TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

June 24, 1982

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

R. L. Tedesco's letter dated December 30, 1981 issued the draft Safety Evaluation Report (SER) for Watts Bar Nuclear Plant units 1 and 2. Comments on the draft SER were provided in my March 5, 1982 letter. Enclosed for your information are additional comments resulting from our review of the March 28, 1982 version of the draft SER which was informally provided to us. These comments were discussed with the NRC Licensing Project Manager for Watts Bar on June 14, 1982. For your convenience, the comments are provided as "marked-up" pages from the draft SER.

If you have any questions concerning this matter, please get in touch with D. A. Kulisek at FTS 858-2681.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 24th day of June 1982

Bryant M. Lowery
Notary Public
My Commission Expires 4/8/86

B001

Enclosure
cc: U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

An electrically heated pressurizer connected to the hot-leg piping of one of the loops will establish and maintain the reactor coolant pressure and provide a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation.

The steam produced in the steam generators will be utilized to drive a tandem compound double-stage reheat turbine and will be condensed in a triple-shell single-pass deaerating condenser. Cooling water from the cooling tower basin ~~drawn from Chickamauga Lake~~ will be pumped through the tubes of the condenser to remove the heat from, and thus condense, the steam after it has passed through the turbine. The condensate will then be pumped back to the steam generator to be heated for another cycle. ~~Depending on conditions in Chickamauga Lake~~, The condenser cooling water will ~~either~~ be ~~returned directly to the lake,~~ passed through two natural draft cooling towers and ~~then returned to the lake, or passed through the cooling towers and returned to the intake channel.~~ return to the cooling tower basin.

The reactor will be controlled by a coordinated combination of a soluble neutron absorber (boric acid) and mechanical control rods whose drive shafts will allow the plant to accept step load changes of 10 percent and ramp load changes of 5 percent per minute over the range of 15 to 100 percent of full power under normal operating conditions. With steam bypass, the plant will also have the capability to accept a 50-percent step load rejection without reactor trip.

Plant protection systems are provided that automatically initiate appropriate action whenever a monitored condition approaches preestablished limits. These protection systems will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

Supervision and control of both the NSSS and the steam and power conversion system for both units will be accomplished from the main control room.

The emergency core cooling system for the plant consists of accumulators, upper head injection, and both high- and low-pressure injection subsystems with provisions for recirculation of the borated water after the end of the injection

The nearest railroad is a spur line of the CNO&TP (Southern Railway System), which runs parallel to State Route 68 about 1 mi north of the plant and which terminates at the Watts Bar steam plant. This spur is used to ship heavy components to both the steam plant and the nuclear plant. The railroad will continue to be used as a coal supply route for the steam plant. The nearest major railroad is the main line of the CNO&TP which runs parallel to U.S. Highway 27 about 6.5 mi west of the site.

The Chickamauga Reservoir of the Tennessee River, which forms the eastern boundary of the site, is used for barge transportation of commercial cargo. The intake structure is protected against a potential barge impact by virtue of its location approximately 800 ft from the edge of the Chickamauga Reservoir. The intake structure is connected to the reservoir by an excavated channel that is perpendicular to the main flow of the river. On the basis of the intake structure location, the staff concludes that it is very unlikely that a drifting or runaway barge will impact the intake structure.

Fuel oil is shipped by barge past the Watts Bar site. In case of a fuel oil barge accident, fire and dense smoke may result. The intake pumping station is protected against fire by virtue of design and location. The pump suction is taken from the bottom of the channel. Thus, even if fuel oil from a spill should reach the intake station, entry of oil into the intake is unlikely because the oil will float, and the pump suction is about 10 ft below the water surface. All pumps and essential cables and instruments are protected from fire because they are enclosed within concrete walls. The control room air intake system will automatically isolate (see Section 6.4 of this report) when it detects high smoke concentrations in the outside air supply. The staff concludes that barge accidents involving spillage of fuel oil and resulting fires will not adversely affect the safe operation of the plant.

Although chlorine is not frequently shipped past the Watts Bar site, ~~The~~ the control room air intake system will also automatically isolate (see Section 6.4 of this report) when it detects high chlorine concentrations in the outside air supply. Consequently, accidents involving chlorine release will not adversely affect the safe operation of the plant.

is evident for winds at the 300-ft level, with winds from the southwest and south-southwest occurring almost 30 percent of the time with winds from the north-northeast and northeast occurring about 25 percent of the time. Winds from the south-southwest at the 30-ft level have been observed to persist for 37 consecutive hours, and winds from the north-northeast have been observed to persist for 26 consecutive hours.

The median wind speed at the 30-ft level is about 3 mph, and more than 95 percent of the winds occur with speeds less than 12 mph. Calm conditions (defined as an hourly average wind speed below the starting threshold of the anemometer) were reported only about 0.4 percent of the time for the 2-year period July 1973 to June 30, 1975. This frequency of observed calm conditions for this 2-year period of record contrasts markedly with the observed frequency of calm conditions for data collected previously at the site. For the period July 1, 1971 to June 30, 1972, calm conditions were reported for 11.6 percent of the time. The applicant believes that the higher frequency of calm conditions and low wind speed conditions in general for this period of record is a consequence of tower location. A temporary onsite meteorological tower was apparently located in a slight topographic depression for the data collection period July 1971 to June 1972, resulting in sharply reduced wind speeds. A permanent onsite meteorological tower was installed in a different location in May 1973, and the temporary tower was decommissioned in September 1973. The applicant performed a comparison of data collected while both towers were in operation from June to September 1973 and concluded that the temporary tower was located in an area affected by low-level "drainage" airflow (principally a nighttime phenomenon where differential cooling of the ground surfaces causes cooler, more dense air to flow towards lower terrain) resulting in abnormally high frequencies of low wind speeds and very stable atmospheric conditions.

Inversions predominate at the Watts Bar site. Slightly stable (Pasquill type "E"), moderately stable (Pasquill type "F") and extremely stable (Pasquill type "G") conditions occur about 16 and 9 percent, respectively.
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As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of

The staff has reviewed and concurred with the applicant's sedimentation estimates. The staff has also made conservative estimates of sediment accumulation during extreme flood events (less than 1 ft) and concludes that sedimentation in the intake canal will not be significant and can be controlled under a formal inspection and maintenance program as described in Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants."

The staff has reviewed the applicant's provisions for the safety-related water supply and concludes that all provisions are acceptable and meet the suggested criteria of Regulatory Guide 1.27. Therefore, the staff concludes that the hydrologic aspects of the UHS meet the requirements of GDC 44.

2.4.7 Groundwater

WBN FSAR Fig. 2.5-11 depicts TVA's interpretation of the subsurface

The materials immediately beneath the site are older terrace deposits and alluvium that are poor waterbearing strata. At deeper levels, discontinuous beds of Conasauga shale (more properly a combination of about 84 percent shale and 16 percent limestone), Chickamauga limestone, and Knox dolomite as found. The Knox dolomite constitutes the primary regional aquifer where water is found in solution channels and openings formed along bedding planes and joints. The general region is known as a Karst area where flowing water is sometimes found in significant solution channels. At the site, groundwater in contact with plant structures move generally toward the river.

TVA has developed three wells 2.5 mi northwest of the plant drawing from the Knox dolomite. The requirements placed on these wells include 16,000 gpd for potable uses at the plant and 200,000 gpd for offsite uses.

2.4.8 Design Basis for Subsurface Hydrostatic Loading

The applicant has constructed an underdrain system around the structures in the power block to reduce water pressures on the buildings and potential inleakage. The system is a virtual duplicate of that provided at TVA's Sequoyah plant. However, because of different groundwater and foundation conditions at Watts Bar, the Watts Bar system is located at relatively high levels compared to basement levels. A single pipe conduit, fed and discharged by gravity, and porous elevation 726.0 ft msl. The associated review of structural design capabilities to withstand groundwater loads is contained in Section 3.8 of the SER.

The applicant has ~~not~~ identified a design-basis groundwater level for the ERCW pipeline ~~Consequently, the staff concludes that ground elevation must be used as the design-basis groundwater level for the ERCW pipeline.~~ route for use in the liquefaction evaluation of soils along the pipeline route. This design basis is contained in a March 19, 1982 letter from the applicant. For design of the pipeline, a design-basis ground water level was not required.

The staff finds that the above provisions for design-basis groundwater level meet the requirements of GDC 2 and are acceptable.

2.4.9 Transport of Liquid Releases

Normal releases of low level radioactivity in water from the plant will be made through the plant discharge system. In addition, some accidental releases could occur via the same route. TVA has evaluated the dilution of such releases in the Tennessee River using conservative assumptions. At the closest major downstream intake, at Dayton, plant releases to the river would be likely to be diluted by a factor of 3.6×10^8 or more.

TVA also evaluated the travel time and dilution that would occur if an accidental release were to be made to groundwater. TVA concluded that contaminated water would take 812 days to reach the nearest point of known possible use, which is a tributary of Yellow Creek. The point of groundwater discharge is 2600 ft from the plant. TVA also estimated a dilution factor of about 9.8×10^6.

The staff has independently reviewed the potential for accidental release of radioactive liquid to the groundwater at the Watts Bar plant and has concluded that the applicant-identified pathway to Yellow Creek may not be the most critical in terms of the potential for offsite contamination. The shortest groundwater pathway to a potential user is to the southeast about 1600 ft to the Tennessee River/Chickamauga Reservoir. However, there is a permanent, passive, dewatering system at about elevation 710.0 ft msl that would capture any radioactive liquid that might seep through the walls above the 710.0 ft msl level. The dewatering system discharges, by gravity, to a holding pond and then to the cooling tower blowdown discharge facilities. A radioactivity monitor on cooling tower blowdown is provided downstream of the holding pond and

2.5.1.2 Tectonic Setting

Because earthquake activity cannot be reasonably associated with known geologic structure in the Southern Appalachians, earthquakes are instead identified with the tectonic province in which the site is located. In a report, Southern Appalachian Tectonic Study (Seay, 1979), the applicant attempted to establish new tectonic provinces based on geophysical and seismic information. The subdivisions proposed divided the Southern Appalachians lengthwise into northern and southern parts along the New York-Alabama magnetic lineament and then transverse to the Appalachian trend along northwest-southeast lines. An important result of these proposed province boundaries was to isolate the TVA nuclear power plants from the Giles Gounty seismicity. However, staff consultants, the U.S. Geologic Survey (USGS), in a status review (Nelson and McDowell, 1980) rejected the TVA study conclusions because they did not provide an adequate basis for the tectonic subdivision's for regulatory purposes. The NRC staff adopted the USGS position. The applicant, therefore, indicates that the site is located in the Southern Appalachian Tectonic Province. As defined, this province is bounded on the east by the western margin of the Piedmont Province, on the west by the western ~~eastern~~ limits of the Cumberland Plateau, on the south by the overlap of the Gulf Coastal Plain Province, and on the north by the re-entrant in the Valley and Ridge Province near Roanoke, Virginia.

In its review, the staff determined that the proposed site is within the Southern Valley and Ridge Tectonic Province based on provinces which are more in accord with those proposed by King (1969), Eardley (1962), Rodgers (1970) and Hadley and Devine (1974) for eastern North America. This province is bounded on the east by the western extent of the Piedmont Province,* on the west by the Cumberland Plateau, on the south by the Gulf Coastal Plain, and on the north by the northern part of the Valley and Ridge Province.

*In the staff's view, for purposes of nuclear power plant siting, the Blue Ridge Province is considered as part of the Piedmont Province.

and operability of mechanical components, component supports, and piping systems are adequate and in compliance with SRP Section 3.6.2. High energy pipe rupture analysis has been provided for systems both inside and outside the containment. Dynamic analysis has been provided for the mainsteam and feedwater system. No break exclusion zones in the containment penetration region have been postulated in these systems. Therefore, SRP Section 3.6.2 acceptance criteria relative to augmented inservice inspection is not applicable to these systems. A listing of the high energy systems that are considered for pipe rupture analysis has been provided. A summary has been provided of the results of the analyses of these systems to demonstrate that essential systems, components, and supports will not be impaired as a result of high energy pipe break. SRP Section 3.6.2 specifies that exceptions taken by the applicant to the pipe break location and configuration criteria in the section must be identified and the bases clearly justified. As discussed in FSAR Question 010.26, the applicant has used definitions of high energy fluid systems and moderate energy fluid systems that differ from those of BTP ASB 3-1 Appendix A.

Regulatory Guide 1.46 high energy piping definition per Regulatory Guide 1.46 considers two regions of temperature and pressure combinations that TVA has not classified as high energy. One of these pressure temperature combinations has a maximum operating temperature $\geq$ 200°F and pressure < 275 lb/in.^{2}. This condition does not exist in any of the piping analyses and, therefore, would not be of any consequence. The other pressure temperature combination has a maximum operating pressure $\geq$ 275 lb/in.^{2} and temperature < 200°F. This region of operation occurs in piping subject to pump discharge pressure, or subject to leakage past pressure isolation valves in lines connecting the reactor coolant system to low pressure systems. Piping fitting these categories was investigated and it was determined that ~~in all cases insufficient energy was available to produce equipment damage~~ no unacceptable damage would occur.

The assumption of a jet profile expansion half angle 20° as stated in the FSAR is less conservative than the assumption of a half angle not exceeding 10° as stated in SRP Section 3.6.2-III.3. The applicant has, however, provided the following additional justification for the design adequacy of safety-related

In the evaluation of the steel containment buckling, the applicant assumed an axisymmetric shell for simplicity even though equipment hatches and other penetrations introduce discontinuities to the shell at these openings. The applicant stated that the shell was reinforced around the openings and, hence, the use of the axisymmetric model was justified. The staff expressed concern that adequacy of such reinforcement was not well demonstrated. For this reason, the staff has initiated an experimental research program at Los Alamos Scientific Laboratory; the results are expected within a few months. The applicant agreed to verify his code with respect to experimental data when the results from Los Alamos become available.

Los Alamos published its preliminary results in NUREG/CR-2165 dated June 1981. Tests in the report indicated that reinforcement of the opening by the area replacement method of ASME, which is used in the design by the applicant, restored original buckling strength. At the same time, a theoretical calculation in the same report indicated that complete restoration of the buckling strength is unlikely. The staff is continuing research for a further understanding. However, the applicant introduced a load factor for LOCA load to compensate such uncertainty as opening reinforcement effect. The load factor is 1.25 and the LOCA load is the primary load for causing compressive stress in the containment shell. Moreover, the shell buckling design was based on ~~ASME NB-3222 in which an average safety factor of three is provided.~~ criteria which is judged to provide a safety factor of two. The current staff position and Regulatory Guide 1.57 recommend a minimum safety factor of two, which is an additional conservatism in design. Therefore, the staff concludes that there is no likelihood of buckling and the Watts Bar containment design is acceptable.

The staff has requested the applicant to inform it of the progress of confirmatory verification of TVA buckling methodology with the previously mentioned research program. Data needed for the verification, such as shell geometry and loading conditions, are available at this time.

The calculated LOCA pressures were increased by 45 percent for use in containment vessel design, safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an accepatable basis for satisfying the applicable requirements of GDC 2 and 4.

The general requirements with respect to materials, testing, analysis, design, construction, and inspection related to the design and construction of Category I masonry walls conform to the ACI 531-79 code. Conformance with the ACI 531-79 code is acceptable to the staff.

The loads and load combinations used in the analysis and design of Category I walls are in conformance with staff criteria and are, therefore, acceptable.

The criteria used in the analysis and design of Category I masonry walls to account for anticipated loadings that may be imposed on the structures during their service lifetime are in conformance with the staff's criteria for masonry walls, and with codes, standards and specifications acceptable to the staff. The staff concludes that in the event of earthquakes and various postulated accidents, the Category I masonry walls will withstand the specified design condition with out impairment of structural integrity. Conformance with these criteria constitutes an acceptable basis for satisfying, in part, the requirements of GDC 2 and 4.

3.8.4 Foundations

The foundation of the containment is a concrete mat. It was analyzed to determine the effects of the various combinations of loads expected during the life of the plant. Analysis was accomplished by means of selected structural codes taking into account bending moment, shear, and soil pressure for a plate on a elastic foundation. Foundations of the other major structures, such as the fuel building, auxiliary building, and main control areas consist, likewise, of reinforced concrete mats. Foundations were designed in accordance with the American Concrete Institute Standard 318..

The use of these criteria as defined by applicable codes, standards, and specifications, the loads and loading combinations, the design and analysis

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

The safety evaluation of the seismic and dynamic qualification of safety-related equipment consists of (1) a review of the methodology standards and procedures as described in the relevant sections of the FSAR and (2) an onsite audit of selected equipment to examine installation and to verify the completeness and adequacy of the qualification program including documentation. The objective of the onsite audit is to develop the basis for the staff judgement on the adequacy of the applicant's entire equipment qualification program based on the results of a detailed review of a limited number of selected equipment. A satisfactory finding by the staff can only be based on the acceptability of the qualification methodology for the entire list of safety-related equipment as judged by the site audit.

The staff has performed a review of the methodology and procedures for the seismic and dynamic qualification program described in FSAR Sections 3.9.2 and 3.10 for the Watts Bar facility. The applicant's seismic and dynamic qualification criteria are generally based on IEEE 344-1971. Because the currently acceptable standard is the IEEE 344-1975, various approaches are described in the FSAR for equipment supplied by the reactor system vendor and those procured under the applicant's purchase specifications to indicate how the requirements of the IEEE 344-1975 standard can be met. The most significant approach is the conservatism of the input used for the original qualification. The conservatism argument is equipment specific, that is, depending on how conservative the input loading is with respect to the design loading, and whether or not the equipment response is dominated by response from a single natural mode.

The seismic hazard for the Watts Bar site was redefined by the staff during the operating license review. The seismic hazard is now specified in terms of a site-specific spectrum which corresponds to the 84th percentile spectrum shape derived from a number of recorded time histories of ground motion. The site-specific spectrum is applicable for rock supported structures. For structures that are supported by soil at the Watts Bar site, the applicant has performed

4.3.2 Design Description

The FSAR contains the description of the first cycle fuel loading, which consists of three different enrichments and has a first cycle length of approximately
1 year. The enrichment distribution, burnable poison distribution, soluble poison concentration, and higher isotope (actinide) content as a function of core exposure are presented. Values presented for the delayed neutron fraction and prompt neutron lifetime at beginning and end of cycle are consistent with those normally used and are acceptable.

4.3.2.1 Power Distribution

The design bases affecting power distribution are

(1) the peaking factor in the core will not be greater than ~~2.32~~ 2.31 during normal operation at full power in order to meet the initial conditions assumed in the LOCA analysis

(2) under normal conditions (including maximum overpower) the peak fuel power will not produce fuel centerline melting

(3) the core will not operate during normal operation or anticipated operational occurrences, with a power distribution that will cause the departure from nucleate boiling ration to fall below 1.3 (W-3 correlation with modified spacer effect)

The applicant has described how the core will be operated and power distributions monitored to ensure that these limits are met. The core will be operated in the constant axial offset control mode, which has been shown to result in peaking factors less than ~~2.32~~ 2.31 for both constant power and load following operation.

SEE SER PAGE 4-10 (section 4.2.3)

Two types of instrumentation systems are provided to monitor core power distribution measurements. Excore detectors are used to monitor core power, axial

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

5.4.1.1 Pump Flywheel Integrity

GDC 4, "Environmental and Missile Design Bases," requires, in part, that nuclear power plant structures, systems, and components important to safety be protected against the effects of missiles that might result from equipment failures. Because reactor coolant pump flywheels have large masses and rotate at speeds of approximately 1200 rpm during normal operation, a loss of flywheel integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features.

Adequate margins of safety and protection against the potential for damage from flywheel missiles can be achieved by the use of suitable material, adequate design, and inspection. The flywheels have been fabricated from SA-533 Grade B, Class 1 steel. This material has been produced by a process that will minimize flaws and improve fracture toughness, and has been cut, machined, finished, and inspected in accordance with Section III of the ASME and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1.

The reactor coolant pump has been designed for a speed 125 percent that of the normal synchronous speed of the motor (approximately 1500 rpm). The margin against failure for the flywheel is significantly higher because the minimum speed for ductile failure is estimated to be much higher than 125 percent of operating speed for flywheels of the design used at Watts Bar. The ~~lowest~~ predicted ~~design~~ operating temperature is specified to be 110°F. The applicant has stated that the NDTT of the flywheel material is no higher than 10°F and CVN impact energy at 70°F is greater than 50 ft-lbs. These fracture toughness data indicate that the RT_{NDT} of the flywheel material is less than 10°F and that the normal operating temperature of the flywheel will be 100°F above RT_{NDT}. Based on the fracture toughness data and normal operating temperatures of the

SEE FSAR SECTION 5.2.6.3

The design provisions and staff requirements noted above (see Section 3.9.6) satisfy the staff requirements for system isolation as specified in BTP RSB 5-1. Provisions for detecting leakage into the RHR system are discussed in Section 5.2.7 of this report.

The planned preoperational and startup test program provides for demonstrating the operation of the RHR system. Conformance with Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," is discussed in Chapter 14. Any additional testing requirements which result from the staff review of the applicant's compliance with RSB 5-1 are discussed below.

The RHR system is housed within a structure that is designed to withstand tornadoes, floods, and seismic phenomena in accordance with GDC 2, as discussed in Section 3.0. The system seismic requirements (Regulatory Guide 1.29) and quality standards (10 CFR 50.55a and Regulatory Guide 1.26) are discussed in Chapter 3.

The RHR system capability to withstand pipe whip inside containment, as required by GDC 4 and Regulatory Guide 1.46, is discussed in Section 3.0. Protection against piping failures outside of containment in accordance with GDC 4 is discussed in Section 3.0.

As noted above, the RHR system serves both during normal shutdown cooling, and emergency low pressure cooling as part of the emergency core cooling system (ECCS). However, both functions are mutually exclusive, because the RHR system is aligned for ECC except for normal cooldown below reactor coolant conditions of 350°F and 425 psig. When the RHR system is aligned for normal shutdown cooling, the suction paths from the refueling water storage tank (RWST) are closed and the suction paths from the hot legs are opened. When the RHR system is aligned for ECC operation the suction paths from the RWST are kept open, and the ~~two~~ suction path~~s~~ (from ~~separate~~ one hot leg~~s~~) ~~are~~ is isolated ~~each~~ by two motor-operated valves in series. A separate residual heat removal system is provided for each unit, thus satisfying GDC 5.

SEE FSAR Fig. 5.5-4

The annulus vacuum control subsystem is a fan and duct network designed to keep the annulus at a negative pressure (-5.0 in. wg) with respect to atmoshere during normal operation; this will be included as a limiting condition for operation in the facility Technical Specifications. There is no need for this subsystem following an accident. The annulus vacuum control subsystem contains two 100-percent-capacity (~~100~~-cfm) fans.
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The air cleanup subsystem operates following a LOCA to maintain the annulus at a negative pressure with respect to auxiliary building and primary containment atmospheres and to remove particulates and vapors that may contain radioactive nuclides. Annulus pressure control is accomplished by adjusting the fraction of the airstream that is returned to the annulus with that exhausted to the atmosphere. An exhaust rate of ~~1000~~ 100 cfm is required to maintain the annulus pressure at -0.5 in. wg. The air cleanup subsystem consists of two air cleanup units (ACUs), each having 100-percent capacity at ~~200~~ 4000 cfm. The ACUs are started either by a Phase A isolation signal or manually.

SEE FSAR SECTION 6.2.3

The ABGTS operates to filter and exhaust air from all areas of the ABSCE while maintaining a building pressure of 0.25 inch wg during building isolation. The ABGTS uses the same ventilation ductwork that is used for normal operation. Each ABGTS ACU is controlled by one airflow control module (consisting of a modulating damper, differential pressure sensor, and transmitter) which controls the auxiliary building negative pressure by varying the amount of air drawn from the auxiliary building. The ABGTS is composed of two 100-percent-capacity ACUs, with each rated at 9000 cfm.

There are two basic control modes for the ABGTS: (1) one ACU operating and the other on standby (the standby unit comes on automatically upon a low flow signal from the operating fan) and (2) starting both ACUs manually and manually placing one unit in a standby status approximately 30 minutes after startup.

The ABGTS is started automatically upon receipt of one of the following:

(1) Phase A containment isolation signal from either reactor unit

with the isolation of fluid systems which penetrate the containment boundary, including the design and testing requirements for isolation barriers and actuators. The isolation barriers include valves, closed piping systems, and blind flanges.

The containment isolation systems for the Sequoyah and Watts Bar plants are designed to meet the criteria of Appendix A to 10 CFR 50 and SRP Section 6.2.4. The Sequoyah and Watts Bar plants have very similar containment isolation systems; the only differences result from plant system differences. The major differences between the plants are discussed below.

(1) Penetrations X-8A, 8B, 8C and 8D - In addition to the main and auxiliary feedwater lines, Watts Bar has four feedwater bypass lines (one per steam generator). These feedwater bypass lines provide another path for supplying feedwater to the steam generator. These feedwater bypass lines penetrate primary containment (penetrations X-8A through X-8D) and attach to the secondary side of the steam generator. The auxiliary feedwater lines at Watts Bar connect to the feedwater bypass lines. The auxiliary feedwater lines to steam generators 1 and 4 join to the corresponding feedwater bypass lines outside containment, and the auxiliary feedwater lines to steam generators 2 and 3 penetrate containment (X-40A, B) and connect to the respective feedwater bypass line inside containment.

With respect to Sequoyah, feedwater bypass lines are not used and penetrations X-8A through X-8D do not exist. At Sequoyah, the auxiliary feedwater lines connect directly to the main feedwater lines outside containment and downstream of the feedwater isolation valves.

The isolation provisions for the Watts Bar feedwater system have deficiencies similar to those found in the Sequoyah feedwater system (see Section 6.2.4, Sequoyah Safety Evaluation Report (NUREG-0011), Supplement No. 5, June 5, 1981). These deficiencies are discussed below.

GDC 57 requires, for a closed system inside containment (such as the feedwater system), that each line have at least one isolation valve outside

As mentioned above, this situation at Watts Bar is very similar to the situation at Sequoyah, and Sequoyah is being required to make similar modifications to its feedwater system.

When the modifications described above are in place at Watts Bar, the isolation provisions for these lines will be acceptable.

(2) Penetration X-17 - The remote manual isolation valve in this RHR line is outside ~~inside~~ containment at Watts Bar and inside ~~outside~~ containment at Sequoyah. Containment isolation for both systems is composed of a closed system outside containment and check valves inside containment.

Additionally, at both plants, a water seal is placed on the penetration by using the remote manual valves and the RHR pumps. These provisions are acceptable.

(3) Penetrations X-19A and 19B - The containment spray suction lines connect to the RHR sump suction lines outboard of the RHR sump suction isolation valves at Sequoyah. At Watts Bar, however, the containment spray suction lines connect inboard of the RHR sump suction isolation valves. For this reason, each containment spray suction line at Watts Bar has a motor-operated containment isolation valve which is remote manually operated from the main control room. This is acceptable.

(4) Penetrations X-27A, 27B, 27C, and 27D - At Sequoyah, the steam generator sample lines connect to the steam generator blowdown lines outboard of primary containment and inboard of the outer isolation valve. Each of these sample lines is isolated by an air-operated gate valve which receives the Phase A isolation signal.

At Watts Bar, the steam generator sample lines penetrate primary containment (penetrations X-27A-D) and connect directly to the steam generator. Each sample line has two containment isolation valves which receive the containment isolation signal - one valve immediately inside containment and one valve immediately outside containment. These provisions are acceptable.

Included are those penetrations that have resilient seals and expansion bellows, such as airlocks, emergency hatches, refueling tube blind flanges, and electrical penetrations.

The applicant has designed the Watts Bar plant containments so that there is no potential path by which containment leakage could bypass both the emergency gas treatment system and the auxiliary building gas treatment system and reach the environs untreated. The applicant has identified systems for which through-line or penetration leakage could bypass the annulus and be released within the areas of the auxiliary building which are treated by the auxiliary building gas treatment system. The applicant has committed to perform local leak rate tests in accordance with the requirements of Appendix J to 10 CFR 50 and to limit the total potential leakage, which could bypass the emergency gas treatment system and be treated by the auxiliary building gas treatment system, to ~~10~~ 25 percent of the containment design leakage rate (0.25 percent per day by weight of the containment atmosphere) at 15.0 psig.

The proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR 50. Such compliance provides adequate assurance that containment leak-tight integrity can be verified periodically throughout service lifetime on a timely basis to maintain such leakage within the limits of the Technical Specifications.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site.

Based on the foregoing, the staff concludes that the containment leak testing program is acceptable and meets the requirements of GDC 52, 53, and 54; Appendix J to 10 CFR 50; and 10 CFR 100.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The staff safety evaluation review assessed the ferritic materials in the Watts Bar Units 1 and 2 containment system that constitute the containment pressure

Each reactor unit has a separate ECCS; however, portions are housed in a common auxiliary building. The individual components within the building are separated by barriers, and the installation has been reviewed for possible flooding, as discussed in Section 3.4 and in preceding paragraphs. The design constitutes demonstration that the ECCS is not shared by the two units, in compliance with GDC 5.

Instrumentation and Control

The ECCS is initiated automatically on: (1) low pressurizer pressure, (2) high containment pressure, (3) high differential pressure between any two out of three steam generators, or (4) high steam flow coincident with low average temperature or low steam pressure. As noted above, the cold leg accumulator and upper head injection subsystems actuate automatically when the reactor coolant pressure decreases to a value below that at which the ECCS subsystems are maintained. This meets the requirements of GDC 20.

Equipment status indication provided in accordance with the requirements of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," is discussed in Section 7.3. Automatic acuation is provided by redundant signals, whose diversity is noted above. The ECCS may also be manually actuated, monitored, and controlled from the control room, as required by GDC 19. The instrumentation needed to monitor and control and ECCS equipment following a LOCA has been reviewed. The applicant has committed to provide an alarm (low flow) to alert the operator to a postulated degradation of ECCS performance (such as, sump clogging). In addressing procedures that would assist the operator in responding to an ECCS flow degradation, the applicant has stated that emergency operating instructions would include checks for inadequate core cooling and transfers to inadequate core cooling procedures if indicated by the checks. Complemented by appropriate procedures, the instrumentation at Watts Bar can provide sufficient information so that the operator can maintain adequate core cooling following an assumed LOCA. Environmental qualification of electrical components of the emergency core cooling system will be discussed in Section 3.11 upon receipt of TVA's response to NUREG-0588.

(1) at 1000 psig, the operator will maintain pressure and cool down the RCS to 425°F
less then

(2) at 1000 psig and 425°F, the operator will close and lock out the accumulator isolation valves

The applicant has presented an evaluation of a LOCA during shutdown assuming no credit for the accumulators and a worst single failure in the remaining ECCS equipment. This evaluation, based on a LOCA 2-1/2 hours after shutdown, has shown a peak cladding temperature of 1671°F at time of downcomer filling, which is below the typical design basis LOCA analysis temperature (~1775°F) for a corresponding event time. In the scenario, after downcomer filling, the LOCA during shutdown case would experience more rapid core reflooding and lower power in the fuel than the design basis event and thus a smaller subsequent temperature rise. Therefore the calculated peak-cladding temperature for the shutdown case would be lower and the event (LOCA at shutdown) less limiting than the design basis LOCA. Because the calculated results of a LOCA at shutdown are less than those anticipated for the design basis LOCA, the staff finds the proposed accumulator lock out procedure acceptable.

The applicant has also analyzed the consequences of a moderate energy line break in the RHR system immediately after RHR initiation (at about 400 psig). The analyses indicate that an alarm on low level would alert the operator to the event 62 minutes before core uncovery, giving him sufficient time to manually initiate ECCS.

6.3.5 Conclusions

Subject to resolution of the above open items, the staff finds that the applicant meets the requirements of 10 CFR 50.46(b) and GDC 5, 25, 35, and the recommendations in Regulatory Guide 1.1 and BTP RSB 6-1, and, therefore, this item is acceptable.

6.4 <u>Control Room Habitability</u>

Based on its evaluation, the staff finds that the calculated toxic gas and radiological consequences are within the acceptance criteria contained in SRP

Section 6.4 and the design of the control room emergency ventilation system is acceptable for preventing significant toxic gas and radiological exposure to operating personnel in the control room.

The control room design meets GDC 4, "Environmental and Missile Design Bases," with respect to "structures, systems and components shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents...." This conclusion is based on the following:

> Isolation of the control room occurs automatically upon the actuation of a safety injection signal or upon indication of high radioactivity or temperature ~~x~~ or chlorine or smoke concentrations in the outside air supply. The operators have the capability of manually purging smoke or fumes from the control room.

Specifically with respect to GDC-19, the applicant has protected the control room operators against radiation by the use of shielding and by the installation of a filtration system to remove airborne contaminants. After an accident, isolation occurs automatically in response to the accident signal (safety injection) or the high gaseous radioactivity signal for inlet air. This places the control room ventilation system in a pressurization mode such that 200 cfm of pressurized air is supplied through adsorbers while 4000 cfm is recirculated through redundant particulate and carbon filtration components.

In summary, the staff review was performed in accordance with SRP Sections 2.2.1, 2.2.2, 2.2.3, and 6.4, and Regulatory Guides 1.78 and 1.95. The staff finds that the control room habitability systems are adequate to provide safe, habitable conditions within the control room under both normal and accident conditions without personnel receiving radiation exposures in excess of 5 rems whole body, or its equivalent to any part of the body, for the duration of the accident.

As a result, the staff concludes that the control room satisfies the requirements of: NUREG-0737 and (2) GDC-19 and is, therefore, acceptable for a full power license.

6.5.2.2 Auxiliary Building Gas Treatment System (ABGTS)

The function of the auxiliary building gas treatment system (ABGTS) is to collect and process leakage from the fuel handling and waste packaging areas of the auxiliary building during accidents. The system is designed to maintain a slight negative pressure in the auxiliary building following an accident. The AGBTS is a redundant system. Each train has a design capacity of 9,000 cfm of air and includes the following components: heater, prefilter, HEPA filter, carbon adsorber, and fan. The equipment and components are designed to Quality Group C and seismic Category I and are located in a seismic Category I structure.

The staff has reviewed the ABGTS in accordance with the guidelines of Regulatory Guide 1.52. In its evaluation, the staff has assigned the system decontamination efficiencies of 99 percent for elemental and organic iodine and 99 percent for particulates. Based on this evaluation, the staff finds that the ABGTS is designed to control the releases of radioactive materials in gaseous effluents in accordance with applicable regulations following a postulated DBA.

6.5.2.3 Reactor Building Purge Ventilation System (RBPVS)

The function of the reactor building purge ventilation system (RBPVS) is to ensure that activity released inside containment from a refueling accident or fuel-handling accident is treated prior to discharge to the environment. The ESF portions of the RBPVS are redundant. Each train has a design capacity of 14,000 cfm of air and includes the following components: prefilter, HEPA filter, carbon adsorber, and fan. The equipment and components are designed to Quality Group C and seismic Category I and are located in a seismic Category I structure.

The staff has reviewed the RBPVS in accordance with the guidelines of Regulatory Guide 1.52. In its evaluation, the staff has assigned the system decontamination efficiencies of 95 percent for elemental and organic iodine and 99 percent for particulates. Based on this evaluation, the staff finds that the ~~ABGTS~~ RBPVS is designed to control the releases of radioactive materials in gaseous effluents in accordance with applicable regulations following a postulated DBA.

6.5.2.4 Main Control Room Emergency Air Cleanup System

The function of the main control room emergency air cleanup system (MCREACS) is to supply nonradioactive air to the control room after a DBA and to pressurize the control room. This system will permit operating personnel to remain in the control room following a DBA. The MCREACS is a redundant system, with each system having an intake design capacity of 200 cfm of air and recirculating design capacity of 4000 cfm of air. Each system contains the following components: HEPA filter, carbon adsorber, and fan. Cooling coils are also provided for relative humidity control. The equipment and components are designed to Quality Group C and seismic Category I and are located in a seismic Category I structure.

The staff has reviewed the MCREACS in accordance with the guidelines of Regulatory Guide 1.52. In its evaluation, the staff has assigned the system decontamination efficiencies of 95 percent for elemental and organic iodine and 99 percent for particulates. Based on this evaluation, the staff finds that the MCREACS is designed to control a suitable control room environment following a postulated DBA.

6.5.4 Ice Condenser as a Fission Product Removal System

The ice condenser is designed to remove iodine from the postaccident atmosphere passing through the ice beds. Sodium tetraborate is added to the ice to enhance the iodine adsorption characteristics of the ice. Technical Specifications require a minimum ice pH of 8.5 whenever the reactor is critical.

The ice condenser iodine removal effectiveness is a function of the flow rate through the alkaline ice beds and the mole-fractions of air and steam in the flow. Based on the expected conditions following a postulated LOCA, the ice condenser iodine removal effectiveness is expected to be high from the initiation of the accident until melt out of the ice beds has occurred. However, it is difficult to establish assured minimum values for the flow rate and mole-fraction of air prior to startup of the recirculation fans. Therefore, in its model of ice condenser effectiveness, the staff has assumed that the alkaline

high differential pressure between steamlines	2/3 in any steam line
pressurizer low pressure	~~2/3~~ 2/3
high steam flow in 2/4 steamlines coincident with low-low T_{avg} or low steamline pressure	1/2 in any two lines SEE FSAR Table 7.3-1

(2) Containment Spray and Containment Isolation, Phase B

manual (two sets, two switches per set)	1/2 sets
containment pressure high-high	2/4

(3) Containment Isolation, Phase A

manual	1/2
automatic safety injection (see second through fifth items for function (1) above)	

(4) Steamline isolation

manual	1/1 for any loop
high steam flow in 2/4 steamlines coincident with low-low T_{avg} or low steam pressure	1/2 in any two lines
containment pressure high-high	2/4

(5) Feedwater Line Isolation

Safety injection (see function (1) above)	
Steam generator level high-high	2/3

In addition to actuation of various equipment as needed to fulfill the functions listed above, some of the ESFAS signals are also employed in the actuation or realignment of the following systems:

(1) diesel generators
(2) ventilation systems (containment and control room)
(3) essential raw cooling water system

On the basis of its review, the staff finds that the ESFAS conforms to the applicable regulations, guides, BTPs and industry standards and is acceptable, subject to confirmation of the items identified above.

7.4 Systems Required for Safe Shutdown

7.4.1 System Description

The applicant states that securing and maintaining the plant in safe condition can be achieved by appropriate alignment of selected systems that normally serve a variety of operational functions. The capabilities that the selected systems must provide to maintain a safe shutdown are

(1) boration
(2) adequate supply of auxiliary feedwater
(3) residual heat removal

The systems and components that are required to be functioning to achieve and maintain hot shutdown include

(1) auxiliary feedwater pumps
(2) charging and boric acid transfer pumps
(3) essential raw cooling water pumps
(4) component cooling water pumps
(5) ~~instrument~~ control air compressors
(6) reactor containment fan cooler units
(7) control room ventilation unit, including the air inlet dampers
(8) charging flow control valves
(9) letdown orifice isolation valves
(10) auxiliary feedwater control valves
(11) pressurizer heater control
(13) diesel generators

Bar plant. The applicant concluded that no deficiencies existed based on the capability to achieve shutdown conditions using plant procedures. The staff agrees with this conclusion and finds it acceptable.

7.5.4 Conclusions

On the basis of its review, the staff finds that the safety-related display instrumentation conforms to the applicable regulations, Regulatory Guides, Branch Technical Positions, and industry standards, and is acceptable, subject to imposition of the license condition discussed in Section 7.5.2.

7.6 All Other Systems Required for Safety

7.6.1 System Description

The other systems required for safety are

120-V AC and 125-V DC Vital Plant Control Power System

Four electrically and physically independent channels of 120-V ac vital power are provided. Each channel consists of an inverter and a distribution panel. Each channel has access to a normal, a standby, and a maintenancy supply, and is supplied by a separate battery. The four 125-V vital dc power batteries are located in individual rooms to provide physical separation.

There are two motor-operated isolation valves in series in the inlet line from the reactor coolant system to the residual heat removal system. In addition, bypass valves are provided for each of those valves to provide an alternate path in case of a valve failure. The isolation valves are normally closed. Interlocks, provided by pressure monitoring channels, prevent thse valves from opening if the pressure is greater than 425 psig, and close the valves automatically when the pressure increases above ~~600~~ psig.
750

SEE SER SECTION 5.4.3

output signals is provided to the operator, such a failure could remain undetected. Furthermore, even if such a failure would be detected, the system would remain inoperative because no capability to manually arm the system to replace a failed permissive signal from the auctioneer is provided. By letter dated February 12, 1982, the applicant has committed to install switches on the main control board for the operator to manually arm this system. The manual arming will be included in the operating procedure when the reactor coolant temperature is equal to or below the set point and before beginning the filling operation. The system arming will be reset when the system is brought back above the system temperature set point for arming. The staff will define the arming set point in the Technical Specifications, and the instruments for overpressure protection will be under periodic surveillance test. The staff finds this design acceptable subject to its review of the updated drawings and FSAR descriptions to be submitted by the applicant. The staff will report on its final conclusions in a supplement to the SER.

7.6.6 Valve Power Lockout

BTP IPSB-18 addresses power lockout during normal reactor operation for valves whose inadvertent operation could affect plant safety. In the Watts Bar design, this requirement is satisfied by adding a special lockout breaker in the power feed to the valve below the main circuit breaker. It is also required that redundant valve position indication must be provided to the reactor operator regardless of the power lockout. For all such valves, redundancy of position indication is provided by independently powered limit switches mounted on the valve stems, which actuate annunciators on the control board when the valves are not correctly positioned for ESF actuation. The staff finds this design acceptable.

7.6.7 Cold Leg Accumulator Valve Interlocks and Position Indication

A motor-operated isolation valve is provided between each safety injection tank and the reactor coolant (primary) system. The valve receives an open signal ~~opens automatically~~ when either the primary coolant system pressure exceeds the safety injection unblock pressure as specified in the Technical Specification, or when the safety

This valve is already open per Tech Specs

RCS (pressure boundary components) and associated pressurizer and pressure relief systems; the residual heat removal system; ESF systems; ESFs electric power systems; and cooling water systems necessary to operate the above systems.)

(2) Changes in the alignment of any system important to safety be recorded ~~on a system status sheet~~. in the Configuration Control System.

(3) Shift personnel being relieved communicate information on any abnormal plant condition including temporary conditions.

(4) System operability be demonstrated before a system is returned to service.

(5) Approval by the shift supervisor or his/her representative be received prior to the performance of any activitiy on any systems important to safety or any activity that may affect systems important to safety. The shift supervisor or his/her representative is notified when an activity authorized to be performed on a system important to safety is completed or a change occurs in the scope of the activity.

Plant operating instructions require completion of a startup checklist before unit startup. This checklist is used to verify correct alignment of all systems important to safety. In addition, alignment of systems important to safety are reviewed each shift. Any time a critical component is changed from its normal position or condition, ~~a system status sheet is completed and placed in a system status folder~~ it is recorded in the Configuration control system. Panel checklists are reviewed each shift to verify that proper panel alignment exists for all systems important to safety.

It is TVA's opinion that this verification function can be performed adequately by an assistant unit operator (AUO) and that the use of licensed unit operators is not necessary. TVA contends that the AUO has sufficient training and familiarity with plant systems to ensure correct system alignment and that this policy will allow the licensed operator to remain in the control room.

generator. The transmitters are safety-grade and are powered from separate power sources connected to the emergency power system. The staff finds that the automatic initiation of the auxiliary feedwater flow and the flow indication satisfy the Action Plan guidelines, and are, therefore, acceptable.

7.8.3 Proportional Integral Derivative Controller Modification (II.K.3.9)

This NUREG-0737 item calls for implementation of a Westinghouse recommendation to modify the PORV proportional integral derivative controller to prevent derivative pressure action from opening the PORV. Two options are provided.

The applicant has satisfied this requirement by implementing the option of setting the derivative time constant equal to zero.

7.8.4 Proposed Anticipatory Trip Modification (II.K.3.10)

~~The applicant has not proposed a change in the interlock for reactor trip on turbine trip; therefore this item is not applicable to Watts Bar.~~

7.8.5 Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (II.K.3.12)

The licensee has confirmed that the Watts Bar facility has an anticipatory reactor trip on turbine trip, which satisfies this NJREG-0737 item.

This was provided in our Sept. response to item II.K.3.12

assemblies. The auxiliary building protects the new fuel storage facility from the effects of tornadoes, tornado-generated missiles, and flooding (refer to Sections 3.4.1 and 3.5.2 of this SER). The new fuel storage facility is located above the possible maximum flood (PMF) level and therefore will not be affected if the auxiliary building experiences flooding. Thus, the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and the guidance of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied.

The new fuel storage facility is not located in the vicinity of any moderate or high energy lines or rotating machinery. Physical protection for the new fuel from internally generated missiles and the effects of pipe breaks is provided by separation (See Sections 3.5.1.1 and 3.6.1 of this SER), thus satisfying the requirements of GDC 4, "Environmental and Missile Design Bases."

The facility is designed to store unirradiated fuel assemblies. Accidental damage to the fuel would release relatively minor amounts of radioactivity that would be accommodated by the fuel handling building ventilation system. Thus, the requirements of GDC 61, "Fuel Storage and Handling and Radioactivity Control," are satisfied.

The new fuel storage racks are designed to store the fuel assemblies at a center to center spacing of 21 in. This spacing if sufficient to ensure that the effective multiplication factor K_{eff} does not exceed 0.98 with fuel of the highest enrichment in place, assuming possible sources of moderation such as aqueous foam or mist. The value of K_{eff} is less than or equal to 0.95, assuming fully flooded conditions. The design of the racks precludes the insertion of fuel assemblies in other than the prescribed locations. The racks are designed to withstand an uplift force equal to the maximum force which can be exerted by the fuel handling bridge crane; they are also designed to withstand the impact of a fuel assembly dropped from the maximum lift height of the fuel-handling bridge crane without causing an unsafe geometric spacing of the new fuel assemblies. Seismic Category I sectional steel covers are provided over the new fuel vault to protect the racks and fuel from dropped objects. Thus, the requirements of GDC 62, "Prevention of Criticality in Fuel Storage and Handling," are satisfied.

These statement are true, however we dont handle new fuel with the Fuel handling Bridge crane. The fuel handling bridge crane doesn't go over the New Fuel Storage Vaults.

Radiation monitoring equipment for the new fuel storage area is provided and is evaluated in Section 12 of this SER. It satisfies the requirements of GDC 63, "Monitoring Fuel and Waste Storage."

Based on its review, the staff concludes that the new fuel storage facility is in conformance with the requirements of GDC 2, 4, 61, 62, and 63 as they relate to new fuel protection against natural phenomena, missiles, pipe break effects, radiation protection, prevention of criticality, and radiation monitoring, and to the guidelines of Regulatory Guide 1.29 relating to seismic design. It is, therefore, acceptable.

9.1.2 Spent Fuel Storage

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions. The staff has reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water.

The pool liner, rack ~~lattice~~ structure, and fuel storage tubes are of welded stainless steel construction with a neutron absorber sandwiched between the stainless steel sheets. The neutron absorber is marketed under the trade name of Boraflex. Boraflex is composed of boron carbide powder in a rubber-like silicone polymeric matrix. The pool is filled with borated demineralized water.

The stainless steel is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.00×10^{-5} in. in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack ~~lattice~~ structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. The Boraflex poison material is composed of guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.29, "Seismic Design Classification"; 1.102, "Flood Protection for Nuclear Power Plants"; and 1.117, "Tornado Design Classification," are satisfied.

The fuel pool is not located in the vicinity of any high energy lines or rotating machinery. Physical separation is provided to protect the spent fuel from internally generated missiles and the effects of pipe breaks (see Sections 3.5.1.1 and 3.6.2 of this SER). Thus, the requirements of GDC 4, "Environmental and Missile Design Bases," are satisfied.

The high density fuel racks are designed to store the fuel assemblies in an array that limits the effective multiplication factor to 0.95 or less, assuming the array is fully flooded with nonborated water. The racks are designed to preclude the inadvertent placement of a fuel assembly in other than a design storage location. The racks can withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel and can withstand the maximum uplift forces exerted by the ~~fuel handling machine~~, Fuel Handling Bridge Crane.

The applicant has stated that, under normal plant operating conditions, no loads will be carried over the spent fuel pool. Thus, the requirements of GDC 61 and 62 and the guidelines of Regulatory Guide 1.13 concerning fuel storage facility design, are satisfied.

The design of the storage pool includes a leak chase network behind the pool liner welds to detect and collect leakage through the welds, a pool water level monitoring system, and radiation monitoring systems with indications and alarms in the control room. These features satisfy the requirements of GDC 63.

Based on its review, the staff concludes that the spent fuel storage facility is in conformance with the requirements of GDC 2, 4, 61, 62 and 63 as related to protection against natural phenomena, missiles, pipe break effects, radiation protection, prevention of criticality, and monitoring provisions, and the guidelines of Regulatory Guides 1.13 and 1.29 concerning design and protection against seismic events, and is, therefore, acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by stored spent fuel assemblies from the fuel pool water. Additional functions of the system include clarifying and purifying the water in the spent fuel pool, transfer canal, and refueling water storage tanks. The system design incorporates two essential trains of equipment, each train consisting of one fuel pool pump and one heat exchanger, a single separate nonessential purification train with two refueling water purification pumps, two refueling water purification filters, one spent fuel pool filter and one demineralizer, and a single nonessential skimmer train consisting of one pump, filter, and strainer. An essential spare fuel pool pump is provided and is capable of operation in either cooling train.

Chemical analysis downstream of the filter and ion exchanger will be made by batch sampling at 1-week intervals for chemical impurities, twice per month for pH and boron concentration, and once a month for gross beta and gamma radioactivities. The applicant provided the radioactivity and chemical impurity limits to be maintained in the pool water. The ion exchanger will be ~~regenerated~~ replaced when the chemical impurities exceed the specified limits ~~or when the decontamination factor becomes less than 10.~~ A high pressure drop in the filter is the basis for changing the cartridge in the filter. Area radiation monitors are also provided.

The staff determined that the spent fuel pool cleanup system (1) provides the capability and capacity of removing radioactive materials, corrosion products and impurities from the pool water, and thus meets the requirements of GDC 61 as it relates to appropriate filtering systems for fuel storage; (2) is capable of reducing occupational exposure to radiation by removing radioactive products from the pool water, and thus meets the requirements of Section 20.1(c) of 10 CFR 20, as it relates to maintaining radiation exposures ALARA; (3) confines radioactive materials in the pool water into the demineralizer and filters, and thus meets Regulatory Position C.2.f(2) of Regulatory Guide 8.8, "Information Relevant To Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," as it motors, screen wash pump motors, backwashing strainer motors, and motor-operated valves can be powered from emergency sources. The design of the ERCW system ensures that system function is maintained assuming a single active component failure coincident with a loss of offsite power. Thus, the requirements of GDC 5, "Sharing of Structures, Systems and Components," and 44, "Cooling Water," respectively, are met.

SEE ATTACHED

~~The ERCW system is designed to Quality Group B and C and seismic Category I requirements.~~ However, during construction, portions of piping leading to HVAC coolers or chillers which service areas containing essential equipment were not installed to Quality Group B or C requirements. The staff determination of the system's acceptability to these regards is discussed in Section 3.0 of this SER. Components of the system are located in seismic Category I structures that provide protection against tornadoes, tornado-generated missiles, and flooding (see to Sections 3.4.1 and 3.5.2 of this SER). ERCW piping between the IPS and the auxiliary building and between the auxiliary building and the ~~cooling tower basin~~ discharge overflow structure is seismic Category I and is buried to protect the piping from tornado missiles. Pump motors, valve operators, and controls are located above the level of the PMF in the seismic Category I IPS.

In a December 24, 1981, letter, the applicant has proposed the use of cement mortar lining in the carbon steel yard piping portion of this system to inhibit corrosion. The applicant has agreed to address the seismic qualification of the lining, because the ERCW system is seismic Category I. Resolution to this concern will be addressed in a supplement to this SER.

The ERCW pumps and pump motors are housed in a Category I structure that shields against horizontal and vertical tornado missiles. Although the roof of the structure shields the pump motors from vertical missile, the motors are exposed to the effects of the environment. The ERCW pump motors are designed and weatherproofed to operate in such environmental conditions. The staff finds, this acceptable. Pumps and pump motors inside the pumphouse are physically separated from each other to preclude coincident damage to redundant equipment from pipe rupture, equipment failure, and missile generation.

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Safety related portions of the ERCW System are designed to Quality Groups B & C and seismic Category I requirements. Other portions of the systems, such as the station Air Compressor Supply, Yard Discharge Header beyond the Overflow Stand-pipe, etc, are designed to Quality Groups G or H.

15P (OS-9-80) INTEROFFICE MAILING SLIP

The ERCW travelling screens are located in the same Category I structure that houses the ERCW pumps. These screens are protected from the effects of tornado-generated missiles, and are designed to function in an exposed atmospheric environment. The applicant has stated that, through the use of administrative procedures, all travelling screens will be turned out during periods when conditions for surface water freezing water present. Thus, the staff concludes, that adverse weather conditions will not affect the cooling function of the ERCW system. Although the ERCW system operates during normal plant operations, components and controls and periodically inspected and tested. All portions of the system are accessible to permit inservice inspections and testing as required. Thus, the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Basis," 45, "Inspection of Cooling Water Systems," and 46, "Testing of Cooling Water System," are satisfied.

Based on its review, the staff concludes that the ERCW system, with the exception of the cement mortar lined carbon steel piping, meets the requirements of GDC 2, 4, 5, 44, 45, and 46 with respect to protection against natural phenomena, missiles and environmental effects, the sharing of essential systems, heat removal capability, inservice inspection and functional testing, and the guidelines of Regulatory Guide 1.102 with respect to flood protection. The staff's findings regarding the seismic qualification of the cement mortar lined carbon steel piping will be addressed in a supplement to the SER.

The raw cooling water (RCW) system which services the balance of plant cooling and makeup requirements (cooling water to the turbine-generator, auxiliary equipment within the turbine building and nonessential air conditioning equipment within the auxiliary building, makeup to the water treatment plant, a source of makeup to the condenser circulating water system, the primary nonqualified source of cooling water for the ice condenser system, ~~and miscellaneous water requirements such as maintenance and cleaning)~~ serve as a source for filling and maintaining pressurization of the Raw Service Water (RSW) System) is a shared, nonsafety-related system. The RCW system is designed so that no component can adversely affect the function of any safety-related system. The RCW pumps located in the ERCW intake pumping station are completely separated from any safety-related equipment. RCW system piping with seismic Category I structures are seismically qualified to the extent required to ensure that a safe shutdown earthquake in combination with normal operating conditions will not cause flooding, water impingement, or damage as a result of falling onto safety-related equipment, thus conforming to the guidelines of Regulatory Guide 1.29, Position C.2, "Seismic Design Classification."

The applicant has provided a comparison which states that the design of the system is essentially the same as that of Sequoyah. The only significant difference in the RCW systems between the two plants ~~is that Watts Bar maintains the use of seven RCW pumps as compared to five pumps used at Sequoyah. This difference is the result of the lesser capacity of the Watts Bar pumps.~~ SEE Attached Insert The staff has reviewed the information provided in the comparison and concur with the applicant's findings.

Based on its review, the staff concludes that a failure within the nonseismic RCW system will not compromise the ability of safety-related systems to perform their intended functions; is in conformance with the guidelines of Regulatory Guide 1.29, Position C.2; and is, therefore, acceptable.

9.2.2 Component Cooling System (Reactor Auxiliaries Cooling Water System)

The component cooling system (CCS), a safety-related system designed to seismic Category I and Quality Group B and C requirements, provides cooling water to various plant components, and rejects heat to the ERCW system. It serves as an intermediate cooling loop between radioactive or potentially radioactive heat sources and the ERCW. The systems served by the CCS are: RHR, CVC, safety injection system (SIS), waste disposal, spent fuel pool cooling and cleaning, sampling, and containment spray.

The CCS consists of five component cooling system pumps, four thermal barrier booster pumps, three heat exchangers, two surge tanks, and one component cooling system pump seal water collection unit, and associated valves, piping and instrumentation. The component cooling system is a shared system of two trains, each train having the capability to provide the maximum cooling requirement for both units under all design-basis plant conditions. A "spare" train is available and can be remotely aligned to supplement either the Unit 1 or the

Insert for page 9-14

... the two plants are:

1. At Watts Bar the RCW is supplied by seven RCW pumps located in the intake pumping station and at Sequoyah the RCW is supplied by five pumps which receive supply from the condenser cooling water conduit.
2. At Watts Bar the strainers are after the pumps and at Sequoyah they are before the pumps.
3. At Watts Bar the RCW system maintains the pressurization on the RSW system. At Sequoyah the RSW pumps are used to maintain the pressure on the RSW system since the take off is on the RCW pump suction header.
4. At Sequoyah booster pumps are used to maintain the pressure to the equipment within the Auxiliary Building.

guidelines of Regulatory Guide 1.27, Positions C.1, C.3, and C.4 regarding the ability of the UHS to maintain proper system temperature under all modes of operation are met.

The immenseness of the UHS in conjunction with its geographic location precludes the impairment of cooling capability as a result of tornado-missile impact and environmental conditions, and additional protection is not required. Therefore, the UHS is consistent with the requirements of GDC 2 and 4.

The applicant has provided a comparison of the Watts Bar system to that which was designed for Sequoyah. The staff has reviewed the comparison and concurs with the applicant's findings. Thus the staff reaffirms that the conclusions stated in the Sequoyah SER (NUREG-0011) are applicable to Watts Bar.

The staff has further reviewed the system for compliance with the applicable GDC, Regulatory Guides, and BTP, and concludes that the UHS design conforms to the requirements of GDC 2, 4, and 44 with respect to the need for protection from natural phenomena and missiles, and heat removal capability, and the guidelines of Regulatory Guide 1.27 as related to the functional and design requirements of the UHS, and is, therefore, acceptable.

9.2.6 Condensate Storage Facilities

The nonsafety-related (Quality Group D, nonseismic) condensate storage facility stores and supplies treated water for various plant functions; it includes all components and piping associated with the system from the storage tank to the points of connection or interfaces with other systems. The staff review has determined that the system is capable of fulfilling the normal operating requirements of the facility for storage of condensate ~~and primary water~~ with the necessary component redundancy. The system was evaluated and found to have no functions necessary for achieving safe reactor shutdown or for accident prevention or mitigation.

The two outdoor condensate storage tanks (CSTs), which reserve 200,000 gal for each auxiliary feedwater (AFW) system, are not seismic Category I, flood, or

Watts Bar doesn't have a chemical additive tank
SEE FSAR SECTION 9.3.2

boundary; (3) the requirements of GDC 26 to control the rate of reactivity changes by sampling the reactor coolant, the refueling water storage tank, and the boric acid mix tank for boron concentration; (4) the requirements of GDC 41 to monitor variables that can reduce the concentration and quality of fission products released to the environment following postulated accidents by sampling the chemical additive tank for chemical additive concentrations to ensure an adequate supply of chemical for meeting the elemental iodine removal requirements of the containment spray and recirculation solutions following a postulated accident; and (5) the requirements of GDC 64 to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents, by sampling the reactor coolant, the pressurizer tank, the steam generator blowdown, the secondary coolant condensate treatment waste, the sump inside containment, the containment atmosphere, and the gaseous radwaste storage tank for radioactivity.

The staff further determined that the proposed process sampling system meets (1) the requirements of 10 CFR §20.1(c) to keep radiation exposures ALARA and of GDC 60 to control the release of radioactive materials to the environment by purging and draining sample streams back to the system or origin or to an appropriate radwaste treatment system and by providing redundant isolation valves that fail in the closed position; and (2) the requirements of GDC 63 to detect conditions that may result in excessive radiation levels in fuel storage and radioactive waste systems by sampling the spent fuel pool water and the gaseous radwaste storage tank for radioactivity.

The staff also determined that the proposed process sampling system meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2 by designing the sampling lines and components of the process sampling system to conform to the classification of the system to which each sampling line and component is connected, in accordance with the regulatory positions C.1, C.2, and C.3 of Regulatory Guide 1.26, the regulatory positions C.1, C.2, C.3, and C.4 of Regulatory Guide 1.29, and the guidelines of Regulatory Guide 1.97.

Based on its evaluation, the staff concludes that the process sampling system meets the relevant requirements of 10 CFR §20.1(c) and GDC 1, 2, 13, 14, 26, 41, 60, 63, and 64, and is, therefore, acceptable.

review determines a specific procedure is unacceptable, the staff will require that the applicant to make modifications as determined by its generic review. The operating license should be conditioned for the items stated above.

ALARA for Postaccident Sampling

By submittal dated August 21, 1981, TVA committed to procedural changes and plant modifications which will assure that radiation exposures during post-accident sampling will be ALARA. The commitment and interim procedures described by the applicant should enable postaccident sampling that is within the guidelines of GDC 19. These actions meet the staff ALARA position for this item and are acceptable for fuel loading and fuel power operations.

9.3.3 Equipment and Floor Drainage System

The nonsafety-related (Quality Group D, nonseismic Category I) equipment and floor drainage system (EFDS) include all piping from equipment or floor drains to the sump, sump pumps, collector tanks, and piping necessary to carry effluents through separate systems.

we don't have a nonchromated category

The liquid drain system is segregated into three basic systems, each designed to accommodate and collect a particular type of effluent. Effluents are categorized as tritiated, nontritiated and nonchromated, and component cooling system effluents. In the auxiliary building, discharges are collected in the floor and equipment drain sump or tritiated sump, and are then subsequently pumped to their respective drain collector tanks. Turbine building discharge is collected in the sump and sampled before it is discharged to the environment. In the reactor building, most equipment drains are for tritiate-deaerated liquids, which are pumped to the reactor coolant drain tank. All other floor and equipment drains are piped to the containment floor and equipment drain sump. Component cooling system (CCS) effluents are returned to the CCS surge tank. The containment penetration for the containment sump pump discharge line is designed to seismic Category I and Quality Group B requirements, and is located in a seismic Category I, flood-, tornado-, missile-, and environmentally protected structures, thereby satisfying the requirements of GDC 2 and 4.

system is designed to maintain the control building at a positive pressure relative to the outdoors and to the adjoining building at all times.

The control room HVAC system components are provided with two 100-percent capacity units. Each meets the single failure criterion, and automatic switchover occurs if one of the units fail. These components include

(1) main control room air-conditioning system, water chillers, air handling units, and piping

(2) control building pressurizing air supply fans

(3) main control room emergency air cleanup supply fans and filter assemblies

(4) main control room emergency pressurizing air supply fans

The control building outside air intakes are provided with radiation monitors, high temperature detectors chlorine detectors, and smoke detectors. Indicators are provided with the chlorine detectors and radiation monitors. Main control room annunciation is provided for each type of monitor or detector. Isolation of the main control room air intakes occurs automatically upon the actuation of a safety injection signal from either unit or upon indication of high radiation, high temperature, chlorine, or smoke concentrations in the outside air supply stream to the building. Ventilation is then provided by recirculation of air through emergency filters. Main control room isolation may also be accomplished manually at any time by the control room operators. Thus, the design conforms to the requirements of GDC 19, "Control Room," and the guidelines of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

Two minor differences in the system design exist between the Watts Bar and Sequoyah facilities: (1) at Watts Bar the capacity of the air handling units (AHUs) is greater than those at Sequoyah, and (2) chilled water is used as the

(3) 24- and 48-V battery room

(4) communications room

(5) corridor

(6) secondary alarm station

Elevation 729 ft

(1) spreading room

Diesel Generator Building

Elevation 742 ft

(1) pipe gallery and corridor

Intake Pumping Station

Elevation 710 ft

(1) electrical equipment room

Reactor Building

(1) REACTOR COOLANT PUMPS

(2) ANNULAS AREA (divisional interactions)

~~Reactor Coolant Pumps~~

Turbine Building

(1) NUMEROUS AREAS OF BUILDING

~~Numerous Areas of Building~~

Auxiliary Building

Elevation 772 ft

(1) 480-V board rooms

(2) 125-V vital battery rooms

(3) 480-V transformer rooms

(4) mechanical equipment rooms

(5) HEPA filter plenum rooms

Elevation 782 ft

(1) control rod drive equipment rooms

(2) pressure heater transfer rooms

Elevation 757 ft

(1) auxiliary control room

(2) 6.9-kV and 480-V shutdown board rooms

(3) 125-V vital battery rooms

(4) personnel and equipment access

(5) reverse osmosis equipment room

(6) reactor building equipment hatches

(7) reactor building access rooms

(8) emergency gas treatment filters

(9) auxiliary control instrument rooms

Elevation 737 ft

(1) common area

(2) hot instrument shop

(3) heating and vent

(4) ventilation and purge air

(5) GF fuel detector room

(6) auxiliary building gas treatment system filters

Elevation 713 ft

(1) valve gallery

(2) decontamination room

Elevation 729 ft

(1) waste package areas

(2) fuel transfer valve room

Elevation 713 ft

(1) auxiliary building common area

(2) air lock

(3) ~~tritration~~ titration room

(4) radiochemical laboratory

(5) counting room

(6) pipe gallery

(7) volume control tank rooms

(8) sample rooms

(9) pipe gallery

(10) containment purge air exhaust filters

Elevation 692 ft

(1) auxiliary feedwater pump rooms

(2) pipe gallery

(3) charging pump~~s~~ rooms

(4) safety injection pump~~s~~ rooms

(5) cast decontamination collection tank room

(6) spent resin tank room

(7) valve gallery

(8) waste evaporator package room

(9) auxiliary waste evaporator packaging room

(10) corridor

(11) chemical drain tank room

During its site visit, the staff observed that throughout the auxiliary building elevations 713 ft and 737 ft, numerous cable trays, conduit, and equipment obstruct the overhead sprinkler discharge. The staff was concerned that the floor elevation might not be protected against a fire by the overhead sprinkler system. By letter dated August 28, 1981, the applicant agreed to modify design so that the sprinkler heads be either relocated or additional heads installed to enable complete sprinkler coverage on the floor below.

The staff has reviewed the design criteria and bases for the water suppression systems and concludes that these systems, with the additional sprinkler systems and standpipe system modifications to be installed, meet the guidelines of Appendix A to BTP 9.5-1 and are, therefore, acceptable.

<u>Gas Suppression System</u>

A low pressure total flooding carbon dioxide (CO_2) system is provided for the following areas:

(1) standby diesel generator rooms 1A-A, 2A-A, 1B-B, 2B-B

(2) turbine lube oil dispensing room

(3) computer room

(4) paint shop and storage room

(5) auxiliary instrument rooms

(6) spreading room

(7) 480-V board room

audible alarm circuits. A wiring fault in the above circuits results in an audible and visual trouble indication at both the local and control locations. The fire detection system is powered from a single 120-V ac distribution panel. The panel is provided with a manual transfer switch to allow normal or alternate power feed from the Class 1E 480-V ac control and auxiliary building ventilation boards. The ventilation boards are automatically connected to the emergency diesel generators on loss of offsite power.

By letter dated September 9, 1980, the applicant stated that the fire alarm system is electrically supervised for ground and open wiring faults in the detection, power supply, alarm, and MUX data transmission circuits. Supervision is Class A in the detection and data transmission circuits and Class B in local audible alarm circuits. A wiring fault in these circuits results in an audible and visual trouble indication at both the local and control locations. In a February 18, 1982 meeting, the applicant committed to provide supervision of valve actuation circuits (for preaction sprinkler system and deluge systems) so that a single break or ground fault condition will initiate a visual and audible trouble indication in the control room. Pending confirmation of this commitment, the staff finds this supervision complies with the requirements of NFPA Standards 13 and 72 D, and is, therefore, acceptable.

The fire detection systems have been installed or will be installed according to NFPA Standard 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protection Signalling Systems."

The staff has reviewed the fire detection systems to ensure that fire detectors are adequate to provide detection and alarm of fires that could occur. It has also reviewed the fire detection system's design criteria to ensure that it conforms to the applicable sections of NFPA 72D. Once the applicant has confirmed his commitment, the staff will report on the supervision of the control function circuits in a subsequent safety evaluation report.

The TVA position is stated in a letter from L.M. Mills to E. Adensam dated March 1, 1982, item #4.

Fire Doors, Dampers, and Fire Barrier Penetrations

The staff has also reviewed the placement of the fire doors to ensure that fire doors of proper fire rating have been provided.

Doors separating the control building from the turbine building are normally closed. Heavy equipment doors are locked and operated by card readers. Operation of these doors is alarmed in the main control room. Strict administrative procedures will be used to ensure that the doors are not left open or propped open during maintenance or plant operation.

In the diesel generator building at elevation 742 ft, the lube oil storage room is enclosed in 3-hour-fire-rated construction; however, the 3-hour fire doors are in the open position and close only upon melting of the thermal link above the door or by discharge of the CO_2 system for the room. The staff is concerned that fire may spread to adjacent areas before the fire doors are closed, or that some obstruction in the doorway may keep the fire door from closing completely. To meet the guidelines of Section 6-6.3.2 of NFPA 101 (1976) as well as of Section 4-4.1.2 of NFPA 30, "Flammable and Combustible Liquids Code," these doors should be self-closing. In addition to the guidelines of NFPA Codes, Section N of Appendix R to 10 CFR 50 requires the doors to be self-closing. The staff require the applicant to keep these doors closed and locked or alarmed, with alarm and annunciation in the control room.

At elevation 760 ft of the diesel generator building, each switchgear room is separated from the other by 3-hour-fire-rated construction. The doors separating these rooms are not labeled. By letter dated August 28, 1981, the applicant has agreed to verify that these doors are 3-hour fire doors (including their frames and hardware) that have been tested and approved by a nationally recognized laboratory or to replace them with 3-hour fire doors.

Documentation verifying 3-hour fire rating was provide in a letter from L.M. Mills to E. Adensam dated Nov. 10, 1981.

Fire doors in most of the fire-cell and fire-area boundaries are UL labeled. The special purpose doors in the auxiliary building, such as flood doors and pressure doors, are not UL labeled. These doors are designed to ASME Standard and are of heavy-welded-steel construction. The applicant has evaluated these doors and determined that they will provide a fire rating commensurate to the fire loading in the areas or cells they separate. The security doors in the main control room are not UL labeled. They are made of bullet-resistant, heavy-gauge steel, and the door manufacturer has certified that the doors are equivalent to UL-tested fire doors rated for 3 hours. The applicant considers these non-tested doors equivalent. Similar doors were found acceptable for the Sequoyah nuclear plant. Therefore, the staff finds these acceptable.

Penetrations, including electrical penetration seals through rated barriers, are sealed to provide fire resistance equivalent to the barrier itself.

Ventilation penetrations through fire-rated barriers are protected by standard fire doors dampers. All of the fire doors/dampers are UL listed.

Some of the 1½-hour-fire-rated dampers are not installed according to the manufacturer's instruction or NFPA 80, "Fire Doors and Windows." By letter dated August 28, 1981, the applicant agreed to modify these installations in the same way they were modified for the Sequoyah nuclear plant. A Letter from L.M. Mills to E. Adensam dated Nov. 10, 1981, submitted revised Fig. 12.6(11)-4 indicating pyrocrete application to the damper sleeves.

By letter dated March 1, 1982, the applicant has stated and provided supporting justification that the Watts Bar electrical fire barrier penetrations meet the requirements of ASTM E-119. The staff will compare the applicant's test methods to the requirements of ASTM E-119 to confirm this statement, and will report its findings in a supplement to the SER.

The staff has reviewed the fire doors and dampers. The applicant has verbally agreed to modify the design and installation to be identical with those accepted at the Sequoyah plant as meeting the guidelines of Appendix A to BTP ASB 9.5-1. The doors and dampers are, therefore, acceptable.

9.5.1.4 Emergency Lighting

By letter dated August 28, 1981, the applicant stated that the 1½-hour fixed emergency seal-beam lighting units are modified throughout the plant with small capacity bulbs, so that a lighting duration of approximately 8 hours is provided. The staff finds this arrangement unacceptable because the amount of light in a given area is reduced below acceptable levels.

The staff will require the applicant to provide adequate illumination of not less than 1.0 ft-candle measured at the floor at all points of equipment operation needed for shutdown, as well as at all points on the floor, including angles and intersections of corridors, passageways, stairways to valves, and for routes to and from all of the areas. The staff will report on this item in a supplement to this SER.

9.5.1.5 Fire Protection for Specific Areas

Component Cooling Water Pump Area (El 713)

On elevation 713 ft of the auxiliary building, all five (two from each unit and one swing) component cooling water (CCW) pumps are located together. Adjacent to these safety-related pumps are the two motor-driven auxiliary feedwater pumps (both trains) of Unit 1, which are also safety related. Both Unit 2 auxiliary feedwater pumps are located approximately 125 ft away, down the corridor. Power-operated control valves for the CCW pumps are located immediately above the CCW pumps on an open-grating mezzanine. Various safety-related cable trays are also located in the area. A preaction sprinkler system is proposed for the ceiling level only; this would not offer adequate protection against an exposure fire because of the many obstructions between the ceiling-level sprinkler and the floor below.

At the request of the staff, the applicant verbally has agreed to provide

(1) ~~automatic sprinkler coverage under the pipe-break barrier for the Units 1 and 2 motor-driven auxiliary feedwater pumps~~ No pipe break barrier is required at WBN due to different type motor driven AFW pumps than at SNP.

(2) automatic sprinkler coverage under the mezzanine for all five CCW pumps

(3) a 1-hour-fire-rated barrier separating the train A CCW pump from the train B CCW pumps, so that the barrier will extend approximately 3 ft above the highest point of each pump

Fire Protection Inside Containment

The major fire hazard within the containment is the reactor coolant pump (RCP) lube oil system. To prevent a fire as a result of oil leakage, the applicant has provided an oil collection system for each RCP. The oil collection system includes enclosing the RCP oil lift pump and providing an oil collection basin at the access platform elevation of each pump to collect and drain away any combustible liquid and/or suppression system discharge. Any discharge is drained from the collection basin into the containment floor and equipment drain sump located inside primary containment. Each RCP is provided with a heat collection hood which also will act as a heat collector to reduce the response time of the thermal detectors and the thermal-actuated-water-spray nozzles installed inside the housing. The fixed automatic water spray system is designed in accordance with NFPA 15. The oil collection system meets the requirements of Section III.O of Appendix R to 10 CFR 50 and is, therefore, acceptable.

~~An automatic fixed-water-spray system has been provided for the charcoal HEPA filters in the lower containment air cleaning units. The water spray system is designed according to NFPA 15.~~ Containment ACU deleted from WBN design

Areas of divisional interaction within the annulus area will be protected by an automatic-fixed-water-spray system designed according to NFPA 15 with the exception that conventional sprinkler heads will be used. In addition, all exposed cables within this area will be coated with a flame retardant. The staff also will require the applicant to provide a 1-hour-fire-rated barrier for all divisional interactions involving redundant functions of equipment or circuits necessary to achieve safe shutdown in the event of a fire.

A standpipe and hose system, designed according to NFPA 14, has been provided to complement the fixed-water-suppression system in the reactor building. The standpipe system within its containment will be normally dry and arranged to admit water to the system through manual operation of remote control devices located at each hose station.

The fire detection system is designed according to NFPA 72D with Class A supervision. ~~Thermal detectors are provided for the charcoal filters and HEPA filters~~. Thermal-rate-compensated and flame detectors are provided for the RCP motors. Smoke, photoelectric, and/or thermal-rate-compensated detectors are provided for divisional cable interaction areas.

Photoelectric smoke-duct detectors are provided for each lower containment cooling unit and each upper compartment cooling unit. In addition, photoelectric smoke-duct detectors are provided for the exhaust ducts serving the containment-purge-air-exhaust systems and the emergency gas treatment system. In the annulus area, heat and smoke collectors are provided for fire detection so that a quick response can be obtained.

The staff has reviewed the applicant's fire hazard analysis and fire protection provided for the area inside containment. The staff concludes that appropriate fire protection is provided for this area and meets the guidelines of Appendix A to BTP ASB 9.5-1 Appendix R to 10 CFR 50. It is, therefore, acceptable.

Control Room

The control room complex is separated from other areas of the plant by 3-hour-rated fire-resistant construction. Doors between the control room and turbine building and auxiliary building are 3-hour-rated-fire doors. All other doors in the complex are 1½-hour fire rated. Three-hour fire dampers are installed in ducts that penetrate the wall from the control building to the auxiliary building.

Fire detection, fire extinguishers, standpipe hose station, automatic preaction sprinkler system (at selected locations), emergency lighting and a manual smoke-venting system are provided for the control room complex.

Both carpeting and a dropped suspended ceiling with a vinyl dust cover are to be installed in the control room. By letter dated April 30, 1981, the applicant stated that the dropped ceiling panels in the main control room will be a UL-listed material having a flame spread classification of 15 and a smoke density

In the view of the staff, plant operation following such procedures would provide assurance that licensees would devote proper attention to controlling secondary water chemistry, while also providing the needed flexibility to allow them to deal effectively with an offnormal condition that might arise.

Discussion

The applicant has stated that the Watts Bar and Sequoyah Plant secondary water chemistry monitoring and control program and implementing systems are ~~identical~~ similar. The Sequoyah secondary water chemistry program has been previously evaluated by the staff and found acceptable. The staff evaluation of the Sequoyah secondary water chemistry program confirmed that the program addressed the six program criteria of the staff position and is based on the steam generator water chemistry program recommended by the NSSS vendor (Westinghouse).

TVA intends to follow the EPRI program at WBN

Conclusion

On the basis of the above evaluation, the staff concludes that the proposed secondary water chemistry monitoring and control program for Watts Bar meets (1) the requirements of GDC 14 insofar as secondary water chemistry control ensures primary boundary material integrity; (2) Acceptance Criterion 3 of SRP Section 5.4.2.1, Revision 1; (3) Positions 2 and 3 of BTP MTEB 5-3, Revision 1; and (4) the program criteria in the staff's position. Therefore, it is acceptable. The staff will condition the operating license to require that the proposed secondary water chemistry monitoring and control program be carried out.

10.4 Other Features

10.4.1 Main Condenser

The main condenser is designed to function as a heat sink for the turbine exhaust steam, turbine bypass steam, and other turbine cycle flows, and to receive and

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

The process and effluent radiological monitoring systems are designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs.

Table 11.2 provides the proposed locations of continuous monitors. Monitors on certain effluent release lines automatically terminate discharges should radiation levels exceed a predetermined value. Systems which are not amenable to continuous monitoring or for which detailed isotopic analyses are required are periodically sampled and analyzed in the plant laboratory.

The staff has reviewed the locations and types of effluent and process monitoring provided. Based on the plant design and on continuous monitoring locations and intermittent sampling locations, the staff has concluded that all normal and potential release pathways are monitored. The staff has also determined that the sampling and monitoring provisions are adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which could affect radioactivity releases. On this basis the staff considers the monitoring and sampling provisions to meet the requirements of GDC 60, 63, and 64 and the guidelines of Regulatory Guide 1.21, and, therefore, are acceptable.

The Technical Specifications for the process effluent radiological monitoring systems, instrumentation, controls, and the sampling and analysis programs for Watts Bar will ~~be the same as~~ similar to those at Sequoyah.

11.6 Evaluation Findings

In its evaluation, the staff has calculated releases of radioactive materials in liquid and gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant.

Table 11.2 Process and effluent radiation monitoring system

Stream monitored*	Number	Monitor classification	Monitor sensitivity µC/ml
LIQUID			
Component Cooling Water	3/plant	Gamma-Scintillation	3×10^{-7} (Co-60)
Service Water	2/plant	Gamma-Scintillation	3×10^{-7} (Co-60)
Stream Generator Blowdown (Process)	2/plant	Gamma-Scintillation	3×10^{-7} (Co-60)
Waste Disposal System**	1/plant	Gamma-Scintillation	3×10^{-7} (Co-60)
Steam Generator Blowdown Liquid Discharge**	2/plant	Gamma-Scintillation	3×10^{-7} (Co-60)
GASEOUS			
Auxiliary Building Exhaust Vent**			
Particulate	1/plant	Beta-Scintillation	5.7×10^{-11} (Co-60)
I-131	1/plant	Gamma-Scintillation	7.4×10^{-10} (I-131)
Noble Gas	1/plant	Beta-Scintillation	4.1×10^{-7} (Kr-85)
Gaseous Waste Process System Discharge**	1/plant	Beta-Scintillation	1×10^{-6} (Kr-85)
Condenser Vacuum Pump, Low Range Exhaust	2/plant	Beta-Scintillation	1.4×10^{-7} (Kr-85)
Condenser Vaccum Pump, High Range Exhaust	2/plant	Beta-Scintillation	1×10^{-3} (Gross)
Containment Purge Exhaust**	4/plant	Beta-Scintillation	5.7×10^{-4} (Gross)
Shield Building Vent Exhaust	2/plant	Beta-Scintillation	5.7×10^{-11} (Co-60)

*All monitors have instrument malfunction and high radiation visual and audible alarms in the main control room.
**These monitors terminate the release when the radiation level exceeds a predetermined level.

These values were changed in FSAR Amendment 42 - SEE Table 11.4-1

Selected supervisors, engineers, technicians, and craftsment are provided training and specialized courses to satisfy the applicable requirements of their particular positions.

General Employee Training

All persons regularly employed at the Watts Bar nuclear plant will be trained in the following areas commensurate with their job duties.

(1) general description of plant and facilities
(2) job-related procedures and instructions
(3) radiological health and safety program
(4) station emergency plans
(5) industrial safety program
(6) fire protection program
(7) security program
(8) quality assurance program

Specialized Courses

Some of the specialized courses include

(1) IBM Selectric input/output devices course
(2) PWR station nuclear engineering STA Training will replace this Training
(3) PWR instrumentation and control
(4) basic nuclear instrumentation program
(5) computer programming and review
(6) Prodac computer maintenance
(7) electro-hydraulic control system course
(8) shift technical advisor training
(9) mitigating core damage training
(10) fire protection training

Guidance on Procedures for Verifying Correct Performance of Operating Activities

The applicant has stated in the FSAR that the current plant administrative procedures require that:

(1) The alignment of all systems and components important to safety be verified prior to unit startup

(2) Changes in the alignment of any system important to safety be recorded ~~on a system status sheet~~ in the configuration control program.

(3) Shift personnel being relieved communicate information on any abnormal plant condition including temporary conditions

(4) System operability be demonstrated before a system is returned to service

(5) Approval by the shift supervisor or his representative be received before the performance of any activity on any systems important to safety or any activity that may affect systems important to safety. The shift supervisor or his representative is notified when any activity authorized to be performed on a system important to safety is completed or a change occurs in the scope of the activity.

Plant operating instructions require completion of a startup checklist before unit startup. This checklist is used to verify correct alignment of all systems important to safety. In addition, alignment of systems important to safety is reviewed each shift. Any time a critical component is changed from its normal position or condition, ~~a system status sheet is completed and placed in a system status folder.~~ an entry is made in the configuration control system Panel checklists are reviewed each shift to verify proper panel alignment exists for all systems important to safety.

It is the applicant's opinion that this verification function can be performed adequately by an assistant unit operator (AUO) and that the use of licensed unit operators is not necessary. The AUO has sufficient training and familiarity with plant systems to ensure correct system alignment, and this policy