Docket Nos. 50-390
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Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

Ten copies of the Safety Evaluation prepared by the Directorate of Licensing concerning your application for the Watts Bar Nuclear Plant, Units 1 and 2 are enclosed for your use. The Safety Evaluation will be made a part of the record for the public hearing to be scheduled in this matter.

Sincerely,

Original signed by R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Safety Evaluation (10)
see rpt jacket

cc w/enclosure:
Robert H. Marquis, Esquire
General Counsel
629 New Sprankle Building
Knoxville, Tennessee 37919

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Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

SAFETY EVALUATION OF THE WATTS BAR NUCLEAR PLANT UNITS NO. 1 & 2

Docket Nos: 50-390
50-391

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ROOM 016

ATOMIC ENERGY COMMISSION
UNITED STATES OF AMERICA

U.S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF LICENSING
WASHINGTON, D.C.

Issue Date:

AUG 2 6 1972

August 28, 1972

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT

RHEA COUNTY, TENNESSEE

DOCKET NOS. 50-390 AND 50-391

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APPENDICES

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The Tennessee Valley Authority (hereinafter referred to as TVA or the applicant) by application dated May 14, 1971, and as subsequently amended, requested licenses to construct and operate its proposed Watts Bar Nuclear Plant, Units 1 and 2 on federally-owned land in Rhea County, Tennessee. The site for these reactor units is located on the west bank of the Tennessee River approximately 50 miles northeast of Chattanooga, Tennessee.

The applicant will be the owner of the proposed facility and will be responsible for its overall design and construction. The nuclear steam supply systems, each utilizing a closed-cycle pressurized water reactor, will be supplied by the Westinghouse Electric Corporation.

Each of the proposed reactors is designed to operate at 3411 thermal megawatts (MWt) with an expected ultimate capability of producing 3582 MWt. The design of the engineered safety features and the consequences of postulated accidents have been analyzed by the applicant and evaluated by the regulatory staff at the higher power level of 3582 MWt. The nuclear, thermal and hydraulic characteristics of the core were evaluated on the basis of a maximum core power level of 3411 MWt. Before operation at any power level above 3411 MWt is authorized, the regulatory staff will perform a safety evaluation to assure that the facility can be operated safely at the higher power level.

Our technical safety review with respect to issuing construction permits for the Watts Bar Nuclear Plant has been based on the applicant's Preliminary Safety Analysis Report and subsequent Amendments 1 through 14 inclusive, all of which are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, NW, Washington, D. C. and at the Dayton Public Library, First Avenue, Dayton, Tennessee. In the course of our review of the material submitted, we held several meetings with the applicant and the Westinghouse Electric Corporation. A chronology of our review is attached as Appendix A to this evaluation.

The review and evaluation of the proposed design of the facility for a construction permit is only the first stage of a continuing review by the Atomic Energy Commission's regulatory staff of the design, construction, and operating features of the Watts Bar plant. Construction will be accomplished under the surveillance of the Commission's regulatory staff. Prior to issuance of an operating license, we will review the final design to determine that all of the Commission's safety requirements have been met. The facility would then be operated only in accordance with the terms of the operating license and the Commission's regulations under the continued surveillance of the Commission's regulatory staff.

1.2 General Plant Description

The nuclear steam supply system for each Watts Bar unit will consist of a pressurized water reactor and a four-loop reactor coolant system. The reactor core will be composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The fuel tubes will be grouped and supported in assemblies. The reactor core will be initially loaded in three regions, each having a different enrichment of U-235. Water will serve as both the moderator and the coolant and will be circulated through the reactor vessel and core by four coolant pumps. The water, heated by the reactor, will flow through four steam generators where heat will be transferred to the secondary (steam) system. The water will then flow back to the pumps to repeat the cycle. An electrically heated pressurizer 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 nuclear steam supply system for each plant unit will be housed in individual containment structures. The primary containment will consist of a free-standing steel structure with an ice condenser. A separate reinforced reactor shield building will enclose the primary containment. The primary containment, including its penetrations will be designed to safely confine the radioactive material that could be released in the event of an accident. The annulus between the containment and shield building will confine any leakage that might occur from penetrations and through the containment walls. This leakage will be filtered and exhausted to the atmosphere by the emergency gas treatment system.

An auxiliary building, to be located between the two containment structures, will house the radioactive waste treatment facilities, components of engineered safety features, and various related auxiliary systems for each reactor unit. The fuel handling facilities will contain the spent fuel pool and new fuel storage provisions.

The steam and power conversion system for each unit will be designed to remove heat energy from the reactor coolant in the four steam generators and convert it to electrical energy. The waste heat will be rejected to the atmosphere through two natural-draft hyperbolic cooling towers.

The reactor will be controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. The control elements, whose drive shafts penetrate the top head of the reactor vessel, will be moved vertically within the core by individual control rod drives. A reactor protection system will be provided that automatically initiates appropriate action whenever a plant condition monitored by the system approaches pre-established limits. The plant protection system 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.

Redundant and independent standby cooling systems will be provided to maintain reactor cooling and to provide containment cooling in the unlikely event of an accident.

The plant will be capable of being supplied with electrical power from two independent offsite power sources and will be provided with independent and redundant onsite emergency power supplies capable of supplying power to engineered safety features.

1.3 Comparison with Similar Facility Designs

Many features of the design of this plant are similar to those we have evaluated and approved previously for other reactors now under construction or in operation. To the extent feasible and appropriate, we have made use of our previous evaluations to expedite our review of those features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our Safety Evaluation reports for those other facilities have been published and are available for public inspection at the Atomic Energy Commission's Public Document Room.

Each of the proposed units will employ a current-generation 4-loop Westinghouse nuclear steam supply system substantially the same as those approved for the Trojan and McGuire plants and for TVA's Sequoyah facility. In this regard it should be noted that the applicant has made strong efforts, consistent with good engineering practice and a desire to minimize environmental impact, to make the design of Watts Bar conform as closely as possible to that of Sequoyah. It is the official policy of the AEC to encourage such attempts at standardization. The proposed Watts Bar Nuclear Plant resembles the Sequoyah Nuclear Plant in every significant engineering sense important to safety.

1.4 Identification of Agents and Contractors

The Tennessee Valley Authority has engaged the Westinghouse Electric Corporation to design and fabricate two nuclear steam supply systems including the first fuel loading. Westinghouse will also furnish the turbine-generators.

TVA will specify and procure the remaining systems, components and elements of the plant and will design, fabricate and construct the complete integrated plant using these and the Westinghouse-furnished items.

1.5 Summary of Principal Review Matters

This safety evaluation report summarizes the results of our technical evaluation of the information submitted by the applicant with regard to the following principal matters:

1. We evaluated the population density and land use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to establish that these characteristics had been determined adequately and will be given appropriate consideration in the plant final design, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100) taking into consideration the design of the facility including the engineered safety features provided.
2. We evaluated the design, fabrication, construction, and testing criteria, and expected performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Safety Guides, and other appropriate rules, codes and standards, and that any departures from these criteria, codes and standards have been identified and justified.

3. We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered credible. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.
4. We evaluated the applicant's engineering and construction organization, plans for the conduct of plant operations, including the proposed organization, staffing and training program, the plans for industrial security, and the scope of the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, and we have reviewed the application to determine that the applicant is technically qualified to design and construct the plant and has proposed an acceptable organizational structure and plan for safe operation of the plant.

5. We evaluated the design criteria for the systems that will be provided for control of the radiological effluents from the plant to determine that these systems will be capable of controlling the release of radioactive wastes from the facility within the limits of the Commission's regulations and that the equipment to be provided will be capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as practicable.
6. We evaluated the financial structure of the applicant to determine that the applicant's financial resources are adequate to design and construct the Watts Bar Nuclear Plant Units 1 and 2 in accordance with the activities that would be permitted by the construction permit.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

The Watts Bar Nuclear Plant site of approximately 1770 acres in Rhea County, Tennessee, is located on the west bank of the Tennessee River, at Tennessee River mile 528, approximately 50 miles northeast of Chattanooga and 31 miles northeast of the Sequoyah Nuclear Plant site. The site, owned by the United States and in the custody of TVA, is a moderately wooded area with rolling hills whose elevations range from 682.5 feet above mean sea level (MSL) at the water surface to approximately 735 feet above MSL. The site is penetrated by a railroad spur used solely by TVA, Tennessee State Highway Route 68, and a road currently used for access to a small boat launching ramp and camping area.

2.1.2 Population

The distance to the residence nearest the nuclear plant is 4800 feet and, originally, the minimum exclusion distance was approximately 2600 feet. The applicant has now incorporated the width of the Tennessee River and extended the minimum exclusion boundary to 3940 feet (1200 meters). The state highway remains outside the exclusion radius, but this extension will result in the inclusion of the small boat launching ramp and camping area, which are 3100 feet east of the plant location, in the minimum exclusion area. We have notified the applicant that if the camping activity is to continue during plant operation, the applicant must determine accident doses to persons at this location and develop evacuation procedures as a part of the plant emergency plan.

The 1970 census indicated that there were 570 people within the 3-mile low population zone. TVA estimates this will grow to 645 people by the year 2000. The nearest population center with a 1970 population of greater than 25,000 people is Oak Ridge which is 40 miles from the plant site. On the basis of the projected population data supplied by the applicant, it is unlikely that a new population center will develop which would not meet the guidelines for population center distance as stated in § 100.11 of 10 CFR Part 100. Table 2.1.2-1 shows the applicant's cumulative population to 10 miles from the plant for the year 1970 and projected for the year 2000. The applicant has identified the only institution within the low population zone as a school within the 2- to 3-mile zone which had a 1970 enrollment of 175 and has a projected enrollment of 200 for the year 2000.

On the basis of our evaluation of the population data, we conclude that the distances established for the exclusion area, the low population zone, and the population center distance comply with the guidelines given in 10 CFR 100.

2.1.3 Uses of Adjacent Land and Water

The land within a 10-mile radius of the plant site consists of about 65% forested land, 25% non-forested farmland, and 10% used for urban, residential and recreational purposes. About 8% of the total area of the two counties in which this land is located is innundated by the Watts Bar and Chickamauga Reservoirs which are used for recreational purposes and potable and industrial water supplies.

The only industrial facilities within 5 miles of the site are the Watts Bar Steam Plant and the Watts Bar Hydroelectric Plant, which are respectively, about 0.65 and 1.9 miles from the plant. Transportation facilities are limited mainly to barge and highway traffic and possible use of the railroad spur to the Watts Bar Steam Plant. The applicant has evaluated the threat to the facility posed by the shipment of explosives by barge, rail, and truck. Each of these has been eliminated as a credible hazard by virtue of regulations (e.g., Department of Transportation, Interstate Commerce Commission) regarding shipment sizes for each mode of transportation and the distances separating such shipments from the facility.

TABLE 2.1.2-1

WATTS BAR POPULATION DATA

Distance from the Site, Miles	Cumulative Population	
	1970	2000
1	60	35
2	210	230
3	570	645
4	1185	1345
5	1805	2010
10	10515	11995

2.2 Meteorology

The plant will be situated on the west shore of Chickamauga Lake on the Tennessee River near the center of a northeast-southwest aligned valley, 10 to 15 miles wide, with ridges to 1,800 feet MSL on the valley's west side and a series of ridges to 1,000 feet MSL on the east side. The region is dominated much of the year by the Bermuda anticyclonic circulation system which produces extended periods of fair weather and widespread atmospheric stagnation. Therefore the average wind speed is low and the wind direction is influenced primarily by the valley orientation.

The accident and annual average diffusion conditions expected for the plant site have been evaluated from measurements of wind direction, wind speed and vertical temperature difference (ΔT) on a 130-foot tower at a temporary onsite location. Wind direction, wind speed and temperatures have been measured by sensors at the 30-and 130-foot levels of this tower since June 23, 1971. The applicant has presented 10 months of hourly data as joint frequency distributions of wind speed and wind direction at the 30-foot level by atmospheric stability determined from classes of ΔT data. Data recovery for this period was 91%. It should be noted that the 10 months of onsite data show stable atmospheric conditions ($\Delta T \geq -5°C/100m$) existing 77% of the time and that the average wind speed at the 30-foot level is 1.6 meters/second. This average wind speed is less than half of the average wind speeds reported for other stations in the region.

Since the accidental and routine releases of effluents to the atmosphere will be either through vents near building rooftop level or from the buildings themselves, we have used the diffusion equations for ground level sources for estimating relative concentration[1] values. The joint frequency distributions

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[1] X/Q : Where X is the short term average centerline value of the ground level concentration, and Q is the amount of material released.

for the 10-month period of record were used as input data for providing the appropriate relative concentrations.

In evaluation of diffusion of short-term accidental releases from the plant, a ground-level release with a building wake factor, cA, of $815m^2$ was assumed. The relative concentration which is exceeded 5% of the time was calculated to be 3.4×10^{-3} sec/m^3 at the minimum site boundary distance of 1200m. This relative concentration is equivalent to dispersion conditions produced by extremely stable atmospheric conditions accompanied by a wind speed of 0.2 meters/second. The applicant has used a value which is in essential agreement with our value. For longer time period accidental releases, we estimate that the relative concentrations presented in Safety Guide No. 4 should be increased by a factor of five to assure that adequately conservative accident dose estimates are obtained at the outer boundary of the low population zone (4827m). A limiting annual average relative concentration estimate of 2.6×10^{-5} sec/m^3 was found at the 1200m site boundary southeast of the plant. This value is about a factor of two higher than the one calculated by the applicant.

Our consultant, the National Oceanic and Atmospheric Administration (NOAA) has independently calculated concentrations for accidental and annual average releases which are in substantial agreement with our values. Our consultant's report is attached as Appendix B.

2.3 Hydrology

2.3.1 Hydrologic Description

The site is on the west bank of the Tennessee River about 1.9 miles south southwest of Watts Bar Dam, about 0.65 miles southwest of Watts Bar Steam Plant, and along the upper reaches of the Chickamauga Reservoir. Water supply is to be taken from the Tennessee River for cooling tower makeup at about 133 cubic feet per second (cfs). The Sequoyah Nuclear Plant is also on the banks of Chickamauga Reservoir, 44 miles downstream of the Watts Bar site. There are 11 major TVA and six Aluminum Company of America dams upstream of the site.

Plant grade is proposed at elevation 728 feet MSL. The Watts Bar Dam just upstream from the nuclear plant has a normal reservoir elevation of 741 feet MSL and a nominal top-of-dam at elevation 752 feet MSL. The main dam is a combination concrete-earth structure. The concrete powerhouse, spillway and navigation lock span about 1638 feet of the river valley, and the earth section is over 1200 feet long on the east side of the river. Another earth embankment, 2.5 miles west of the main structure, closes a low point in the rim of the reservoir. The project was authorized in 1940 and first used under emergency wartime conditions in 1942. Chickamauga Dam, 57 miles downstream, maintains a normal reservoir level of 682.5 feet MSL. A minimum Tennessee River channel depth of 9 feet is maintained in the area for navigation.

Four major existing (and one proposed) public water supplies are taken from Watts Bar or Chickamauga Reservoirs, and 17 public and 7 industrial ground water supply users are located within 20 miles of the site. Springs and shallow wells in the site area are known to supply local domestic water users.

2.3.2 Floods

The greatest flood of record occurred in March, 1867 (before dam construction) and reached an estimated level of elevation 716 feet MSL at the site. The 17 major dams upstream of the site provide some flood control capability for all floods approaching the severity of a probably maximum flood (PMF). The applicant has proposed constructing most of the plant facilities above all but the more severe flood levels, and has provided appropriate design bases and emergency shutdown procedures for the more severe floods.

2.3.3 Probable Maximum Floods

The applicant has estimated a probable maximum flood (PMF) having a peak flow rate at the site of 1,225,000 cubic feet per second which would reach a relatively steady water surface elevation of approximately 737.5 feet MSL. The evaluation is based on the estimated probable maximum precipitation for the region as determined by the Hydrometeorological Branch of the Weather Bureau, and the suggested rainfall has been applied to a verified runoff model of the basin. The analysis is complicated by the conclusion that dams both upstream and downstream of the site would be incapable of safely passing such a severe flood and could fail. The analysis of the flood, and potential upstream and downstream dam failures, indicates that the PMF at the site would have two important peak flow and water level conditions. The first would be caused by the surge of flood waters as a result of the potential rapid failure of the eastern Watts Bar Dam embankment. The second, of approximately the same flow rate at the site, would result from the maximum upstream runoff pouring over, through, and around what would be left of the dam. The applicant has also determined that a failure of upstream Fort Loudoun Dam would contribute to the flood, but that the potential failure of downstream (57 miles) Chickamauga Dam might reduce the flood level at the site by only 0.5 feet.

The first major flood peak, that associated with the rapid failure of Watts Bar Dam, might be attended by a wave front analogous to a hydraulic bore. TVA considers that such a wave would strike the ridge, or valley wall, on the other side of the river and might be reflected across the stream. The applicant's analysis indicates a bore as high as 2 feet could approach the site at a velocity of 30 feet per second, and has proposed that such a wave become part of the design bases for safety-related facilities. From the applicant's analysis this wave could occur in the final stages of embankment failure with a reservoir-tailwater level difference of 32.4 feet; that is, the water level behind the dam would be 32.4 feet greater than downstream. The applicant estimates that this last stage of embankment failure also leads to an increase in river flow of over 350,000 cubic feet per second which results in a very rapid rise in river level at the plant site of approximately 7 feet. The applicant has agreed to study the possibility of a bore accompanying this rapid rise in level and should it materialize, include it as well as the bore mentioned above in the design bases for safety-related plant features.

The applicant has analyzed the effects of wind-generated wave activity which might occur coincidentally with a maximum PMF water level. Originally, TVA chose an over-water wind speed of 32 miles per hour from a critical direction with respect to the plant, and estimated runup to elevation 741.7 feet MSL on the south walls of the diesel generator building and the pumping station. The applicant selected a 32-mile per hour wind speed based on a coincident flood-wind probability of

2×10^{-5}. Wind speeds were based upon an analysis of recorded wind speeds in the general plant region, and the assumption was made that the PMF might occur in March once during the anticipated 40-year life of the project.

This approach was not acceptable to us. We suggested, after the fashion of the Corps of Engineers (the developers of the PMF as a design basis), that it would be more appropriate to assume a wind speed of 45 miles per hour. Accordingly, the applicant has now agreed to protect safety-related structures and equipment to elevation 743.5 feet MSL to account for the combined effects of the PMF and a 45-mile per hour over-water wind from directions critical to the plant.

The applicant has provided a summary of an analysis of local drainage which indicates that acceptable provisions will be made to prevent a local probable maximum flood from reaching the critical safety-related plant grade elevation of 729 feet MSL.

2.3.4 Potential Dam Failures (Seismically-Induced)

The applicant has presented a summary of the analysis of the capability of the Watts Bar Dam to sustain severe earthquakes and has concluded that the dam is capable of sustaining a safe shutdown earthquake (SSE) without failure. In addition, the applicant has concluded that even the instantaneous removal of the dam would not cause water levels at the site approaching plant grade, and that the only safety-related structure which could be affected (the intake pumping station) would be capable of withstanding such an event. The applicant is continuing the investigation of the potential flood which could result from the possible seismically-induced failure of dams upstream of Watts Bar and has committed to document this study this year. It is not unlikely that the result of this study will indicate that additional flood protection measures will be required. Should this study identify seismic-related floods for which the present PMF emergency provisions are not adequate, TVA has proposed three alternative means of protecting the plant. The three means are (1) to construct a dike or levee around the plant, (2) raise the structures and plant grade, or (3) seal safety-related structures below potential flood levels. Our evaluation indicates that these means are practical solutions and, if needed, we will require the applicant to submit a design of the scheme selected for our review prior to issuance of the construction permit.

2.3.5 Low Water Considerations

Cooling water is to be provided via closed-loop cooling towers with makeup water supplied from an intake pumping station set into the river bank. The estimated minimum water requirement from the Tennessee River - Chickamauga Reservoir source is 67 cubic feet per second. The reservoir control exerted by the applicant on low-flow adjacent to the plant will assure a normal flow of more than 2000 cubic feet per second. In the event the reservoirs cannot be counted on to assure the minimum 2000 cubic feet per second, the applicant believes minimum natural flow should be sufficient to assure a dependable water supply. This view is reinforced by the recorded pre-dam construction minimum flow of 2600 cubic feet per second. The minimum controlled flow of 2000 cubic feet per second will provide a minimum depth of 5.9 feet in the intake channel and approximately 10 feet in the river. Protection of the channel to the intake structure from the river against sudden flood-produced sediment deposition will be provided by the adjacent Watts Bar Dam. The applicant has also stated that the intake channel will be monitored periodically to assure that the channel will not be silted gradually.

We agree that sufficient water supply will be available from the adjacent Tennessee River under all conditions. The applicant maintains control of substantial natural river flows and levels through an extensive reservoir system and natural runoff should be adequate for any situation when such control could not be exercised.

2.3.6 Ground Water

Local ground water is mined from an interbedded, folded, and contorted formation of limestone and shale which is labeled the Conasauga formation. The primary rock type at the site is shale, and the general slope of the formation is toward the adjacent river. Rock outcrops and springs occur in the shallow soil deposits in the site vicinity.

The applicant has reported approximately 55 wells and 5 springs within 2 miles of the site. Most of the wells are low yield at depths ranging from about 6 to 257 feet. There are 17 public and 5 industrial water supplies taken from wells within 20 miles of the site. One public and one industrial supply are within 2 miles of the site, but both are upgradient. The applicant has estimated the range of permeability of the surficial materials to be between 10^{-3} and 10^{-6} centimeters per second.

We have concluded that the location of ground water users with respect to the plant, and the hydrologic characteristics of the local ground water environment are such as to make it unlikely that any well can be contaminated as the result of liquid radioactive releases from the plant.

2.3.7 Emergency Operation Requirements

The applicant has proposed shutting down the plant and flooding the auxiliary building to prevent uplifting in the event any storm-related flood occurs that would exceed plant grade. As in the case of Sequoyah, an alternate decay heat removal method will be provided to cope with this situation. This method is discussed in Section 8.2.1.

2.4 Geology, Seismology, and Foundation Engineering

2.4.1 Geology

The proposed site is in the Valley and Ridge Province of the Appalachian Mountains. The geologic setting of the plant site is similar to that of the Sequoyah plant site.

A dominant feature of the geologic structure of the region is the Kingston thrust fault which trends northeasterly about 1 mile northwest of the site. The fault dips to the southeast in the direction of the plant site, and underlies the site at a depth of about 2000 to 3000 feet. There is no indication of faulting or structural activity in the region since Paleozoic time.

Our U.S. Geologic Survey consultant concludes and the staff agrees that there are no active faults or other geologic structures in the area that are thought potentially capable of localizing seismicity in the vicinity of the site. The earthquakes that have occurred in the region cannot be related directly to any faults in the area. Consequently, we assumed that the largest earthquake previously experienced in the region might also occur again anywhere in the region.

Our consultant's report is attached as Appendix C.

2.4.2 Seismology

Our NOAA consultant states that the largest historic earthquake that occurred in the region was the Giles County earthquake of May 31, 1897. This event is listed as being of Intensity VIII.

Also, there have been three Intensity VII events recorded in the region. It is believed that events such as these present the greatest earthquake hazard to the proposed plant.

We and our consultant agree with the applicant that an acceleration of 0.09g, resulting from an Intensity VII event, would be adequate to represent ground motions resulting from the operating basis earthquake; and an acceleration of 0.18g, resulting from an Intensity VIII event, would be adequate to represent ground motions resulting from the safe shutdown earthquake.

Our consultant's report is attached as Appendix D.

2.4.3 Foundation Engineering

The site area includes unconsolidated river terrace deposits, averaging approximately 40 feet in thickness, overlying the Conasauga formation of Cambrian age which composes the bedrock.

The applicant is continuing its investigations to completely define the river terrace deposits at the site. Soils exploration and laboratory testing programs are currently underway. The information obtained from these programs will enable the applicant to complete slope stability and soils liquefaction analyses for the site and confirm the acceptability of the presently proposed slope designs. Further, it will enable the applicant to define the foundation design for the diesel generator building, the only Category I structure that will not be founded on unweathered Conasauga shale. We have concluded that the applicant's continuing investigations are sufficient in nature and scope to confirm the acceptability of the foundation provisions to be made for the facility or establish modifications to these provisions that will make them acceptable. We have informed the applicant that we will require the results of these investigations and related facility design information to be submitted to us for our review and approval prior to issuance of the construction permit.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with AEC General Design Criteria

The applicant has stated that the Watts Bar Nuclear Plant will be designed, constructed, and operated in accordance with the Commission's General Design Criteria for Nuclear Power Plants of July 7, 1971. Detailed evaluations of each system for compliance with the appropriate criteria are presented in the PSAR.

We find that the proposed Watts Bar design meets the intent of the General Design Criteria.

3.2 System Quality Group Classifications

The applicant has applied the ANS system of safety classes to those water and steam-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety. ANS Safety Classes 1, 2a and 3 generally correspond to Quality Groups A, B and C in Safety Guide 26. In addition, the ANS system has a Safety Class 2b which is based on those component codes within Group C and the Quality Assurance (administration-management and documentation) requirements normally associated with components of Quality Group B. In Appendix B to the PSAR, the applicant has identified for the respective safety classes, the applicable specific codes and standards for system components.

For those fluid systems identified in Safety Guide 26, we and the applicant are in general agreement on the application of the quality group classification system. The applicant has supplied piping and instrumentation diagrams identifying the boundary limits of each classification group within those fluid systems identified in Safety Guide 26.

We find that the system quality group classifications as specified by the applicant are acceptable.

3.3 Wind and Tornado Design Criteria

The wind speed used for the design of essential plant structures will be 95 mph. Wind pressure, shape factors, gust factors, and variation of winds with height will be determined in accordance with ASCE Paper 3269, "Wind Forces on Structures."

Tornado loadings will consist of a pressure drop of 3 pounds per square inch in 3 seconds, and a lateral force caused by a funnel of wind having a 300 mile per hour radial velocity plus a 60 mile per hour translational velocity.

These criteria are acceptable to us.

3.4 Water Level (Flood) Design Criteria

A discussion of flooding criteria and design bases was presented in Section 2.4.

3.5 Missile Protection Criteria

The design of essential structures and vital equipment will consider the effects of a spectrum of tornado-borne missiles. Also, internally generated missiles associated with component overspeed failures and missiles which could originate from high-pressure system ruptures will be considered. The design will assure that no loss of essential function can occur.

We find these criteria to be acceptable.

3.6 Criteria for Protection Against Dynamic Effects Associated With a Loss-of-Coolant Accident

The applicant has stated that engineered safety feature systems and components located within the containment vessel will be protected from the dynamic effects resulting from credible piping failures to the extent that:

(a) the reactor will be shutdown.

(b) minimum ECCS requirements will be satisfied.

(c) minimum performance requirements of other engineered safeguards will be satisfied.

(d) containment vessel integrity will be maintained.

(e) maximum break size and type will not exceed the design basis of the engineered safeguard systems.

With respect to protection against pipe whip, the applicant has stated that a low stress relative to the maximum allowable stress for the material in question below which a break would be highly improbable and below which a crack would have no potential to propagate will be established. Breaks in the pipe will then be postulated at points with stress intensity greater than the level thus established. In those instances where piping failures and/or their effects violate any of the requirements in the first paragraph above, protection requirements will be established. We expect that the applicant will establish the exact stress levels which will be used for locating potential pipe breaks within the next 6 months. We have asked that these stress levels be submitted when they are available.

We find these criteria to be acceptable for the construction permit stage.

3.7 Seismic Design

We have been assisted in our evaluation of seismic design by Nathan M. Newmark, Consulting Engineering Services. Our consultant has reviewed the Watts Bar PSAR including applicable amendments and finds the seismic design criteria for structures, systems, and components documented therein to be acceptable.

Our consultant's report is attached as Appendix E.

3.7.1 Seismic Input

The seismic design response spectra as modified by Amendment 6, produce amplification factors of 3.5 between the period range of 0.15 to 0.5 seconds and of greater than 1 in the period range 0.15 to 0.033 seconds for 2% damping. The structure and equipment damping is in accordance with the damping factors which have been accepted for all recently licensed plants. The modified time history to be used for component equipment design is adjusted in amplitude and frequency to envelope the response spectra specified for the site. We and our seismic consultants conclude that the seismic input criteria proposed by the applicant provide an acceptable basis for seismic design.

3.7.2 Seismic System Analyses

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods will be used for all major Category I structures, systems, and components. Governing response parameters will be combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. The absolute sum of responses is used for in-phase closely-spaced frequencies. Floor spectra inputs to be used for design and test verification of structures, systems and components are generated from the normal mode-time history method. A vertical seismic-system dynamic analysis will be employed for all structures, systems and components. We and our consultants conclude that the seismic system dynamic methods and procedures proposed by the applicant provide an acceptable basis for the seismic design.

3.7.3 Seismic Instrumentation

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structures correspond to the recommendations of Safety Guide 12.

Supporting instrumentation will be installed on Category I structures, systems, and components in order to provide data for the verification of the seismic responses determined analytically for such Category I items.

A plan for the utilization of the acquired seismic data will be submitted for the FSAR review.

3.7.4 Seismic Design Control Measures

The quality assurance requirements for Category I structures, systems, and components are stated in Amendment 5 to the PSAR. We believe that these quality assurance provisions which will be implemented for all items designated as Category I for design, comply with the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants" of 10 CFR Part 50.

3.8 Design of Category I Structures

3.8.1 Structural Foundations and Concrete Supports

The containment and the surrounding shield building will be supported on a common reinforced concrete mat foundation of approximately 132 feet in diameter which will bear on rock materials. The other Category I structures that will be supported on mat type foundations bearing on rock material are the auxiliary building and the intake pumping station.

The various design parameters for the underlying materials such as the modulus of elasticity and allowable bearing stress values have been evaluated by the applicant based on analytical and empirical results as well as observations from other structures supported by the rock materials of the site. This information was used to determine allowable bearing capacities and the settlement criteria to be used in the foundation design.

The designs for the foundations will be based on the tolerable differential settlements since the factor of safety on bearing failure for the worst case in a Category I rock supported structure is approximately 5.0. The foundations will be designed to behave independently under the specified loads and accommodate 1-inch differential settlements. The analysis and design for these reinforced concrete mats will be executed on the basis of an elastic plate on an elastic foundation and the materials proportioned on the basis of a working stress design. The stress allowables for the concrete and reinforcing steel have been judged to be acceptable.

The only Category I structure supported on soil will be the diesel generator building which will be on a 20-foot layer of soil above bedrock.

The concrete supports such as the primary shield of the reactor vessel, will be analyzed and designed on an elastic basis as Category I structures. The design will use the stress allowable values specified in the PSAR for the various combinations of loads. The design criteria including the loads, load combination and stresses as presented in the PSAR for the structural foundations and concrete supports are consistent with the criteria being utilized for other nuclear facilities and are judged to be adequate. The analysis and design procedures presented in the document are acceptable and if followed should result in a safe facility.

3.8.2 Containment Base Slab Liner and Internal Structures

The bottom liner plate will be constructed from plate conforming to the requirements of SA 516, Grade 70. Seam welds will be checked by either dye penetrant or magnetic particle examination in accordance with Appendix VIII and Appendix VI of the ASME Code. Welds at seams will also be 100% vacuum box tested and 100% visually inspected and also have leak chase channels welded on them prior to concrete floor placement. The liner will be welded to embedded structural steel that meets ASTM A-36 specifications. These criteria meet the requirements of AEC Safety Guide 19.

The critical internal structure is the divider barrier which is essential for the containment function. (See Section 5.2 for a description of the containment system.) The divider barrier separates the upper and lower compartments of the containment and ensures that the steam from a ruptured coolant loop is directed into the bottom of the ice condenser. The barrier structure is composed of a series of slabs and walls arranged to fit the layout of the major equipment within the containment. The various sections and compartment pressures have been provided in diagram form to illustrate the pressure loads imposed on the various parts of the internal structures. The various loading combinations that represent the hypothesized worst conditions have also been provided.

The divider barrier will be designed by the working stress method of ACI 318-63 for the various combinations of loads including accident pressure and temperature loads and the earthquake loads. The differential pressure loads range from about 10 psig to 19 psig in the areas away from the immediate vicinity of the reactor vessel. Localized plastic action will be utilized for resisting the effects of fluid jet loads, pipe whip loads and missile loads. The analytical methods include the use of ICES-STRUDL-II, GENDEK and a program developed by TVA engineers for thermal stress analysis. The divider barrier design will be subjected to an independent analysis by a group within TVA that is separate from the group having primary design responsibility.

The three compartments in the immediate vicinity of the reactor are designed in the same manner for loadings of differential pressure. The three areas are designed for the following pressures; the sump pit - 23 psig, the vessel annulus - 100 psig and, the above-vessel compartment - 30 psig.

The other major internal structure consists of the system utilized to support the ice condenser ice beds. The system of piers, pedestals, columns, beams, and slabs utilize ACI 318-63 and the AISC 1969 specifications for their design.

The fuel transfer facility's structures are designated as Category I structures and those internal to the shield building are protected from the effects of winds, tornadoes, and the related missiles by the shield building. The polar crane of the containment building is provided with rail yokes to prevent dislodgement from the rails. These provisions satisfy the portions of Items 1, 2, and 3 of Safety Guide 13 (Fuel Storage Facility Design Basis) that relate to structural engineering.

It is concluded that the analysis and design criteria for the conditions specified are adequate. Execution in accordance with the cited references should result in structures that are safe for their intended use. The appropriate Safety Guides related to structural engineering have also been met.

3.8.3 Other Category I Structures

The other Category I structures include (1) the shield building, (2) the auxiliary building, (3) the diesel generator building, and (4) the intake pumping station.

The shield building will be a reinforced concrete structure, the geometry of which will be a right cylinder with a shallow domed roof. The wall thickness will be 3 feet and the dome will be 2 feet thick. The structure will be designed to resist the loads resulting from dead load, snow load, wind load, tornado load, uplift forces, water pressure, earth pressures, missile loads, seismic loads, and the design basis accident loads. Various load combinations will be used in order to proportion the shell structure on the basis of an allowable stress procedure. Under certain combinations that include the design basis earthquake, the concrete compressive stress will be allowed to reach $0.75\ f_c'$ and the reinforcing steel tensile stress will be allowed to reach $0.90\ f_y$. The design will be a duplicate of the Sequoyah plant shield building which is based on ACI 318-63. If Cadweld splices are used the applicant's proposed program of testing will be in accordance with AEC Safety Guide No. 10. User testing of reinforcing steel will be in accordance with AEC Safety Guide No. 15. The proposed program for the control and testing of concrete is acceptable. The applicant will sample every 175 cubic yards of concrete when the required strengths are 3000 psi or greater.

The auxiliary building is primarily a reinforced concrete structure designed in accordance with ACI 318-63 as a duplicate of the Sequoyah plant. This structure will house the spent fuel storage pit and will meet the structural criteria set forth in AEC Safety Guide No. 13.

The diesel generator building and the intake pumping station will be constructed of reinforced concrete and will use new designs based on ACI 318-71.

The criteria related to structural engineering that have been provided for the other Category I structures are judged to be adequate to design and construct the Watts Bar facility.

3.8.4 Metal Containment System - Design

The metal containment system, which includes the containment vessel, penetration assemblies, and access openings, is a low leakage steel shell which will be designed to sustain the combination of loads resulting from the loss-of-coolant accident, the operational basis earthquake, and the conventional live and dead loads within the stress limits defined in Subsection B of the ASME Section III Nuclear Vessels Code for the normal and upset operating condition categories. For the combination of loadings which include those calculated to result from the loss-of-coolant accident and the design basis earthquake, the functional integrity of the metal containment system will be assured by design within the stress limits for the emergency operating condition category of the specified code. We find the design stress limits for the metal containment system to be acceptable.

The containment will be designed for an external design pressure of 0.5 psig. Automatic vacuum relief devices will be used to prevent the containment vessel from being subjected to an external pressure in excess of design requirements.

Containment "hot" piping penetrations will utilize a multiple flued fitting to accommodate the use of a guard pipe concentric to the process line (e.g., steam piping) in the shield building annulus. The guard pipe design will be subjected to an independent analysis by a group within the TVA organization that is separate from the group having the primary design responsibility. The guard pipe will protect the bellows expansion joint and maintain the penetration seal in the event of a rupture of a process line within the annulus between the containment vessel and shield building.

Pneumatic overpressure testing of the containment system will be in accordance with the applicable code requirements. All weld seams and gaskets, including both doors of the personnel air lock will be soap-bubble tested. These leakage tests will be conducted with the containment vessel pressurized to 5 psig and again at the maximum containment internal pressure of 15 psig upon completion of the pneumatic overpressure test at 16.9 psig.

The structural acceptance testing proposed by the applicant for the metal containment system is acceptable.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

The applicant has designated Indian Point Unit No. 2 as the prototype for Westinghouse four-loop plants from which vibration test data is applicable in evaluating the adequacy of the Watts Bar Nuclear Plant, Units 1 and 2 reactor internals to withstand flow induced vibration effects. The hot functional testing period on the prototype plant has been completed and a topical report documenting these tests was recently submitted. Based on a preliminary evaluation of the test data, the tests appear to satisfy the requirements of Safety Guide 20, Vibration Measurements on Reactor Internals. The applicant is aware that, in the highly unlikely event that a prototype is not established for the Watts Bar plant, a complete vibrations test program for a non-prototype plant will need to be performed for the Watts Bar facility. The design does not preclude the performance of such a program.

If, as expected, an acceptable prototype plant is established a program of preoperational functional vibration tests will be conducted in order to subject the the Watts Bar Nuclear Plant reactor internals to all significant flow modes expected during power operation. These tests will be conducted under the same test conditions that were imposed on the prototype design. Subsequent rigorous inspection will confirm the structural integrity of the Watts Bar reactor internals from the standpoint of vibration. We find the planned program of tests and inspection for the Watts Bar Nuclear Plant to be acceptable.

The reactor internals of the Watts Bar Nuclear Plant will be analyzed to determine the effects of postulated accidents, the design basis earthquake and the loads which would result from the concurrent occurrence of these events. The applicant has referenced Topical Report WCAP-7332-L, Indian Point No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation as the LOCA analytical study. This matter is currently undergoing generic evaluation by the regulatory staff. We will notify the applicant of our findings when this review has been completed.

3.9.2 ASME Code Class 2 and 3 Components

All seismic Category I components, equipment and systems in ASME Code Class 2 and 3 and outside of the reactor coolant pressure boundary, will be designed, fabricated and inspected in accordance with the requirements of the applicable codes mentioned in Section 3.2, System Quality Group Classification.

They will be designed to sustain normal loads, anticipated transients and the operational basis earthquake within the appropriate code allowable stress limits and the design basis earthquake within stress limits which are comparable to those associated with the emergency operating condition category. We consider that these stress criteria provide an adequate margin of safety for seismic Category I systems, components, and equipment.

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

The reactor protection system, engineered safety feature circuits and the emergency power system are designed to meet Category I design criteria. The seismic requirements established by the seismic system analysis will be incorporated into the equipment specifications to insure that the equipment purchased or designed will meet seismic requirements equal to or in excess of the requirement for Category I components.

We evaluated Topical Report WCAP-7397-L, Seismic Testing of Electrical and Control Equipment, referenced in this application. We have concluded that the seismic tests described therein are suitable for demonstrating the seismic resistance of the essential electrical and electronic equipment provided for Watts Bar.

3.11 Environmental Design of Mechanical and Electrical Equipment

The applicant has stated that all engineered safety feature motors, cables, and instruments located inside containment which must operate during or following a loss-of-coolant accident will be capable of functioning under the post-accident temperature, pressure, and humidity conditions for the time periods required. This capability has been demonstrated by testing and has been documented in Topical Report WCAP-7744, Environmental Testing of Engineered Safety Feature Related Equipment. We conclude that this is acceptable.

In addition, the solenoid valves used within containment as isolation valve pilots will be qualified to survive the accident environment. The containment air return fans will meet IEEE-334 requirements and be qualified to operate in the accident environment. The applicant has also stated that cable splices and terminations will be qualified.

Engineered safety feature electrical equipment and instrumentation located in containment will be fabricated of material having a threshold for radiation damage higher than the postulated sum of the accumulated long-term and accident doses. Equipment and instrumentation located outside of containment in areas of lower accident doses will also be qualified to perform their function in the postulated environment.

The applicant has committed to satify the requirements of IEEE 317, 1971 for qualification of containment electrical penetrations.

We have concluded that the environmental test program will provide acceptable means of assuring that equipment and systems required to be operable following an accident will be qualified.

4.0 REACTOR AND REACTOR COOLANT SYSTEM

4.1 Summary Description

The nuclear and thermal-hydraulic design of the Watts Bar nuclear steam supply systems is the same as for a number of previously reviewed and approved Westinghouse 4-loop PWR systems namely, those for the Sequoyah, McGuire and Trojan facilities. On the basis of these earlier reviews, we have concluded that the Watts Bar design is acceptable. However, consistent with the approach we followed in our evaluation of the proposed thermal performance changes for the Trojan and McGuire plants, we intend to limit the core thermal parameters to those approved for the Sequoyah core until additional evidence from tests conducted on reactors of similar design is provided to verify the conservatism of the proposed increase in core thermal performance for Watts Bar. We anticipate that the results of these tests will be available prior to operation of the Watts Bar plant.

Our review of Watts Bar has stressed certain important mechanical design and fabrication aspects of the system as well as testing, surveillance, and inspection programs.

4.2 Integrity of Reactor Coolant Pressure Boundary

4.2.1 Design Criteria, Methods, and Procedures

The reactor coolant pressure boundary will be a seismic Category I system designed, fabricated, and inspected in accordance with the requirements of the applicable codes discussed in

Section 3.2. The applicable codes and code editions comply with the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards. The stress limit criteria specified for the normal and upset operating condition categories of the applicable codes will apply for all normal loads and anticipated transients including the operational basis earthquake.

Under the loads calculated to result from the design basis accident, the design basis earthquake, and the combination of these postulated events, the components of the reactor coolant pressure boundary will be designed to the applicable emergency and faulted operating condition category limits of the appropriate codes or where the appropriate codes do not provide explicit design limits for these operating condition categories, to the criteria submitted in Appendix B of the PSAR. The plastic instability limits allowed by NB-3200 of the Code will not be employed for pumps and valves under any loading conditions. In addition, active components, i.e., pumps and valves required to operate reliably in order to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a pipe break will be designed to deformation limits that are consistent with operational requirements. Under these restrictive deformation criteria, calculated primary stresses will be in the elastic range. We find the above stress and deformation criteria acceptable.

In accordance with Paragraph I.701.5.4 of the ANSI B31.7 Nuclear Power Piping Code, which requires that piping shall be supported to minimize vibration and that the designer is responsible to observe that vibration is within acceptable levels, a vibration operational test program to verify that the piping and piping restraints within the RCPB have been designed to withstand dynamic effects due to valve closures, pump trips, etc., will be performed during startup and initial operating conditions. The proposed tests and the associated actions e.g., pump trips, valve actuations, that are to be used in this program will be similar to those experienced during reactor operation and provide an acceptable basis for conducting the vibration operational test program.

4.2.2 Material Considerations

4.2.2.1 Fracture Toughness

We have reviewed materials testing and operating limitations proposed by the applicant to assure that ferritic materials of pressure-retaining components of the reactor coolant pressure boundary will exhibit adequate fracture toughness under normal reactor operating conditions, sytem hydrostatic tests, and during transient conditions to which the system may be subjected.

The applicant has stated that acceptance testing for ferritic materials will be performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1968 edition). Dropweight NDT data as well as Charpy V-notch energy curves will be obtained for the plates and major forgings in the reactor vessel.

To establish operating pressure and temperature limitations during startup and shutdown of the reactor coolant system, the applicant has agreed to follow Appendix G, Protection Against Non-Ductile Failure, of the recently revised ASME Code, Section III, fracture toughness rules (Code Case 1514). The applicant will submit specific operating limitation curves at the operating license stage.

We conclude that the selected materials and planned operation of the reactor coolant system will assure adequate margins of safety with respect to fracture toughness considerations.

4.2.2.2 <u>Sensitized Stainless Steel</u>

Stainless steel that has been sensitized has an increased susceptibility to stress corrosion cracking. The applicant has stated that significant sensitization of all non-stabilized austenitic stainless steel within the reactor coolant pressure boundary will be avoided through materials selection and control of all welding and heat treating processes. The precautions will include control of preheat and interpass temperatures and control of heat input during the welding operations.

Stainless steel components and piping will be joined to the reactor vessel ferritic steel nozzles by buttering the ferritic steel with Inconel, prior to post-welded heat treatment, and by later shop-welding an annealed stainless steel safe-end to the Inconel buildup using Inconel filler metal.

We conclude that the planning to avoid sensitization of austenitic stainless steel during the fabrication period is acceptable.

4.2.3 Leakage Detection System

Coolant leakage within the reactor containment may be an indication of a small through-wall flaw in the reactor coolant boundary.

The leakage detection system proposed for the reactor coolant pressure boundary will include diverse leak detection methods, will have sufficient sensitivity to measure small leaks, and will be provided with suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate and gaseous radioactivity monitors, and level indicators on the containment sump. Indirect indication of leakage can be obtained form the containment humidity, pressure and temperature indicators. We conclude that the proposed leakage detection system will have the capability to detect small through-wall flaws in the reactor coolant pressure boundary.

4.2.4 Inservice Inspection Program - Primary System

Selected welds and weld heat-affected zones must be inspected periodically to assure continued integrity of the reactor coolant pressure boundary during the service lifetime of the plant.

The applicant has stated that the inservice inspection program for the reactor coolant pressure boundary will comply with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for In-Service Inspection of Reactor Coolant Systems" 1970 edition. Access for inservice inspection is being considered in the design and arrangement of pressure-containing components.

The reactor vessel will be examined from the inside with a remotely operable reactor vessel inspection tool capable of performing inspections of the circumferential, longitudinal and nozzle welds. Collection of data during inservice and preservice inspections will be by an electronic system.

The structural integrity of the reactor coolant system boundary is to be maintained at the level of the original acceptance standards.

We conclude that the access provisions and planning for inservice inspection are acceptable.

4.2.5 Inservice Inspection Program - Other Category I Systems

The applicant is planning access to the Group B and C fluid systems such as the engineered safety systems, reactor shutdown systems, cooling water systems and the radioactive waste treatment systems outside the limits of the reactor coolant pressure boundary for inservice inspection. We conclude that the planning for an inservice inspection program for the Group B and C fluid systems is adequate.

4.3 Reactor Vessel Material Surveillance Program

A material surveillance program is required to monitor changes in the fracture toughness properties of the reactor vessel material as a result of neutron irradiation.

The applicant has stated in the PSAR that the material surveillance program will comply with the proposed AEC §50.55a Appendix H, Reactor Vessel Material Surveillance Program Requirements, and ASTM E-185-70. The program specification is acceptable with respect to the number of capsules, number and type of specimens, withdrawal schedule, and retention of archive material. We conclude that the proposed program will adequately monitor neutron radiation induced changes in the fracture toughness of the reactor vessel beltline material.

4.4 Reactor Vessel Internals (Mechanical Design)

For normal design loads of mechanical, hydraulic, and thermal origin, including anticipated plant transients and the operational basis earthquake, the reactor internals will be designed to the stress limit criteria of Article 4 of the ASME Boiler and Pressure Vessel Code Section III.

Under design basis accident conditions, which include the combined loads from a recirculation line break or a steam line break plus the design basis earthquake, the reactor internal components will be designed to the criteria submitted in Section 14 of the PSAR. These criteria are consistent with comparable code emergency and faulted operating condition category limits and the criteria which have been accepted for all recently licensed plants. We find these criteria acceptable. The dynamic analyses of the Watts Bar nuclear reactor internals were discussed in Section 3.9.1 Dynamic System Analysis and Testing.

4.5 Pump Flywheel Integrity

The probability of a loss of pump flywheel integrity, which could result in high energy missiles and excessive vibration of the reactor coolant pump assembly, can be minimized by the use of suitable material, adequate design and inservice inspection.

The applicant has stated that the specifications for the design, fabrication, and inspection of the pump flywheels are in general accord with AEC Safety Guide 14, Reactor Coolant Pump Flywheel Integrity. We conclude that the planning for design, fabrication, and inspection of the flywheels is acceptable.

4.6 Power Distribution Monitoring

As mentioned in Section 5.3 the applicant now plans to limit the peak linear power density to 14.9 kW/ft rather than

18.8 kW/ft as originally proposed, without a reduction of the total reactor design thermal power level of 3411 MW. The change was made to achieve conformance with the acceptance critieria set forth in the Commission's Interim Policy Statement on emergency core cooling. The reduction in peak linear power density will be accomplished by a 21% reduction in the axial power peaking factor (2.1 versus 2.67 as originally specified in the PSAR). There will be no increase in the average power density over that in the original design. This change in the peak linear power density will not result in any physical core design change, but represents an operational restriction.

The maintenance of a peaking factor of 2.1 will require close surveillance of the core axial power distribution. To achieve this the applicant will be prepared to use an appropriate in-core monitoring system. We will evaluate the acceptability of the improved in-core instrumentation that will be required for this plant at the operating license review stage, when the detailed design information is available.

We have reviewed the information provided by the applicant and have concluded that, with the inclusion of a suitably sensitive in-core monitoring system, there is reasonable assurance that the applicant can develop appropriate operating procedures to assure that the maximum peaking factor can be maintained less than 2.1.

5.0 ENGINEERED SAFETY FEATURES

5.1 General

The engineered safety features of the Watts Bar and Sequoyah plants will be essentially the same. The containment and containment-related systems, emergency core cooling, and auxiliary feedwater systems will be designed to the same criteria.

As originally proposed, the containment leakage processing systems were also the same. During our review of Watts Bar, however, the emergency gas treatment system and the auxiliary building gas treatment system were completely redesigned to achieve overall higher performance in terms of dose reduction. This improvement became necessary when it was learned from meteorological data taken at the Watts Bar site that the site diffusion characteristics were poorer than predicted. It is our understanding that these system changes will be incorporated into the Sequoyah design.

5.2 Containment System

The containment for each of the reactors consists of a free-standing steel containment vessel with an ice condenser surrounded by a separate reinforced concrete shield building with an emergency gas treatment system. The system is designed to reduce the offsite consequences of a loss-of-coolant accident (LOCA). The assumed accident is a sudden rupture of the reactor coolant system.

The containment vessel is a low-leakage, steel structure designed for an internal pressure of 15 psig. The vessel including its penetrations is designed to confine the radioactive material that could be released in the event of an accident. The interior of this primary containment is divided into three major volumes or compartments, a lower compartment which houses the reactor and reactor coolant system, an intermediate compartment housing the energy absorbing ice bed in which steam is condensed, and an upper compartment which accommodates the air displaced from the other two compartments during a loss-of-coolant accident. The lower compartments and to a lesser extent the upper compartment are divided into subcompartments.

The ice condenser concept involves the very rapid absorption of the energy released, in the event of a loss-of-coolant accident, by condensing the steam in a low temperature heat sink. This heat sink, located inside the containment, consists of a suitable quantity of borated ice in a cold storage compartment.

The shield building is a medium-leakage concrete structure surrounding the containment vessel that is designed to provide for the collecting, mixing, holdup, and controlled release of containment vessel fission product leakage following an accident.

The design of the concrete and steel structures was dealt with earlier in Section 3.8.

5.2.1 Containment Functional Design

The Watts Bar ice condenser containment design and parameters that bear on the functional performance of the pressure suppression system are substantially the same as those previously reviewed and approved for the Sequoyah and the D. C. Cook plants. Our review of the Watts Bar ice condenser containment has been accomplished primarily by comparing the design and performance parameters of the Watts Bar system with those of the previously-reviewed plants. The differences that have been identified and other areas that required emphasis during our review are discussed and evaluated in the following paragraphs.

5.2.1.1 Initial Pressure Peak

The performance analysis of the Watts Bar ice condenser for the design basis LOCA predicts an initial pressure peak of about 8.5 psig occuring at approximately 250 milliseconds into the LOCA. This initial peak is due to an initial high rate of mass and energy discharge into the lower compartment causing much of the air in the lower compartment to be exhausted through the ice compartment and into the upper compartment. It is during this early portion of the LOCA transient that the maximum containment subcompartment pressure differentials and maximum loadings on the ice compartment inlet doors are encountered. The applicant has examined the lower compartment on a region by region basis (6 elements or regions) to determine the local peak pressures and the severity of the pressure differentials should the LOCA occur within regions most restrictive to the air and steam flow and to its distribution about the lower compartment. This analysis performed with the Westinghouse Transient Mass Distribution Code (TMD), reveals that a LOCA in a corner region of the lower compartment results in a local peak pressure of 9.8 psig and a local peak pressure differential on the subcompartment walls of approximately 9.0 psig. These local peaks are experienced within the lower compartment and are not "seen" by the containment shell.

In recognition of the fact that such transient spatial calculations are complex and are needed to establish the local peak pressure loadings for the actual plant subcompartment design and because the ice condenser full-scale section test program could not fully confirm the transient, spatial propagation of steam and air around the lower compartment of an actual plant configuration (which includes flow obstructions such as steam generators, pumps, etc.), the staff in late 1971 required that additional independent confirmatory analyses for subcompartment calculations be provided for ice condenser containment systems. This requirement, emphasizing subcompartment calculations was made a requirement of all applicants who have proposed use of ice condenser containments. These confirmatory analyses have now been completed. Our findings with respect to the confirmatory analyses and to the lower compartment pressure predictions indicate that an acceptable level of confidence exists as to the adequacy of the TMD subcompartment calculations. In view of the foregoing and considering that at least a 20% pressure margin in the subcompartment design above the calculated peak pressure or pressure differentials that result from the most severe blowdown mass and energy discharge from a pipe rupture occurring either within or adjacent to the subcompartment of interest will be provided, we have concluded that an acceptable level of lower compartment design will be attained for the Watts Bar containment. We have also reviewed the proposed design pressures for the reactor vessel annulus and pipe sleeves, the compartments above and below the reactor vessel, the steam generator enclosures and the pressurizer enclosure. We find the proposed design pressure levels to be reasonably consistent with those proposed for previously-reviewed ice condenser containments except that the pressures specified for the compartments above and below the reactor vessel (30 psi and 23 psi, respectively) appear to be on the order of 10 psi less than those for the McGuire compartments.

We recognize that layout and compartment volume differences and compartment vent opening sizes can account for such differences. We have concluded that the design techniques to be used by the applicant are able to assure an acceptable analysis to establish compartment and subcompartment design pressures. We will require the applicant to verify that the pressures used for design purposes were determined appropriately at the operating license stage of review.

The operating deck structural design and integrity is vital to the ice condenser containment performance and although insensitivity to large steam bypass areas has been demonstrated by the ice condenser full scale section test program, the plant operating deck structure is not testable. Consequently the applicant will conduct independent reviews on the structural design and analyses of the operating deck including the deck structures enclosing the pressurizer and steam generator vessels. This is similar to the course of action taken by other applicants using ice condenser containment systems, and we consider it acceptable for the Watts Bar facility.

5.2.1.2 Final Peak Pressure

The predicted final peak pressure for the Watts Bar containment is less than about 8 psig. This pressure results from the compression of air into the upper compartment during the LOCA blowdown period. This final pressure peak occurs in about 10 seconds as the blowdown nears completion. Prediction of the final pressure peak is based on polytropic air compression processes observed in the extensive full scale section test program and is readily amenable to a check by simple hand calculations. We consider that the final peak pressure level has been suitably described and that the reference containment design parameters that determine the polytropic air compression processes for the Watts Bar containment are essentially the same as those previously reviewed and accepted for the Cook, Sequoyah, and McGuire containments.

5.2.1.3 Long Term Pressure Peak

The "long term" pressure peak for the Watts Bar containment establishes the containment design pressure of 15 psig. The magnitude of this peak is determined principally by the maximum quantity of ice within the ice bed; the capacity of the containment spray system, and the energy released over the post-LOCA period to exhaust the ice. Presently the applicant predicts exhaustion of the ice at about 3000 seconds where upon the balance of the energy released is to be handled by the containment sprays. A "long term" pressure peak of about 12 psig is predicted to be attained at about 4000 seconds into the LOCA. We have reviewed the design basis input assumptions for these calculations and have concluded that they are reasonably conservative. For example an undefined energy release equivalent to 50 x 10^6 has been included into the design basis calculations and no structural heat sinks were considered in the 12 psig determination. An additional energy release of 68 x 10^6 Btu (representative of a hypothetical 33% zirconium-water reaction with the hydrogen burning as it is evolved) was also postulated in order to further demonstrate the containment heat removal capability. The resultant pressure from this capability study was 14.5 psig. In view of the ability of the containment system to sustain additional postulated energy releases of at least 118 x 10^6 Btu and remain within the 15 psi design pressure, we have concluded that the Watts Bar ice condenser containment system is acceptable and the intent of General Design Criterion 50 has been met.

5.2.2 Containment Heat Removal System

The containment heat removal system designed as engineered safety features will be provided to remove heat from the containment after a loss-of-coolant accident so as to reduce the containment pressure to essentially the ambient conditions. The principal components of the proposed Watts Bar containment spray system will be substantially the same as those in systems previously approved for construction in connection with ice condenser containments. The principal difference will be a slightly higher spray system flow rate and a reduction in the heat exchanger type and rating. The proposed Watts Bar containment spray heat exchanger rating is, however, one of the lowest reviewed to date for an ice condenser containment system. The proposed heat exchanger design also reverses the shell-tube flow arrangement from those plants previously reviewed. We have discussed this matter with the applicant and have identified no unique technical factors or problems associated with this reversed shell-tube flow arrangement. We will examine the final design of this component prior to operation of the Watts Bar plant. As in previously approved plants the containment spray system will be designed to accomodate the failure of any single component and still fulfill its pressure-limiting design function. We conclude that the proposed system is acceptable.

5.2.3 Containment Combustible Gas Control

Following a loss-of-coolant accident hydrogen gas could be generated inside the containment from a chemical reaction between the fuel rod cladding and steam (metal-water reaction), coating off-gassing, corrosion, and radiolysis. Both hydrogen and oxygen would be generated as a result of radiolytic decomposition of recirculating coolant solutions. If a sufficient amount of hydrogen is generated, and oxygen is available in stoichiometric quantities, the subsequent reaction of hydrogen with oxygen at rates rapid enough to lead to a significant over-pressure could lead to failure of the containment to maintain low leakage integrity. In this regard the AEC has published Safety Guide No. 7 that describes an acceptable method of controlling combustible gas concentrations in containment following a loss-of-coolant accident.

A hydrogen control system, designed to engineered safety feature standards, for the post-accident control of hydrogen in the containment will be provided for the Watts Bar plant. For this purpose, the applicant will use a newly developed Westinghouse electric hydrogen recombiner system. Each of the two recombiners in the system will be capable of recombining all of the hydrogen generated, using the assumptions of AEC Safety Guide No. 7. In accordance with AEC Safety Guide No. 7, the applicant will also install a controlled purge system as a backup to the recombiner for controlling post-accident hydrogen.

The Westinghouse designed electric hydrogen recombiner system will consist of two recombiner units to be located within the containment building and an associated control panel to be located outside the containment in an area that will be accessible following a loss-of-coolant accident. The hydrogen recombiner system will be operated only after a loss-of-coolant accident. Operation will be initiated from the control station. The heating elements within the unit will be energized, increasing the temperature of the atmosphere within the recombiner to produce a natural draft through the system. The temperature of the containment atmosphere drawn through the unit by natural convection will be raised to a level sufficient for recombination of the hydrogen and oxygen to occur (approximately 1160°F). Recombination will take place without producing a flame.

We have reviewed the information presented in regard to the design basis, performance, and the effects of containment parameters on recombiner performance. We have also reviewed the proof-of-principle tests and the prototype tests that have been conducted by Westinghouse, and have concluded that an acceptable system for hydrogen control has been developed and is suitable for use in the proposed Watts Bar containments. Details regarding the Watts Bar final system design and installation will be reviewed prior to plant operation.

On the basis of our evaluation, we have concluded that the design criteria for the control of combustible gas in the containment in the event of a loss-of-coolant accident meet the recommendations of AEC Safety Guide No. 7 and are acceptable.

5.3 Emergency Core Cooling System (ECCS)

The ECCS will be designed to provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the reactor coolant system piping resulting in a loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The ECCS subsystems to be provided will be of such number, diversity, reliability, and redundancy that no single failure of ECCS equipment occurring during a loss-of-coolant accident will result in inadequate cooling of the reactor core. Each of the ECCS subsystems will be designed to function over a specific range of reactor coolant piping system break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe.

The Watts Bar emergency core cooling system will consist of two high pressure injection subsystems (two centrifugal charging pumps and two safety injection pumps), an injection system employing accumulator tanks, and a low pressure injection system with external (to the containment) recirculation capability. Various combinations of these systems will be employed to assure core cooling for the complete range of postulated break sizes.

All of the ECCS subsystems will be designed to accomplish their functions when operating on emergency (onsite) power as well as offsite power. In the event of a loss-of-offsite power concurrent with a single failure in the emergency power supply system, the minimum ECCS requirement of the accumulators (which require no electrical power), plus one centrifugal charging pump, one safety injection pump and one low head injection pump would be available for operation and capable of providing the required performance.

With respect to performance of the ECCS, the AEC regulatory staff has conducted a general reevaluation of the ECCS for light water cooled reactors.

On June 19, 1971, the AEC issued an Interim Policy Statement containing interim acceptance criteria for the performance of emergency core cooling systems for light water cooled nuclear power reactors. The Interim Policy Statement includes a set of conservative assumptions and procedures to be used in conjunction with the Westinghouse codes to analyze the ECCS performance for pressurized water reactor plants incorporating a dry containment.

As did the Duke Power Company in the case of the McGuire application, TVA has provided the results of an analysis of the ECCS performance capability for the Watts Bar plant using the Westinghouse evaluation model described in the Commission's Interim Policy Statement to account for differences between a low-pressure containment system (ice condenser) and the standard dry containment. The AEC Interim Policy Statement on ECCS permits modifications where changes to the evaluation model are justified. We evaluated these modifications during our review of the McGuire station application and found them acceptable.

To meet the acceptance criterion of the Interim Policy Statement limiting the calculated peak clad temperature to less than 2300°F, the applicant proposes to limit the maximum linear power density to 14.9 kW/ft. The calculated peak clad temperatures for a spectrum of pipe break sizes up to and including the double-ended rupture of the largest coolant pipe assuming plant operation at 102% of the design power level of 3411 MWt, are shown in Table 5.3-1.

TABLE 5.3-1

Break	Peak Clad Temperature (°F)
Double-ended Hot Leg (8.24 ft^2)	1205
Double-ended Cold Leg (8.24 ft^2)	2300
0.6 Double-ended Cold Leg (4.94 ft^2)	2210
Cold Leg (3.0 ft^2)	2030
Cold Leg (0.5 ft^2)	2245

The results of the analyses indicated that for each of the assumed pipe breaks, the total core metal-water reaction is less than the 1% limit specified in the interim acceptance criteria.

The clad temperature transient is terminated while the core is still amenable to cooling and before it becomes excessively embrittled, such that its essential heat transfer geometry is preserved and it can be cooled to remove decay heat for an extended period of time.

On the basis of our evaluation, we have concluded that the predicted functional performance of the Watts Bar ECCS for the full spectrum of postulated break sizes is in accord with the Commission's Interim Policy Statement and acceptance criteria and is acceptable.

The applicant has stated that Westinghouse is continuing to seek an optimum solution to the ECCS problem. Limitation of peak power density, model improvements, and system modifications are being studied. When these studies are completed in late 1973, the final design of the ECCS will be submitted to the Commission for review and approval. TVA has stated that in the meantime the Watts Bar ECCS design will be kept sufficiently flexible to incorporate the essential features of the final solution as approved by the Commission. This commitment is acceptable to us.

5.4 Auxiliary Feedwater System

The auxiliary feedwater system will supply water to the steam generators for decay heat removal if normal feedwater is lost through loss of power or other malfunction. Two electric pumps and one steam-driven pump will be provided for each unit. Any one pump will be capable of supplying sufficient water to the minimum required steam generator water level for the removal of decay heat.

In addition to the normal supply from the condensate storage tanks, an emergency source of river water will be provided by the fire protection system. This portion of the fire protection system will have suitable redundancy and will be seismically designed.

Following a loss-of-coolant accident, the auxiliary feedwater system will also serve to maintain a sufficient head of water in the steam generators to prevent radioactive leakage through any existing steam generator tube leaks.

We have concluded that the proposed design of the auxiliary feedwater system, including the provisions for alternate water supply, is acceptable.

5.5 Emergency Gas Treatment System (EGTS)

The purpose of the emergency gas treatment system (EGTS) is two-fold: (1) to maintain the pressure in the shield building annulus negative with respect to the containment, the auxiliary building, and the atmosphere at all times, and (2) to hold up and filter annulus in-leakage prior to releasing it to the atmosphere. These objectives are met by the combination of two subsystems: the annulus pressure control and air cleanup subsystems.

Annulus Pressure Control Subsystem (APCS)

The annulus pressure control subsystem is a fan and duct network provided to maintain the shield building annulus pressure below atmospheric during normal plant operation. In the event of an accident, this subsystem will be shut down and redundant isolation dampers will automatically close all ducts. The function of the subsystem will, therefore, be anticipatory, i.e., it will establish an initial annulus pressure sufficiently low that throughout the period immediately following the accident and before the main leakage processing chain (the air cleanup subsystem) is activated, the annulus leakage will remain inward. The pressure increases that must be accommodated during this period will be due principally to thermal expansion of annulus air and containment vessel dimensional growth.

Air Cleanup Subsystem

The air cleanup subsystem is a completely redundant system of ducts, fans, and filters. It will be designed to draw air from the shield building annulus through an inlet located above the primary containment and process it through a series of filters. Enough of the filtered stream to maintain a negative annulus pressure will then be exhausted via the shield building vent. The remainder will be returned in a distributed fashion to the bottom of the annulus.

Each redundant filter train will consist of a demister and heating elements, a particulate filter, an absolute filter, two charcoal absorbers, and another absolute filter. The filter assemblies will remain isolated between automatically controlled dampers during standby periods to reduce the potential for contamination.

Pressure-controlled modulating dampers located at the inlet to the annulus air distribution header and in the duct to the shield building vent will be designed to maintain the annulus air pressure at the appropriate negative level. The applicant will conduct a series of initial preoperational tests to confirm the predicted performance of the EGTS. We will review the results of this testing program in detail, at the operating license stage. In the meantime, we have concluded that the applicant has developed sufficient preliminary design information on which to base confidence that the EGTS will function as intended.

5.6 Auxiliary Building Gas Treatment System (ABGTS)

The auxiliary building gas treatment system (ABGTS) is designed to collect and process potential containment leakage that bypasses the shield building annulus following a loss-of-coolant accident, and as such is an adjunct to the EGTS.

During normal plant operation the auxiliary building will be held at a slight negative pressure by the general auxiliary building ventilation system. Following an accident this function will be taken over by the ABGTS, which, in addition, will process all auxiliary building in-leakage through a filter train similar to those provided for the EGTS and exhaust it through the shield building vent.

All ducts, fans, and filters are suitably redundant.

The applicant proposes to test the system following construction to assure that a negative pressure can be maintained during isolated conditions.

We have concluded that the ABGTS can be designed and built so as to perform its intended function.

6.0 INSTRUMENTATION AND CONTROL

6.1 General

The protection system for Watts Bar Nuclear Units 1 and 2 have been designed to satisfy the requirements of the Commission's General Design Criteria (GDC) and of the IEEE-279 Criteria for Protection Systems for Nuclear Generating Stations, (1971).

The recommendations of recent Safety Guides No. 11 and No. 22 and recent IEEE Standards have also been adopted by the applicant. The acceptability of the applicant's implementation of these documents is addressed in this evaluation.

We have also evaluated the seismic and environmental qualification of electrical equipment and instrumentation, quality assurance provisions, cable separation, identification and installation design criteria, Class IE electric and protection systems testability, and instrumentation provided for incident and post-accident monitoring.

6.2 Plant Protection and Control Systems

6.2.1 Comparison of Protection Systems

Our evaluation of the Watts Bar protection system consists of a comparison with the previously evaluated Sequoyah design. The applicant has not identified any feature of the Watts Bar protection system that differs from the Sequoyah plant. The protection system designs for Watts Bar and Sequoyah meet the requirements of the 1971 version of the IEEE 279 Standard.

The applicant has stated that both plants, Watts Bar and Sequoyah, will be provided with a new reactor protection system power range flux-rate trip. The applicant has submitted a description of the system and has stated that it will conform with IEEE-279 requirements. We have concluded that this is acceptable.

The applicant has stated that the Watts Bar reactor protection system trips and engineered safety features will be testable in accordance with the requirements of Safety Guide No. 22.

We have concluded that the protection system design is acceptable and affords protection equivalent to that of the Sequoyah plant.

Comparison of Control Systems

The applicant has stated that the design of the Watts Bar control systems would be functionally the same as that for the Sequoyah plant. The applicant has not identified any differences. This commitment is acceptable and satisfies our evaluation requirements.

Bypass Indication for the Reactor Protection System and Engineered Safety Features

The reactor protection system bypasses are in accordance with paragraph 4.13 of IEEE-279 and are acceptable.

While the applicant has agreed that indication of bypass of engineered safety features (ESF) is essential we do not consider that the documentation provided is acceptable. We, therefore, will require the applicant to document its intent to have the designs of these circuits include control room annunciators whenever operator actions result in the loss of an ESF function or a reduction in system redundancy. The applicant will be required to demonstrate at the operating license review that its design complies with this requirement.

Four channels of containment pressure instrumentation will be provided. The containment pressure transmitters will be connected to four pressure taps on containment. The applicant has stated that protection system sensing lines will meet the requirements of Safety Guide No. 11. This commitment is acceptable.

6.2.4 Periodic Testing of the Reactor Protection System and Engineered Safety Features

The applicant's reactor protection system design provides for testability of individual channels, logic, and final actuation devices. Similarly, the engineered safety feature initiation channels, logic and final actuation devices will be testable. We have concluded that this satisfies Safety Guide No. 22 and is acceptable.

6.3 Post-Accident and Incident Monitoring

The applicant has provided a listing of instrumentation that would be available to the control room operator to follow an accident or an incident condition. The instrumentation provided is sufficiently comprehensive and of the required range to permit an operator to make decisions. Further, the instrumentation will be qualified for the accident environment, will be redundant, and will be energized from the emergency power system, and at least one channel will be recorded. The applicant's commitments in this regard are adequate to satisfy our requirements.

6.4 Accumulator and RHR/RCS Interlocks

The applicant has agreed to design the valve circuitry to include certain interlocks necessary to conform with the position given below with the understanding that the topic may be pursued by Westinghouse toward developing a less complex and more reliable system. This is acceptable.

a. Accumulator Motor-Operated Valves

An acceptable degree of protection would be provided if the control circuit for the motor-operated isolation valves between the accumulators and the primary coolant system were designed to meet the intent of IEEE-279 and to incorporate the following features.

(1) Automatic opening of the valves when the primary coolant system pressure exceeds a preselected value (specified in the Technical Specifications).

(2) Valve position visual indication that is actuated by sensors on valve ("open" and "closed").

(3) An audible alarm, independent of item 2, which is actuated by a sensor on the valve when the valve is not in the fully open position.

(4) Utilization of a safety injection signal to automatically remove (override) any bypass feature that may be provided to allow a motor-operated valve to be closed, for short periods of time, when the primary system is at pressure (in accordance with the provisions of the Technical Specifications).

b. <u>RHR/RCS Motor-Operated Valves</u>

The following design features for the motor-operated valves in the letdown line between the high pressure primary coolant system and the relatively low pressure RHR system would, in our opinion, provide an acceptable degree of protection.

(1) Provision of at least two valves in series, with each valve interlocked to prevent valve opening unless the primary system pressure is below the RHR system design pressure.

(2) Interlocks of diverse principles, and designed to meet the intent of IEEE-279.

(3) Provision for automatic closure of the two series valves whenever the pressure in the primary coolant system exceeds a selected fraction of the design pressure of the RHR system. These closure devices should be designed to the intent of IEEE-279.

6.5 <u>Cable Separation and Identification Criteria for Protection and Emergency Power Systems</u>

The applicant's design criteria for separation of redundant cable routing in the cable spreading area are acceptable. A minimum separation of 3 feet horizontally and 5 feet vertically will be maintained between redundant cable trays.

Also, the applicant has developed an acceptable program for identification of protective system equipment and cabling.

7.0 ELECTRIC POWER

7.1 General

The electric power system has been designed to satisfy the requirements of General Design Criteria 17 and 18, the IEEE 308 Standard, Criteria for Class IE Electric Systems for Nuclear Power Generating Stations, dated 1971; and the recommendations of Safety Guide No. 6 and No. 9.

The applicant has stated that all the electrical systems and equipment required by Unit 1 and shared by Unit 2 will be available prior to startup of Unit 1. This includes the diesel-generator system, 125-volt dc system and the offsite power system.

7.2 Offsite Power

The Watts Bar hydro switchyard will be interconnected with the TVA system by eight 161-kV transmission lines and also by five Watts Bar hydro-generators. The applicant has shown that the transmission lines will maintain sufficient physical independence to meet the requirements of General Design Criterion 17.

Preferred power will be supplied from the existing Watts Bar hydro switchyard over two lines approximately 1.5 miles in length to Watts Bar Units 1 and 2. The 161-kV lines will be supported on separate structures separated sufficiently to ensure that the failure of any single tower will not endanger the redundant line.

Each of the 161-kV circuits will be connected to separate bus sections of the hydro switchyard double Z bus arrangement. The switchyard will be designed so that loss of any one of the four main bus sections will not cause a loss of power to either of the two preferred lines supplying the nuclear units.

The Watts Bar Nuclear Unit 1 will constitute approximately 3.8% of TVA's total system capacity and Unit 2 will represent approximately 3.7%. The loss of either or both units should not cause a significant disturbance of the TVA 500-kV transmission network. Power generated at the Watts Bar Nuclear Plant will be delivered to TVA's load centers over five 500-kV transmission lines. The applicant has evaluated the loss of the two 500-kV non-independent transmission lines from the generators to the 500-kV switchyard and has determined that a loss of preferred power at 161 kV to the Watts Bar units would not occur. The applicant has stated that although this 161 kV system would be fragemented, it would still be available to the Watts Bar substation.

We have concluded that the stability of the grid is acceptable.

The two 161/6.1-kV common service station transformers energized from the two physically independent lines from the Watts Bar hydro switchyard will each provide power to both startup buses A and B (nonsafety), which will be common to Units 1 and 2. Each common service transformer and each startup bus will have the capacity to supply the engineered safety features of one unit under LOCA conditions and the power required for safe shutdown of the non-accident unit. During normal operation, power will be supplied to each unit by two 22.5/6.9-kV unit station service transformers. The unit station service transformers of each unit will be energized from its main generator iso-phase bus. On loss of a unit generator, a fast-transfer to the startup buses will occur. The startup buses will be continuously energized through the redundant common station service transformers from offsite power sources.

We have concluded that the applicant's offsite power system design is acceptable and in accord with GDC 17 requirements.

7.3 Onsite Power

7.3.1 AC Power System

The design of the auxiliary power system will utilize the split-bus concept. Two independent 6.9-kV switchgear units for each unit, which are not a part of the Class IE electric system, will be fed from both startup buses and from a unit station service transformer. They in turn will energize the two redundant Class IE 6.9-kV shutdown boards for each unit. In addition, two standby diesel-generators, one connected to each 6.9-kV shutdown board, will be provided for each unit. In the event of an accident a single diesel would be started and on a loss of voltage it would be connected to its shutdown board.

The emergency power systems for Unit 1 and Unit 2 will be independent of each other except for the essential raw cooling water and component cooling water pumps and some 480-volt loads which will be shared equipment. In these cases, three of four shutdown boards will be required to serve both units with one under LOCA conditions and the other unit shutdown. We have concluded that this design meets the single-failure criterion and is acceptable.

The four diesel generator sets to be provided will be physically separated in a seismic Category I building located above the maximum flood level assumed for design of the facility. Each set will be provided with independent auxiliary systems such as the starting, fuel oil, cooling water, and dc control power systems.

The applicant has agreed to satisfy Safety Guide No. 9. The tabulation of loads in the PSAR indicates that the short time diesel-generator loading, following a LOCA, will be within the continuous rating specified for the diesel-generator.

Standby emergency power for the redundant Class IE equipment at 480 volts and at lower voltages will be supplied from each 6.9-kV shutdown board by two 6.9/0.48-kV stepdown transformers through nine 480-volt boards. A third stepdown transformer will be provided as a backup for either train.

The 120-volt ac vital instrument power system will consist of four boards for each unit and conforms with the recommendations of Safety Guide No. 6.

The onsite emergency ac power system meets the requirements of GDC 17 and 18, IEEE-308, and Safety Guide No. 6 and No. 9. We have concluded that it is acceptable.

7.3.2 DC Power System

The vital dc system for Units 1 and 2 will consist of four 125-volt batteries and buses. A charger will be assigned to each bus and a spare charger with manual breakers will be shared between each pair of batteries. The batteries will be located in separate Category I areas above the maximum flood level assumed for design of the facility, will be separately ventilated, and will be designed to comply with Safety Guide No. 6 and applicable IEEE-308, 1971 criteria.

We have concluded that the 125-volt vital battery system is acceptable.

8.0 AUXILIARY SYSTEMS

8.1 General

The auxiliary systems are described in Section 9 of the PSAR. These process systems normally provide plant services that have an auxiliary function to the production of power. The systems proposed for the Watts Bar facility are substantially the same as those we have reviewed in the context of our reviews of Trojan, McGuire, and most notably, Sequoyah. In our review of Sequoyah, we directed our attention to the design bases of these systems, including any safety-related objectives of the respective systesm, and the manner in which these objectives will be achieved. For those auxiliary systems that are safety-related, we reviewed the requirements for redundancy, independence, and physical separation, and the criteria that establish the quality of the systems. During our review of Watts Bar, we have checked the appropriateness of the seismic design classification, and the acceptability of the codes, standards, and specifications to be used for the design, fabrication, and inspection of the piping and other components within each system.

On the basis of our current and earlier reviews, we have concluded that the auxiliary systems to be provided for Watts Bar are acceptable.

We have given two auxiliary functions additonal attention: the auxiliary building gas treatment system which was discussed in Section 5.6, and the essential plant cooling provisions.

8.2 Plant Cooling: Emergency High- and Low-Water Conditions

8.2.1 Flood Above Plant Grade

TVA has developed an alternate method for decay heat removal from the reactors in the unlikely event of flood conditions that exceed plant grade. This method was also proposed in connection with the Sequoyah application. At that time we reviewed it in detail and concluded that it was acceptable. The same plan is proposed for Watts Bar.

Briefly, should probable maximum flood conditions threaten (it has been shown that a minimum of 36 hours warning would be available), the plants would be shut down and preparations would be made to flood their auxiliary buildings. Decay heat removal would be accomplished by natural circulation in the primary system and steam blowdown to the atmosphere from the secondary system. Fire pumps would provide makeup to the steam generators during the flood and until normal heat removal functions could be restored.

To cope with this situation, the Watts Bar intake structure pump deck has been designed to remain above the maximum flood level so that the essential raw cooling water (service water) pumps and the fire pumps remain operational. For the same situation, Sequoyah has a special system of auxiliary essential raw cooling water pumps, mechanical draft auxiliary cooling towers, and submersible fire pumps.

On the basis that the Watts Bar emergency high-water provisions are similar to those for Sequoyah, while of much simpler design, and that they will be designed to safety standards of performance and reliability, we have concluded that they are acceptable.

8.2.2 Downstream Dam Failure

Loss of the downstream dam is handled differently for the Sequoyah and Watts Bar plants. Watts Bar will be designed so that its essential pump suctions remain below minimum river level, whereas Sequoyah will use a closed-cycle system employing the auxiliary cooling towers mentioned above. The two approaches provide an equivalent level of protection.

We have concluded that adequate consideration has been given to the minimum water level condition.

8.3 Spent Fuel Cask Handling System

The design of the fuel storage pit and the remainder of the auxiliary building for the Watts Bar facility is essentially complete by virtue of the applicant's effort to make it a duplicate of that for the Sequoyah facility. Spent fuel shipping casks will be handled by the same auxiliary building crane that will handle other heavy loads in the auxiliary building, and that occasionally will be used to transfer such loads to and from the shipping area. To perform this dual function the crane must be able to pass over the spent fuel storage pit. To provide the required degree of protection against the potential dropping of a heavy object into the pit and onto stored spent fuel the applicant has agreed to:

(1) Provide electrical interlocks designed to meet the requirements of IEEE-279 to prohibit movement of heavy loads over the spent fuel area.

(2) Establish a safe-passage corridor along one side of the spent fuel storage pit. If a portion of the pit is included in this corridor, spent fuel will be excluded from it and, it will be separated from the storage area by an underwater wall.

(3) Provide removable mechanical stops to be installed on the crane to prohibit crane hook passage over the fuel storage area whenever the crane is in the safe-passage corridors. The installation and removal of these stops would be under strict administrative control.

We have concluded that the design of the fuel storage pit and the provisions being made to limit the potential for dropping heavy objects into the spent fuel storage pit are acceptable.

9.0 RADIOACTIVE WASTE MANAGEMENT

The original Watts Bar radwaste systems were designed to comply with the AEC regulations (10 CFR Part 20) in effect at the time the construction permit application was filed with the Commission. During our review of the application, the applicant modified the gaseous and liquid radioactive waste treatment systems to meet the requirements of proposed Appendix I to 10 CFR Part 50, which was published in the Federal Register on June 9, 1971.

This entailed the following design changes:

(1) Addition of three gas decay tanks, for a total of nine, to increase the minimum gas holdup time to 60 days.

(2) Addition of high-efficiency particulate air (HEPA) filters plus two charcoal absorbers in series to treat activity in the condenser off-gas associated with a primary-to-secondary leak.

(3) Addition of a full-flow HEPA filter to the auxiliary building exhaust system.

(4) Addition of full-flow charcoal absorbers to the containment purge system.

(5) Segregation of equipment drains into tritiated and non-tritiated sources with provision to process tritiated water for reuse in the primary cooling system.

(6) Addition of a 15-gallon per minute evaporator to process steam generator blowdown, and aid in removing liquid-borne activity from the secondary system.

The capacities of the revised Watts Bar radwaste systems for treating both liquid and gaseous sources of activity are considerably greater than those originally proposed. The equipment capacities for these modified systems are shown in Table 9.0-1. The applicant estimates that the annual total quantity of radioactive material to be released from this plant will be less than 5 curies. Our review of the revised preliminary design of the liquid radwaste system indicates that this estimate can be achieved and that with proper operation of the system releases will be low. On this basis the system is acceptable.

Liquid Radwaste

As with similar plants such as the Sequoyah and McGuire facilities, the radioactive waste batch operation type treatment system and storage systems are to be sized on the basis of continued reactor operation with clad defects in 1% of the fuel rods. Liquid effluents to the Tennessee River will be reduced to as low as practicable and will be continuously monitored by a radiation detector.

The Watts Bar waste treatment system will be designed to recycle as much reactor-grade water entering the system as possible. This will be accomplished by segregation of equipment drains and waste streams to reduce the intermixing of tritiated and non-tritiated liquids.

Most of the radioactive liquids discharged from the primary cooling system will be processed through the 2-gallon per minute waste evaporator and retained within the plant by the chemical and volume control system (CVCS). This will reduce the input to the waste disposal system designed to process the low-volume activity wastes originating from equipment drains, radiochemical and laboratory drains, decontamination area drains, demineralizer flushing and back washing, and the sampling system. This non-tritiated water will be collected in the 23,500-gallon floor drain collector tank, pumped through the floor drain filter to the 15-gallon per minute waste auxiliary waste evaporator. The condensate will flow to one of three 2,000-gallon waste condensate tanks for release to the condenser cooling water system. Prior to release of liquid from the waste condensate tanks, a laboratory analysis will be made to determine the type and amount of activity, and that the release will be within the limits imposed by the Technical Specifications which will be developed during the operating license review. The discharge valve will be interlocked with a process radiation monitor and will close automatically when the radioactive concentration in the liquid discharge exceeds preset limit.

The effluents from the laundry and shower drains will constitute the largest volume of liquid wastes to be processed by the radioactive waste treatment system. We do not expect this to be a significant contributing source of activity. This water will be transferred to one of the waste condensate tanks via the waste condensate filter, sampled and released to the condenser circulating water discharge line if the activity is below acceptable levels. If the analysis indicates that further processing is required, the contents of the liquid waste tank will be pumped to the floor drain collector for cycling through the auxiliary waste evaporator.

In the event primary-to-secondary leakage results in high secondary side activity, the steam generator blowdown will be routed through the auxiliary blowdown cooler to the floor drain collector tank for processing through the auxiliary waste evaporator prior to release to the condenser circulating water discharge.

Based on the performance of operating plants of similar design we have concluded that the liquid activity released from the Watts Bar radwaste system will be less than 5 curies/year/unit and therefore, we have concluded that with proper operation of the system, releases will be as low as practicable.

9.2 Gaseous Waste Control

9.2.1 Gaseous Radwaste

The Watts Bar gaseous radwaste system collects and processes gases stripped and purged from the reactor coolant, cover gases displaced from liquid storage tanks, and gases collected from equipment vents during sampling operations. These wastes can be stored in the gas decay tanks for 60 days prior to controlled release to the atmosphere. All releases will be continuously monitored by three separate systems which will measure and record gaseous, particulate and radioactive-iodine releases. A trip valve in the discharge line will be closed automatically by a high-activity signal.

9.2.2 Containment Purging

The Watts Bar containment purging cleanup system will consist of an internal recirculation system, containing both HEPA filters and charcoal absorbers, designed to reduce the iodine and particulate activity prior to venting.

The iodine activity will be further reduced by charcoal absorbers and filters to be located in the purge exhaust system. The applicant has estimated 12 purges/year and has shown that the release of activity from this source will be reduced to levels that will be as low as practicable.

9.2.3 Condenser Off-Gas System

During normal operations the air ejector off-gas will be vented from the turbine building roof. In the event of high secondary radioactivity levels that could accompany a steam generator tube leak, a radiation monitor on the off-gas line will indicate an abnormal situation. During periods when the reactor is operated with such a primary-to-secondary leak in a steam generator, the air ejector off-gas from the condenser will be manually routed through both HEPA filters and charcoal absorbers in series prior to release to the environment. The condenser off-gas system will be designed to maintain the releases from this source to levels that will be as low as practicable.

9.2.4 Auxiliary Building Leakage

In addition to the gaseous activity releases described previously, activity can become airborne in the auxiliary building from equipment leaks. This air will be routed through HEPA filters prior to release to the environment.

9.2.5 Steam Leakage

Steam leakage in the turbine building will result in some iodine activity becoming airborne. This will be exhausted to the atmosphere from the turbine building ventilation system. Calculations, however, show that this source of iodine activity will be small and will not require filtration prior to release to the environment.

9.3 Solid Wastes

The solid waste disposal system will provide for the collection, packaging and shipment of the solid radioactive wastes resulting from plant operation. Shipments will be to a licensed burial site in accordance with AEC and DOT regulations. The bulk of the material will consist of evaporator concentrates and spent ion-exchange resins from the liquid waste treatment systems. The remaining wastes will consist of filters, contaminated rags, paper, glassware, and miscellaneous materials.

TABLE 9.0-1
RADWASTE TREATMENT COMPONENTS

Component	Number	Flow Rate, Gallons Per Minute	Volume (Each), Gallons
Reactor Coolant Drain Tank	2	150	600
Laundry Drain Tank	2	20	600
Chemical Drain Tank	1	20	600
Sump Tank	1		600
Tritiated Drain Tank	1	20	24,700
Floor Drain Tank	1	20	23,526
Spent Resin Storage Tank	1		2,250
Waste Condensate Tank	2	20	1,500
Waste Condensate Tank	1		2,000
Gas Decay Tank	9		600
Waste Evaporator	1	2	
Auxiliary Waste Evaporator	1	15	

9.4 Radiation Monitoring System

All important fluid streams discussed earlier in this section which could become significant pathways for radioactivity from the plant will be continuously monitored with suitable detectors. The process radiation monitoring system will provide status indication, and alarms when conditions rise above preset levels and, where appropriate, automatic remedial action. This system will be similar in design to those we previously reviewed and approved for several other plants.

Similarly, the area radiation monitoring system will monitor radiation in various portions of the plant, normally accessible to operating personnel. Radiation levels will be indicated locally and in the control room and alarms will be actuated in both locations in the event allowable radiation limits are exceeded.

9.5 Environmental Monitoring

The applicant has described an environmental radiological monitoring program which will begin 2 years prior to plant startup and will continue throughout the life of the plant. The program will include sampling and analysis of air, milk, vegetation, crops, soil, fish, clams, bottom sediment, plankton, and water from wells, surface sources, and public water supplies. The applicant has presented tentative sampling locations and frequencies, types of analyses and quality control measures.

Our consultant, the Fish and Wildlife Service of the U. S. Department of the Interior, has reviewed the proposed program and considers it adequate to protect fish and wildlife resources from significant damage. The report of the Fish and Wildlife Service is attached as Appendix H.

We conclude that the applicant's program will be adequate for monitoring the radiological impact of plant operation on the environs and assessing the health and safety aspects of the release of radioactivity to the environment from the proposed operation of the plant.

10.0 CONDUCT OF OPERATIONS

10.1 Organization and Technical Qualifications

The applicant has a large in-house corporate technical support base in direct support of its multi-unit nuclear construction program. The corporate technical support base is centered in the TVA Nuclear Operations Coordinator and the professional personnel comprising the Divisions of Power Production, Engineering Design, and Construction. The nuclear operations coordinator is responsible for reviewing the design and plans for the nuclear units for compliance with licensing regulations, safety, and operating economy. Within the Division of Power Production, the Power Plant Engineering Branch and Power Plant Maintenance Branch provide multi-discipline corporate technical support to the nuclear plant. The Division of Engineering Design serves as the plant architect-engineer and principal contractor with the nuclear steam supply system vendor (Westinghouse). The Division of Construction is responsible for constructing the plant in accordance with design specifications provided by the Division of Engineering Design. Other TVA Divisions, Chemical Development, Power Resource Planning (Fuels), Transmission Planning and Engineering, Power Systems Operations and the Office of Health and Environmental Science are available to support the design and construction effort.

We have concluded that the applicant and its contractor are qualified to design and construct the Watts Bar Nuclear Plant.

The applicant has proposed a total station complement of approximately 170 personnel to staff the facility. These personnel will function in three main groups: Operations, Results (Technical Support), and Maintenance.

The Operations Group will consist of five operating shifts supervised by a Power Plant Operations Supervisor (Licensed Senior Reactor Operator) and an Assistant Power Plant Operations Supervisor (Licensed Senior Reactor Operator) and the Power Plant Superintendent and his assistant. Each normal shift for single-unit operation will be composed of six personnel; a Shift Engineer (Licensed Senior Reactor Operator), an Assistant Shift Engineer (Licensed Senior Reactor Operator or Reactor Operator), a Unit Operator (Licensed Reactor Operator) and three Assistant Unit Operators. One additional Assistant Shift Engineer, one additional Unit Operator, and one additional Assistant Unit Operator will be required for two-unit operation.

Approximately 84 plant level maintenance and technical support personnel will be assigned to the plant. The maintenance group will be headed by a Power Plant Maintenance Supervisor (under the Power Plant Superintendent) who will supervise two sub-groups headed by Assistant Power Plant Maintenance Supervisors in charge of Electrical and Mechanical Maintenance. The Results (Technical Support) Group will be headed by a Power Plant Results Superviosr (under the Power Plant Superintendent) who will supervise engineering and technical personnel in the areas of nuclear engineering, mechanical engineering, chemical engineering, and instrumentation engineering. Health Physics (Radiation Protection) functions will be carried out by a separate Health Physics Group directly under the Power Plant Superintendent and his Assistant.

We have concluded that the applicant's plans for staffing the facility are in conformance with current guidance and will provide an adequate operating organization and an adequate plant-level technical support capability.

10.2 Selection and Training of Personnel

The applicant has indicated its intent to meet the requirements of American National Standards Institute N18.1, Standard for Selection and Training of Personnel for Nuclear Power Plants. The initial training program will be divided into several phases: (a) Basic Nuclear Course; (b) Plant Technology Course; (c) Plant Systems and Operations Training; (d) Training Period at an Operating Reactor in Connection with Simulator Training; (e) Simulator Training; (f) On-the-Job Training at an Operating Plant; and (g) Control Board Experinece. Additionally, plant personnel will be given training in radiation protection, emergency procedures, industrial security and first-aid.

We have concluded that the program being developed for the selection and training of station personnel is adequate to ensure that a qualified capable staff will be trained for the Watts Bar plant.

10.3 Emergency Planning

The applicant has submitted information in accordance with Section 50.34 of 10 CFR Part 50 with regard to the requirements for emergency plans at the construction permit stage. The applicant has outlined an organization for coping with emergencies, and has described contacts and arrangements which will be developed with local, State, and Federal agencies with responsibilities for coping with emergencies. The applicant intends to develop offsite and onsite protective measures. An agreement has been made with the Oak Ridge Associated Universities Hospital for the definitive care of any severely injured victims of a radiological accident. Onsite first-aid facilities will be provided at the facility. An agreement for emergency medical treatment for any accident victims will be culminated between TVA and a local hospital.

10.4 Industrial Security

The Security Program for the Watts Bar Nuclear Plant will be directly supervised by TVA's Division of Reservoir Properties and the Division of Power Production. Central authority for the conduct of the security program will be vested in TVA's Office of General Manager. The plant site and its structures will be protected by security fencing, lighting, surveillance equipment, physical barriers and a trained security force. A system of personnel identification, access control and administrative arrangements will be established to limit access to the plant and its equipment.

We have discussed with the applicant our concern over the adequacy of security provisons for the plant water intake pumping station area. TVA has agreed to provide adequate security arrangements for this portion of the facility.

We have concluded that the applicant has provided sufficient information in the form of an overview of the Industrial Security Program to ensure that an Industrial Security Plan will be developed that will provide plant protection and reasonable assurance that the risk associated with potential acts of sabotage that could lead to a significant threat to the public health and safety is acceptably low.

11.0 ACCIDENT ANALYSIS

11.1 General

In order to assess the safety margins of the plant design, the following plant operating transients were considered by the applicant: rod withdrawal during startup and from power, moderator dilution, loss-of-coolant flow, loss of electrical load, and loss of ac power. The applicant's criterion for detailed design of the reactor control and protection system is to be able to automatically take corrective action to cope with any of these transients. Our previous evaluations of other PWR plant designs at the operating license stage have demonstrated that anticipated transients will be terminated with adequate margin to a minimum departure from nucleate boiling ratio of 1.3, and we have concluded that this limit can be met in the Watts Bar units.

The postulated design basis accidents analyzed for offsite radiological consequences by the applicant are the same as those analyzed for previously licensed PWR plants, including a steam line break accident, a steam generator tube rupture accident, a loss-of-coolant accident, a fuel-handling accident, and a rupture of a radioactive gas storage tank in the gaseous radioactive waste treatment system.

On the basis of our experience with the evaluations of steam generator tube rupture and the steam line break accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the primary and secondary coolant system radioactivity concentrations so that potential offsite doses are small. At the operating license stage, we will include limits in the Technical Specifications on primary and secondary coolant activities such that the calculated 2-hour doses at the exclusion radius will be small relative to the 10 CFR Part 100 guideline values. Because of the poor diffusion conditions which are currently postulated for this site on the basis of initial onsite measurements, the primary coolant activity may be limited to less than 1 μCi/cc of equivalent I-131.

The radiological consequences of the other accidents presented in Table 11.1-1 are within 10 CFR Part 100 guideline values.

TABLE 11.1-1
CALCULATED OFFSITE DOSES (REM)

ACCIDENT	2-Hour Dose at Exclusion Distance		Course-of-Accident Dose at Low Population Zone	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant Accident	130	23	49	6
Fuel Handling Accident	61	16	15	4
Waste Gas Decay Tank Rupture	-	21	-	5

11.2 Waste Gas Decay Tank Rupture

The waste gas decay tank rupture is postulated to occur in the one (of nine) gas decay tank that has just been filled with the radioactive gases stripped from the reactor primary coolant during refueling. No significant iodine inventory will be in the gas decay tank because of required processing of the gases prior to storage in the tanks. Therefore, only the whole body dose resulting from a postulated release of noble gases was determined. The relative concentration (χ/Q) at the minimum exclusion boundary distance for a short-term release is calculated to be 3.4×10^{-3} sec/m^3, resulting in a potential whole body dose to an individual of 21 Rem. At the low population zone (LPZ) distance, the potential whole-body dose is calculated to be 5 Rem. A technical specification will be set at the operating license stage to restrict tank inventories such that a release resulting from a single active failure (such as the lifting and sticking of a relief valve) will have consequences which are a small fraction of the 10 CFR Part 100 guideline values.

11.3 Fuel Handling Accident

As in other similar facilities, irradiated fuel assemblies will be handled and stored under water. For the postulated fuel handling accident, it is assumed that a fuel assembly is dropped during the handling operation in the spent fuel pit and all the fuel rods fail. The fission product gases from the damaged fuel rods are assumed to be released from the water and collected by the auxiliary building gas treatment system which provides iodine filtration. The Technical Specifications will require the reactor building to be closed and the normal reactor ventilation systems to be operating during fuel handling operations.

In our evaluation of the accident, we assume that the dropped fuel assembly has been removed from a region of the core which has been generating 1.65 times the average core power. We assume that 10% of the noble gases and 10% of the iodine in the damaged fuel rods are released to the pool water, and that 1% of the iodines and all of the noble gases in the pool water are released to the building atmosphere. We assume an effective filter efficiency of 95% for the removal of inorganic iodines by the two 2-inch thick charcoal beds in series and 90% for organic forms. For the 2-hour dose computation we used a relative concentration (χ/Q) at the 1200 meter site boundary of 3.4×10^{-3} sec/m^3. It was assumed that all fission products were released instantaneously. For this accident we calculate that the potential doses at the site boundary are 61 Rem to the thyroid and 16 Rem to the whole body. At the LPZ distance the potential dose is 15 Rem to the thyroid and 4 Rem to the whole body.

11.4 Loss-of-Coolant Accident (LOCA)

The design basis loss-of-coolant accident (LOCA) for the Watts Bar plant is a double-ended break of the largest pipe in the reactor coolant system. The emergency core cooling systems are designed to limit fuel cladding temperatures to well below melting, and to limit fission product release from the fuel. We nevertheless require that the containment and its associated engineered safety features shall be capable of limiting potential doses in conformance to 10 CFR Part 100 guidelines assuming significant releases of fission products from the fuel and using conservative assumptions for the transport of fission products.

Based on the systems described in Section 5.5 and 5.6 which have been provided to process leakage from the containment, and on our current policy with respect to mixing credit for secondary containments, the following model was employed to estimate the potential doses at the site boundary as a result of a postulated loss-of-coolant accident:

1. The Safety Guide No. 4 source term and iodine form fractions were used.
2. An exclusion boundary distance of 1200 meters was used, and the relative concentrations (χ/Q) of fission products at the exclusion boundary and LPZ were based on the 10 months of onsite meteorological data (See Section 2.3).

3. Filter efficiencies for both the auxiliary building gas treatment system and the emergency gas treatment system were assumed to be 90% for organic iodines and 95% for inorganic and particulate iodines (two 2-inch deep charcoal beds in series).
4. The primary containment leak rate was assumed to be 0.25%/day for the first 24 hours and one-half this value for subsequent time periods.
5. Of this primary containment leakage, 10% was assumed to be through-line leakage into the auxiliary building where it was assumed to be filtered by the auxiliary building gas treatment system before being exhausted to the atmosphere. (We will assure this fraction of leakage to the auxiliary building through an appropriate technical specification at the operating license stage).
6. The remaining 90% of the primary containment leakage is assumed to leak to the shield building annulus. During the first 20 minutes, the shield building annulus is exhausted through charcoal filters at an exponentially varying rate, R(t), of from 4000 cubic feet per minute (cfm) to 100 cfm with the difference (4000 - R(t)) being recirculated and mixed in 50% of the shield building annulus free volume. After 20 minutes, the exhaust flow is held constant at 100 cfm; and the recirculation rate is 3900 cfm. It has been assumed that the

initial leakage from the primary containment is directly to the inlet for the emergency gas treatment system and is not mixed with the annulus volume on the first pass. This is consistent with our treatment of BWR secondary containments with recirculation systems.

7. The applicant has stated that there will be no leakage which totally bypasses the shield building annulus and the auxiliary building. This will be insured by maintaining a water seal in the steam generators system in the post accident condition to preclude leakage through the steam lines. Furthermore, the applicant has proposed to maintain the shield building annulus at a negative pressure during operation to preclude leakage which bypasses the filters during the approximate 1/2-minute immediately following the accident before the emergency gas treatment system has been placed in operation. These provisions substantially reduce the dose which would otherwise be computed.

8. The credit allowed for the secondary containment using this model is approximately a factor of 110 for iodines and 7 for the whole-body dose.

With this model, the 2-hour doses at the site boundary are 130 Rem to the thyroid and 23 Rem whole body; and the 30-day doses at the LPZ boundary are 49 Rem to the thyroid and 6 Rem whole body. These limits meet the guideline values of 10 CFR Part 100. The whole-body dose includes the dose from beta radiation, which is essentially a skin dose.

Because of the extremely poor meteorological diffusion conditions for this site, which are reflected in an assumption of very low wind speeds for the accident conditions, lower doses would be calculated if credit were allowed for transit time and decay in the cloud for the dose computed for the first 2 hours after the accident. For example, at a wind speed of 0.25 meters/sec, it would take 1-1/3 hours to reach the 1200-meter exclusion area boundary, resulting in an exposure time of only 2/3 hour with 1-1/3 hour decay. Despite better diffusion at a higher wind speed (smaller χ/Q), a higher dose could be computed for a higher wind speed than 0.25 meters/sec for this model because of the increase in exposure time and decrease in decay time. The shine dose from the cloud while enroute would have to be analyzed before credit could be given for the delay time. To date, the applicant has not addressed this matter in enough detail for us to reach a decision. We do, however, consider the doses as presently computed to be very conservative and acceptable for this site in view of the above.

12.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. The applicant has provided preliminary Technical Specifications for Watts Bar Units 1 and 2 containing an identification and justification for the selection of those variables, conditons, and other requirements that are considered to influence final design. They incorporate the type, number, and capacity of safety-significant components that are known from preliminary design studies and preliminary safety analyses.

We have concluded that the status of the applicant's development of Technical Specifications is adequate for issuance of a construction permit.

13.0 QUALITY ASSURANCE

The Directorate of Regulatory Operations has examined the applicant's quality assurance (QA) program to determine its conformance with the requirements of Appendix B to 10 CFR Part 50, Quality Assurance Criteria for Nuclear Power Plant. The evaluation included; (1) a review of Appendix A to the PSAR, (2) an in-depth review of the quality assurance organization and present quality assurance program procedures, (3) a review of the program as implemented for the Sequoyah Nuclear Plant, (4) an assessment of past performance of the quality assurance organizations at TVA's other nuclear plants, and (5) detailed discussions with the applicant of the findings.

Regulatory Operations found initially that although the QA program conformed in general to the requirements of Appendix B, there were several shortcomings regarding the authority, independence, and staffing of the QA organization.

We discussed these deficiencies further with the applicant and have been informed of certain major organizational changes that are underway which will remove them. For the most part, these improvements will flow from the establishment of a stronger position for QA management directly under the Manager of Engineering Design and Construction.

We intend to follow the implementation of these improvements through the inspections of the Directorate of Regulatory Operations.

We have concluded that with the changes being implemented by the applicant, the QA program is acceptable.

14.0 THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The application for the Watts Bar plant is being reviewed by the ACRS. We intend to supplement this Safety Evaluation when the Committee's report to the Commission relative to its review is available. The supplement will append a copy of the Committee's report and will address the significant comments made by the Committee, and will also describe steps taken by the staff to resolve any issues raised as a result of the Committee's review.

15.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are United States citizens. TVA is a corporate agency of the Federal Government. The applicant is not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

16.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility operating license are Part 33(f) of 10 CFR Part 50 and 10 CFR Part 50, Appendix C. We have reviewed the financial information presented in the application and have concluded that the Tennessee Valley Authority is financially qualified to design and construct the Watts Bar Nuclear Plant. A detailed discussion of the basis for our conclusion is presented in Appendix G.

17.0 CONCLUSIONS

Based on the proposed design of the Watts Bar Nuclear Plant Units 1 and 2, on the criteria, principles and design arrangements for systems and components thus far described that include all of the important safety items, on the calculated potential consequences of routine and accidental releases of radioactive material to the environs, on the scope of the development program which will be conducted, and on the technical competence of the applicant and the principal contractors, and assuming favorable resolution of outstanding unresolved items described above, we have concluded that, in accordance with the provisions of paragraph 50.35(a), 10 CFR Part 50 and paragraph 2.104(b), 10 CFR 2:

(1) The applicant has described the proposed design of the facility, including the principal architectural and engineering criteria for the design, and has identified the major features and components for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration, will be supplied in the final safety analysis report;

(3) Safety features or components, which require research and development have been described by the applicant, and the applicant has identified, and there will be conducted, research and development programs reasonably designed to resolve any safety questions associated with such features or components;

(4) On the basis of the foregoing, there is reasonable assurance that (i) such questions will be satisfactorily resolved at or before the latest date stated in the application for completion or construction of the proposed facility and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

(5) The applicant is technically qualified to design and construct the proposed facility; and

(6) The issuance of a permit for the construction of the facility will not be inimical to the common defense and security or the health and safety of the public.

APPENDIX A

CHRONOLOGY

REGULATORY REVIEW OF TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT

UNITS 1 AND 2

May 18, 1971	Application (PSAR) filed.
August 10, 1971	Letter to applicant requesting updated ECCS analysis.
August 24, 1971	Amendment 1: comparison of Watts Bar and Sequoyah plants.
November 23, 1971	Letter to applicant requesting additional information.
January 5, 1972	Letter to applicant requesting additional information.
January 7, 1972	Letter to applicant requesting additional information.
January 23, 1972	Amendment 2: partial response to AEC letter of November 23, 1971; site and structural matters.
February 11, 1972	Amendment 3: additional responses to AEC letters of November 23, 1971 and January 5, 1972; radwaste, mechanical design, conduct of operations, hydrogen recombiner.
March 14, 1972	Amendment 4: additional responses to AEC letter of November 23, 1971.
March 29, 1972	Amendment 5: final responses to AEC letters of November 23, 1971, January 5, 1972 and January 7, 1972; hydrology, ECCS performance, mechanical design.
March 30, 1972	Amendment 6: financial information.
May 22, 1972	Amendment 7: revisions to quality assurance program.

June 14, 1972	Meeting with applicant concerning system changes to reduce accident doses.
June 20, 1972	Amendment 8: antitrust information.
June 26, 1972	Amendment 9: additional information on the probable maximum flood.
June 29, 1972	Amendment 10: miscellaneous page revisions.
July 7, 1972	ACRS subcommittee meeting at the site.
August 1, 1972	Amendment 11: sensitivity analysis of LOCA calculations, revised information for EGTS and ABGTS designs, mechanical design, I&C and electrical design.
August 14, 1972	Amendment 12: additional financial information.
August 21, 1972	Amendment 13: additional information regarding hydrology, electrical and mechanical design.
August 25, 1972	Amendment 14: additional financial information.

APPENDIX B

U.S. DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
ENVIRONMENTAL RESEARCH LABORATORIES
Silver Spring, Maryland 20910

50-390
50-391

July 18, 1972

R323

Dr. Joseph M. Hendrie
Deputy Director for Technical Review
Directorate of Licensing, USAEC
Washington, D. C. 20545

Dear Dr. Hendrie:

This refers to the letter of June 2, 1971, from R. C. DeYoung, Assistant Director for Pressurized Water Reactors, Division of Reactor Licensing, requesting comments on the following:

Watts Bar Nuclear Plant Units 1 and 2
Tennessee Valley Authority
Preliminary Safety Analysis Report
Volumes 1 through 4 dated 5/18/71

These comments are attached.

Sincerely,

Isaac Van der Hoven

Isaac Van der Hoven, Chief
Air Resources Environmental Laboratory
Air Resources Laboratories

Attachment

cc: E. H. Markee, USAEC

3961

U.S. DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
ENVIRONMENTAL RESEARCH LABORATORIES

Comments on
Watts Bar Nuclear Plant Units 1 and 2
Tennessee Valley Authority
Preliminary Safety Analysis Report
Volumes 1 through 4 dated 5/18/71
Amendment No. 3 dated 2/8/72
Amendment No. 10 dated 6/22/72

Prepared by
Air Resources Environmental Laboratory
National Oceanic and Atmospheric Administration
July 18, 1972

Since the release of effluents to the atmosphere is either from exhaust ducts located on the roofs of the various buildings of the reactor complex or from the buildings themselves, we have assumed an effective ground source in our evaluation of the atmospheric diffusion at the Watts Bar site. The basis for our evaluation was the six months of wind speed and direction data at 30 feet and the temperature difference between 30 and 130 feet above the ground at the site.

The most striking features of the onsite meteorological data are the extremely low wind speeds and the very high frequency of inversion conditions. The average wind speed over the six-month period is 2.7 mph (1.2 m/sec), which is about half the value of any station in the United States as shown on page 74 of the Climatic Atlas of the United States (1). In part, this difference may be explained by the data sample being taken from July through December, when somewhat lower speeds can be expected in comparison to the other half of the year. However, coupled with a 33 percent frequency of calms, the speeds as measured at 30 feet above the ground seem unusually low when compared to the regional average (1) of 6 to 8 mph.

The inversion statistics for 10 months of data (Amendment No. 10) show a 77 percent frequency of temperature lapse rates greater than -0.5°C/100 m and 35 percent greater than +1.5°C/100 m. Furthermore, 10 percent of the time winds at 30 feet were reported as calm (less than 0.6 mph) with lapse rates greater than +1.5°C/100 m.

Extrapolating the joint frequency distribution of the wind speed and diffusion categories presented in Tables Q 2.17-2 through Q 2.17-8 (Revised), we have computed that for the short-term (0-2 hours) release, concentrations at the exclusion distance of 790 m will exceed a value of 3×10^{-3} sec m^{-3} 5 percent of the time. A building wake factor of cA 815 m^2 was assumed.

No onsite data were presented with regard to joint wind persistence and diffusion category. Consequently, we have estimated that for a 24 hour release Type F diffusion with a wind speed of 1 m/sec would prevail over a 22 1/2 degree sector resulting in a concentration of 2×10^{-4} sec m^{-3} at the exclusion distance.

2

For the 30-day, low population zone concentration we have assumed an 80 percent frequency of inversion conditions (Type F and 1 m/sec) and a prevailing wind frequency of 15 percent for winds from the northeast. The resulting value at a distance of 4800 m was 1.5×10^{-6} sec m^{-3}.

For the average annual concentration calculation we have used the diffusion wind roses presented in Figures Q 2.17-1 through 7 (Revised) assuming that the controlling condition occurs 7 percent of the time with winds from the northeast and Type F diffusion at an average speed of 1 m/sec. The average annual concentration at the exclusion distance was 1.4×10^{-5} sec m^{-3}.

In summary, we feel that the onsite meteorological data given in Amendments Nos. 3 and 10 do not agree with the meteorological assumptions given in Table 2.6-9 of the original Preliminary Safety Analysis Report. Our concentration estimates are from a factor of 2 to 8 times higher than those given by the applicant in Table 2.6-10, with the greatest difference being in the short-term release.

Reference

(1) U. S. Dept. of Commerce, 1968 "Climatic Atlas of the United States", ESSA, Environmental Data Service

APPENDIX C

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

RECEIVED
AUG 17 1972
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Mr. L. Manning Muntzing
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

50-390
50-391

Dear Mr. Muntzing:

Transmitted herewith in response to a request by Mr. Richard DeYoung, is a review of the geologic and hydrologic aspects of the Watts Bar Nuclear Plant - AEC Docket Nos. 50-390 and 50-391, proposed by the Tennessee Valley Authority.

The review was prepared by F. A. Kilpatrick and F. M. Byers, Jr. and has been discussed with members of your staff. We have no objections to your making this review a part of the public record.

Sincerely yours,

W. A. Radlinski

ActingDirector

Enclosure

cc: A. J. Pressesky, AEC

Watts Bar Nuclear Plant
Tennessee Valley Authority

AEC Docket Nos. 50-390 and 50-391

This plant site is on the west bank of the Tennessee River on Chickamauga Reservoir at Tennessee River mile 528, 1.9 miles below the Watts Bar Hydroelectric Dam. Chattanooga and Oak Ridge, Tennessee are 50 miles southwest and 40 miles northeast respectively of the plant site. The plant is to consist of two reactors each having a capacity of 3,411 Mwt (1,180 Mwe).

Geology

The analysis of the geology of the Tennessee Valley Authority's Watts Bar Nuclear Site, Units 1 and 2, as presented in AEC Docket Nos. 50-390 and 50-391 and supplements, was reviewed and compared with the available literature. The site was visited on February 1, 1972. In general the analysis appears to present an adequate appraisal of those aspects of the geology that are pertinent to an engineering evaluation of the site.

The proposed nuclear plant at Watts Bar is in the western part of the Valley and Ridge physiographic province of the Appalachian Mountains about 8 miles southeast of the northeasterly trending Eastern Cumberland Escarpment. This is an irregular southeasterly facing escarpment, as much as 1,000 feet high, that marks the physiographic and structural boundary between the Cumberland Plateau to the northwest and the folded and thrust-faulted Appalachians to the southeast.

The geologic structural setting of the proposed Watts Bar Nuclear Plant is closely similar to the Sequoyah Nuclear Plant of TVA, now under construction. Both sites are underlain by the Conasauga Shale of Cambrian age that dips generally southeastward at moderate angles. The regional structural setting is mainly one of imbricate thrust faulting and minor folding involving generally southeasterly dipping lower Paleozoic rocks. The northeasterly trending Kingston thrust fault parallels the attitude of the beds and reaches the surface about 1 mile northwest of the site. None of the thrust faults, however, under either the Cumberland Plateau or the Valley and Ridge province have been active since Paleozoic time.

There are no known active faults or other major geologic structures in the area that are thought potentially capable of localizing seismicity in the immediate vicinity of the site. The nearest probable active fault zone, about 275 miles west of the site, is along the axis of the Mississippi Embayment, a large depositional syncline approximately centered on the Mississippi River (Stearns and Marcher, 1962). Movements on this fault zone probably occurred during the earthquakes of 1811-1812 near New Madrid, Missouri. The faults that cut the Paleozoic rocks of the Valley and Ridge and Cumberland Plateau structural subprovinces ceased movement over 200 million year ago. Although many earth tremors have been felt in the

Southern Appalachian Structural Province, none can be related directly to movement on any of the faults within the area. It is assumed, therefore, that the maximum earthquake intensity previously experienced in the region might also occur again anywhere in the region, including the vicinity of the proposed Watts Bar site.

Bedrock at the plant site consists of about 1,000 feet of the lower third of the Middle Cambrian Conasauga Shale, underlain by at least 2,000 feet of the Lower Cambrian Rome Formation. According to the applicant's report, drill cores showed the Conasauga Shale to be dominantly fine-grained clastics, the limestone content of which ranges from 9 to 25 percent and averages about 16 percent; none of the limestone beds, however, exceed 6 inches in thickness. These figures appear to be representative, based on the examination of drill cores at the site, and on published descriptions of the lower part of the Conasauga Shale as exposed in foundation excavations during construction of the Watts Bar Dam (Fox, 1943, p. 168-169). The underlying Lower Cambrian Rome Foundation, which is partly exposed in the west bank of the Tennessee River at Watts Bar Dam, consists of shale, siltstone and sandstone in order of abundance.

Assuming a general southeasterly dip of about 30° for the Kingston thrust fault and the sedimentary rocks, the Kingston thrust fault underlies the site at a depth of about 3,000 feet; minor isoclinal folds in the Rome Formation adjacent to the fault would increase rather than decrease the apparent thicknesses of these formations underlying the site.

The applicant proposes to site all Class I buildings on unweathered Conasauga Shale, which will require removal of about 40 feet of overburden at the site. Based on the boring logs, the elevation of the weathered top of the shale under the proposed nuclear plant ranges between 688 and 701 feet above sea level. Final foundation grade for the proposed plant will be 690 feet. In situ and dynamic seismic testing by the applicant indicates that the unweathered Conasauga Shale is adequate to support the proposed nuclear plant. Boring logs, however, indicate a weathered zone 1 to 3 feet thick beneath the upper surface of the Conasauga Shale. The top of unweathered shale, therefore, would generally be below foundation grade at holes 21, 29, 36, and 43 (fig. 2.8-59, rev. 1, FSAR). Moreover, it is possible that in some places between drill holes the top surface of unweathered Conasauga Shale may be possibly several feet below proposed foundation grade for Class I structures, owing to a local deeper scour of the former course of the Tennessee River.

Hydrology

Cooling towers are to be utilized for cooling condenser water with about 133 cfs (cubic feet per second) of makeup water to be taken from Chickamauga Reservoir. Minimum flow at the site prior to the construction of dams on the Tennessee River was 2,600 cfs. Water should be adequate since the applicant has specified that flows in excess of 2,000 cfs will be released from Watts Bar Dam.

Plant foundation grade is to be at approximately elevation 690 ft MSL (mean sea level) in the sedimentary rock of the Conasauga Formation. At the site the upper surface of the Conasauga is at about 700 ft MSL. This is overlain by high level terrace deposits about 30 feet thick. Ground water to approximately elevation 720 ft exists in the overlying terrace deposits and hence is above the grade of the plant foundation.

Numerous wells and springs exist within the vicinity of the plant site. The data on well and spring water levels supplied by the applicant indicates that the water table slopes toward Watts Bar Lake. No groundwater users are presently downgradient from the plant. Measures should be taken to prevent the development of any new ground-water supplies downgradient between the plant and the lake.

The applicant should be more specific at the FSAR stage in spelling out the location, type and frequency of sampling of the water environment.

While it is not felt that any problems exist, the applicant's dilution and dispersion analysis of accidentally released liquid wastes into the Tennessee River is inadequate.

The applicant has stated that "inadvertent release from the radioactive liquid waste system to the environment does not occur;" such statements are likely to undermine the credibility of the applicant's other analyses.

At the request of the AEC, the potential, safety related effects of site flooding were not reviewed.

References

Fox, P. P., 1943, Character of the Rome and Rutledge Formations at Watts Bar Dam: Jour. Tenn. Academy Science, v. 18, p. 157-171.

Stearns, R. G., and Marcher, M. V., 1962, Late Cretaceous and subsequent structural development of the northern Mississippi Embayment Area: Tenn. Dept. Conservation and Commerce, Div. Geology, Rept. Inv. No. 18; also pub. in Geol. Soc. Amer. Bull., v. 73, p. 1387-1394.

APPENDIX D

U.S. DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
Rockville, Md. 20852
ENVIRONMENTAL RESEARCH LABORATORIES

50-390
50-391

July 10, 1972

Reply to
Attn of: R1030
809.82

Mr. L. Manning Muntzing
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Muntzing:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of the Watts Bar Nuclear Plant Units 1 and 2 in Rhea County, Tennessee.

If we may be of further assistance to you, please contact us.

Sincerely,

Leonard M Murphy

Leonard M. Murphy
Director, Seismological
Investigations Group

Enclosure

RECEIVED
JUL 11 1972
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

REPORT ON THE SITE SEISMICITY FOR THE WATTS BAR NUCLEAR PLANT UNITS 1 & 2

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismological Investigations Group, NOAA, has evaluated the seismicity of the area around the proposed Watts Bar Nuclear Plant Units 1 & 2 adjacent to the TVA Watts Bar Dam Reservation in Rhea County, Tennessee. The Group has reviewed a similar evaluation submitted to AEC by the Tennessee Valley Authority in its Preliminary Safety Analysis Report and Amendments.

The historical seismic activity considered to have an effect on this site evaluation is the intensity VII (MM) earthquakes that have occurred in the southern half of the Ridge and Valley Province, the intensity VIII (-) (MM) earthquake in Giles County, Virginia, the 1811-1812 series of very large earthquakes near New Madrid, Mo., and the numerous smaller events near Chattanooga, Tenn., and elsewhere in the Appalachian Mountains.

The U. S. Geological Survey report on this site states, "The regional structural setting is mainly one of imbricate thrust faulting and minor folding involving generally southeasterly dipping lower Paleozoic rocks. The northeasterly trending Kingston thrust fault parallels the attitude of the beds and reaches the surface about 1 mile northwest of the site. There is no evidence that thrust faults, under either the Cumberland Plateau or the Valley and Ridge Province, have been active since Paleozoic time."

The geological report continues, "There are no known active faults or other major geologic structure in the area that are thought potentially capable of localizing seismicity in the immediate vicinity of the site."

However, the southern half of the Ridge and Valley Province has experienced earthquake activity throughout the time that historical records have been maintained. The largest event in this region was the Giles County earthquake of May 31, 1897, during which some structural damage (listed as intensity VIII) occurred. Also, there have been three intensity VII events; the January 27-28, 1905, activity near Gadston, Alabama; the March 28, 1913, Knoxville, Tennessee, earthquake; and the October 18, 1916 earthquake near Birmingham, Alabama. In addition, more than 60 earthquakes with intensities from III to VI have occurred with epicenters throughout the Province.

Since the seismic activity in this region cannot be associated with specific structures, it must be assumed that earthquakes with intensities comparable with those characteristics of the southern half of the Ridge and Valley Province might also occur in the vicinity of the plant site.

While the major events of the New Madrid, Mo., and

Charleston, S. C., area were probably felt at the proposed plant site, a repeat of these events is considered to be less of a hazard than the events occurring within the Ridge and Valley Province. The site evaluation and recommendation of acceleration values are premised by the fact that the applicant proposes to locate all Class 1 buildings on unweathered Conasauga shale bedrock.

As a result of this review of the seismological and geological characteristics of the area around the proposed plant site, the Seismological Investigations Group agrees with the applicant that an acceleration of 0.09g, resulting from an intensity VII earthquake, would be adequate for representing the earthquake disturbance likely to occur within the lifetime of the facility. The Group also agrees with the applicant that an acceleration of 0.18g, resulting from an intensity VIII earthquake, would be adequate for representing the ground motion from the maximum earthquake likely to affect the site. It is believed that these values would be adequate for designing protection against the loss of function of components important to safety.

Seismological Investigations Group
Earth Sciences Laboratories
Rockville, Maryland 20852

APPENDIX E

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

10 July 1972

Mr. Edson G. Case, Director
Division of Reactor Standards
U.S. Atomic Energy Commission
Washington, D.C. 20545

Re: Contract No. AT(49-5)-2667
<u>Commentary</u>
<u>Summary Comments</u>
Watts Bar Nuclear Plant Units 1 and 2
Tennessee Valley Authority
AEC Docket Nos. 50-390 and 50-391

Dear Mr. Case:

Dr. N. M. Newmark and I have completed our review of the Preliminary Safety Analysis Report for the Watts Bar Nuclear Plant Units 1 and 2, and are transmitting herewith our Commentary and Summary Comments.

Sincerely yours,

W. J. Hall

W. J. Hall

pg
Enclosures

cc: N. M. Newmark

NATHAN M NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

10 July 1972

COMMENTARY

ON

PRELIMINARY SAFETY ANALYSIS REPORT

FOR

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

TENNESSEE VALLEY AUTHORITY

AEC Docket Nos. 50-390 and 50-391

by W. J. Hall and N. M. Newmark

1. Seismic Hazards

The seismic hazards for which the Watts Bar Nuclear Plant is being designed correspond to a Design Basis Earthquake characterized by a maximum horizontal transient ground acceleration of 0.18g and an Operating Basis Earthquake of half this magnitude. A peak vertical acceleration of two-thirds the horizontal ground acceleration is to be employed in the design and is assumed to act simultaneously with the horizontal acceleration. We concur in this approach.

2. Structural Foundations

A summary of the field exploration program and criteria studies leading to the foundation schemes employed in the construction of the plant is presented in Section 2 of the PSAR. It is indicated there that the foundations for plant structures will be founded on the shales and limestones of the Conasagua Formation.

From the material presented in the PSAR, including inspection of the boring logs and exploratory information, it appears that cavities will not be a problem in these foundations. It is indicated in Appendix 2.8C that a major portion of the plant will be founded 5 to 10 feet below the surface of the weathered rock, which will significantly reduce the settlements to be expected under the structures.

An intake canal southeast of the pumping station is shown in Fig. 2.2-4. We assume that this intake canal is a Class I item and that standard computational and review techniques will be employed. None-the-less, it would be our recommendation that the applicant describe in the FSAR the dynamic analysis approach used to evaluate the dynamic stability and/or liquefaction potential of the slopes when subjected to seismic excitation. The answer to Question 2.19 outlines slope stability which is acceptable.

On page B.2-4 it is indicated that for Class I structures founded upon soil, the surface acceleration will be considered to be amplified or attenuated through the soil. Further information on this point is contained in the answer to Question B.9 and in Section B.2.5. The only Class I structure founded on soil is noted to be the Diesel generator building which rests on 20 ft of soil overlying rock. It is indicated in the answer to Question B.9 that studies of soil amplification will be made and probably employed in the design. It is our belief that the calculation of soil amplification for such a thin layer holds little meaning since the soil is constrained to move with the underlying rock material; thus the approach outlined is acceptable to us so long as the seismic motions equal or exceed those corresponding to the response spectra presented in Appendix B (Amendment 5), Figs. B.2-1 and B.2-2.

3. Seismic Analysis and Design

Seismic Analyses

The containment for each of the reactors is noted to consist of a free-standing steel containment vessel surrounded by a shield building constructed of reinforced concrete. An ice condenser is located within the containment vessel.

The response spectrum method of analysis will be employed for the containment vessel, shield building, and ice condenser, as outlined in Section B.2.4 of Appendix B.

It is noted on page 5.1-26 that due to the method of supporting the ice condenser, and the procedure used in the design of the air locks, there is no significant coupling between the structures above the base slab.

The criteria applicable to other Class I structures, including the auxiliary building, the Diesel generator building, and the intake pumping station, are discussed in Section 5.2 and it is indicated there that the provisions of Appendix B will be applicable for the seismic design.

The load combinations for which the analysis will be carried out are described in Section 5 of the PSAR.

It is indicated on page 5.2-3 that stresses determined by seismic analysis, as described in Appendix B, will be added linearly to the stresses resulting from the analyses for other loadings. In Appendix B it is indicated that horizontal and vertical loadings will be considered to act simultaneously.

Time history analysis may be employed in the design. In the answer to Question B.6, it is noted that the time histories will lead to response spectra which equal or exceed the response spectra specified as design criteria for the plant.

We concur in these approaches.

Response Spectra

The response spectra to be employed in the design are presented in Appendix B, Figs. B.2-1 and B.2-2 (as revised in Amendment 5). We concur in the spectra adopted.

Damping Values

The damping values to be employed in the analysis are summarized in Table B.2-1. We concur in the use of the values given there and to the use of the damping value of 10 percent of critical for soil-supported structures.

Design Stresses

The design stresses for the containment vessel will be governed by Section III of the ASME Pressure Vessel and Boiler Code. In the case of the shield building, the stress criteria are presented in Tables 5.1.4-1 and 5.1.4-2. In the case of other Class I structures, the allowable stress criteria are presented in Table 5.2-1. We concur in the criteria presented.

Junction with Adjacent Buildings

The shield building and the auxiliary building are not structurally connected. Separation is accomplished through a joint between the shield building and the auxiliary building, which is designed to accommodate the deflections of the two buildings, plus an additional thickness to preclude their interaction during the Design Basis Earthquake. The details of the sealing materials to be used below grade and above grade are described on page 5.1-85 (Amendment 7) and we believe the approach described there to be acceptable.

Equipment and Personnel Hatches and Other Penetrations

The discussions beginning on page 5.1-30 et seq. pertain to penetrations of various types and indicate the design is to be carried out to accommodate the thermal and mechanical stresses and for accommodating the differential motions between the containment vessel and the shield building under normal operating and accident conditions, including the Design Basis Earthquake. The general design approach and criteria presented are acceptable.

Piping

The stress criteria for handling the design of the piping are in accordance with Report WCAP 5890 Revision 1. It is noted on page B.3-9 that the damping value to be employed for the DBE will not exceed 1 percent for stresses at or near yield.

The answer to Question B.7 indicates that floor response spectra will be employed in the seismic analysis of the piping, and that relative floor deformations will be accommodated.

We concur in the criteria presented.

Buried Piping

It is noted on page B.3-9 that for underground piping, special studies for buried piping will be carried out; further details on the criteria to be employed are contained in the answer to Question B.5. It is noted in the answer to that question that the piping will be designed to accommodate the relative deflection between structures and to avoid overstress at points where the piping enters major structures. We concur in the approach outlined.

Equipment

Certain aspects of the seismic design approach for equipment are contained in Appendix B and further information is contained in the answer to Question B.8. The design of the equipment will be based in part on the calculated floor response spectra and will include provisions for both horizontal and vertical excitation. The design approach for cranes is presented in several places in Section 5 of the PSAR, and the criteria indicate that the cranes will be designed to resist seismic overturning and dislodgement forces. We concur in the approach outlined.

Class I Controls and Safety-Related Instrumentation

The approach for handling procurement and design of Class I equipment is described briefly in Section 7, Appendix A of the PSAR, and in the answer to Question B.3. The approach outlined generally in the PSAR is acceptable.

REFERENCES

"Preliminary Safety Analysis Report, Watts Bar Nuclear Plant, Tennessee Valley Authority, Vols. 1-5 and Amendments 1 through 5, 7, 9-10", AEC Docket Nos. 50-390 and 50-391.

W. J. Hall

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

10 July 1972

SUMMARY COMMENTS

ON

PRELIMINARY SAFETY ANALYSIS REPORT

FOR

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

TENNESSEE VALLEY AUTHORITY

AEC Docket Nos. 50-390 and 50-391

by W. J. Hall and N. M. Newmark

As a result of our review of the PSAR, including Amendments 1 through 7, we believe that the design criteria described in the PSAR for the Watts Bar Nuclear Plant Units 1 and 2 can be considered adequate in terms of provisions for safe shutdown for a Design Basis Earthquake of 0.18g maximum transient horizontal ground acceleration and capable otherwise of withstanding the effects of an Operating Basis Earthquake of half this intensity.

Our review was based on consideration, among other things, of the seismic adequacy of the structural foundations, seismic analyses, response spectra, damping values, design stresses, junction with adjacent buildings, equipment and personnel hatches and other penetrations, piping, buried piping, equipment, Class I controls and safety-related instrumentation.

We believe that the criteria and procedures presented in the PSAR as being applicable to the design of this plant are in accord with the state-of-the-art and that the design should incorporate an acceptable margin of safety for the earthquake hazards considered.

W. J. Hall

APPENDIX F

United States Department of the Interior

FISH AND WILDLIFE SERVICE

BUREAU OF SPORT FISHERIES AND WILDLIFE

WASHINGTON, D.C. 20240

ADDRESS ONLY THE DIRECTOR, BUREAU OF SPORT FISHERIES AND WILDLIFE

50-390

50-391

RECEIVED NOV 23 1971 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

NOV 2 ? 1971

Mr. L. Manning Muntzing
Director of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Muntzing:

This is in reply to Mr. DeYoung's letter of June 2 which requested our comments on the Preliminary Safety Analysis Report submitted by the Tennessee Valley Authority for the proposed Watts Bar Nuclear Plant, Units 1 and 2, Chickamauga Lake, Rhea County, Tennessee, AEC Docket Nos. 50-390 and 50-391.

TVA prepared a draft environmental statement, dated May 14, on this project. Our comments were requested and the Department of the Interior letter of comment was forwarded to TVA on July 22. We commented that a good discussion of the environmental impacts of the proposed generating plant was presented in the draft statement.

The project will be located at Tennessee River Mile (TRM) 528 on the west shore of Chickamauga Lake 2 miles below Watts Bar and about 8 miles southeast of Spring City, Tennessee. It will utilize two closed cycle pressurized water reactors, each designed for a power output of 3,425 megawatts thermal and a total electrical generating capacity of 1,270 megawatts.

Condenser cooling will be provided by a closed-cycle system including natural draft cooling towers. Makeup water for evaporative losses in the towers, cooling water for plant auxiliaries, and blow-down water (between 55 cfs and 134 cfs) will be withdrawn from the head of a channel feeding from the Chickamauga Reservoir at TRM 528. This water will be held in a 20-acre reservoir prior to use in the cooling system. Return water from the plant auxiliary and blow-down will discharge into this reservoir. The temperature of this water will have a maximum increase of 10° F. Some of the water in the storage reservoir will recycle through the plant and some will be returned to Chickamauga Lake. Some cooling can be expected to occur in the storage reservoir before the water is discharged into the lake. Heat discharged to the reservoir will be less than 1.0 percent of the waste heat discharged by the plant. Water entering the reservoir will meet applicable water quality standards.

8125

Chickamauga Lake and the surrounding area support important fish and wildlife resources. The fishery consists of a variety of game and food fish species, including largemouth bass, smallmouth bass, spotted bass, white bass, channel catfish, blue catfish, flathead catfish, black crappie, white crappie, sauger, bluegill, freshwater drum, buffalo, suckers, and carp.

Fish population surveys conducted in 1970 indicated production to be about 182 pounds per acre. Game and pan fish made up 12 percent of this amount. There is an annual commercial harvest of about 144,000 pounds of fish from Chickamauga Lake. The Watts Bar tail water area is considered favorable spawning habitat for sauger, white bass, and smallmouth bass. This area supported about 6.1 percent of the fishing done in TVA's 12 reservoir tail waters during the period 1965 - 1969. On July 1, 1965, the State of Tennessee designated a 3-mile area downstream from Watts Bar Dam (TRM 526.9) as a mussel sanctuary.

The lake and the surrounding areas, including the Yellow Creek Management Area located one mile from the plant site and the Hiawassee Management Area 27 miles away, support white-tailed deer, gray squirrel, raccoon, wild turkey, ruffed grouse, cottontail rabbit, bobwhites, ducks and geese. Hunting pressure in the vicinity of the project is light to moderate, but is increasing.

In most respects, the monitoring program outlined by TVA is adequate. It is expected that the release of radioactive materials will not exceed allowable limits set by the AEC. The effects of radioactivity on fish and wildlife are poorly understood. Acceptable dose rates and body burdens of radiation for fish and wildlife have not been established. In view of the probable existence of two nuclear generating stations on Chickamauga Reservoir, it is imperative that the monitoring programs be well planned, coordinated, and carefully executed. Toward this end, TVA should include samples of common upland game and waterfowl and some of their foods in the monitoring program. Aquatic plants and animals as well as water and sediments should be sampled within 500 feet of the effluent outfall. Every effort must be made to safeguard these resources. Therefore, we recommend that the monitoring studies be coordinated with the appropriate Federal and State agencies and conducted as planned. These studies also should include:

1. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the effluent outfall.

2. Beta and gamma radioactivity analysis of selected plants and animals as near the effluent outfall as possible.

3. Radioactivity analysis of wildlife and waterfowl samples together with some samples of their foods in the project area.

The project has the potential of affecting the fish and wildlife resources and the environment adversely. We have been concerned about the possibility of damage to aquatic life from the heated effluent; the radioactive wastes and chemicals that will or may be released to the receiving waters; the velocity of the waters approaching the fish screening device on the intake; the movement into and entrapment of aquatic animals in the cooling system; the possibility of terrestrial animals and birds being adversely affected by radioactive effluents of the project; and by the project transmission lines.

These concerns have been allayed for the most part because the TVA has expressed assurance of compliance with all applicable Federal and State regulations and that any unforeseen problems that may become apparent through the environmental and radiological program studies will be corrected. Therefore, we have no objection to the issuance of the construction permit for this project.

Sincerely yours,

Willis King

Assistant
Director

APPENDIX G

FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to the financial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR 50.33(f) and 10 CFR 50 Appendix C. The application of the Tennessee Valley Authority (TVA), as amended, and the accompanying certified annual financial statements provide the financial information required by the Commission's regulations.

The Tennessee Valley Authority is a corporate agency of the Federal Government created by the TVA Act of 1933, as amended. As part of its program, TVA is engaged in the generation, transmission, distribution and sale of electricity. Financially, the power program is separate from the other activities. It is required to be self-supporting and self-liquidating.

We have reviewed the financial information presented in the application and Amendments No. 6, 12, and 14 thereto of the Tennessee Valley Authority to construct two nuclear power units each with an initial net electrical output of about 1,160 megawatts (3,411 Mwt) to be located at its Watts Bar site in Rhea County, Tennessee. Based on this review, we have concluded that the Tennessee Valley Authority is financially qualified to design and construct the proposed facility to be known as the Watts Bar Nuclear Plant, Units 1 and 2.

Our conclusion is based on the following facts and considerations:

1. Applicant estimates the costs of construction of the plant, including certain transmission facilities and other associated costs, and initial

reactor cores will total about $685.0 million. The details of these estimates are contained in the application and are summarized below:

Total nuclear production plant capital costs	$609.9 million
Transmission, distribution and general plant costs	15.1
Nuclear fuel inventory cost for first cores	60.0
	$685.0 million

The Division of Reactor Licensing has reviewed the details of the estimated plant capital costs for construction and has found them to be reasonable.

2. TVA expects to finance the proposed Watts Bar facility from the proceeds of the sale of bonds and notes and from available revenues of the Power Program. The August 1972 estimates of the applicant reflect that construction expenditures for power facilities for the eight-year period 1972-1979, including nuclear fuel inventories, will total about $5,414.0 million, ranging from $667.7 million in 1972 through a low of $486.6 million in 1975 to a high of $905.6 million in 1979. Of this total of $5,414.0 million, the applicant estimates that about 26% will be available for the construction program from current proceeds (internally generated funds) and the remainder of 74% will be financed through borrowings.

3. The Authority has had no apparent problems in the past in financing expansions to the power facilities. It has a borrowing limitation of $5.0 billion established by Congress in 1970. When bonds outstanding

approach this limit, it is expected that legislation will be introduced in Congress to increase this amount. It is a reasonable assumption that the borrowing limitation will be fully adequate to permit issuance of the required bonds and notes. In view of the magnitude of the Power Program's resources, its earnings record, the strength of its financial position, and the high regard held for the Power Bonds as evidenced by Moody's rating of Aaa (gilt-edge) for all issues, there is little question as to the Authority's ability to finance the nuclear facility, including the nuclear fuel inventory buildup.

4. We have examined the certified financial statements of TVA to determine whether it is financially qualified to meet the estimated costs. The information contained in TVA's fiscal year 1971 financial report indicates that operating revenues for 1971 totaled $598.0 million; operating expenses were $449.5 million, of which $80.0 million represented depreciation. The interest on long-term debt was earned 4.0 times; and the net income for the year was $119.0 million, of which $65.1 million was repaid to the U. S. Treasury as a return on the net appropriation investment and the remainder of $53.8 million was transferred to retained earnings. As of December 31, 1971, the TVA's assets totaled $3,352.4 million, most of which was invested in utility plant ($3,183.8 million); retained earnings amounted to $714.7 million. Financial ratios computed from the 1971 statements indicate a sound financial condition, e.g., long-term debt to total

capitalization - .27, and to net utility plant - .46; net plant to capitalization - 1.29; the operating ratio - .75; and the rates of return on proprietary capital - 6.7%, and on total investment - 5.9%. The record of TVA's operations over the past 5 years reflects that operating revenues increased from $326.8 million in 1966 to $598.0 million in 1971; net income increased from $47.9 million to $119.0 million; and net investment in plant from $2,166.6 million to $3,183.8 million. Moody's Investors Service rates the TVA's first mortgage bonds as Aaa (gilt-edge). A copy of the staff's financial analysis of the TVA is attached as an appendix.

TENNESSEE VALLEY AUTHORITY (POWER PROGRAM)

DOCKET NOS. 50-390 AND 50-391

FINANCIAL ANALYSIS

	(dollars in millions) Fiscal Year Ended June 30		
	1971	1970	1969
Debt (including short-term notes)	$1,455.3	$1,096.0	$ 827.7
Utility plant (net)	3,183.8	2,785.1	2,507.7
Ratio - debt to fixed plant	.46	.39	.33
Utility plant (net)	3,183.8	2,985.1	2,507.7
Capitalization	2,460.9	2,424.2	2,120.8
Ratio of net plant to capitalization	1.29	1.15	1.18
Proprietary capital	1,785.9	1,749.2	1,745.8
Total assets	3,352.4	2,933.9	2,632.0
Proprietary ratio	.53	.60	.66
Net income	119.0	74.6	50.7
Proprietary capital	1,785.9	1,749.2	1,745.8
Rate of return on proprietary capital	6.7%	4.3%	2.9%
Net income before interest	196.7	136.9	89.5
Liabilities and capital	3,352.4	2,933.9	2,632.0
Rate of return on total investment	5.9%	4.7%	3.4%
Net income before interest	196.7	136.9	89.5
Interest on long-term debt	48.6	30.7	38.8
No. of times long-term interest earned	4.0	4.4	2.3
Net income	119.0	74.6	50.7
Total revenues	646.2	511.1	419.3
Net income ratio	.18	.15	.12
Operating expenses	449.5	374.2	329.8
Operating revenues	598.0	479.6	403.3
Operating ratio	.75	.78	.82
Utility plant (gross)	4,181.7	3,709.5	3,363.7
Utility operating revenues	598.0	479.6	403.3
Ratio of plant investment to revenues	6.99	7.73	8.34
Annual payment of return on appropriation investment	65.1	57.6	53.1
Annual repayment of appropriation investment	20.0	15.0	15.0

	1971		1970	
Capitalization:	Amount	% of Total	Amount	% of Total
Power bonds	$ 675.0	27.4%	$ 675.0	27.8%
Proprietary capital	1,785.9	72.6%	1,749.2	72.2%
Total	$2,460.9	100.0%	$2,424.2	100.0%

Moody's Bond Ratings: Aaa