TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

70-7018

November 12, 2009

10 CFR 70.5

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Director, Office of Nuclear Material Safety and Safeguards
Washington, D C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Docket No. 50-391

Subject: **APPLICATION FOR A SPECIAL NUCLEAR MATERIAL LICENSE FOR WATTS BAR NUCLEAR PLANT UNIT 2 IN ACCORDANCE WITH 10 CFR 70, "DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL"**

The Tennessee Valley Authority (TVA) is submitting an application requesting authorization to receive, possess and store special nuclear material (SNM) in the form of 193 new fuel assemblies for the Watts Bar Nuclear Plant , Unit 2 located in Spring City, Tennessee.

In accordance with 10 CFR 70.21, "Filing," and 10 CFR 70.5, "Communications," TVA submits Enclosures 1 through 4 which contain the required license application information. The enclosed application meets the applicable requirements of 10 CFR 70.22, "Contents of Applications," for nuclear reactors being licensed under 10 CFR Part 50, "Domestic Licensing Of Production And Utilization Facilities." Additionally, the content and format of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility" was used as guidance to develop the application.

Receipt, handling, inspection and storage of new nuclear fuel for Watts Bar Nuclear Plant, Unit 2 will utilize shared systems, structures, components and administrative controls currently in place supporting the operation of Watts Bar Nuclear Plant Unit 1. As a result, the majority of the information contained, or incorporated by reference, in this application has been previously submitted and approved by the NRC. This information includes but is not limited to, the Final Safety Analysis Report, Final Environmental Impact Statement (NUREG-0498 and Supplement 1), TVA's Final Supplemental Environmental Impact Statement for the Completion and Operation of Watts Bar Nuclear Plant Unit 2, Radiological Emergency Plan, Physical Security Plan, and Safeguards Contingency Plan.

Designate as original
Rafael

NMSS01

The level of detail provided in the application is based on a comparison of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed using the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided. This approach was presented by TVA to the Nuclear Material Safety and Safeguards (NMSS) Staff during a pre-submittal Public Meeting on October 22, 2009.

The fission chambers and the startup sources for both units at Watts Bar Nuclear Plant were originally licensed under Byproduct Materials License No. 41-17572-01. This license was subsequently terminated March 5, 1996 following the issuance of the Watts Bar Nuclear Plant, Unit 1 Operating License on February 7, 1996 which encompassed the byproduct materials covered by License 41-17572-01. Fission chambers and startup sources will be acquired using the programs established for Watts Bar Nuclear Plant, Unit 1. Therefore, this application requests no other authorization for SNM or byproduct material requiring an NRC license.

Delivery of the first shipment of the Watts Bar Nuclear Plant, Unit 2 new fuel to the Watts Bar Nuclear Plant site is currently scheduled to begin during the second calendar quarter of 2011. Therefore, we request the special nuclear material license be issued by March 31, 2011.

Enclosures 1 ~~and 2~~ contain the Safety Analysis Report for this 10 CFR 70 license application. Enclosure 3 provides a summary of SNM control to be applied to the Watts Bar Nuclear Plant Unit 2 new fuel. Enclosure 4 provides a summary of the Physical Security Plan/Contingency Plan.

There are no new commitments contained in this letter. If you have any questions, please contact Gordon P. Arent at (423) 365-2004.

Respectfully,

R.M. Krich
Vice President
Nuclear Licensing

Enclosures:
1. Watts Bar Nuclear Plant Unit 2, 10 CFR 70 Safety Analysis Report.
2. ~~CD containing TVA's 2008 10-K Report to U.S. Securities and Exchange Commission, Document Components: 2,954,000 bytes~~ PER PM
3. Special Nuclear Material Control Summary
4. Physical Security Plan/Contingency Plan Summary

Chapter 1 Table of Contents

1 GENERAL INFORMATION

This section of the application contains a general description of the Watts Bar Nuclear (WBN) Plant Site and the types of activities that will be performed when receiving, possessing, inspecting, and storing special nuclear materials in the form of 193 fully assembled fuel assemblies for the initial core of the WBN Unit 2 reactor.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed using the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 1 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 1.1 Facility and Process Description				
Facility Location, Site Layout, and Surrounding Characteristics	70.22(a)(7) & 70.65(b)(1)	1.1.4.3(2)	2.1.1 Site Location and Description	1.2: General Plant Description
Facilities Description	70.22(a)(7) & 70.65(b)(2)	1.1.4.3(2)	2.1.1 Site Location and Description	1.2.2: Facility Description
Process Descriptions	70.22(a)(7) & 70.65(b)(3)	1.1.4.3(3)	9.1.1 – Criticality Safety of Fresh and Spent Fuel Storage and Handling 9.1.2 – New	9.1.1: New Fuel Storage 9.1.2: Spent Fuel Storage 9.1.4: Fuel Handling System

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 1 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
			and Spent Fuel Storage	
Raw materials, by-products, wastes, and finished products		1.1.4.3(4)	None	None
Section 1.2 Institutional Information				
Corporate identity	70.22(a)(1)	1.2.4.3(1)		See Section 1.2.1
Financial information	70.22(a) note	1.2.4.3(2)		See Section 1.2.2
Type, quantity, and form of licensed material	70.22(a)(4)	1.2.4.3(3)		See Section 1.2.3
Requested licenses and authorized uses	70.22(a)(2) & 70.22(a)(3)	1.2.4.3(4)		See Section 1.2.4
Special exemptions or special authorizations	70.17	1.2.4.3(5)		See Section 1.2.5
Security of classified information	10 CFR Part 95	1.2.4.3(6)		See Section 1.2.6
Section 1.3 Site Description				
Site Geography	70.65(b)(1)	1.3.3(1)	2.1.1 Site Location and Description	2.1: Geography And Demography
Demographics	70.65(b)(1)	1.3.3(2)	2.1.3 Population Distribution	2.1 Geography And Demography 2.2: Nearby

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 1 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
			2.2.1-2.2.2 Identification of Potential Hazards in Site Vicinity	Industrial Transportation, And Military Facilities
Meteorology	70.65(b)(1)	1.3.3(3)	2.3.1 Regional Climatology/ Local Meteorology	2.3: Meteorology
Hydrology	70.65(b)(1)	1.3.3(4)	2.4.1 Hydrologic Description	2.4: Hydrologic Engineering
Geology	70.65(b)(1)	1.3.3(5)	2.5.1 Basic Geologic an Seismic Information	2.5 Geology, Seismology, and Geotechnical Engineering Summary of Foundation Conditions

1.1 FACILITY AND PROCESS DESCRIPTION

1.1.1 Facility Location, Site Layout, and Surrounding Characteristics

The Watts Bar Nuclear Plant (WBN) Site is located on a site of approximately 1770 acres in Rhea County, Tennessee on the west bank of the Tennessee River at river mile 528. The site is just south of the Watts Bar Dam, approximately 50 miles northeast of Chattanooga, and 31 miles north-northeast of the Sequoyah Nuclear Plant site. WBN Unit 1 was licensed on February 7, 1996, Docket No. 50-390 NFP-90 and operates at 3459 MWt. WBN Unit 2 is presently under construction as authorized by Construction Permit CPPR 92, Docket NO. 50-391 issued by the Atomic Energy Commission on January 23, 1973. On July 7, 2008, the NRC issued an Order extending the construction completion date of Watts Bar Unit 2 to March 31, 2013.

1.1.2 Facilities Description

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 Final Safety Analysis (FSAR) Figures 2.1-1 through 2.1-5. Plant arrangement plans and cross sections are presented in FSAR Figures 1.2-1 through 1.2-15.

The fuel storage and handling area is located in the Auxiliary Building of WBN. All handling and storage will be within this defined area. The fuel will be inspected in the fuel-handling area and stored in the new fuel storage vault and the spent-fuel storage pit. Detailed elevation and plan views of the Auxiliary Building showing the fuel-handling areas are shown on Figures 1.2-3, 1.2-4, and 1.2-8 of the WBN FSAR, subsection 1.2.3.

New fuel is stored in racks (WBN FSAR Figure 9.1-1). Each rack is composed of individual vertical cells which can be fastened together in any number to form a module that can be firmly bolted to anchors in the floor of the new fuel storage pit. The new fuel storage racks are designed to include storage for 1/3 core for each unit at a center to center spacing of 21 inches. Space between storage positions is blocked to prevent insertion of fuel. The center-to-center distance between new fuel assemblies is sufficient to assure $k_{eff} < 0.98$ when the new fuel storage area is dry or fogged (optimally moderated). For the fully flooded condition assuming cold, clean, unborated water, the value of k_{eff} is less than or equal to 0.95. Under these conditions, a criticality accident during refueling and storage is not considered credible.

All surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel, whereas the supporting structure may be painted carbon steel.

The racks are designed to withstand nominal operating loads as well as SSE and OBE seismic loads in accordance with Regulatory Guides 1.29 and 1.13.

Floor areas of the Auxiliary Building elevation 757 are designated critical lift zones. Lifting of heavy loads in this area is controlled by site procedures.

The new fuel storage racks are located in the new fuel pit area which has a cover that protects the racks from dropped objects. The new fuel storage vault is a reinforced concrete structure. A three inch drain is provided in the new fuel storage vault. This vault is a part of the Auxiliary Building, which is a Seismic Category I Structure. The new fuel storage vault opens on to the elevation 757 floor, but is normally covered by a series of hatches which are designed to withstand the effects of an OBE or SSE. These hatches are removed as necessary during handling of the new fuel. Administrative controls are utilized when a section of the protective cover is removed for handling of the new fuel assemblies.

The spent fuel storage pool is a reinforced concrete structure with a stainless steel liner for leak tightness. This storage pool is a part of the Seismic Category I Auxiliary Building, and is shared between units one and two. Both the liner and pool walls are designed to withstand the effects of an OBE and SSE. The location of the spent fuel storage pool is shown on WBN FSAR Figures 1.2-3 and 1.2-8. The spent fuel storage pool opens onto the elevation 757 floor, and is protected by a guard rail which surrounds the pool. The depth of the pool is sufficient to allow some 26 feet of water shielding (nominally) above the spent fuel. This water depth ensures that the doses on the operating floor from stored spent fuel are negligibly small. The spent fuel storage racks consist of stainless steel structures with cells or receptacles for nuclear fuel assemblies as they are used in a reactor. Twenty-four of these flux trap racks provide 1386 storage positions in eighteen 7 x 8 cell array modules and six 7 x 9 cell array modules. Each rack is supported by four pedestals (one rack has five pedestals) sitting on two-inch thick stainless steel bearing pads which spread the load on the pool floor imposed by the dead load of the fuel assemblies, the maximum uplift force from the spent-fuel bridge hoist, thermal loads, and loads from SSE and OBE.

The spent fuel racks are designed in accordance with the following listed criteria:

(1) The spent fuel storage racks were designed for storage of 1386 fuel assemblies. The design meets all the structural and seismic requirements of Category I equipment as defined by the NRC Position Paper dated April 14, 1978, on spent fuel storage and handling applications and the references listed in Table 9.1-3.
(2) Burnup credit and fuel assembly placement controls are used to ensure the fuel array in the spent fuel racks is maintained subcritical assuming the array is fully flooded with non-borated water, the fuel is new with a maximum anticipated enrichment of 5.0 weight percent U-235, and the geometric array is the worst possible considering mechanical tolerances and abnormal conditions.
(3) The spent fuel storage facility is designed to prevent severe natural phenomena, including missiles generated from high winds, from causing damage to the spent fuel. The spent fuel storage facility, including the spent fuel racks, is Seismic Category I.
(4) The spent fuel storage racks are designed to withstand handling and normal operating loads and the maximum uplift forces generated by the fuel handling equipment.
(5) A loss of pool cooling accident is not considered a credible accident because the pool cooling system is Seismic Category I and single failure proof.
(6) The spent fuel storage racks are designed to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pit bridge hoist.
(7) The spent fuel storage facilities provide the capability for limiting the potential offsite exposures, in the event of significant release of radioactivity from the stored fuel, to well less than 10 CFR 100 guidelines.

Design of these storage racks is in accordance with Regulatory Guide 1.13 and ensures a safe condition under normal and postulated accident conditions. The distance between spent fuel assemblies is maintained to ensure a $k_{eff} < 0.95$ even if unborated water is used to fill the spent fuel storage pool.

The spent fuel racks are designed as free standing and are qualified as seismic Category I structures. The seismic design considered fully loaded racks in water at less than boiling temperature undergoing a safe shutdown earthquake (SSE). Composite, dynamic simulations which modeled all racks in the pool were utilized to determine limiting loads and displacements for each rack in the pool, to establish limiting relative motion between racks, and to evaluate the potential for and the consequences of inter-rack and rack-wall phenomena in the entire assemblage of racks. The racks were also checked for operating basis earthquake (OBE) loads and found to be satisfactory. The racks can withstand the drop of a fuel assembly from its maximum supported height and the drop of tools used in the pool. The racks are also capable of withstanding accidental drops of the gates which cover the slots between the spent fuel pool and the transfer canal and cask loading pit from a height of eight feet above the top of the racks. Electrical and mechanical stops prevent the movement of heavy objects over the spent fuel pool including the shipping casks. The movement of the casks is restricted to areas away from the pool. The wall which separates the fuel storage area from the cask loading area has been designed to restrict damage to the cask loading area if a cask were dropped even in a tipped position in the cask loading area.

1.1.3 Process Descriptions

All fuel handling will be performed with cranes and hoists located in the auxiliary building which is common to both WBN Units 1 and 2. These will include the Auxiliary Building crane, the 6-ton overhead crane in the cask loading area, the spent-fuel pit bridge hoist, and the new fuel elevator. The new fuel assemblies and their inserts are handled with handling fixtures designed specifically for this purpose and with a special sling suspended from the Auxiliary Building cranes or bridge hoist. All handling devices have provisions to avoid dropping or jamming of fuel assemblies during fuel movement. The Auxiliary Building crane, the spent-fuel pit bridge hoist, and the associated handling devices are capable of supporting maximum loads under SSE conditions. The equipment is inspected and tested for safe operation before use in fuel handling activities.

When the fuel arrives onsite, the shipping containers will be unloaded and placed on the refuel floor. The shipping containers will be opened, and the fuel will be removed one at a time and inspected in the fuel handling area. After inspection, the fuel will be placed in either the new fuel storage vault or the spent fuel storage pool.

1.1.4 Raw Materials, By-Products, Wastes, and Finished Goods

There are no reflectors or moderators with special characteristics associated with this license application. There are no raw materials associated with this license application.

1.2 INSTITUTIONAL INFORMATION

1.2.1 Corporate Identity

TVA is a wholly owned corporate agency and instrumentality of the United States of America established pursuant to the Tennessee Valley Authority Act of 1933, as amended ("TVA Act"). As an agency of the United States Government, TVA is neither owned, controlled, nor dominated by an alien, a foreign corporation, or a foreign government. A copy of TVA's latest annual report (2008 Form 10-K) filed with United States Securities and Exchange Commission which provides a current description of TVA is attached. See, in particular, pages 7 and 8 which describe TVA's service area. The address of TVA's principal executive offices is 400 W. Summit Hill Drive, Knoxville, Tennessee.

TVA is administered by a board of nine part-time members appointed by the President of the United States with the advice and consent of the Senate. The Chairman of the TVA Board is selected by the members of the TVA Board. Under the terms of the TVA Act, in order to be eligible to be appointed as a member of the Board of Directors, an individual must be a citizen of the United States. A list and description of the members of the TVA

Board of Directors appears on pages 155 and 156 of the attached 2008 Form 10-K. The only significant changes regarding the list of Board members are that Robert M. Duncan currently serves as the Chairman and Donald R. DePriest no longer serves as a member of the Board. There are currently three vacant positions on the TVA Board. A list and description of TVA's Executive Officers appears on pages 156-158 of the attached 2008 Form 10-K. The only significant change insofar as the organization and governance of TVA's nuclear program is concerned is that the current Chief Nuclear Officer and Executive Vice President is Preston D. Swafford. All of TVA's Executive Officers are citizens of the United States.

The applicant is not acting as agent or representative of another person in filing this application.

The Watts Bar Nuclear Plant (WBN) Site is located on a site of approximately 1770 acres in Rhea County, Tennessee on the west bank of the Tennessee River at river mile 528. The site is just south of the Watts Bar Dam, approximately 50 miles northeast of Chattanooga, and 31 miles north-northeast of the Sequoyah Nuclear Plant site. The site address is P O Box 2000, Spring City, Tennessee.

For Unit 2 construction completion, Bechtel Power Corporation provides the engineering, procurement, and construction services with TVA oversight. Bechtel uses major specialty subcontractors such as Siemens and Westinghouse. Westinghouse will supply the initial fuel loading for WBN Unit 2.

WBN is owned by the United States and operated by TVA.

1.2.2 Financial Qualifications

Information to demonstrate TVA's financial qualification is contained in the annual reports filed with the Securities and Exchange Commission. TVA's 2008 annual report is attached.

1.2.3 Type, Quantity and Form of Licensed Material

The maximum quantity of SNM for WBN Unit 2 including the initial core of 193 fuel assemblies and allowance for extra material onsite will be 2600 kg of U-235. A more detailed description of the fuel assemblies to be stored is set forth in section 4.2 of the WBN FSAR.

The average core enrichment is approximately 2.70 wt. percent U-235. A nominal enrichment is the design enrichment plus or minus a manufacturing tolerance. The maximum enrichment under this license will be 5 wt. percent U-235. Each fuel assembly will contain approximately 462 kilograms (kg) of uranium.

This information is summarized as follows.

Special Nuclear Material	Form	Maximum Amount
Uranium enriched in isotope U-235 up to 5% by weight and uranium daughters	Physical: Solid Chemical: UO_2	91,800 kg

1.2.4 Authorized Uses

TVA hereby applies for a SNM license to provide for receipt, possession, inspections, and storage of 193 fully assembled fuel assemblies for the initial core of the WBN Unit 2 reactor. This license is requested until June 30, 2013, or until the receipt of an operating license for WBN Unit 2.

1.2.5 Special Exemptions or Special Authorizations

None.

1.2.6 Security of Classified Information

This license application does not contain any classified National Security Information, Restricted Data, or Formerly Restricted Data (FRD) nor does it result in a change in access to such information. In addition, it is not expected that activities conducted in accordance with the proposed license will involve classified National Security Information, Restricted Data or Formerly Restricted Data (FRD).

1.3 SITE DESCRIPTION

1.3.1 Site Geography

The Watts Bar Nuclear Plant is located on a tract of approximately 1770 acres in Rhea County on the west bank of the Tennessee River at river mile 528. The site is approximately 1-1/4 miles south of the Watts Bar Dam and approximately 31 miles north-northeast of the Sequoyah Nuclear Plant. The site (about 700 feet MSL) is near the center of a northeast-southwest aligned valley, 10 to 15 miles wide, flanked to the west by Walden Ridge (900 to 1,800 feet MSL and to the east by a series of ridges reaching elevations of 800 to 1,000 feet MSL. FSAR Figure 2.1-3 consists of a map of the topographic features (as modified by the plant) of the site area for 10 miles in all directions from the plant. Profiles of maximum elevation versus distance from the center of the plant are shown in FSAR Figures 2.3-14 through 2.3-29 for the sixteen compass point sectors (keyed to true north) to a radial distance of 10 miles.

The 1770 acre reservation is owned by the United States and is in the custody of TVA. Also located within the reservation are the Watts Bar Dam and Hydro-Electric Plant, the Watts Bar Steam Plant, the TVA Central Maintenance Facility, and the Watts Bar Resort Area.

The resort area buildings and improvements have been sold to private individuals and the associated land mass leased to the Watts Bar Village Corporation, Inc. Due to this sale and leasing arrangement no services are provided to the resort area from the Watts Bar Nuclear Plant.

The location of each reactor is given below:

LONGITUDE AND LATITUDE (degrees/minutes/seconds)

UNIT 1	35°36' 10.430" N	84°47' 24.267" W
UNIT 2	35°36' 10.813" N	84°47' 21.398" W

UNIVERSAL TRANSVERSE MERCATOR (Meters)

Northing	Easting
UNIT 1 N3, 941,954.27	E 700,189.94
UNIT 2 N3, 941,967.71	E 700,261.86

FSAR Figure 2.1-2 shows the Watts Bar site location with respect to prominent geophysical and political features of the area. This map is used to correlate with the population distribution out to 50 miles. The population density within 10 miles is keyed to FSAR Figure 2.1-3.This map shows greater detail of the site area. FSAR Figures 2.1-4a and 2.1-4b are maps of the Watts Bar Site Area. The Watts Bar reservation boundary and the exclusion area boundary are boldly outlined. Details of the site and the plant structures may be found on FSAR Figure 2.1-5.

The only significant nearby industrial facility is the Watts Bar Steam Plant. The nearest land transportation route is State Route 68, about one mile north of the Site. The Tennessee River is navigable past the site. A main line of the CNO&TP (Norfolk Southern Corporation) is located approximately 7 miles west of the site. A TVA railroad spur track connects with this main line and serves the Watts Bar Steam Plant and Watts Bar Nuclear Plant. The spur has fallen into disuse and would need to be repaired prior to use. No other significant industrial land use, military facilities, or transportation routes are in the vicinity of the nuclear plant.

1.3.2 Demographics

Historical and projected population information is contained in this section. Both resident and transient populations are included. For 2000, population was based on data from the U.S. Census Bureau, Census of Population, 2000, including block group, block, and census track data. Projections were based on county projections by Woods& Poole. Sub-county population estimates were prepared using a constant share of the 1990 county total. County Census maps and 1:250,000 topographic maps were used to desegregate sub-county population data into the annular segments.

Considerations included municipal limits, topography, road system, land ownership (e.g., National Forest), and land use (e.g., strip mines). Transient population consists of two components - recreation visitation and school enrollments. Peak hour visitation to recreation facilities is based on the maximum capacity of the facility plus some overflow. School enrollments for 2008 are from the Tennessee Department of Education Report Card 2008 (http://www.state.tn.us/education/). Projected enrollments are based on projected population growth in the respective counties.

About 18,900 people lived within 10 miles of the Watts Bar site in 2000, with more than 75% of them between five and 10 miles from the site. Two small towns, Spring City and Decatur, which in 2007 had populations of 2,002 and 1,456 respectively, are located between five and 10 miles from the site. Decatur is south and south-west of the site, while Spring City is northwest and north-northwest. Most of the remainder of the area is sparsely populated, especially within five miles of the site. The pattern is expected to continue. FSAR Tables 2.1-2 through 2.1-8b show the estimated and projected population distribution within ten miles of the site for 2000, 2010, 2020, 2030, 2040, 2050, and 2060. FSAR Figure 2.1-3 shows the area within ten miles of the site overlaid by circles and sixteen compass sectors.

The area between 10 and 50 miles from the site lies mostly in the lower and middle portions of east Tennessee, with small areas in southwestern North Carolina and in northern Georgia. The population of this area is projected to increase by about 62%, or 660,000 persons, between 2000 and 2060. About 71% of this total increase is expected to be in the area between 30 and 50 miles from the site. The largest urban concentration between 10 and 50 miles is the city of Chattanooga, located to the southwest and south-southwest. This city had a population in 2007 of 169,884; about 80% of this population is located between 40 and 50 miles from the site, while the rest is located beyond 50 miles. The city of Knoxville is located to the east-northeast of the site and is slightly larger than Chattanooga. However, only a small share, less than 10 percent, of its population of 183,546, is located between 40 and 50 miles of the site with the remainder beyond 50 miles. There are three smaller urban concentrations in this area with population greater than 20,000. The city of Oak Ridge, which had a 2007 population of 27,514, is located about 40 miles to the northeast. The twin cities of Alcoa and Maryville, which had a combined population in 2007 of about 35,300, are located between 45 to 50 miles to the east-northeast. Cleveland, with a 2007 population of 39,200, is located about 30 miles to the south. Most of the population growth is expected to occur around these and the larger population centers.

There are, in addition, a number of smaller communities dispersed throughout the area, surrounded by low-density rural areas. FSAR Tables 2.1-8 through 2.1-14 contain the 2000, 2010, 2020, 2030, 2040, 2050, and 2060 population distribution at various distances and directions from the site out to 50 miles. FSAR Figure 2.1-2 shows the area within 50 miles of the site overlaid by the circles and 16 compass sectors.

Maps showing the area are found on FSAR Figures 2.1-2 and 2.1-3. The only significant nearby industrial facility is the Watts Bar Steam Plant. The nearest land transportation route is State Route 68, about one mile north of the Site. The Tennessee River is navigable past the site. A main line of the CNO&TP (Norfolk Southern Corporation) is located approximately 7 miles west of the site. A TVA railroad spur track connects with this main line and serves the Watts Bar Steam Plant and Watts Bar Nuclear Plant. The spur has fallen into disuse and would need to be repaired prior to use. No other significant industrial land use, military facilities, or transportation routes are in the vicinity of the nuclear plant.

The Watts Bar Steam Plant is a coal-fired electric generating facility with a total capacity of 240,000 kW which during normal operation has about 100 employees. The plant is not currently operating, but could be reactivated in the future.

The Watts Bar Nuclear Plant site is located on a 9-foot navigable channel on Chickamauga Reservoir. Its intake structure is located approximately two miles downstream of Watts Bar Lock and Dam. Watts Bar lock is located on the left bank of the Tennessee River with dimensions of 60' wide x 360' long. Towboat sizes vary from 1500 to 1800 horsepower for this section of the Tennessee River (Chattanooga to Knoxville). The most common type barge using the water way is the 35'x 195' jumbo barge with 1,500 ton capacity. There were also numerous liquid cargo (tank) barges of varying size with capacity to 3,000 tons.

1.3.3 Meteorology

Short-term site-specific meteorological data from the TVA meteorological facility at the Watts Bar Nuclear Plant site are the basis for dispersion meteorology analysis. Data representative of the site or indicative of site conditions for temperature, precipitation, snowfall, humidity, fog, or wind were also obtained from climatological records for Chattanooga, Dayton, Knoxville, Oak Ridge, and Watts Bar Dam, all in Tennessee. Short-term records for the Sequoyah Nuclear Plant site were used.

These data source locations are shown relative to the plant site in FSAR Figure 2.3-3.
(1) 300 mph = Rotational Speed
(2) 60 mph = Translational Speed
(3) 360 mph = Maximum Wind Speed
(4) 3 psi = Pressure Drop
(5) 1psi/sec = Rate of Pressure Drop (3 psi/3 sec is assumed)
(1) 290 mph = Rotational Speed
(2) 70 mph = Translational Speed
(3) 360 mph = Maximum Wind Speed
(4) 3 psi = Pressure Drop
(5) 2 psi/sec = Rate of Pressure Drop (3 psi/1.5 sec is assumed)

Temperature data for Dayton and for Chattanooga are presented in FSAR Tables 2.3-2 and 2.3-3, respectively. The Chattanooga and Dayton data are provided as reasonably representative and recent (1971-2000) temperature information. Mean temperatures have ranged from the low 40s in the winter to the upper 70's in the summer at both locations. Mean maxima ranged from about 50°F in mid winter to about 90°F in midsummer. The mean minima ranged from about 24°F for both locations to about 74°F for Dayton and 75°F for Chattanooga. The extreme maxima recorded for the respective data periods were 107°F at Decatur and 106°F at Chattanooga, while the extreme minima recorded were -15°F and -10°F, respectively.

Precipitation data for Watts Bar Dam are presented in FSAR Table 2.3-4. Rain or snow has fallen on an average of 110 days per year, and the annual average precipitation for 1941 through 1970 was nearly 53 inches. The maximum monthly rainfall has ranged from about seven inches to nearly 15 inches. The minimum monthly amount for September 1939 through September 1989 was zero. The maximum in 24 hours was 5.3 inches on January 6-7, 1946. Mean monthly data reveal the wettest period as late fall through early spring, with March normally the wettest month of the year. The data show a secondary peak of rainfall in July. Thunderstorm activity is most predominant in the spring and summer seasons, and the maximum frequency of thunderstorm days (FSAR Table 2.3-1) is normally in July.

Appreciable snowfall is relatively infrequent in the area. Snowfall data are summarized in FSAR Table 2.3-5 for Dayton and in FSAR Table 2.3-6 for Chattanooga and Knoxville. The Dayton, Chattanooga and Knoxville records provide current information and offer a complete picture of the pattern of snowfall in the Tennessee River Valley from Chattanooga to Knoxville. Mean annual snowfall has ranged from 4.8 inches at Chattanooga to about 10 inches at Knoxville. Dayton, about halfway between those locations, averaged about 4 inches annually for an earlier period of record. Generally, significant snowfalls are limited to November through March. For the data periods presented in the tables, respective 24-hour maximum snowfalls have been 18.5, 8, and

11.1 inches at Chattanooga, Dayton, and Knoxville. Severe ice storms of freezing rain (or glaze) are infrequent, as discussed in the regional climatology section. Atmospheric water vapor content is generally rather high in the site area, as was indicated in the discussion of the regional climatology.

Long-term relative humidity and absolute humidity data for Chattanooga are presented in FSAR Tables 2.3-7 through 2.3-9. Short-term humidity data based on measurements at the onsite meteorological facility are summarized in FSAR Tables 2.3-10 and 2.3-11 for comparison with the data in FSAR Tables 2.3-8 and 2.3-9. A typical diurnal variation is apparent in FSAR Table 2.3-7. Relative humidity and absolute humidity are normally greatest in the summer. Fog data for Chattanooga, Knoxville, and Oak Ridge, Tennessee, and from Hardwick are presented in FSAR Table 2.3-12. These data indicate that heavy fog at the Watts Bar site likely occurs on about 35 days per year with the fall normally the foggiest season. Sources of data on fogs with visibilities significantly less than 1/4 mile and on durations of fogs which can be considered representative of the site have not been identified.

Wind direction patterns are strongly influenced by the northeast-southwest orientation of the major topographic features, as evidenced in the onsite data, Sequoyah Nuclear Plant data, and the records for Knoxville and Oak Ridge. The Watts Bar wind direction and wind speed data are summarized in FSAR Tables 2.3-13 and 2.3-14 (annual at 10 and 46 meters); FSAR Tables 2.3-15 and 2.3-16 (directional persistence at 10 and 46 meters); and FSAR Tables 2.3-17 through 2.3-40 (monthly at 10 and 46 meters). The annual wind roses for each level are shown in FSAR Figures 2.3-4 and 2.3-5. The most frequent wind direction at 10 meters has been from south-southwest (about 16%). The next highest frequencies (about 8%) are from the north-northeast and northwest wind. The data in FSAR Table 2.3-41 and the data in FSAR Table 2.3-13 show a predominance of wind from the north-northwest and northwest, respectively, for wind speeds less than about 3.5 mph. More discussion of this very light wind speed pattern is contained in FSAR Section 2.3.3.3. It is very significant that the frequencies of calms differ so markedly between the two sets of onsite data. It appears that the higher frequency of calm conditions is primarily a consequence of the location of the temporary meteorological facility in a "sink." The maximum wind direction persistence period at 10 meters is shown in FSAR Table 2.3-15 as 44 hours from the south-southwest direction.

The monthly summaries show some minor variation in the wind direction patterns, but the upvalley-downvalley primary and secondary frequency maxima generally are fully evident.

In the FSAR summary tables for 46 meters, the upvalley-downvalley wind direction pattern is very clear and dominant. The two highest frequencies are 19% from the south-southwest wind direction and 11% from the north-northeast wind direction. The maximum wind direction persistence (FSAR Table 2.3-16) during the 17-year period was 48 hours from the south-southwest.

The site is located in Region I for Design Basis Tornado considerations. The design conditions assumed for the Watts Bar Nuclear Plant reactor shield building (and other safety-related structures) are the following:

(1) 300 mph = Rotational Speed
(2) 60 mph = Translational Speed

(3) 360 mph = Maximum Wind Speed
(4) 3 psi = Pressure Drop
(5) 1psi/sec = Rate of Pressure Drop (3 psi/3 sec is assumed)

For the additional Diesel Generator Building and structures initiated after July 1979, the design basis tornado parameters are as follows:
(1) 290 mph = Rotational Speed
(2) 70 mph = Translational Speed
(3) 360 mph = Maximum Wind Speed
(4) 3 psi = Pressure Drop
(5) 2 psi/sec = Rate of Pressure Drop (3 psi/1.5 sec is assumed)

These and tornado-driven missile criteria are discussed in FSAR Sections 3.3 and 3.5. The fastest mile of wind at 30 feet above ground is about 95 mph for a 100-year return period in the site area. The vertical distribution of horizontal wind speeds at 50, 100, and 150 feet above ground is 102, 113, and 120 mph on the basis of the speed at 30 feet and a power law exponent of 1/7. A gust factor of 1.3 is often used at the 30-foot level, but this would be conservative for higher levels. The wind load for the Shield Building is based on 95 mph for that level, as discussed in FSAR Section 3.3. Estimates of the probable maximum precipitation (PMP) and the design considerations for the PMP are discussed in FSAR Section 2.4.

1.3.4 Hydrology

Watts Bar Nuclear Plant is located on the right bank of the Chickamauga Lake at Tennessee River Mile (TRM) 528 with plant grade at elevation 728 MSL. The plant has been designed to have the capability for safe shutdown in floods up to the computed maximum water level, in accordance with regulatory position 2 of Regulatory Guide 1.59, Revision 2, August 1977.

Determination of the maximum flood level included consideration of postulated dam failures from seismic and hydrologic causes. The maximum flood Elevation 734.9 would result from an occurrence of the probable maximum storm. Allowances for concurrent wind waves could raise lake levels to Elevation 736.2 with run up on the 4:1 slopes approaching the plant reaching about Elevation 736.9.

The nearest surface water user located downstream from Watts Bar Nuclear Plant is Dayton, Tennessee, at TRM 503.8, 24.2 miles downstream. All surface water supplies withdrawn from the 58.9 mile reach of the mainstream of the Tennessee River between Watts Bar Dam (TRM 529.9) and Chickamauga Dam (TRM 471.0) are listed in FSAR Table 2.4-4.

The probable minimum flow past the site is estimated to be 2000 cubic feet per second (cfs), which is more than adequate for plant water requirements.

Ground water sources within a two-mile radius of the site are listed in FSAR Table 2.4-10 and their locations are shown on FSAR Figure 2.4-102. Of the 89 wells listed, only 58 are equipped with pumps. Two of the thirteen spring sources listed are equipped with pumps. Seventy-nine residences are supplied by ground water, with one well supplying five houses. Assuming three persons per residence and a per capita use rate of 75 gallons per day (gpd), total ground-water use is less than 10,000 gpd.

Drawdown data are available only for the Watts Bar Reservation wells, as listed in the previous section. Water-level fluctuations have been observed monthly in six observation wells since January 1973. Data collection for wells 7, 8, & 9 began in December 1981. The locations of these wells are shown on FSAR Figure 2.4-104. Data for the period January 1973 through December 1975 is shown on FSAR Figure 2.4-103.

As elsewhere in the region, water levels normally reach maximum elevations in February or March and are at minimum elevations in late summer and early fall. Depth to the water table is generally less than 20 feet throughout the plant site.

FSAR Figure 2.4-105 is a water-table contour map of the area within a two-mile radius of the plant site, based on 48 water-level measurements made in January 1972. The water table conforms fairly closely to surface topography, so that directions of ground-water movement are generally the same as those of surface-water movement. The water-table gradient between plant site and Chickamauga Lake at maximum water-table elevation and minimum river stage is about 44 feet in 3200 feet, or 0.014.

Water occurs in the Conasauga Shale in very small openings along fractures and bedding planes. Examination of records of 5500 feet of foundation exploration drilling showed only one cavity, 0.6-foot thick, penetrated.

Water occurs in the terrace deposit material in pore spaces between particles. The deposit is composed mostly of poorly-sorted clay- to gravel-sized particles and is poorly water bearing, although an approximately six-foot-thick permeable gravel zone is locally present at the base of the terrace deposit. The foundation excavation required only intermittent dewatering after initial drainage. The excavation was taken below the base of the terrace deposit into fresh shale. No weathered shale was found to be present; the contact between the terrace deposit and fresh shale is sharp.

The average depth to the water table in the plant area, based on data collected during August through December 1970, is 17 feet; the average overburden thickness is 40 feet; the saturated overburden thickness is therefore, 24 feet. No weathered zones or cavities were penetrated in the Conasauga Shale below a depth of 85 feet, so that the average saturated thickness of bedrock is assumed to be less than 50 feet.

The plant site is hydraulically isolated by Yellow Creek and Chickamauga Lake to the west, south, and east; it is hydraulically isolated to the north by the relatively impermeable Rome Formation underlying the site. Therefore, it is believed that any off-site groundwater withdrawals could not result in altered groundwater movement at the site.

No attempt was made to measure hydraulic properties of overburden or of bedrock at this site because of the very limited occurrence of ground water and the heterogeneity and anisotropy of the materials underlying the site.

1.3.5 Geology

The Watts Bar Nuclear Plant site is located in Rhea County, Tennessee, on the right side of the Tennessee River at river mile 528, about two miles south of Watts Bar Dam (FSAR Figure 2.5-9). Physiographically, the site is located in the Tennessee section of the Valley and Ridge Province of the Appalachian Highlands (FSAR Figure 2.5-1). This section is the southernmost of the three sections comprising the Valley and Ridge Province and extends from the Tennessee River-New River divide southwestward into central Alabama. It is bounded on the west by the Appalachian Plateaus Province and on the east by the Blue Ridge Province.

The site is located along the northeast-southwest trending portion of the Tennessee River drainage basin. At the site area the elevation of the flood plain is approximately 700 feet and to the west of the plant location is a series of knobs reaching elevation 900 feet. The plant lies on an alluvial terrace of approximate elevation 735 feet (FSAR Figures 2.5-9 and 2.5-10).

The site is located in the folded and faulted Southern Appalachian Structural Province and in the Southern Appalachian Tectonic Subdivision. A Modified Mercalli Intensity (MM) VIII earthquake is assumed to occur at the site with accelerations of 0.18 G's horizontal and 0.12 G's vertical for SSE requirements.

Regional and sub-regional fault maps are provided as FSAR Figures 2.5-7 and 2.5-8 and cover radii of 200 and 100 miles, respectively.

The plant site is situated in a bend of the Tennessee River that has been covered by alluvial terrace deposits (FSAR Figures 2.5-9 and 2.5-10). Beneath these deposits lies the Middle Cambrian Conasauga Formation, an inter-bedded shale and limestone upon which the Category I structures are founded. The regional strike of this formation is approximately N35°- 40°E (Figure 2.5-110) and beds for the most part dip to the southeast. However, because of relatively complex folding at the site, the attitudes of the bedding range from horizontal to vertical. Photographs, sections, and maps of the excavations of the plant showing rock quality and structural complexity are included in the Watts Bar FSAR as Figures 2.5-110 through 2.5-122.

Chapter 2 Table of Contents

2 ORGANIZATION AND ADMINISTRATION

This section of the application contains a description of the management systems and administrative procedures at TVA and the Watts Bar Nuclear Plant (WBN) in place to assure that corporate management is involved with, informed about, and dedicated to the safe design, and operation of the nuclear plant and that sufficient technical resources have or are being provided to adequately accomplish these objectives. WBN is an existing Site that has management systems and procedures already established to support the receipt, handling, inspection and storage of fuel. WBN Unit 1 personnel shall be responsible for receipt, handling, inspection, and storage of the new fuel for WBN Unit 2.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed using the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 2 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 2.1 Organizational Structure				
Functional description of specific organization groups responsible for operating and managing the design changes to the facility	70.22(a)(6)	2.4.3(1) & 2.4.3(7)	13.1.1 - Management and Technical Support Organization	13.1 Organization Structure of Applicant TVA-NPOD89-A Rev. 18
Section 2.2 Key Management Positions				
Qualifications, responsibilities, and	70.22(a)(6)	2.4.3 (2) 2.4.3(3) &	13.1.2, 13.1.3 - Operating	13.1.3: Qualification

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 2 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
authorities for key management personnel		2.4.3(4)	Organization	Requirements for Nuclear Facility Personnel
Section 2.3 Administration				
Effective implementation of HS&E functions using written procedures	70.22(a)(8)	2.4.3(5)	13.5.1 - Administrative Procedures	13.5: Site Instructions
Reporting of unsafe conditions or activities	70.62(a)	2.4.3(6)	17.2 – QA During the Operations Phase 17.3 – Quality Assurance Program Description	17.2: Quality Assurance for Station Operation
Commitment to establish formal management measures to ensure availability of IROFS	70.62(d)	2.4.3(8)	13.5.1 - Administrative Procedures	13.5: Site Instructions
Written agreements with offsite emergency resources	70.22(i)	2.4.3(9)	13.3 - Emergency Planning	13.3: Emergency Planning

2.1 Organizational Structure

The TVA and the WBN organizational structures are briefly described in the following sections. The summaries are based on FSAR Chapter 13.1, Organizational Structure of Applicant, the Nuclear Quality Assurance Plan, TVA-NQA-PLN89-A and the Organization Topical Report TVA-NPOD89-A, TVA Nuclear Power Group Organization Description.

Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements are documented in the Nuclear Power Organization Topical Report (TVA-NPOD89-A).

2.1.1 Corporate Functions, Responsibilities and Authorities

TVA is an agency of the federal government whose major policies, programs, and organization are determined by a part-time, nine member Board of Directors (BOD) structure pursuant to the TVA Governance Restructuring provisions of the Consolidated Appropriations Act, 2005. The BOD members are appointed by the President of the United States and confirmed by the Senate for five-year terms. The BOD selects a Chief Executive Officer (CEO) who also serves as President to manage TVA's day-to-day business. The BOD shapes the long-term business strategies, recommends major program initiatives, and guides TVA's day-to-day operations.

The Chief Operating Officer (COO) is responsible for pulling together all the operational elements of TVA with a clear focus on the operational excellence of the organization. This organization is faced with the challenges of meeting environmental pressures, growing power demand, and stakeholder expectations.

The Office of Inspector General (OIG) supports TVA in addressing its challenges and meeting its goals through the conduct of a comprehensive Audit and Inspection Programs designed to focus on areas of high risk and strategic importance. In addition, OIG responds to allegations of fraud, waste, and abuse affecting TVA. The OIG works along side, yet independent of TVA.

The CEO is responsible for managing all aspects of TVA, including power production, transmission, power trading, resource management programs, and economic development, as well as TVA's corporate functions. The CEO heads TVA's Executive Committee and chairs its Business Council.

The Office of General Counsel (OGC) provides legal services to TVA in all aspects of operations, including offering guidance and advice to the BOD on the legal ramifications of TVA activities and operations and representing them in litigation.

The Chief Administrative Officer (CAO) is responsible providing corporate support functions for all of TVA through TVA's Human Resources, Information Systems, Procurement, TVA Police, and Facilities Management.

The Executive Vice President and Chief Nuclear Officer (CNO) is responsible for the overall safety, efficiency, and economy of TVA's Nuclear Power Program and the overall Nuclear Power Group (NPG) organization.

The Senior Vice President (VP) Nuclear Generation Development and Construction is accountable for the development and construction of additional nuclear generation assets and technologies to meet demands for safe, clean, reliable and low cost power.

The Corporate Organization leadership and reporting relationships are shown in Figure 1-1 of the Organizational Topical Report TVA-NPOD89-A.

2.1.2 Operating Organization

The Vice President WBN is responsible and accountable for activities at the site, including Unit operations, modifications, maintenance, support, training, and engineering services. To accomplish these activities, the following departments report to the Vice President WBN:

- Plant Management
- Engineering
- Training
- Project Management
- Safety and Licensing
- Site Human Resources
- Site Quality Assurance
- Site Concerns Resolution

The Plant Manager reports to the Vice President WBN and is responsible for overall plant safe operation and shall have control over the onsite resources necessary for safe operation and maintenance of the plant. He directs the activities of the following departments:

- Operations
- Maintenance
- Radiological Protection
- Chemistry/Environmental
- Work Control
- Safety

The organization structure, position responsibilities and authorities are contained in the Organizational Topical Report TVA-NPOD89-A. Figure 1-12 of the Organizational Topical Report represents the Watts Bar Nuclear Plant operating organization.

2.2 Key Management Positions

This section describes the functional positions responsible for managing the operation of the WBN. The responsibilities, authorities, and lines of communication for each key management position are provided in this section. Responsible managers have the authority to delegate tasks to other individuals; however, the responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements.

2.2.1 Operating Organization

The functions and responsibilities of key personnel are described in the following paragraphs. Additional detailed responsibilities are provided in Organizational Topical Report TVA-NPOD89-A and corporate and site procedures.

2.2.1.1 Executive Vice President and Chief Nuclear Officer (CNO)

The Executive Vice President and CNO is the senior nuclear manager with direct authority and responsibility for the management, control, and supervision of TVA's Nuclear Power Group (NPG) and for the execution of nuclear programs, policies, and decisions that the Board of Directors approves or adopts. The Vice President and CNO has corporate responsibility for overall plant nuclear safety and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

2.2.1.2 Senior Vice President Nuclear Operations

This position reports directly to the Executive Vice President and CNO. Responsibilities of this position include oversight of the licensed NPG nuclear plants. The Senior Vice President Nuclear Operations direct reports are the three site Vice Presidents.

2.2.1.3 Vice President Watts Bar Nuclear Plant (WBN)

This position is responsible and accountable for activities at the site, including unit operations, modifications, maintenance, support, training, and engineering services. This includes determining the nature and extent of onsite and offsite support services required to support site operations and activities in accordance with TVA NPG's policy and procedures. This also includes quality of work activities.

2.2.1.4 Director Site Engineering

This position is responsible for integrated management and execution of site projects to provide overall management of the Engineering Design, Systems Engineering, Engineering Support, Technical Support, and Components Test and Inspection functions at the site.

2.2.1.5 Director Site Training

This position directs the planning, development, implementation, and evaluation of federally-regulated and nationally-accredited Training Programs to ensure sufficient qualified personnel to operate, maintain, and modify the nuclear power plant. The nuclear industry's training organization, the National Academy for Nuclear Training, is managed by INPO, the industry's self-governance organization.

2.2.1.6 Director Project Management

This position is responsible for planning and scheduling of major modifications and projects. This position ensures scope of work is appropriately defined and planned to minimize impact on site operations.

2.2.1.7 Director Safety and Licensing

This position is responsible for the Safety and Licensing functions at the site. This position reports to the Vice President WBN, but is provided governance and oversight direction from the Vice President Nuclear Licensing.

2.2.1.7.1 Manager Site Licensing and Industry Affairs

This position provides licensing services associated with the site operating license. This position serves as the primary interface with the NRC Region II for site-related matters. This manager is responsible for developing the vision and strategy for the site in the areas of the NRC, INPO, NEI, and other industry interfaces.

2.2.1.7.2 Manager Site Emergency Preparedness

This position is responsible for directing the technical professionals of the Site Emergency Preparedness (EP) organization which provides technical direction and support the site staffs in managing the development, maintenance, and implementation of the site-specific portions of the Radiological Emergency Plan (REP), site Emergency Plan implementing procedures, site response organization, facilities, and communications programs to meet NRC Federal regulations for maintaining an operating license and to provide protective measures to ensure the health and safety of TVA employees and the general public in the event of an accident at a NPG facility.

2.2.1.7.3 Manager Management Services

This position is accountable for planning, managing and directing all Document Control, Records Management, and Administrative Services at the site.

2.2.1.7.4 Manager Site Nuclear Security

This position is responsible for the management and direction of the Site Nuclear Security Program to ensure security at the nuclear plant sites and compliance with TVA and NRC requirements. This position reports to the Manager Security Operations (Corporate) and has a reporting relationship (dotted line) to the Director Safety and Licensing.

2.2.1.8 Manager Site Human Resources

This position serves as an advisor for Human Resource Program delivery to the nuclear site. In conjunction with line management, this position administers Human Resource policies and practices and consults with line management to develop workforce plans, staffing and recruiting plans, and succession plans. This position also provides consultation in areas such as performance management, compensation and labor relations. This position reports to the Human Resources Service Manager (Corporate) and has a reporting relationship (dotted line) to the Vice President WBN.

2.2.1.9 Manager Site Quality Assurance

This position provides oversight of quality activities associated with the operation of Watts Bar. Responsibilities are described in detail in TVA's Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A). This position reports to the General Manager, Quality Assurance (Corporate) and has a reporting relationship (dotted line) to the Site Vice President.

2.2.1.10 Specialist Site Concerns Resolution

This position is responsible for implementing and managing the Site Concerns Resolution Program to receive, evaluate, and initiate actions for resolution of employee concerns regarding NPG activities. Responsibilities also include nurturing an environment free of intimidation, harassment, or discrimination. This position reports directly to the Concerns Resolution Program Manager (Corporate), which provides the program is sufficiently independent and freed to ensure that employee concerns are properly addressed. This position has a reporting relationship (dotted line) to the Vice President WBN.

2.2.1.11 Plant Manager

The position's primary responsibility and authority is ensuring safe, reliable, and efficient plant operations in conformance and compliance with all federal, state, and local laws and regulations. The Plant Manager shall be responsible for overall plant safe operation and shall have control over the onsite resources necessary for safe operation and maintenance of the plant.

2.2.1.11.1 Manager Maintenance

This position is responsible for planning, directing, and managing the plant's Maintenance Program to ensure that equipment and systems are maintained in accordance with operability and reliability engineering practices and requirements.

2.2.1.11.2 Manager Radiological Protection

This position guides programs and activities at the plant ensuring that all operations, maintenance, modifications and engineering activities are conducted in a radiologically safe manner and protect plant systems and equipment. This includes developing, implementing, and managing the Site Radiological Program; provides technical assistance (guidance) and project management activities in support of the site consistent with regulatory requirements; develops and maintains procedures and applies standards necessary for the Radiological Protection Program; supports the Site Training Program and provides specialized training in radiological disciplines; ensures compliance with personnel radiation requirements; maintains continuing records of personnel exposure, plant radiation and contamination levels and; implementation of effective site programs for radiochemistry and radiological compliance.

2.2.1.11.3Manager Chemistry/Environmental

The position guides programs and activities at the plant ensuring that all operations, maintenance, modifications, and engineering activities that potentially impact plant chemistry/environmental are conducted in a manner consistent with applicable federal and state regulations and protect the plant systems, equipment, and the environment.

2.2.1.11.4 Manager Work Control

This position provides overall responsibility for planning, coordination, scheduling and monitoring of all on line and outage work. Responsible for establishing work priorities and coordinating shift turnover; managing the plant scheduling processes; and ensuring efficient and effective management of the work control function that is the basis of the site schedule.

2.2.1.11.5 Manager Operations

This position has responsibility for planning, organizing, and setting policy, and support activities (e.g., fire protection surveillances). These activities include operational strategies for generation, water and waste usage, approval authority for system enhancements, and prioritization of maintenance activities.

2.2.1.11.5.1 Superintendent Operations

This position is responsible for all plant operations. The superintendent, through the Shift Manager, manages the day-to-day operation of the facility, refueling operations, start-up, operational testing, water and waste processing, and plant operations. Other responsibilities include coordinating and scheduling the Training Program for all Operations personnel as well as providing the nucleus for emergency response teams.

2.2.1.11.5.2 Superintendent Operations Support

This position is responsible for budget preparation, training oversight, performance monitoring, and assists the Manager, Operations, in overall program direction for operations. The Supervisor, Fire Operations, with the overall responsibility for the Fire Protection Program, reports to the Superintendent, Operations Support.

2.2.1.12 Safety Consultant

This position delivers a tactical and consolidated safety and health program for the site. Delivers progressive programs and initiatives including safety program design and implementation, emergency response and security planning, and other programs designed to promote a skilled and safe workplace.

2.2.2 Shift Crew Composition

The shift crew for one unit operating normally consists of the Shift Manager, Unit Supervisor, Nuclear Unit Operators, and Assistant Unit Operators. Additional licensed and non-licensed personnel are required for two-unit operation. Additional operators are assigned as required by the Technical Specifications to meet the requirements of 10 CFR 50.54(m)(2). Plant management and technical support personnel will be present or on call at all times.

2.2.3 Safety Review Committee

The WBN has a Plant Onsite Review Committee (PORC) that functions to advise the Plant Manager in matters related to nuclear safety. This advisory function is performed by the PORC acting in a formal meeting periodically and as situations demand. The PORC Chairman and members are appointed in writing by the Plant Manager. PORC members meet the experience requirements of ANSI N18.1-1971 and ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2, April 1987, "Qualification and Training of Personnel for Nuclear Power Plants,".

Technical reviewers and PORC are qualified, organized, and conduct business as described in the Nuclear Quality Assurance Plan, TVA-NQA-PLN89A.

The PORC is used to conduct, as a minimum, reviews of the following. The PORC may delegate the performance of reviews, but shall maintain cognizance over and responsibility for them (e.g., subcommittees).

- New procedures or changes to existing procedures recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 that require an evaluation in accordance with 10 CFR 50.59.
- The emergency operating procedures which implement NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.
- Physical Security Plan.
- Radiological Emergency Plan.
- Offsite Dose Calculation Manual (ODCM).
- Process Control Program (radwaste packaging and shipping).
- Additional PORC reviews specifically required by site-specific technical specifications or the plant's licensing basis.
- Proposed changes to Technical Specifications; Technical Requirements Manual; their Bases; and amendments to the Operating License.
- Selected 50.59 evaluations and 72.48 evaluations.

2.2.4 Personnel Qualification Requirements

NPG personnel at the WBN meet the qualification and training requirements of NRC Regulatory Guide 1.8 (ANSI N18.1-1971 and ANSI/ANS 3.1-1981) with the alternatives as outlined in the Nuclear Quality Assurance Plan, TVA-NQA-PLN89-A.

2.3 Administration

This section provides the requirements for fuel and fuel related components (FRCs). Fuel supply, fuel design, plant operations, refueling outages, dry cask storage, and other related activities are controlled and managed to comply with applicable Technical Specification and regulatory requirements, licensing commitments, licensing and design bases, or additional commitments made due to industry practices. Further detailed discussions of management measures are contained in Chapter 11 of this application.

2.3.1 Configuration Management

Configuration management is a critical element of the engineering standard programs as well as essentially all other functional areas involved with operating, maintaining and modifying a nuclear plant. It encompasses and is implemented through various plant organizations' procedures that are established to ensure the objectives below are achieved. The detailed aspects of configuration management are integrated into many of the engineering processes and procedures to ensure that (1) plant structures, systems, components, and computer software conform to approved design requirements, and (2) the plant's physical and functional characteristics are accurately reflected in plant documents, plant simulator, and other data systems.

Configuration management philosophies are incorporated into processes for (1) operating and maintaining the plant systems and components, (2) evaluation of hardware and components to meet the plant design basis, (3) generation of design output and changes to plant configuration, (4) installation and testing of plant systems and components, and (5) revision, updating, storage, and retrieval of documents which document the configuration of the plant.

The controls established in these processes ensure that design bases are maintained, design output is consistent with the defined bases, the as-built plant configuration meet design output requirements, and the as-built documents accurately reflect the plant's configuration.

2.3.1.1 Fuel Related Aspects

Plants are operated with a strategic objective of zero fuel defects. Fuel supply, fuel design, plant operations, refueling outages, and other related activities are controlled and managed to comply with applicable Technical Specification and regulatory requirements, licensing commitments, licensing and design bases, or additional commitments made due to industry practices.

Fuel is stored only in approved locations. Approved locations are those licensed by the NRC. These are the reactor cores, the fuel storage racks and shipping containers. Requirements, restrictions, limitations, and controls for these locations are given in site Technical Specifications for cores and racks and in Certificates of Compliance for containers. To preclude the possibility of accidental criticality when fuel is outside of these locations, limited quantities of fuel are allowed out of approved storage locations.

The maximum quantity of fuel assemblies allowed out of approved storage locations per approved plant procedures are as follows:

- One un-irradiated fuel assembly shall be allowed within the fuel-handling area. The fuel handling area includes all areas of the refueling floor where un-irradiated fuel assemblies are handled outside of metal shipping containers. The fuel-handling area also includes the new fuel storage vault and the truck bay where metal shipping containers are unloaded.
- One fuel assembly shall be allowed within the spent fuel storage pool boundary (excluding the inspection, reconstitution, or cleaning locations with appropriate evaluation for each configuration that must be performed prior to implementation). The spent fuel storage pool boundary includes the cask loading area, fuel transfer canal (excluding the transfer cart), and spent fuel pool.
- Three fuel assemblies shall be allowed within the refueling canal. The refueling canal includes the fuel transfer tube boundary (including the transfer cart) and the rod cluster control changing fixture. This allows for two fuel assemblies to be in the rod cluster control changing fixture while the third fuel assembly is being transferred through the fuel transfer tube, is in the upender, or is in transit to or from the reactor cavity.
- One fuel assembly shall be allowed within the reactor cavity.
- Loose fuel rods or pellets must be evaluated for criticality before removal from a fuel assembly or storage at the site.

2.3.1.2 Maintenance

The maintenance and modification (M/M) program assures that equipment, systems, and structures (1) are maintained and modified in accordance with applicable requirements, (2) supports safe, reliable, and efficient operation of the nuclear power plants, and (3) are maintained at a quality level required for them to perform their intended functions as specified in the original design, material specifications, and inspection requirements. In the context of this program, the modification process refers only to the physical implementation of design changes.

2.3.1.3 Corrective Maintenance (CM)

Corrective maintenance is the classification of any work on systems, structures, or components (SSCs) where the SSC has failed or is significantly degraded to the point that failure is imminent (within its operating cycle/preventive maintenance interval) and no longer conforms to or is incapable of performing the SSC's design function.

2.3.1.4 Preventive Maintenance (PM)

PM consists of predictive, periodic, and planned maintenance actions taken to maintain equipment within design operating conditions and extend its life. PM is performed before equipment failure. This work is controlled by the work order (WO) process. The program requires that site PM activities be performed on critical components and be re-evaluated, revised, or updated periodically based on industry experience, plant equipment history, or trend analysis.

2.3.1.5 Long-Term Maintenance Plan (Rolling Schedule)

The long-term maintenance plan is a product of the preventive and surveillance process, and specifies the frequency for implementation of maintenance and surveillance activities necessary for the reliability of components in each system. The rolling schedule includes the preliminary defense-in-depth assessment, which documents the allowable combinations of system and Functional Equipment Groups (FEGs) that may be simultaneously worked on line or during shutdown conditions. FEGs are common sets of boundaries encompassing equipment that has been evaluated for acceptable out-of-service combinations. They are used to schedule planned maintenance and establish equipment clearances.

2.3.2 Training and Qualification

It is the policy of TVA NPG to develop and implement performance-based training programs which promote and support the safe, reliable, and efficient operation of TVA's nuclear power plants. This demands that the personnel who operate, maintain, and support those plants be fully qualified to perform their duties. Effective training is an essential element of achieving and maintaining such qualifications. Effective training thus requires definition of the skills, knowledge, and competencies necessary to perform required duties; establishment and implementation of learning opportunities which develop those desired skills, knowledge, and competencies; and, documentation of attainment of such skills, knowledge, and competencies

The training and qualification program at TVA and WBN for personnel in operating and support organizations has been developed to ensure the safe and efficient operation of nuclear power plants. The program also addresses the qualifications of personnel who occupy positions to which TVA is committed through its licensing documents. This commitment is to ensure that the minimum qualifications, which are contained in national standards, for positions in operating and support organizations are appropriate for the safe and efficient operation of TVA's nuclear power plants.

TVA will meet the requirements of Regulatory Guide 1.8, Revision 2 (4/87) for all new personnel qualifying on positions identified in regulatory position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (5/77). As specified in regulatory position C.2, all other positions will meet the requirements of ANSI/ANS N18.1-1971.

The objective of the training programs is to provide qualified personnel to operate and maintain the facility in compliance with its license, technical specifications, and appropriate governmental regulations.

Training programs are kept up-to-date by the responsible training manager to reflect plant modifications, changes in procedures, and lessons learned from in-house and industry operating experience. Training materials are updated and approved prior to use.

Continuing training is established by responsible managers to ensure that individuals performing safety-related functions remain cognizant of changes to the facility, procedures, governmental regulations, and quality assurance requirements as well as industry operating experience, Licensee Event Reports (LERs), historical INPO SOERs and other past significant facility experience, and personnel errors as applicable for their area of responsibility.

2.3.3 Procedures

The hierarchy of NPG procedures is defined in the NPG Procedure and Document Control Program. Descriptions of the major types of procedures are given below to assist in the determination of where a particular procedure fits in this hierarchy. Procedures for receipt, handling, inspection and storage of fuel are contained in the following procedure categories.

2.3.3.1 Standard Programs and Processes (SPPs)

These procedures describe administrative controls for processes that cross organizational boundaries, and based on their content, must be available, understood, and followed by all personnel. For example, clearance administration and fitness for duty procedures are SPPs since all employees need to be aware of these processes and their requirements. The following functional areas are provided from TVA's Administration of Standards Programs and Processes (TVA-SPPs). This list provides a sample of the types of procedures developed to support TVA Nuclear Power Group functions associated with nuclear plant operations:

2.3.3.2 Standard Department Procedures (SDPs)

SDPs describe administrative controls for processes that normally do not cross organizational boundaries and are generally contained within one organization. SDPs, like SPPs, are applicable to all NPG sites and locations unless reduced or limited applicability is specified in the procedure.

2.3.3.3 Site Instructions

Site Instructions are used to specify implementing instructions in the operation and maintenance of the plant. These instructions are normally technical in nature and are not administrative procedures. Examples of Site Instructions are Surveillance Instructions, Maintenance Instructions, Physical Security Instructions, Radiological Control Instructions, and Operating instructions. Site instructions are typically site-specific, but in some cases they may be common procedures used at all sites. If common procedures are used, licensing and Technical Specification requirements must be met for all sites.

2.3.4 Incident Investigation

The corrective action process is the primary tool used to document problems, analyze why problems exist, correct problems, and to document and close our gaps to excellence. As station leaders, all members of the WBN staff understand the importance of effectively using this process. Every opportunity is to be taken to ensure we and others are documenting issues. Strong and timely actions to correct and improve our equipment, performance and processes are expected from all organizations.

The WBN Unit 1 Technical Specifications (TSs) require that a system, structure, or component (SSC) be operable given the plant condition (operational mode); thus there should be a reasonable expectation that the SSC in question is operable while an operability determination is being made, or an appropriate TS action requirement should be entered.

Operations and/or Site Licensing shall take action when Problem Evaluation Reports are determined reportable and document the reportability determination.

2.3.5 Audits and Assessments

2.3.5.1 Audits

Measures are established to implement a comprehensive audit program which consists of internal audits, including NPG and other TVA organizations, which support the nuclear program and contractor/supplier audits to determine and assess the adequacy and effectiveness of the QA program.

2.3.5.2 Assessments

Quality Assurance Assessments are performed as a type of verification to ensure that observed quality-related activities are performed in accordance with requirements and desired results are achieved.

A detailed description of the program elements related to Audits and Assessments is contained in Chapter 11 of this application.

2.3.6 Quality Assurance Department

The Quality Assurance department is responsible for developing and administering the Nuclear Quality Assurance Plan (NQAP) and the Nuclear Assurance organization procedures required to ensure that TVA activities provide the required degree of safety and reliability.

Providing oversight of TVA activities by auditing, inspecting, assessing and observing the conduct of activities at Corporate and nuclear plant sites to ensure that they provide the required high degree of safety and reliability and are carried out consistent with applicable laws, regulations, regulatory commitments, licenses, and other requirements. The depth and scope of oversight is dependent on the item's or subject's importance to safety and performance history.

The TVA NQAP addresses and complies with the 18 criteria provided in 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." In addition, changes to the TVA NQAP are performed in accordance with 10 CFR 50.54, "Conditions of licenses," paragraph (a).

2.3.7 Operating Organization

At WBN, the fundamental approach to operating the plant safely requires a site operational focus. Operational Focus consists of three elements: Operational Safety, Operational Decision Making and Organizational Alignment around the roles and responsibilities of the organization in meeting the needs of Operations and minimizing operational challenges.

A licensed Unit Operator is designated as the "Operator at the Controls" or OATC whose primary focus is the monitoring of the critical parameters necessary to support safe reactor operation (typically power, level, pressure and other critical parameters as determined by Operations or Shift Management

2.3.8 Employee Concerns

The Concerns Resolution Program (CRP) is established to ensure (1) that all employees supporting NPG are free to express safety issues, concerns, or differing views to NPG management without fear of reprisal and (2) all such concerns and issues are investigated and resolved in a timely manner.

NPG places special emphasis on resolving concerns which are important to the safe and reliable operation of its nuclear plants. The normal process for resolving concerns and differing views is through the responsible line management. Employees are encouraged to use the chain of command so that corrective actions can be handled promptly at the working level. Use of the Corrective Action Program (CAP) is the preferred avenue to identify, evaluate, and resolve issues related to the safe operation of NPG plants. In addition to the CAP, the following additional avenues are available:

- NPG Concerns Resolution Staff (CRS)
- Office of the Inspector General (OIG)
- Nuclear Regulatory Commission (NRC), and
- Other governmental agencies with jurisdiction

2.3.9 Records Management

The QA program requires that for activities affecting quality, measures shall be established to ensure that documents prescribing the activity, including changes, are approved for release by authorized personnel, reviewed for adequacy, and made available to personnel performing the prescribed activity prior to commencing work.

2.3.10 Written Agreements with Offsite Emergency Resources

Interfaces between TVA, State, and local governmental agencies, and emergency response organizations are defined in the TVA REP and in the emergency plans of the affected State and local governments.

The State Radiological Emergency Plans, as well as the plans for those portions of states within the 50-mile ingestion pathway, are referenced in the TVA REP, Appendix E. These plans provide for the coordinated response of the State and affected local governments as well as the States and local governments within the 50-mile ingestion pathway.

Agreements have been established for services of outside organizations during an emergency. Agreement letters for offsite law enforcement support are maintained by the site Nuclear Security Services and are updated annually. The following provides the types of agreements established:

- Agreements maintained with ambulance services for 24-hour availability of EMT-staffed ambulances for the transport of irradiated/contaminated patients:
- Agreements maintained with medical centers to provide 24-hour availability of medical treatment for patients who may have been exposed to or contaminated with radioactive material:
- Agreements maintained with fire departments with 24-hour assistance capabilities:
- DOE Radiation Emergency Assistance Center/Training Site (REAC/TS), Oak Ridge, Tennessee - 24-hour availability of backup assistance to TVA for medical/radiological emergencies which exceed in-house and commercially available capabilities.

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3 INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY

Based on TVA's review of the regulations and discussions with the NMSS staff during a public meeting on October 22, 2009, the regulations requiring an Integrated Safety Analysis are not applicable to licensing the use of special nuclear material in a nuclear power plant. The regulations in 10 CFR 70.61 through 10 CFR 70.76 apply, in addition to other applicable Commission regulations, to each applicant or licensee that is or plans to be authorized to possess greater than a critical mass of special nuclear material, and engaged in enriched uranium processing, fabrication of uranium fuel or fuel assemblies, uranium enrichment, *enriched uranium hexafluoride conversion, plutonium processing, fabrication of mixed-oxide fuel* or fuel assemblies, scrap recovery of special nuclear material, or any other activity that the Commission determines could significantly affect public health and safety.

Section 50.34 of 10 CFR Part 50 specifies the technical information required to be contained in an application for an Operating License (OL). 10 CFR 50.34 (b) requires that an application for an OL include a final safety analysis report (FSAR) that includes information that describes the facility, presents the design bases and the limits of its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. Since the areas where fuel handling is conducted are shared between Watts Bar Nuclear Plant Unit 1 and Unit 2, the fuel handling accident analyses currently contained in Chapter 15 of the WBN Updated Final Safety Analysis Report (FSAR) are applicable to this application. Criticality analyses associated with fuel handling and storage are described in WBN FSAR Section 4.3.2.7.

The following refueling accident cases were evaluated by TVA: (1) two cases for drop of a fuel assembly with its handling tool, which impacts the baseplate (deep drop scenario) and (2) one case for drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario). An analysis of the drop of the spent fuel pool gate was performed. Fuel handling accident dose consequence analysis was performed assuming rods in a fuel assembly were damaged. In addition, criticality analyses were performed for fuel storage in the new fuel vault and the spent fuel pool. These analyses are considered to bound potential accidents associated with the receipt, inspection, handling, and storage of the new fuel for WBN Unit 2.

The FSAR states that the fuel handling system devices and equipment have provisions to avoid dropping or jamming fuel assemblies while conducting refueling operations. The combined weight of a fuel assembly plus handling tool is approximately 2100 lbs. Controls on crane movement are such that the top of an active fuel assembly can only be raised to within approximately 10 feet of the top of normal water level. Despite the handling system provisions and the controls imposed on the crane, a conservative accident evaluation of the fuel racks should include the effect of a fuel assembly falling. Drop accidents focusing on the integrity of the rack structure due to such drops are considered for the bounding rack cases. The consequences of dropping a fuel assembly as it is being moved over stored fuel are discussed below. Based on the highest lift of a fuel assembly, the maximum distance from the bottom of a fuel assembly, traveling over fuel racks, to the top of the rack is 36 inches.

Dropped Fuel Assembly - Accident I
A fuel assembly plus its handling tool (2100 lbs.) is dropped from 36 inches above the top of an empty storage location away from a rack support pedestal and impacts the base of the rack module. Local failure of the baseplate or bottom casting is acceptable; however, the rack design should ensure that gross structural failure of the rack does not occur and that the subcriticality of

the adjacent stored fuel assemblies is not violated. Calculated results show that there will be no change in spacing between cells. The load transmitted to the pool liner through the support pedestal by such an accident is well below the loads caused by the seismic event results provided in FSAR Chapter 6. Local failure of the rack bottom casting occurs during a "straight deep drop" accident away from the pedestal locations. The rack design allows local failure in that the amount of casting material present at the base of each cell is insufficient to support the postulated impact load. A finite element analysis using DYNA 2-D3 shows that the local failure of the bottom casting grid structure absorbs only 12 percent of the total impact energy. The pool liner is impacted following failure of the bottom casting. Local damage of the liner and its supporting concrete structure in the leak chase area was investigated using the LS-DYNA3D computer code to address the nonlinear elasto-plastic problem. The results show that there is no rupture of the liner.

Dropped Fuel Assembly - Accident II
Pedestal parameters were used to address the case of the "straight deep drop" accident over a pedestal. The resulting impact transmits a load of 191,000 lbs. to the slab through the pedestal. The magnitude of this impact is less than the peak pedestal load, 300,000 lbs., obtained from the seismic analysis for the racks. Furthermore, the impact load is less than the calculated peak pedestal load from the single rack analyses under OBE conditions (198,000 lbs.). In that analysis, the pedestals were shown to satisfy the allowable stress limits for Level A conditions. This accident, therefore, is not limiting. The bearing pressure on the pool slab, 2,432 psi, is below the allowable concrete pressure, 2,890 psi.

Dropped Fuel Assembly - Accident III
For the "straight shallow drop" of a fuel assembly and its handling tool on the top of the rack modules, a very conservative energy balance calculation was used together with the more conservative physical parameter values from the rack. Permanent deformation of the rack is acceptable, but such deformation is required to be limited to the top region such that the rack cross-sectional geometry at the level of the top of the active fuel region (and below) is not altered. Analysis results demonstrate that permanent damage to any fuel storage cell is limited to a maximum depth of 3.06 inches below the top of the rack. This is less than the distance from the top of the rack to the beginning of the active fuel region (approximately 20 inches). Therefore, there will be no effect on the subcriticality of fuel stored in adjacent cells as a result of this accident.

Dropped Gate
The drop of the 3820 lb. spent fuel pool gate from eight feet above the top of the racks was also evaluated. It was determined that permanent damage to a fuel storage cell is limited to a maximum depth of 5.325 inches below the top of the rack. Again, there will be no effect on the subcriticality of fuel stored in adjacent cells as a result of this accident.

The analysis results of Dropped Fuel Assembly - Accident I show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results of Dropped Fuel Assembly - Accident III and the dropped gate show that damage will be restricted such that there is no effect on the subcriticality of fuel stored in adjacent cells. The NRC staff reviewed TVA's analysis results in its submittal of October 23, 1996 and concurred with the findings. This is acceptable based on the TVA's structural integrity conclusions supported by the parametric studies.

Consequences of a Fuel Handling Accident (FHA)
The analysis of a postulated fuel handling accident is based on Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors ," and NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors."

The parameters used for this analysis are listed in FSAR Table 15.5-20.

The bases for the Regulatory Guide 1.25 evaluations are:

1. In the Regulatory Guide 1.25 analysis the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

2. In the Regulatory Guide 1.25 analysis damage was assumed for all rods in one assembly.

3. The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in FSAR Table 15.5-21. In the Regulatory Guide 1.25 analysis, a radial peaking factor of 1.65 is used.

4. For the Regulatory Guide 1.25 analysis all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 10% of the total noble gases and radioactive iodine inventory in the rods at the time of the accident with the following gap percentage exceptions which are based on NUREG/CR 5009 as appropriate: 14% of the Kr-85, 5% of the Xe-133, 2% of the Xe-135, and 12% of the I-131.

5. Noble gases released to the spent fuel pool are released through the Shield Building vent to the environment.

6. In the Regulatory Guide 1.25 analysis, the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

7. In the Regulatory Guide 1.25 analysis, the spent fuel pool decontamination factors for the inorganic and organic iodine are 133 and 1, respectively.

8. All iodine escaping from the pool is exhausted to the environment through charcoal filters.

9. A filter efficiency of 99% is used for elemental and organic iodine for the Auxiliary Building Gas Treatment System (ABGTS) filters and 90% for inorganic iodine and 30% for organic iodine for the purge air exhaust filters.

10. No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

11. The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in FSAR Table 15A-2 are used. The thyroid dose utilizes ICRP-30, "Limits for Intakes of Radionuclides by Workers," iodine dose conversion factors. Doses are based on the dose models presented in FSAR Appendix 15A.

12. Two Tritium-Producing Burnable Absorber Rods in the assembly are assumed to break and release the entire contents of tritium. All of the tritium is conservatively assumed to evaporate into the air.

The thyroid, gamma, and beta doses for FHAs in the Auxiliary and Reactor Buildings are given in FSAR Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10 CFR 100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body. These doses are calculated by using Revision 4 of the computer code FENCDOSE.

The ventilation function of the reactor building purge ventilating system (RBPVS) is not a safety-related function. However, the filtration units and associated exhaust ductwork do provide a safety-related filtration path following a fuel-handling accident prior to automatic closure of the associated isolation valves. The RBPVS contains air cleanup units with prefilters, HEPA filters, and 2-inch-thick charcoal adsorbers. This system is similar to the auxiliary building gas treatment system except that the latter is equipped with 4-inch-thick charcoal adsorbers. Anytime fuel handling operations are being carried on inside the primary containment, either the containment is isolated or the reactor building purge filtration system is operational. The assumptions listed above are, therefore, applicable to a fuel handling accident inside primary containment except that the assigned filter efficiency is 90% for inorganic iodine and 30% for organic iodine since no relative humidity control is provided.

The radiation dose results of the Regulatory Guide 1.25 fuel handling accident (FHA) analysis are provided in FSAR Table 15.5-23. For a FHA inside containment, no allowance has been made for possible holdup or mixing in the primary containment or isolation of the primary containment as a result of a high radiation signal from monitors in the ventilation system for the case where containment penetrations are closed to the Auxiliary Building. However, the containment purge filters are credited. For a FHA inside containment when containment penetrations and/or the annulus are open to the Auxiliary Building Secondary Containment Enclosure (ABSCE) spaces, the containment is isolated by a high radiation signal from monitors in the ventilation system and no credit is assumed for the containment purge filters. The result of a FHA inside primary containment is well below the limits of 10 CFR 100.

The whole body, beta and thyroid doses to control room personnel from the radiation sources discussed above are presented in FSAR Table 15.5-23. The doses are calculated by the COROD computer code. The gamma and beta doses are based on a one time burn of a Tritium Production Core fuel element whereas the thyroid dose is based on a three times burned element. This selection of sources produces higher doses. Parameters for the control room analysis are found in FSAR Table 15.5-14. The dose to whole body is below the 10 CFR 50, Appendix A, GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844, ""Calculation of Distance Factors for Power and Test Reactors," were used to determine the dose. Dose conversion factors in ICRP-30 were used to determine thyroid doses in place of those found in TID-14844.

Criticality Accidents

See Chapter 5 of this license application for a discussion of the criticality analyses performed for the new fuel storage vault and spent fuel storage pool.

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4 RADIATION PROTECTION

This section of the application describes the radiation protection program at the Watts Bar Nuclear (WBN) Plant.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed based on the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 4 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 4.1 Commitment to Radiation Protection Program Implementation	10 CFR 20.1101, Subpart B	4.4.1.3	12.5 Operational Radiation Protection Program	12.5 Radiological Control (RADCON) Program
Section 4.2 Commitment to an ALARA Program	10 CFR 20.1101	4.4.2.3	12.1 Assuring that Occupational Radiation Exposure Are ALARA	12.1 Assuring that Occupational Radiation Exposure Are ALARA
Section 4.3 Organization and Personnel Qualifications	10 CFR 70.22	4.4.3.3	13.1.2, 13.1.3 Operating Organization	13.1.3 Qualification Requirements for Nuclear Facility Personnel
Section 4.4 Commitment to Written Procedures	10 CFR 70.22(8)	4.4.4.3	13.5.1 Administrative Procedures	13.5 Site Instructions

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 4 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 4.5 Training Commitments	10 CFR 19.12 & 10 CFR 20.2110	4.4.5.3	13.2.2 Non-Licensed Plant Staff Training	13.2.1 Accredited Training Programs
Section 4.6 Ventilation and Respiratory Protection Programs Commitments	10 CFR 20, Subpart H	4.4.6.3	12.3 & 12.4 Radiation Protection Design Features, 12.5 Operational Radiation Protection Program	12.3.3 Ventilation, 12.5.2 Equipment, Instrumentation, and Facilities
Section 4.7 Radiation Surveys and Monitoring Programs Commitments	10 CFR 20, Subpart F, C, L, M	4.4.7.3	12.3 & 12.4 Radiation Protection Design Features, 12.5 Operational Radiation Protection Program	12.3.3 Ventilation, 12.5.2 Equipment, Instrumentation, and Facilities
Section 4.8 Additional Program Commitments	10 CFR 20, Subpart L, M, 10 CFR 50.72 & 10 CFR 50.73	4.4.8.3	N/A	See Section 4.8

4.1 COMMITMENT TO RADIATION PROTECTION PROGRAM IMPLEMENTATION

The TVA Nuclear Power Group (NPG) Radiation Protection (RP) program, which is applicable to WBN, implements the requirements of 10 CFR 19, 20, and 30 through 34. The RP Program is further established to meet, to the extent practicable, the guidelines contained in INPO 05-008, and ANI Inspection Criteria 8.1 through 8.10.

The RP Program consists of four elements that are directed toward essential support to TVA's nuclear power program.

- Radiological impact assessments.
- Radiation protection planning and radiological safety evaluation, including preliminary safety analysis reports, final safety analysis reports, and radiological emergency plans.
- Radiological environmental monitoring.
- Radiological control activities

The RADCON Section is under the supervision of the Plant Manager.

The RADCON Section is responsible for the radiological control activities at the plant. It applies radiation standards and procedures; reviews proposed methods of plant operation; participates in development of plant documents; and assists in the plant training program, providing specialized training in radiation protection. It provides coverage for all operations involving radiation or radioactive materials including maintenance, fuel handling, waste disposal, and decontamination. It is responsible for personnel and inplant radiation monitoring, and maintains continuing records of personnel exposures, plant radiation, and contamination levels.

4.2 COMMITMENT TO AN ALARA PROGRAM

Consistent with TVA's overall commitment to keep occupational radiation exposures as low as reasonably achievable, specific plans and procedures are followed by operating and maintenance staff to assure that ALARA goals are achieved in the operation of the plant. Operational ALARA policy and procedures are formulated at the corporate level in Nuclear Power and are implemented at each nuclear plant through the issuance of division procedures and plant instructions for the purpose of maintaining Total Effective Dose Equivalent (TEDE) ALARA. These procedures and instructions are consistent with the intent of Section C.1 of Regulatory Guide 8.8 and Regulatory Guide 8.10. Included in these operating procedures and plant instructions are the provision that employee radiation exposure trends are reviewed periodically by management staff at the plant and in the central office. Summary reports are prepared that describe: (a) major problem areas where high radiation exposures are encountered; (b) which worker group is accumulating the highest exposures; and (c) recommendations for changes in operating, maintenance, and inspection procedures or modifications to the plant as appropriate to reduce exposures.

An ALARA committee composed primarily of supervisory personnel is established to review periodically the effectiveness of implementation of the ALARA Program. Reviews include the site performance against ALARA goals, employee ALARA suggestions, ALARA planning documents, and trends. The Plant Manager or Assistant Plant Manager will normally serve as chairman of the site ALARA committee.

4.3 ORGANIZATION AND PERSONNEL QUALIFICATIONS

The TVA and Watts Bar Nuclear Plant specific organizations are discussed in Chapter 2 of this application.

As described previously in section 2.2.4 of this application, the site Radiation Protection Manager shall have the education and experience as described in Regulatory Guide 1.8, Revisions 1 and 2 in the context of Regulatory Guide 1.8 and the endorsed ANSI N18.1-1971 and ANSI/ANS-3.1-1981. Because of TVA's commitment to both documents, the Radiation Protection Manager must meet the more restrictive of the composite qualifications and training of both documents.

The Radiation Protection Manager shall have a bachelor's degree in a science or engineering subject, including formal training in radiation protection. At the time of initial core loading or appointment to the active position, whichever is later, the responsible individual shall have five years of experience in applied radiation protection. At least three of the five years shall be professional-level experience in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power plants, preferably in a nuclear power plant. During the three years, the individual shall participate in the radiation protection section of an operating nuclear power plant during the following periods: (1) routine refueling outage (one to two months); and (2) two months operation above 20 percent power. The Radiation Protection Manager shall have at least six months experience onsite.

4.4 COMMITMENT TO WRITTEN PROCEDURES

Radiation control instructions are maintained and made available to all site personnel. These instructions are written to implement the requirements of 10 CFR 20, applicable codes and standards, and commitments to outside agencies (American Nuclear Insurers, Institute of Nuclear Plant Operations, etc.). Chapter 11 of this application provides a detailed discussion of procedure controls and implementation.

Radiation protection procedures are prepared, reviewed and approved to carry out activities related to the Radiation Protection Program. Procedures are used to control radiation protection activities in order to ensure that the activities are carried out in a safe and effective manner. Radiation protection procedures are reviewed and revised as needed to incorporate facility or operational changes.

4.4.1 Radiation Work Permit Procedures

A Radiation Work Permit (RWP) system shall be established to document radiological conditions and prescribe appropriate protective requirements for work in radiologically controlled areas.

- Site Radiation Protection shall be responsible for establishing entry requirements for radiological areas via the RWP.

- The area in which the work is to be performed is surveyed for radiological hazards before the start of work and/or as appropriate during work to ensure that radiological hazards are properly identified.

- Protective clothing and equipment, dosimetry, and work limitation requirements are specified for all workers entering the area.

RWPs will normally be required for all work in radiologically controlled areas. RWPs shall always be required for areas where radiological conditions meet or exceed the criteria listed below.

- Entering a "Radiation Area."
- Entering a "High Radiation Area" or "Very High Radiation Area."
- Entering a "Contaminated Area."
- Entering an "Airborne Radioactivity Area."
- Breaching a contaminated system or component.
- RP discretion to provide adequate radiological control.
- For radiographic examinations conducted at licensed nuclear facilities.
- Entering an area or component where radiological conditions are unknown.

Each worker shall be responsible for awareness and compliance with the radiation protection requirements of an RWP and for meeting the prerequisites for RWP entry.

4.5 TRAINING COMMITMENTS

A radiation protection training program shall be developed, documented, and administered consistent with expectations as outlined in NEI 95-04, "Guideline for General Access Training". This program is implemented in General Employee Training for NPG power plant facilities. All individuals who in the course of employment are likely to receive an occupational exposure to radiation from licensed and unlicensed radiation sources under the control of the licensee in excess of 100 mrem in a year shall receive radiation protection training commensurate with their duties and responsibilities (10 CFR 19.12) and instructions on U.S. NRC Regulatory Guides 8.13 and 8.29.

A training program for RP personnel shall be developed by Nuclear Training. Nuclear Training shall issue procedures detailing the program. The Program Manager of Radiological Services will concur with the initial issuance and any change to procedures for the training of RP personnel. The National Voluntary Laboratory Accreditation Program (NVLAP), Technical Director will concur with the training requirements and procedures involving NVLAP accredited activities.

4.6 VENTILATION AND RESPIRATORY PROTECTION PROGRAMS

Internal occupational dose is controlled though facility design, engineering controls, confinement and reduction of contaminated areas, limiting access to radiological controlled areas, and the use of respiratory protective equipment. Personnel are not routinely monitored for internal deposited radioactive material. Confirmatory monitoring (by licensee) is performed for individuals through the assessment and tracking of DAC-h. Radio-bioassay (in vitro and in vivo measurement and analysis) is employed to confirm and/or evaluate probable intake.

4.6.1 Respiratory Protection Program

A respiratory protection program shall be established and maintained in accordance with 10 CFR 20. Workers shall have respiratory protection training before wearing respiratory protection equipment.

TVA is responsible for providing a workplace environment in which individuals are adequately protected from hazards, including hazards from exposure to ionizing radiation. As part of TVA's program to maintain exposures ALARA, the TEDE is to be ALARA for activities subject to the 10 CFR 20 "Standards for Protection Against Radiation." These requirements allow intakes of radioactive material by workers, if such intakes result in lower external dose and maintain TEDE.ALARA. Under these requirements intakes of radioactive material are permissible if evaluations predict that use of respiratory protection equipment will result in a higher TEDE. Additionally, other factors may be considered in the evaluation for maintaining TEDE ALARA. These factors may include, but are not limited to, environmental conditions, safety conditions, accessibility conditions, worker comfort, wear times, and the type of respiratory equipment specified or available. All TEDE ALARA evaluations shall be documented and retained as a Facility based Radiological Control Program record. Dose calculations/investigations are reviewed and approved by radiation protection supervision.

Unplanned intakes (no documented TEDE ALARA evaluation) of radioactive material by workers that result in an internal dose of 10 mrem or greater shall be documented in the Corrective Action Program.

Respiratory Protection Program elements include:
- Air sampling sufficient to identify the potential hazard, permit proper equipment selection, and estimate exposures;
- Surveys and bioassays, as appropriate, to evaluate actual intakes;
- Testing of respirators for operability immediately prior to each use;
- Written procedures shall be established that address: selection, fitting, issuance, maintenance, and testing of respirators, including testing for operability immediately prior to each use; program audits; minimum qualifications of program supervisors and implementing personnel; limitations on periods of respirator use and relief from respirator use; maintaining TEDE ALARA and performing evaluations; supervision and training of personnel; monitoring (including air sampling and bioassays), and recordkeeping; a description of the applications of respirators for routine, non-routine, and emergency respirator use; and periodic medical evaluation (NRC Regulatory Guide 8.15).
- Determination by a physician prior to the initial fitting of respirators, and annually (quarter ending) thereafter or periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection equipment.

4.6.2 Ventilation Systems

The fuel handling area ventilation system, a subsystem of the Auxiliary Building ventilating system, serves the fuel-handling area at Elevation 757, the penetration rooms at Elevation 737, Elevation 757 and Elevation 782, and the fuel, waste, and cask handling areas at Elevation 729 and Elevation 692.

The system is designed to: (1) maintain acceptable environmental conditions for personnel access, operation, inspection, maintenance, and testing, (2) protect mechanical and electrical equipment and controls, and (3) control airborne activity during normal operation. The environmental control system is designed to maintain building temperatures between 60°F minimum and 104°F maximum.

During accident conditions, the fuel handling area ventilation system is shut down and all environmental control is handled by the Auxiliary Building Gas Treatment System (ABGTS), described in FSAR Section 6.2.3.

All ductwork, dampers, and grilles of the fuel handling area ventilation system essential to operation of the ABGTS are designed to Seismic Category I and Safety Class 2b requirements. Each fueling handling area exhaust fan is provided with a primary circuit breaker and a shunt trip isolation switch which is tripped by a signal of the opposite train from that for the primary circuit breaker to ensure that power is isolated from the fan. All other system components, including exhaust fans and remaining ductwork and dampers, are designed to Seismic Category I(L) requirements.

To control airborne activity, ventilation air is supplied to clean areas, then routed to areas of progressively greater contamination potential. The fuel handling area is maintained at a slightly negative pressure to limit out leakage, and can be physically isolated from the outdoors in case of radiological contamination.

Air utilized to ventilate the fuel handling area, waste packaging, and cask shipping areas is exhausted through the fuel handling area exhaust fans. An exhaust duct system from the waste packaging area and cask loading area is connected to a duct system around the periphery of the spent fuel pit and fuel transfer canal. Thus, exhaust air from the fuel handling area passes across the spent fuel pit forming an air curtain across the pool. During periods of irradiated fuel movement in the fuel transfer canal, air curtain exhaust flow at the fuel transfer canal area is required to be interrupted. The fuel transfer canal exhaust flow is isolated to prevent the uptake of source terms emitted during a postulated fuel handling accident in the fuel transfer canal and to support proper spent fuel pool accident radiation monitor operation.

Exhaust is provided by two 100% capacity fuel handling area exhaust fans. During normal operation one fan is in operation with the other on standby. Both fans discharge to the Auxiliary Building exhaust stack.

An inlet damper furnished with each fuel handling area exhaust fan is used to regulate the volume of air exhausted as required to maintain a ¼-inch negative pressure within the building. These dampers are automatically operated by static pressure controllers.

During periods of high radiation in the fuel handling area or upon initiation of a containment isolation signal, or for high air temperature at the supply intake the Auxiliary Building supply and exhaust fans and the fuel handling exhaust fans are automatically stopped and isolation dampers located in the ducts that penetrate the Auxiliary Building Secondary Containment Enclosure (ABSCE) are closed. Additionally, during refueling operations when containment and/or the annulus is open to the ABSCE spaces, a Containment Vent Isolation (CVI) signal will automatically stop the above described fans and close the same isolation dampers as described above. Similarly, the high radiation signal in the fuel handling area can also automatically initiate a CVI during refueling operations when containment and/or the annulus is open to the Auxiliary Building ABSCE spaces. An isolation barrier is thus formed between the building and the outdoor environment, and the ABGTS is started up automatically (see FSAR Section 6.2.3) to maintain the ABSCE at less than a 1/4-inch water gauge negative pressure during these high radiation or accident periods.

The fuel-handling area ventilation system is located completely within Seismic Category I structures and all safety-related components are fully protected from floods and tornado-missile damage.

4.7 RADIATION SURVEYS AND MONITORING PROGRAM COMMITMENTS

Prospective monitoring determinations for internal and external dose monitoring are performed for individuals or group of individuals entering the restricted area. Personnel monitoring, for dose from sources external to the body, is conducted using appropriate dosimeters as required by 10 CFR 20. TVA maintains accreditation as a processing laboratory for dosimeters, as described in American Standards Institute (ANSI) N13.11-1983, "Personnel Dosimeter - Criteria for Performance". This accreditation is under the National Voluntary Laboratory Accreditation Program conducted by the National Institute of Standards and Technology. Dosimeters may be processed onsite by WBN, an accredited sub-facility, or by another processing laboratory within the scope of TVA's accreditation. Dose information for whole body (total effective dose equivalent), external exposure of the skin, lens of the eye, and extremities is recorded in a dose tracking system and retained in a permanent historical database for generating required reports. Real time control is generally implemented using information from direct reading dosimeters. Official doses of record are taken from dosimeters. However, doses are calculated when dosimeter results are not available or do not accurately represent actual dose received.

Personnel monitoring and confirmatory monitoring for dose from intakes of radioactive material is conducted using DAC-HR tracking and bioassays, including whole body counting. Monitoring is performed for each person required to be monitored by 10 CFR 20. The whole body counter is calibrated with standard radioisotopes in configurations that approximate the human body. It is able to detect expected gamma emitting radionuclides per ANSI-N13.30, September 1989, Table-1, "Acceptable Minimum Detectable Activities."

Routine radiological surveys to detect radiation, radioactive contamination, and airborne radioactivity are performed throughout the plant on periodic schedules. Survey frequencies are determined by the RADCON Superintendent based upon the actual or potential radiological conditions. Schedules for completion of routine surveys are issued to the technicians. As plant conditions change, the schedule will be updated. Radiological surveys may be performed whenever personnel enter potential or actual radiological areas and there is any doubt as to the existing conditions. Retention of survey records follows the requirements of 10 CFR 20.2103 and Regulatory Guide 1.88

Radiation and contamination surveys will be made on the new fuel shipments by Radiological Control personnel. The purpose of the survey is to protect personnel from unnecessary exposure to radiation and/or contamination. Smears shall be counted for alpha and beta-gamma radiation.

The designated fuel receiving areas will be zoned according to 10 CFR 20. When the fuel arrives onsite, radiation and contamination surveys will be taken on the transport vehicle. Dose rate at contact and 2 meters from the vehicle will be taken, Contact dose rates, dose rates at 1 meter, and smears will be taken on the external surfaces of the shipping containers. After the shipping containers are opened, smears will be taken of the fuel assembly covering and the inside of the container. The dose rate of each fuel assembly will be obtained, and the fuel assembly will be smeared when the polyethylene covering has been removed for inspection. When all fuel containers are removed from the truck, radiation and smear surveys will be taken on the truck before allowing it to leave.

Periodic surveys will be performed within the storage/handling area. Upon detection of contamination, a personnel monitoring station will be established and the area controlled to prevent the spread of the contamination. The work controlling document will describe the protective clothing, dosimetry, and methods to be followed to prevent unnecessary exposure to personnel. The contaminated area or item will be cleaned and/or disposed of appropriately.

Portable survey instruments are calibrated and checked periodically with standard radioactive sources in accordance with instrument specific calibration and maintenance procedures. Accurate records on the performance of each instrument during each calibration are maintained. Each laboratory counting system is checked at regular intervals with standard radioactive sources for proper counting efficiencies, background count rates, and operating parameters.

4.7.1 Radiological Zones

Radiological zones at WBN have been established to (1) control the spread of contamination, (2) control personnel access to avoid unnecessary exposure of personnel to radiation, and (3) to control access to radioactive sources present in the facility. The following definitions of areas are provided to describe how the facility Radiation Protection Program is implemented to protect workers and the general public on the site.

- **Owner Controlled Area** - An area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

- **Restricted Area** - Any area access to which is limited by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials (10 CFR 20.1003).

- **Radiologically Controlled Area (RCA)** - An area within (or that may coincide with) the Restricted Area (defined in 10 CFR 20.1003) boundaries that may have increasing radiological hazards

- **Radiation Area** - An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

- **High Radiation Area (HRA)** - An area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mrem in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

- **Very High Radiation Area** - An area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads in 1 hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates.

4.7.2 Access and Egress Control

Controls have been established for entry into and exit from radiological controlled areas (RCA). Prior to entry, workers are provided training, radiation monitoring devices (thermoluminescent dosimeter TLD and electronic dosimetry) and are required to have a radiation work permit (RWP) applicable to the assigned work activity. Upon exiting a RCA, workers are expected to proceed to the nearest frisker station and perform a self-survey of their hands and feet at a minimum. Once frisking is completed, workers will exit the RCA via a personal contamination monitor (PCM). Prior to leaving the WBN protected area workers exit through a portal monitor that again measures the individual for contamination.

Access controls to prevent unplanned exposures in high radiation areas are implemented in accordance with WBN Unit 1 Technical Specifications. In addition to the access control requirements for high radiation areas, the following control measures are implemented to control access to very high radiation areas in which radiation levels could be encountered at 500 rads or more in 1 hour at 1 meter from a radiation source or any surface through which the radiation penetrates:

- Conspicuously posted with a sign(s) stating GRAVE DANGER - VERY HIGH RADIATION AREA

- Area is locked. Each lock shall have a unique core. The keys shall be administratively controlled by the RADCON Superintendent.

- Plant manager's (or designee) approval required for entry.

- RADCON personnel shall be in accompaniment of the person(s) making the entry.

4.7.3 Posting for Radiation Protection Awareness

Each RCA shall be posted by yellow and magenta signs bearing the standard radiation warning symbol and the words "Caution - Radiologically Controlled Area." The posting shall also state that a monitoring device is required (unless it has been determined that monitoring is not required).

Contamination areas shall have conspicuous boundaries consisting of such items as rad-ribbon, rad-rope, rad-tape, and step-off pads and be posted by yellow and magenta signs bearing the standard radiation warning symbol and the words "Caution-Contaminated Area" or "Caution-Contamination Area." Where, due to physical space limitations, it is impractical to post a contaminated area as described above, the area may be noted with radiation tape and/or radiation hazard tags. Physical space limitation is intended to apply to such areas as floor drains, electrical panels, sample sinks, etc.

Radiological postings shall be displayed with yellow and magenta colors in accordance with 10 CFR 20.1901.

4.7.4 Protective Clothing and Equipment

TVA provides protective clothing for use in radiological areas. Clothing required for a particular instance is prescribed by RADCON based upon the actual or potential radiological conditions. Protective clothing is cleaned, surveyed for contamination, checked for physical condition, and returned to service if acceptable. Additional protective clothing stock is available from the plant warehouse as required. Protective clothing available for use includes but is not limited to:

1. Coveralls
2. Lab coats
3. Gloves
4. Head covers
5. Foot covers

4.7.5 Personnel Monitoring for External Exposures

All individuals who are expected to work in a radiologically controlled area (RCA) shall process through RP when arriving, transferring, or terminating at a Nuclear Power Group (NPG) site. In addition, monitored and NPG staff individuals who will visit another licensee or TVA plant, and require a thermoluminescent dosimeter (TLD) on that visit, must check out prior to leaving their respective sites unless exempted by RP. If an employee is assigned to work at a non-TVA installation where an exposure to radiation is incurred, the employee shall inform RP of this assignment. The employee shall turn in their dosimetry, obtain any required bioassays, and complete any requested documentation. When the employee returns, they must report to RP to obtain any required bioassay and update their exposure records.

TVA will provide each worker entering an RCA with dosimetry capable of measuring the worker's dose. This is accomplished by using a dosimeter of record (for example, a TLD), appropriate for the radiological environment, provided by a National Voluntary Laboratory Accreditations Program (NVLAP) certified processor (utility or vendor).

Administrative dose levels (ADLs) to be used as guidelines for maintaining doses below regulatory limits have been established within the NPG and shall be observed for routine work. This program is not applicable to minors or declared pregnant women. Obtain appropriate station supervision and radiation protection management approval to increase a worker's administrative dose level. Examples of a bona fide need for a dose extension are that 1) the unique ability or experience of the individual will minimize collective dose; and 2) other qualified individuals with lower doses are not available.

The RPM shall prepare a report for the TVA Chief Nuclear Officer and Executive

Vice President for submittal within 30 days to INPO's Radiological Protection and Emergency Preparedness Division and the NRC (10 CFR 20.2105) if a regulatory limit is exceeded or a Planned Special Exposure (PSE) is used (10 CFR 20.2203, 20.2204, and 20.2205).

Any worker who exceeds a regulatory dose limit shall not be permitted to enter any RCA until all investigations surrounding the event are completed. The RPM or designee must approve reentry.

Any personnel exposure received which is in excess of the limits of 10 CFR 20.1201 shall be reported by the RPM to Radiation Effects Advisory Group (REAG) and the appropriate area chief physician for an examination.

Information regarding an individual's occupational radiation exposure is maintained pursuant to and in accordance with the Privacy Act of 1974, 5 U.S.C. 552a and TVA's Privacy Act regulations (18 CFR 1301 Subpart B).

4.7.6 Personnel Monitoring for Internal Exposures

Internal occupational dose is controlled though facility design, engineering controls, confinement and reduction of contaminated areas, limiting access to radiological controlled areas, and the use of respiratory protective equipment. Personnel are not routinely monitored for internal deposited radioactive material. Confirmatory monitoring (by licensee) is performed for individuals through the assessment and tracking of DAC-h. Radio-bioassay (in vitro and in vivo measurement and analysis) is employed to confirm and/or evaluate probable intake.

The primary means to minimize the intake of airborne radioactive materials is to control the generation of airborne radioactivity. This is best accomplished at its source and by process or other engineering controls. These controls include identification and repair of leaks, process modification, decontamination, containment, and ventilation control. Routine and special tasks should be planned such that potential sources of airborne radioactive material are managed by repair, decontamination, process, or other engineering controls.

If it is impractical to repair, decontaminate, apply process or other engineering controls or while these processes are being implemented, other measures should be taken to limit the uptake of radioactive materials. These measures include increased surveillance, limitation of working times, use of respiratory protective devices, or combination thereof.

Internal Exposure Monitoring and Control Program elements, at a minimum, are to include:

- Air sampling sufficient to identify the potential hazard, permit proper equipment selection, and estimate exposures;
- Surveys and bioassays, as appropriate, to evaluate actual intakes;
- Testing of respirators for operability immediately prior to each use;

- Written procedures shall be established that address: selection, fitting, issuance, maintenance, and testing of respirators, including testing for operability immediately prior to each use; program audits; minimum qualifications of program supervisors and implementing personnel; limitations on periods of respirator use and relief from respirator use; maintaining TEDE ALARA and performing evaluations; supervision and training of personnel; monitoring (including air sampling and bioassays), and recordkeeping; a description of the applications of respirators for routine, non-routine, and emergency respirator use; and periodic medical evaluation (NRC Regulatory Guide 8.15).

- Determination by a physician prior to the initial fitting of respirators, and annually (quarter ending) thereafter or periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection equipment.

Internal dose monitoring (DAC-hr tracking including bioassay) is required for:
Adult workers that are likely to receive an occupational intake in excess of 0.1 ALI or 200 DAC-h in a year.

4.7.7 Evaluation of Dose

A *dose record system shall be implemented* by RP *for purposes of maintaining historical* dose records for all persons for whom personnel monitoring or dose calculations are performed. These records are collected and maintained pursuant to and in accordance with the Privacy Act of 1974, 5 U.S.C. 552a and TVA's Privacy Act regulations (18 CFR 1301 Subpart B). The records maintained shall include: the deep-dose equivalent to the whole-body, lens dose equivalent, shallow-dose equivalent to the skin, and shallow-dose equivalent to the extremities; the estimated intake of radionuclides; the committed effective dose equivalent assigned to the intake of radionuclides; and the specific information used to assess the committed effective dose equivalent pursuant to 10 CFR 20.1204(a) and (c), and when required by 10 CFR 20.2106.

Deep Dose Equivalent, Lens Dose Equivalent, Shallow Dose Equivalent (Whole-body), Shallow Dose Equivalent (Maximum extremity), Committed Effective Dose Equivalent, Committed Dose Equivalent, Total Effective Dose Equivalent, and Total Organ Dose Equivalent dose information shall be calculated, maintained, and reported to the NRC and individuals according to NRC Regulatory Guides 8.7 and 8.34 and NRC Technical Communication RADIATION RECORDS DATA COLLECTION AND ANALYSIS to TVA dated January 4, 1994. The dose record system shall make a clear distinction among the quantities entered on the records (e.g., total effective dose equivalent, shallow-dose equivalent, lens dose equivalent, deep-dose equivalent, committed effective dose equivalent).

Those individuals who receive occupational exposure and require monitoring per 10 CFR 20.1502 shall have their doses reported annually to the NRC and the individuals with greater than 100 mrem of TEDE, EDE, DDE, LDE, SDE, SDEME, CEDE, or CDE on an NRC FORM-5 or an electronic record containing all the information required by a FORM-5.

4.8 ADDITIONAL PROGRAM COMMITMENTS

TVA has developed the following program to track, trend and report attributes of the radiation protection program.

4.8.1 Records and Reporting

A tracking system shall be implemented which will track radiation exposure for purposes of trend analysis and work planning, and provide data for management evaluations of the ALARA program.

A. Exposure Control System

An exposure control system will be implemented which will:

- Keep up-to-date exposure data from dosimeters, calculated doses, and DAC-hr.
- Compare individual dose data with TVA Administrative Dose Limits and regulatory limits.
- Keep the supervisor informed of workers' exposure.
- Keep employees informed of their own exposure.

B. Dose Record System

A dose record system shall be implemented by RP for purposes of maintaining historical dose records for all persons for whom personnel monitoring or dose calculations are performed. These records are collected and maintained pursuant to and in accordance with the Privacy Act of 1974, 5 U.S.C. 552a and TVA's Privacy Act regulations (18 CFR 1301 Subpart B). The records maintained shall include: the deep-dose equivalent to the whole-body, lens dose equivalent, shallow-dose equivalent to the skin, and shallow-dose equivalent to the extremities; the estimated intake of radionuclides; the committed effective dose equivalent assigned to the intake of radionuclides; and the specific information used to assess the committed effective dose equivalent pursuant to 10 CFR 20.1204(a) and (c), and when required by 10 CFR 20.2106.

Deep Dose Equivalent, Lens Dose Equivalent, Shallow Dose Equivalent (Whole-body), Shallow Dose Equivalent (Maximum extremity), Committed Effective Dose Equivalent, Committed Dose Equivalent, Total Effective Dose Equivalent, and Total Organ Dose Equivalent dose information shall be calculated, maintained, and reported to the NRC and individuals according to NRC Regulatory Guides 8.7 and 8.34 and NRC Technical Communication RADIATION RECORDS DATA COLLECTION AND ANALYSIS to TVA dated January 4, 1994. The dose record system shall make a clear distinction among the quantities entered on the records (e.g., total effective dose equivalent, shallow-dose equivalent, lens dose equivalent, deep-dose equivalent, committed effective dose equivalent).

The system includes:

- All official dose records for each individual, including externally measured or calculated doses, whole-body counting results and internal dose commitment calculation, personnel contamination reports, and investigation reports as appropriate.

- Means to store and retrieve records in accordance with NPG's quality assurance program requirements.

- Means to retrieve individual dose records by name or employee identification number.

- Means for RP personnel to obtain individual records.

- Means to generate all required reports.

C. Dose Record Reporting

- Those individuals who receive occupational exposure and require monitoring per 10 CFR 20.1502 shall have their doses reported annually to the NRC and the individuals with greater than 100 mrem of TEDE, EDE, DDE, LDE, SDE, SDEME, CEDE, or CDE on an NRC FORM-5 or an electronic record containing all the information required by a FORM-5.

- These reports are generated and reported by licensee as required by 10 CFR 20.2206.

- External exposures as measured with a NVLAP accredited device will be recorded and reported at a 10 mrem threshold value.

- When determining the dose from airborne radioactive material, NPG shall include the contribution to the deep-dose equivalent, lens dose equivalent, and shallow-dose equivalent from external exposure to the radioactive cloud. External exposures as calculated for noble gas submersion dose will be integrated in the Radiation Protection Records system. Doses calculated by the RP Computer system will be reported at a 1 mrem monitoring period threshold value.

- Internal exposures as calculated for derived air concentration (DAC-hrs) exposures and/or bioassay data will be integrated in the Radiation Protection Records system. Doses calculated by the Radiation Protection Computer system are reported at a 1 mrem threshold.

4.8.2 Abnormal Events and Reporting

All plant abnormal occurrences shall be investigated in accordance the WBN Corrective Action Program.

TVA is required by 10 CFR 50.72 to notify NRC immediately if certain types of events occur. The WBN Unit 1 Operations Department is responsible for making the reportability determinations for 10 CFR 50.72 and 10 CFR 50.73 reports. Operations is responsible for making the immediate notification to NRC in accordance with 10 CFR 50.72.

- 10 CFR 50.72(b)(3)(xii) - Any event requiring the transport of a radioactively contaminated person to an offsite medical facility for treatment.

- 10 CFR 50.73(a)(2)(viii)(A) - Any airborne radioactivity release that, when averaged over a time period of 1 hour, resulted in airborne radionuclide concentrations in an unrestricted area that exceeded 20 times the applicable concentration limits specified in Appendix B to Part 20, table 2, column 1.

- 10 CFR 50.73(a)(2)(viii)(B) - Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentrations specified in Appendix B to Part 20, table 2, column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases.

TVA is required by its various NRC licenses to report certain events or conditions. 10 CFR Part 20 contains reporting requirements for events involving licensed byproduct, source, or special nuclear material. 10 CFR 30.50 contains reporting requirements for events involving licensed byproduct material. 10 CFR 40.60 contains reporting requirements for events involving licensed source material. 10 CFR Part 70 contains reporting requirements for events and conditions involving licensed special nuclear material.

Chapter 5 Table of Contents

5. NUCLEAR CRITICALITY SAFETY

This section of the application contains an overview of the criticality design and administrative controls in place at the Watts Bar Nuclear Plant (WBN). The methodologies and analyses discussed are currently in-place and licensed in support of WBN Unit 1 fuel receipt, handling and storage operations. No changes to the criticality methodologies, analysis or system, structures and component design are required to receive and store new fuel for WBN Unit 2.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed based on the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 5 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 5.1 Nuclear Criticality Safety (NCS) Program				
Management of the NCS Program	70.61(d) 70.64(a)	5.4.3.1	9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling	See Section 5.1
Control Methods for Prevention of Criticality	70.61	5.4.3.4.2	9.1.1 – Criticality Safety of Fresh and Spent Fuel Storage and Handling	See Section 5.1

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 5 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Safe Margins Against Criticality	70.61	5.4.3.4.2	9.1.1 – Criticality Safety of Fresh and Spent Fuel Storage and Handling 9.1.2 – New and Spent Fuel Storage	See Section 5.1
Description of Safety Criteria	70.61	5.4.3.4.2		See Section 5.1
Organization and Administration	70.61	5.4.3.2		See Section 5.1
Section 5.2 Methodologies and Technical Practices				
Methodology	70.61	5.4.3.4.1 5.4.3.4.4 5.4.3.4.6		4.3.2.7 Criticality of Fuel Assemblies
Section 5.3 Criticality Accident Alarm System				
Criticality Accident Alarm System	70.24	5.4.3.4.3	9.1.1 – Criticality Safety of Fresh and Spent Fuel Storage and Handling 9.1.2 – New and Spent Fuel Storage	See Section 5.3
Section 5.4 Reporting				

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 5 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Reporting Requirements	10 CFR 50.72 & 10 CFR 50.73	5.4.3.4.7(7)		See Section 5.4

5.1. NUCLEAR CRITICALITY SAFETY PROGRAM

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and fuel storage facilities and by administrative control procedures in accordance with 10 CFR 50.68, "Criticality accident requirements," paragraph (b). This section identifies those criteria important to criticality safety analyses.

5.1.1 Management of the Nuclear Criticality Safety Program

It is the policy of the TVA Nuclear Power Group to operate its nuclear plants in a safe, conservative and cautious manner such that the health and safety of the public and employees are protected at all times. It is the intent of this policy that reactivity be controlled and managed in a conservative and cautious manner such that the integrity of the fuel cladding and the reactor system pressure boundary is not challenged.

This policy requires that nuclear fuel be operated, handled, and stored in a monitored and defined condition within the bounds of fuel and/or core design limits and analyses assumptions. All activities potentially affecting reactivity must be performed in a well-planned and deliberate manner in accordance with approved procedures. Before any actions are undertaken which could affect reactivity, the effects of the reactivity changes must be known and indications of the effects must be monitored during the changes. All responses to anomalous reactivity indications are required to be conservative actions.

Individuals with reactivity-related responsibilities are required to be capable of recognizing potential reactivity events or conditions and when an unexpected situation occurs, know and take conservative actions. It is not possible to provide procedural guidance for all possible reactivity-related situations; therefore, the key elements of the Reactivity Management Program are a reactivity consciousness and the implementation of conservative actions. The program includes the following.

- Criticality safety requirements have been developed, implemented and maintained to comply with 10 CFR 50.68.
- The criticality analyses are maintained consistent with current configuration by means of the configuration management function described in Chapter 11 of this application.
- Criticality safety limits and requirements are established in Technical Specifications and procedures and maintained consistent with the criticality analyses.

- Modifications to design and to operations procedures are evaluated to ensure that nuclear criticality safety is not adversely impacted.

- Nuclear criticality safety deficiencies are promptly identified by means of operational inspections, audits, and investigations. Deficiencies are entered into the corrective action program so as to prevent recurrence of unacceptable performance deficiencies related to nuclear criticality safety.

Additional discussion of management measures is provided in Chapter 11 of this application.

5.1.2 Control Methods for Prevention of Criticality

The controls implemented at WBN to prevent criticality during the handling and storage of fuel assemblies include WBN Unit 1 Technical Specifications requirements for the storage of new and spent fuel assemblies, plant procedures to control of handling and storage of fuel assemblies to ensure that the assumptions of the criticality safety analyses are satisfied, and procedural requirements to ensure independent verification of certain required activities, e.g., verification of storage of fuel assemblies in proper locations.

5.1.3 Safe Margins Against Criticality/Safety Criteria

The following safe margins/safety criteria are established for the criticality analyses used for new fuel and spent fuel storage.

- The k_{eff} of new fuel in the new fuel storage racks is calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

- If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k_{eff} corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level.

- If no credit for soluble boron is taken, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k_{eff} must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

5.1.4 Organization and Administration

The WBN Unit 1 Shift Manager has the direct responsibility for controlling reactivity. The WBN Unit 1 Reactor Engineers are responsible for performance of the required criticality analyses and establishing the required Technical Specification and procedural limits and controls consistent with assumptions of the criticality analyses. Refer to Chapter 2 of this application for additional information regarding the TVA and WBN organizations.

5.2 METHODOLOGIES AND TECHNICAL PRACTICES

5.2.1 New Fuel Storage

New fuel is normally stored dry in the new fuel storage vault. The design basis for preventing criticality within the new fuel storage vault is that, including uncertainties, there is a 95% probability at a 95% confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.95 under full moderator density conditions and less than 0.98 under low water density (optimum moderation) conditions.

The new fuel rack criticality analysis demonstrated that this rack will meet the design basis limits for k_{eff} for storage of Westinghouse 17x17 STANDARD and VANTAGE 5H fuel assemblies with nominal enrichments up to 4.3 wt% U-235 utilizing all (130) available storage cell locations. The analysis also showed that nominal enrichments above 4.3 wt% and up to 5.0 wt% U-235 can be stored provided that only 120 specific cells of the 130 available locations are utilized. When fuel enrichment above 4.3 wt% are to be stored in the new fuel vault, ten physical restricting devices such as insert plates will be placed in the proper locations to provide additional assurance, over procedural controls, that the fuel will only be stored in the 120 analyzed positions. The insert plates may have a non-fuel bearing component stored in them such as thimble plugging assemblies, rod cluster control assemblies, burnable poison rod assemblies, or tritium producing burnable absorber rod assemblies which are described in FSAR Sections 4.2.3.2.1 and 4.2.4. The allowed location for the 120 usable cells is described in the new fuel storage rack criticality report.

The design method which ensures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes for cross-section generation and KENO IV for reactivity determination. The 227 energy group cross-section library that is the common starting point for all cross-sections used for the benchmarks and the storage rack analysis is generated from ENDF/B-V data. The NITAWL program includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM program which is a one-dimensional Sn transport theory code. These multigroup cross-section sets are then used as input to KENO IV which is a three dimensional Monte Carlo theory program designed for reactivity calculations.

Under normal conditions, the fresh fuel racks are maintained in a dry environment. The introduction of water into the fresh fuel rack area is the worst case accident scenario.

The full density and low density optimum moderation cases are bounding accident situations which result in the most conservative fuel rack k_{eff}.

Other accidents can be postulated which would cause some reactivity increase (i.e., dropping a fuel assembly between the rack and wall or on top of the rack). For these other accident conditions, the double contingency principle of ANSI N16.1-1975, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these other accident conditions, the absence of a moderator in the fresh fuel storage racks can be assumed as a realistic initial condition since assuming its presence would be a second unlikely event.

The maximum reactivity increase for these kinds of postulated accidents is less than 10% delta-k/k, and since the normal, dry fresh fuel rack reactivity is less than 0.70, these postulated accidents will not result in a k_{eff} which is more limiting than the analyzed worst case accident scenarios of full density and optimum moderation water flooding. Thus, using the method described above, the maximum k_{eff} was determined to be less than 0.95, which meets the criteria stated in Section 4.3.1.65.

5.2.2 Spent Fuel Storage - Wet

The high density spent fuel storage racks for WBN are designed to assure that the effective neutron multiplication factor (k_{eff}) is equal to or less than 0.95. Design calculations model the racks fully loaded with fuel of the highest anticipated reactivity, and with a margin for uncertainty in reactivity calculations including mechanical tolerances. Uncertainties are statistically combined, such that the final k_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level.

The layout of storage cells in the WBN spent fuel pool is shown in FSAR Figure 9.1-15. The criticality analysis of the WBN spent fuel pool configuration assures that the maximum k_{eff} will be less than or equal to 0.95 with fuel up to 4.95 $\pm$ .05 wt% U-235 enrichment.

Analysis of the WBN spent fuel rack configuration was performed using the SCALE system of codes for cross section generation and reactivity calculations, and CASMO was used for depletion calculations. The design basis fuel is a 17x17 Westinghouse VANTAGE-5H assembly containing a maximum initial enrichment of 4.95 $\pm$ .05 wt% U-235. The calculations were performed with a moderator temperature of 4°C.

Margin for uncertainty in the reactivity calculations and manufacturing tolerances were included such that the final k_{eff} for allowed storage configurations will be less than or equal to 0.95 with a 95% probability at a 95% confidence level. In order to store fuel with U-235 enrichment as high as 4.95 $\pm$ .05 wt%, administrative controls and burnup credit must be applied. Therefore, the analysis takes credit for the reactivity decrease due to burnup of the stored fuel and for administrative controls on fuel placement. Burnup in discharged fuel was treated using CASMO4, performing depletion calculations which explicitly describe the fission product nuclide concentration. This methodology incorporates approximately 40 of the most important fission products. The fission product nuclide concentrations obtained from the CASMO4 depletions were then modeled in three-dimensions using KENO5a.

The VANTAGE 5H fuel design was modeled as the design basis fuel. The VANTAGE 5H design contains a smaller guide tube outer diameter and thus slightly increased neutron moderation compared with the Westinghouse Standard 17x17 fuel assembly. In addition, VANTAGE 5H fuel assemblies have zircaloy spacer grids as opposed to the more neutron-absorbing material Inconel found on the Standard 17x17 fuel assembly. As a result of these differences, VANTAGE 5H fuel has a higher reactivity for a given enrichment than Standard fuel. Therefore, analysis of VANTAGE 5H fuel also covers storage of Standard 17x17 fuel. VANTAGE 5H fuel assembly data is provided in FSAR Table 4.3-12. The analysis model bounds the design basis fuel assembly using the data provided in FSAR Table 4.3-12 or a more conservative value depending on the specific calculation.

WBN 2 uses Robust Fuel Assembly (RFA)2. An analysis showed the RFA2 fuel design is less reactive than the VANTAGE 5H fuel design at the same enrichment. The ZIRLO material used in the midgrids, fuel cladding and guide tubes has a slight reactivity penalty relative to ZIRC-4. Therefore, the analysis of VANTAGE 5H also covers and is considered bounding for the RFA2 fuel design.

5.2.3 Analytical Technique and Results

As previously discussed, the criticality analysis for the WBN racks were performed primarily with KENO5a, a three-dimensional Monte Carlo computer code, using the 238-group SCALE cross-section library and the Nordheim integral treatment for resonance shielding effects found in NITAWL. Depletion analyses were performed using CASMO4, a two-dimensional transport theory code. The models included explicit descriptions of the fission product nuclide concentrations, incorporating approximately 40 of the most important fission products.

Analysis of the spent fuel racks confirmed the racks can safely and conservatively accommodate storage of fuel up to 5 wt% U-235 enrichment with the following storage conditions:

1. Fuel assemblies with 3.8 wt% or less U-235 enrichment may be stored in Region 1 without restrictions.

2. Fuel assemblies with initial with enrichment greater than 3.8 wt% and up to 5.0 wt% (4.95 ± 0.05) U-235 and less than a maximum of 5.0 wt% (4.95 ± 0.05) may be stored in one of four arrangements with the limits specified below:

   A. Fuel assemblies may be stored in the racks without further restrictions provided the burnup of each assembly is in the acceptable domain identified in FSAR Figure 4.3-46, depending on the specified initial enrichment.

   B. New and spent fuel assemblies may be stored in a checkerboard arrangement of 2 new and 2 spent assemblies, provided the accumulated burnup of each spent assembly is in the acceptable domain identified in FSAR Figure 4.3-47, depending on the specified initial enrichment.

C. New fuel assemblies may be stored in 4-cell arrays with 1 of the 4 cells remaining empty of fuel (containing only water or water with up to 75% by volume of non-fuel bearing material).

D. New fuel assemblies with a minimum of 32 integral fuel burnable absorber (IFBA) rods may be stored in the racks without further restrictions provided the loading of ZrB_2 in the coating of each IFBA rod is a minimum of 1.25x (1.9625 mg/in).

A water cell is less reactive than any cell containing fuel and therefore may be used at any location in the loading arrangements. A water cell is defined as a cell containing water or non-fissile material with no more than 75% of the water displaced.

The WBN Unit 1 Technical Specifications include curves defining the limiting burnup for fuel of various initial enrichments for both unrestricted storage and checkerboard arrangements assuming the fresh fuel region is enriched to 4.95$\pm$0.05 wt% U-235. The calculated maximum reactivity is 0.948, which is within the regulatory limit of a k_{eff} of 0.95. This maximum reactivity includes calculational uncertainties and manufacturing tolerances (95% probability at the 95% confidence level), an allowance for uncertainty in depletion calculations, and the evaluated effect of the axial distribution in burnup. Fresh fuel of less than 4.95% enrichment would result in lower reactivity.

Accounting for biases and uncertainties, the maximum k_{eff} values for the above spent fuel storage rack conditions are less than 0.95. The maximum k_{eff} was determined as follows:

$$k_{eff} = k_{eff}\text{ (KENO)} + \text{BIASES} + \text{UNCERTAINTIES}$$

Biases include the CASMO and KENO method biases, a boron particle self-shielding allowance, and a bias for the extrapolation of enrichment from the critical benchmark comparisons. The uncertainties include the KENO statistical uncertainty, the KENO and CASMO method uncertainties, and the mechanical tolerance uncertainty.

The analyses conservatively do not take credit for presence of borated water, presence of discrete burnable absorbers, lower enrichment and higher burnup which would decrease reactivity. Other conservative assumptions include:

- Ignoring radial neutron leakage from the spent fuel storage racks
- Ignoring the presence of control rods
- Ignoring the presence of spent burnable absorber assemblies in storage
- Ignoring the higher water temperature of the spent fuel pool
- Maximizing burnable poison history effects
- Maximizing water density history effects
- Minimizing the ^{10}B content in the Boral

A water gap of 1.5 inches between Region 1 and Region 2 racks, two rack modules with Boral panels on both sides of the water gap (i.e., a flux trap), precludes any adverse interaction between the two regions modules.

The effect of various parameters on reactivity was determined to ensure the conservatism of the analysis. This was accomplished by performing sensitivity studies on these parameters with either KENO or CASMO-3. Parameters evaluated were axial burnup distribution, water temperature/density, assembly placement, mechanical tolerances, poison loading, pellet density, cell dimensions/bow, boron particle self shielding effect, borated water activity worth, Boral width tolerance, cell lattice spacing tolerance, stainless steel thickness tolerance, and fuel enrichment and density tolerance.

5.2.4 Credit for Soluble Boron

Although credit for soluble poison normally present in the spent fuel pool water is permitted under abnormal or accident conditions (double contingency principle), most abnormal or accident conditions will not result in exceeding the limiting reactivity (k_{eff} = 0.95) even in the absence of soluble poison. However, the inadvertent misplacement of a fresh fuel assembly in a location intended to be a water cell has the potential for exceeding the limiting reactivity and results in the worst-case accident scenario, should there be a concurrent loss of all soluble boron. Misplacement of a fuel assembly outside the periphery of a storage module, or a dropped assembly lying on top of the rack would have a smaller reactivity effect. Under this worst-case accident condition, calculations show that approximately 55 ppm of soluble boron would be sufficient to ensure that the limiting k_{eff} of 0.95 is not exceeded. Assuring the presence of soluble boron during fuel handing operations will preclude the possibility of the simultaneous occurrence of the two independent accident conditions. Administrative controls require that the spent fuel pool boron concentration be monitored (to ensure at least 2000 ppm) during operations requiring fuel moves in the pool until verification is made of assembly locations.

5.3 CRITICALITY ACCIDENT ALARM SYSTEM

In accordance with 10 CFR 50.68(b)(3), radiation monitors are provided in the storage and associated handling areas when fuel is present. These radiation monitors are capable of detecting excessive radiation levels and allow appropriate safety actions to be taken in accordance with plant procedures.

5.4 REPORTING

Reports to NRC associated with nuclear criticality safety shall be made in accordance with 10 CFR 50.72, “Immediate notification requirements for operating nuclear power reactors,” and 10 CFR 50.73, “Licensee event report system,” as applicable.

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6 CHEMICAL PROCESS SAFETY

As described in NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," Chapter 6, "Chemical Process Safety," the primary purpose of the NRC review is to determine with reasonable assurance that the applicant has designed a facility that will provide adequate protection against chemical hazards related to the storage, handling, and processing of licensed materials. Chapter 6 of NUREG-1520 also states that the facility design must adequately protect the health and safety of workers and the public during normal operations and credible accident conditions from the chemical risks of licensed material and from hazardous chemicals produced from licensed material.

The activities associated with this license application include receiving, possessing, inspecting, and storing special nuclear materials in the form of 193 fully assembled new fuel assemblies for the initial core of the WBN Unit 2 reactor. The special nuclear material is fully contained within the ZIRLO cladding of these fuel assemblies and does not represent a chemical hazard. For the scope of this license application, i.e., to receive, inspect, handle and store new fuel assemblies, there are no credible accident conditions from the chemical risks associated with the contained special nuclear material within the new fuel assemblies. In addition, since the special nuclear material is contained within the new fuel assembly cladding, there are no hazardous chemicals produced from the contained special nuclear material. As a result, there are no chemical process safety hazards associated with the receipt, inspection, handling, and storage of new fuel assemblies for WBN Unit 2.

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7 FIRE SAFETY

This section of the application contains an overview of the fire protection design and associated administrative controls at the Watts Bar Nuclear (WBN) Plant Site and the types of activities that will be performed when receiving, possessing, inspecting, and storing special nuclear materials in the form of 193 fully assembled fuel assemblies for the initial core of the WBN Unit 2 reactor.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) *and information previously docketed in the Watts Bar* Nuclear Plant Final Safety Analysis Report (FSAR), which was developed using the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 7 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 7.1 Fire Safety Management Measures	70.62(a),(d) & 70.64(b)	7.4.3.1	9.5.1 Fire Protection Program	9.5.1 Fire Protection System
Section 7.2 Fire Hazards Analysis	70.61(a),(c) & 70.62(a) & (c)	7.4.3.2	9.5.1 Fire Protection Program	9.5.1 Fire Protection System
Section 7.3 Facility Design	70.62(a),(c) & 70.64(b)	7.4.3.3	9.5.1 Fire Protection Program	9.5.1 Fire Protection System
Section 7.4 Process Fire Safety	70.64(b)	7.4.3.4	9.5.1 Fire Protection Program	9.5.1 Fire Protection System

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 7 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 7.5 Fire Protection and Emergency Response	70.62(a),(c) & 70.64(b)	7.4.3.5	9.5.1 Fire Protection Program & 13.3 Emergency Planning	9.5.1 Fire Protection System & 13.3 Emergency Planning

The purpose of the WBN Fire Protection Report (FPR) is to consolidate a sufficiently detailed summary of the WBN regulatory required Fire Protection Program into a single document and to reflect the design as-constructed at the time of fuel load. The Final Safety Analysis Report (FSAR) references this report as detailing WBN's Fire Protection Program. This report is updated in conjunction with updates to the FSAR. The Fire Protection Report has been developed in accordance with the guidelines of NRC Generic Letter 86-10, "Implementation of Fire Protection Requirements" and NRC Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications". The FPR brings WBN into compliance with NRC recommendations for documenting the Fire Protection Plan and commitments.

7.1 Fire Safety Management Measures

WBN administers and ensures fire safety in accordance with the WBN Fire Protection Report (FPR). The fire safety management measures included in the FPR applicable to the receipt, inspection, handling and storage of new fuel are as follows

7.1.1 Fire Brigade

Effective handling of fire emergencies is an important aspect of the WBN Fire Protection Program. This is accomplished by trained and qualified emergency response personnel. The fire response organization is staffed and equipped for firefighting activities. The fire brigade is comprised of a fire brigade leader and four fire brigade members. The fire brigade shall not include the Shift Manager nor the other members of the minimum shift crew necessary for safe shutdown of the unit, nor any personnel required for other essential functions during a fire emergency. Additional support is available when needed through an agreement with a local fire department.

An Incident Commander is available to direct each shift fire brigade. The Incident Commander meets the requirements of a Unit Supervisor, Shift Technical Advisor or Shift Support Supervisor and has sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability.

7.1.2 Training and Qualifications

WBN fire brigade training ensures that the fire brigade's capability to combat fires is established and maintained. Prior to training and annually thereafter (with a 25% allowable extension), each fire brigade member and leader receives medical evaluation to ensure the ability to perform strenuous physical activity, to wear special respiratory equipment, and for unescorted access to nuclear plants.

The training program consists of initial (classroom and practical) training and recurrent training which includes periodic instruction, fire drills and annual fire brigade training. In addition, fire brigade leaders receive additional training that provides the fire brigade leader with the knowledge and skills necessary to supervise and direct the activities of the fire brigade during an incident.

7.1.3 Availability of Firefighting Equipment

Firefighting equipment for the Fire Brigade is provided throughout the plant. The availability of firefighting equipment is such that delays in obtaining equipment by the fire brigade for fire emergencies will be minimized. Firefighting equipment may, alternatively, be staged adjacent to or at the access to areas/locations to facilitate equipment availability. This may be necessary to address equipment surveillance test concerns relative to life safety and ALARA practices.

7.1.4 Fire Emergency Procedures and Pre-fire Plans

Fire emergency procedures and pre-fire plans specify actions taken by the individual discovering a fire and actions considered by the emergency response organization. Included in these procedures are operational instructions for response to the fire detection system annunciation. These procedures provide different levels of response based on whether there is an actual fire or an annunciation (e.g., a single zone annunciation in a cross zoned area will not carry the same level of response as a cross zone annunciation in the same area). An annunciation may or may not carry the same level of response as the report of a fire by site personnel. Pre-fire plans are not intended to establish a procedure or step-by-step process but to provide guidance, depending upon the particular circumstances, to aid in firefighting efforts. It is recognized that many different firefighting techniques or strategies exist which would be acceptable for fire suppression efforts.

7.1.5 Control of Combustibles

Combustibles are controlled to reduce the severity of a fire which might occur in a given area and to minimize the amount and type of material available for combustion. The use and application of combustible materials at WBN are controlled utilizing the following methods:

- Instructions/guidelines provided during general employee training/orientation programs.
- The chemical traffic control program.
- Periodic plant housekeeping inspections/tours by management and/or the plant fire protection organization.
- Design/modification review and installation process.
- Administrative procedures (e.g., Transient Combustible Control Program).

The fire protection organization performs a periodic fire safety inspection of the safety-related areas of the plant to identify and minimize potential fire hazards. The use and handling of combustible materials such as fire retardant-treated lumber, paper, plastic, and flammable/combustible gases and liquids are controlled in safety-related areas. The use of untreated lumber requires specific approval of the fire protection organization. Combustible materials generated as a result of work activities are removed/cleaned up from the work area at the end of the shift or at the conclusion of the work activity, whichever is sooner. The storage of combustible materials within safety-related areas is controlled by the fire protection organization. The control of hazardous waste and hazardous materials is conducted in accordance with the chemical control and hazardous material processes.

Design considerations in the control of combustibles is utilized when appropriate. For example, these considerations include the application of noncombustible or limited combustible construction materials or components, use of noncombustible fluids in operating equipment, dikes, or containments provided for equipment containing combustible liquids, etc.

Combustible Control Zones (CCZs) are established at WBN to strictly control or prohibit the placement of transient combustibles. Transient combustibles brought into CCZs require an evaluation in accordance with site administrative procedures. The strict control or prohibition of combustibles by site procedures within the combustible control zone provides reasonable assurance that fire will not propagate and jeopardize required equipment or components.

7.1.6 Control of Ignition Sources

The use of ignition sources such as welding, flame cutting, thermite welding, brazing, grinding, arc gouging, torch applied roofing, and open flame soldering within safety-related areas are controlled through the approval and issuance of an ignition source permit. Permits are reviewed and approved by appropriate plant personnel. The ignition source permit is valid for one job. Job area inspection shall be performed and documented at the start of each shift that ignition source activities are being performed. If no ignition sources activities are performed, then reinspection is not required.

Designated ignition source activity areas are located and approved by the fire protection organization. A fire watch system shall be established for all ignition source work activities that are performed in safety-related areas of the plant. Ignition source fire watches are established and will remain for 30 minutes following the elimination of the ignition source, unless other durations are approved by the fire protection organization.

7.2 Fire Hazards Analysis

As discussed above the Fire Hazards Analysis (FHA) is part of the FPR. The FHA results are documented on a fire area basis, broken down into separate discussions of classical fire protection features and safe shutdown analysis for each fire area. The FHA includes the following:

- A summary of the evaluation performed to determine the adequacy of the fire protection features for each fire area.
- A discussion of the ability to achieve safe shutdown in case of a fire in each fire area.

The fire hazards and safe shutdown evaluation were performed by qualified nuclear, mechanical, electrical and fire protection engineers. The deviation requests and evaluations applicable to each fire area are also summarized.

7.3 Facility Design

7.3.1 Building Construction

The facility is designed in accordance with 10 CFR 50, Appendix A, General Design Criteria 3, which requires that noncombustible and fire-resistant materials be used throughout the facility. Noncombustible materials are used to the extent practicable.

7.3.2 Fire Area Determination and Fire Barriers

Fire area barriers are 2-hour or 3-hour rated. WBN fire areas and room compartmentation does not always comply with the specific fire barrier rating guidelines contained in BTP 9.5-1 Appendix A. The differences are judged acceptable given the extensive use of suppression systems at WBN, the low combustible loading in many areas of the plant, the detailed and rigorous Appendix R analysis performed, the conservative nature of the plant design evaluations and the fire hazards analysis performed.

Penetrations in these barriers, including conduit and piping, are generally sealed or evaluated to provide a fire-resistant rating equivalent to the required rating of the barrier.

Normally, doors, frames and hardware in required fire barriers have a fire rating equivalent to the required rating of the barrier, and have been tested and approved by a nationally recognized laboratory. Fire doors have been evaluated per the requirements of NFPA 80-1975.

7.3.3 Electrical Installation

Plant design minimizes the use of combustible material. Cables within certain areas are generally coated with a fire resistant coating or are qualified to the requirements of IEEE 383-1974. Noncombustible material is used for cable tray construction. Where appropriate, in situ plastics are included in the fire area combustible inventories utilized in the FHA.

High amperage transformers are not installed within building spaces. Transformers installed within safety-related buildings are either dry-type or insulated and cooled with "high fire point" liquid.

7.3.4 Life Safety

Access and egress routes are established in the Prefire Plans and are included as part of the drills practiced by operating and fire brigade personnel. Stairwells in the Control Building are enclosed and designed to minimize smoke infiltration.

Emergency lighting and communication are provided. Fixed emergency lighting consists of sealed beam units with individual 8-hour minimum battery power supplies are provided for access and egress routes. An alternate emergency communication system consisting of sound powered phones with head sets is provided.

NIOSH-approved self-contained full-face positive pressure breathing apparatus is available for the fire brigade, damage control and control room personnel. The operating life of the self-contained units is a minimum of one-half hour.

7.3.5 Ventilation

Plant ventilation systems are generally used for smoke removal, or manual smoke venting can be performed with portable smoke ejectors located on site. Non recirculating ventilation systems are provided for fire areas which may contain airborne radioactive materials. Smoke from fires which might occur in areas containing radioactive materials are monitored for radioactivity.

7.3.6 Drainage

Means of drainage is provided in the main buildings. In areas containing fire suppression systems or hose stations, drainage provided removes expected fire protection water flows or controls accumulations or such water could not cause unacceptable damage to equipment in the area. Water drainage from areas which may contain radioactivity are sampled and analyzed before discharge to the environment.

7.3.7 Lightning Protection

Lightning protection is incorporated in the facility design. A direct low impedance path for the lightning to travel to ground, rather than through structures and / or equipment, is provided. The lightning protection system consists of three basic parts which provide the low impedance path:

- The air terminals on roofs and other elevated locations
- The ground grid
- The conductors connecting the air terminals to the ground grid

7.3.8 Criticality Concerns

Criticality analyses of new fuel assemblies, under the analyzed worst case accident scenarios of full density and optimum moderation water flooding, have been performed. These analyses demonstrate, under these conditions, that the new fuel assemblies remain subcritical. Refer to FSAR Section 4.3.2.7 for discussion of these analyses. As such, actuation of the plant automatic fire suppression systems or use of manual suppression systems will not result in new fuel assembly criticality.

7.4 Process Fire Safety

The Fire Hazards Analysis summarizes the engineering evaluations performed to determine the adequacy of the fire protection features for the fire areas and rooms identified for WBN to ensure process fire safety. The Fire Hazards Analysis also summarizes the physical characteristics of required fire barriers (including fire doors and fire dampers), combustible loading and fire severity, suppression and detection capabilities, deviations and evaluations, and fire safe shutdown capability for each area and room.

7.5 Fire Protection and Emergency Response

7.5.1 Fire Protection

The fire protection equipment in the fuel handling area of the Auxiliary Building is common to both WBN Unit 1 and WBN Unit 2.

Equipment available during fuel receipt and movement for the fuel cask receipt area (Auxiliary Building, elevation 729) consists of the following:

a) *A minimum of five 10-pound dry chemical fire extinguishers located in the cask* receiving area and adjacent nitrogen storage area.

b) Two 1 1/2-inch hose stations equipped with 100 feet of hose and fog nozzles (ABC-rated). One hose station is located in the cask fuel receipt area and the other is located in the adjacent nitrogen storage area.

Equipment available during fuel storage inside the new fuel storage vault and/or the spent-fuel storage pit (Auxiliary Building, elevation 757) consists of the following:

a) A minimum of four 10-pound dry chemical fire extinguishers located strategically on the refueling floor.
b) One 100-pound C02 or dry chemical wheeled extinguisher located in the area.
c) Two 1 1/2 inch hose connections equipped with 100 feet of hose and adjustable fog nozzles (ABC-rated). One hose station is located south of stairway No. 4, and the other is available from the 1 1/2 inch Siamese connection in the Unit 1 Reactor Building access room.

A fire pump, with a flow path to the hose stations listed above, will be available.

Site procedures for the maintenance and surveillance testing of the above-listed equipment, including fire pump, fire mains, standpipes, and hoses, have been developed and will be performed as described in the FPR. In addition, the compensatory actions described in the FPR will be used should any of the listed fire equipment become unavailable.

7.5.2 Emergency Response

Effective handling of fire emergencies is an important aspect of the WBN Fire Protection Program. This is accomplished by trained and qualified emergency response personnel. The fire response organization is staffed and equipped for firefighting activities. The fire brigade is composed of a fire brigade leader and four fire brigade members. The fire brigade does not include the Shift Manager or other members of the minimum shift crew necessary for safe shutdown of the unit, nor any personnel required for other essential functions during a fire emergency. Additional support is available when needed through an agreement with a local fire department.

Training ensures that the fire brigades capability to combat fires is established and maintained. The training program consists of initial (classroom and practical) training and recurrent training which includes periodic instruction, fire drills and annual fire brigade training.

Firefighting equipment is provided throughout the plant. Fire emergency procedures and prefire plans specify actions taken by the individual discovering the fire and by the emergency response organization. A specific pre-fire plan has been prepared for the fuel receipt area and the fuel storage area. Discussion of this pre-fire plan is included in the periodic classroom instruction's training program provided for the emergency response team.

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8 EMERGENCY MANAGEMENT

This section contains a brief discussion of the radiological emergency plan (REP) developed for the Watts Bar Nuclear Plant. Based on the shared nature of the areas where new fuel will be received, handled, inspected and stored, the existing Appendix C to the REP is applicable to this application.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed using the guidance in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 8 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Emergency Plan	70.22(i)(3)	8.4.3.1	13.3 Emergency Planning	13.3 Emergency Planning and REP
Facility Description	70.22(i)(3)(i)	8.4.3.1.1	2.1.1 Site Location and Description	1.2.2: Facility Description
Onsite and Offsite Emergency Facilities	70.22(i)(3)(i)	8.4.3.1.2	13.3 Emergency Planning	13.3 Emergency Planning and REP
Types of Accidents	70.22(i)(3)(ii)	8.4.3.1.3	13.3 Emergency Planning	13.3 Emergency Planning and REP

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 8 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Classification of Accidents	70.22(i)(3)(iii)	8.4.3.1.4	13.3 Emergency Planning	13.3 Emergency Planning and REP
Detection of Accidents	70.22(i)(3)(iv)	8.4.3.1.5	13.3 Emergency Planning	13.3 Emergency Planning and REP
Mitigation of Consequences	70.22(i)(3)(v)	8.4.3.1.6	13.3 Emergency Planning	13.3 Emergency Planning and REP
Assessment of Releases	70.22(i)(3)(vi)	8.4.3.1.7	13.3 Emergency Planning	13.3 Emergency Planning and REP
Responsibilities	70.22(i)(3)(vii)	8.4.3.1.8	13.3 Emergency Planning	13.3 Emergency Planning and REP
Notification and Coordination	70.22(i)(3)(viii)	8.4.3.1.9	13.3 Emergency Planning	13.3 Emergency Planning and REP
Information to be Communicated	70.22(i)(3)(ix)	8.4.3.1.10	13.3 Emergency Planning	13.3 Emergency Planning and REP
Training	70.22(i)(3)(x)	8.4.3.1.11	13.3 Emergency Planning	13.3 Emergency Planning and REP
Safe Shutdown (Recovery and Facility Restoration)	70.22(i)(3)(xi)	8.4.3.1.12	13.3 Emergency Planning	13.3 Emergency Planning and REP
Exercises and Drills	70.22(i)(3)(xii)	8.4.3.1.13	13.3 Emergency Planning	13.3 Emergency Planning and REP

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 8 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Responsibilities for Developing and Maintaining the Emergency Program and Procedures Correct	N/A	8.4.3.1.14	13.3 Emergency Planning	13.3 Emergency Planning and REP

The TVA Radiological Emergency Plan (REP) and Emergency Plan Implementing Procedures have been developed to provide protective measures for TVA personnel and to protect the health and safety of the public in the event of a radiological emergency resulting from an accident at WBN. The REP fulfils the requirements set forth in 10 Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities. It also satisfies the requirements of NUREG-0800, Chapter 13.3 Emergency Planning. The REP contains site-specific appendices for each TVA plant. WBN's radiological emergency information is in Appendix C of the REP. Changes to the REP are processed in accordance with 10 CFR 50.54(q).

For events related to fuel handling, Appendix C contains emergency action levels that are common to both Watts Bar Unit 1 and Unit 2. These emergency action levels address events occurring in the common spent fuel pool area and include, loss of water level in the spent fuel pool, loss of spent fuel pool cooling, and elevated radiation levels.

A detailed description of the Watts Bar Nuclear Plant is contained in Chapter 1, General Information.

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9 ENVIRONMENTAL PROTECTION

This section of the application describes the Watts Bar Nuclear (WBN) Plant Site environmental protection measures associated with the receipt, possession, inspection, and storage special nuclear materials in the form of 193 fully assembled fuel assemblies for the initial core of the WBN Unit 2 reactor.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed using the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 9 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 9.1 Environmental Report	70.21(h)	9.4.3.1.1	None	None
9.1.1 Date of Application	70.21(f)	9.4.3.1.1(1)	None	None
9.1.2 Environmental Considerations	51.45(b)	9.4.3.1.1(2)	None	None
9.1.3 Analysis of Effects of Proposed Actions and Alternatives	51.45(c)	9.4.3.1.1(3)	None	None
9.1.4 Status of Compliance	51.45(d)	9.4.3.1.1(4)	None	None

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 9 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
9.1.5 Adverse Information	51.45(e)	9.4.3.1.1(5)	None	None
Section 9.2 Environmental Protection Measures	70.22(a)(8)	9.4.3.2	-	-
9.2.1 Radiation Safety	20.1101(a)	9.4.3.2.1	12.3	12.3 Radiation Protection Design Features and 12.5 Radiological Control Program
ALARA Controls and Monitoring	20.1101(d)	9.4.3.2.1(1)-(3)	12.1	12.1 Assuring that Occupational Radiation Exposures are as Low as Reasonably Achievable
Waste Minimization	20.1406	9.4.3.2.1(4)	11.2, 11.3 and 11.4	11.2 - Liquid Waste Systems, 11.3 - Gaseous Waste Systems, and 11.5 Solid Waste Management System
9.2.2 Effluent and Environmental Controls and Monitoring	70.59(a)(1)	9.4.3.2.2	-	-
9.2.2.1 Effluent Monitoring	20.1501(a)	9.4.3.2.2(1)	11.5	11.4 Process and Effluent Monitoring and Sampling System
9.2.2.2 Environmental	20.1501(a)	9.4.3.2.2(2)	None	None

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 9 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Monitoring				
9.2.2.3 ISA Summary	70.65(b)	9.4.3.2.2(3)	15	15 Accident Analysis

TVA's Final Supplemental Environmental Impact Statement (FSEIS) for the Completion and Operation of Watts Bar Nuclear Plant Unit 2 was issued on June 23, 2007. The TVA Board authorized completion of WBN Unit 2 on August 1, 2007. Subsequently, TVA informed the NRC of its intention to reactivate and complete construction activities at WBN Unit 2. The TVA Board Record of Decision was posted in the Federal Register on August 15, 2007. The FSEIS was submitted to the NRC on February 15, 2008. The FSEIS includes an evaluation of the need for increased baseload power; an analysis of potential socioeconomic, cultural, and environmental effects of completing WBN Unit 2; and it identifies potential mitigation measures.

This FSEIS supplements TVA's original 1972 "Final Environmental Statement, Watts Bar Nuclear Plant Units 1 and 2." In December 1978, NRC issued a "Final Environmental Statement Related to the Operation of Watts Bar Nuclear Plant Units 1 and 2, NUREG-0498." In 1993, TVA conducted a review to determine whether additional environmental review was needed to inform decision makers about whether to complete both units and concluded that neither plant design nor environmental considerations had changed in a manner that materially altered the environmental impact analysis set forth in its 1972 Final Environmental Statement (FES). TVA provided additional analyses and information in support of NRC's "Final Environmental Statement Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2, NUREG-0498," which was issued in April 1995. Following an independent review of NRC's analyses and a new analysis of the need for additional power, TVA adopted NRC's 1995 FES in July 1995.

Other major reviews of WBN environmental impacts include TVA's cooperation with the U. S. Department of Energy in evaluating the production of tritium in commercial light water reactors, which resulted in a 1999 "Final Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor." Also, in February 2004, TVA issued its "Reservoir Operations Study Final Programmatic Environmental Impact Statement" evaluating the impacts of alternative ways of operating TVA's reservoir system, the water supply needs of TVA's generating facilities, including WBN, and compliance with environmental permits. A more detailed description of environmental reviews and studies pertaining to the operation and construction of WBN is provided in the FSEIS.

TVA's assessment of the actions required to complete WBN Unit 2 as described in the FSEIS remains valid, and no additional environmental reviews are anticipated at this time. TVA will, of course, review and assess any supplemental environmental review completed by the NRC in connection with the completion and operation of WBN Unit 2 in the future. Background information and analyses used in the preparation of TVA's FSEIS, including that associated with the severe accident analysis section, are available at the WBN site for review.

9.1 Environmental Report

TVA's Final Supplemental Environmental Impact Statement (FSEIS) for the Completion and Operation of Watts Bar Nuclear Plant Unit 2 was submitted to the NRC on February 15, 2008.

9.1.1 Date of Application

The 10 CFR 70 license application, requesting approval to receive, possess, inspect, and store special nuclear materials in the form of 193 fully assembled fuel assemblies for the initial core of the WBN Unit 2 reactor, was submitted in November 2009. TVA expects to receive the first shipment of new fuel for WBN Unit 2 in the second quarter of 2011 at the earliest.

9.1.2 Environmental Considerations

The impact of the activities in this license application are bounded by the NRC FES. The environmental considerations of the entire fuel cycle were analyzed as part of NUREG-0498 Environmental Statement related to operation of Watts Bar Nuclear Plant Units Nos. 1 and 2. In NUREG-0498 Supplement 1, the NRC concluded that there were no significant changes in the environmental impacts since the NRC 1978 FES-OL.

9.1.3 Analysis of Effects of Proposed Actions and Alternatives

TVA's FSEIS provides a description of the proposed action (Chapter 1), the purpose of the proposed action (Chapter 1), a description of the affected environment (Chapter 3), and a discussion of considerations (Chapter 3). TVA's FSEIS provides an analysis of the effects of the proposed action and alternatives (Chapters 2 and 3).

9.1.4 Status of Compliance

TVA's FSEIS provides a discussion of the environmental permits and approval required for the operation of WBN Unit 2 (Chapter 1). Because WBN Unit 1 is already operating, there should be few additional permits and approvals required. The FSEIS documents TVA's compliance with the National Historic Preservation Act (Section 3.7)

9.1.5 Adverse Information

Various sections of the FSEIS discuss adverse effects. TVA's FSEIS Table 2-1 provides a summary of the environmental effects from completing WBN Unit 2.

9.2 Environmental Protection Measures

TVA is committed to protecting the public, plant workers and the environment from the harmful effects of ionizing radiation due to plant operation.

9.2.1 Radiation Safety

FSAR section 12.5 provides details of the radiological control program including the organization, equipment and procedures. FSAR section 12.3 describes specific design features to limit in plant radiation exposures. TVA has a formal program to ensure that occupational exposure to employees is kept as low as reasonably achievable. This program is discussed in FSAR section 12.1.

9.2.2 Effluent and Environmental Controls and Monitoring

9.2.2.1 Effluent Monitoring

FSAR section 11.4 describes the process and effluent radiological monitoring and sampling system. Specific monitoring capability applicable to fuel assembly handling and storage is as follows.

Spent Fuel Pool Accident Radiation Monitors

These monitors continuously monitor the fuel pool area. Two Geiger Mueller tubes with preamplifiers are mounted above the fuel pool. A high radiation signal initiates Auxiliary Building ventilation isolation. In addition, a high radiation signal from these monitors during refueling operations with containment and/or the annulus open to the Auxiliary Building ABSCE spaces will result in a containment valve Isolation (CVI). The two fuel pool monitors are supplied from separate Class 1E power supplies. The setpoint of these monitors is selected to prevent exceeding a significant fraction of the 10 CFR 100 limits subsequent to a fuel handling accident in the Auxiliary Building. These monitors are safety related.

Auxiliary Building Vent Monitor

The Auxiliary Building Vent Monitor assembly continuously monitors the Auxiliary Building Vent stack exhaust for radioactivity. The effluent stream is sampled by an isokinetic sampling probe assembly fitted with 72 sample nozzles. The nozzles are arranged such that a representative sample of the effluent stream is taken. The monitor consists of a particulate, gas, and iodine channel. The monitor noble gas and particulate detectors are beta scintillators. The iodine detector is a gamma scintillator. Particulate and iodine filters are available for laboratory analysis. Monitor setpoints for the gas channel are established using the methodology provided in the Offsite Dose Calculation

Manual. Setpoints for the particulate and iodine channels are based on plant personnel protection requirements.

9.2.2.2 Environmental Monitoring

Environmental monitoring requirements are included in the Watts Bar Nuclear Plant National Pollutant Discharge Elimination System (NPDES) permit. In accordance with Appendix B "Environmental Protection Plan" of the WBN Unit 1 Operating License, TVA provides an annual nonradiological environmental operating report. This report provides a summary of the reports submitted as specified in the NPDES permit and other, non-routine and special biological monitoring, reports.

9.2.2.3 ISA Summary

FSAR Chapter 15 addresses accident analysis. Specific sections of this chapter address normal operation and operational transients, faults of moderate frequency, infrequent faults, and limiting faults. FSAR section 15.5 addresses the environmental consequences of accidents.

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10 DECOMMISSIONING

At the time of new fuel receipt, decommissioning funding may not be in place. Since the new fuel will not have been activated in the reactor, residual radioactivity from operation will not exist. In the event a decision was made to delay or defer Watts Bar Unit Two after fuel receipt, the fuel could be returned to the vendor.

Reasonable assurance of decommissioning funding will be provided in accordance with the requirements of 10 CFR §50.33(k)(1), as part of the application for an operating license (OL) that will contain information in the form of a report, as described in 10 CFR §50.75, indicating how funds will be available to decommission the facility.

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11 MANAGEMENT MEASURES

It is the policy of the Tennessee Valley Authority (TVA) that activities which affect quality be accomplished in a planned and systematic manner to achieve compliance with pre-established quality objectives and acceptance criteria. Accordingly, Nuclear Assurance has established and will maintain a Nuclear Quality Assurance Program (NQAP). The NQAP includes the Nuclear Quality Assurance Plan and the approved documents which are used to implement the Plan. The quality assurance program and requirements for specific items and activities are applied commensurate with their importance to safe, reliable nuclear operations, construction, and independent spent fuel storage.

Management policies and requirements for the TVA NQAP are established by the Chief Operating Officer through the Chief Nuclear Officer and Executive Vice President, Nuclear Power Group (NPG), for operating units and the Senior Vice President, Nuclear Generation Development and Construction, for units with construction permits. These management policies and requirements provide the controls that must be applied to the activities performed by and for the agency to ensure implementation of TVA commitments.

This section contains a brief discussion of the management measures described in the final safety analysis report and the TVA Quality Assurance Plan and the Organizational Topical Report. Both documents are applicable to the Watts Bar Nuclear Plant and the receipt, handling, inspection and storage of new fuel.

The level of detail provided in this chapter is based on a comparison of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (July 1981) and information previously docketed in the Watts Bar Nuclear Plant Final Safety Analysis Report (FSAR), which was developed using the guidance provided in NUREG-0800, and the guidance recommended by NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (NRC, 2002). Where a comparison is made, a brief discussion of the area is provided with the detailed discussion incorporated by reference. Where no direct comparison is available, a detailed discussion is provided.

The following table provides the requested information, the corresponding regulatory requirement, the applicable section of NUREG-1520, the applicable section of NUREG-0800 and the applicable section(s) of the WBN FSAR.

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 11 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 11.1 Configuration Management	70.62(d) & 70.72	11.4.3.1	17.1 Quality Assurance During the Design and Construction	17.1 Quality Assurance for Design and Construction 17.2 Quality

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 11 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
			Phases 17.2 Quality Assurance During the Operations Phase 17.3 Quality Assurance Program Description	Assurance for Station Operation
Section 11.2 Maintenance	70.62(d)	11.4.3.2	17.6 Maintenance Rule	17.2 Quality Assurance for Station Operation TVA Nuclear Quality Assurance Plan
Section 11.3 Training and Qualifications	70.62(d) & 10 CFR 19	11.4.3.3	13.1.2, 13.1.3 Operating Organization 13.2.1 Reactor Operator Requalification Program, Reactor Operator Training 13.2.2 Non-Licensed Plant Staff Training	13.1.3 Qualification Requirements for Nuclear Facility Personnel 13.2 Training Programs
Section 11.4 Procedures Development and Implementation	70.62(d) & 70.22(a)(8)	11.4.3.4	13.5.1 Administrative Procedures	13.5 Site Procedures
Section 11.5 Audits and Assessments	70.62(d)	11.4.3.5	13.4 Operational Programs	13.4 Review and Audit

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 11 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
Section 11.6 Incident Investigations and Corrective Action Process	70.74(a)& (b) 70.62(a)(3)	11.4.3.6	17.1 Quality Assurance During the Design and Construction Phases 17.2 Quality Assurance During the Operations Phase	17.1 Quality Assurance for Design and Construction 17.2 Quality Assurance for Station Operation
Section 11.7 Records Management	70.62(a)(2) & (3) 70.62(d)	11.4.3.7	17.1 Quality Assurance During the Design and Construction Phases 17.2 Quality Assurance During the Operations Phase 17.3 Quality Assurance Program Description	13.6 Plant Records
Section 11.8 Other QA Elements	70.62(d)	11.4.3.8	17.1 Quality Assurance During the Design and Construction Phases 17.2 Quality Assurance During the Operations Phase 17.3 Quality Assurance	17.1 Quality Assurance for Design and Construction 17.2 Quality Assurance for Station Operation

Information Category and Requirement	10 CFR Citation	NUREG-1520 Chapter 11 Reference	NUREG-0800 Comparable Chapter	WBN FSAR CHAPTER / TITLE
			Program Description	

11.1 Configuration Management

Configuration management is a critical element of the engineering standard programs as well as essentially all other functional areas involved with operating, maintaining and modifying a nuclear plant. It encompasses and is implemented through various plant organizations' procedures which are established to ensure the objectives below are achieved. The detailed aspects of configuration management are integrated into many of the engineering processes and procedures to ensure that (1) plant structures, systems, components, and computer software conform to approved design requirements, and (2) the plant's physical and functional characteristics are accurately reflected in plant documents, plant simulator, and other data systems.

Configuration management philosophies are incorporated into processes for (1) operating and maintaining the plant systems and components, (2) evaluation of hardware and components to meet the plant design basis, (3) generation of design output and changes to plant configuration, (4) installation and testing of plant systems and components, and (5) revision, updating, storage, and retrieval of documents which document the configuration of the plant.

The controls established in these processes shall ensure that design bases are maintained, design output is consistent with the defined bases, the as-built plant configuration meet design output requirements, and the as-built documents accurately reflect the plant's configuration.

Design Basis Management and Control

The design basis for plant and system performance shall be established and maintained for systems and components critical to the safe and reliable operation of the plant. Controls and requirements for the establishment and maintenance of the design bases will address identification, establishment, and maintenance of design related configuration documents and design/licensing basis documents.

Design Change Control

The design change control process is established to ensure that new designs, as well as changes to existing designs, satisfy plant design bases and established design requirements. The process ensures that additional design input considerations such as constructability, system and component operability and maintainability, radiological protection, and operating experience are included in the design. The design process ensures effective resolution of plant problems and enhancement of plant safety and reliability.

The key elements of the design change process include the following:

- Issue identification and analysis

- Evaluation of alternative solutions
- Authorization
- Detailed design development and change package issuance
- Installation
- Testing
- Return to service
- Documentation updates
- Package close-out activities

Design Input Control
Design input consists of design requirements that govern the design of plant structures, systems, and components. These design inputs are used to develop and support design output. Examples of design input includes laws, regulations, industry codes and standards, design bases, interface requirements, design criteria, documented tests and NPG design standards, design guides, and standard procurement specifications. Mechanisms are established to ensure appropriate design inputs are incorporated into engineering designs for plant systems and facilities.

Design Output Control
Design outputs:

- Correlate the technical and design requirements applicable to structures, systems and components to the required physical configuration in the plant, and/or
- Communicate engineering requirements which affect plant activities (e.g., construction, installation, operation, maintenance, modification, surveillance, and testing).

Documents which constitute design output are defined by engineering management and are based on approved and issued design input. Other organizations shall take engineering requirements only from those documents identified as design outputs.

Design Verification
Design verification is the process of reviewing, confirming or substantiating design inputs and outputs by one or more methods to provide assurance that safety-related and, where specifically required, quality-related designs meet the specified design inputs and will not unacceptably increase the probability or consequences of potential adverse events.

Use and Control of Design Standards and Guides
Engineering utilizes standards and guides to provide input to and support the design process. The term standards and guides is the term used to collectively denote Design Standards, Design Guides, drafting standards, standard drawings, standard specifications, general engineering specifications, and any other documents that provide proven and accepted engineering and design parameters, practices and/or approaches, designs, or technical requirements. Requirements and controls for the use, development, review, and approval of the standards and guides are specified in order to ensure they are authorized, applicable, accurate, and up to date.

Operational Configuration Controls

- **Fuel Related**
  Plants are operated with a strategic objective of zero fuel defects. Fuel supply, fuel design, plant operations, refueling outages, and other related activities will be controlled and managed to comply with applicable Technical Specification and regulatory requirements, licensing commitments, licensing and design bases, or additional commitments made due to industry practices.

  Fuel shall be stored only in approved locations. Approved locations are those licensed by the NRC. These are the reactor cores, the fuel storage racks and shipping containers. Requirements, restrictions, limitations, and controls for these locations are given in site Technical Specifications for cores and racks and in Certificates of Compliance for containers. To preclude the possibility of accidental criticality when fuel is outside of these locations, limited quantities of fuel are allowed out of approved storage locations.

  The maximum quantity of fuel assemblies allowed out of approved storage locations per approved plant procedures are as follows:

  - One un-irradiated fuel assembly shall be allowed within the fuel-handling area. The fuel handling area includes all areas of the refueling floor where un-irradiated fuel assemblies are handled outside of metal shipping containers. The fuel-handling area also includes the new fuel storage vault and the truck bay where metal shipping containers are unloaded.
  - One fuel assembly shall be allowed within the spent fuel storage pool boundary (excluding the inspection, reconstitution, or cleaning locations with appropriate evaluation for each configuration that must be performed prior to implementation). The spent fuel storage pool boundary includes the cask loading area, fuel transfer canal (excluding the transfer cart), and spent fuel pool.
  - Three fuel assemblies shall be allowed within the refueling canal. The refueling canal includes the fuel transfer tube boundary (including the transfer cart) and the rod cluster control changing fixture. This allows for two fuel assemblies to be in the rod cluster control changing fixture while the third fuel assembly is being transferred through the fuel transfer tube, is in the upender, or is in transit to or from the reactor cavity.
  - One fuel assembly shall be allowed within the reactor cavity.
  - Loose fuel rods or pellets must be evaluated for criticality before removal from a fuel assembly or storage at the site.

- **System Status Control**
  The responsibilities and programmatic methods have been established for obtaining, maintaining, and documenting system status as well as documenting off-normal alignments not controlled by other administrative or procedural control.

- **Clearance Program**
  Processes have been established to ensure that, before any employee performs any service or maintenance on a machine or equipment where the unexpected energizing, startup, or release of stored energy could occur and cause injury or

equipment damage, the machine or equipment is isolated from the energy source and rendered inoperative.

11.2 Maintenance

The maintenance and modification (M/M) program assures that equipment, systems, and structures (1) are maintained and modified in accordance with applicable requirements, (2) supports safe, reliable, and efficient operation of the nuclear power plants, and (3) are maintained at a quality level required for them to perform their intended functions as specified in the original design, material specifications, and inspection requirements. In the context of this program, the modification process refers only to the physical implementation of design changes.

Corrective Maintenance (CM)

Corrective maintenance is the classification of any work on systems, structures, or components (SSCs) where the SSC has failed or is significantly degraded to the point that failure is imminent (within its operating cycle/preventive maintenance interval) and no longer conforms to or is incapable of performing the SSC's design function. An SSC should be considered failed or significantly degraded if the deficiency is similar to any of the following examples:

- Is removed from service because of actual or incipient failure
- Significant component degradation that affects system operability-The SSC may be determined operable by engineering assessment, but the degradation is significant and requires immediate corrective action. This normally includes any deficiency that requires a basis for continued operation as defined in NRC Generic Letter 91-18, and should be considered as corrective maintenance.
- Creates the potential for rapidly increasing component degradation (for example, leaks of borated water, steam leaks where cutting degradation is possible)
- Releases fluids that create significant exposure or contamination concerns (or has the potential to under postulated accident conditions)
- Adversely affects controls or process indications that directly or indirectly impair operator ability to operate the plant or that reduce redundancy of important equipment
- Significant component degradation identified from the conduct of predictive, periodic, or preventive maintenance which, if not resolved, could result in equipment failure or significant additional damage prior to its next scheduled preventive maintenance period

Preventive Maintenance (PM)

PM consists of predictive, periodic, and planned maintenance actions taken to maintain equipment within design operating conditions and extend its life. PM is performed before equipment failure. This work is controlled by the work order (WO) process. The program requires that site PM activities be performed on critical components and be re-evaluated, revised, or updated periodically based on industry experience, plant equipment history, or trend analysis.

Predictive Maintenance

Predictive maintenance results from vibration analysis, thermography, etc., should be used to trend and monitor equipment performance so that needed planned maintenance can be performed before equipment failure or to prevent equipment failure, and that periodic maintenance can be modified to prevent future equipment failures.

Periodic Preventive Maintenance
Periodic PM activities are performed on a routine basis on equipment to prevent breakdown and involve servicing such as lubrication, filter changes, cleaning, and adjustments.

Planned Preventive Maintenance
Planned PM activities are performed before equipment failure but not necessarily on a routine basis like periodic PM activities. Planned PM can be initiated by predictive or periodic maintenance results, vendor recommendation, experience, or identification in the field such as during operator rounds.

Long-Term Maintenance Plan (Rolling Schedule)
The long-term maintenance plan is a product of the preventive and surveillance process, and specifies the frequency for implementation of maintenance and surveillance activities necessary for the reliability of components in each system. The rolling schedule includes the preliminary defense-in-depth assessment, which documents the allowable combinations of system and Functional Equipment Groups (FEGs) that may be simultaneously worked on line or during shutdown conditions. FEGs are common sets of boundaries encompassing equipment that has been evaluated for acceptable out-of-service combinations. They are used to schedule planned maintenance and establish equipment clearances.

Surveillance/monitoring
A Surveillance Test Program has been established to ensure that plant equipment and components will continue to operate or operate on demand in accordance with design and other regulatory requirements. Technical requirements are specified in Plant Technical Specifications, Technical Requirements Manuals, Offsite Dose Calculation Manuals, and plant Fire Protection Plans/Reports.

Within the Surveillance Test Program, controls have been established to ensure that required testing is identified test instructions are prepared which satisfy regulatory requirements, tests are scheduled and conducted within prescribed frequencies, and tests results are documented and reviewed to ensure that system/component performance satisfies the identified acceptance criteria.

Equipment and Maintenance Activities Requiring Post Maintenance Testing (PMT)
Post-maintenance testing shall be based on the extent of maintenance performed. The PMT shall be sufficiently comprehensive to ensure that the maintenance performed does not adversely affect the equipment's ability to perform its intended function, that the original deficiency has been corrected, and that no new or related problems were created by the maintenance activity.

Equipment within the scope of the PMT program is plant safety-related, quality related and non-quality-related equipment necessary for plant operations.

All work orders (WOs) do not require PMT; e.g., WOs which do not perform physical work such as inspection activities. Maintenance activities on plant equipment under Operations' control which require PMT are exemplified by the following:

- Maintenance that affects the integrity or operation of a fluid or gas system, or components within those systems,
- Maintenance that affects the wall thickness of pressure boundaries or affects mechanical strength of components or fittings,
- Maintenance that affects the function of electrical distribution equipment,
- Maintenance that affects the function of electrical control circuitry or electronic components,
- Maintenance that affects the function of instrument detectors or components in an instrument loop,
- Maintenance that affects the engineered or design function of a system or component such as pressure, flow rate, etc.,
- Maintenance that requires the development of pre-maintenance tests, e.g., containment isolation valves requiring a local leak rate test before maintenance.

Return to operability (RTO) testing shall be considered for maintenance activities on equipment with Technical Specification operability requirements.

11.3 Training and Qualifications

The purpose of the TVA qualification and training program is to provide criteria for the training and qualification of personnel for TVA's nuclear power plants. The program addresses the training of personnel in operating and support organizations to ensure the safe and efficient operation of nuclear power plants. The program also addresses the qualifications of personnel who occupy positions to which TVA is committed through its licensing documents. This commitment is to ensure that the minimum qualifications, which are contained in national standards, for positions in operating and support organizations, are appropriate for the safe and efficient operation of TVA's nuclear power plants.

To ensure the qualifications for these positions, TVA will meet the requirements of Regulatory Guide 1.8, Revision 2 (4/87) for all new personnel qualifying on positions identified in regulatory position C.1 after January 1, 1990. Personnel qualified on these positions prior to this date will still meet the requirements of Regulatory Guide 1.8, Revision 1-R (5/77). As specified in regulatory position C.2, all other positions will meet the requirements of ANSI/ANS N18.1-1971. TVA's Nuclear Power Group (NPG) is committed to comply with the requirements of ANSI N18.1-1971 and ANSI/ANS 3.1-1981 as endorsed by the Regulatory Guide 1.8, Revision 2, April 1987, "Qualification and Training of Personnel for Nuclear Power Plants" except as outlined in the Nuclear Quality Assurance Plan, Appendix B. Qualifications of key members of the organization are contained in Section 2.2.4 of this application.

It is the policy of TVA Nuclear Power Group (NPG) to develop and implement performance-based training programs which promote and support the safe, reliable, and efficient operation of TVA's nuclear power plants. This demands that the personnel who operate, maintain, and support those plants be fully qualified to perform their duties. Effective training is an essential element of achieving and maintaining such qualifications. Effective training thus requires definition of the skills, knowledge, and competencies necessary to perform required duties; establishment and implementation of learning opportunities which develop those desired skills, knowledge, and competencies; and, documentation of attainment of such skills, knowledge, and competencies.

The programs for training of nuclear power plant personnel subject to accreditation by the National Nuclear Accrediting Board are developed and maintained by the responsible training managers using a Systems Approach to Training (SAT). Program guidelines promulgated by Federal regulation, pertinent ANSI/ANS Standards, the Academy, and the Institute for Nuclear Power Operations (INPO) are used to develop those training programs.

Training program procedures (TRNs) shall be developed by the Training Managers Peer Team for the following Academy-accreditable training programs:

- Non-licensed operator (initial and continuing training)
- Reactor operator (initial training)
- Senior reactor operator (initial training)
- Shift manager (initial training)
- Continuing training for licensed personnel (including simulator training and control room team training)
- Shift technical advisor
- Instrument and control technician
- Electrical maintenance personnel
- Mechanical maintenance personnel
- Chemistry technician
- Radiological control technician
- Engineering Support personnel
- Maintenance supervisor

11.4 Procedures

The hierarchy of NPG procedures is defined in the NPG Procedure and Document Control Program. Descriptions of the major types of procedures are given below to assist in the determination of where a particular procedure fits in this hierarchy.

Standard Programs and Processes (SPPs)

These procedures describe administrative controls for processes that cross organizational boundaries, and based on their content, must be available, understood, and followed by all personnel. For example, clearance administration and fitness for duty procedures are SPPs since all employees need to be aware of these processes and their requirements. The following functional areas are provided from TVA's Administration of Standards Programs and Processes (TVA-SPPs). This list provides a sample of the types of procedures developed to support TVA Nuclear Power Group functions associated with nuclear plant operations:

- Policy and Management
  Includes administrative controls necessary to ensure consistency in TVA policies, programs, procedures, process documentation, process improvement and assessment methods.

- Performance Planning
  Includes business planning, comparative analysis, benchmarking, project justification, performance and resource planning, measurement and analysis, improvement initiative justification and cost analysis.
- Regulatory Compliance
  Includes legislative and regulatory legal requirements, licensing and corrective action program.
- Supply Chain Management
  Includes all activities and supporting processes and systems related to sourcing strategy; supplier relations; contracting for products and services; transportation and TVA logistics; materials management, including receipts, warehousing, distribution, inventory strategy, inventory management, disbursement, and disposal of all surplus material. This excludes contracts for purchase and sale of power, purchase and transportation of fossil fuel, the sale of fossil operation byproducts, the purchase and sale of land, the sale of services, loan agreements, and cooperative agreements.
- Environmental Management
  Includes Environmental Management System (EMS), environmental compliance, pollution prevention and control, environmental reviews (NEPA), hazardous material management, waste management, air & water quality, environmental stewardship, emergency preparedness, and environmental auditing.
- Asset Maintenance and Modification
  Includes asset maintenance, modification and unit optimization.
- Work Management
  Includes planning and scheduling system administration, work control and outage planning and management.
- Fuel Management
  Includes supply planning, purchase, transport and management of both fossil and nuclear fuel.
- Engineering and Technical Support
  Includes project design and management, configuration and design change control, Includes specialized services related to TVA's core businesses and delivered to external customers and internal TVA organizations. These specialized services include: energy and environmental technologies and services, integrate resource management tools and power production and delivery technologies.
- Asset Operations
  Includes plant operations and clearance procedures

Standard Department Procedures (SDPs)
SDPs describe administrative controls for processes that normally do not cross organizational boundaries and are generally contained within one organization. SDPs, like SPPs, are applicable to all NPG sites and locations unless reduced or limited applicability is specified in the procedure.

Site Instructions
Site Instructions are used to specify implementing instructions in the operation and maintenance of the plant. These instructions are normally technical in nature and are not administrative procedures. Examples of Site Instructions are Surveillance Instructions, Maintenance Instructions, Physical Security Instructions, Radiological Control Instructions, and Operating instructions. Site instructions are typically site-specific, but in some cases they may be common procedures used at all sites. If common procedures are used, licensing and Technical Specification requirements must be met for all sites.

Review and Approval Process
A procedure review and approval process will be established to meet all applicable regulatory and NQAP requirements. It shall include provisions for affected organization reviews, reviews required by individual site technical specifications and other regulatory documents, and independent technical reviews for quality-related procedures and major revisions to these procedures.

Verification and Validation of Procedures
The Procedure Control Program includes a Verification and Validation (V&V) Program for, as a minimum, critical quality-related, man-machine interface procedures. For Emergency Operating Instructions (EOIs), the V&V Program will meet the requirements of the applicable Owner's Group Guidelines for EOIs. The verification process ensures a thorough and detailed review of the procedure before approval to ensure, to the extent practical, that the procedure is complete, accurate, and can be performed as written. The validation process will normally be conducted after approval by actual performance of the procedure, and provides (1) validation that the procedure can be performed as written and (2) a mechanism for the performer to provide feedback to the author on ways to improve the procedure from a performer's perspective.

11.5 Audits and Assessments

Audits
Measures shall be established to implement a comprehensive audit program which consists of internal audits, including NPG and other TVA organizations, which support the nuclear program and contractor/supplier audits to determine and assess the adequacy and effectiveness of the QA program.

Program Elements
- An audit plan shall be prepared identifying the audits to be performed and their frequencies and schedule.
- Audits shall include: a determination of the effectiveness of QA program elements; evaluation of work areas, activities, processes, and items; review of documents and records; review of audit results with responsible management; follow-up on corrective action taken for deviations identified during the audit; and escalation to appropriate senior management of any safety significant disagreement between the auditing organization and the organization or function being audited.
- Audits shall be performed in accordance with written procedures or checklists by qualified, certified, and appropriately trained personnel not having direct responsibilities in the areas being audited.
- Audited organizations shall provide access to facilities, documents, and personnel needed to perform the audits. They shall take necessary action to correct deviations identified by the audit in a timely manner.

Assessments
Quality Assurance Assessments are performed as a type of verification to ensure that observed quality-related activities are performed in accordance with requirements and desired results are achieved.

Program Elements
- Assessment procedures and instructions shall address assessment techniques.

- Assessment frequencies shall be based on such factors as the status and safety significance of the activity or process, frequency of occurrence, degree and acceptability of previous experience, adverse trends, and testing or operation sequences.
- The results of assessments shall be documented and reported to appropriate levels of management.
- Records shall be maintained in sufficient detail to provide adequate documentation of assessed activities.
- Follow-up verifications or additional assessments shall be conducted as necessary to ensure that required corrective action has been taken.
- Assessments shall be performed in accordance with written procedures and instructions by qualified and appropriately trained personnel not having direct responsibility in the areas being assessed.

11.6 Incident Investigations

Measures shall be established to ensure that items that do not conform to requirements are controlled to prevent their inadvertent installation or use. Adverse conditions, including nonconforming items or non-hardware problems such as failure to comply with operating license, technical specifications, or procedures, shall be identified, evaluated, corrected, tracked, trended, and when required, reported to appropriate levels of management. Procedures or instructions implementing the corrective action program shall establish the criteria for documenting and tracking adverse conditions.

NPG organizations, NGDC, and onsite non-NPG service organizations performing quality-related activities at nuclear facilities shall promptly identify and resolve adverse conditions.

Minor deficiencies which may be brought into compliance within an acceptable timeframe shall be corrected on the spot in accordance with established instructions.

Adverse conditions shall be dispositioned by organizations with defined responsibility and authority and shall be corrected in accordance with documented plans.

Disposition actions for nonconforming items may be accept-as-is, repair, rework, scrap, or return to vendor. Dispositions of accept-as-is or repair shall be reviewed and approved by Corporate or Site Engineering or, for nuclear fuel-related items, Nuclear Fuels. Reworked or repaired, and replaced items shall satisfy the original inspection and test requirements or acceptable alternatives.

The cause of significant adverse conditions shall be determined and corrective action taken to preclude recurrence. Significant adverse conditions shall be reported to appropriate levels of management.

The satisfactory completion of corrective actions shall be verified and documented by the appropriate organization. Independent verification of corrective action implementation is performed as specified within the corrective action program.

11.7 Records Management

The QA program requires that for activities affecting quality, measures shall be established to ensure that documents prescribing the activity, including changes, are approved for release by authorized personnel, reviewed for adequacy, and made available to personnel performing the prescribed activity prior to commencing work.

Identification and Distribution

- The types of documents to be controlled shall be identified.
- Master document indexes shall be established and maintained for identifying all controlled documents and their revision status.
- The distribution of documents shall be controlled and maintained to assist in preventing the use of obsolete or superseded documents.

Controlled Use

- Quality related activities shall be performed in accordance with approved and controlled instructions, procedures, and drawings.
- Organizations shall ensure through procedures or instructions that those participating in an activity are made aware of and use proper and current documents.

Control of Equipment Technical Information

- Administrative controls shall provide for control and distribution of equipment technical information (ETI) supplied to TVA.

11.8 Other QA Elements

Other QA elements and their application are as described in the TVA Nuclear Quality Assurance Plan (NQAP). The TVA NQAP addresses and complies with the 18 criteria provided in 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." In addition, changes to the TVA NQAP are performed in accordance with 10 CFR 50.54, "Conditions of licenses," paragraph (a).

Enclosure 3

Special Nuclear Material Control Summary

Special Nuclear Material Control Summary

This 10 CFR 70 materials license will permit TVA to own, acquire, receive, possess, use, and transfer the special nuclear material (SNM) at its Watts Bar Nuclear Plant (WBN) site for use at WBN Unit 2. The SMN will be controlled using WBN Unit 1 procedures. WBN Unit 1 is a 10 CFR 50 Licensee (License Number NPF-90). TVA maintains a procedure for control and accountability of SNM in accordance with the applicable regulations. This procedure will be utilized to maintain the material addressed in this 10 CFR 70 license application. The procedure includes provisions to:

1. Establish, maintain, or follow written material control and accounting procedures to account for SNM.
2. Maintain adequate records of the initial receipt or current inventory of SNM, including records of isotopic content, material received, material shipped, and material lost (material balance reports and physical inventory listing reports DOE/NRC forms 742 and 742C).
3. Develop adequate inventory procedures or maintained adequate perpetual inventory records, including traceability of items to ultimate disposal.
4. Inventory SNM within the 12-month prescribed frequency.
5. Report SNM inventories on the applicable forms.
6. Establish an individual responsible for the control and accountability of SNM.
7. Report the loss of or inability to find SNM items in a timely manner.
8. Control access to SNM.
9. Control the shipping and transfer of SNM.

Enclosure 4

Physical Security Plan/Contingency Plan Summary

Physical Security Plan / Contingency Plan Summary

TVA's plan for protection of the Watts Bar Nuclear Plant is contained in a separate controlled document (i.e., the Watts Bar Nuclear Plant Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan). This document provides a comprehensive description of the physical security program for the plant site (composed of WBN Units 1 and 2) which includes physical barriers and means of detecting unauthorized intrusions; provisions for monitoring access to vital equipment and access control; communication systems for security; provisions for maintenance and testing of security systems; arrangements for law enforcement assistance; provisions for responding to security threats; and required organizational charts and drawings that depict the site layout. Based on the shared nature of the areas where new fuel will be received, handled, inspected and stored, the existing Watts Bar Nuclear Plant Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan is applicable to this application and is incorporated by reference.