common station service transformers, the diesel generators, the batteries, and the electric distribution system. Under normal operating conditions the main generators supply electrical power through isolated-phase buses to the main step-up transformers and through the unit station service transformers (located adjacent to the Turbine Building) to the nonsafety auxiliary power system. Offsite electrical power supplies Class 1E circuits through the 161-kV system via Common Station Service Transformers (CSST) C and D. The primaries of the unit station service transformers are connected to the isolated-phase bus at a point between the generator terminals and the low-voltage connection of the main transformers. During normal operation, station auxiliary power is taken from the main generator through the unit station service transformers and from the 161-kV system through the common station service transformers. The standby onsite power is supplied by four diesel generators.

The safety-related plant distribution system receives ac power from CSST C and D through the shutdown boards (which are powered from the offsite power system), or four 4400 kW diesel-generator standby (onsite) power sources, and distributes it to both safety-related and nonsafety-related loads in the plant. The two preferred circuits have access to the TVA transmission network which in turn has multiple interties with other transmission networks.

The safety-related loads for the plant are divided into two redundant load groups. Each redundant load group has access to each of the two preferred offsite sources. One load group with its two associated diesel generators can provide the safety functions. The electrical systems are described in Sections 8.2 and 8.3.

The vital ac and dc control and instrument power system consists of five 125V batteries (fifth vital battery can be switched for any of the other four), eight battery chargers (two pairs of spares), and twelve 120V ac inverters (four spares) with their respective safety-related loads. The 125V dc Distribution System is a safety-related system which receives power from independent battery chargers and 125V dc batteries and distributes it to safety-related loads. The 120V ac Distribution System receives power from eight independent inverters and distributes it to the safety-related loads of both units. These systems are described in Sections 8.2 and 8.3.

1.2.2.8 Cooling Water

The condenser circulating water system (CCW) provides cooling water for the dissipation of waste heat for the power generation cycle while meeting applicable effluent limitations and water quality standards. The CCW includes the circulating water pumps, circulating water conduits, yard holding pond, main condensers, hyperbolic natural draft cooling towers, and the desilting basin and the supplemental condenser cooling water (SCCW) system. The SCCW system supplies water from the Watts Bar Reservoir to provide a source of cooler water to the existing Unit 2 cooling tower discharge flume.

The blowdown from the CCW is used to dilute and dispense low-level radioactive liquid wastes. The CCW pumping station is located in the yard between the Turbine Building and the cooling towers. There are eight circulating pumps. Four pumps for each unit operate in parallel and circulate water from the cooling tower cold water basin, through the condenser, and back into the heat exchanger section of the tower.

The essential raw cooling water system (ERCW) provides the essential auxiliary support functions to the engineered safety features (ESF) of the plant. The system is designed to provide a continuous flow of cooling water to those systems and components necessary to plant safety either during normal operation or under accident conditions. The ERCW system consists of eight ERCW pumps, four traveling screens, four traveling screen wash pumps, and four strainers located in the intake pumping station.

1.2.2.9 Component Cooling System

The component cooling system (CCS) is the closed cooling system designed to remove residual and sensible heat from the RCS, via the RHR system; cool the spent fuel pool water and the letdown flow of the Chemical and Volume Control System (CVCS); provide cooling to dissipate waste heat from various plant components; and provide cooling for safeguard loads after an accident.

1.2.2.10 Chemical and Volume Control System

The CVCS, discussed in Section 9.3.4, is designed to provide the following services to the RCS:

(1) Maintenance of programmed water level in the pressurizer, i.e., maintain required water inventory in the RCS.

(2) Maintenance of seal water flow to the reactor coolant pumps.

(3) Control of reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration and makeup.

(4) Processing of excess reactor coolant to effect recovery and reuse of boric acid and primary makeup water. This operation is not performed by CVCS for Unit 1. Liquid waste will be processed through the waste disposal mobile demineralizer.

(5) Emergency core cooling (part of the system is shared with the Emergency core cooling system).

During power operation, a continuous feed-and-bleed stream is maintained to and from the RCS. Letdown water leaves the RCS and flows through the shell side of the regenerative heat exchangers where it gives up its heat to makeup water being returned to the RCS. The letdown water then flows through the orifices where its pressure is reduced, then through the letdown heat exchanger, followed by a second pressure reduction by a low-pressure letdown valve. After passing through a mixed bed demineralizer, where ionic impurities are removed, the water flows either through the cation demineralizers or directly through the reactor coolant filter, and into the volume control tank (VCT) via a nozzle. The vapor space in the VCT contains hydrogen which dissolves in the coolant. Any fission gases present are removed from the system by venting of the VCT when required.

The charging pumps take the coolant from the VCT and send it along two parallel paths: 1) to the RCS through the tube side of the regenerative heat exchangers; and 2) to the seals of the reactor coolant pumps. The streams divide with some water flowing into the RCS and the remainder leaving the pumps as seal leakage. From the pumps, the leakage water goes to the seal water heat exchanger and then returns to the VCT for another circuit. If the normal letdown and charging path through the regenerative heat exchanger is not operable, water injected into the RCS through the reactor coolant pump seals is returned to the VCT through the excess letdown heat exchanger.

Surges from the RCS accumulate in the VCT unless a high water level in the tank causes flow to be diverted to the Boron Recycle or waste processing systems.

Makeup to the CVCS comes from the following sources:

(1) Demineralized water supply, when the concentration of dissolved neutron absorber is to be reduced.

(2) Boric acid tank, when the concentration of dissolved neutron absorber is to be increased.

(3) A blend of demineralized water and concentrated boric acid to match the reactor coolant boron concentration for normal plant makeup.

(4) Refueling water storage tank for emergency makeup of borated water.

The chemical mixing tank is used to inject small quantities of hydrazine for oxygen scavenging or lithium hydroxide for pH control.

1.2.2.11 Sampling and Water Quality System

The sampling and water quality system provides the equipment necessary to provide required process samples for laboratory analysis. These analyses provide the essential chemical and radiochemical data required for the operation of the various process systems in each of the two units.

1.2.2.12 Ventilation

The internal environments of the various buildings of the plant are controlled within acceptable limits for safety, comfort, and equipment protection by several heating, cooling, and ventilating systems. Filtration is provided in exhaust systems as required to reduce contaminants.

Heating systems involve both electric and hot water systems while cooling systems utilize fan coil units supplied with direct expansion, chilled water, or raw water coils.

Ventilation is by both supply and exhaust with central intakes and exhausts for proper treatment of the air.

Redundant equipment is provided for safety related equipment.

1.2.2.13 Fire Protection System

The fire protection systems are designed to provide early detection and suppression of fires. The suppression systems provide a reliable water and Co_2 distribution system to control and extinguish fires both inside and outside the buildings. The water base suppression systems consist of pumps, headers, yard hydrants, automatic spray systems for outdoor transformers, automatic sprinkler system inside buildings and hose stations at strategic locations within the buildings.

The Co_2 systems consist of storage tanks, supply headers, and suppression system piping and nozzles for protection inside the buildings.

1.2.2.14 Compressed Air Systems

The compressed air system is common to both units and is divided into three subsystems: the station control and service air system, and two auxiliary control air systems for emergency use. The station control and service air system, supplies compressed air for general plant service, instrumentation, testing, and control. The auxiliary control air systems provide, as a minimum, sufficient air for an orderly plant shutdown under conditions such as safe shutdown earthquake and maximum possible flood. Only the auxiliary control air systems are considered to be Engineered Safety Features. For detailed description see Section 9.3.1.

1.2.2.15 Engineered Safety Features

Several ESF have been incorporated into the plant design to reduce the consequences of a loss-of-coolant accident (LOCA). One of these safety features is an emergency core cooling system (ECCS) which automatically delivers borated water to the reactor core via the cold legs to cool it under high and low reactor pressure conditions and inserts negative reactivity during plant cooldown following a steam line rupture or other accidental steam release. Another safety feature is the ice condenser containment system. Basically, this system involves the very rapid absorption of the energy released from the RCS in the improbable event of a LOCA. The energy is absorbed by condensing the steam in a low temperature heat sink, consisting of a suitable quantity of ice permanently stored, in a cold storage compartment, inside the containment. This containment system results in markedly reducing the peak pressure

that would result in the containment in the event of LOCA and reduces this peak to an even lower value within a few minutes. The system also removes iodine radioactivity from the containment atmosphere by the action of sodium tetraborate impregnated ice.

There are several other systems which help mitigate the consequences of a LOCA by aiding the systems mentioned above or by the performance of other specific functions. The containment spray system sprays cool water into the containment atmosphere to ensure that the containment pressure limit is not exceeded. The air return fans also aid in the operation of the containment spray system and the ice condenser by circulating air from the upper compartment of the containment through the ice condenser. This system also limits hydrogen concentration by ensuring a flow of air in potentially stagnated regions. The containment isolation systems maintain containment integrity by isolating fluid systems that pass through the containment. The radioactivity that may be released in the containment will be confined there by this system.

To help reduce radioactive nuclide releases to the atmosphere this plant is provided with gas treatment systems. The emergency gas treatment system (EGTS) and the Auxiliary Building gas treatment system (ABGTS) establish and maintain the air pressure below atmospheric in the Shield Building annulus and the Auxiliary Building secondary containment enclosure (ABSCE), respectively. These systems also reduce the concentration of radioactive nuclides in the air released from the annulus and the ABSCE.

1.2.2.16 Shared Facilities and Equipment

Separate and similar safety-related systems and equipment are provided for each unit of the two unit Watts Bar Nuclear Plant except as noted below. In those instances where some components of a safety-related system are shared by both units, only those major components which are shared are shown. Also listed are major components of the non-safety-related radioactive waste disposal system.

	System/Components	Number Shared
a.	Chemical and Volume Control System	
	Boric Acid Tanks	3
	Boric Acid Transfer Pumps	4
	Hold-up Tanks	2
	Gas Stripper Feed Pumps	3
b.	Component Cooling System (only the train B components are shared)	
	Component Cooling Heat Exchangers (all components)	1
	Component Cooling Water Pumps	3
	Component Cooling Surge Tanks	2

System/Components	Number Shared
c. Spent Fuel Pit Cooling and Cleaning System (whole system is shared)	
d. Fuel Handling System	
Spent Fuel Storage Pit	1
New Fuel Storage Area	1
Decontamination Area	1
Spent Fuel Pit Bridge	1
e. Plant Fire Protection System	1
High Pressure Fire Protection Pumps	4
f. Cooling Water System	
Essential Raw Cooling Water Pumps	8
Traveling Water Screens	4
Screen Wash Pumps	4
Strainers	4
g. Radioactive Waste Disposal System	1
Tritiated Drain Collector Tank	1

	System/Components	Number Shared
	Tritiated Drain Collector Tank Pumps	2
	Floor Drain Collector Tank	1
	Floor Drain Collector Tank Pumps	3
	Monitor Tank	1
	Monitor Tank Pumps	2
	Laundry & Hot Shower Tank	1
	Laundry & Hot Shower Tank Pump	1
	Cask Decontamination Collector Tank	1
	Cask Decontamination Collector Tank Pumps	2
	Waste Condensate Tanks	3
	Waste Condensate Tank Pumps	2
	Spent Resin Storage Tank	1
	Chemical Drain Tank	1
	Chemical Drain Tank Pump	1
	Waste Gas Compressor Packages	2
	Waste Gas Decay Tanks	9
	Nitrogen Supply	1
h.	Emergency Gas Treatment System	
	Air Cleanup Units	2
i.	Auxiliary Building Gas Treatment System	
	Air Cleanup Units	2
j.	Control Building Main Control Room HVAC and Pressurizing Air System (whole system is shared)	
k.	Control Building Electrical Board Room HVAC System (whole system is shared)	
l.	Auxiliary Building Shutdown Board Room HVAC System (whole system is shared)	

	System/Components	Number Shared
m.	Fuel Oil System (for each Diesel Generator)	4
n.	Electrical System Train A and Train B	2
	Diesel Generator Systems	4
	Normal Auxiliary Power System	1
	Class 1E DC Systems	4
o.	Structures, Building, and Miscellaneous	
	Control Building	
	Auxiliary Building	
	Service Building	
	Intake Pumping Station	
	Auxiliary Control Air Subsystem	1
	Auxiliary Control Air Subsystem Compressors	2
	Plant Heating Steam System	1
	Makeup Water Supply and Treatment System	
p.	Flood Mode Boration Makeup System	
	Auxiliary Boration Makeup Tank	1
	Auxiliary Charging Booster Pumps	2
	Flood Mode Boration Demineralizer	1
	Flood Mode Boration Filters	2

In the main text, it is stated in each system description whether that system is provided as either (a) a common facility which is shared by the two units, or (b) as a separate identical facility for each unit.

1.2.3 General Arrangement of Major Structures and Equipment

The major structures are two Reactor Buildings, a Turbine Building, an Auxiliary Building, a Control Building, a Service and Office Building, Diesel Generator Buildings, an Intake Pumping Station, and two natural draft cooling towers. The arrangement of these structures is shown in Figure 2.1-5. Plant arrangement plans and cross sections are presented in Figures 1.2-1 through 1.2-15.

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Figure 1.2-1 Powerhouse Units 1 & 2 Equipment Plans - Roof

Figure 1.2-2 Powerhouse Units 1 & 2 Equipment Plan - EL. 772.0 and Above

Figure 1.2-3 Powerhouse Units 1 & 2 Equipment Plan - EL. 757.0 and EL. 755.0

Figure 1.2-4 Powerhouse Units 1 & 2 Equipment Plan - EL. 737.0 and EL. 729.0

Figure 1.2-5 Powerhouse Units 1 & 2 Equipment Plan -EL. 713.0 and EL. - 708.0

Figure 1.2-6 Powerhouse Units 1 & 2 Equipment Plan - EL. 692.0 and EL. 685.5

Figure 1.2-7 Powerhouse Units 1. & 2 Equipment Plan -EL. 750.5, EL. 730.5,' EL. 676.0, EL. 670.0, and EL. 666.0

Figure 1.2-8 Powerhouse Units I & 2 Equipment Transverse Section A8-A8

Figure 1.2-9 Powerhouse Units I & 2 Equipment Longitudinal Section B9-B9

Figure 1.2-10 Powerhouse Units 1 & 2 Equipment Longitudinal Section CIO-C10

Figure 1.2-11 Powerhouse Units I & 2" Equipment Reactor Building - Plan Upper and Lower Compartments

Figure 1.2-12 Powerhouse Units I & 2 Equipment Reactor Building -Plan EL. 674.69, and 702.78 and Above

Figure 1.2-13 Powerhouse Units I & 2 Equipment Reactor. Building -Section D13-D13

Figure 1.2-14 Powerhouse Units I & 2 Equipment Reactor Building -Section E14-E14

Figure 1.2-15 Powerhouse -CDWE Building Units 1 & 2 -Mechanical General Arrangement Condensate Demineralizer Waste Evaporator Building -Equipment -Elevation 730.5 and 750.5

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1.3 COMPARISON TABLES

1.3.1 Comparisons With Similar Facility Designs

Table 1.3-1 presents a design comparison of the Watts Bar Nuclear Steam Supply System design with that of Sequoyah, Donald C. Cook, and Trojan. Table 1.3-2 presents a detailed design comparison of the Watts Bar Nuclear Plant Secondary Cycle with that of Sequoyah, D. C. Cook, and Zion.

1.3.2 Comparison Of Final And Preliminary Designs

Table 1.3-3 lists the significant design changes that have been made since the submittal of the Preliminary Safety Analysis Report.

Table 1.3-1 DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)
Inspections, Tests, Analyses and Acceptance Criteria Nuclear Plant Units 1 and 2 -
Comparison with Donald C. Cook, Trojan, and Sequoyah (Sheet 1 of 5)

Chapter Number	Chapter Title System/component	References (FSAR)	Significant Similarities	Significant Differences
3.0	Containment	Section 3.8.2	D. C. Cook, Sequoyah	Watts Bar and Sequoyah use of freestanding steel primary containment vessel.
4.0	Reactor Fuel	Section 4.2.1	Trojan, Sequoyah	None.
	Reactor Vessel Internals	Section 4.2.2	D. C. Cook, Sequoyah, Trojan	D. C. Cook Units 1 and 2 and Sequoyah Units 1 and 2 have thermal shields. Trojan has neutron pads. Sequoyah and Watts Bar have the inverted top hat upper internals design.
	Reactivity Control	Section 4.2.3	D. C. Cook, Sequoyah, Trojan	None.
	Nuclear Design	Section 4.3	D. C. Cook, Sequoyah, Trojan	None.
	Thermal-Hydraulic Design	Section 4.4	D. C. Cook, Sequoyah, Trojan	The total primary heat output and coolant temperatures are higher for Sequoyah, Watts Bar, and Trojan than for D. C. Cook Plant.
5.0	Reactor Coolant System	Sections 5.1, 5.2	D. C. Cook, Sequoyah, Trojan	The following have been added or changed for Sequoyah and Watts Bar; New requirements for fracture toughness testing, New means of determining heat-up and cool-down rates.
	Reactor Vessel*	Section 5.4	D. C. Cook, Sequoyah, Trojan	None.

Table 1.3-1 DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)
Inspections, Tests, Analyses and Acceptance Criteria Nuclear Plant Units 1 and 2 - Comparison with Donald C. Cook, Trojan, and Sequoyah (Continued) (Sheet 2 of 5)

Chapter Number	Chapter Title System/component	References (FSAR)	Significant Similarities	Significant Differences
5.0 (Cont'd)	Reactor Coolant Pumps*	Section 5.5.1	D. C. Cook, Sequoyah, Trojan	None.
	Steam Generators*	Section 5.5.2	D. C. Cook, Sequoyah, Trojan	None.
	Piping*	Section 5.5.3	D. C. Cook, Sequoyah, Trojan	None.
	Residual Heat Removal System	Section 5.5.7	D. C. Cook, Sequoyah, Trojan	None.
	Pressurizer*	Section 5.5.10	D. C. Cook, Sequoyah, Trojan	None.
6.0	Engineered Safety Features			
	Emergency Core Cooling System	Section 6.3	D. C. Cook, Sequoyah, Trojan	None.
	Ice Condenser	Section 6.7	D. C. Cook, Sequoyah	Trojan does not use an ice condenser.
7.0	Instrumentation and Controls			

Table 1.3-1 DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)
Inspections, Tests, Analyses and Acceptance Criteria Nuclear Plant Units 1 and 2 - Comparison with Donald C. Cook, Trojan, and Sequoyah (Continued) (Sheet 3 of 5)

Chapter Number	Chapter Title System/component	References (FSAR)	Significant Similarities	Significant Differences
7.0 (Cont'd)	Reactor Trip System	Section 7.2	System functions are similar to D. C. Cook, Sequoyah, Trojan	Sequoyah and Watts Bar have a Westinghouse EAGLE 21 digital Process Protection System; Trojan and D. C. Cook use an analog system. Sequoyah's low-low steam generator level trip function is processed through an environmental allowance modifier/trip time delay (EAM/TTD) functional algorithm in the EAGLE 21 system. This allows a lower low-low level setpoint when an adverse containment environment does not exist as determined by monitoring containment pressure. Watts Bar uses the TTD without EAM.
	Engineered Safety Features System	Section 7.3	System functions are similar to D. C. Cook, Sequoyah, Trojan	None.
	Systems Required For Safe Shutdown	Section 7.4	System functions are similar to D. C. Cook, Sequoyah, Trojan	None.
	Safety Related Display Instrumentation	Section 7.5	Parametric display is similar to that of D. C. Cook, Sequoyah, Trojan	Actual physical configuration may differ due to customer design philosophy.
	Other Safety Systems	Section 7.6	Operational Functions are similar to D. C. Cook, Trojan, Sequoyah	None.

Table 1.3-1 DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)
Inspections, Tests, Analyses and Acceptance Criteria Nuclear Plant Units 1 and 2 - Comparison with Donald C. Cook, Trojan, and Sequoyah (Continued) (Sheet 4 of 5)

Chapter Number	Chapter Title System/component	References (FSAR)	Significant Similarities	Significant Differences
7.0 (Cont'd)	Control Systems	Section 7.7	Operational Functions are similar D. C. Cook, Trojan, Sequoyah	The Sequoyah Nuclear Plant has a 50% load rejection capability while that of the D. C. Cook Plant is 100%. The rod position indication for the Sequoyah Nuclear Plant and the D. C. Cook Plant is an analog system; Trojan's RPI is a digital system.
8.0	Electric Power			
	Offsite Power	8.2	Sequoyah - 2 offsite sources 161 kV/6.9 kV	None
	Onsite Power	8.3	Sequoyah - Tandem diesel generator arrangement	Sequoyah diesel generator rated at 4000 kW. Watts Bar diesel generator rating is 4400 kW.
			Sequoyah - Four 125V dc batteries for supplying vital dc power	None
9.0	Auxiliary Systems			
	Chemical and Volume Control System	Section 9.3.4	D. C. Cook, Trojan, Sequoyah	The Sequoyah and Watts Bar do not have deboration demineralizers.
11.0	Radioactive Waste Management			
	Source Terms	Section 11.1	D. C. Cook, Trojan, Sequoyah	Differences are based upon plant operational influences.

Table 1.3-1 DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)
Inspections, Tests, Analyses and Acceptance Criteria Nuclear Plant Units 1 and 2 - Comparison with Donald C. Cook, Trojan, and Sequoyah (Continued) (Sheet 5 of 5)

Chapter Number	Chapter Title System/component	References (FSAR)	Significant Similarities	Significant Differences
11.0 (Cont'd)	Liquid Waste Processing	Section 11.2	Performance characteristics similar to D. C. Cook, Trojan, Sequoyah	The Sequoyah and Watts Bar have similar segregated liquid drain systems.
	Gaseous Waste Processing	Section 11.3	D. C. Cook, Trojan, Sequoyah	None.
	Solid Waste Processing	Section 11.5	Functionally similar to D. C. Cook, Trojan, Sequoyah	None.
15.0	Accident Analysis	Chapter 15	Similar to D. C. Cook, Trojan	The Accident Analysis sections have been updated. New sections have been added, e.g., single RCCA withdrawal, accidental depressurization of the RCS, compare code descriptions, etc.

* All components designed and manufactured to Code edition in effect at date of purchase order.

Table 1.3-2 DESIGN COMPARISON - SECONDARY CYCLE
(Sheet 1 of 2)

Feature	Referenced FSAR Section	Sequoyah Nuclear Plant	Watts Bar Nuclear Plant	D. C. Cook	Zion
Turbine Generator					
Net Generator Output (kW)	10.1, 10.2	1,183,192	1,218,225	1,100,000	1,050,000
Turbine Cycle Heat Rate (Btu/kW-Hr)	10.1	9,871	9,593	*10,208;**10,232	***
Type/LSB Length	10.2	TC6F/44	TC6F/44	*TC6F/43; **TC6F/52	TC6F/44
Steam Conditions at Throttle Valve					
Flow (lb/hr)	10.2	14,254,200	15,143,600	14,120,000	13,989,300
Pressure (psia)	10.2	832	1000	728	690
Temperature (°F)	10.2	522.7	544.6	507.5	501.5
Moisture Content (%)	10.1, 10.2	0.34	0.39	N/A	.25
Turbine Cycle Arrangement					
Steam Reheat Stages (No.)	10.1	2	2	1	1
Feedwater Heating Stages (No.)	10.1,10.4.7,	7	7	6	6
Strings of Feedwater Heaters (No.)	10.4.9	3	3	3 Lowest Pressure 2 All Others	3
Heaters in Condenser Neck (No.)	10.1,10.4.7, 10.4.9	3	3	0	1
Heater Drain System Type	10.4.9	All Drains Pumped Forward	High Pressure Pumped Forward; Low Pressure Cascaded	High Pressure Pumped Forward; Low Pressure Cascaded	High Pressure Pumped Forward; Low Pressure Cascaded
Hotwell Pumps (No.)	10.1, 10.4.7	3	3	3	4
Condensate Booster Pumps (No.)	10.1, 10.4.7	3	3	3	4
Heater Drain Pumps (No.)	10.1, 10.4.10	3 H.P.- 3 L.P.	3 H.P. - 2 L.P.	3	3
Main Feed Pumps (No. & Type)	10.1	2-Turbine Driven	2-Turbine Driven 1 Motor Driven	2-Turbine Driven	2-Turbine Driven

Table 1.3-2 DESIGN COMPARISON - SECONDARY CYCLE
(Sheet 2 of 2)

Feature	Referenced FSAR Section	Sequoyah Nuclear Plant	Watts Bar Nuclear Plant	D. C. Cook	Zion
Main Steam Bypass Capacity (%)	10.4.4	40%	40%	85%	40%
Final Feedwater Temperature	10.4.7	434.3	441.6	*434.8;**430.5	NA
Condenser					
Type	10.1, 10.4.1	Single Pressure	Three Pres. Zone	Single Pressure	Single Pressure
Number of Shells	10.1, 10.4.1	3	3	3	3
Design Back Pressure (In. Hg Abs)	10.1, 10.4.1	2	1.63, 2.38, 3.40	*1.71; **1.41	1.5
Total Condenser Duty (Btu/Hr)	10.1, 10.4.1	7.829×10^9	7.789×10^9	2.5×10^9 (Approx.)	7.18×10^9 (Approx.)

* Unit 1

** Unit 2

*** Commonwealth Edison will not release these heat rates.

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Sheet 1 of 9)

System	Reference Section	Section Changes
Containment Ice Condenser	6.7	Design of the following has been modified:
		(1) Ice Baskets
		(2) Lower inlet door and hinges
		(3) Lower support structure
		(4) Lattice frames
		(5) Lattice frame support columns
		(6) Wall panels
		(7) Intermediate deck floors
		(8) Top deck doors
		(9) Air handling unit supports
		(10) Top deck beams
		(11) Ice condenser crane, crane rail, and supports
		(12) (12)Stud material and diameter in containment, end walls, and crane wall
		(13) Number or air handling units
		(14) Number of refrigeration packages and associated hardware
		The following have been deleted:
		(1) Floor air-cooling duct
		(2) Lower section of outer three rows of ice basket
		(3) Access platform to lower inlet doors

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 2 of 9)

System	Reference Section	Section Changes
Containment Ice Condenser (Cont'd)	6.7	The following have been added: (1) Ice basket tie-down (2) Lattice frame tangential-tie-member (3) Closer spacing of lattice frames (4) Lower inlet door arrester (5) Turning vanes on lower support structure and floor (6) Jet impingement plate (7) Foam concrete in floor (8) Glycol cooling of floor (9) Defrosting capability of wall panels and floor (10) Floor support columns (11) Wall panel cradle (12) Rounded entrance to lower doors
Containment Spray	6.2	Separate Containment Spray Systems suction lines have been routed to the containment sump. The containment spray pumps design flow rate has been increased to 4000 gpm. Check valves have been added to the containment spray pumps discharge header.
Fuel	4.2.1	Unit 1 will be fueled with recaged VANTAGE 5H 17 x 17 fuel assemblies in lieu of 15 x 15 fuel assemblies.
Reactor Internals	4.2.2	The reactor internals have been modified to accept 17 x 17 fuel assemblies.

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 3 of 9)

System	Reference Section	Section Changes
Emergency core cooling	6.3	Safety injection pumps will normally inject into the four cold legs of the reactor coolant system but provision for injection into the hot legs has been retained.
AC Power	8.1	Two additional RCP start buses were added to feed the 8 7000HP reactor coolant pumps. The RCP's were originally proposed to be powered from the 6.9 kV unit boards.
		The 12 69kV - 480 shutdown transformers were changed from a 1500 kVa rating to 2 2000 kVa rating.
		An additional 480V intake pumping station board and two 2000 kVa transformers were added at the intake pumping station.
Diesel Generator	8.4	The four diesel generators were each up graded from a 4000 KW rating to 4400 KW.
	3.8	The Diesel Generator Building was strengthened to withstand the additional required missile spectrum.
Onsite DC power	8.3	Battery test equipment has been added to vital batteries.
Essential raw cooling water	9.2.1	Missile barriers are added to the pump deck.
CO_2 fire protection	9.5.1	CO_2 storage has been moved to a storage vault.
Main steam supply	10.3	Main steam isolation valves are uni-directional. Check valves associated with the bidirectional valves have been removed.
Condensate-Feedwater	10.4	Auxiliary feedwater system uses modulating valves instead of off/on control valves.
		Full flow polishing condensate demineralizers were added secondary chemistry was changed to AVT.
		A standby motor driven feed pump has been added to the Main feedwater System.

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 4 of 9)

System	Reference Section	Section Changes
Steam Generator Blowdown	10.4.8	Steam generator blowdown system was redesigned from the flash tank to the condensate demineralizer system. Manual throttling valves and regulating valves were added.
Steam Generator Blowdown	10.4.8	Condensate demineralizers have been added to process blowdown.
Waste disposal	11.2	The drains have been segregated into tritiated and non-tritiated systems.
		An auxiliary waste evaporator has been provided.
	11.3	Holdup time for the gaseous waste system has been increased to 60 days.
CVCS	9.3.4	Differential pressure across the labryinth seals of the reactor coolant pumps is not alarmed. Total seal water flow is alarmed.
Post Accident Monitoring	7.5	A Post Accident Monitoring System has been added.
Source and Intermediate Range Monitors	7.2	The source and intermediate range neutron monitoring systems are replaced with seismically qualified systems to meet Reg. Guide 1.97 Rev. 2 requirements.
Process Protection System	7.0	The Foxboro analog instrumentation in the Process Protection System racks has been replaced with Westinghouse EAGLE 21 digital system. Concurrently, some functional changes were made which improve plant availability and reliability.

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 5 of 9)

System	Reference Section	Section Changes
Raw Water Corrosion Program	9.2.8	Watts Bar Nuclear Plant (WBN) has a comprehensive chemical treatment program to treat raw water systems. This new treatment is a major part of WBN Raw Water Corrosion Program. The chemical treatment is used to control corrosion in carbon steel and yellow metals, to control organic fouling, including slime, and to minimize the effect of microbiologically induced corrosion (MIC). Zinc sulfate is used as a corrosion inhibitor in the control of carbon steel corrosion. Butyl Benzotriazole is used for the corrosion protection of yellow metals. Macrofouling and microbiological control will be accomplished through the use of dodecylguanide hydrochloride (DGH) and alkydimethyl benzylammonium chloride (quat). That is, the DGH and quat are used as a non-oxidizing biocide to control Asiatic clams populations, Zebra mussels, and to prevent MIC. All raw water systems are also being treated with 1-Bromo, 3-chloro, 5, 5-dimethyl hydantoin (BCDMH). BCDMH is a biocide that replaced NaHCl that adds hypobromous and hypochlorous acid to control clams and help prevent MIC.
Auxiliary control air	9.3.1	Credit is now taken for auxiliary air system as a safety feature.
Compressed air system	9.3.1	Several portable breathing air stations have been provided.
Heating, ventilating and air conditioning	9.4.1	Ventilation, heating and air conditioning provided for the reactor auxiliary board rooms.
	9.4.2	Shutdown Board Room air conditioning system outside air is taken from intake on roof of Auxiliary Building and filtered thru HEPA filters only. In the event of an accident or high radiation signal, operator will close isolation dampers from main control room.
	9.4.3	Auxiliary Building Ventilation System is assisted by operation of the General Cooling System by providing chilled water to the building air intake coils and various strategically located air handling equipment.
	9.4.7	An annulus vacuum control subsystem was included in the emergency gas treatment system to continuously maintain the shield building annulus space at a negative pressure during plant operation.

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 6 of 9)

System	Reference Section	Section Changes
Hydrology	2.4	Implementation of New Flood Plan
Containment	3.8.1.1.1	Install new Equipment Hatch doors
Dynamic Testing and Analysis	3.9.2.5	Use New Analytical; Methods for Robust Fuel Assembly-2 (RFA-2) Upgrade
Fuel	4.3	Increase Spent Fuel Storage
	4.3.2.7	Make changes for Fuel Storage greater than 4.3% enrichment
Ice Condenser	6.1.1.2, 6.1.3.3, 6.1.3.4	Reduction of Ice Condenser Ice Weight
	6.2.1.3.3	Re-gear specific Valves in GL 89-10 Program
	6.2.3.2.1, 6.2.4.3	Install modification to allow Ice Blowing during Fuel Handling
Accident Analysis	6.2.4.2	Revise Failure Modes and Analysis Report - Use of Operator Action
Technical Specifications	6.2.4.2.3	Incorporate part of TSTF 51, Revision 2, into the Technical Specifications to Eliminate Certain ESF Operability Requirements During Core Alterations.
Containment Leak Rate Testing	6.2.6.2, 6.2.6.3, Table 6.2.6-3	Implementation of 10 CFR 50, Appendix J, Option B, Performance-Based Containment Leakage testing
RCS	7.1.2.1.9	Using Reactor Coolant System (RCS) Flow Measurement Using Elbow Tap Methodology NOTE: Unit 2 will verify reactor coolant flow Technical Specification requirements using the precision flow calorimetric methodology until sufficient data is collected to correlate elbow tap ΔP measurements with actual flow.
Reactor	4.0	New Westinghouse Fuel Assemblies
	7.2.1.1.2, Tables 7.2-1, 7.2-3, 7.2-4	Deletion of Neutron Flux Negative Rate Trip
Condensate - Feedwater	7.2.1.1.7, 7.2.2.2	Alternate method for use of Condenser Dump Valves

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 7 of 9)

System	Reference Section	Section Changes
Plant Computer	7.5.4.1.3, 7.5.1.4.4, 7.5.1.4.5, 7.5.1.6, 7.5.2, 7.5.2.1.2, 7.5.2.3.2, Table 7.5-1	P2500 and ERFDS Computer Replacement with an Integrated Computer System
Instrumentation and Controls	7.5.1.5.1, Table 7.5-2	New Containment Sump Level Transmitter
	Table 7.5-2	New Safety Injection Cold Leg Accumulator Tank Level Measurement System
	7.7.1.3.2	Alternative means for monitoring Control or Shutdown Rod position
	7.7.1.5	Eliminate Pressurizer Backup Heaters of high level signal
	7.7.1.12	ATWS Mitigation System Actuation circuitry (AMSAC) Replacement
Onsite DC Power	8.1.2	Change in the number of inverters
AC Power	8.2.2	Increase time delay setting of 6.9kV Emergency Bus degraded function from 6 to 10 seconds.
Spent Fuel Pool	9.1.3.1.1, 9.1.3.3.1, 9.1.3.3.3, Table 9.1-1	Change in Spent Fuel Pool Cooling Methodology
Fuel Handling System	9.1.4.1, 9.1.4.2, 9.1.4.2.2, 9.1.4.3.1	Fuel Handling System Upgrade
	9.1.4.1, 9.1.4.2, 9.1.4.2.2	Use of Existing New Fuel Elevator
	9.1.4.3.1	Upgrade Spent Fuel Bridge Crane
HVAC	9.2.1.2	Install Temporary Outage Cooling System
Essential Raw Cooling Water	9.2.8.2	Raw Cooling Water Discharge routed to Unit 2 Cooling Tower Flume during Plant Outages
Equipment and Floor Drainage	9.3.3.1	Replacement of Alum Sludge Pond Sump Pumps with gravity drain

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 8 of 9)

System	Reference Section	Section Changes
HVAC	9.4.3.2.5, Table 9.4-5	Install Fifth Vital Battery Room Heating and Ventilation System Modification
Containment	9.4.5.3.4, 9.4.6.1, 9.4.6.2, 9.4.6.3	Install Modification to allow for venting of the Containment into the Annulus
Auxiliary Feedwater System	10.4.9.3, Table 10.4-5	Use of Motor-Driven Auxiliary Feedwater Pump pressure switches to detect loss of CST and initiate transfer of Turbine-Driven Auxiliary Feedwater Pump supply to Essential Raw Cooling Water.
Solid Waster Management System	11.2	Solid Radwaste Disposal System
Accident Analysis	Table 11.2-4	Steam Generator Tube Rupture (SGTR), Fuel Handling Accident (FHA) and Effluent Releases Updated
Radiological Controls	11.4.2.2.2, 11.4.2.2.4, Tables 11.4-1 and 11.4-2	Deletion of Radiation Monitors
	12.3.4.2.2, Table 12.3-5	Delete Continuous Air Monitors and Install Portable monitors
Instrumentation and Controls	7.7, 5.2.7.3.2, 9.3.4 and 10.4.7.2	Installation of Foxboro IA Distributed Control System for non-safety related instrumentation and control functions. Additional changes made to eliminate selected single point failures in the previous design
	7.7	Installation of WINCISE/Power Distribution Monitoring System (Beacon)
Reactor	7.1 and 7.5	Installation of Common Q system Post Accident Monitoring system. Saturation Monitor, Core Exit Thermocouple Monitor and Reactor Vessel Level Indication
Instrumentation and Controls	Table 7.5-2, 6.2.5 and 9.3.2	Containment Hydrogen Monitor down graded to non-safety related and only 1 monitor installed.
		Hydrogen Recombiners are abandoned in place
		Post Accident Sampling System is abandoned in place

Table 1.3-3 DESIGN CHANGES SINCE SUBMITTAL OF THE PSAR
Inspections, Tests, Analyses and Acceptance Criteria (Continued) (Sheet 9 of 9)

System	Reference Section	Section Changes
	7.3	ESFAS, safety-related analog Bailey and Robert Shaw instrument and controls replaced with Foxboro Spec 200 hardware
	3.10	Replacement of Westinghouse supplied Foxboro and most Barton transmitters with Rosemount transmitters
	7.7	Installation of Westinghouse CERPI computer enhance rod position indication system
	7.6	Replacement of the Loose Parts Monitoring system with Westinghouse DIMMS-DX digital system.
	7.7	Replacement of the turbine generator and reactor coolant pump vibration monitoring system with a Bentley Nevada 3500 system
	7.5, 11.4	Replacement of the Containment High Range Radiation monitors with digital monitors
	7.5	Relocation of the Core Exit Thermocouples to the top of the Incore Instrument Thimble Assemblies and reducing the number of thermocouples from 65 to 58 as part of the WINCISE installation.
	7.0	Replacement of the control room annunciator system with a Ronan X110 Serial Controllers, X500F Central Control Units and X501 NET Multiplexers digital annunciator system.
	10.2	Upgrade of the generator voltage regulator with an ABB Unitrol 5000 digital voltage regulator
Turbine Generator Controls	10.2	Elimination of the following turbine generator trips High Turbogenerator Vibration Trip Low EHC Tank Level Low Lube Oil Tank Pressure Low EHC Fluid Pressure

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1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The Westinghouse Electric Corporation has been contracted to design and fabricate the NSSS components including the two reactors. In addition, they are contracted to supply the initial fuel loading for Watts Bar Unit 1 and Unit 2. TVA's Nuclear Power (NP) has the overall responsibility for the remainder of the plant, with Nuclear Engineering (NE) responsible for the design, Nuclear Construction (NC) responsible for the construction, and Nuclear Power Production (NPP) responsible for operation.

TVA utilized consultants, as necessary, to perform selected design work and to obtain specialized services. Weston Geophysical Engineering, Inc., was contracted to assist in soil foundation dynamic analyses. EDS Nuclear Inc., of San Francisco, was retained to assist in seismic analysis of piping. Chicago Bridge and Iron Company, Chicago, Illinois, was contracted to design and construct the free standing steel containments for both units. Certification of material used for containment flexible seals to withstand extreme radiation and temperature conditions was done by the Presray Corporation, Pawling, New York.

For Unit 2 construction completion, Bechtel Power Corporation provides the engineering, procurement, and construction services with TVA oversight. Bechtel uses major specialty contractors such as Siemens and Westinghouse.

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1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The design of the Watts Bar Nuclear Plant is based upon proven concepts which were developed and successfully applied to the design of pressurized water reactor systems.

Reference [1] presents descriptions of the safety related Research and Development Programs which have been carried out for, or by, or in conjunction with, Westinghouse Nuclear Energy Systems, and which are applicable to Westinghouse Pressurized Water Reactors.

The term 'research and development', as used in this report, is the same as that used by the Nuclear Regulatory Commission (NRC) in 10 CFR 50.2, that is:

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

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

1.5.1 17 x 17 Fuel Assembly

A comprehensive test Program for the 17 x 17 assembly has been successfully completed by Westinghouse. Reference [1] contains a summary discussion of the program. References [7] and [8] provide detailed descriptions and justification of design concepts used in the Watts Bar Nuclear Plant 17 x 17 fuel assemblies. The following sections present specific references documenting individual portions of the research and development program.

1.5.1.1 Rod Cluster Control Spider Tests

Rod cluster control spider tests have been completed. For a further discussion of these tests, refer to Section 4.2.3.4.

1.5.1.2 Grid Tests

Verification tests of the structural adequacy of the grid design have been completed. Refer to Section 4.2.1.3.4 and References [2] and [8] for a discussion of these tests.

1.5.1.3 Fuel Assembly Structural Tests

Fuel assembly structural tests have been completed. Refer to References [2], [3] and [8] for a discussion of these tests.

1.5.1.4 Guide Tube Tests

Verification tests of the structural adequacy of the guide tubes have been completed. Refer to references [3] and [4] for a discussion of these tests.

1.5.1.5 Prototype Assembly Tests

Verification tests of the integrated fuel assembly and rod cluster control performance have been completed. Refer to references [3], [4], and [8] for a discussion of these tests.

1.5.2 Heat Transfer Tests (17 x 17)

1.5.2.1 17 x 17 LOCA Heat Transfer Tests

Verification tests on simulated 17 x 17 assemblies to determine behavior under Loss of Coolant Accident (LOCA) have been completed. Refer to References [5] and [6] for a discussion of these tests and resultant models.

1.5.2.2 Departure from Nucleate Boiling (DNB)

The 17 x 17 fuel assembly thermal hydraulic tests have been completed and DNB correlations developed based on rod bundle data. Refer to References [7], [8], [9],and [10] for a discussion of testing and resultant DNB correlations.

REFERENCES

(1) Eggleston, F. T., 'Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries,' WCAP-8768, Latest Revision.

(2) Gesinski, L. and Chiang, D., 'Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident,' WCAP-8236 (Proprietary) and WCAP-8288 (Non-Proprietary), December 1973.

(3) DeMario, E. E., 'Hydraulic Flow Test of the 17 x 17 Fuel Assembly,' WCAP-8278 (Proprietary) and WCAP-8279 (NonProprietary), February 1974.

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

(5) 'Westinghouse ECCS Evaluation Model - October 1975 Version,' WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November 1975.

(6) Eicheldinger, C., "Westinghouse ECCS Evaluation Model, 1981 Version," WCAP-9220-P-A (Proprietary) February 1979, and WCAP-9221-A (Non-Proprietary) February 1981, Revision 1.

(7) Davidson, S. L., ed., et al., "VANTAGE 5H Fuel Assembly," WCAP-10444-P-A, Addendum 2A, April 1988.

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(9) Motley, F. E., Hill, K. W., Cadek, F. F., and Shefcheck, J., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A, July 1984.

(10) Smith, L. D. et al, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Bundles with Modified LPD Mixing Vane Grids," WCAP-15025-PA, April 1999.

Table 1.5-1 Deleted by Amendment 76

and

Table 1.5-2 Deleted by Amendment 76

Figure 1.5-1 Deleted by Amendment 76

Figure 1.5-2 Deleted by Amendment 76

1.6 MATERIAL INCORPORATED BY REFERENCE

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

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 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
0	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.
R	Used for reference only
V	Currently valid; older generation report; not formally reviewed by NRC.

Report	Review Status	Section
"Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries Fall 1974," WCAP 8485, March 1975.	B	4.2, 4.3
"Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP 8236, December 1973 (Proprietary) and WCAP 8288, December 1973 (Non Proprietary) and Addendum 1.	A	1.5
"Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," WCAP 8082 P A, January 1975 (Proprietary) and WCAP 8172-A, January 1975 (Non Proprietary).	A	3.6
"Fuel Assembly Safety Analysis For Combined Seismic and Loss of Coolant Accident," WCAP-7950, July 1972.	R	3.7
"Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation," WCAP-7332-L-AR, November 1973 (Proprietary) and WCAP-7822-AR, December 1973 (Non-Proprietary).	A	3.9

Report	Review Status	Section
"Seismic Vibration Testing with Sine Beats," WCAP-7558, October 1971.	V	3.10
"Seismic Testing of Electrical and Control Equipment," WCAP-7397-L, February 1970 (Proprietary) and WCAP-7817, December 1971 (Non-Proprietary) and Supplements 1, 2, 3, 4, 5, 6.	B	3.10
"Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-A, March 1975 (Proprietary) and WCAP-8219-A, March 1975 (Non-Proprietary).	A	4.1, 4.2, 4.3, 4.4, 15.3,
"CYGRO-2, A Fortran IV Computer Program for Stress Analysis of the Growth of Cylindrical Fuel Elements with Fission Gas Bubbles," WAPD-TM-547, November 1966.	N	4.2
"Neutron Shielding Pads," WCAP-7870, June 1972	A	4.2
"Operational Experience - Westinghouse Cores," WCAP-8183, Revision 19, January 1992.	B	4.2
"Fuel Rod Bowing," WCAP-8691 (Proprietary) and WCAP-8692, December 1975 (Non-Proprietary).	A	4.2, 4.4
"Westinghouse Anticipated Transients Without Reactor Trip Analysis," WCAP-8330, August 1974.	R	4.3, 15.2
"Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7308-L, April 1969 (Proprietary) and WCAP-7810, December 1971 (Non-Proprietary).	A	4.3
"Verification Testing of Analysis of 17 x 17A Optimized Fuel Assembly", WCAP-9401, August 1991.	A	3.7
"Morita, T., et al., "Topical Report, Power Distribution Control and Load Following Procedures," WCAP-8385, September 1974 (Proprietary) and WCAP-8403, September 1974 (Non-Proprietary).	A	4.3
"Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-7208, September 1968 (Proprietary) and WCAP-7811, December 1971 (Non-Proprietary).	O	4.3
"Power Peaking Factors," WCAP-7912-P-A, January 1975 (Proprietary) and WCAP-7912-A, January 1971 (Non-Proprietary).	A	4.3, 4.4
"Xenon-Induced Spatial Instabilities in Large PWRs," WCAP-3680-20, (EURAEC-1974), March 1968.	O	4.3
"Control Procedures for Xenon-Induced X-Y Instabilities in Large PWRs," WCAP-3680-21, (EURAEC-2111), February 1969.	O	4.3
"Xenon-Induced Spatial Instabilities in Three-Dimensions," WCAP-3680-22 (EURAEC-2116), September 1969.	O	4.3

Report	Review Status	Section
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"Application of the THINC-IV Program to PWR Design," WCAP-7359, August 1969 (Proprietary) and WCAP-7838, January 1972 (Non-Proprietary)	O	4.4
"THINC-IV - An Improved Program for Thermal- Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.	A	4.4
"Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973.	V	5.5
"Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, Revision 1, August 1975 (Proprietary) and WCAP-8312-A, Revision 2, August 1975 (Non-Proprietary).	A	6.2
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Report	Review Status	Section
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Report	Review Status	Section
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"Seismic Operability Demonstration Testing of the Foxboro H Line Series Process Instrumentation System Bistables," WCAP 8848 (Proprietary) and WCAP 8849 (Non Proprietary), November 1976.	V	3.10
"Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)," WCAP 7817, Supplement 8, June 1975.	V	3.10
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Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)," WCAP-7817, Supplement 1, December 1971.	B	3.7, 3.10

Report	Review Status	Section
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Vogeding, E. L., "Seismic Testing of Electric and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)," WCAP-7817, Supplement 3, December 1971.	B	3.10
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Malinowski, D. D., "Iodine Removal in the Ice Condenser System," WCAP-7426, April 1970.	A	15.5
WCAP-12375, Rev 1 (Proprietary Class 3) and WCAP-12374, Rev 1 (Proprietary Class 2), "Topical Report Eagle-21 Microprocessor-Based Process Protection System".	B	7.2
WCAP-17044, "Westinghouse Setpoint Methodology for Protection Systems - Watts Bar Unit 2 Only”.	B	7.2
WCAP-7671, "Topical Report Process Instrumentation for Westinghouse Nuclear Steam Supply Systems (4 Loop Plants)".	B	7.2
WCAP-8584, Rev 1 (Proprietary Class 2) and WCAP-8760, Rev 1 (Proprietary Class 3), "Failure Mode and Effects Analysis (FMEA) of the Engineered Safety Features Actuation System".	B	7.2
WCAP-11733 (Proprietary Class 2) and WCAP-11896 (Proprietary Class 3), "Noise, Fault, Surge, and Radio Frequency Interference Test Report for Westinghouse Eagle-21^{TM} Process Protection Upgrade System Upgrade System".	B	7.2

Report	Review Status	Section
WCAP-8687, Supp. 2-E69A, "Equipment Qualification Test Report, Eagle 21 Process Protection System (Environmental and Seismic Testing)" (Westinghouse Proprietary Class 2), Revision 0, May 1988.	B	7.2
WCAP-12417, (Westinghouse Proprietary Class 2), October 1989; and WCAP-12418, (Westinghouse Class 3), October 1989, "Median Signal Selector for Foxboro Series Process Instrumentation Application to Deletion of Low Feedwater Flow Reactor Trip".	B	7.2
WCAP-13632-P-A, "Elimination of Pressure Sensor Response Time Testing Requirements," (Westinghouse Proprietary Class 2C), Revision 2, January 1996.	B	7.2
WCAP-14036-P-A, "Elimination of Periodic Protection Channel Response Time Tests," (Westinghouse Proprietary Class 2C), Revision 1, December 1995.	B	7.2
WCAP-10271-P-A, Supplement 1 and Supplement 2, Rev 1 (Both are Westinghouse Proprietary Class 2) "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrument System," dated May 1986 and June 1990, respectively.	B	7.2
WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," (Westinghouse Proprietary Class 2), Revision 1, March 2003.	B	7.2
WCAP-8746-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trips," (Westinghouse Class 3), March 1977.	B	7.2
WCAP-12945-P-A "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", WCAP-16009-P-A, January 2005 (Westinghouse Proprietary).	AE	15.4
"Best-Estimate Analysis of the Large-Break Loss-of-Coolant Accident for Watts Bar Unit 2 Nuclear Power Plant Using the ASTRUM Methodology", WCAP-17093-P, December 2009	R	15.4

1.7 ELECTRICAL, INSTRUMENTATION, AND CONTROL DRAWINGS

A list of proprietary and non-proprietary electrical, instrumentation, and control (EI&C) drawings is presented in Table 1.7-1.

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 1 of 15)

System or Title	Drawing Nos.*	System or Title	Drawing Nos.*	System or Title	Drawing Nos.*
120V AC Vital Instrument Power System 1 - Single Line	45N706-1	6900V Shutdown Bd 1B-B Single Line	45W724-2	6900V Shutdown Power Schematic Diagram	2-45W760-211-2
120V AC Vital Instrument Power Sys 2 - Single Line	45N706-2	6900V Shutdown Bd 2A-A Single Line	2-45W724-3	6900V Shutdown Power Schematic Diagram	2-45W760-211-3
120V AC Vital Instrument Power Sys 3 - Single Line	45N706-3	6900V Shutdown 2B-B Single Line	2-45W724-4	6900V Shutdown Power Schematic Diagram	2-45W760-211-4
120V AC Vital Instrument Power Sys 4 - Single Line	45N706-4	6900V Diesel Generators Single Lines	45W727	6900V Shutdown Power Schematic Diagram	2-45W760-211-5
125V Vital Battery Board I - Single Line	45W703-1	6900V Diesel Generators Single Lines	45W728-1	6900V Shutdown Power Schematic Diagram	2-45W760-211-6
125V Vital Battery Board II - Single Line	45W703-2	6900V Diesel Generators Single Lines	45W728-2	6900V Shutdown Power Schematic Diagram	2-45W760-211-7
125V Vital Battery Board III - Single Line	45W703-3	6900V Unit Boards Schematic Diagrams	2-45W760-201-1	6900V Shutdown Power Schematic Diagram	45W760-211-8
125V Vital Battery Board IV - Single Line	45W703-4	6900V Unit Boards Schematic Diagrams	2-45W760-201-2	6900V Shutdown Power Schematic Diagram	45W760-211-9
Key Diagram 125V DC & 120V AC Vital Power	45N700-1	6900V Unit Boards Schematic Diagrams	2-45W760-201-3	6900V Shutdown Power Schematic Diagram	45W760-211-10
Key Diagram 48V & 250V DC & 120V AC Power	45N700-2	6900V Start & Common Boards Schematic Diagrams	45W760-200-1	6900V Shutdown Power Schematic Diagram	45W760-211-11
Key Diagram	45N700-3	6900V Start & Common Boards Schematic Diagrams	2-45W760-200-2	6900V Shutdown Power Schematic Diagram	2-45W760-211-12
Key Diagram Station Power System	15E500-1	6900V Start & Common Boards Schematic Diagrams	45W760-200-3	6900V Shutdown Power Schematic Diagram	2-45W760-211-13
Key Diagram Station Power System	15E500-2	6900V Start & Common Boards Schematic Diagrams	45W760-200-4	6900V Shutdown Power Schematic Diagram	2-45W760-211-14
Key Diagram Station Power System	15E500-3	6900V Start & Common Boards Schematic Diagrams	45W760-200-5	6900V Shutdown Power Schematic Diagram	2-45W760-211-15
6900V Shutdown Bd 1A-A Single Line	45W724-1	6900V Shutdown Power Schematic Diagram	2-45W760-211-1	6900V Shutdown Power Schematic Diagram	2-45W760-211-16

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 2 of 15)

System or Title	Drawing Nos.*	System or Title	Drawing Nos.*	System or Title	Drawing Nos.*
6900V Shutdown Power Schematic Diagram	2-45W760-211-17	480V Diesel Aux Bd 1B1-B Single Lines	2-47W732-3	480V Reactor MOV Bd 1A2-A Single Line Sh 2	2-45W751-5
6900V Shutdown Power Schematic Diagram	2-45W760-211-18	480V Diesel Aux Bd 1B2-B Single Line	2-47W732-4	480V Reactor MOV Bd 1A2-A Single Line Sh 3	2-45W751-6
6900V Shutdown Power Schematic Diagram	2-45W760-211-19	480V Shutdown Bd 1B1-B Single Line	2-45W749-3	480V Reactor MOV Bd 1B1-B Single Line Sh 1	2-45W751-7
6900V Shutdown Power Schematic Diagram	2-45W760-211-20	480V Shutdown Bd 1B2-B Single Line	2-45W749-4	480V Reactor MOV Bd 1B1-B Single Line Sh 2	2-45W751-8
6900V Shutdown Power Schematic Diagram	2-45W760-211-21	480V Shutdown Bd 1A1-A Single Line	2-45W749-1	480V Reactor MOV Bd 1B1-B Single Line Sh 3	2-45W751-9
6900V Shutdown Power Schematic Diagram	45W760-211-22	480V Shutdown Bd 2A1-A Single Line	45W749-1A	480V Reactor MOV Bd 1B2-B Single Line Sh 1	2-45W751-10
6900V Shutdown Power Schematic Diagram	2-45W760-211-23	480V Shutdown Bd 1A2-A Single Line	2-45W749-2	480V Reactor MOV Bd 1B2-B Single Line Sh 3	2-45W751-11
480V Diesel Aux Supply Bd Single Line	45W733-7	480V Shutdown Bd 2A2-A Single Line	45W749-2A	480V Reactor MOV Bd 1B2-B Single Line Sh 3	2-45W751-12
480V Diesel Aux Board 1A1-A Single Line	2-45W732-1	480V Shutdown Bd 2B1-B Single Line	45W749-3A	480V Reactor Vent Bd 1A-A Single Line Sh 1	2-45W755-1
480V Diesel Aux Board 1A2-A Single Line	2-45W732-2	480V Shutdown Bd 2B2-B Single Line	45W749-4A	480V Reactor Vent Bd 1A-A Single Line Sh 2	2-45W755-2
480V Diesel Aux Board C1-S Single Line	45W733-3	480V Reactor MOV Bd 1A1-A Single Line Sh 1	2-45W751-1	480V Reactor Vent Bd 1B-B Single Line Sh 1	2-45W755-3
480V Diesel Aux Board C1-S Single Line	45W733-4	480V Reactor MOV Bd 1A1-A Single Line Sh 2	2-45W751-2	480V Reactor Vent Bd 1B-B Single Line Sh 2	2-45W755-4
480V Diesel Aux Board C2-S Single Line	45W733-5	480V Reactor MOV Bd 1A1-A Single Line Sh 3	2-45W751-3	480V Control & Aux Bldg Vent Bd 1A1-A Single Line	2-45W756-1
480V Diesel Aux Board C2-S Single Line	45W733-6	480V Reactor MOV Bd 1A2-A Single Line Sh 1	2-45W751-4		

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 3 of 15)

System or Title	Drawing Nos.*	System or Title	Drawing Nos.*	System or Title	Drawing Nos.*
480V Control & Aux Bldg Vent Bd 1A1-A Single Line Sh 2	2-45W756-2	Instruments & Control (Layout of Control Panel)	47W600-50	Layout of Control Panels and Cabinets	47W605-50
480V Control & Aux Bldg Vent Bd 1A2-A Single and Cabinets	2-45W756-3	Instruments & Control (Layout of Control Panel)	47W600-52	Layout of Control Panels and Cabinets	47W605-51
480V Control & Aux Bldg Vent Bd 1A2-A Single Line Sh 2	2-45W756-4	Instruments & Control (Layout of Control Panel)	47W600-55	Layout of Control Panels and Cabinets	47W605-52
480V Control & Aux Bldg Vent Bd 1B1-B Single Line Sh 1	2-45W756-5	Instruments & Control (Layout of Control Panel)	47W600-56	Layout of Control Panels and Cabinets	47W605-53
480V Control & Aux Bldg Vent Bd 1B1-B Single Line Sh 2	2-45W756-6	Instruments & Control (Layout of Control Panel)	47W600-58	Layout of Control Panels and Cabinets	47W605-54
480V Control & Aux Bldg Vent Bd 1B2-B Single Line Sh 1	2-45W756-7	Instruments & Control (Layout of Control Panel)	47W600-59	Layout of Control Panels and Cabinets	47W605-55
480V Control & Aux Bldg Vent Bd 1B2-B Single Line Sh 2	2-45W756-8	Instruments & Control (Layout of Control Panel)	2-47W600-141	Layout of Control Panels and Cabinets	47W605-56
480V Shutdown Power Schematic Diagram	2-45W760-212-1	Instruments & Controls (Layout of Control Panel)	47W600-2052	Layout of Control Panels and Cabinets	47W605-57
480V Shutdown Power Schematic Diagram	2-45W760-212-2	Layout of Control Panels and Cabinets	47W605-1	Layout of Control Panels and Cabinets	47W605-58
480V Shutdown Power Schematic Diagram	2-45W760-212-3	Layout of Control Panels and Cabinets	2-47W605-2	Layout of Control Panels and Cabinets	47W605-59
480V Shutdown Power Schematic Diagram	2-45W760-212-4	Layout of Control Panels and Cabinets	47W605-28	Layout of Control Panels and Cabinets	47W605-156

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 4 of 15)

System or Title	Drawing Nos.*	System or Title	Drawing Nos.*
Layout of Control Panels and Cabinets	47W605-158	Layout of Control Panels and Cabinets	47W605-2054
Layout of Control Panels and Cabinets	47W605-160	Layout of Control Panels and Cabinets	47W605-2059
Layout of Control Panels and Cabinets	47W605-162	Layout of Control Panels and Cabinets	47W605-2172
Layout of Control Panels and Cabinets	47W605-172	Layout of Control Panels and Cabinets	47W605-2181
Layout of Control Panels and Cabinets	47W605-2001	Layout of Control Panels and Cabinets	47W605-2183
Layout of Control Panels and Cabinets	47W605-2018	Layout of Control Panels and Cabinets	47W605-2187
Layout of Control Panels and Cabinets	47W605-2032	Layout of Control Panels and Cabinets	47W605-2191
Layout of Control Panels and Cabinets	47W605-2050		
Layout of Control Panels and Cabinets	47W605-2051		
Layout of Control Panels and Cabinets	47W605-2052		
Layout of Control Panels and Cabinets	47W605-2053		
Layout of Control Panels and Cabinets	47W605-2055		
Layout of Control Panels and Cabinets	47W605-2056		
Layout of Control Panels and Cabinets	47W605-2057		
Layout of Control Panels and Cabinets	47W605-2058		

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 5 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Main Steam	2-47W610-1-1	2-47W611-1-1	2-45W600-1-1	2-45W760-1-1
	2-47W610-1-2	2-47W611-1-2	2-45W600-1-2	2-45W760-1-2
	2-47W610-1-3	2-47W611-1-3	2-45W600-1-3	2-45W760-1-3
	2-47W610-1-4		2-45W600-1-4	2-45W760-1-4
			2-45W600-1-5	
			2-45W600-1-6	
			2-45W600-1-7	
Condensate	2-47W610-2-1	2-47W611-2-1	2-45W600-2	2-45W760-2-1
& Demineralizer	2-47W610-2-2	2-47W611-2-2		2-45W760-2-2
Water	2-47W610-2-3	2-47W611-2-3		2-45W760-2-3
	2-47W610-2-4			2-45W760-2-4
				2-45W760-2-5
Main & Aux	2-47W610-3-1	2-47W611-3-1	2-45W600-3-1	2-45W760-3-1
Feedwater	2-47W610-3-2	2-47W611-3-2	2-45W600-3-2	2-45W760-3-1A
	2-47W610-3-3	2-47W611-3-3	2-45W600-3-3	2-45W760-3-2
	2-47W610-3-4	2-47W611-3-4	2-45W600-3-4	2-45W760-3-3
	2-47W610-3-5	2-47W611-3-5	2-45W600-3-5	2-45W760-3-4
	2-47W610-3-7	2-47W611-3-6	2-45W600-3-6	2-45W760-3-5
			2-45W600-3-7	2-45W760-3-6
			2-45W600-3-8	2-45W760-3-7
			2-45W600-3-9	2-45W760-3-8
			2-45W600-3-10	2-45W760-3-9
			2-45W600-3-11	2-45W760-3-10
				2-45W760-3-11
Htr Drains &	2-47W610-6-1	2-47W611-6-1	2-45W600-6-1	2-45W760-6-1
Vents	2-47W610-6-2	2-47W611-6-2	2-45W600-6-2	2-45W760-6-2
	2-47W610-6-3			2-45W760-6-3
	2-47W610-6-4			2-45W760-6-4
	2-47W610-6-5			
Auxiliary Boiler	47W610-12-1	47W611-12-1	45W600-12	45W760-12-1
		47W611-12-2		45W760-12-2

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 6 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Fire Detection		2-47W611-13-1 2-47W611-13-2 47W611-13-3 2-47W611-13-4 2-47W611-13-5 2-47W611-13-6 2-47W611-13-7		
Fuel Oil	47W610-18-1 47W610-18-2	1-47W611-18-1		45W760-18-1 45W760-18-2 45W760-18-3
High Pressure Fire Protection	47W610-26-1 47W610-26-2 47W610-26-3 47W610-26-4 47W610-26-5 47W610-26-6 2-47W610-26-7 47W610-26-8 47W610-26-9 47W610-26-10	47W611-26-1 47W611-26-2 2-47W611-26-3	45W600-26-1 45W600-26-2 45N600-26-3 45W600-26-4 45W600-26-5 45W600-26-6 45W600-26-7 45W600-26-8 45W600-26-9 45W600-26-10 45W600-26-11 45W600-26-12 45W600-26-13 45W600-26-14 45W600-26-15 45W600-26-16	45W760-26-1 45W760-26-2 45W760-26-3 45W760-26-4 45W760-26-5
Condenser Circulating Water	2-47W610-27-1 2-45W610-27-2	2-45W711-27-1		45W760-27-1 45W760-27-2 45W760-27-4 45W760-27-5 45W760-27-6

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 7 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Ventilating	2-47W610-30-1	2-47W611-30-1	2-45W600-30-1	45W760-30-1
	47W610-30-1A		2-45W600-30-2	45W760-30-2
	2-47W610-30-2	2-47W611-30-3	45W600-30-3	45W760-30-3
	2-47W610-30-3	2-47W611-30-4	45W600-30-4	45W760-30-4
	47W610-30-4	47W611-30-5	45W600-30-5	45W760-30-5
	2-47W610-30-5	2-47W611-30-6	45W600-30-6	2-45W760-30-6
	47W610-30-5A	2-47W611-30-7	2-45W600-30-7	2-45W760-30-7
	2-47W610-30-6	47W611-30-8	2-45W600-30-8	2-45W760-30-8
	47W610-30-6A	47W611-30-9	2-45W600-30-9	2-45W760-30-9
	47W610-30-7	47W611-30-10	2-45W600-30-10	2-45W760-30-10
	47W610-30-8		2-45W600-30-11	2-45W760-30-11
	47W610-30-8A		2-45W600-30-12	2-45W760-30-12
			2-45W600-30-13	2-45W760-30-13
			2-45W600-30-14	45W760-30-14
				2-45W760-30-15
				45W760-30-15A
				2-45W760-30-16
				2-45W760-30-17
				45W760-30-17A
				2-45W760-30-18
				2-45W760-30-19
				2-45W760-30-20
				45W760-30-21
				2-45W760-30-22
				2-45W760-30-23
				2-45W760-30-24
				2-45W760-30-25
				2-45W760-30-26
				45W760-30-27
				45W760-30-28
				45W760-30-29
				2-45W760-30-33

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 8 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Air Conditioning	47W610-31-1	2-47W611-31-1	2-45W600-31-1	45W760-31-1
	47W610-31-2	47W611-31-2	45W600-31-2	45W760-31-2
	47W610-31-3	47W611-31-3	45W600-31-3	45W760-31-3
	47W610-31-4	47W611-31-4	45W600-31-4	45W760-31-4
	2-47W610-31-5	47W611-31-5	45W600-31-5	45W760-31-5
	47W610-31-6	47W611-31-6	45W600-31-6	2-45W760-31-6
	47W610-31-7	2-47W611-31-7	2-45W600-31-7	2-45W760-31-7
	47W610-31-7A	47W611-31-8		45W760-31-8
	47W610-31-8	2-47W611-31-9		45W760-31-9
	47W610-31-8A			45W760-31-10
	2-47W610-31-9			45W760-31-11
				45W760-31-12
				45W760-31-13
				45W760-31-14
				2-45W760-31-15
				45W760-31-16
				2-45W760-31-17
				45W760-31-17A
				45W760-31-18
				45W760-31-19
				2-45W760-31-21
Control Air	47W610-32-1	47W611-32-1	2-45W600-32	45W760-32-1
	2-47W610-32-2	2-47W611-32-2		45W760-32-2
	47W610-32-3			45W760-32-3
Feedwater	47W610-36-1		45W600-36	45W760-36-1
Secondary Treatment	47W610-36-2			
CO_2 Storage &	47W610-39-1	47W611-39-1	45W600-39-1	
Fire Protection	47W610-39-2	47W611-39-2	45W600-39-2	
& Purging			45W600-39-3	
			45W600-39-4	

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 9 of 15)

System or Title	Drawing Nos.* Control	Logic	Schematics	Schematics
Station Drawings	47W610-40-1 47W610-40-2	2-47W611-40-1 2-47W611-40-2	45W600-40	45W760-40-1 45W760-40-2 2-45W760-40-3 2-45W760-40-4 45W760-40-5 45W760-40-6
Flood Mode Boration Makeup Sys Wtr Trtmt	2-47W610-41-1 2-47W610-41-2	2-47W611-41-1		2-45W760-41-1
Sampling & Wtr Quality	2-47W610-43-1 2-47W610-43-2 2-47W610-43-3 2-47W610-43-4 2-47W610-43-5 47W610-43-5A 2-47W610-43-6 2-47W610-43-7 2-47W610-43-8 2-47W610-43-9	2-47W611-43-1 2-47W611-43-2	2-45W600-43-1 2-45W600-43-2 2-45W600-43-3 2-45W600-43-4 2-45W600-43-5 2-45W600-43-6	45W760-43-1
Feedwater Control	2-47W610-46-1 2-47W610-46-2 2-47W610-46-3 2-47W610-46-4		2-45W600-46-1 2-45W600-46-2 2-45W600-46-3 2-45W600-46-4 2-45W600-46-5 2-45W600-46-6 45W600-46-6A 2-45W600-46-7 2-45W600-46-8	
Turbogenerator	2-47W610-47-1 47W610-47-1A 2-47W610-47-2 47W610-47-2A 2-47W610-47-3 47W610-47-3A		2-45W600-47-1 2-45W600-47-2 2-45W600-47-3 2-45W600-47-4 2-45W600-47-5 2-45W600-47-6 2-45W600-47-7 2-45W600-47-8 2-45W600-47-9 2-45W600-47-10	2-45W760-47-1

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 10 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Separation & Miscellaneous Aux Relay S/D			2-45W600-57-1	
			2-45W600-57-2	
			2-45W600-57-3	
			2-45W600-57-4	
			2-45W600-57-5	
			2-45W600-57-6	
			2-45W600-57-7	
			2-45W600-57-8	
			2-45W600-57-9	
			2-45W600-57-10	
			2-45W600-57-11	
			45W600-57-12	
			2-45W600-57-13	
			2-45W600-57-14	
			2-45W600-57-15	
			2-45W600-57-16	
			2-45W600-57-17	
			2-45W600-57-18	
			2-45W600-57-19	
			2-45W600-57-20	
			2-45W600-57-21	
			2-45W600-57-22	
			2-45W600-57-23	
			2-45W600-57-24	
			2-45W600-57-25	
			2-45W600-57-26	
			45W600-57-27	
			2-45W600-57-28	
			2-45W600-57-29	
			2-45W600-57-31	
			2-45W600-57-32	
			2-45W600-57-33	
			45W600-57-34	
			2-45W-600-57-36	
			2-45W-600-57-37	

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 11 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Ice Condenser	2-47W610-61-1	2-47W611-61-1	2-45W600-61-1	2-45W760-61-1
	2-47W610-61-2	2-47W611-61-2	1-45W600-61-22	2-45W760-61-2
	2-47W610-61-3			
Chemical &	2-47W610-62-1	2-47W611-62-1	2-45W600-62-1	
Volume Control	2-47W610-62-2	2-47W611-62-2	2-45W600-62-2	2-45W760-62-1
	2-47W610-62-3	2-47W611-62-3	45W600-62-2A	45W760-62-1A
	2-47W610-62-4	2-47W611-62-4	2-45W600-62-3	2-45W760-62-2
	2-47W610-62-5	2-47W611-62-5	2-45W600-62-4	45W760-62-2A
	-47W610-62-6	2-47W611-62-6	2-45W600-62-5	2-45W760-62-3
		2-47W611-62-7		2-45W760-62-4
				-45W760-62-5
				2-45W760-62-6
				2-45W760-62-7
				2-45W760-62-8
Safety Injection	2-47W610-63-1	2-47W611-63-1	2-45W600-63-1	2-45W760-63-1
	2-47W610-63-2	2-47W611-63-2	2-45W600-63-2	2-45W760-63-1A
	2-47W610-63-2B	2-47W611-63-3		2-45W760-63-2
	2-47W610-63-1A	2-47W611-63-4		2-45W760-63-3
		2-47W611-63-5		2-45W760-63-4
		2-47W611-63-6		2-45W760-63-5
		2-47W611-63-7		2-45W760-63-6
		2-47W611-63-8		2-45W760-63-7
				2-45W760-63-8
				2-45W760-63-9
Emergency Gas Trtmt	2-47W610-65-1	2-47W611-65-1	2-45W600-65-1	1-45W760-65-1
	2-47W610-65-1A	2-47W611-65-2	2-45W600-65-2	2-45W760-65-2
		2-47W611-65-3	2-45W600-65-3	
Essential Raw	47W610-67-1	47W611-67-1	2-45W600-67-1	1-45W760-67-1
Cooling Water	47W610-67-1A	47W611-67-2	2-45W600-67-2	1-45W760-67-2
	2-47W610-67-2	2-47W611-67-3		2-45W760-67-3
	47W610-67-2A	2-47W611-67-4		2-45W760-67-4
	2-47W610-67-3	2-47W611-67-5		2-45W760-67-5
	47W610-67-3A			2-45W760-67-6
	2-47W610-67-4			2-45W760-67-7
	2-47W610-67-5			
	47W610-67-5A			
	2-47W610-5A			
	2-47W610-67-6			

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 12 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Essential Raw Cooling Water (Continued)				2-45W760-67-8
				2-45W760-67-9
				45W760-67-9A
				2-45W760-67-10
				2-45W760-67-11
				45W760-67-12
				45W760-67-13
				2-45W760-67-14
				2-45W760-67-15
				2-45W-760-67-17
Reactor Coolant	2-47W610-68-1	2-47W611-68-1	2-45W600-68-1	2-45W760-68-1
	47W610-68-1A	2-47W611-68-2	2-45W600-68-2	45W760-68-1A
	2-47W610-68-2	2-47W611-68-3		2-45W760-68-2
	47W610-68-2A			45W760-68-2A
	2-47W610-68-3			2-45W760-68-3
	47W610-68-3A			45W760-68-3A
	2-47W610-68-4			2-45W760-68-4
	47W610-68-4A			45W760-68-4A
	2-47W610-68-5			2-45W760-68-5
	2-47W610-68-5A			45W760-68-5A
	2-47W610-68-6			2-45W760-68-6
	2-47W610-68-7			
	2-47W610-68-8			
	2-47W610-68-9			
	2-47W610-68-10			
	2-47W610-68-11			
Component Cooling	47W610-70-1	47W611-70-1	2-45W600-70	2-45W760-70-1
	47W610-70-1A	2-47W611-70-2		45W760-70-2
	2-47W610-70-2	2-47W611-70-3		2-45W760-70-3
	47W610-70-2A	2-47W611-70-4		2-45W760-70-4
	2-47W610-70-3			2-45W760-70-5
	47W610-70-3A			2-45W760-70-6
				2-45W760-70-7
				2-45W760-70-8
				2-45W760-70-9
				2-45W760-70-10

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 13 of 15)

System or Title	Drawing Nos.* Control	Logic	Schematics	Schematics
Containment Spray	2-47W610-72-1	2-47W611-72-1		2-45W760-72-1 45W760-72-1A 2-45W760-72-2 2-45W760-72-3 2-45W760-72-4
Residual Heat Removal	2-47W610-74-1	2-47W611-74-1 2-47W611-74-2	2-45W600-74	2-45W760-74-1 45W760-74-1A 2-45W760-74-2 2-45W760-74-3 2-45W760-74-4
Waste Disposal	47W610-77-1 47W610-77-2 2-47W610-77-3 2-47W610-77-4 47W610-77-5 47W610-77-6	2-47W611-77-1 47W611-77-2 47W611-77-3 47W611-77-4 2-47W611-77-5 47W611-77-6 2-47W611-77-7 1-47W611-77-8	45W600-77-1 45W600-77-2 45W600-77-3 45W600-77-4 45W600-77-5 45W600-77-6 45W600-77-7	45W760-77-1 2-45W760-77-2 2-45W760-77-3 2-45W760-77-4 45W760-77-5 2-45W760-77-6
Spent Fuel Pit Cooling	2-47W610-78-1	47W611-78-1	2-45W600-77	45W760-78-1 45W760-78-2
Primary Makeup Water	2-47W610-81-1	2-47W611-81-1	45W600-81-1 2-47W600-81	2-45W760-81-1

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 14 of 15)

System or Title	Drawing Nos.*			
	Control	Logic	Schematics	Schematics
Diesel Generator	47W610-82-1	47W611-82-1		45W760-82-1
	47W610-82-2	47W611-82-2		45W760-82-2
	47W610-82-3	47W611-82-3		45W760-82-3
	47W610-82-4	47W611-82-4		45W760-82-4
	47W610-82-5	47W611-82-1A		45W760-82-5
	47W610-82-6	47W611-82-1B		45W760-82-6
	47W610-82-7	47W611-82-1C		45W760-82-7
	47W610-82-8	47W611-82-1D		45W760-82-8
	47W610-82-9	47W611-82-2A		45W760-82-9
	47W610-82-10	47W611-82-2B		45W760-82-10
	47W610-82-11	47W611-82-2C		45W760-82-11
	47W610-82-12	47W611-82-2D		45W760-82-12
	47W610-82-13	47W611-82-3A		45W760-82-13
		47W611-82-3B		45W760-82-14
		47W611-82-3C		45W760-82-15
		47W611-82-3D		45W760-82-16
		47W611-82-3A		45W760-82-17
		47W611-82-3B		45W760-82-18
		47W611-82-3C		45W760-82-19
		47W611-82-3D		45W760-82-20
		47W611-82-4A		45W760-82-21
		47W611-82-4B		45W760-82-22
		47W611-82-4C		45W760-82-1A
		47W611-82-4D		45W760-82-1B
				45W760-82-1C
				45W760-82-2A
				45W760-82-2B
				45W760-82-2C
				45W760-82-3A
				45W760-82-3B
				45W760-82-3C

Table 1.7-1— Electrical, Instrumentation, and Control Drawings (Page 15 of 15)

System or Title	Drawing Nos.* Control	Logic	Schematics	Schematics
Diesel Geneator (Cont'd)				45W760-82-4A 45W760-82-4B 45W760-82-4C 45W760-82-5A 45W760-82-5B 45W760-82-5C 45W760-82-6A 45W760-82-6B 45W760-82-6C 45W760-82-7A 45W760-82-7B 45W760-82-7C 45W760-82-8A 45W760-82-8B 45W760-82-8C 45W760-82-9A 45W760-82-9B 45W760-82-9C 45W760-82-10A 45W760-82-10B 45W760-82-10C
Flood Mode Boration Makeup				 2-45W760-84-1
Control Rod Drive				2-45W760-85-1
Containment Isolation		2-47W611-88-1		
Radiation	2-47W610-90-1 2-47W610-90-2 2-47W610-90-3 2-47W610-90-4 2-47W610-90-5		2-45W600-90-1 2-45W600-90-2 2-45W600-90-3 45W600-90-4	
Reactor Protection		2-47W611-99-1 2-47W611-99-2 2-47W611-99-3 2-47W611-99-4 1-47W611-99-5 2-47W611-99-6 2-47W611-99-7	2-45W600-99-1	

1.8 TECHNICAL QUALIFICATION OF APPLICANT

The TVA power system is the largest in the United States. Power generating facilities operated by TVA at September 30, 2009, included 29 coventional hydroelectric sites, one pumped storage hydroelectric site, 11 coal-fired sites, three nuclear sites, 11 combustion turbine sites, two diesel generator sites, one wind energy site, one digester gas site, one biomass cofiring site, and 15 solar energy sites. TVA has three nuclear sites consisting of six units in operation. The units at Browns Ferry Nuclear Plant are boiling water reactor units and the units at the Sequoyah and Watts Bar Nuclear Plants are pressurized water reactor units. At September 30, 2009, these facilities accounted for 6,624 MW of summer net capability. TVA is primarily a wholesaler of power, operating generating plants, and transmission facilities, but no retail distribution systems. The TVA transmission system contains over 16,500 miles of lines. TVA supplies power over an area of about 90,000 square miles in parts of 7 southeastern states, containing about eight million people, and more than 2.3 million residential, farm, commercial and industrial customers.

The Tennessee Valley Authority has been engaged in the business of designing, constructing, and operating large power-producing hydro, steam, and nuclear units for over 50 years. The applicant's technical qualifications to construct and operate Watts Bar Units 1 and 2 are evidenced by the skills and experience gained over many years in the power business. This experience is supplemented by the skills and experience of TVA's consultants and its contractors in assisting in the design, construction, and operation of the Sequoyah Nuclear Plant.

TVA has acted as its own engineer/constructor and as such has pioneered in erecting large generating units. Examples are the 1,150 megawatt electric (MWe) unit placed in operation at the Paradise Steam Plant; the 1,300 MWe units in operation at the Cumberland Steam Plant; the three 1,100 MWe units at the Browns Ferry Nuclear Plant; the 1,218 MWe unit placed in operation at the Watts Bar Plant; and the two 1,170 MWe units at the Sequoyah Nuclear Plant. Over 60 individual steam generating units have been designed, constructed, and placed in operation by TVA in the past 35 years.

TVA has an experienced competent nuclear plant design organization, including engineers with many years of experience in the design and construction of large plants, including the design of the Browns Ferry, Sequoyah, Watts Bar, and Bellefonte Nuclear Plants.

Much of TVA's experience has been gained from early and continuing participation in nuclear power studies. In 1946, TVA took part in the Daniels Power Pile Study at Oak Ridge and the work of the Parker Committee, which surveyed prospects of nuclear power application. In 1953, TVA started developing a nuclear power staff and began a more detailed study of possible uses of nuclear power on its system.

In 1960, TVA agreed to operate the Experimental Gas-Cooled Reactor for the AEC at Oak Ridge and developed a technical and operating staff. Many of these trained and experienced people were assigned to TVA engineering and operating organizations

that have been directly involved in the planning, design, and construction of the Watts Bar Nuclear Plant.

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1.9 NUCLEAR PERFORMANCE PLAN

In 1985, in response to various problems identified in the Tennessee Valley Authority (TVA) nuclear program and numerous employee concerns raised with respect to the Watts Bar Nuclear Plant (WBN), TVA shut down all of its operating nuclear units and delayed its pursuit of an operating license for WBN Unit 1. Subsequently, TVA embarked on a long-term effort to comprehensively review its nuclear program. This effort, as it specifically relates to WBN resulted in the Watts Bar Nuclear Performance Plan (WBNPP) Volume 4, which was endorsed by the NRC by letter to TVA dated December 28, 1989[1]

The WBNPP describes the actions taken or planned by TVA to identify, document, investigate, and correct problems for WBN Unit 1. The WBNPP specifically provides further assurance that upon completion of these actions, WBN will be designed and constructed in accordance with applicable regulatory requirements and TVA commitments.

In a letter dated August 3, 2007 [28], TVA stated its intention to resolve the Unit 2 Corrective Action Programs (CAPs) and Special Programs (SPs) using the WBNPP [1], NUREG-0847 and applicable regulations.

1.9.1 Corrective Action Programs

Corrective Action Program plans to resolve the Unit 2 CAPs and SPs involve implementation of processes for Unit 2 design, construction and testing based upon design standards, processes or general specifications that were corrected for use of Unit 1. A different process may be used for Unit 2 than Unit 1 if design criteria, design output or licensing basis requirements have changed. Therefore, except for isolated historical cases (involving Unit 2 hardware issues), implementation of Unit 2 CAP and SP corrective actions are controlled by execution of design standards, standard processes and procedures, or general specifications. Corrective action and self-assessments processes may be used to ensure historical Unit 1 CAP and SP issues are reviewed for impact on Unit 2 to support CAP and SP completion. Completion includes preparation of a closure package documentation the completion for Unit 2.

The WBNPP provides a summary description and listing of 18 CAPS. The CAPS were submitted to NRC to obtain their concurrence with the approach described in the CAPs. A formal presentation of selected CAPs was also made to the NRC to address the NRC staff's specific questions as well as questions regarding TVA's overall approach in using CAPs as a tool for resolving nonconforming issues. The 18 CAPs are briefly described in Sections 1.9.1.1 through 1.9.1.18.

1.9.1.1 Cable Issues

This CAP provides methods for analyzing cable issues identified in Employee Concerns, conditions adverse to quality (CAQs), and NRC findings. This effort resolved prior discrepancies and ensures the adequacy of existing and future cable installations. The NRC endorsed the approach by SER, dated April 25, 1991(Unit 1) [2] and August 31, 2009 (Unit 2) [29].

1.9.1.2 Cable Tray and Cable Tray Supports

This CAP assures the structural adequacy and compliance with design criteria and licensing requirements of existing safety-related cable tray and cable tray supports required for Unit 1 operation.

This CAP assures that WBN safety-related cable tray and cable tray supports meet licensing requirements and program improvements are in place to ensure the adequacy of new or modified cable tray and cable tray supports. The CAP includes the review and revision of design criteria as necessary to ensure technical adequacy and compliance with licensing commitments. Also, design output requirements are revised or developed to comply with design criteria and to adequately translate TVA design requirements to the NRC. The NRC endorsed the approach by SER dated September 13, 1989 (Unit 1) [3] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.3 Design Baseline and Verification Program (DBVP)

This CAP is an integrated effort to ensure that the plant licensing basis is consistent with plant design and that the plant design basis is supported by adequate analysis. The DBVP ensures that an effective design change control process is implemented in order to maintain configuration control. The approach was endorsed by NRC in Inspection Report 390, 391/89-12 dated November 20, 1989 (Unit 1) [4] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.4 Electrical Conduit and Conduit Support

This CAP includes a critical case evaluation program to assure the structural adequacy of existing safety-related conduit and conduit supports that are required for Unit 1 operation.

Design output documents are revised or developed to comply with design criteria and to adequately translate design requirements. Any specific attributes not meeting these design criteria are modified as necessary. Where changes to licensing commitments are necessary, technical justification is provided and the FSAR revised accordingly. The NRC endorsed the approach by SER dated September 1, 1989 (Unit 1) [5] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.5 Electrical Issues

Implementation of this CAP ensures that the identified electrical issues are resolved in conformance with WBN licensing requirements. CAP activities provide the means to resolve the discrepancies and ensure the adequacy of existing and future electrical installations.

This effort documents resolution of electrical issues by issuing or revising calculations, procedures, design output documents, corrective actions for existing CAQs, topical reports, and test procedures. New CAQs are issued if additional deficiencies are

identified as part of this effort. Walk-down data is collected and documented in accordance with walk-down procedures. The NRC endorsed the approach by SER dated September 11, 1989 (Unit 1) [6] and August 31, 2009 (Unit 2) [29].

1.9.1.6 Equipment Seismic Qualification

Implementation of this CAP ensures that equipment seismic qualification is in conformance with WBN licensing requirements.

Field data is gathered in accordance with approved engineering walkthrough procedures. Calculations are performed and documented in accordance with TVA procedures. The justification for any equipment installation discrepancies that are determined to be "not significant to equipment qualification" and left installed as-is, is documented. The NRC endorsed the approach by SER dated September 11, 1989 (Unit 1) [7] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.7 Fire Protection

Provides assurance that WBN complies with Appendix A to BTP 9.5.1 and 10 CFR Part 50, Appendix R, Sections III.G, III.J, III.L and III.O. Deviations/exemptions are documented and justified, or corrected.

The results and conclusions of the CAP are incorporated into the Fire Protection Report, which is referenced by the FSAR. The NRC endorsed the approach by SER dated September 7, 1989 (Unit1) [8] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.8 Hanger and Analysis Update Program (HAAUP)

The program assures that the subject piping and associated pipe support installations are structurally adequate, meet the design criteria reflected in the FSAR, and comply with licensing requirements. The NRC endorsed the approach by SER dated October 6, 1989 (Unit 1) [9] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.9 Heat Code Traceability

This CAP assures that the Unit 1 piping and attachment materials of concern are in compliance with licensing requirements. Where changes to licensing commitments are necessary, technical justification is provided and the FSAR revised accordingly. Improvements have been made to ensure material traceability is maintained for future installations of ASME Code and reclassified ASTM material. Final NRC acceptance of this CAP was provided by letter dated March 29, 1991 (Unit 1) [10] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.10 Heating, Ventilation, and Air Conditioning (HVAC) Duct Supports

Implementation of this CAP demonstrates design criteria and FSAR compliance by assuring that the subject piping and associated pipe support installations are structurally adequate, meet design criteria in the FSAR, and comply with licensing requirements. Program documentation demonstrates design criteria and FSAR compliance. The NRC endorsed the approach by SER dated October 24, 1989 (Unit 1) [11] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.11 Instrument Lines

This CAP identifies the major technical issues and provides corrective actions necessary to assure that the instrument lines and associated supports are functionally and structurally adequate, and comply with WBN licensing and design basis requirements. Where changes to licensing commitments are necessary, technical justification is provided and the FSAR revised accordingly. The NRC endorsed the approach by SER dated September 8, 1989 (Unit 1) [12] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.12 Prestart Test Program

The Prestart Test Program CAP plan was withdrawn with the resubmittal of Chapter 14 of the FSAR to conform to the requirements of Regulatory Guide 1.68, Revision 2. The entire program is described in the current revision to Chapter 14. The NRC endorsed Chapter 14 in SSER-14 dated December, 1994.[13]

1.9.1.13 QA Records

This CAP resolves recognized records issues in a controlled program which: (a) provides appropriate records storage; (b) allows timely and reliable retrieval of site records commensurate with the importance of the record; (c) resolves WBN construction and operations record deficiencies; and (d) provides recurrence control for ongoing activities.

This program is documented through the implementation of procedures. Open CAQs are tracked in accordance with site procedures. The NRC endorsed the approach by SER dated April 25, 1994 (Unit 1) [14] and September 8, 2009 assessment (Unit 2) [31].

1.9.1.14 Q-LIST

The Q-list CAP provides a differentiation between features with full QA Program Requirements and those with limited QA Program requirements. The NRC endorsed the approach in SSER 13 (Appendix-AA) of NUREG-0847 dated April 1994 (Unit 1) [15] and February 11, 2009 assessment (Unit 2) [30].

1.9.1.15 Replacement Items Program (RIP-CAP)

The WBN RIP-CAP evaluates replacement items that are currently installed or will be installed to ensure that the components' ability to perform intended safety function has not been degraded.

The technical and quality requirements provided by the WBN Procurement Engineering Group is documented in output packages and dedication packages for commercial grade items. The compilation of previous maintenance activities is documented in a computer database. The engineering evaluations performed for those individual parts reviewed from inventory, or installed in the plant are documented in item evaluation QA records (including dedication documentation for QA Level II items). Other reviews are documented in QA record task summary reports. The NRC endorsed the approach by letter dated July 27, 1992 (Unit 1) [16] and September 9, 2009 assessment (Unit 2)[32].

1.9.1.16 Seismic Analysis

The purpose of this CAP was to confirm that the seismic analyses of structures and the Amplified Response Spectra generated from the analyses are technically adequate and satisfy licensing requirements. In addition, related employee concern and CAQs dealing with seismic analysis issues were resolved.

Seismic data produced as a result of this CAP is utilized by several different disciplines to calculate component-specific seismic requirements. The NRC endorsed the approach by SER dated September 7, 1989[17] and in SSER 6 of NUREG-0847 dated April 1991 (Unit 1) [18] and February 11, 2009 assessment (Unit 2)[30].

1.9.1.17 Vendor Information

This CAP provides reasonable assurance that vendor requirements for the installation, operation, maintenance, and testing of safety-related equipment are verified to be current, complete, and appropriately updated for the life of the plant. Also, the CAP confirms that correct vendor documents have been used as input to TVA design output documents, and plant instructions and procedures when appropriate. The consistency between vendor technical manuals, TVA documents, and plant configuration is confirmed as a result of direct Vendor Information CAP activities and by review/analysis of other WBN recovery and corrective action programs. The NRC endorsed the approach in SSER 11 of NUREG-0847 dated April 1993 (Unit 1) [19] and February 11, 2009 assessment (Unit 2)[30].

1.9.1.18 Welding

This CAP provided reasonable assurance that existing welds at WBN are adequate, that future welding activities will meet licensing requirements, and that a welding program is in place that can demonstrate compliance with these requirements.

This CAP resulted in three reports (Phase I, Phase II, and a Final Report) that provided TVA's bases for determining that welding of structures, systems, and components at WBN are adequate and satisfy licensing requirements. The NRC endorsed the approach in Inspection Report Nos. 50-390/89-04 and 50-391/89-04 dated August 9, 1989[20] and in Inspection Report Nos. 50-390/90-04 and 50-391/90-04 dated May 17, 1990.[21] The CAP was subsequently revised on July 31, 1990,[22] and the revisions were accepted by NRC in a letter dated March 5, 1991 (Unit 1) [23] and February 11, 2009 assessment (Unit 2)[30].

1.9.2 Special Programs (SPs)

The WBNPP provides summary descriptions and a listing of 11 SPs. Since many of the SPs were narrow in scope, and for many others substantial progress had already been made and several reports submitted to NRC, SPs were not sent to NRC for prior endorsement of approach. These programs have been reviewed and accepted by NUREG-1232, Volume 4.[1]

The SPs are described in Sections 1.9.2.1 through 1.9.2.11

1.9.2.1 Concrete Quality Program

Verified that plant/construction procedures met FSAR commitments regarding concrete compressive strength and frequency of sampling. The NRC accepted TVA's conclusions for this program in SER NUREG-1232, Volume 4 (Unit 1) [1] and February 11, 2009 assessment (Unit 2)[30].

1.9.2.2 Containment Cooling

Ensured that the WBN time-dependent environmental qualification temperature profile for the lower compartment had adequately considered the long-term effects of an MSLB inside containment for a plant going to hot standby conditions (as opposed to cold shutdown). The NRC endorsed the approach for the Containment Cooling SP by SER dated May 21, 1991 (Unit 1) [24] and February 11, 2009 assessment (Unit 2) [30].

1.9.2.3 Detailed Control room Design Review

This SP involved the performance of a detailed control room design review consistent with NUREG-0737, Supplement 1, and other commitments to the NRC regarding human factors-related control room issues. The NRC endorsed the approach for the DCRDR Special Program in SSER 6 of NUREG-0847 dated April 1991 (Unit 1) [25] and February 11, 2009 assessment (Unit 2) [30].

1.9.2.4 Environmental Qualification Program

To satisfy 10 CFR 50.49 requirements and the intent of Regulatory Guide 1.89, Revision 1, as appropriate.

Auditable documentation is compiled (EQ binders), and program controls are implemented to ensure compliance with EQ-related regulations. The NRC endorsed the approach for the Environmental Qualification Special Program by SER NUREG-1232, Volume 4 (Unit 1) [1] and February 11, 2009 assessment (Unit 2) [30] .

1.9.2.5 Master Fuse List

The purpose of the SP is (1) develop a list of both Class 1E safety-related fuses that are under TVA design control and non-Class 1E penetration protection fuses, (2) identify areas were Bussman KAZ actuators were incorrectly used and replace as necessary, and (3) correct design problems associated with EPA fuses. The NRC endorsed the approach to resolve these issues in SER NUREG-1232, Volume 4 (Unit 1) [1] and February 11, 2009 assessment (Unit 2)[30].

1.9.2.6 Mechanical Equipment Qualification

The purpose of this SP is to define WBN actions that were necessary to ensure that active safety-related mechanical equipment located in a harsh environment will perform its intended function during both normal and accident conditions. The NRC endorsed the approach for the Mechanical Equipment Qualification Special Program by SER NUREG-1232, Volume 4 (Unit1) [1] and February 11, 2009 assessment (Unit 2) [30].

1.9.2.7 Microbiologically Induced Corrosion (MIC)

The purpose of this SP is to develop a program for control of microbiologically induced corrosion in all raw water systems susceptible to this phenomenon (i.e., essential raw cooling water, condenser circulating water, raw service water, raw cooling water, and high pressure fire protection water systems). The MIC program was approved for both units in Appendix Q of SSER 8 (NUREG-0847) dated January 1992[26] and SSER 10 dated October 1992.[27]

1.9.2.8 Moderate Energy Line Break Flooding (MELB)

This SP documents TVA's evaluation of the effects of flooding in Category I structures outside containment following an MELB and the associated plant upgrades. The NRC accepted the approach for the MELB Special Program in SER NUREG-1232, Volume 4 (Unit 1) [1] and February 11, 2009 assessment (Unit 2) [30].

1.9.2.9 Radiation Monitoring System

This SP ensures that programmatic corrective actions are implemented regarding sample line, radiation monitoring system hardware, technical evaluations of RMS equipment, and correction of calibration deficiencies. NRC accepted this special program is NUREG 1232, Volume 4 (Unit1) [1] and February 11, 2009 assessment (Unit 2) [30].

1.9.2.10 Soil Liquefaction

This SP addressed concerns involving the west side of the intake pumping station regarding use of an alternative material, incomplete excavation of potentially liquefiable material, and leakage between the intake pumping station and Trench B. The NRC endorsed the approach for the Soil Liquefaction SP by SER NUREG-1232, Volume 4 (Unit 1) [1] and February 11, 2009 assessment (Unit 2) [30] .

1.9.2.11 Use-As-Is CAQs

This SP ensured that all "use-as-is" or "repair" CAQs reflected in design documents have adequate engineering justifications, meet ASME Code requirements, and the cumulative effects of all CAQs on design documents have been considered. Procedure revisions have been made as necessary. The Use-as-is SP was accepted by the NRC in SER NUREG-1232, Volume 4 (Unit 1) [1] and February 11, 2009 assessment (Unit 2) [30] .

1.9.3 REFERENCES

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(3) U.S. Nuclear Regulatory Commission, Letter from S. Black, Assistant Director for TVA Projects, TVA Projects Division, Office of Nuclear Reactor Regulation, to O. D. Kingsley, Senior Vice President, Nuclear Power (TVA). Subject: Safety Evaluation of the Watts Bar Corrective Action Program (CAP) Plan for Category I Cable Tray and Cable Tray Supports. September 13, 1989.

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