TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

March 5, 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

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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. Enclosed for your information are comments resulting from our review of the draft SER. 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. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 5th day of March 1982

Bryant M. Lawery
Notary Public
My Commission Expires 4/4/82

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

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number thereafter. They are not included as reference lists or Appendix B. Appendix C is a discussion of how various ACRS (Advisory Committee on Reactor Safeguards) generic concerns relate to the application. Appendix D is an evaluation of the applicant's preliminary control room assessment. Appendix F is a list of abbreviations used in this report. Appendix G discusses the guidelines for demonstration of operability of purge and vent valves. Appendix H is a list of principal contributors.

As part of the NRC review of the Watts Bar facility for compliance with the Commission's regulations, the staff asked the applicant to verify that the Watts Bar facility meets the pertinent regulatory requirements in 10 CFR Parts 20, 50, and 100. The applicant's response to this request, which was submitted on 12/28/81, stated that the Watts Bar facility is in compliance with all applicable regulations and requirements. Subject to the applicant's adoption of the additional requirements imposed by the staff in this Safety Evaluation Report, and the exemptions granted, the staff concurs that the facility is in compliance with these regulations and requirements.

In accordance with the provisions of the National Environmental Policy Act (NEPA) of 1969, Draft and Final Environmental Statements that set forth the considerations related to the proposed construction and operation of the Watts Bar Nuclear Plant were prepared by the staff. The Draft Environmental Statement was issued on June 5, 1978 and the Final Environmental Statement on December 29, 1978.

The review and evaluation of the Watts Bar facility for an OL is only one stage in the continuing review by the staff of the design, construction, and operating features of the facility. The proposed design of the facility was reviewed as part of the CP review. Construction of the facility has been monitored in accordance with the inspection program of the staff. During the OL review stage, the NRC staff reviewed the final design to determine that the Commission's safety requirements have been met. If an OL is granted, the Watts Bar facility must be operated in accordance with the terms of the OL and the Commission's regulations and will be subject to the continuing inspection program of the staff.

1.2 General Plant Description

Unit 1 and 2 of the Watts Bar Nuclear Plant each utilize a nuclear steam supply system (NSSS) incorporating a pressurized water reactor (PWR) rated at 3411 MWt and a 4-loop reactor coolant system. In each of the identical units, the reactor core is composed of fuel rods made of slightly enriched uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs that are grouped and supported into assemblies. The mechanical control rods consist of clusters of stainless steel-clad boron carbide (B_4C) ~~silver-indium-cadmium alloy~~ absorber rods that are inserted into Zircaloy guide tubes located within the fuel assemblies. The core fuel is loaded in three regions, each utilizing fuel of a different enrichment of U-235, the new fuel is introduced into the outer region, moved inward at successive refuelings, and removed from the inner region to spent fuel storage.

Water will serve as both the moderator and the coolant, it will be circulated through the reactor vessel and core by four vertical, single-stage centrifugal pumps, one located in the cold leg of each loop. The coolant water heated by the reactor will be circulated through the four steam generators where heat will be transferred to the secondary system to produce saturated steam, and then be returned to the pumps to repeat the cycle.

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 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 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.

Regulatory Guide 1.76; however, the differences are not considered significant in the determination of an acceptable design basis tornado for missile generation. Therefore, the requirement of GDC 4 is met.

2.3.2 Local Meteorology

Climatological data from Chattanooga, Knoxville, Decatur, and Watts Bar Dam, in addition to available onsite data, have been used to assess the local meteorological characteristics of the site.

Extreme maximum temperatures of 108°F and 106°F and extreme minimum temperatures of -20°F and -10°F have been observed at Decatur and Chattanooga, respectively. Mean monthly precipitation at Watts Bar Dam varies from about 5.6 in. in March to about 2.9 in. in October. The maximum monthly precipitation reported at Watts Bar Dam was 14.8 in., while the minimum monthly precipitation reported is 0.0 in. The maximum 24-hr rainfall reported at Chattanooga was about 7.6 in., while at Watts Bar Dam the maximum 24-hr rainfall reported was about 5.3 in. Annual average snowfall in the area is about 9 in., although about 18 in. of snow has been reported in 24 hours in Knoxville. The site area is generally humid, particularly in the summer. Heavy fog can be expected on about 35 days annually.

The plant is located along a stretch of the Tennessee River with a valley orientation generally southwest to northeast. The terrain rises rapidly away from the plant site in most directions, with increases on the order of several hundred feet within 2 mi of the plant. The local airflow pattern is bimodal, reflecting channelling up and down the river valley. Wind data from the 30-ft level of the onsite meteorological tower for the period July 1, 1973 through June 30, 1975 indicate winds from the southwest and south-southwest together occur about 25 percent of the time while winds from the north-northeast and northeast occur about 16 percent of the time. Winds from the east-southeast and southeast occur least frequently, totalling only about 3 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.

A similar pattern 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.

The median wind speed at the 3-ft level is 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 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 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.

SEE "Inversion" column of table 2.3-42 of the FSAR

Inversions predominate at the Watts Bar site, occurring almost 60 percent of the time. Moderately stable (Pasquill type "F") and extremely stable (Pasquill type "G") conditions occur about 16 and 9 percent, respectively.

As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of this plant. Based on this review, the staff concludes that the applicant has identified and considered appropriate local meteorological conditions in the design and siting of this plant, and, therefore, meets the requirements of 10 CFR Part 100.10 and GDC 2.

2.3.3 Onsite Meteorological Measurements Program

The onsite meteorological measurements program at the Watts Bar site was initiated in June 1971 with the installation of a 120-ft temporary tower used to provide data in support of the Construction permit application. A permanent 273-ft meteorological tower was installed in May 1973 in a location about 2400 ft west-southwest of the Unit 1 reactor building. Two 450-ft high natural draft cooling towers are located about 3000 ft northeast of the meteorology tower. Wind speed and direction are measured at the 30-, 130-, and 270-ft levels of the permanent tower; dry bulb temperature is measured at the 3, 30-, 130-, and 270-ft levels; dewpoint temperature is measured at the 3 and 30-ft levels; and solar radiation, atmospheric pressure, and precipitation are measured at about 3 ft above the ground. Atmospheric stability is defined by the vertical temperature gradient determined between the 30-ft and 130-ft levels and between the 30- and 270-ft levels. Data are recorded digitally. Apparently, prior to January 31, 1975, temperature values were recorded only once per hour. After that time, temperature values were recorded once per minute and hourly averages were determined based on 60 values. The representativeness of hourly averages based on just one value is suspect. The impact of this averaging technique on the determination of vertical temperature gradient is unknown; however, the stability distribution for the period July 1973 to June 1975 appears reasonable. The applicant has also examined the effects of the large natural draft cooling towers on wind speed and wind direction measurements and concluded that no effects are evident to date, although the cooling towers are not in operation.

SEE Amendment 44 of FSAR

The applicant has provided 2 years (July 1, 1973 through June 30, 1975) of meteorological data on magnetic tape. A joint frequency distribution for this period was compiled using wind speed and direction from the 30-ft level, and using vertical temperature gradient between the 30- and 130-ft levels as the indicator of atmospheric stability. The joint data recovery for wind speed, wind direction, and atmospheric stability was 93 percent.

The onsite meteorological measurement system now conforms to the guidance of Regulatory Guide 1.23. The program in operation for the period July 1, 1973 to June 30, 1975 appears to have provided adequate data to represent onsite meteorological conditions used for evaluations required by 10 CFR Part 100.10 and 10 CFR Part 50, Appendix I. The onsite data appear to provide a reasonable basis for making estimates of atmospheric dispersion conditions for assessing consequences of design basis accident and routine releases from the plant.

To address the meteorological requirements for emergency preparedness planning outlined in 10 CFR Part 50.47 and Appendix E to 10 CFR Part 50, the applicant will be required to upgrade the operational meteorological measurements program to meet the criteria in NUREG-0654, Appendix 2, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The upgrades must be in accordance with the schedule of NUREG-0737, III.A.2, "Clarification of TMI Action Plan Requirements." Although the effects of the cooling towers on measurements made on the meteorological tower appear to be negligible, the representativeness of the meteorological measurements with respect to airflow from likely release points towards and around the cooling towers may be questionable. Distortions in airflow trajectories because of the presence of the cooling towers should be considered in the development of the upgraded meteorological measurements program and modeling capability.

SEE Attached sheet

2.3.4 Short-Term (Accident) Diffusion Estimates

Short-term (less than 30 days) accidental releases from buildings and vents were evaluated using the direction-dependent atmospheric dispersion model described in Regulatory Guide 1.145, with consideration of increased lateral dispersion during stable conditions accompanied by low wind speeds. Two years (July 1, 1973 to June 30, 1975) of onsite data were used for this evaluation. Wind speed and wind direction were measured at the 30-ft level and atmospheric stability was defined by the vertical temperature gradient determined between the 30- and 130-ft levels. A ground-level release with a building wake factor, cA, of 850 m^2 was assumed. The χ/Q values for appropriate time periods at the outer boundary of the low population zone (4828m) are:

The relative concentration (χ/Q) for the 0-2 hour time period was determined to be 3.6×10^{-4} sec/m^3 at a distance of 3600 ft in both the southeast and south-southeast sectors from the reactor buildings.

TVA concurs that the cooling towers will affect airflow from possible gaseous release points near the cooling towers. Such releases should experience additional mixing in the wake cavity of the cooling towers, but should be transported in the mean wind direction. Therefore, the meteorological measurements should indicate a representative transport direction and, due to the additional mixing, the modeling results should be conservative. TVA does not beleave that the cooling tower distortion warrants additional consideration in the meteorological measurements program upgrade.

2.4.6 Groundwater

The materials immediately beneath the site are older terrace deposits and alluviam that are poor waterbearing strata. At deeper levels are discontinuous beds of Conasauga shale (more properly a combination of about 84 percent shale and 16 percent limestone), Chickamauga Limestone, and Knox Dolomite. 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 flows generally toward the river.

TVA has developed three wells, 2.5 mi northwest of the plant, that draw 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.

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 backfill constitute the primary features of the system. TVA has claimed the system is not safety related under the provision of Branch Technical Position (BTP) HMB/GSB1, SRP 2.4.13. TVA states that credit for the system's ability to lower groundwater levels has not been taken in the design of safety-related structures. However, it appears to the staff that the applicant has taken credit for the dewatering systems' ability to keep groundwater levels at or below elevation 710 ft msl (see Section 2.4.8). The staff also concludes that while the underdrain may not be important to the design of safety-related structures, the system provides a pathway for accidental releases of radioactivity in liquid form.

There is no underdrain system at SQN. WBN has an underdrain system to maintain an equivalent groundwater elevation as that designed for at SQN.

ERCW Pipeline

The ERCW piping runs from the intake structure on Lake Chickamauga to the powerblock area and furnishes water for cooling the reactor during emergency condition. There is additional fire protection piping along the same alignment. The ERCW piping is approximately ~~600~~ 5000 ft long. A major portion of this piping has been founded in alluvial terrace deposits of loose to medium density. The natural ground in the vicinity of the pipeline generally slopes downward on one or both sides of the finished grade above the pipeline. The soil profile along the seismic Category I ERCW pipeline was developed by the applicant from the results of soil borings along the alignment of the pipeline. The borings are approximately 100 ft apart. The applicant has presented the blow count and soil classification data along with the results of anisotropically consolidated stress-controlled dynamic triaxial test data on soils susceptible to liquefaction along this route. The staff has evaluated the results of the exploration and field and laboratory test data for the soild beneath the ERCW pipelines. The available data indicate the presence of very loose to loose deposits of alluvial silty sands and sandy silts as evidenced by blow counts ranging from 1 to 12 for thicknesses ranging from 5 to 12 ft. Organic soil samples, indicating the possible existence of unsuitable materials, were also recovered by the applicant during the field investigations.

The staff reviewed the applicant's submittals and has independently assessed the liquefaction potential of the very loose to loose alluvial, granular soils underneath the ERCW pipeline and has concluded that these soils would be susceptible to liquefaction for a SSE of 0.18 g horizontal ground motion.

The staff concerns and the bases for these concerns have been provided to the applicant. Unless some other acceptable solution is developed, the staff will require the applicant to take state-of-the-art, remedial field measures to provide stable, safe and acceptable foundation soil conditions underneath the ERCW pipelines. The staff will review the applicant's submittal and provide, in a supplement to this SER, an evaluation of the liquefaction potential of soils underneath the seismic Category I ERCW pipeline.

this report.) The applicant has committed to provide an evaluation of possible tornado missile impact and its consequences and of proposed additional missile protection. Resolution of these items will be addressed in a supplement to this report.

missile protection slabs or

Essential piping from the outdoor and ERCW intake structure is protected from missiles thoughout its length by seismic Category I pipe tunnels. The UHS, which is the Tennessee River waterway complex, does not require protection from externally generated missiles. (The UHS and the ERCW systems are discussed in Sections 9.2.5 and 9.2.1 of this report.) Thus, the requirements of GDC 4 with respect to missile protection and the specific guidance of Regulatory Guide 1.13, 1.27, and 1.117 concerning tornado-missile protection for safety-related SSC including stored fuel and UHS are met.

Based on its review, the staff concludes that those safety-related structures systems and components identified by the applicant as requiring protection from externally generated missiles with the exception of the 480-V shutdown transformers, 125-V vital batteries, and the diesel generator auxiliary control boards and exhaust stacks, conform to the requirements of GDC 4 with respect to missile and environmental effects and the guidelines of Regulatory Guide 1.27 concerning the protection of safety-related plant features from tornado missiles, and are, therefore, acceptable. Resolution to the concerns regarding the shutdown transformers vital batteries, control boards and exhaust stacks will be addressed in a supplement to this report.

3.5.3 Barrier Design Procedures

The analysis of structures, shields, and barriers to determine the effects of missile impact was accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact was investigated. This was accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles were prevented by fixing the target thickness well above that determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally square root sum of squares

[redacted]s was discussed at the May 28, 1981 meeting as discussion item B6.

The system and subsystem analyses were performed by the applicant on an elastic basis. Model response spectrum multidegree of freedom and time history methods form the bases for the analyses of all major seismic Category I structures, systems and components. When the model response spectrum method was used, governing response parameters were combined by the square root of the sum of squares (SRSS) rule. However, the absolute sum of the modal responses was used for modes with closely spaced frequencies. Three components of seismic motion were considered: two horizontal and one vertical. The total response was obtained by the absolute sum of one horizontal and one vertical. Floor spectra inputs to be used for design and test verifications of structures, systems, and components are generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis was employed for all structural amplification in the vertical direction. Torsional effects and stability against overturning were considered.

The lumped soil-spring approach was used to evaluate soil-structure interaction effects on structural responses.

The staff concludes that, with the exception of the site specific spectra discussed in Sections 3.7.1 and 2.5, the seismic system and subsystem analysis procedures and criteria used by the applicant as discussed above provide an acceptable basis for the seismic design.

3.7.3 Seismic Subsystem Analysis

The scope of review of the seismic subsystem analysis for Watts Bar included the seismic analysis methods for all seismic Category I piping systems, equipment, and components. It included review of procedures for modeling, combination of the three components of earthquake motion, combinaton of modal responses, inclusion of torsional effects of eccentric masses, and determination of composite damping. The review included design criteria and procedures for evaluation of the interaction of nonseismic Category I piping systems with seismic Category I piping systems, for evaluation of relative support motions of piping systems that interconnect two or more components or span two or more floors. The review also has included criteria and seismic analysis procedures for reactor internals and seismic Category I buried piping outside containment.

The design criteria and the testing program conducted verifies that the mechanical operability and life cycle capabilities of te control rod drive mechanism are in conformance with SRP Section 3.9.4 and specifications acceptable to the staff. The use of these criteria provide reasonable assurance that the reactivity control system will function reliably when required and form an acceptable basis for satisfying the mechanical reliability stipulations of GDC 27.

3.9.5 Reactor Pressure Vessel Internals

The staff evaluated the load combinations, allowable stress, and deformation limits that were used in the design of the reactor internals. The staff review also included the structural integrity of reactor internals under the combination of loads which could result from postulated events such as the SSE and loss of coolant accident.

Although the reactor internals were not constructed to subsection NG of the ASME Code,
all of the Code requirements were satisfied with the exception of the application of a Code N-Symbol stamp and the preparation of "Code" specific-plant stress report. The staff reviewed a summary of stresses, deformation, and usage factors for the Watts Bar rector internals and agrees that allowable limits were satisfied. The applicant has agred to modify the FSAR to clearly state the extent of compliance with the ASME Code and to indicate that allowable limits have been met. This was done in Amendment 45

Subject to documentation as discussed above, the staff finds that the specified transient, and service loadings, and combination of loadings as applied to the design of the reactor interals provide reasonable assurnace tht in the event of ana earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposes on these reactor internals would not exceed allowable stresses and deformation limits for the materials of construction. Lmiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these reactor internals to withstand the most adverse loading events which have been postulated to occur

All aspects of the Westinghouse fuel design are based upon mechanical tests, inreactor operating experience, and engineering analyses. Additionally, the inreactor performance of the design is subject to the continuing surveillance programs of Westinghouse and individual utilities. These programs provide confirmatory and current design performance information.

4.2.2 Thermal Performance

In its evaluation of the thermal performance of the reactor fuel, the staff assumes that densification of the uranium oxide fuel pellets may occur during irradiation in light-water reactors. The initial density of the fuel pellets and the size, shape, and distribution of pores within the fuel pellets influence the densification phenomenon.

Briefly stated, inreactor densification (shrinkage) of oxide fuel pellets (1) may reduce gap conductance, and hence increase fuel temperatures, because of a decrease in pellet diameter; (2) may increase the linear heat generation rate because of the decrease in pellet length; and (3) may result in gaps in the fuel column as a result of pellet-length decreases (these gaps produce local power spikes and the potential for cladding creep collapse).

The engineering methods to be used by Westinghouse to analyze the densification effects on fuel thermal performance have been previously submitted to the staff in Westinghouse Topical Report WCAP-8219 and approved for use in licensing. The methods include testing, mechanical analyses, thermal and hydraulic analyses, and accident analyses. The results of the staff's review are the AEC report, "Technical Report on Densification of Westinghouse PWR Fuel" (Mah 1974), and additional information on densification methods can be found in "The Analysis of Fuel Densification," NUREG-0085.

A Westinghouse fuel thermal-performance code known as PAD-3.1 described in attachments to correspondence from Westinghouse to the AEC* was used for the

SEE NEXT PAGE

*Letters from R. Salvatori, Westinghouse to D. Knuth, AEC, NS-SL-518 (December 22, 1972), NS-SL-521 (December 29, 1972), and NS-SL-543 (January 12, 1973) (proprietary), and NS-SL-527 (January 1, 1973) and NS-SL-544 (January 12, 1973).

Watts Bar safety analysis. A more recent Westinghouse fuel thermal performance code, known as PAD-3.3 (Westinghouse Topical Report WCAP-8785), has also been approved by the NRC. The more recent code, which contains revised models for fission gas release, helium solubility, fuel swelling, and fuel densification, was not used for the Watts Bar safety analysis.

The PAD-3.3 code addresses a concern over enhanced fission gas release at high burnup. Because the previous version of the code, PAD-3.1, does not contain the models necessary to analyze this effect, the staff's safety evaluation of PAD-3.3 stated that future fuel performance analyses must be done with the revised version of the code, PAD-3.3, according to a February 9, 1979 letter from J. Stolz, NRC, to T. Anderson, Westinghouse.

THE W PAD 3.3 code (NOT PAD 3.1) has been used for the WBN fuel performance analysis.

The use of the earlier PAD-3.1 code, rather than the more recent PAD-3.3 code, is generally acceptable because the earlier code produces more conservative thermal conditions than the revised code. This margin, however, does not exist for high burnup fission gas release and fuel rod internal pressure calculations. In a letter dated September 22, 1981, the applicant stated that the more recent code, PAD-3.3, will be used to analyze the fuel thermal performance (including fission gas release and rod internal pressure) at Watts Bar. The staff, therefore, concludes that the Watts Bar fuel performance analysis will be performed in an acceptable manner. The staff will confirm the acceptability of the result of these calculations when they are submitted and report its findings in a supplement to this report.

For the Watts Bar safety analysis, revised internal fuel rod pressure criteria, as described in an approved Westinghouse Topical Report WCAP-8963-A, were used. Briefly stated, these criteria allow the fuel rod internal pressure to exceed the external system pressure. The approved criteria are: (1) the internal pressure is limited so that the fuel-to-cladding gap does not increase during steady-state operation, and (2) extensive departure from nucleate boiling (DNB) propagation does not occur during postulated transients and accidents. Provided that the as-docketed analyses are bounded by the conditions predicted by PAD-3.3 (in WCAP-8785), the staff concludes that these rod pressure criteria will be satisfied for fuel burnups up to the peak target burnup.

Watts Bar LOCA analysis already takes into account the error in the zirc-water reaction as documented in the FSAR.

(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 so as to assure 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 for both constant power and load following operation. A recently discovered error* in the LOCA analysis may lead to a requirement for operation with a peaking factor less than 2.32. A reanalysis of this event using the corrected model will be required before an operating license is granted. In this event, operation at full power may be performed with the axial power distribution monitoring system. This mode of operation has been required in several operating Westinghouse-designed reactors and is acceptable. The requirement for this mode of operation will be inserted into the Technical Specification, if necessary. Another option is the performance of a plant specific analysis to support operation with a lower power peaking factor using excore monitoring.

Two types of instrumentation systems are provided to monitor core power distribution measurements. Excore detectors are used to monitor core power, axial offset, and aximuthal tilt, and moveable incore detectors permit detailed power distributions to be measured. These systems are used in operating reactors supplied by Westinghouse, and the staff finds their use acceptable for Watts Bar.

4.3.2.2 Reactivity Coefficients

The reactivity coefficents are expression of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature,

*The error in the Zirconium-water reaction calculation discovered early in 1978.

4.4.2.2 Fuel Temperature

The fuel temperature design basis given in Section 4.4.1.2 is

> During modes of operation associated with Condition I and Condition II events, the maximum fuel temperature shall be less than the meltng temperature of UO_2. The UO_2 melting temperature for at least 95 percent of the peak kW/ft fuel rods will not be exceeded at the 95 percent confidence level.

This design basis is evaluated in Section 4.2, of this report.

4.4.2.3 Core Flow

Section 4.4.1.3 of the FSAR states the core flow design basis:

92.5 SEE AMEND. 46 in FSAR

> A minimum of ~~95.5~~ percent of the thermal flow rate will pass through the fuel rod region and be effective for fuel rod cooling.

4.4.2.4 Hydrodynamic Stability

The hydrodynamic stability design basis in FSAR Section 4.4.1.4 is:

> Modes of operation associated with Condition I and II events shall not lead to hydrodynamic instability.

4.4.3 Thermal-Hydraulic Design Methodology

4.4.3.1 Departure from Nucleate Boiling

The thermal-hydraulic design analysis was performed using the W-3 Critical Heat Flux (CHF) correlation in conjunction with a THINC-IV analysis. THINC-IV is an open channel computer code which determines the coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distribution along parallel flow channels within a reactor core.

The W-3 correlation was developed from data obtained from experiments conducted with fluid flowing inside single heated tubes. As test procedures progressed to the use of rod bundles instead of tubes, the correlation was modified to include the effects of "R" mixing vane grids and axially nonuniform power distributions.

The applicant has proposed a minimum departure from nucleate boiling of 1.30 to ensure that there is a 95 percent probability at a 95 percent confidence level that critical heat flux will not occur on the limiting fuel rod. The use of the W-3 CHF correlation with a minimum DNBR of 1.30 was previously approved by the staff.

A description of the THINC-IV computer code is given in WCAP-7956, "THINC-IV An Improved Program For Thermal-Hydraulic Analysis of Rod Bundle Cores." The design application of the THINC-IV program is given in detail in WCAP-8054, "Application of the THINC-IV Program to PWR Design." Both WCAP-7956 and WCAP-8054 have been reviewed and approved by the staff.

The design calculational procedure, using THINC-IV, is to perform a core-wide analysis followed by a hot assembly and hot subchannel analysis. For the hot assembly and hot subchannel analyses, a set of hot channel factors is used to account for deviations as a result of manufacturing tolerances. A reload review of a pressurized water reactor, not of Westinghouse design, showed that the hot channel factors used in the thermal-hydraulic analysis of the initial core did not bound future cycles (that is, beyond the first cycle). The staff asked the applicant to determine if his methods appropriately bound future cycles and is awaiting the applicant's response.

SEE OUR RESPONCE, TO Question 221.14 PROVIDED IN AMEND. 45

During the course of its review, the staff requested that the applicant provide the radial pressure gradient in the upper plenum and at the core outlet and that he discuss and support, by calculations, the differences in hot channel pressure drop, flow, enthalpy rise, and minimum DNBR relative to the assumption of a uniform pressure at the core boundaries.

The staff has recently reviewed, under a different docket, a November 2, 1977 letter from C. Eicheldinger (Westinghouse) to J. Stolz (NRC) which described THINC-IV analyses using a cosine upper plenum radial pressure gradient with a maximum value of 5 psi at the core center and o psi at the periphery. The results of these analyses showed that the effects of a core pressure distribution on the minimum DNBR is negligible. The staff conducted a similar sensitivity study using COBRA-IV. The staff results also showed that the effects are small. Based on these analyses, the staff concludes that the use of a nonuniform exit pressure gradient in the Watts Bar thermal-hydraulic design is acceptable.

Before the staff can complete its review of the Watts Bar DNB design methodology, the applicant must demonstrate that the analyses appropriately bound future cycles; however, the staff concludes that the thermal-hydraulic analyses are acceptable for preliminary design approval.

4.4.3.2 Core Flow

92.5 (per FSAR Amend. 46)

The core flow design basis requires that the minimum flow which will pass through the fuel rod region and be effective for fuel rod cooling is ~~95.5~~ percent of the primary coolant flow rate. The remainder of the flow, called bypass flow, will be ineffective for cooling because it will take the following bypass paths:

(1) flow through the spray nozzles into the upper head

(2) flow into the rod cluster control rod guide thimbles

(3) leak from the vessel inlet nozzle directly to the vessel outlet nozzle

(4) flow between the baffle and barrel

(5) flow in the gaps between the fuel assemblies

The amount of bypass flow is determined by a series of hydraulic resistance calculations on the core and vessel internals and verified by model flow tests. Because the amount of bypass flow is consistent with approved plants of similar design, the staff concludes that the core bypass flow used in the design analysis, ~~4.5~~ percent, is acceptable.

7.5
SEE FSAR Amendment 46

4.4.3.3 Hydrodynamic Instability

For steady-state, two-phase heated flow in parallel channels, the potential for hydrodynamic instability exists. The applicant used the HYDNA program to demonstrate that the core is stable. The applicant also referenced WCAP-7240, "An Experimental Investigation of the Effects of Open Channel Flow on Thermo-Hydro-Dynamic Flow." WCAP-7240 presents experimental data intended to show that simulated fuel assemblies without an enclosing shroud will provide a larger stability margin than would fuel assemblies with an enclosing shroud. Although such data are useful as background information, they are not sufficient to predict the onset of flow instability in the core. Therefore, the staff will require the applicant to provide a discussion excluding the HYDNA code that supports the contention that the core is thermal-hydraulically stable.

SEE FSAR SECTION 4.4.3.5 (AMEND. 41)

4.4.4 Operating Abnormalities

4.4.4.1 Fuel Rod Bowing

A significant parameter that influences the thermal-hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The Westinghouse methods for predicting the effects of rod bow on DNBR are under review by the staff. Therefore, the magnitude of rod bow as a function of burnup was evaluated based on interim methods which have been previously accepted. The resultant reduction in the departure from nucleate boiling ratio due to rod bow is given in Table 4.1.

Table 4.1 Rod bow penalties

Burnup (MWD/MTU)	DNBR penalty (%)
0	0
3,500	0
5,000	0
10,000	2.15
15,000	4.64
20,000	6.74
25,000	8.59
30,000	10.27
35,000	13.07
40,000	19.09

Prior to issuance of the Technical Specifications, the staff will ensure that the thermal margin reductions given above have been accommodated using an acceptable method.

For plants designed by Westinghouse, the staff has approved the following generic margins in Table 4-2 (see "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," December 1976), which may be used to offset the reduction in DNBR as a result of rod bowing.

Table 4-2 Generic margins

Margin	% reduction
Use of a design minimum DNBR of 1.30 instead of the 95/95 DNBR limit of 1.28	1.6
Reduction in fuel rod pitch for the hot channel analysis	1.7
Use of a thermal diffusion coefficient (TDC) of 0.038 instead of a TDC of 0.051	1.2
Addition of an extra grid in the design of the Westinghouse 17 x 17 fuel assembly relative to the 15 x 15 fuel design	2.9
0.86 SEE FSAR AMEND. 46 Use of a $\boxed{0.88}$ multiplier on the modified spacer factor (F_s) of the W-3 correlation instead of a 0.865 multiplier	1.7
Maximum generic margin which may be claimed	9.1

Plant-specific margins which could be available are

(1) The Technical Specification minimum flow rate is greater than the design flow rate

(2) The Technical Specification maximum T_{avg} is less than the design T_{avg}

(3) The trip setpoints are more limiting than the thermal-hydraulic analysis indicates

Table 4.3 Reactor Design Comparison

	Watts Bar Units 1&2	Trojan SER	Sequoyah SER
Performance characteristics			
Reactor core heat output, (MWt)	3411	3411	3411
System pressure, psia	2250	2250	2250
Minimum DNBR	1.30	1.30	1.30
Typical cell	2.08	2.04	2.22
Thimble cell	~~1.74~~ 1.72	1.71	1.81
Critical heat flux correlation	W-3	W-3	W-3
Coolant flow			
Total flow rate (10^6 lb/hr)	144.8 ~~140.3~~	132.7	133.8
Effective flow rate for heat transfer (10^6 lb/hr)	133.9 ~~134.0~~	126.7	127.8
Average velocity along fuel rods, (ft/s)	16.6 ~~16.7~~	15.7	15.6
Effective core flow area (ft^2)	51.1	51.1	51.1
Coolant temperature, °F			
Nominal reactor inlet	559.1 ~~558.1~~	552.7	545.7
Average rise in core	62.5 ~~62.7~~	66.9	67.8
Pressure drop across core (psi)	~~27.9±5.6~~ 25.8±3.9		24.3±2.4
Heat transfer, 100 % power			
Active heat transfer surface area, (ft^2)	59.700	59.700	59.700
Average heat flux, (BTU/hr-ft^2)	189,800	189,800	189,800
Maximum heat flux, (BTU/hr-ft^2)	440,300	574,500	474,500
Average linear heat rate (kW/ft)	5.44	5.44	5.44
Maximum thermal output (kW/ft)	12.6	13.6	12.2

SEE TABLE 4.4-1 (revised by amend. 46) in FSAR be tested to ensure compatability with the operating environment. Alert levels will be set to detect a loose part having an impact of 0.5 ft-lb during plant shutdown. Because operating conditions will cause variation in the amount of background noise, the sensitivity will be the same percentage of the background noise.

The applicant has stated that the training program for the operators of the LPMS will be the same as that for Sequoyah.

The staff has evaluated the Watts Bar LPMS by comparing it with the equipment and procedures used on other comparable plants, taking into account pertinent differences, and the requirements of Regulatory Guide 1.133. Based on these comparisons, the staff concludes that before it can determine the acceptability of the Watts Bar LPMS, it will require the applicant to submit a detailed description of the operator's training program.

4.4.6 Thermal-Hydraulic Comparison

Table 4.3 lists the thermal-hydraulic design parameters for the Watts Bar facility and compares them to values for the Trojan and Sequoyah facilities.

The Watts Bar units were designed to operate at the same thermal power as the Sequoyah and Trojan plants. The W-3 CHF correlation and THINC-IV computer program were used in the design of all of the plants. The major differences between the Watts Bar and Trojan plants are a higher nominal inlet temperature, a lower flow rate, and a lower maximum heat flux. The higher nominal inlet temperature for the Watts Bar units results in a decrease in the thermal margin. However, the higher flow rate and the lower maximum heat flux compensate and result in the minimum DNBR, at nominal conditions, increasing from 1.71 for Trojan to ~~1.74~~ for Watts Bar. Therefore, the net change for Watts Bar is a slightly greater thermal margin.
1.72

The differences in the Watts Bar and Sequoyah designs are a higher flow rate and inlet temperature for Watts Bar. The higher inlet temperature results in a lower minimum DNBR for Watts Bar, ~~1.74~~ for Watts Bar as compared to 1.81 for
1.72

The staff concludes that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on the coating systems and their applications meeting (1) the positions of Regulatory Guide 1.54, with an acceptable alternate to ANSI N101.4 (1972), and (2) the testing requirements of ANSI N101.2. The coating systems chosen by the applicant have been qualified under conditions which take into account the postulated DBA conditions.

The control of combustible gases that can potentially be generated from the organic materials and from qualified and unqualified paints is reviewed in Section 6.2.5 of this report. The consequences of solid debris that can potentially be formed from unqualified paints are reviewed in Section 6.2.2.

6.1.3 Postaccident Emergency Cooling Water Chemistry

This review is related to providing and maintaining the proper pH of the containment sump water and recirculated containment spray water following a design basis accident to reduce the likelihood of stress corrosion cracking of austenitic stainless steel.

The applicant will use borated water (concentration $\geq$2000 ppm boron) from the refueling water storage tank during the injection phase of containment spray. The borated water from the containment spray drains to the sump. The ice in the ice condenser contains sodium tetraborate equivalent to a concentration of 1800 ~~2000~~ ppm as boron. The melted ice also drains to the containment sump and, after mixing, will raise the sump water pH to $\geq$7.0. Mixing is achieved as the solution is continuously recirculated from the sump to the containment spray nozzles during the recirculation phase of containment spray.

PER DRAFT T.S.

The staff evaluated the pH of the water (mixture of refueling water storage tank and ice condenser borated water) in the containment sump. The staff verified by independent calculations that sufficient sodium tetraborate is available to raise the containment sump water pH above the minimum 7.0 level to reduce

The sump water level is monitored by four level measurement channels utilizing differential pressure transmiters. the staff was concerned that debris in the sump could block the inlets to the differential pressure transmitters and result in a loss of the permissive signal to the switchover logic. The applicant has initiated a design change to provide protection to prevent debris from entering the level sensors. This is acceptable; however, formal documentation is required.

7.3.3 Auxiliary Feedwater Initiation and Control

In the event of a loss of the main feedwater supply, the auxiliary feedwater (AFW) system supplies sufficient feedwater to the steam generators to remove the energy stored in the primary system. The two plant units have separate AFW systems except for certain shared support facilities. Each system has two 470-gpm electric-motor-driven pumps, and one 940-gpm turbine-driven pump. Each of the electric pumps serves two steam generators; the turbine-driven pump serves all four. The preferred sources of water for the AFW system are the to condensate storage tanks. The backup water supply is provided by the essential raw cooling water (ERCW) system.

The pumps start automatically on a loss of offsite power, loss of both main feedwater pumps, or a safety-injection signal. The electric-motor-driven pumps also start automatically on a two-out-of three low-low water level signal from any steam generator; the turbine-drive pump starts automatically on a two-out-of-three low-low level signal from any two steam generators. All pumps also can be started either remote-manually or locally. A modulating level control valve (which is normally closed) between each pumps and each steam generator fed by the pump receives an opening signal on a low-low water level in the steam generator. After an accident, the operator can take manual control of the system-level operation by blocking the initiating signal with a handswitch. However, if on other initiation signal occurs, the control will again revert to automatic. Manual controls of individual components will not be overridden.

The staff reviewed applicable information in Sections 7.3 and 10.4.9 of the FSAR. In response to a request for additional information, the applicant garic a detailed presentation of the automatic initiation, operation, reset, and control of the AFW system. The applicant stated that the auxiliary feedwater

This sentence is not correct and does not reflect WBN FSAR section 7.3.2.2. It should be deleted.

control circuitry, is safety-grade, Class 1E, is powered from a power source connected to the emergency power system, and is testable on line. A manual-initiation capability that is independent of the automatic initiation is also provided. The design of the auxiliary feedwater system for the Watts Bar plant is similar to that for the Sequoyah plant for which a failure mode and effects analysis has been performed and documented in the FSAR. The staff finds that the design of the auxiliary feedwater system is acceptable. Action Plan requirements are addressed in Section 7.8.3.

7.3.4 Falure Modes and Effects Analysis

After the staff advised the applicant that the FSAR did not provide a facilure mode and effects analyses of the safety features actuation system (SFAS) as required by Regulatory Guide 1.70, the applicant referenced Westinghouse Topical Report WCAP-8584. The applicant stated that ~~the Topical Report covers the SFAS, including the post TMI modifications and that~~ the BOP design satisfies the interface criteria of WCAP-8584 and the topical report does not cover the post TMI modifications to the SFAS system since none were required. The staff finds this acceptable.

7.3.5 IE Bulletin 80-06

IE bulletin 80-06 calls ESF for review of, with the objective of ensuring that no device will change position solely because of the reset action. The applicant has stated that the requested reviews have been performed, and has committed to perform the confirmatory tests requested by the Bulletin. The staff has asked the applicant to provide a listing of all devices found to change state upon reset during the drawing review, along with justification for any such devices for which corrective action is not taken. The staff will confirm that adequate justification is provided for any items that change state on reset. Additional information and formal documentation are required.

7.3.6 Conclusions

The ESFAS includes the instrumentation and controls used to detect a plant condition requiring operation of an ESF system, to initiate action of the ESF, and to control its operation. The scope of the review of the ESFAS included

The staff has audited the following and found them acceptable: conformance to system redundancy and diversity; single failure; both electrical and physical separation; identification of control boards, equipment, cables and cable trays; and system testing and inoperable status surveillance criteria.

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 resolution of the concerns identified in Sections 7.1.3.1, 7.3.2.1, and 7.3.2.4.

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 air compressors
(6) reactor containment fan cooler units
See WBN FSAR p. 7.4-3 Amendment 45
(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

The following variables need to be monitored and indicated:

(1) water level for each steam generator
(2) pressure for each steam generator
(3) pressurizer water level
(4) pressurizer pressure

The controls for all of the equipment and the indicators listed above are in the main control room. In addition, an auxiliary control room is provided that allows the plant to be maintained in a hot shutdown condition or taken to cold shutdown should the main control room become uninhabitable. Transfer switches allow transferring the controls from the main control room to the auxiliary control system ~~room~~. Placing a transfer switch in the auxiliary room in the auxiliary ~~operating~~ position gives an audible alarm in the main control room. Systems requiring infrequent alignment have their auxiliary controls at the motor control centers.

7.4.2 Shutdown from Auxiliary Control Room

The auxiliary control room provides a means to shut down the reactor in case the control room is uninhabitable. The staff questioned the applicant about the need for a test to prove the adequacy of design of the auxiliary control room controls and instrumentation to safely shut down the reactor and to maintain the plant in this condition. In his response, the applicant stated that a test to shut down the reactor using instrumentation to the auxiliary control room was performed on Sequoyah Unit 1 from an initial power level of 30 percent. The test was successful, and the cold shutdown was reached in 36 hours as required by BTP RSB 5.1 using safety-grade equipment. Because the design of the systems required for safe shutdown of the Watts Bar plant is very similar to that of Sequoyah, the applicant does not consider it necessary to perform a similar test at Watts Bar. The staff finds this acceptable.

electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies; (3) physical independence of circuits, and (4) availability of circuits.

The Watts Bar plant is interconnected to the electric grid system through SIX ~~seven~~ 161-kV transmission lines that terminate on an existing 161-kV switchyard (Watts Bar Hydro Plant Switchyard) 1.5 mi from the plant. The 161-kV lines enter the switchyard by way of a number of physically separate and independent rights of way. In addition, five hydro generators and four steam-driven generators terminate at the switchyard.

The 161-kV switchyard consists of circuit breakers, disconnect switches, transformers, buses, and associated equipment arranged so that each incoming or outgoing transmission line can be connected to one or to both main buses through circuit breakers. Switchyard protective relays include transmission line protective relays and switchyard bus differential relays. These relays are backed up by switchyard bus breakup relays and by switchyard circuit breaker failure relays.

Offsite power from the switchyard to the onsite Class 1E distribution system is from two independent immediate-access circuits. Each of the two circuits is routed from the switchyard through a 161-kV transmission line and 161-to-6.9 kV transformer (common station service transformer) to the onsite Class 1E distribution system.

The onsite Class 1E distribution system consists, in part, of two redundant and independent 6.9 kV buses, each capable of being fed from either of the above-described offsite circuits. Offsite power is normally supplied to the onsite distribution system from the plants main generator through a 22.5-to-6.9 kV transformer (unit station service transformer) and 6.9-kV switchgear (unit start board) to the onsite Class 1E distribution system. For any unit generator trip, offsite power is automatically transferred from the normal supply to the two preferred offsite circuits.

The ERCW piping from the standpipe to the cooling tower is TVA class H (non-seismic). The ERCW piping

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.

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 The determination of the system's acceptability to these regards has been deferred to the Mechanical Engineering 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 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. ERCW standpip

The ERCW pumps and pump motors are housed in a Category I structure that shields against horizontal and vertical tornado missiles. Though 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.

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

The turbine generator is manufactured by the Westinghouse Company and is a tandem-compound type (single shaft) with one double-flow high pressure turbine and three double-flow low pressure turbines. The rotational speed is 1800 rpm and is designed for a gross generator output of 1218 MWe at a nominal plant exhaust pressure of ~~20.0~~ in. mercury (absolute).
2.0

The turbine generator is equipped with an digital electrohydraulic control (EHC) system. The EHC system consists of an electronic governor using solid state control techniques in combination with a high pressure hydraulic actuating system. The system includes electrical control circuits for steam pressure control, speed control, load control, and steam control valve positioning.

Overspeed protection is accomplished by three independent systems (normal speed governor, mechanical overspeed, and electric backup overspeed control systems). The normal speed governor modulates the turbine control valves to maintain desired speed load characteristics, and it will close the intercept valves and control valves at 103 percent of rated speed. The mechanical overspeed sensor trips the turbine stop, control, and combined intermediate valves by deenergizing the hydraulic fluid systems when 111 percent of rated speed is reached. The turbine steam valves close in 0.15 sec., and the extraction valves close in less than a second after overspeed condition is detected. These valves are designed to fail closed on loss of hydraulic system pressures. The electrical backup overspeed sensor will trip these same valves when 111 percent of rated speed is reached by independently deenergizing the hydraulic fluid system. Both of these actions independently trip the energizing trip fluid system. The overspeed trip systems can be tested while the unit is on line. Therefore the requirements of GDC 4 are met.

To protect the turbine-generator, the following signals will shut down the turbine: (1) low bearing oil pressure, (2) low vacuum trip, (3) high thrust bearing temperature, (4) high turbogenerator vibration (5) low differential water pressure across generator starter coils, (6) high stator coil outlet water temperature, (7) low EHC fluid tank level, (8) low lube oil tank pressure, (9) low EHC fluid pressure, (10) low auto stop oil pressure, (11) turbine overspeed at 111 percent of rated speed (electrical trip), (12) turbine overspeed history of similar turbine disc cracking, and results of laboratory tests. This prediction method takes into account two main parameters, the yield strength of the disc, and the temperature of the disc at the bore area where the cracks of concern are occurring. The higher the yield strength of the material and the higher the temperature, the faster the crack growth rate will be.

In July 1981, NRC advised all licensees and applicants by letter that they were requested to continue their current programs in cooperation with the turbine suppliers to properly monitor the condition of the disc bore and keyway areas. The staff considers that these programs meet the intent of current staff guidelines.

The turbine meets staff criteria regarding the use of materials with acceptable fracture toughness and adequate design. Preservice and inservice inspection criteria are in accordance with current staff guidelines. The materials, processes, and designs used by the applicant are therefore considered acceptable. The staff concludes that these provisions provide reasonable assurance that the probability of disc failure with missile generation is low during normal operation, including transients up to design overspeed.

10.3 Main Steam Supply System

The function of the main steam supply system is to convey supply steam from the steam generators to the high-pressure turbine and other auxiliary equipment for power generation. Section 10.3.1 evaluates the safety-related portion of the main steam system including the main steam isolation valves (MSIVS). Section 10.3.2 evaluates the nonsafety-related portion of the main steam system downstream of the MSIV up to and including the turbine stop valves.

10.3.1 Main Steam Supply System (Up to and Including the Main Steam Isolation Valves)

The main steam supply system (MSSS) routes the steam generated in each of the four steam generators to the high pressure, low pressure, and main feedwater

The main feedwater regulator valves and the main feedwater isolation valves, both upon receipt of the feedwater isolation signal (high-high steam generator level, ESF actuation signal, or reactor trip), will close in 5.0 seconds and 6.5 seconds respectively.

system serves no safety function and is therefore classified as nonsafety related (Quality Group D, nonseismic Category I), except for the portion of the system between the check valves located outside the containment, including the containment isolation valves and up to the steam generators. This portion is classified as safety related and is designed to seismic Category I, Quality Group B requirements in order to ensure feedwater system isolation in accident situations. This portion of the system is located in seismic Category I, flood- and tornado-protected structures. The structures provide protection against tornado missiles. The essential equipment is separated from the effects of internally generated missiles and is not affected by failures in high energy piping, thus satisfying the requirements of GDC 2 and 4 and the guidelines of Regulatory Guide 1.17. Complete isolation of the main feedwater system is provided when it is required to mitigate the consequences of a steam or feed-water line break. ~~The main feedwater isolation valves close within 5 sec on receipt of a "high-high" steam generator level signal, ESF actuation signal, or reactor trip signal.~~ Feedwater is isolated from any one steam generator on indication of high level in that particular steam generator.

The use of the standby feedwater pump is the normal means for starting up and shutting down the plant. This pump is also automatically activated in the event of the loss of one main feedwater pump. This is accompanied by an automatic turbine runback to 85 percent of load if the power level is above 80 percent of full power. Should main feedwater flow continue to decrease, the auxiliary feedwater system will automatically activate when the low-low steam generator level is reached. The auxiliary feedwater system (see to Section 10.4.9) automatically provides flow to the steam generators for decay heat removal upon the loss of normal feedwater supply.

The applicant has stated that the feedwater system has been designed to prevent waterhammer induced by the piping system, and has committed to perform a test utilizing standard plant procedures which demonstrates that unacceptable damage will not occur (see Section 10.4.9 of this SER).

The applicant provided a comparison of the system to that which was designed for Sequoyah. The comparison stated that the design and design philosophy of

Each AFWS consists of two trains, each train utilizing one electric motor-driven pump to supply two steam generators. One pump is capable of providing sufficient flow to its steam generators to prevent the release of reactor coolant via the pressurizer safety valves, thus satisfying the requirements of GDC 44. The turbine-driven pump provides AFW flow to all four steam generators via redundant trains.

The preferred water sources for all AFW pumps are the two nonseismic, Quality Group D condensate storage tanks (CST). An unlimited backup water source, the essential raw cooling water (ERCW) system can provide the AFW pumps with the required supply in the event the CST is unavailable. The ERCW supply is automatically (or remote-manual) initiated when low pressure is sensed in any one of the three AFW pump suction lines.

The applicant has provided verification (through analysis) that the AFW pumps can survive the transition to the backup water source in the event the preferred source is unavailable. The staff will require that the vendor of the pumps concur with the results of the analysis that verify pump survivability or that the applicant perform a suitable test which demonstrates that the pumps can survive the transfer.

The electric motor-driven pumps and all associated controls, valves, and other supporting systems can be powered from onsite ac sources. The turbine-driven pump and all associated valves, controls, and other supporting systems are powered by steam, compressed air, and, if necessary, dc power. The turbine-driven pump is designed to be independent from all ac power for 2 hr.

The AFW system is housed within seismic Category I, tornado-missile-protected structures. The system is not protected against flooding. During flood operations, the fire protection system will directly supply the steam generator with water. Thus, the system conforms to the requirements of GDC 2 and the guidelines of Regulatory Guide 1.117, "Tornado Design Basis." Protection from internal flooding due to pipe rupture is discussed in Section 9.3.3 of this SER.

Based on discussions with the NRC's ASB (August 26 + 27, 1981) TVA committed to provide verification (analysis and FSAR change) only. A test or vendor concurrence was not discussed.

3 Based on its evaluation, as described below, the staff finds the proposed liquid, gaseous, and "dry" solid radioactive waste systems and associated process and effluent radiological monitoring and sampling systems to be acceptable.

11.2 Liquid Waste Management

The liquid waste processing system (LWPS) for Watts Bar is shared between Units 1 and 2. The LWPS consists of process equipment and instrumentation necessary to collect, process, monitor, and recycle or dispose of radioactive liquid wastes. The liquid radwaste system is designed to collect and process wastes based on the origin of the waste in the plant and the expected levels of radioactivity. All liquid waste is processed on a batch basis to permit optimum control of releases. Prior to being released, samples are analyzed to determine the types and amounts of radioactivity present. Based on the results of the analyses, the waste is recycled for eventual reuse in the plant, retained for further processing, or released under controlled conditions to the environment.

A radiation monitor in the discharge line automatically terminates liquid waste discharges if radiation measurements exceed a predetermined level. The liquid radioactive waste processing system consists of the tritiated and nontritiated waste subsystems and a condensate regenerant waste subsystem. In addition, the chemical and volume control system (CVCS) processes letdown from the primary system to control boron concentration and reactor water purity. In its evaluation model, the staff assumed that a portion of the CVCS flow will be released through the LWPS for tritium control. A deep-bed regenerable demineralizer system is provided for treatment of turbine condensate. Steam generator blowdown will be cooled and normally sent directly to the condensate cleanup system for processing and resuse in the plant. Laundry, hot shower, and decontamination wastes are normally released without treatment; the floor drain (dirty waste) subsystem is used to treat effluents from these sources when radioactivity concentrations are in excess of pre-established limits.

The LWPS consists of the boron recycle system, the tritiated waste system, the floor drain (dirty waste) system that handles nontritiated waste and condensate

Procedures and training for post-accident sampling and analysis are provided under the Watts Bar Health Physics Instruction Manual, Radiation Control Instruction Manual, and inplant Training Program.

The post-accident radioiodine sampling and analysis provisions described for the Watts Bar facility satisfactorily meet the staff positions for fuel loading and full-power operations as outlined in NUREG-0737 and are acceptable.

IS THIS CONSISTENT WITH LAST PARAGRAPH ON PAGE 12-8 the control room. They will also have variable alarm setpoints and local audible alarms. The detectors will be calibrated quarterly. Therefore, the staff concludes that the area radiation monitoring system design is acceptable.

The applicant has provided area radiation monitors around the fuel storage areas to meet the requirements of 10 CFR 70.24 and to be consistent with the guidance of Regulatory Guide 8.12, "Criticality Accident Alarm Systems."

The applicant will rely on the area radiation monitoring system and portable radiation monitoring instruments to assess the radiation hazard to personnel in areas that may be accessed during the course of an accident. The area monitors will receive backup power from the diesel generators. The portable instruments will be placed to be readily accessible to personnel responding to an emergency. The portable instruments will be designed with a sufficient instrument range for use in the event of an accident.

The airborne radioactivity monitoring system is designed to provide a clear indication locally and to operations personnel when abnormal amounts of airborne radioactive material occur, and to provide information so that action can be taken to ensure that inhalation of airborne particulates and iodine is within the limits of 10 CFR Part 20. These airborne radioactivity monitors have the capability to detect 10 maximum-possible-concentration hours (mpc-hr) of particulate and iodine radioactivity in any compartment which has a possibility of containing airborne radioactivity and which may be occupied by personnel. The applicant's portable airborne radioactivity monitoring systems will monitor air in areas not provided with fixed airborne radioactivity monitors. The objectives and location criteria for these monitors are in conformance with 10 CFR Part 20 and 50 and Regulatory Guides 8.2 and 8.8. The staff concludes that the radiation protection design features for Watts Bar are acceptable with the SRP Section 12.3 criteria, however, evaluations in accordance with TMI Lessons Learned post-accident conditions established in Items 2.1.8.c/III.D.3.3 of NUREGs-0578, 0660, 0694, and clarification letters remain to be completed.

IS THIS CONSISTENT WITH LAST PARAGRAPH ON PAGE 12-12

(1) Power range high neutron flux
(2) High pressure
(3) Low pressure
(4) Overpower ΔT
(5) Overtemperature ΔT
(6) Low coolant flow
(7) Pump undervoltage/underfrequency
(8) Low steam generator level
(9) High steam generator water level

Watt Bar FSAR section 15.2.10.2 states that the steam generator high-high level trip setpoint initiates a turbine trip. The turbine trip signal initiates a reactor trip only when reactor power is greater than 50-percent (P-9 interlock).

Time delays to trip, calculated for each trip signal, are included in the analyses.

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control insertion accounts for a stuck rod in accordance with GDC 26.

The flow coastdown code PHOENIX has been found satisfactory for evaluating transients and accidents in the Watts Bar plant. Staff reviews of the FACTRAN, BLKOUT, MARVEL, and LOFTRAN codes have progressed to the point that there is reasonable assurance that analyses results dependent on the codes will not be appreciably altered by any method revisions that may be required by the staff.

Transients and accident were analyzed for Watts Bar using a procedure for evaluating fuel performance, which conservatively bounds the consequences of the event by accounting for fabrication and operating uncertainties directly in the calculations. DNBRs were calculated using the W-3 correlation, with a minimum DNBR of 1.3 used as a design limit.

The applicant accounts for errors in initial conditions by making the following assumptions as appropriate for the event being considered:

(1) Core power, 3411, +2% MW
(2) Average reactor coolant system temperature (T_{AV}), 572.2, $\pm$6.5°F
(3) Pressure (at pressurizer), 2250, $\pm$30 psi

Table 15.2 (Continued)

Time step	Recirculation flow ft^3/min	Exhaust flow ft^3/min
0-30 sec	0	0
30-180 sec	770	3230
180-360 sec	1700	2300
360-600 sec	2500	1500
600-1200 sec	3340	660
1200-1800 sec	3810	190
1800 sec-30 days	3900	100
Filter efficiencies (%)		
Elemental iodine		99
Organic iodine		95
Particulate iodine		99
Ice condenser removal efficiency (%)		
Elemental iodine		30
Flowrate through ice condenser (ft^3)		40,000
Period of ice condenser effectiveness (min)		10-60
Primary containment leak rates (%)		
0-24 hr		0.25
24-hr-30 days		0.125
Bypass leakage fraction (%)		0
Minimum exclusion area boundary distance (m)		1200
Low population zone distance (m)		4828
Atmospheric diffusion (χ/Q) values (sec/n13)		
0-2 hr		3.6 E-4
0-8 hr		5.0 E-5
8-24 hr		3.3 E-5
1-4 days		1.3 E-6
4-30 days		2.7 E-6

These values should be checked since it is not expected tha the 4-30 days value would be higher tha the 1-4 days value at the LPZ.

In the analysis of the design basis LOCA, the primary containment was assumed to leak at the design leak rate of 0.25 percent per day for the first 24 hours following the accident and at 0.125 percent per day thereafter. The applicant established to the satisfaction of the staff that the shield building annulus and auxiliary building would not experience a period of positive pressure and, hence, no exfiltration. Therefore, no leakage was assumed to bypass the gas treatment systems throughout the course of the accident.

The QA program is implemented under the direction of the Manager, Nuclear Regulation and Safety. Implementation is carried out through the Office of Power Quality Assurance Manager and the directors of other involved divisions. The Quality Assurance Manager is responsible to develop, coordinate, monitor, audit, and evaluate the QA program to meet regulatory requirements and guidance as well as licensing commitments.

Within the Office of Power, there are full-time QA staffs at several organization levels. These QA personnel have the authority to identify quality problems, to provide solutions, to verify implementation of solutions, and to stop an activity through the appropriate channels in accordance with TVA procedures when the work fails to comply with approved specifications and plans.

The Director of the Division of Nuclear Power is responsible for the operation and maintenance of the Watts Bar nuclear plant during the operations phase. He has delegated the responsibility of the day-to-day operation and maintenance activities for the plant to the ~~Assistant Director (Operations)~~ Production Manager. The Plant Superintendent, who reports to the ~~Assistant Director (Operations)~~ Production Manager, has primary responsibility for operating and maintaining the Watts Bar Nuclear Plant in compliance with the requirements of the Operating License and the plant Operational Quality Assurance Manual. The resolution of any disputes on QA program requirements arising between QA personnel and other department personnel that cannot be resolved locally are referred to higher management for resolution, with eventual resolution by the Manager of Power, if necessary.

The Supervisor of the plant QA staff, who reports ~~to the Plant Superintendent,~~ offsite to the Chief, Field QA Staff of the Quality Assurance and Compliance Branch. ~~communicates directly with the Office of Power In-Plant Quality Assurance Coordinator, who reports to the Quality Assurance Manager, in matters relating to the policies and practices of the operational QA program.~~ This Supervisor and his plant QA staff perform QA functions relative to plant operations and provide inspections and verification of those activities. They review drawings, specifications, purchase requisitions, and plant instructions and procedures covering activities such as test, calibration, special processes, maintenance, modification, and repair for compliance with the QA program requirements. They are responsible for developing and implementing the inspection program covering operations, maintenance, repair, and test.

skill in the performance of their quality-related activities. The training program also provides for retraining of personnel to ensure proficiency of performing activities affecting quality is maintained.

Quality is verified through checking, review, surveillance, inspection, testing, and audit of quality-related activities. The QA program requires that quality verification be performed by individuals who are not directly responsible for performing the actual work activity. Inspections are performed in accordance with procedures, instructions, and/or checklists reviewed ~~approved~~ by the plant QA staff, and approved by the ~~plant~~ Plant Superintendent ~~the Plant Superintendent,~~ and the Plant Operations Review Committee. Inspections are performed by qualified personnel who are trained in accordance with TVA training programs.

External audits of vendors and service contractors and internal audits of all aspects of the QA program are conducted by ~~the~~ QA organizations. Audits are performed in accordance with pre-established written procedures by appropriately trained personnel not having direct responsibilities in the areas being audited. Audits, which are conducted at scheduled intervals and/or on a random unscheduled basis, include an objective evaluation of (1) the effectiveness of implementation of the QA program; (2) the adequacy of and compliance with QA policies, practices, procedures, and instructions; (3) the adequacy of work areas, activities, processes, items, and records; and (4) product compliance with applicable engineering drawings and specifications. The QA program requires documentation of audit results and review by management having responsibility in the area audited to determine and take any needed corrective action. Followup audits are performed to determine that nonconformances are effectively corrected and that the corrective action precludes repetitive occurrences. Audit findings, which indicate performance trends and the effectiveness of the QA program, are also reported to responsible management for review and assessment.

In addition to audits, ~~there is continual monitoring of onsite activities by the Office of Power In-Plant Quality Assurance Coordinator, and~~ there are annual independent management reviews of parts of the QA program with the total program being reviewed biennially.

This listing of abbreviations ~~are~~ are not fully applicable to WBN. They seem to be based on a CE plant

APPENDIX F

ABBREVIATIONS

ACCWS - auxiliary component cooling water system
ACFM - actual cubic feet per minute
ACRS - Advisory Committee on Reactor Safeguards
ADV - atmospheric dump valve
AFW - auxiliary feedwater
AFWS - auxiliary feedwater system
ALARA - as low as reasonably achievable
ANO-2 - Arkansas Nuclear One, Unit 2
ANSI - American National Standards Institute
ASB - Auxiliary Systems Branch
ASI - axial shape index
ASLB - Atomic Safety and Licensing Board
ASME - American Society of Mechanical Engineers
ANS - American Nuclear Society
ASOEA - Assistant Secretary of the Office of Environmental Affairs
ATWS - anticipated transient without scram
BIL - basic impulse insulation level
B&O - Bulletins and Orders (Task Force)
BOL - beginning of life
BS - Bachelor of Science degree
B&W - Babcock and Wilcox
BWR - boiling water rector
BTP - Branch Technical Position
CACS - containment air cooler system
CAD - containment atmosphere dilution
CARS - containment atmosphere release system
CCAS - containment cooling actuation signal
CCS - containment cooling system
CCW - component cooling water
CD - consolidated drained
CE - Combustion Engineering Inc.
CEA - control element assembly
CEADS - control element assembly drive system
CEDM - control element drive mechanism
CEPADAS - Computerized Emergency Planning and Data Acquisition System
CFR - Code of Federal Regulations
CESSAR - Combustion Engineering Standard Safety Analysis Report
CHF - critical heat flux
CGCS - combustible gas control system
CIAS - containment isolation actuation signal
CIS - containment isolation signal
COLSS - core operating limit supervisory system
CP - construction permit
CPCS - core protection calculator system
CPIS - containment purge isolation signal
CRD - control rod drive