Safety Evaluation Report

related to operation of

Sequoyah Nuclear Plant Units 1 and 2

Tennessee Valley Authority

NUREG-0011

U. S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

Docket Nos. 50-327
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NUREG-0011
March 1979

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

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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 October 15, 1968, as amended, requested a license to construct and operate the Sequoyah Nuclear Plant, Units 1 and 2 (Sequoyah Nuclear Plant or, the facility), at a site in Hamilton County, Tennessee, on the west bank of Chickamauga Lake, approximately 9.5 miles northeast of Chattanooga, Tennessee.

The Atomic Energy Commission, now Nuclear Regulatory Commission (Commission), reported the results of its review prior to construction in a Safety Evaluation Report dated March 24, 1970. Following a public hearing before an Atomic Safety and Licensing Board, Provisional Construction Permits Nos. CPPR-72 and CPPR-73 were issued on May 27, 1970, for Units 1 and 2 respectively.

The applicant tendered an application for operating licenses for the Sequoyah Nuclear Plant by letter dated December 3, 1973. After our acceptance review, the application was docketed on January 31, 1974.

Our technical safety review with respect to issuing an operating license for the plant has been based on the Final Safety Evaluation Report and Amendments 15 through 58 to the application, all of which are available for review at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. and at the Chattanooga-Hamilton County Bicentennial Library, 1001 Broad Street, Chattanooga, Tennessee. In the course of our review, we have held a number of meetings with representatives of the applicant and the nuclear steam system supplier to discuss plant design, construction, and proposed operation. As a consequence, additional information was requested, which the applicant provided in amendments to the Final Ssafety Analysis Report. A chronology of the principal actions relating to the processing of the application is attached as Appendix A to this Safety Evaluation Report.

This Safety Evaluation Report summarizes the results of the radiological safety review of the Sequoyah Nuclear Plant performed by the staff.

In accordance with the provisions of the National Environmental Policy Act of 1969, Draft and Final Environmental Statements, which set forth the considerations related to the continuation of construction and the proposed operation of the Sequoyah Nuclear Plant, were prepared by TVA, as the lead Federal agency, and reviewed by the staff. The Final Environmental Statement was issued on February 13, 1974.

The review and evaluation of the Sequoyah plant for operating licenses is only one stage in the continuing review by the staff of the design, construction and operating features of the facility. The proposed design of the facility was reviewed as part of the construction permit review. Construction of the facility has been monitored in accordance with the inspection program of the staff. At this, the operating license review stage, we have reviewed the final design to determine that the Commission's safety requirements have been met. If operating licenses are granted, the Sequoyah Nuclear Plant must be operated in accordance with the terms of the operating licenses and the Commission's regulations and will be subject to the continuing inspection program of the staff.

1.2 General Plant Description

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

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

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

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

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

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

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

The emergency core cooling system for the plant consists of accumulators, upper head injection, and both high and low pressure injection subsystems with provisions for recirculation of the borated water after the end of the injection phase. Various combinations of these features will assure core cooling for the complete range of postulated coolant pipe break sizes.

The two nuclear steam supply systems are each housed in a separate free standing steel containment structure within a reinforced concrete shield building. The containment employs the ice condenser pressure-suppression concept. A common auxiliary building located adjacent to the containment structure for Units 1 and 2 houses the radioactive waste treatment facilities, components of the engineered safety features, and various related auxiliary systems for each unit. Both units share a common fuel handling facility which contains a spent fuel pool and a new fuel storage facility.

The plant is supplied with electrical power by independent transmission lines from offsite power sources and is provided with independent and redundant onsite emergency power supplies capable of supplying power to shut down the plant safely or to operate the engineered safety features in the event of an accident.

1.3 Comparison with Similar Facility Designs

Many features of the design of the Sequoyah Nuclear Plant are similar to those we have evaluated and approved previously for other nuclear plants now under construction or in operation. To the extent feasible and appropriate, we have relied on our earlier reviews for 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 these other facilities have been published and are available for public inspection at the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C.

1.4 Identification of Agents and Contractors

The Westinghouse Electric Corporation (Westinghouse) is supplying the nuclear steam supply system, including the first fuel loading, and the turbine-generators. TVA is responsible for the design of the remainder of the plant, and all other architect-engineer functions, and for the construction and operation of the plant.

Principal consultants utilized by TVA to perform selected design work and other specialized services include Western Geophysical Engineering, Inc. for soil foundation dynamic analyses, Engineering Data Systems, Inc. for seismic analysis of piping, Chicago Bridge and Iron Company for design and construction of the free-standing steel containments, and Pressay Corporation for certification of material for containment flexible seals.

1.5 Summary of Principal Review Matters

The evaluation performed by the staff included a review of the information submitted by the applicant, particularly with regard to the following matters:

We evaluated the population density and 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 had been given appropriate consideration in the final design of the plant, 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.

We evaluated the design, fabrication, construction, and testing and 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, Regulatory Guides, and other appropriate rules, codes, and standards, and that any departure from these criteria, codes, and standards has been identified and justified.

We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of 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. Conservative analyses were performed 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.

We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operations, including the proposed organization, staffing and training program, the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to safely operate the plant.

We evaluated the design of the systems provided for control of the radiological effluents from the plant to determine that these systems are capable of controlling the release of radioactive wastes from the facility within the limits of the Commission's regulations, and that the equipment provided is capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as reasonably achievable.

We will evaluate the financial position of the applicant to determine that the applicant is financially qualified to operate the Sequoyah Nuclear Plant, and will report on this matter in a supplement to this Safety Evaluation Report.

1.6 Outstanding Issues

We have identified outstanding issues in our review which have not been resolved with the applicant. We will complete our review of these items prior to issuance of an operating license, and will discuss the resolution of each of these items in a supplement to this report. These items are listed below and are discussed further in the sections of this report as indicated.

1. Bolted Connections in Component Supports (Section 3.9.2)

   The applicant has not yet furnished requested information on bolted connections in linear component supports in safety-related systems regarding support plate flexibility considerations in determining maximum bolt loads. We will report on our evaluation of this matter when the information is available.

2. Seismic Qualification of Instrumentation and Electrical Equipment (Sections 3.10, 7.2.2, 7.8.1)

   We have not yet completed our review of the Westinghouse-supplied Class 1E instrumentation and electrical equipment. For balance of plant equipment, confirmatory information is required on containment isolation valve motor operators. We will report further on this matter in a supplement to this report.

3. Fire Protection (Section 9.5)

We have not yet completed our review of the applicants fire protection program. We will complete this review prior to issuance of an operating license and will condition the operating license to assure implementation of all required modifications. We will report further in a supplement to this report.

4. Radiological Emergency Plan (Section 13.3)

The applicant has not yet provided responses to our request for additional information on this matter. All issues will be resolved prior to issuance of an operating license, and we will report further in a supplement to this report.

5. Acceptance Criteria for Plant Trip Test (Section 14.0)

The applicant has not yet provided information we requested on acceptance criteria for the turbine trip and generator load reject portions of the plant trip test from 100 percent power. We will report further in a supplement to this report.

1.7 Confirmatory Issues

As a result of our review, there are a number of matters for which we have completed our review and have determined positions which are acceptable to the staff and for which there appears to be no significant disagreement between the applicant and the staff. The applicant has been advised of our positions and we are awaiting confirmation of the applicant's commitment to comply with these positions and to provide appropriate information. These items will be reported in a supplement to the Safety Evaluation Report. These items, with appropriate references to subsections of this report, are stated below.

1. Single Failure in the Residual Heat Removal System (Section 5.3.2)

The applicant has not yet provided formal documentation of its agreement to provide a dedicated operator to monitor flow to the residual heat removal pumps during decay heat removal operations, pending installation of a flow alarm (See section 1.8 below).

2. Pressure-Temperature Limits for Heatup and Cooldown (Section 5.2.3)

The applicant has not yet provided confirmation of its statement that the proposed pressure-temperature limits for reactor vessel heatup and cooldown use an acceptable prediction for temperature shift.

3. Inservice Inspection of Steam Generator Tubes (Section 5.2.6)

The applicant has not yet provided formal documentation of an inservice inspection program for the steam generator tubes. We will verify that an acceptable program is in place, and will report further on this matter in a supplement to this report.

4. Cold Shutdown Using Safety-Grade Equipment (Section 5.3.2)

The applicant has discussed with us the capability of the system to achieve cold shutdown using only safety-grade equipment and will provide appropriate confirmatory documentation. We will report further in a supplement to this report.

5. Design of Steam Generator and Pressurizer Supports (Sections 3.9.1, 6.2)

The applicant has not yet confirmed the assumption that, as in other plants, the pressure response to line breaks in the steam generator and pressurizer subcompartments has been utilized in evaluating the design of the equipment supports. We will report further in a supplement to this report.

6. Containment Response to Steam Line Break and Environmental Qualification of Westighouse Equipment (Sections 6.2.1, 7.2.2, 7.8.2)

Westinghouse has indicated that the containment temperature response to the small line break already analyzed will bound the response for the additional breaks we have requested be examined, but the applicant has not yet provided confirmatory information. Additional information is also forthcoming on environmental qualification of Westinghouse equipment. We will report further in a supplement to this report.

7. Upper Head Injection Preoperational Tests (Section 6.3.4)

The applicant has not yet submitted confirmatory documentation on tests already performed which reportedly demonstrated acceptable flow performance of the upper head injection system. We will report further in a supplement to this report.

8. Containment Sump (Section 6.3.4)

In fulfillment of the applicable requirements of Regulatory Guide 1.79, the applicant has performed scale model tests of the containment emergency sump performance and submitted reports which we have reviewed. The applicant has not yet responded formally to our requests for additional information to verify sump performance in the event of certain line breaks. We will report fully on these matters in a supplement to this report.

9. Bypassed Safety Injection Signal (Section 6.3.5)

The applicant has indicated that sufficient time is available to respond effectively to postulated line breaks in the residual heat removal system when in the normal shutdown cooling mode when the safety injection signal is blocked, but has not yet provided information verifying actions required and time available. We will report further on this matter in a supplement to this report.

10. Loss-of-Coolant Accident Analysis (Sections 6.3.5, 15.3.2)

We have reviewed the loss-of-coolant accident analysis provided by the applicant and have requested information confirming that the most limiting case has been analyzed. We will report further in a supplement to this report.

11. Response Time Testing (Section 7.2.2)

The applicant has committed to measure channel response time including the sensors, but has not yet submitted the confirmatory information requested to assure acceptable implementation of this commitment.

12. Isolation Valve Interlocks and Position Indication (Section 7.3.2)

The applicant has not yet submitted documentation to confirm verbal information that position indication of two safety-related valves will be maintained when power is removed from the valves.

13. Post Accident Monitoring Separation Criteria (Section 7.5.2)

The applicant has not yet provided information varifying implementation of agreed criteria for separation and independence of post-accident monitoring channels.

14. Environmental Qualification of Balance-of-plant Equipment (Section 7.8.2)

The applicant has not yet provided confirmatory information on an environmental monitoring system or on the correction of errors in several tables in the Final Safety Analysis Report.

15. Diesel Generator and Remote Shutdown Testing (Section 14.0)

We require that the applicant perform tests in accordance with regulatory guides covering diesel generators and remote shutdown capability, or provide justification for exceptions to these guides. Confirmatory information has not yet been provided by the applicant. We will report further on this matter in a supplement to this report.

16. Boron Dilution (Section 15.2)

The applicant has not yet provided documentation confirming his procedures associated with alarm setpoints for the high flux alarm which provides protection against a boron dilution event during startup or shutdown.

17. Long Term Effects of Steam Line Break (Section 15.33)

The applicant has not yet provided information requested to verify operator actions related to long-term reactor vessel repressurization.

1.8 Staff Positions - Licensing Conditions

The staff has taken positions on certain issues requiring implementation and/or documentation after issuance of an operating license. The license will be conditioned as necessary to assure acceptable implementation of our positions. These items are listed below and are discussed further in the sections of this report as indicated.

1. Seismic Design of Structures and Components (Section 2.5)

   The operating license will be conditioned to require evaluations showing margins available in structures and components to function during and after a design earthquake.

2. Inservice Testing After Commercial Operation (Section 3.9.1)

   The operating license will be conditioned to assure implementation of an acceptable inservice testing program for pumps and valves after commercial operation.

3. Reactor Vessel Overpressurization (Section 5.2.2)

   If equipment is not installed prior to initial fuel load to protect against startup and shutdown overpressurization transients, the operating license will be conditioned as necessary to require installation of such equipment at a later date. The applicant must provide acceptable justification for operation prior to installation of such equipment.

4. Loose Parts Monitor (Section 5.2.8)

   We require installation of an acceptable loose parts monitoring system before initiation of startup testing after the initial fuel loading.

5. Flow Alarm in Residual Heat Removal System (Section 5.3.2)

The operating license will be conditioned to assure installation of a flow alarm to indicate loss of flow in the suction line to the residual heat removal pumps prior to startup following the first refueling outage.

6. Instrument Trip Setpoints (Section 7.2.7)

The operating license will be conditioned to assure receipt of requested information on the determination of instrument trip setpoints.

7. Effect of Power Transients on Safety Related Equipment (Section 7.3.2)

The operating license will be conditioned to require provision of an additional level of under- and over-voltage protection prior to startup following the first refueling outage.

1.9 Generic Issues

The Advisory Committee on Reactor Safeguards periodically issues a report listing various generic matters applicable to light water reactors. A discussion of these matters is provided in Appendix C to this report which includes references to sections of this report for more specific discussions concerning this facility.

The Nuclear Regulatory Commission staff continuously evaluates the safety requirements used in its review against new information as it becomes available. In some cases immediate action or interim measures are taken by the staff to assure safety. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing staff requirements should be modified. These issues being studied are sometimes called generic safety issues because they are related to a particular class or type of nuclear facility. A discussion of our program for the resolution of these generic issues will be presented in a supplement to ths report.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Description and Exclusion Area Control

The Sequoyah Nuclear Plant is located in Hamilton County in southeastern Tennessee approximately 17 miles northeast of the center of Chattanooga, Tennessee. The site occupies a 525-acre tract of land on a peninsula on the western shore of Chickamauga Lake, a reservoir formed by the Chickamauga Dam on the Tennessee River. A general map of the region is shown in Figure 2-1.

The exclusion area consists of the site property plus small inlets of Chickamauga Lake which penetrate the site property. The minimum exclusion area boundary distance is 556 meters (1,824 feet) measured from the reactor building nearest the exclusion area boundary to the nearest point on the boundary. The site exclusion area is shown in Figure 2-2 along with the principal plant structures.

All of the land within the exclusion area, including the mineral rights, is owned by the United States and is in the custody of the applicant. There are no residences or designated recreational areas within the site boundary. No railroads or major highways traverse the exclusion area. Two rural county roads run just inside and adjacent to the western boundary of the exclusion area. The applicant has made arrangements with local and state authorities to control traffic on these roads in case of an emergency. The applicant has also made arrangements with the Tennessee Division of Water Safety for assistance in clearing the water areas within the exclusion area in case of an emergency. These arrangements are described in the site Radiological Emergency Plan. A visitor's center and a TVA training center will be located within the exclusion area. All activities associated with these facilities will be under the applicant's control.

We conclude that the applicant has the authority to determine all activities within the exclusion area and has made appropriate arrangements to control traffic within the exclusion area in case of an emergency, as required by 10 CFR Part 100.

2.1.2 Population and Population Distribution

Approximately 32,000 people resided within 10 miles of the site in 1970. The majority of this population is located in the southwest through northwest directions from the site and is attributed primarily to the influence of suburban Chattanooga and the town of Soddy-Daisy which is located about six miles west of the site. The applicant projects that future population growth within 10 miles of the site will be concentrated in the same area.

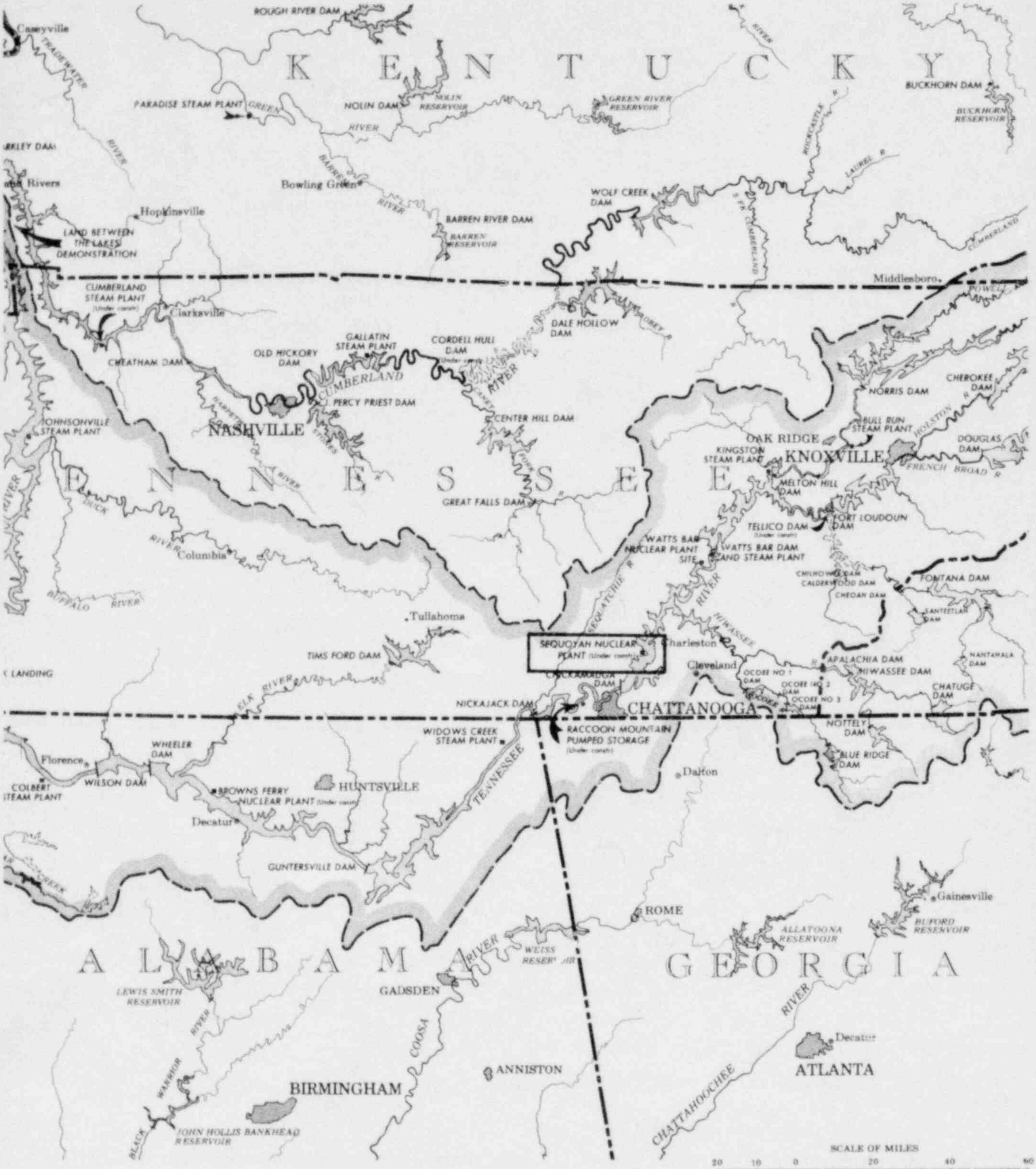

FIGURE 2-1 GENERAL SITE LOCATION

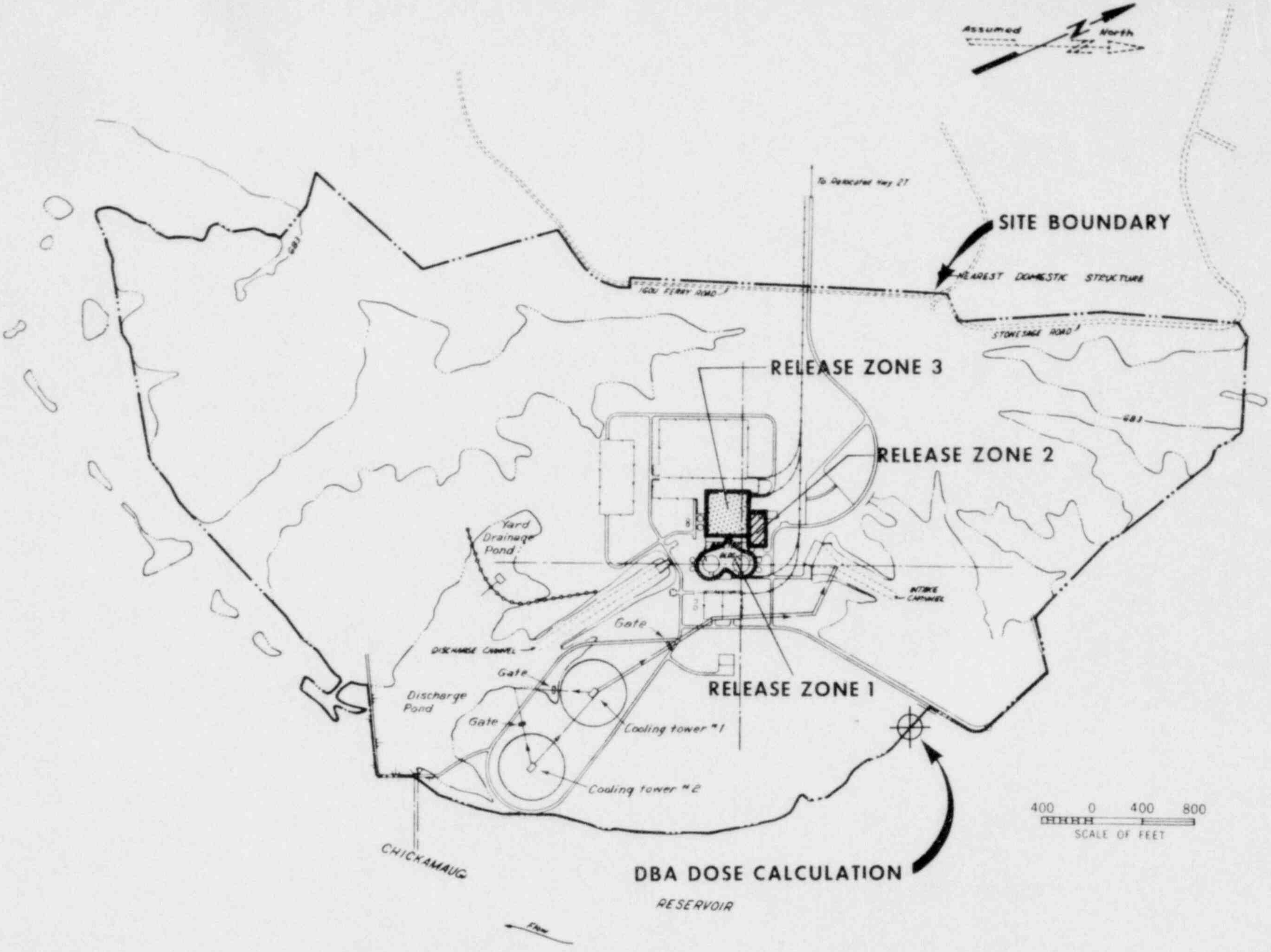

FIGURE 2-2 SITE EXCLUSION AREA BOUNDARY

2-3

The total 1970 resident population within 50 miles was 659,000 and the applicant projects that the 50-mile population will increase to 736,000 by 1980 and 1,057,000 by 2010. This corresponds to a population increase of 12.5 percent per decade. In order to verify the applicant's population data, we obtained an independent estimate of the 1970 population within 50 miles of the site from U. S. Bureau of the Census data and compared this value to the applicant's 50-mile population figure for 1970. We found that the U. S. Bureau of the Census value of 688,178 agreed reasonably well with the applicant's value of 659,015. We also compared the applicant's population projections to the population projections of the U. S. Bureau of Economic Analysis for Economic Area 48, an area comprising metropolitan Chattanooga and the surrounding counties in Tennessee, northwestern Georgia, and northeastern Alabama. This comparison showed that the applicant's growth projection of 12.5 percent per decade was in close agreement to the regional growth projection of 12 percent per decade made by the U. S. Bureau of Economic Analysis.

The applicant has selected a low population zone with an outer radius of three miles. The population within the low population zone in 1970 was determined by the applicant to be 2,005 persons. The applicant projects little or no growth in population within the low population zone over the lifetime of the plant. On the basis of our review and analysis of site information and the proposed emergency plans submitted by TVA (See Section 13.3 of this report), we find that there is reasonable assurance that appropriate protective measures can be taken in behalf of the persons within the low population zone in the event of a serious plant accident.

The nearest densely populated center containing more than about 25,000 residents is Chattanooga, Tennessee, which had a population of 223,580 in 1970. The nearest boundary of the Chattanooga urban area based upon consideration of the population distribution is approximately 7.5 miles southwest of the site. This distance meets the requirement in 10 CFR Part 100 that the population center distance be at least one and one-third times the distance to the outer boundary of the low population zone.

2.1.3 Conclusions

On the basis of the 10 CFR Part 100 definitions of the exclusion area, low population zone, and population center, and the calculated radiological consequences of postulated design basis accidents presented in Section 15.4 of this report, we conclude that the exclusion area, low population zone, and population center specified for the Sequoyah Nuclear Plant meet the requirements of 10 CFR Part 100 and are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

There are no industrial facilities within five miles of the site which pose a potential threat to the safe operation of the Sequoyah Nuclear Plant. The nearest facility of significance is the Volunteer Army Ammunition Plant which is located eight miles south-southwest of the plant site. The distance is adequate to insure that the occurrence of any credible accident at the Volunteer Army Ammunition Plant will not adversely affect the Sequoyah Nuclear Plant. The applicant states that explosives are transported from the Volunteer Army Ammunition Plant by rail or truck only and that no explosives are transported by barge. The closest railroad over which explosives could be transported is the Southern Railway which passes through the town of Soddy-Daisy about 5-1/2 miles west of the plant site. The applicant states that the only highways over which explosives are transported are Interstate 75, U.S. Highway 64, and U.S. Highway 411, and that the closest of these highways, Interstate 75, is over seven miles from the plant site. Other highways in the vicinity of the site are Tennessee Highway 58, about three miles southeast, and U.S. Highway 27, about five miles west of the plant site. An explosion of the maximum amount of explosives which could be carried at any given time over any of these transportation routes would not be of sufficient magnitude to damage safety-related systems and structures of the facility.

The Tennessee River adjacent to the site is a navigable waterway used for the transportation of bulk cargoes by barge. Chlorine has been identified as a hazardous material which is shipped by barge past the site. The plant is protected against an accidental chlorine release by chlorine detectors in the control room air intake which will alarm and automatically isolate the control room air intakes in the event chlorine is detected (see Section 6.4 of this report for further discussion of control room habitability). Gasoline was shipped by barge past the site until construction of a pipeline in 1974 to supply Knoxville eliminated all gasoline barge traffic.

The new essential raw cooling water intake structure will be protected against barge collisions by a dike which will be constructed on the upstream side of the intake structure, and by the skimmer wall on the downstream side. The dike will provide protection for river levels up to an elevation of 705 feet above mean sea level, an elevation 22 feet above the normal pool elevation. Lock operation and hence all river navigation ceases at a flood level lower in elevation then the dike. The applicant has computed that the probability of a flood exceeding an elevation of 705 feet above mean sea level in combination with a drifting barge striking the intake structure is on the order of 4×10^{-8} per year. The location of the essential raw cooling water intake structure on the inside bend of the river will also tend to reduce the likelihood of river traffic collisions with the intake structure. We agree with the applicant that the probability of a river barge causing unacceptable damage to the essential raw cooling water intake structure is sufficiently low that such an event need not be considered as a design basis for the plant.

The nearest pipeline to the plant is a six-inch diameter natural gas distribution line located about four miles west of the site. At this distance a pipeline accident would not affect the operation of the nuclear facility.

The nearest airfield to the plant is Dallas Bay Skypark, a private facility used primarily by private general aviation aircraft, which is located about five miles southwest of the site. Chattanooga Municipal Airport, the nearest airport with scheduled commerical operations, is located 14.5 miles south-southwest of the plant site. Federal Airway V 333 passes over the site area. Airways identified by the "V" designation are low altitude airways which occupy the airspace up to 18,000 feet. The applicant states that aircraft on V 333 in the vicinity of the site operate at a minimum altitude of 4,000 feet. The applicant further states that the peak day traffic recorded on this airway was 15 flights. Based on plant sites reviewed in the past which met our criteria and which had equivalent aircraft traffic in equal or closer proximity, we conclude that aircraft activity near Sequoyah will not adversely affect safe operation of the facility.

We conclude that the nearby industrial, transportation, and military activities in the vicinity of the Sequoyah Nuclear Plant have been properly identified and evaluated and that, with regard to potential accidents which may occur as a result of these activities, the plant is adequately protected and can be operated with an acceptable degree of safety.

2.3 Meteorology

2.3.1 Regional Climatology

The applicant has provided a sufficient description of the regional meteorological conditions of importance to the safe design and siting of the Sequoyah Nuclear Plant.

The Sequoyah site, located in southeastern Tennessee along the Tennessee River, is in an area of complex topography which can result in marked variations in local wind characteristics. The wind pattern within the Tennessee River Valley in the area of the site is distinctly bimodal, northeasterly down-valley and southwesterly up-valley The climate is generally moderate, influenced during much of the year by the anticyclonic circulation of the Azores-Bermuda high pressure system. The site lies near the path of winter cyclones generated along the western edge of the Appalachian Mountains. This circulation pattern results in cold, dry continental air masses predominating during the winter, with the cool periods occasionally broken by warm, moist air pressing northward from the Gulf of Mexico. As a result of the winter storm track and contrasts between alternating air masses, over 40 percent of the normal annual precipitation occurs from December through March. Summers are warm and humid with frequent afternoon thunderstorms.

2.3.2 Local Meteorology

Climatological data from Chattanooga, Tennessee (about 15 miles south-southwest of the site), the TVA rain gauge network, and available onsite data have been used to assess local meteorological characteristics of the site.

Mean monthly temperatures may be expected to range from about 42 degrees Fahrenheit in January to about 81 degrees Fahrenheit in July. Extreme temperatures reported at Chattanooga have been 100 degrees Fahrenheit and -10 degrees Fahrenheit.

Precipitation is primarily associated with the winter and spring seasons, with 56 percent of normal annual precipitation of 57.7 inches occurring from December through May. The maximum precipitation in 24 hours was about 7.6 inches. Average annual snowfall at Chattanooga is 4.5 inches.

Wind data from the 33-foot level of the onsite meteorol gical tower for the period January 1972 through December 1975 indicate the distinc. n biomodal wind characteristics of the river valley site location. Winds from the north-northeast and northeast directions occur about 28 percent of the time, and winds from the south-southwest and southwest direction occur about 29 percent of the time. Winds from the east-southeast direction occur least frequently at 0.9 percent.

Thunderstorms are most frequent in June, July, and August, which account for about 56 percent of the 55 thunderstorm days expected annually.

During the period 1955-1967, 15 tornadoes were reported in the one-degree latitude-longitude square containing the site giving a mean annual tornado frequency of 1.2. The computed recurrence interval for a tornado at the plant site is 1200 years. The "fastest mile" wind speed recorded at Chattanooga was 82 miles per hour.

In the period 1936-1970, about 80 atmospheric stagnation cases totalling about 300 days were reported in the site area. About 10 of these cases lasted seven days or more.

2.3.3 Onsite Meteorological Measurements Program

The onsite meteorological measurements program, operational since April 1971, consists of a 300-foot tower, located about 4000 feet southwest of the plant at an elevation about 50 feet above plant grade. Wind speed and direction are measured at 33, 150 (since 1976), and 300 feet. Ambient temperature is measured at 4, 33, 150, and 300 feet, and dewpoint temperature is measured at 4 feet and 33 feet (since 1976). Precipitation, solar radiation, and barometric pressure are measured at four feet. Prior to 1975, the measurements of vertical temperature gradient were based on only one measurement per hour. TVA performed a correlation study over a one-year period (May 1975-April 1976) to examine differences between the vertical temperature gradient determined using only one measurement per hour and that determined using a longer averaging time (15 minutes or more) more representative of an hourly average value. The study showed that the resulting atmospheric stability distribution developed using each technique varied only slightly over an annual cycle. For example, about one percent more extremely-stable atmospheric stability conditions were indicated by the one-measurment-per-hour technique than by using the longer averaging time. About one percent less moderately-stable atmospheric conditions were indicated by the one-measurement-per-hour technique.

There was also an indication that the accuracy of the determination of vertical temperature gradient did not meet the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs." TVA has examined the entire data collection system and determined the measurement of vertical temperature gradient is in conformance with the accuracy specified in Regulatory Guide 1.23.

The operational meteorological measurements program will include realtime displays in the control room of wind speed and wind direction measurements from the 33-foot and 150-foot levels, and vertical temperature gradient measurements between the 33-foot and 150-foot levels and between the 33-foot and 300-foot levels. These data will also be transmitted to a meteorological forecast center in Muscle Shoals, Alabama, for use in a meteorological forecast program as support to the radiological emergency plan. The implementation of a meteorological forecast program to provide additional information for incident response is an interesting innovation, and for this reason we expect to review the documentation of the procedures for such a program. As stated in Amendment 47 to the Final Safety Analysis Report, these procedures will be developed prior to initial fuel loading.

The applicant has submitted four years (January 1972 - December 1975) of onsite meteorological data in the form of joint frequency distributions of wind speed and wind direction by atmospheric stability for use in evaluating short-term (accident) and long-term (routine) atmospheric dispersion characteristics. Wind speed and wind direction were measured at the 33-foot level, and atmospheric stability was defined by the vertical temperature gradient measured between the 33-foot and 150-foot levels. The joint frequency distributions also had additional light wind speed classes to more accurately reflect the distribution of light wind speed conditions at this site. We conclude that this measurement program is consistent with the recommendations of Regulatory Guide 1.23.

2.3.4 Short-Term (Accident) Diffusion Estimates

Conservative assessments of atmospheric diffusion conditions for evaluating accidental releases of radoactivity from buildings and vents have been made by us from the applicants meteorological data (see 2.3.3 above) and appropriate diffusion models.

In the evaluation of short-term (0-2 hours at the exclusion distance) accidental releases from buildings and vents, a ground level release with a building wake factor, cA, of 850 square meters was assumed. The relative concentration (X/Q) value for the 0-2 hour time period which is exceeded five percent of the time was calculated, using the model described in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors" (Revision 2, June 1974), to be 1.4×10^{-3} seconds per cubic meter at the exclusion distance of 556 meters. This relative concentration is equivalent to that calculated using Pasquill Type F stability with a wind speed of 0.5 meters per second.

The relative concentration values for various time periods at the outer boundary of the low population zone (4828 meters) are:

Time Period	X/Q seconds per cubic meter
0-8 hours	6.4×10^{-5}
8-24 hours	4.5×10^{-5}
1-4 days	2.1×10^{-5}
4-30 days	6.9×10^{-6}

We also examined relative concentration values calculated using the atmospheric dispersion model described in Regulatory Guide 1.4, modified to incorporate the results of recent atmospheric tracer tests and to consider atmospheric dispersion conditions as a function of direction. Specifically, the modified dispersion model considers the following effects:

(1) Lateral plume meander as a function of atmospheric stability, wind speed, and distance from the source, during periods of light winds and neutral and stable atmospheric conditions,

(2) Boundary distance as a function of direction from the plant,

(3) Atmospheric dispersion conditions when the wind is blowing in a specific direction, and

(4) The fraction of time that the wind can be expected to blow into each of the 16 compass directions.

The highest relative concentration value for the 0-2 hour time period calculated using the modified atmospheric dispersion model and assuming a circular site boundary of 556 meters was about 40 percent lower than that calculated using the model described in Regulatory Guide 1.4. Similarly, the highest relative concentrations for the various time periods at the outer boundary of the low population zone calculated using the modified atmospheric dispersion model are also lower (20 percent or less).

2.3.5 Long-Term (Routine) Diffusion Estimates

Reasonable estimates of annual average atmospheric dispersion conditions used in evaluating atmospheric transport and dispersion characteristics for routine releases of radioactivity have been made using the atmospheric dispersion model presented in NUREG-0324, "Program for Meteorological Evaluation of Routine Effluent Releases at

Nuclear Power Stations," which is based on the "Straight-Line Trajectory Model" described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors" (Revision 1, July 1977). All releases at the Sequoyah site were considered as ground-level, with adjustments for mixing in the building wake. An estimate of increase in calculated relative concentration values due to spatial and temporal variations in airflow, not considered in the straight-line model, was included as presented in NUREG-0324. The calculations also included consideration of intermittent releases during more adverse atmospheric conditions than indicated by an annual average as a function of total duration of release (see NUREG-0324).

2.3.6 Conclusions

We conclude that the four years of onsite meteorological data provided by the applicant are an acceptable basis for our assessment of atmospheric dispersion characteristics at the Sequoyah site. The applicant has modified the procedure for determining a representative hourly average measurement of vertical temperature gradient. The data collected using only one measurement of vertical temperature gradient per hour are reasonable and are considered acceptable for input into calculations of relative concentrations.

The relative concentration values presented in Section 2.3.4 are conservative when compared to those calculated using an atmospheric dispersion model that considers lateral plume meander during certain conditions, boundary distance as a function of direction from the plant, atmospheric dispersion conditions as a function of wind direction, and wind direction frequency.

The proposed control-room display of real-time wind speed and direction measurements from two levels (33 feet and 150 feet) and of real-time vertical temperature gradient measurements between two intervals (33 feet and 150 feet and 33 feet and 300 feet) provides appropriate meteorological information and suitable redundancy of information. In addition, TVA is developing a meteorological forecasting capability which should provide supplemental information for developing improved emergency procedures for incident response.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

The plant site is comprised of about 525 acres on a peninsula on the western shore of Chickamauga Lake at Tennessee River Mile 484.5. Plant grade is elevation 705 feet above mean sea level datum. The plant has been designed to be safely shut down in the event that floods occur which exceed plant grade.

The drainage area of the Tennessee River above the plant site is about 20,650 square miles. At Chickamauga Dam, 13.5 miles downstream, the drainage area is about

20,790 square miles. Rainfall occurs relatively evenly throughout the year and averages about 51 inches per year above Chickamauga Dam. Snowfall on the watershed above the plant averages about 14 inches per year. There are two general types of major flood-producing storms in the Tennessee River Basin -- the cool-season winter-type storm and the warm-season hurricane-type storm. Historically, most floods at the plant site have been produced by winter-type storms occurring from January through early April.

Water surface elevations on Chickamauga Lake at the plant site are controlled by Chickamauga Dam. At normal full pool, the water surface elevation is 682.5 feet above mean sea level, and the water surface area of the lake is about 35,400 acres. The corresponding volume of water impounded by Chickamauga Dam is about 628,000 acre-feet. The lake is about 3000 feet wide at the plant site with depths ranging to 50 feet at normal full pool elevation.

Upstream from the plant there are 20 major reservoirs in the TVA system and six major dams owned by the Aluminum Company of America, but controlled by TVA as part of the TVA flood control system. Flood control with emphasis on protection for the City of Chattanooga, about 20 miles downstream from the plant site, is a prime purpose of the TVA system. Above the plant site, flood control is provided largely by 11 tributary reservoirs and two main river reservoirs, Watts Bar and Fort Loudoun.

Average daily streamflow discharges from Chickamauga Dam approximate the streamflow rates at the plant site. However, upstream releases from Watts Bar Dam can also affect streamflow rates at the plant site. Furthermore, instantaneous flows at the plant site may vary considerably from daily averages, depending upon turbine operations at Watts Bar and Chickamauga Dams. Periods of several hours may occur when there are no releases from either dam, and periods of upstream flow in Chickamauga Lake can occur following rapid turbine shutdown at Chickamauga Dam.

Since the closure of Chickamauga Dam in 1940, the average daily streamflow rate at the plant site has been about 32,800 cubic feet per second. The maximum daily discharge was 219,000 cubic feet per second on March 18, 1973, and the minimum daily discharge was 700 cubic feet per second on November 1, 1953. Water velocities in the river channel at the plant site average about 0.6 feet per second during normal winter conditions and about 0.3 feet per second during the summer months.

Cooling water for both the condenser circulating water system and safety-related systems for Unit 1 operation, before the startup of Unit 2, will be provided by once-through flow taken from the river upstream of the plant. Water will flow from the river under a skimmer wall into an intake channel and condenser circulating water intake pumping station forebay prior to entering the plant. Two natural draft (main) cooling towers and a permanent safety-related essential raw cooling water intake pumping station will be completed prior to startup of Unit 2. The permanent essential raw cooling water intake pumping station will be located offshore in the lake at the skimmer wall, and it will be capable of taking suction from the river channel, even in the event that the downsteam dam, Chickamauga, fails.

Prior to Unit 2 operation, the heated cooling water will be discharged through a 1500-foot embayment (holding pond) and diffuser discharge system into the lake. Upon startup of Unit 2, the condenser circulating water system may be operated in any of the three following modes:

(1) once-through mode, as described above;

(2) helper mode, in which the heated water is passed through the main cooling towers prior to downstream release through the holding pond and diffusion pipes; or

(3) closed-cycle mode, in which the heated water is returned from the main cooling towers to the condenser circulating water pumping station forebay.

During operation of Unit 1, before the completion of the permanent essential raw cooling water pumping station, auxiliary essential raw cooling water mechanical draft cooling towers and pumps will be used in the event that floods occur which exceed plant grade and/or in the event that the downstream dam is lost. For the flood condition, the auxiliary essential raw cooling water pumps and cooling towers will be used to recycle and cool water for the necessary safety-related systems. For the postulated condition considering the loss of the downstream dam, the temporary (prior to Unit 2 operation) emergency raw cooling water sytem will function in a once-through mode until the condenser circulating water intake pumping station forebay (the forebay) is isolated from the lake. At that time the auxiliary essential raw cooling water closed-cycle mode will begin. After the main cooling towers and permanent essential raw cooling water intake pumping station are put into service, the auxiliary essential raw cooling water and the temporary emergency raw cooling water systems will be decommissioned.

The condenser circulating water discharge system diffuser pipes emanate from the holdup pond and are located in Chickamauga Lake at Tennessee River Mile 483.6. Relocation of the Savannah Utility District Water intake, which supplied water to over 2000 people, was required (at the construction permit review stage) before issuance of an operating license. The applicant has conducted a groundwater investigation study and has replaced the Savannah Utility District river intake with wells. As stated in Amendment 37 to the Final Safety Evaluation Report, the wells are located about 4.5 miles northeast of the plant on the other side of Chickamauga Lake at depths of 90-100 feet in the Knox Dolomite formation. This formation is hydraulically separated from the Conasauga shale formation, which underlies the plant, by a trace of the Kingston fault (see Section 2.4.4 of this SER) and Chickamauga Lake. Thus, there will be no effect on the plant of pumping from these wells, and vice versa.

The nearest present downstream public surface water user is the City of Chattanooga, which has a river intake located about 18.3 river miles from the plant discharge diffuser pipes. A future river water supply intake in planned by the East Side Utility District to be located about nine miles below the plant site in the Wolfcreek embayment of Chickamauga Lake. The nearest industrial users of surface water for human consumption are located about 10.6 river miles downstream from the plant discharge diffuser pipes.

The East Side Utility District is the nearest major public user of groundwater from the site aquifer. This supply is obtained from wells located about seven miles from the plant site. In addition, there are about 100 small-yield domestic wells within a two-mile radius of the plant. The applicant estimates that the total domestic groundwater withdrawal within the two-mile radius is about 50,000 gallons per day. Most of these wells obtain water from the Knox Dolomite formation, which is the regional groundwater source for eastern Tennessee.

2.4.2 Flood Potential

Continuous river discharge records dating from 1874 are available for the Tennessee River at Chattanooga about 20 miles downstream from the plant site. Flood flows and corresponding river stages (levels) at the plant site have been altered by TVA's reservoir system beginning in 1936 with the closure of Norris Dam and reaching the present level of control in 1952 with the closure of Boone Dam. The maximum known flood that occurred prior to regulation (March 1867) reached an estimated elevation of 690.5 feet above mean sea level (450,000 cubic feet per second) at the plant site. Under present day regulation, the largest flood occurred on March 18, 1973, and reached elevation 687.0 feet above mean sea level (219,000 cubic feet per second) at the plant site.

The probable maximum flood stillwater (excluding windwave effects) level for the plant is 722.6, feet above mean sea level. Such a flood would result from the occurrence of the probable maximum precipitation on the Tennessee River drainage basin above the plant site. During such an event, the flood crest at the plant could be augmented by the failure of the earthen embankments at Watts Bar Dam upstream and diminished by the failure of the earthern embankments at Chickamauga Dam downstream. The estimated probable maximum flood discharge at the plant site would be 1,370,000 cubic feet per second.

The estimated maximum water surface elevation in the lake, 726.8 feet above mean sea level, results from postulated 45 miles per hour overwater wind wave activity coincident with the probable maximum flood stillwater level. The corresponding maximum water level (design basis flood level) at the plant, including wave runup effects, is estimated to be 726.8 feet above mean sea level. We have reviewed these estimates and concur that they are conservative. The probable maximum flood, as well as other lesser rainfall floods, could exceed plant grade, 705 feet above mean sea level, and will necessitate plant shutdown. Emergency plant operating requirements and technical specifications are described in Section 2.4.5 of this report.

Since there are 20 major dams above the plant site, the applicant examined these dams individually and in groups to determine if arbitrarily assumed seismic failures coincident with river flooding would create critical water levels at the plant. The applicant examined two postulated combinations of natural events acceptable to us:

(1) a one-half safe shutdown earthquake as defined in Section 2.5 of this report, coincident with a one-half probable maximum flood; and

(2) a safe shutdown earthquake coincident with a 25-year flood.

Neither of the above conditions were estimated to result in water levels at the plant site greater than that created by the probable maximum flood. However, one combination of events, dam failures assumed from a critically-centered one-half safe shutdown earthquake coincident with one-half the probable maximum flood, could cause the simultaneous failure of Fontana Dam on the Little Tennessee River and Hiwassee, Blue Ridge, and Apalachia dams on the Hiwassee River. The resulting maximum stillwater elevation in the lake at the plant was estimated to reach 710.9 feet above mean mean sea level. Wind waves could raise this water level to elevation 712.6 feet above mean sea level. Wind wave runup on plant safety-related buildings could reach elevation 715.8 feet above mean sea level. Five other assumed seismic dam failure-flood combinations with concident wind wave activity were estimated to also possibly cause water levels that exceed plant grade. As described in Section 2.4.5 of this report, plant shutdown will be initiated upon notice that any one of nine upstream dams (Norris, Cherokee, Douglas, Fort Loudoun, Fontana, Hiwassee, Apalachia, Blue Ridge, or Tellico) has failed. In the absence of communications for eight hours after an earthquake, shutdown will be initiated, as discussed in Section 2.4.5.

The probable maximum flood and separately combined seismic dam failure and rainfall floods were extensively reviewed and accepted during the construction permit review. The applicant has made minor changes to the hydrologic runoff models to reflect information available from the 1973 flood and improved techniques developed since construction permit review. These changes result in predicted water levels that are up to about four feet higher than those determined in the construction permit licensing stage. We have reviewed these changes and the resultant effect on design basis flood levels and conclude that they are consistent with the Regulatory Guides 1.59," Design Basis Floods for Nuclear Power Plants," and 1.102, "Flood Protection for Nuclear Power Plants."

Site drainage to the Tennessee River has been provided to accommodate runoff from precipitation as severe as a local probable maximum precipitation. Structures that house safety-related facilities are protected from such local flooding by the slope of the plant yard; the local probable maximum flood will not reach or exceed the critical floor elevation, 706 feet above mean sea level.

The applicant conservatively neglected precipitation losses in estimating the local probable maximum flood level. At our request, the applicant assumed all underground drains to be clogged and all surface drains to be full in analytically testing the adequacy of the site drainage system. A peak local probable maximum flood discharge rate of 14 inches per hour (equal to the maximum hourly rainfall rate) from the outlet of each drainage area was used. With all drains clogged, the plant yard perimeter road and railroad embankments control site drainage outflow. Standard backwater methods were used to estimate the corresponding water levels at plant buildings, assuming a flat plant yard slope. Resulting water levels were estimated to be less than 705.5 feet above mean sea level. We have reviewed the applicant's analysis of the adequacy of the site drainage system, and we find it consistent with the criteria of Regulatory Guides 1.59, "Design Basis Floods for Nuclear Power Plants" and 1.102, "Flood Protection for Nuclear Power Plants."

We also asked the applicant to evaluate the ability of the site drainage system to function properly during the period of time between startup of Unit 1 and startup of Unit 2. The applicant stated that the plant would not flood from a local probable maximum precipitation during this period of time, since grading essential to the site drainage system will be complete prior to startup of Unit 1.

We required the applicant to demonstrate that no flooding of safety-related structures can occur in the event of precipitation as severe as a local probable maximum precipitation coincident with clogged roof drains and scuppers. The applicant demonstrated to our satisfaction that water buildup under such conditions would be less than the allowable depth of water the roofs of safety-related structures can withstand.

2.4.3 Low Water Considerations

The applicant's estimated minimum average daily river flow rate past the plant site is 5,000 cubic feet per second. This estimate is based upon a low flow frequency analysis conducted for the period of record since January 1942. Since then, low flows at the site have been regulated by TVA dams. Recorded average daily flows at the plant site have been less than 5000 cubic feet per second only 0.2 percent of the time.

We requested the applicant to compare the estimated minimum water level at the plant site, elevation 673 feet above mean sea level (occurring in the winter flood season as a result of special preflood reservoir drawdown), with a minimum flow rate, level, and frequency of occurrence resulting from the most severe drought considered reasonably possible in the region, to evaluate the dependability of safety-related water supply. The most severe historical drought in the Tennessee Valley region occurred in 1925. The minimum average daily flow at the plant site during this drought was 3,200 cubic feet per second; the applicant's frequency studies indicated that this flow rate had a probability of 10^{-2} of occurring in any given year. The same studies indicated that a daily flow rate of 1,300 cubic feet per second at the plant site would have a probability of 10^{-7} of occurring in any given year.

The corresponding lake level would be about elevation 675 feet above mean sea level.

The plant can operate normally during low water conditions down to elevation 668 feet mean sea level, the level at which the condenser circulating water pumps must be assumed to cease operation. This elevation is three feet above the elevation, 665 feet above mean sea level, where the intake forebay would lose access to the river. We have reviewed the applicant's low water analyses, and conclude that an adequate water supply for safety systems should be available.

Prior to two-unit operation, and in the unlikely event of the loss of the downstream dam, the seismic Category I intake forebay pool (also designed to withstand erosion and sedimentation effects of a probable maximum flood) will provide in excess of 2.5 million gallons storage capacity, of which 2.2 million gallons will be available for supplying makeup water to the closed-cycle auxiliary essential raw cooling water system. This is enough makeup water for about six days, assuming severe environmental conditions in conjunction with a loss-of-coolant accident and a loss of the downstream dam.

Four portable forebay makeup pumps would be deployed prior to total depletion of the forebay pool, to supply makeup water from the river. These pumps, which will be stored onsite above the design basis flood elevation in such a way as to protect them against the safe shutdown earthquake, can supply water at a combined rate of 2100 gallons per minute. The forebay makeup requirement under such postulated conditions does not exceed 300 gallons per minute.

In the event of the loss of the downstream dam, as a result of a severe flood (not the preceding case, in which a seismic event also is postulated to cause a loss-of-coolant accident), the forebay portable makeup system will be deployed immediately following isolation between the river and the forebay, and the system is to be operational within three days.

The permanent emergency raw cooling water pumping station is to be functional upon two-unit operation. Since this seismic Category I structure will have direct communication with the river for all water levels (including any loss of downstream dam) and is above the design basis flood level, the emergency raw cooling water system for two-unit operation will be capable of functioning in an open cooling-cycle mode for all anticipated river conditions. Therefore, makeup by portable means to the condenser circulating water intake forebay will not be necessary, nor will there be a need for dependence upon the auxiliary essential raw cooling water system.

We requested the applicant to provide an analysis of the river water level and flow rate at the plant site resulting from the loss of the downstream dam coincident with a drought of historical severity. The lowest average daily flow of record 3,200 cubic feet per second, occurred prior to reservoir regulation, and this represents the minimum flow of historical severity that could be expected to be available after a loss of the downstream dam. The water surface elevation corresponding to this flow rate is greater than 10 feet above the elevation (625 feet above mean sea level) of the essential raw cooling water pump sump. Only 300 gallons per minute (less than 1 cubic foot per second) is needed for the ultimate heat sink to perform adequately (closed-cycle mode) prior to two-unit operation, and less than 50 cubic feet per second is required for the ultimate heat sink to perform adequately (open-cycle mode) during two-unit operation. Therefore, we conclude that the proposed ultimate heat sink system will be capable of performing adequately during all credible low river flow conditions.

2.4.4 Groundwater

An outcropping of the Conasauga Shale, a poor water bearing formation, underlies the plant. A trace of the Kingston Fault, located about 2000 feet northwest of the plant, separates hydraulically the Conasauga Shale outcropping from a wide belt of the Knox Dolomite formation. The latter is the major water-bearing formation of eastern Tennessee. Because the Knox Dolomite is essentially hydraulically separated from the Conasauga Shale, offsite pumping, including future development, should have little effect upon the groundwater table in the Conasauga Shale at the plant.

Small openings along fractures and bedding planes contain the groundwater in the Conasauga Shale. In the Knox Dolomite, however, groundwater occurs in solutionally enlarged openings formed along fractures and bedding planes and in locally thick cherty clay overburden.

The applicant does not intend to use groundwater at the plant. Local offsite groundwater use was described above in Section 2.4.1.

Recharge to the groundwater system (Conasauga Shale) at the plant site is from local precipitation. Discharge from the system is towards the northeast and southwest into Chickamauga Lake.

At our request, the applicant provided conservative estimates of the permeability and porosity of the Conasauga Shale, the hydraulic gradient, and an acceptable estimate of groundwater travel time from the plant to the nearest downgradient surface water body, Chickamauga Lake. There are no downgradient wells between the plant and Chickamauga Lake, which is about 1000 feet from the plant in a downgradient northeast direction. We concur that the applicant's estimate of about 300 days for groundwater to move along this pathway in the event of a postulated accidental liquid radwaste release to the groundwater system is conservative.

We have conservatively estimated that a groundwater dilution factor of about 2.8 would be applicable to a postulated accidental release prior to entry into Chickamauga Lake. Upon entry and initial mixing with lake water, we estimate that a dilution factor of 9.8×10^4 is appropriate. The total dilution factor applicable at the nearest planned downstream surface water user, East Side Utility District (nine miles downstream from the facility) is 9.9×10^5. There are no existing downstream water users closer than nine miles from the plant. (See Section 15.4.7 for dose evaluation of a postulated liquid tank failure)

Even though the potential for accidental contamination of the groundwater system is extremely low, the applicant plans to monitor radioactivity levels and groundwater levels in five observation wells in the plant area throughout the plant lifetime.

We have reviewed groundwater hydrographs in the plant area and based on these hydrographs we find a groundwater elevation of 691.0 feet above mean sea level to be acceptable as the design basis groundwater level.

2.4.5 Hydrologically-Related Technical Specifications

The applicant has provided an emergency flood protection plan designed to minimize the impact of floods exceeding plant grade on safety related facilities, and a corresponding proposed technical specification outlining the action to be taken to prevent any flood-caused accidents. The applicant's flood protection plan includes procedures for predicting rainfall floods, arrangements to warn of upstream seismically induced dam failure floods, and lead times available and types of action to be taken to meet safety related requirements for both sources of flooding. The applicant's warning scheme for both types of floods is to be divided into two stages. Stage I will allow preparation steps and some damage, but will withhold major economic damage until Stage II warning assures a flood above plant grade.

Reservoir levels for large rainfall floods can be predicted well in advance by the applicant. The applicant estimates that a minimum of 27 hours, divided into the two warning stages, will be available between the time a preflood preparation order is issued and the time the flood water could exceed plant grade. A minimum 10-hour Stage I will begin upon prediction that flood producing conditions might develop. A minimum 17-hour Stage II will be based on a confirmed estimate that conditions will produce a flood above plant grade.

Seismically-induced failure of upstream dams can result in flood surges that exceed plant grade. However, such surges do not have a water level potential as great as the rainfall-induced probable maximum flood water level. A minimum of 27 hours, divided into the two warning stages, is estimated by the applicant to be available to prepare the plant for such flooding.

The applicant defines "flood mode" operation as the means by which the plant will be safely maintained during the time when flood waters exceed plant grade, elevation 705 feet above mean sea level, and are allowed ingress into plant structures, and during the succeeding time period until recovery is accomplished.

Plant cooling requirements during flood mode operation will be met by the essential raw cooling water system, unless flood mode operation is necessary prior to operation of the permanent essential raw cooling water pumping station. If the latter is necessary, the auxiliary essential raw cooling water system will provide closed-cycle water circulation to meet plant cooling requirements. Water supplied by both these systems is discussed in greater detail above in Section 2.4.1 and 2.4.3.

The applicant proposes one kind of warning scheme for rainfall floods and another type of warning scheme for seismically-induced dam failure floods. For rainfall floods, the first stage (Stage I) of shutdown will begin when sufficient rainfall occurs to yield a projected plant site water level of 697.0 feet above mean sea level in the winter months (October 1 through April 15) and 703 feet above mean sea level in the summer months (April 16 through September 30). These water levels assure that any additional rain will not produce water levels in excess of 703 feet mean sea level in less than 27 hours. This level provides a two-foot margin (requested by us) so that waves resulting from high winds cannot disrupt flood protection preparation, i.e., cannot exceed plant grade of 705 feet above mean sea level.

Stage I will be maintained until either Stage II begins, or until the applicant determines that floodwaters will not exceed elevation 703 feet above mean sea level at the plant. Stage II shutdown will begin only when enough additional rain has fallen to yield water levels in excess of 703.0 feet above mean sea level. The applicant estimates that required shutdown procedures will take no longer than 24 hours, which allows a three-hour contingency margin.

As stated in Section 2.4.2 above, the failure of nine upstream dams either singly or in varying combinations can produce floods over plant grade. Stage I shutdown will be started upon notification that any one of these dams has failed, and will continue until it has been determined that critical combinations do not exist. At our request, the applicant committed to initiating Stage II shutdown if communications are lost, or if there is no certainty that critical combinations do not exist in such situations.

Three communication networks are available to the applicant:

(1) the applicant's own microwave network;

(2) the applicant's own powerline carrier system; and

(3) the commercial Bell telephone system.

We have reviewed the applicant's proposed emergency flood protection plan and corresponding plant shutdown technical specifications. We find both acceptable from a hydrologic engineering standpoint. Technical specifications for plant shutdown to minimize the possibility of an accident resulting from hydrologically associated phenomena other than floods are not necessary, since such phenomena should have inconsequential effects upon safety-related facilities.

2.4.6 Conclusions

We conclude that adequate flood design bases have been provided and implemented, including an adequate emergency flood protection plan and corresponding proposed plant shutdown technical specifications. We also conclude that an adequate water supply can be assured for safety-related purposes, and that plant operation and the remainder of plant construction will not adversely affect, or be affected by, local and regional surface or groundwater supplies.

2.5 Geology and Seismology

2.5.1 Geology

The regional and site geological and seismological conditions, as presented in the Final Safety Analysis Report, including amendments, were reviewed by the staff. The Final Safety Analysis Report adequately appraises conditions pertinent to an evaluation of the sites for Units 1 and 2. The information contained therein confirms the conditions as described in the Preliminary Safety Analysis Report for Units 1 and 2 which was reviewed by the staff and its advisors, the U.S. Geological Survey and the U.S. Coast & Geodetic Survey, and reported in the Safety Evaluation Report dated March 24, 1970. We have reviewed inspection records, maps, and reports of the excavations and conclude that there is no evidence of faulting or other geologic features at the site that are unsafe or unacceptable. A great deal of information has been gathered during the review of this site, and reviews of other sites in the Southern Valley and Ridge area (Phipps Bend, Watts Bar, Bellefonte, Clinch River Breeder Reactor) have aided in the current evaluation of this site. The regional aspects which also apply to this site are reasonably well understood and have also been discussed extensively in these other reviews and safety evaluations.

The investigations performed by the applicant have been sufficient to adequately assess site geologic conditions in accordance with "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A to 10 CFR Part 100. We conclude that the regional tectonic structures do not pose any threat to the safety of the plant. There are also no known geologic structures that would cause surface displacement or would tend to localize earthquakes in the site vicinity.

2.5.2 Tectonic Setting

Since earthquake activity cannot be reasonably associated with known geologic structure in the southern Appalachians, earthquakes are instead identified with the tectonic province in which the site is located. The applicant indicates that the site is located in the Southern Appalachian Tectonic Province. As defined, this province is bounded on the east by the western margin of the Piedmont Province, on the west by the eastern limits of the Cumberland Plateau, on the south by the overlap of the Gulf Coastal Plain Province, and on the north by the reentrant in the Valley and Ridge Province near Roanoke, Virginia.

In our review, we determined that the proposed site is within the Southern Valley and Ridge Tectonic Province based on provinces which are more in accord with those proposed by King (1969), Eardley (1973), King (1959), Rodgers (1970), and Hadley and Devine (1974) for eastern North America. This province is bounded on the east by the western extent of the Piedmont Province (in our view, for purposes of nuclear power plant siting, the Blue Ridge Province is considered as part of the Piedmont Province), on the west by the Cumberland Plateau, on the south by the Gulf Coastal Plain, and on the north by the northern part of the Valley and Ridge Province.

2.5.3 Seismology

Introduction

The Sequoyah site lies within the Southern Valley and Ridge tectonic province, a region where earthquake activity cannot be reasonably associated with known faults or other geologic structures. Following the procedures set forth in Appendix A to 10 CFR Part 100, earthquakes are therefore identified with the tectonic province in which they occur, and the controlling earthquake, or that event which defines the safe shutdown earthquake, becomes, in this and most eastern site evaluations, the maximum historical earthquake in the province within which the site is located.

The construction permit for the Sequoyah Nuclear Plant was issued in May 1970. During the construction permit review, we concluded that a modified Housner response spectrum anchored at 0.18g was acceptable as the safe shutdown earthquake. This conclusion was based on the assumption that the maximum historical earthquake (1897 Modified Mercalli Intensity VIII earthquake at Giles County, Virginia) might recur anywhere within the Southern Valley and Ridge tectonic province, and on adoption of recommended response spectra and an acceleration anchor point.

While the seismological and geological evaluation of this controlling earthquake has not been altered since the construction permit review, the staff has in the interim adopted a Standard Review Plan and Regulatory Guides which have the effect of changing the response spectra and the anchor point acceleration for a Modified Mercalli Intensity VIII earthquake from the values accepted for the Sequoyah

Nuclear Plant at the construction permit review stage. Specifically, the staff would now characterize a Modified Mercalli Intensity VIII earthquake with the more conservative spectrum specified in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power PLants," anchored at 0.25g. The higher reference acceleration is determined using the trend of the means relating peak acceleration to intensity shown by Trifunac and Brady (1975). An example of application of this present staff practice in the Southern Valley and Ridge tectonic province is found in the Phipps Bend Safety Evaluation Report (Docket 50-553). The Standard Review Plan and Regulatory Guides represent approaches and practices which the staff considers acceptable to establish conformance with the Nuclear Regulatory Commission regulations, but they are not specified as the only acceptable means of meeting the regulations.

Since the present staff practice, as represented by the Standard Review Plan and Regulatory Guides, resulted in the identification of a seismic design basis acceptable to the staff which exceeded the actual seismic design basis for the Sequoyah Nuclear Plant, we requested the applicant to provide information which would establish the adequacy of the seismic design prior to our completing the operating license review of the Sequoyah Nuclear Plant. We also organized an interdisciplinary working group within the NRC to assess different approaches which might be pursued to establish the seismic design adequacy of the Sequoyah Nuclear Plant, and of other Tennessee Valley Authority plants within the same tectonic province currently being considered for operating licenses (Watts Bar and Bellefonte).

In response to our requests for information, the applicant responded with several reports. The first, "Justification of the Seismic Design Cr. ria Used for the Sequoyah, Watts Bar, and Bellefonte Nuclear Power Plants, Phase I," April 1978, was submitted prior to the completion of the NRC's working group's recommendations. In it, the applicant argues that the present design at Sequoyah is adequate because:

(1) The maximum intensity of the 1897 Giles County earthquake was really VII-VIII rather than VIII.

(2) The intensity ra ing for the 1897 Giles County earthquake is soil biased and that the same earthquake would result in a lower intensity at rock sites such as at Sequoyah.

(3) The intensity-acceleration relationship derived by Murphy and O'Brien (1978) is more appropriate than that found in Trifunac and Brady (1975) and should be used in determining reference accelerations.

(4) At foundation depth, earthquake-induced ground motion is less than that at the surface.

We concluded that while there was some validity in these arguments, there were also a sufficient number of problems associated with each so as to preclude their use in justifying the adequacy of the present seismic design. For example:

(1) The U.S. Geological Survey and National Oceanic and Atmospheric Administration had convened special panels to reevaluate the Giles County earthquake. The majority of the panel members considered the maximum intensity to be VIII even though some felt it to be a "weak" VIII (letter from W. A. Radlinski to E. G. Case, 1976).

(2) Studies of earthquake response spectra suggest that, for similar earthquakes, spectra recorded on rock sites indicate stronger motion at frequencies of interest to nuclear power plants (greater than two Hertz) than those recorded on soil sites (Trifunac, 1978).

(3) While the Murphy and O'Brien (1978) intensity-acceleration relationship may be more correct than the Trifunac and Brady (1975) relationship with regard to peaks of acceleration, this does not necessarily mean that it is more appropriate for use in predicting reference accelerations. Agbabian Associates (1977), for example, compared different design spectra and reference accelerations for given intensities with response spectra determined from accelerograms recorded at the same intensities. When used in conjunction with the Regulatory Guide 1.60 spectrum, the more conservative Trifunac and Brady (1975) relationship was not found to predict an overly conservative design (that is, exceed the mean plus one standard deviation of the data in frequencies of interest).

(4) It is very difficult to predict the reduction in ground motion at shallow depths. Recent staff and Advisory Committee on Reactor Safeguards positions on the Davis Besse Unit 1 Nuclear Power Plant (Docket No. 50-346) advise against allowing for this reduction.

The recommendations of the NRC working group to establish the adequacy of the seismic design basis were to:

(1) Determine site-specific safe shutdown earthquake response spectra from strong-motion records of appropriate magnitude and distance.

(2) Determine site-specific safe shutdown earthquake response spectra from strong-motion records of appropriate intensity.

(3) Reevaluate original seismic structural and floor response analyses using more realistic methods and material properties as well as site-specific safe shutdown earthquake response spectra.

(4) Reevaluate the operating basis earthquake to see whether it meets the reasonability-of-occurrence criterion of Appendix A to 10 CFR Part 100.

(5) Compare the probability of the safe shutdown earthquake being exceeded at Sequoyah to other Tennessee Valley Authority plants that meet the Standard Review Plan.

The working group report was transmitted to the applicant (letter R. Boyd to N. B. Hughes, May 1978). Following interaction with the staff, the applicant submitted a second report, "Justification of the Seismic Design Criteria Used for the Sequoyah, Watts Bar, and Bellefonte Nuclear Power Plants, Phase II," August 1978. Our review is based upon this report and additional information transmitted to the staff addressing the seismological recommendations (1), (2), (4) and (5) listed above. We are awaiting submittal of a final report which incorporates all the additional information, most of which has not yet been submitted formally.

The working group's recommendations and our review of the submitted material are aimed at:

(1) Making a realistic, yet conservative, estimate of ground motion from the controlling earthquake.

(2) Comparing this estimate with the existing seismic design.

(3) Determining the significance of any differences between the above.

Site-Specific Earthquake

In order to compute site-specific response spectra, it is necessary to characterize the earthquake size, the epicentral distance (the distance between the surface location of the earthquake and the site), and the site conditions (soil or rock) being modelled. As mentioned above, the 1897 Giles County earthquake was considered to have an epicentral intensity of VIII (Modified Mercalli). There are relatively few recordings of strong ground motion at intensity VIII and none (at least in the western United States) recorded at rock sites. This fact, plus the dispute over the epicentral intensity (i.e., VIII, weak VIII, or VII-VIII) and the more dependable classification of strong-motion records by magnitude, led the staff to seek a reliable magnitude estimate of the Giles County event for this review. A recent study by Nuttli, Bollinger and Griffiths (1979) has utilized empirically determined relationships between body wave magnitudes of instrumentally-recorded earthquakes and (1) their intensity fall-off with distance, (2) their total felt area, and (3) the area within the intensity IV isoseismals to determine magnitudes for historical earthquakes where only intensity data were available. Particular attention was given in this study to the 1897 Giles County earthquake. The two most reliable estimates in the authors' opinion (intensity fall-off with distance and the intensity IV isoseismal area) each indicated a body wave magnitude of 5.8 for this event. We recommended use of this value, and to account for uncertainty, we suggested the applicant assume a body wave magnitude of 5.8 ± 0.5 (5.3 to 6.3) for the Giles County earthquake. In addition, the suggestion was made to include all strong-motion records in this magnitude range recorded at rock sites (the foundation conditions for most safety-related structures at Sequoyah) at epicentral distances of less than 20 kilometers to 25 kilometers. According to Gupta and Nuttli (1976), this is the approximate distance range to which maximum intensities are felt in the central United States. In addition, at these close distances, the strong differences in seismic wave attenuation between earthquakes east and west of the Rocky Mountains have not yet affected the ground motion. In our opinion, this makes the direct use of strong-motion records from the west for simulating ground motion in the east (where there are no appropriate records) more credible and acceptable.

Aside from reflecting the uncertainties in characterization of the controlling earthquake, establishing a range of magnitude and distance would help insure an adequate amount of data. The applicant was able to collect 13 sets of strong-motion records (26 horizontal components) that fell within this range. Six sets were from western United States earthquakes (Helena, 1935; San Francisco, 1957; Parkfield, 1966; Lytle Creek, 1970; and Oroville, 1975), and seven sets were from a sequence of well-recorded events that occurred in 1976 near Fruili, Italy. The magnitude of the earthquakes used ranged from 5.3 to 6.2 (average 5.7) and the epicentral distance ranged from seven kilometers to 27 kilometers (average 16 kilometers). The set of records recorded at 27 kilometers represented the strongest ground motion in the data and were included to insure conservatism even though it fell slightly outside the 25 kilometer guideline.

The applicant did not include the records recorded at Koyna, India, from the 1967 body wave magnitude 6.0 event. TVA felt that these records could not be considered free-field in that they were recorded at mid-height on the Koyna Dam and that the peaks of strong motion corresponded to the natural period of the monolith in which the instruments were based. The applicant also indicated an attempt was being made to obtain records from another event at Koyna in which both free-field records and dam records were recorded and could be compared. Unless these latter records indicate otherwise, the staff feels that the applicant has supplied sufficient evidence to warrant exclusion of the Koyna Dam records from free-field data sets. The general sensitivity of the data set to the possible inclusion of additional strong-motion records is discussed below.

Response spectra from the 26 records were computed and representative site-specific spectra were calculated. No scaling for magnitude or distance was done since the parameter range chosen is believed to represent the uncertainty in the definition of the controlling earthquake causing maximum damage at the site. The data at different frequencies were found to best fit a log-normal distribution. The 50th and 84th percentile spectra were calculated and then compared to the design spectra used at Sequoyah. When comparing spectra, differences between the more conservative damping values used in the design of the plant and those accepted currently by the staff (Regulatory Guide 1.61) were taken into account. Presently, the staff allows greater damping than previously. The case where this factor contributes the least to the conservatism of the original design (the "worst" case) would be for materials at which the allowed damping has changed the least. For rock supported structures at Sequoyah, this would be reinforced concrete. The plant design assumed five percent damping while Regulatory Guide 1.61 allows for seven percent damping. The comparison was then made between the design spectra at five percent damping with the site-specific spectra at seven percent damping. For soil-supported structures, the applicant utilized techniques which assume conservative amplifications of two to three times the rock motions in the periods of interest (less than 0.5 seconds). In the following discussion, arguments made for reinforced concrete represent the extreme or "worst" case at which any differences between response spectra would be maximized.

We have concluded that the 84th percentile spectra represents an appropriately conservative representation of the site-specific earthquake. Although it is not dictated by the regulations, the choice of this level is a judgement which we consider a reasonable technical approach to establish a standard of acceptability which is conservative. The selection of the 84th percentile level (mean plus on standard deviation) is the level used in deriving the Regulatory Guide 1.60 spectral shape, and is also the level of acceptability being expressed by the staff in revisions of the Standard Review Plan dealing with the use of site-specific spectra.

For periods of interest (less than 0.5 seconds), the present Sequoyah design spectrum for reinforced concrete lies between the 50th and 84th percentile (see Figure 2.3). The average in this period range is about the 74th percentile. The greatest departure from the 84th percentile is at 0.06 seconds (15 Hertz) where the present design is at the 67th percentile. In terms of actual spectral response for this period, the present design is at 0.18g while the 84th percentile (hereafter called the site-specific safe shutdown earthquake) would be at 0.28g. At periods greater than 0.35 seconds, the present design always exceeds the site-specific safe shutdown earthquake. Examination of other materials indicate less deviation of the present design from the site-specific safe shutdown earthquake. For welded steel, for example, the present design is less than the site-specific safe shutdown earthquake only at periods less than 0.08 seconds with the greatest departure being at 0.055 seconds (74th percentile). In terms of peak acceleration, the 84th percentile site-specific spectrum is associated with a peak acceleration of 0.215g.

The applicant contends that site-specific spectra should be chosen using the same procedures as those used in deriving the Regulatory Guide 1.60 spectrum and in the

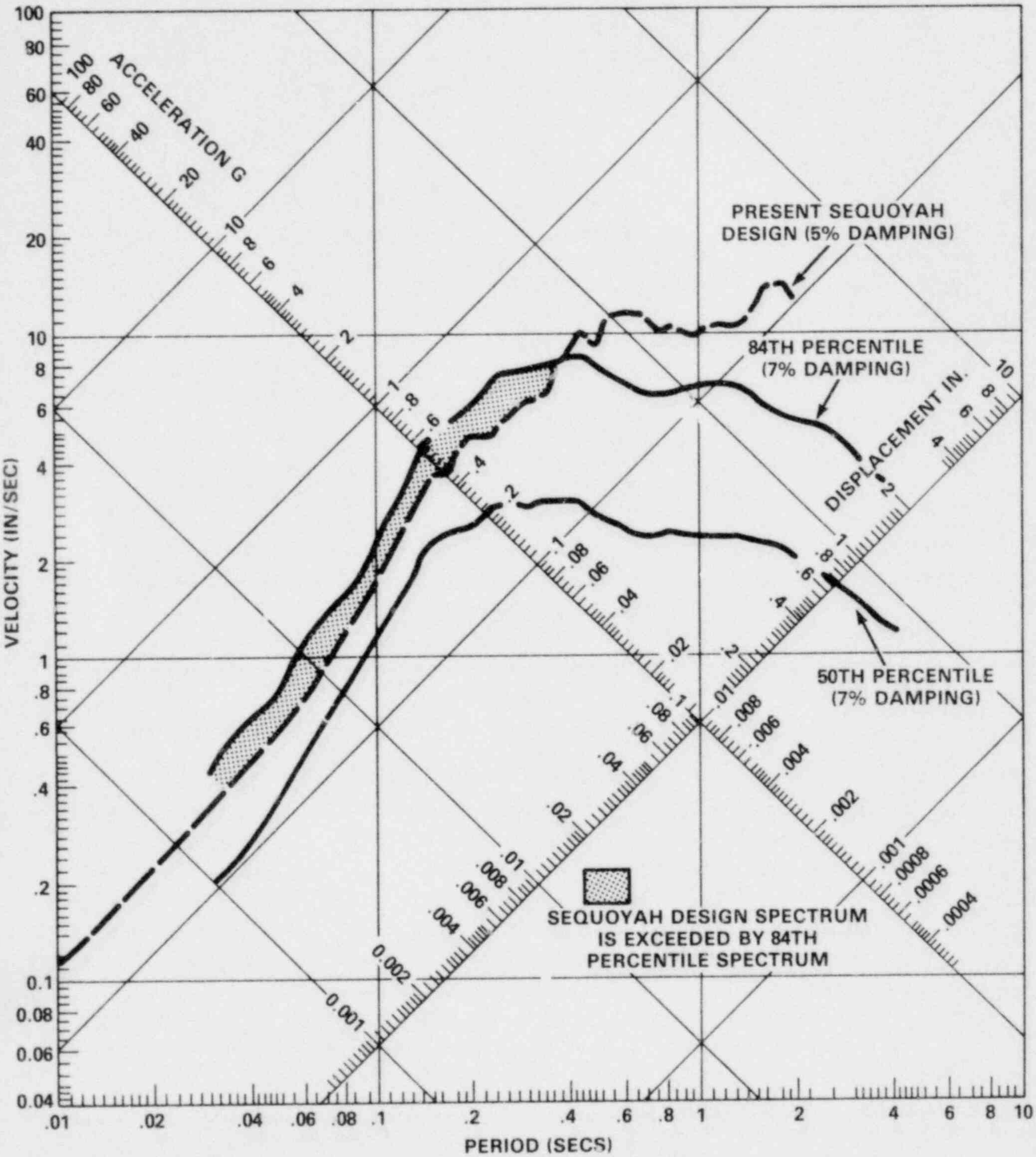

Figure 2-3

Comparison Of The Present Sequoyah Design Spectrum For Reinforced Concrete With Appropriately Damped 50th And 84th Percentile Site Specific Response Spectra.

Standard Review Plan. Following these procedures, they normalized all 26 strong-motion records to the same peak acceleration and then calculated 16th, 50th and 84th percentile spectral shapes, which were then anchored to the 50th percentile peak acceleration (0.10g) of the data set. In this case, the present design spectrum exceeds the 84th percentile spectrum at all freqencies.

The staff does not consider this procedure applicable in site-specific studies. The Regulatory Guide 1.60 spectrum was calculated from strong-motion records of earthquakes of different magnitudes, and recorded at different epicentral distances at different site conditions. It was specifically developed for use with differing reference peak accelerations to estimate different earthquake conditions. Determination of a site-specific spectra from strong-motion records of the appropriate size earthquake, distance and site conditions obviates the use of this approach and allows direct estimation of the response spectrum at each period.

Relative Seismic Hazard

To assess the significance of the differences between the present design and the site-specific response spectrum, we recommended that the applicant utilize probabilistic techniques to compare the relative difference in seismic hazard associated with different response spectra. Certain aspects of probabilistic techniques have been subjected to criticism in the past, particularly detailed risk computations associated with structural performance during earthquakes and specific absolute probabilities. In this review, however, emphasis has been placed upon the much more reliable relative probabilities, and structural performance was not considered. The basic calculations were done utilizing a very widely used and tested earthquake hazard computer code (McGuire, 1976). The input parameters used were:

(1) Earthquake activity levels for the host tectonic province and those surrounding it. The activity rate for each province was determined from the specific earthquake history. The b values (relative recurrence rates) were all assumed to be 0.57 (Chinnery, 1979). The upper intensity cutoff was assumed to be the maximum historical intensity except for the host (and controlling) province where the maximum possible intensity was conservatively assumed to be IX rather than VIII.

(2) The intensity fall-off with distance was taken to be that determined from the 1886 Charleston earthquake (Bollinger, 1977).

(3) Site intensities were converted to peak acceleration utilizing the relationship determined by Murphy and O'Brien (1978). For these relative probabilistic calculations this relationship is preferred to that of Trifunac and Brady (1975) in that it assumes a log-normal distribution and has associated with it a needed measure of dispersion. As indicated previously, however,

this does not make the Murphy and O'Brien relationship preferable when used to predict single-valued non-dispersed reference accelerations to anchor the Regulatory Guide 1.60 spectrum.

(4) Peak accelerations were converted to spectral accelerations at selected periods utilizing spectral amplification factors calculated from the 26 site-specific spectra normalized to the same peak acceleration.

(5) The dispersion associated with each of the last three relationships was included in a total dispersion defined by a standard deviation for each period. There was no limit placed upon this dispersion except that at distances out to 49 kilometers the attenuated intensity was greatly restricted from exceeding the peak intensity. This was accomplished by defining the peak intensity as being the mean intensity plus three standard deviations. Since no such restrictions were placed upon the intensity-acceleration relationship or the spectral-amplification relationship, very high accelerations were still considered possible and indeed allowed.

While the above input parameters were considered reasonable and conservative by the staff, the applicant conducted parametric tests to determine the effects of changes in the b values, upper intensity cutoff, and attenuation functions.

Simplified uniform hazard spectra were computed. These are spectra that have a uniform risk of exceedence, for example, 10^{-2}, 10^{-3}, 10^{-4} or 10^{-5} per year. They were compared to the present design spectrum, the site-specific safe shutdown spectrum, and the Phipps Bend safe shutdown earthquake spectrum which utilized the Regulatory Guide 1.60 response spectrum anchored at 0.25 g (see Figure 2-4). At periods of interest, the present design spectrum was found to be approximately equivalent to the 10^{-3} spectrum (average risk of exceedance at 9.0×10^{-4} per year). Both the site-specific and Phipps Bend spectra were found to lie between the 10^{-3} and 10^{-4} uniform hazard spectra with average risks of exceedance of 4.7×10^{-4} per year and 2.3×10^{-4} per year, respectively. This range (10^{-3} to 10^{-4}) is the general level of annual risk of exceedance of intensity VII-VIII or VIII calculated for various locations in the eastern United States (see for example McGuire, 1977).

Relative seismic hazard (ratios of annual risk of exceedance) were calculated for the different design spectra in the periods of interest. The seismic hazard associated with the present Sequoyah design is on the average twice as great as that associated with the site-specific safe shutdown spectrum. The "worst" case is at 0.06 seconds where the increase in hazard is a factor of 3.1. These relative hazards are very stable. Parametric tests indicated that while changes in recurrence rate, intensity cutoff and attenuation functions could result in variations greater than an order of magnitude in absolute seismic hazard, the relative seismic hazard changed very little.

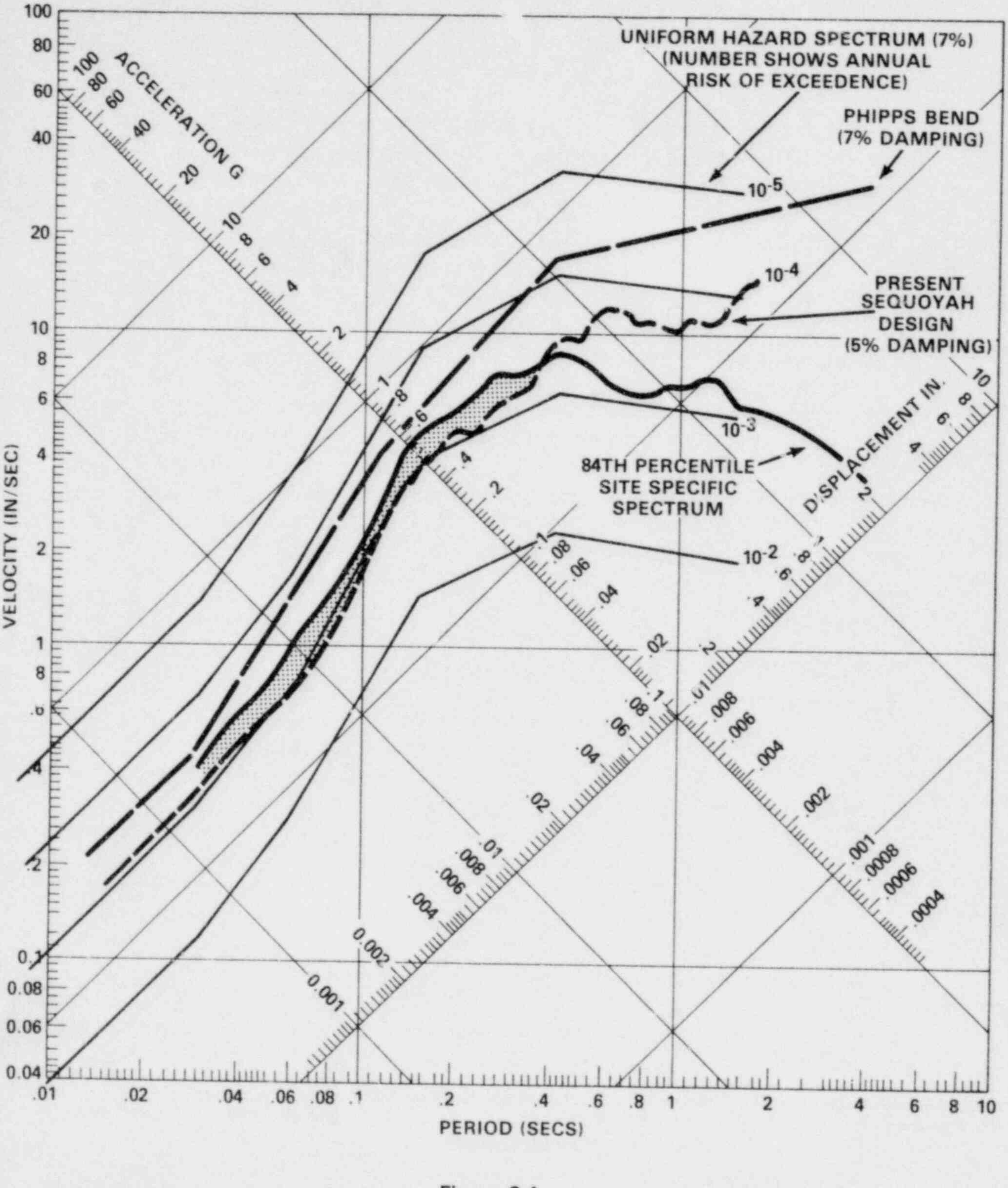

Figure 2-4

Comparison Of 7% Damped Uniform Hazard Response Spectra For The Sequoyah Site With The Present Sequoyah Design Spectrum For Reinforced Concrete, The 7% Damped 84th Percentile Site Specific Spectrum And The Phipps Bend Design Spectrum For Reinforced Concrete.

The seismic hazard associated with the present Sequoyah design is on the average five times greater than that associated with the Phipps Bend safe shutdown earthquake. The worst case is at 0.06 seconds, where the increase in hazard is about a factor of nine.

A factor of two or three in relative seismic hazard (present design versus site-specific safe shutdown earthquake) seems very small when compared to absolute hazards which are of the order of 10^{-3} or 10^{-4}. In our judgment, there must certainly exist differences in seismic hazard between other plants in the eastern United States that also exceed factors of two or three.

The applicant was also asked to study the sensitivity of these results due to a possible incompleteness in the original strong-motion data set. As an extreme case, four additional sets of the same records which showed the strongest ground motion (those recorded at Tolmezzo, Italy, from a magnitude 6.2 earthquake) were added and 50th and 84th percentile response spectra were recomputed. The present seismic design was still found to lie between the 50th and 84th percentile of periods of interests. The greatest departure from the 84th percentile appears to be at 0.06 secon s where the present design lies at about the 55th percentile. The annual risk c exceedance of the 84th percentile spectrum lies between 10^{-3} and 10^{-4} over most of the frequency range of interest. The relative difference in seismic hazard between this site-specific spectrum and the present design ranges from a factor of 4 to 8.5 with an average of 5.4. It should be pointed out that a deficiency in the data set of only high strong-motion records is most unlikely. While strong-motion records equal to or greater than the highest already included in the original data set will most certainly be recorded in the future, so will records showing lower motion. The recording of both "high" and "low" records in the well recorded Fruili, Italy, sequence supports this contention.

Operating Basis Earthquake

As currently defined, the operating basis earthquake used in the design of the Sequoyah Nuclear Plant was one-half the present safe shutdown earthquake. Based upon the seismic hazard calculations discussed above, it has a return period of about 150 to 300 years. This is acceptable in light of the Appendix A to 10 CFR Part 100 definition of the operating basis earthquake as being "that earthquake which . . . .could reasonably be expected to affect the plant site during the operating life of the plant."

Estimates of Response Spectra Associated with Modified Mercalli Intensity VIII

Spectral estimates of ground motion associated with intensity VIII would most likely result in more conservative spectra than that indicated by the 84th percentile site-specific spectrum. For example, use of the Trifunac and Brady (1975) relationship and the Regulatory Guide 1.60 spectrum would indicate that for reinforced concrete the present Sequoyah design in the periods of interest woul' be approximately equivalent to ground motion associated with an intensity VII earthquake while the site-specific spectrum would be approximately equivalent to ground motion associated with an intensity VII-VIII earthquake. The Phipps Bend design, as indicated previously, was determined assuming an intensity VIII earthquake. Most of the reasons for the different estimates using the Trifunac and Brady (1975) and Regulatory Guide 1.60 approach, as outlined in Section 2.5.2 of the present Standard Review Plan, were discussed above, but are worthwhile reiterating.

(1) There is very little strong moti_ data recorded at intensity VIII, and simple intensity-acceleration relationships are based to a large extent upon abundant data at lower intensities.

(2) Most of the data for the intensities of interest come from larger western United States earthquakes such as the magnitude 6.5 San Fernando, California, earthquake of 1971.

(3) There were some indications that the epicentral intensity of the 1897 Giles County earthquake wa a "weak" VIII and may not be equivalent in damage to intensity VIII normally associated with western United States earthquakes.

(4) The epicentral region of the Giles County earthquake and the Sequoyah Nuclear Power Plant have different site conditions.

The determination of a site-specific response spectrum utilizing a more reliable source magnitude, appropriate site conditions and epicentral distance allows a systematic consideration of the factors mentioned above and results in a better estimate of ground motion from the controlling earthquake than is determined utilizing only epicentral intensity.

Conclusions

It is our conclusion that the difference in associated seismic hazard (risk of design spectra being exceeded by earthquake ground motion) between the present design at Sequoyah and the appropriate site-specific response spectrum is not substantial. The reasons for this are:

(1) For reinforced concrete, the present design at Sequoyah represents a more than median description of the controlling site-specific ground motion.

(2) For reinforced concrete, the differences in seismic hazard are factors of two or three. This is small when compared to the absolute seismic hazard, which is on the order of 10^{-3} to 10^{-4}.

(3) In our judgment, there already exist variations in seismic hazard associated with design spectra for other plants in the eastern United States that exceed factors of two or three.

(4) The hazard associated with reinforced concrete represents a worst case and the difference in seismic hazard would be even less for other materials.

Findings

We have concluded above that the difference in seismic hazard between the present design at Sequoyah and the site-specific response spectrum is not substantial. In addition, because of such factors in the plant design as usage of lower-bound material properties, conservative analysis methods, and loading combinations that include low-occurrence-probability secondary events, a substantial additional margin to resist seismic loading exists in the plant's structures and equipment.

Based on all the above, we conclude that the present design basis for the Sequoyah Nuclear Plant is adequate to withstand the effects of earthquakes without loss of capability to perform the required safety functions. However, because the design spectra do fall below the site-specific spectrum in a particular frequency range, and to verify our judgement regarding structural margins, we are initiating a program to quantify the additional margins in representative critical sections of the reactor building and the auxiliary building structures, and in representative components required for safe shutdown. This program will confirm the capability of these structures and components to withstand the effects of the site specific earthquake without loss of capability to perform their required safety functions. The operating license will be conditioned to require such evaluations prior to startup following the first regularly scheduled refueling outage.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with General Design Criteria

In Section 3.0 of the Final Safety Analysis Report, the applicant presented an evaluation of the design bases against the Nuclear Regulatory Commission's General Design Criteria listed in Appendix A of 10 CFR Part 50. We evaluated the final design and the design criteria and conclude, subject to the applicant's adoption of the additional requirements made by us as discussed in this report, that the facility has been designed to meet the requirements of the General Design Criteria.

Our review of structures, systems and components relies extensively on the application of industry codes and standards that have been used as accepted industry practice. These codes and standards, as cited in this report and attached bibliography, have been previously reviewed and found acceptable by us; and have been incorporated into our Standard Review Plans.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

Criterion 2 of the General Design Criteria requires that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to 10 CFR Part 100 guideline exposures. Structures, systems and components important to safety that are required to be designed to withstand the effects of a safe shutdown earthquake and remain functional have been properly classified as seismic Category I items.

All other structures, systems and components that may be required for operation of the facility are designed to other than seismic Category I requirements. Included in this classification are those portions of seismic Category I systems which are not required to perform a safety function. Structures, systems and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in an acceptable manner in Tables 3.2-1 and 3.2-2 of the Final Safety Analysis Report.

The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria and design bases for structures, systems and components important to safety with the Commission's regulations as set forth in

General Design Criterion 2, and to Reg latory Guide 1.29, "Seismic Design Classification (Rev. 2)" technical staff positions, and industry standards.

We conclude that structures, systems and components important to safety that are designed in accordance with seismic Category I requirements provide reasonable assurance that the plant will perform in a manner providing adequate safeguards of the health and safety of the public.

3.2.2 System Quality Group Classification

General Design Criterion 1 requires that nuclear power plant systems and components important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicant has applied a classification system (Safety Classes A, B, C and D) which corresponds to the Commission's Quality Group A, B, C and D in Regulatory Guide 1.26 "Quality Group Classifications and Standards for Water, Steam and Radio-Waste-Containing Components of Nuclear Power Plants (Rev. 3)" to those fluid-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown condition, and (3) to contain radioactive material. These fluid systems have been classified in an acceptable manner in Tables 3.2-2 and 3.2-4 of the Final Safety Analysis Report and on system piping and instrumentation diagrams in the Final Safety Analysis Report based on conformance with Regulatory Guide 1.26, "Quality Group Classification and Standards."

The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves in fluid systems important to safety with the Commission's regulations as set forth in General Design Criterion 1, the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, technical staff positions, and industry standards.

We conclude that fluid system pressure-retaining components important to safety that are designed, fabricated, erected, and tested to quality standards in conformance with these requirements provide reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Wind and Tornado Loading

3.3.1 Wind Loading

All seismic Category I structures exposed to wind forces were designed to withstand the effects of the design wind. The design wind specified has a velocity of 95 miles per hour based on a r rence interval of 100 years.

The procedures that were used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with American Society of Civil Engineers Paper No. 3269.

We find the procedures in this paper utilized to determine the loadings on seismic Category I structures induced by the design wind specified for the plant acceptable, since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of design basis winds, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.3.2 Tornado Loading

All seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant were designed to resist a tornado of 300 mile per hour tangential wind velocity and a 60 mile per hour translational wind velocity. The simultaneous atmospheric pressure drop was assumed to be 3 pounds per square inch in 3 seconds. Furthermore, an appropriate spectrum of tornado-generated missiles was postulated.

The procedures that were used to transform the tornado wind velocity into pressure loadings are similar to those used for the design wind loadings as discussed in Section 3.3.1 of this report. The tornado missile effects were determined using procedures discussed in Section 3.5 of this report. The total effect of the design tornado on seismic Category I structures was determined by appropriate combinations of the individual effects of the tornado wind pressure, pressure drop, and tornado associated missiles. Structures are arranged on the plant site and protected in such a manner that collapse of structures not designed for the tornado will not affect other safety-related structures.

The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of a design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of General Design Criterion 2.

3.4 Water Level (Flood) Design

The design flood level resulting from the most unfavorable condition or combination of conditions that produce the maximum water level at the site is discussed in Section 2.4, Hydrology. The hydrostatic effect of the flood was considered in the design of all seismic Category I structures exposed to the water head.

Procedures following the methods outlined in U.S. Army Coastal Engineering Research Center Technical Report No. 4, "Shore Protection, Planning, and Design," were utilized to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant. These are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The reactor building will be maintained dry during the time when flood waters exceed plant grade. The diesel generator building is expected to remain dry since its lowest floor is at elevation 722 above mean sea level and its doors are on the uphill side facing away from the main body of flood water. All other structures, including the service, turbine, auxiliary, and control buildings, will flood as the water exceeds their grade level entrances. All equipment, including power cables, located in these structures and required for operation in the flood mode is either above the design basis flood or designed for submerged operation.

Based on our review, we conclude that the design meets the requirements of General Design Criterion 2 with respect to the protection of essential equipment from the effects of groundwater and design basis flooding, and is therefore acceptable.

3.5 Missile Protection

In accordance with General Design Criteria 2 and 4, the plant seismic Category I structures, systems and components will be shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado-generated missiles and various missiles that may result from equipment failure both inside and outside the containment.

Adequate information has been provided indicating the structures, shields, and barriers are designed to resist the effect of missiles. The missiles applicable to each of these structures, shields, and barriers are also adequately identified and their characteristics defined.

3.5.1 Missile Selection and Description

Facility Equipment Generated Missiles - Inside Containment

The Sequoyah plant is designed so that missiles inside containment will not cause an accident or increase the severity of an accident. Safety-related systems needed to bring the plant to a safe shutdown are protected against loss of function due to missile impact.

The applicant has confirmed that potential missile sources inside containment and potential missile paths are considered in the design. Pressurized components and rotating machinery are identified as potential missile sources inside containment. These include retaining bolts, control rod drive assemblies, valve bonnets, and valve stems. Protection provided against such potential missiles includes preferential orientation of potential missile sources, missile barriers, and physical separation of redundant safety systems and components. Missile impact calculations were performed to assure the adequacy of the barriers.

We conclude that the applicant's procedures for considering missiles and their consequences provide reasonable assurance that essential structures and systems will be protected against potential missiles inside containment in conformance with General Design Criterion 4 and are acceptable.

Facility Equipment Generated Missiles - Outside Containment

Missile protection is provided to ensure safe shutdown capability of the reactor facility. Pressurized components and rotating machines have the potential to become facility equipment generated missile sources. Protection against missiles outside containment is achieved by proper orientation of components and systems, by use of missile barriers, and by physically separating redundant safety-related systems or components from each other so that a potential missile cannot damage both redundant trains of the system and prevent safe shutdown of the reactor.

As a result of our review, we conclude that the design is in conformance with General Design Criterion 4 as it relates to structures housing essential systems and to the systems being capable to withstand the effects of facility equipment generated missiles, and Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," as it relates to protection of spent fuel pool systems and fuel assemblies from facility equipment missiles, and is therefore acceptable.

Turbine Missiles

The applicant has arranged Unit 1 and 2 turbine generators in a peninsular orientation. With the exception of the essential raw cooling water intake structure, this configuration excludes all systems important to safety from the low trajectory turbine missile strike zones. We have evaluated the strike probability with respect to the intake structure in the event of a turbine failure and find that it is less than 10^{-3}. Thus, the plant configuration is within the guidelines of Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles (3/76)," and we conclude that the risk of turbine missile damage to safety-related plant structures, systems and components for the facility is acceptably low.

Tornado Missiles

In accordance with Revision 1 of Standard Review Plan Section 3.5.1.4, applicants for operating licenses who were not required to design to the missile spectrum described in Revision 0 of that section during their construction permit revie should provide sufficient protection at least against the following postulated tornado missiles:

- Steel Rod, 1 inch diameter, 3 feet long, weight 8 pounds, traveling horizontally at 316 feet per second and vertically at 252 feet per second, at all elevations.

- Utility Pole, 13½ inches diameter, 35 feet long, weight 1690 pounds, traveling horizontally at 211 feet per second and vertically at 169 feet per second, at all elevations less than 30 feet above grade within one-half mile of the facility structures.

We have reviewed the applicant's design with respect to these missiles. Specifically, we have reviewed all areas of seismic Category I structures housing equipment important to safety with respect to barriers which are significantly less than two feet of concrete. We consider concrete at least two feet thick with a strength of 4000 pounds per square inch to be adequate protection against all postulated tornado missiles in this region, and do not require any additional evaluation. The areas found to be in this category are listed in Table 3-1.

TABLE 3-1

	Area	Concrete Thickness (inches)	Elevation Above Grade (feet)
a)	Portions of Auxiliary Building Roofs	9-1/2 to 13-1/2	58 to 86
b)	Diesel Generator Building	10-1/2 to 12	48-1/2 to 51-1/2
c)	West Main Steam Valve Room Roof Blowout Lids	4	24
d)	Portions of East Main Steam Valve Room Walls	12	48 to 53
e)	East and West Main Steam Valve Roofs	*	45 (East) to 60 (West)
f)	Control Building Roof Over New Air Intake	14 to 17	71-1/2

*Roofs consist of diaphragm-type 22 gauge metal decking covered with a built-up roof.

The elevations of the areas listed in Table 3-1, with the exception of a west main steam valve room roof, are over 30 feet above plant grade. This limits the protection considerations for these areas to the potentia' effects of the one inch diameter steel rod. Our calculations as modified by results of independent full-scale tests, indicate that penetration by the one inch diameter steel rod into reinforced concrete could be as much as four inches.

Of the areas above 30 feet elevation listed in Table 3-1, only the main steam valve rooms (item e) are not covered by concrete roofs. The rest (i.e., items a, b, d, and f) are protected by concrete barriers of at least nine and one-half inches of concrete, which is judged to be adequate with respect to the steel rod on the basis of the penetration estimates described above.

The west steam valve room roof consists of a 22 gauge metal decking covered with a built-up roof and supported by three 24 inch wide flange beams and many eight inch steel channels. Below the roof is a one-half inch steel grating floor which is also supported by three 24 inch wide flange beams and many eight inch steel channels. Below this are four additional levels consisting of wide flange beams (33 inch to eight inch size) provided for pipe break restraints and support functions. The main steam and main feedwater isolation valves and auxiliary feedwater supply piping is below all of the above structural components. While significant credit cannot be given to the built-up roof and decking for resisting the one inch steel rod, the one and one-half inch steel floor grating by itself is sufficient to prevent the steel rod from reaching the main steam and main feedwater isolation valves and piping inboard of the valves, without taking credit for the built-up roof or the labyrinth path created by the many levels of wide flanged beams, and the safety and relief valving and piping.

The east steam room valve roof is also of the same type, i.e., a 22 gauge metal decking covered by a built-up roof structure and supported by three 24 inch wide flange beams and many eight inch channels. About one-third of the roof is screened by a 12 inch thick concrete awning. There are four levels of wide flange beams (33 inch to eight inch size) between the roof and the main steam and main feedwater isolation valves. In addition, the main steam relief and safety valves are located above the isolation valves. Therefore, we conclude that the probability of a one inch steel rod entering the east steam valve room in a nearly vertical direction and reaching the valve elevations without interacting with the structural features described above, and ultimately penetrating the main steam or main feedwater guard and inner piping upstream of the isolation valving, is acceptably low.

Based on the above, we conclude that the plant systems within areas a, b, d, e, and f listed in Table 3-1 are protected against missiles descending from elevations greater than 30 feet above plant grade.

The only area in Table 3-1 that is at an elevation less than 30 feet above plant grade is the portion of the west main steam valve room roof which contains several circular four inch thick concrete blowout lids (Item c). This area houses a main feedwater pipe and a main feedwater isolation valve both of which are located below a 36 inch main steamline. At this elevation, the potential effects due to the utility pole as well as the one inch steel rod must be considered. The four inch blowout lids are marginally adequate in stopping the steel rod. However, we would expect that scabbing would occur. Also, we do not believe that there is sufficient assurance that the lids would be effective in completely stopping the utility pole. Thus it is possible that the utility pole or secondary missiles due to the steel rod could enter the west main steam valve room. These missiles, however, would have to penetrate the 36 inch main steam pipe (downstream of the main steam isolation valve) prior to striking the main feedwater piping inboard of the isolation valve. Neither the scabbing fragments formed by the steel rod nor the utility pole penetrating the four inch blowout lids could penetrate the 36 inch main steamline (an effective thickness of two inches of steel) and still have sufficient residual energy to cause damage to the main feedwater valve or inner pipe. Based on this, we conclude that plant systems within area c of Table 3-1 are adequately protected against tornado missiles.

In addition to considering the areas listed in Table 3-1, we have also reviewed the potential for tornado missiles entering portions of the auxiliary building via several of the blowout panels located on the north and south ends of the building roof between Unit 1 and 2 containments. Since the auxiliary building is well over 30 feet above plant grade, we considered only the steel rod as the appropriate missile. The safety related systems that could lie in a missile path are two redundant component cooling surge tanks, two redundant essential control air compressors, and the spent fuel storage assemblies.

The surge tanks and essential control air compressors are shielded from above by a reinforced concrete slab which supports other equipment, and a two inch steel grating below the concrete. This provides adequate protection against the steel rod.

The spent fuel pool is located approximately in the center of the auxiliary building, between the Unit 1 and unit 2 containment buildings. Potential missiles entering through the blowout panels in the west end of the auxiliary building could reach and enter the spent fuel pool. However, the steel rod is not hydrodynamically stable so that it would be expected to tumble in its descent through the 23 feet of pool water. Previous evaluations of this type indicate that no more than a few assemblies would sustain some damage with potential for releasing the gap activity. In addition, any released airborne radioactivity is expected to be highly dispersed by the tornadic winds in the area. Based on the above, we estimate that the radiological consequences of a tornado missile damaging spent fuel would be well within the values of 10 CFR Part 100.

We had previously expressed concern about the old and new essential raw cooling water intake structures, and the diesel generator building doors and exhaust stacks. In response, the applicant has acceptably redesigned these features to resist all appropriate tornado missiles by added missile shielding and shortening the diesel generator exhaust stacks.

Based on all the above, we conclude that all plant systems are adequately protected against tornado missiles.

3.5.2 Barrier Design Procedures

The analysis of structures, shields, and barriers to determine the effects of missile impact is accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact is investigated. This is accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile is determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, are combined with other applicable loads as is discussed in Section 3.8 of this report.

The design procedures used to determine the effects and loading on seismic Category I structures by design basis missiles selected for the plant provide a conservative basis for engineering design to assure adequate protection from the effects of missile impacts.

The use of this information provides reasonable assurance that, in the event of design basis missiles stroking seismic Category I structures, the structural integrity of structures would not be impaired or degraded to an extent that would result in a loss of required protection. Seismic Category I systems and components located within these structures are, therefore, expected to be adequately protected against the effects of missiles. Conformance with these missile protection design procedures is an acceptable basis for satisfying the requirements of General Design Criterion 4.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Inside Containment

The criteria used by the applicant for identifying high-energy fluid piping and for postulating pipe break locations, break orientations, and break flow areas inside containment provide a level of protection equivalent to that provided in Regulatory Guide 1.46. "Protection Against Pipe Whip Inside Containment."

These provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide acceptable assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the safe shutdown earthquake and a concurrent single pipe break of the largest pipe at one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potentially multiple failures of piping.

(2) The reactor emergency core cooling systems can be expected to perform their intended function.

Pipe motion subsequent to rupture and the pipe restraint dynamic interaction analyzed by the use of an elastic-plastic lumped mass beam element model sufficiently detailed to reflect the structural characteristics of the piping system.

The applicant has referenced Topical Report WCAP-8082, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop", as the basis for its conclusion that the proposed quantitative protection criteria for the reactor coolant system piping will provide an equivalent level of protection to that recommended in Regulatory Guide 1.46. The staff has reviewed and approved Topical Report WCAP-8082 (see our letter to Westinghouse dated May 22, 1974). The applicant has stated and we concur that the design of the Sequoyah reactor coolant system piping is similar to the design used in WCAP-8082.

On the basis of our review, we conclude that the above cited criteria used for the identification, design, and analysis of piping systems where postulated breaks may occur constitute an acceptable design basis for meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15.

3.6.2 Outside Containment

The criteria used by the applicant in postulating pipe rupture and leakage locations in high and moderate energy piping systems outside containment provide a level of protection equivalent to that provided by the letter from J. F. O'Leary (NRC) dated July 12, 1973, concerning the same subject.

Based on our review of the information submitted by the applicant, we conclude that the criteria used for postulating pipe rupture and leakage locations in high and moderate energy pipes outside containment constitute an acceptable design basis for satisfying the applicable requirements of General Design Criterion 4 of Appendix A to 10 CFR Part 50.

We have reviewed the applicant's design criteria and bases for protection against postulated piping failures in fluid systems outside containment and found that they are consistent with our positions as stated in this section. Further, the applicant provided the necessary analyses which were performed in accordance with the criteria as delineated in Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,"

Appendix B (criteria of A. Giambusso letter of December 15, 1972) for high energy lines, and APCSB 3-1 Section B3 for moderate energy lines.

The plant design will accommodate the effects of postulated pipe breaks outside containment with respect to pipe whip, jet impingement, and resulting reactive forces for piping systems. The general plant arrangement and the layout design of high-energy systems utilize the possible combinations of physical separation, pipe whip restraints, and piping systems that are enclosed in suitably designed structures or compartments. In addition to pipe whip restraints, pipe sleeves are incorporated as pipe whip and jet impingement protection for postulated longitudinal ruptures of high-energy lines. The design ensures that the consequences of a break can be mitigated so that the reactor can be shut down safely and maintained in a safe shutdown condition. Redundancy of essential equipment, including electrical components, is provided as necessary. Operability of essential systems and components in the vicinity of a pipe break are included in the equipment specifications which list the environmental conditions based on conservative design. The integrity of the control room, diesel rooms, and all other seismic Category I structures housing essential equipment is assured by structural isolation wherever possible. Based on our review, we conclude that the applicant has developed a plant design such that the postulated pipe breaks outside of containment will not prevent the safe shutdown of the reactor facility.

Based on our review we conclude that the protection against dynamic effects of postulated piping failures in fluid systems outside containment is in accordance with our requirements and guidelines, and is acceptable.

3.7 Seismic Design

3.7.1 Seismic Input

The seismic Category I structures, systems and components have been designed for a safe shutdown earthquake having a maximum horizontal acceleration of 0.18 times the acceleration of gravity and a maximum vertical acceleration of 0.12 times the acceleration of gravity.

Ground response spectra similar to and more conservative than that developed by Housner (which have been commonly used and accepted for design of seismic Category I structures) were selected for Sequoyah.

Damping values used in the dynamic analyses are as presented in Table 3.7-2 of the Final Safety Analysis Report.

Our review of the seismic analysis and the seismic design of the Sequoyah plant, as discussed below, was based on the above noted seismic input.

3.7.2 Seismic Analysis

The scope of review of the seismic system and subsystem analysis for the plant included the seismic analysis methods for all seismic Category I structures, systems, and components. It included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, and evaluation of seismic Category I structure overturning. The review included design criteria and procedures for evaluation of interaction of non-seismic Category I structures and piping with seismic Category I structures and piping and effects of parameter variations on floor response spectra. The review also included criteria and seismic analysis procedures for reactor internals and seismic Category I buried piping outside the containment.

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

The lumped soil spring approach is used to evaluate soil-structure interaction effects upon seismic responses.

We conclude that the seismic system and subsystem analysis procedures and criteria used by the applicant as discussed above provide an acceptable basis for the seismic design.

3.7.3 Seismic Instrumentation Program

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 structure comply with Regulatory Guide 1.12, Revision 1, "Instrumentation for Earthquakes." Supporting instrumentation is being installed on seismic Category I structures, systems and components in order to provide data for the verification of the seismic responses determined analytically for such seismic Category I items.

The installation of the specified seismic instrumentation in the reactor containment structure and at other seismic Category I structures, systems and components, which complies with Regulatory Guide 1.12, constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1 12, Revision 1.

3.8 Design of Seismic Category I Structures

3.8.1 Steel Containment

Each reactor and its cooling system is enclosed in a separate containment structure which consists of a free-standing cylindrical steel shell, hemispherical dome and reinforced concrete base located within a separate reinforced concrete reactor (or shield) building. The containment was designed, fabricated, constructed and tested as a Class MC vessel in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III. Loads include an appropriate combination of dead and live loads, thermal loads, seismic, and loss-of-coolant accident-induced loads, including pressure and jet forces.

The analysis of the containment was based on the elastic thin shell theory. The allowable stress and strain limits are generally those delineated in the applicable sections of Subsection NE of the ASME Code, Section III, for the various loading conditions.

We have reviewed the criteria used in the analysis, design, and construction of the steel containment structure. Based on our review, we conclude that the criteria account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime and that they are in conformance with established criteria, codes, standards, and guides acceptable to the staff, in accordance with criteria now included in Section 3.8.2 of the Standard Review Plan.

The use of these criteria; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occuring inside and outside the containment, the structure will withstand the specified conditions without impairment of structural integrity or safety function. A seismic Category I concrete shield building protects the steel containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2, 4, 16, and 50 of Appendix A to 10 CFR Part 50.

3.8.2 Concrete and Structural Steel Internal Structures

The containment interior structures consist of a shield wall around the reactor, secondary shield walls and other interior walls, compartments and floors, and the ice condenser structural systems. The ice condenser unit consists essentially of a well insulated cold storage room in which ice is contained in perforated metal baskets. The space between the baskets forms the channels for steam and air. The baskets are supported by a steel frame. The ice condenser system is contained in the annulus formed by the containment wall and the crane wall, circumferentially over a 300 degrees of arc. A refueling canal and equipment hatch are located in the remaining 60 degrees of arc. The interior structures were designed in accordance with the American Concrete Institute 318 Code for concrete and the Americans Institute of Steel Construction specifications for structural steel, which are acceptable to the staff.

The containment internal structures are designed and proportioned to remain within limits established by us under the various load combinations. These limits are, in general, based on the American Concrete Institute Standard 318-63 Code and on the American Institute of Steel Construction Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The use of these criteria; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.8.3 Other Seismic Category I Structures

The shield building for each unit is a reinforced concrete structure consisting of a flat foundation mat, a cylindrical wall, and a shallow dome. The foundation mat, common to the shield building, containment vessel, and other interior structures, forms the only structural tie between these structures. The shield building was designed in accordance with the applicable criteria of American Concrete

Institute 318, "Building Code Requirements for Reinforced Concrete," including modifications as required by the staff.

Seismic Category I structures other than the containment and its interior structures are all of structural steel and concrete. The structural components consist of slabs, walls, beams, and columns, except for the shield building which is of shell construction. The design method for concrete is in accordance with that specified in the American Concrete Institute 318 Code. Structural steel components are designed in accordance with the American Institute of Steel Construction specifications. These documents are acceptable to the staff.

The criteria used in the analysis, design, and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff, as now included in Standard Review Plan Section 3.8.4.

The use of these criteria; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4 of Appendix A to 10 CFR Part 50.

3.8.4 Foundations

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

The criteria used in the analysis, design, and construction of all the plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff, in accordance with criteria now included in Section 3.8.5 of the Standard Review Plan.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria, the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4 of Appendix A to 10 CFR Part 50.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

Piping Vibration Operational Test Program

In accordance with the ASME Code, Section III, paragraph NB-3622.3 and NC-3622, which requires that the designer be responsible, by observation during startup or initial operation, for ensuring that the vibration of piping systems is within acceptable levels, the applicant will conduct a piping vibration operational test program. The preoperational vibration dynamic effects test program which will be conducted on all ASME Class 1 and Class 2 piping systems and piping restraints during startup and initial operation conditions testing is an acceptable program in accordance with the guidance described in Standard Review Plan Section 3.9.2.

The tests will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design operational transients. The planned tests will develop loads similar to those experienced during reactor operation. Compliance with this test program constitutes an acceptable basis for satisfying the applicable requirement of General Design Criteria 15 of Appendix A to 10 CFR Part 50.

Analysis and Testing of Mechanical Equipment

To seismically qualify all seismic Category I mechanical equipment, the applicant has performed appropriate dynamic testing programs and analyses. Subjecting the equipment and its supports to these dynamic testing and analysis procedures provides reasonable assurance that in the event of an earthquake at the site, the seismic Category I mechanical equipment as identified in the Final Safety Analysis Report will continue to function during and after a seismic event. The combined loading imposed on the equipment and its supports under such loading combinations, in accordance with the guidance described in Standard Review Plan Section 3.9.2, provides an acceptable basis for the design of the equipment supports to withstand the dynamic loads associated with seismic events, as well as operational vibratory loading conditions without gross loss of structural integrity.

Implementation of these dynamic testing and analysis procedures constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 14.

The applicant has provided a description of a proposed program for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The program includes both baseline preservice testing and periodic inservice testing. It provides for both functional testing of components in the operating state and for visual inspection for leaks and other signs of degradation. In accordance with the requirements of Section 50.55a(g) of 10 CFR Part 50, the applicant proposes the period for which the program is applicable as follows: (1) From the issuance of the facility operating license to the start of facility commercial operation, inservice testing of ASME code class 1, 2, and 3 pumps and valves will be performed in accordance with Section XI, 1974 Edition through Summer 1975 addenda; (2) Following the start of facility commercial operation, inservice testing of pumps and valves will be performed in accordance with the ASME Section XI Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g)(4)(iv).

The date of the applicant's construction permit (May 27, 1970) places this plant under 10 CFR 50.55a(g)(1) which requires compliance with Section XI editions of the ASME Boiler & Pressure Vessel Code to the extent practical. Since inservice testing requirements for pumps and valves were not included in the Code until the Summer 1973 addenda of the 1971 edition, the applicant has chosen to meet the requirements of 1974 Edition, through the Summer 1975 Addenda, to the extent practical, and has requested relief from certain Code requirements as permitted by the above cited regulation. Based upon our review of the program, we find that with the requested relief it is in conformance with the Code, and we therefore conclude that it is acceptable for the preservice testing phase of plant operation extending up to the start of commercial operation. Acceptability of the program for the period following commercial operation will be determined prior to the start of facility commercial operation. The license will be appropriately conditioned to assure implementation of an acceptable inservice testing program.

Preoperational Vibration Assurance Program for Reactor Vessel Internals

With regard to flow-induced vibration testing of reactor internals for Sequoyah Units 1 and 2, the applicant has referenced the Indian Point Unit 2 reactor (Docket No. 50-247) as the prototype design for a four-loop plant on which vibrational testing has been performed. The Sequoyah internals design is similar to the Indian Point Unit 2 internals design. Two deviations from Indian Point in the Sequoyah design are the 17 x 17 fuel assembly configuration instead of the 15 x 15 array and the modification of the upper internals to accommodate the upper head injection emergency core cooling system.

To evaluate these differences in design, Westinghouse is presently planning to instrument and test the upper head injection upper internals of the Sequoyah Unit 1 reactor in accordance with Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals" to confirm the adequacy of the 4-loop, 17 x 17 upper head injection upper internals assembly. If acceptable confirmatory tests are not completed, the applicant recognizes that it may be subject to operational limitations and/or other restrictions until acceptable tests are completed either at their facility or another acceptable facility.

If a satisfactory prototype is established, the applicant has proposed additional confirmatory vibration testing and subsequent visual inspection as part of the Sequoyah preoperational tests to provide added confirmation of the capability of the structural elements of the reactor internals to sustain flow-induced vibrations. The proposed program is consistent with Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals."

We have reviewed the preoperational vibration test program proposed by the applicant for verifying the design adequacy of the reactor internals under loading conditions that will be comparable to those experienced during operation. The combination of tests, predictive analysis, and post-test inspection provide adequate assurance that the reactor internals can be expected to withstand flow-induced vibrations without loss of structural integrity during their service lifetime. We have concluded that the proposed preoperational vibration test program, which conforms with Regulatory Guide 1.20, constitutes an acceptable basis for demonstrating the design adequacy of the reactor internals in satisfying the applicable requirements of General Design Criteria 2 and 14 of Appendix A to 10 CFR Part 50.

Analysis Methods for Loss-of-Coolant Accident Loadings

The applicant has performed a dynamic system analysis of the reactor vessel supports, the reactor internals, and the broken and unbroken piping loops. The dynamic system analysis provides an acceptable basis for confirming the structural design adequacy of the reactor supports and internals and the unbroken piping loops to withstand the combined dynamic effects of the postulated occurrence of a loss-of-coolant accident and a safe shutdown earthquake.

TVA has performed structural analyses for the Sequoyah reactor coolant system for loads induced by a loss-of-coolant accident resulting from postulated pipe ruptures. The structural analysis for the reactor pressure vessel support considers simultaneous application of the time-history loads on the reactor vessel resulting from the reactor coolant loop vessel nozzle mechanical loads, internal hydraulic pressure transients, and reactor cavity pressurization.

The applicant employed the Westinghouse analytical code, MULTIFLEX, to calculate the reactor internals hydraulic pressure loads. The reactor cavity pressure loads calculations were performed using TMD code and have been found acceptable as discussed in Section 6.2.1 of this Safety Evaluation Report. The reactor vessel supports, internals, and primary coolant piping were analyzed using an assumed break opening area for the postulated pipe ruptures at the vessel nozzles of 100 square inches and a double-ended rupture at the pump outlet nozzle. The loss-of-coolant accident loads were combined with other applicable faulted condition loads for the analyses. The results of these analyses indicate that the stresses and deformations are all within acceptable values and the structural integrity of these components and supports is assured.

We have reviewed the analytical methods and find that the analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor supports and internals will not exceed the allowable design stress and strain limits for the materials of construction as specified in Appendix F to the ASME Boiler and Pressure Vessel Code, Section III, and that the resulting deformation of the reactor internals is within acceptable limits.

The applicant has also performed structural analyses for steam generator and pressurizer support structures. However, the information provided by the applicant regarding the analyses and results is not complete as discussed in Section 6.2.1 of this Safety Evaluation Report. We will report the resolution of this matter in a future supplement to this Report.

3.9.2 ASME Code Class 2 and 3 Components

Plant Conditions and Design Loading Combinations

All seismic Category I systems, components, and equipment outside of the reactor coolant pressure boundary are designed to sustain normal loads, anticipated transients, the operating basis earthquake, and the safe shutdown earthquake, within design limits which are consistent with those outlined in Regulatory Guide 1.48, "Design Limits and Loading Conditions." The specified design basis combinations of loading as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in systems classified as seismic Category I provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) other upset, emergency, or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity.

We conclude that the applicant's design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components conform with the guidelines of Regulatory Guide 1.48 and constitute an acceptable basis for design in satisfying the applicable requirements of General Design Criteria 1, 2 and 4 of Appendix A to 10 CFR Part 50.

The applicant has recently been requested to submit additional information concerning bolted connections in linear component supports for all ASME Class 1, 2 and 3 piping systems and components which are required for safe shutdown of the plant or to mitigate the consequences of an accident. The information requested concerns the assumptions which were made with respect to support plate flexibility when determining the maximum load that would be applied to the support bolts under steady state and transient dynamic loading. The results of this evaluation will be presented in a future supplement to this report.

The applicant has conducted component test programs, supplemented by analytical predictive methods, which provide adequate assurance that the ASME Code Class 2 and 3 active valves and pumps will (a) withstand the imposed loads associated with, normal, upset, emergency and faulted plant conditions without loss of structural integrity, and (b) perform their "active" function (i.e., valve closure or opening) under conditions and combinations of conditions comparable to those expected when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated.

We have reviewed the component analysis and test programs and conclude that they conform with the guidelines of Standard Review Plan Section 3.9.3, and will provide reasonable assurance of active valve and pump operability.

Design and Installation of Pressure Relieving Devices (Class 2)

The criteria used in developing the design and mounting of the safety and relief valves of ASME Code Class 2 systems provide adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function. The criteria used for the design and installation of over-pressure relief devices in ASME Code Class 2 Systems constitute an acceptable design basis in meetir the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 of Appendix A to 10 CFR Part 50, and are consistent with Regulatory Guide 1.67, "Installation of Over-Pressure Protection Devices."

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

3.10.1 Discussion

The supporting information is contained or referenced in Section 3.10 of the Final Safety Analysis Report. We review this information as detailed in the Standard

Review Plan Section 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment," and also determine the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format for Safety Analysis Reports.

3.10.2 Findings

Our review of the results of the seismic testing and analysis of Class 1E sensors and components indicated that the seismic testing has not been completed (see Section 3.10.3).

We are pursuing the seismic qualification program of the applicant as discussed and referenced below. An onsite seismic audit was conducted as described below.

3.10.3 Qualification Program

Instrumentation and electrical components required to perform a safety function are designed to meet seismic Category I design criteria. Seismic requirements established by the seismic system analysis are incorporated into equipment specifications to assure that the equipment purchased or designed will meet seismic requirements equal to or in excess of the requirements for Seismic Category I components, either by appropriate analysis or by qualification testing.

The applicant has conducted a seismic qualification program for the balance-of-plant seismic Category I instrumentation and electrical equipment and the associated supports for this equipment to provide assurance that such equipment can be expected to function properly and that structural integrity of the supports will not be impaired during the excitation and vibratory forces imposed by the safe shutdown earthquake and the conditions of post-accident operation. The seismic qualification program described by the applicant is consistent with IEEE Standard 344, 1971, "Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations."

In addition, to address the concern of whether or not the original testing or analysis can be justified in light of our current criteria (IEEE Standard 344, 1975, as supplemented by Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants"), we have established a seismic qualification review team. This team visited the Sequoyah plant in 1976. The team inspected selected vital mechanical and electrical equipment as installed and identified concerns about the adequacy of the original qualification per IEEE-344, 1971 for some of the items that were inspected. The review of these items has not been completed. See Section 7.8 of this Safety Evaluation Report for additional comments. The resolution of this issue will be discussed in a supplement to this Safety Evaluation Report.

For the nuclear steam supply system instrumentation and electrical equipment, the staff conducted a generic review of Westinghouse supplied equipment, which applies to the Sequoyah plant, to determine the adequacy of testing previously performed by the vendor. See Section 7.2.2 of this Report for additional information.

3.10.4 Evaluation

We conclude that the seismic qualification testing program which has been implemented for seismic Category I instrumentation and electrical equipment as supplemented by the program described in Subsection 3.10.3, above, will provide adequate assurance that such equipment will function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation. We also conclude that this program constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 2, when those items which remain to be qualified have been seismically qualified and the onsite seismic audit has been completed.

4.0 REACTOR

4.1 General

The nuclear steam supply system design for Sequoyah Units 1 and 2 is similar to that reviewed and approved for the Trojan Nuclear Plant (Docket No. 50-344) with the following exceptions:

(1) The effective flow rate for Sequoyah is slightly higher.

(2) The reactor inlet temperature for Sequoyah is lower.

A comparison of the principal thermal-hydraulic design parameters is presented in Table 4-1.

4.2 Mechanical Design

4.2.1 Fuel Description

Description

Each Sequoyah fuel assembly consists of 264 fueled rods, 24 guide thimbles, and one instrumentation thimble, plus ancillary hardware, arranged in a 17x17 array. The instrumentation thimble is at the center of the assembly and facilitates the insertion of neutron detectors. The guide thimbles provide channels for inserting various reactivity controls. The fueled rods contain uranium dioxide ceramic pellets hermetically clad in Zircaloy-4 tubes. The assembly is supported at both ends by stainless steel nozzles. Alignment and traverse spacings are maintained by eight spacer grids located axially equidistant. A total of 193 fuel assemblies make up an individual core.

All fuel rods are internally pressurized with helium during final welding to minimize cladding compressive stresses during service. The level of prepressurization is designed to preclude cladding flattening. The specific level of prepressurization will be dependent upon the planned fuel burnup and will be determined prior to establishing technical specifications.

The Sequoyah fuel assembly design (17x17) is mechanically identical to the assemblies in Trojan, Farley, and Diablo Canyon, and similar to the previously used Westinghouse fuel assembly (15x15). Those mechanical aspects that differ are shown in Table 4-2. The differences are essentially geometric, resulting in a lower linear power density and other increased safety margins for the 17x17 fuel assembly.

The evaluation of the Westinghouse fuel mechanical design is based upon mechanical tests, in-reactor operating experience and engineering analyses. Additionally,

TABLE 4-1

THERMAL AND HYDRAULIC DESIGN PARAMETERS

	Sequoyah	Trojan
Reactor Core Heat Output, megawatts thermal	3411	3411
System Pressure, Nominal, pounds per square inch	2250	2250
Minimum Departure from Nucleate Boiling Ratio at Nominal Conditions		
Typical Flow Channel	2.22	2.04
Thimble (Cold Wall) Flow Channel	>1.81	1.71 Total
Thermal Flow Rate, pounds per hour	133.3×10^6	132.7×10^6
Effective Flow Rate for Heat Transfer, pounds per hour	127.8×10^6	126.7×10^6
Effective Core Flow Area, square feet	51.1	51.1 Average
Coolant Temperature		
Nominal Inlet, degrees Fahrenheit	545.7	552.5
Average Rise in Core, degrees Fahrenheit	67.8	66.9
Active Heat Transfer Surface Area, square feet	59,700	59,700
Active Heat Flux, Btu per hour-square foot	189,800	189,800
Maximum Heat Flux, for normal operation, Btu per hour-square feet	474,500	474,500
Average Thermal Output, kilowatts per foot	5.44	5.44
Maximum Thermal Output, for normal operation, kilowatts per foot	12.2	12.6
Heat Flux Hot Channel Factor, F_Q	2.25	2.32
Peak Fuel Central Temperature at 100 percent Power, degrees Fahrenheit	3400	3400

TABLE 4-2

FUEL MECHANICAL DESIGN COMPARISON

Design Parameter	Westinghouse Sequoyah Units 1 and 2	Westinghouse Typical Operation Fuel
FUEL ASSEMBLY		
Rod Array	17x17	15x15
Number of Fueled Rods	264	204
Number of Spacer Grids	8	7
Number of Guide Thimbles	24	20
Inter-rod Pitch, inches	0.496	0.563
Average Thermal Output (4 loop), kilowatts per foot	5.4	7.0
FUEL PELLETS		
Density (theoretical), percent	95	94
Fuel Weight/Unit Length (per rod) pounds per foot	0.364	0.462
FUEL CLADDING		
Outside Radius, inches	0.187	0.211
Thickness, inches	0.0225	0.0243
Radius/Thickness Ratio	8.31	8.68

the in-reactor performance of the fuel design will be subject to the continuing surveillance programs of Westinghouse and individual utilities. These programs continually provide confirmatory and current design performance information.

Thermal Performance

In our evaluation of the thermal performance of reactor fuel, we assume that densification of uranium dioxide fuel pellets may occur during irradiation in power reactors. The initial density of the fuel pellets and the size, shape, and distribution of pores within the fuel pellets influence the densification phenomenon. In-reactor densification (shrinkage) of oxide fuel pellets (a) may reduce gap conductance, and hence increase fuel temperatures, because of a decrease in pellet diameter; (b) increases the linear heat generation rate because of the decrease in pellet length; and (c) may result in gaps in the fuel column as a result of pellet length decreases -- these gaps produce local power spikes and the potential for cladding creep collapse.

The engineering methods used by Westinghouse to analyze the densification effects on fuel thermal performance have been previously submitted to the staff and approved by us for use in licensing. The methods include testing, mechanical analyses, thermal and hydraulic analyses, and accident analyses. The results of our review are reported in "Technical Report on the Densification of Westinghouse PWR Fuel" dated May 14, 1974. Additional information on densification methods can be found in NUREG-0085 "The Analysis of Fuel Densification."

Fuel performance calculations for Sequoyah have been performed with PAD-3.1, a Westinghouse fuel model that incorporates the effects of fuel densification. Subsequently, Westinghouse submitted new data that showed the fission gas release at high burnup (greater than 20,000 megawatt days per ton) was modeled in a nonconservative manner. An improved version of the fuel thermal performance code, PAD-3.3, was submitted in topical report WCAP-8720, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations." The improved code contains a revised fission gas release model to account for increased release at high burnup, and revised models for helium solubility, fuel swelling and fuel densification. In our letter to Westinghouse dated February 9, 1979, we indicated our acceptance of WCAP-8720, but required some restrictions for the application of PAD-3.3. These restrictions also apply to the previous code (PAD-3.1). Therefore, we now find that without further review the previous code is no longer acceptable for use in licensing applications. Since these restrictions for either fuel code will become significant only at high burnup, the analyses performed with the earlier model remain valid for the safety analysis of Sequoyah until the fuel burnup exceeds 20,000 megawatt day per ton. The operating license for Sequoyah will be conditioned to require a reanalysis of the fuel performance using the approved version of the improved code (PAD-3.3) prior to attaining burnups in excess of 20,000 megawatt days per ton.

We have reviewed Westinghouse topical report WCAP-8377, "Revised Clad Flattening Model," July 1974, which describes the details of a revised cladding flattening model and, for a given fuel region, predicts initial flattening time and the flattened rod frequency for pressurized rods containing relatively stable fuel. This revised analysis was based on the results of television examinations of irradiated fuel rods which indicated that the original flattening model significantly underpredicted the time and frequency of collapse. The "COLLAP" computer code is used to perform these calculations. The revised model was accepted for use in safety analyses related to licensing in our letter to Westinghouse dated February 14, 1975.

Mechanical Performance

Although limited operating experience exists on 17x17 fuel assemblies, substantially all of the in-reactor operating experience with Westinghouse fuel rods and assemblies is applicable to the Sequoyah fuel design since the 17x17 fuel assembly is only a slight extrapolation mechanically from the 15x15 fuel assembly. The current use of similar fuel rods and assemblies has yielded operating experience that provides confidence in the acceptable performance of the fuel assembly design. The range in design parameters for which in-reactor experience is specifically applicable has been tabulated in Table 4-3. The assemblies referred to in Table 4-3 have been irradiated for up to six years and have had peak exposures of 30 gigawatt days per metric ton, totaling more than 70 million megawatt hours of power generation.

During this power reactor service, a small fraction of the fuel rods have experienced defects. However, there has been no instance where cladding defects have threatened either the plant or the public safety. Cladding defects were caused by excessive manufacturing impurities, excessive coolant cross-flow velocities, and fuel pellet densification. Excessive manufacturing impurities have been eliminated by modifications to the manufacturing procedures and cross-flow velocities were reduced by modifications to baffle joints. Densification effects are discussed earlier in this section.

Verification tests on the 17x17 assemblies have been completed and reported in Topical Reports WCAP-8279, "Hydraulic Flow Test of the 17x17 Fuel Assembly," February 1974, and WCAP-8288, "Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," December 1973 and Addendum No. 1, March 1974. We have reviewed these topical reports and have approved them for use in the safety analysis in our letters to Westinghouse of October 22, 1975 (interim approval) and February 6, 1979, respectively.

The consideration of fuel rod bowing in the 17x17 design was previously analyzed by Westinghouse and documented in Topical Report WCAP-8346, "An Evaluation of Fuel Rod Bowing; December 1975. The topical report described an analysis of rod bowing

TABLE 4-3

RANGE OF DESIGN PARAMETER EXPERIENCE

Parameter	Range of Power Reactor Experience
Fuel Rod Array	14 x 14, 15 x 15, and 17 x 17
Rods per Assembly	179 and 264
Guide Thimbles per Assembly	16 to 24
Assembly Envelope, inches	7.76 to 8.43
Inter-rod Pitch, inches	0.563 to 0.463
Plenum Length, inches	3.27 to 6.69
Prepressurization, pounds per square inch absolute	14.7 to over 400
Diametral Gap, inches	0.0065 to 0.0075
Spacer Grids/Assembly	7 to 9
Fuel Column Height, inches	120 to 144

based upon deliberation of the potential mechanisms causing fuel rod bowing. The analysis appeared rigorous and compatible with the available data. The methodology of the topical report was approved in our letter to Westinghouse of January 9, 1976, with the requirement that observations of fuel rod bowing in modified fuel assemblies (rod-off-bottom) substantiate this methodology. Subsequent observations, however, indicated that the magnitude of rod bow was underpredicted. Consequently, Westinghouse has reassessed its analysis in light of this new information and has documented its findings in Topical Report WCAP-8692, "Fuel Rod Bowing," December 1975. In this report, Westinghouse has documented its rod bowing experience which to date is based upon the inspection of 26 different regions of fuel (about 25,000 fuel rods) including more than 70 assemblies at burnups beyond 27,000 megawatt days per ton. This experience has demonstrated the exposure (burnup) dependence of rod bowing.

The staff issued an interim safety evaluation report on WCAP-8692 in April 1976, entitled "Interim Safety Evaluation Report in Westinghouse Fuel Rod Bowing." In this report the staff accepted the burnup-dependent approach to rod bowing used by Westinghouse with modifications to account for extension to the 17x17 design and an increase in rod bow from as-measured values (cold dimensions) to those in-reactor (hot dimensions). The effects of rod bowing on thermal hydraulic effects (departure from nucleate boiling) due to reduction in hot channel pitch are discussed in Section 4.4 below.

Seismic effects and vertical loads from postulated double-ended hot and cold leg breaks during the loss-of-coolant accident were analyzed in Topical Report WCAP-8288. We found this analysis acceptable. However, Westinghouse subsequently postulated a new asymmetric (horizontal) hydraulic load caused by a postulated pipe break within the biological shield. Westinghouse has performed a preliminary analyses which indicated that the fuel assemblies will be able to accommodate this load. In a letter dated March 1, 1976 (C. Eicheldinger to D. Vassallo, NRC), Westinghouse stated that although the experiments and calculational techniques supplied in WCAP-8288 may be applicable in assessing the adequacy of the fuel assembly to withstand these loads, it would be expected that they would be reviewed on a plant-by-plant basis. In Amendment 44 to the Sequoyah Final Safety Analysis Report, the evaluation of the response of the Sequoyah fuel assemblies to this load was presented. The results of the anaysis showed the fuel assembly deflections and the associated stresses and the spacer grid impact forces resulting from this load were below those that would cause permanent deformation, and are therefore acceptable.

We have reviewed the safety aspects of waterlogging fuel rod failures. A recent survey of available information included (a) results of tests in the capsule driver core at SPERT and the Japanese test reactor NSRR, and (b) observations of waterlogging failures in test and commercial reactors. It was concluded that (a) operating restrictions to reduce pellet/cladding interactions also reduce the potential for waterlogging failures during transients, (b) tests to simulate accident conditions produced the worst waterlogging failures, and (c) there is no apparent threat from waterlogging failures to the overall coolability of the core or to safe reactor shutdown. We will continue to monitor the waterlogging test programs, and if any modifications are indicated to maintain overall core coolability or to assure safe shutdown capability, we will require them to be made on Sequoyah.

Limitations on power rate changes could also affect pellet/cladding interaction, which is being reviewed generically. The Westinghouse 17x17 fuel rod design used in Sequoyah incorporates features which reduce cladding strain due to pellet/cladding interaction. These include pellet chamfering, rod prepressurization, and lower linear heat rating and cladding diameter-to-thickness ratio than the 15x15 design. Based on the available experimental and commercial reactor data, these design features should result in a reduction or delay of pellet/cladding interaction failures to later in the fuel design life. While the failure thresholds are probably lower at high burnup than at low burnup, the fuel duty is also less severe. Our review of the consequences of pellet/cladding interaction failures has so far not resulted in the identification of safety problems. Therefore, no restrictions are currently warranted. If any safety issues are identified in the future, however, appropriate restrictions will be implemented.

Fuel assembly fretting and wear test results for 17x17 fuel assemblies were reported in Westinghouse WCAP-8279. These tests indicated that fuel rod wear under both normal and transient operating conditions was within the Westinghouse predicted values and that even for fuel rods with deliberately damaged grid cells the wear was within acceptable limits. The staff reviewed the results of these tests and concluded that they provide an acceptable basis for demonstrating the overall adequacy of the 17x17 fuel assemblies. However, since these tests were performed on fuel assemblies with seven grids, we informed Westinghouse that further justification for applying the results to eight grid assemblies was needed. Westinghouse has since submitted the results of an eight grid 17x17 fuel assembly loop test with their letter of May 15, 1975, C. Eicheldinger to V. Stello. These tests showed no anomalous vibrations could be induced or were observed which simulated actual in-reactor conditions and therefore no modification to the 17x17 fuel assembly design was required. We concur with this conclusion.

Surveillance

Performance of the fuel is indirectly monitored by measurement of the activity of the primary coolant for compliance with technical specification limits. Westinghouse has proposed a fuel surveillance program for several plants that will use the 17x17 fuel assemblies. A summary of this program is given in the fuel rod bowing report, WCAP-8692. This program includes lead assemblies in the second fuel cycles for Surry Units 1 and 2 and the initial core loadings for Trojan, Beaver Valley Unit 1, Farely Unit 1, and Salem Unit 1.

The Surry Units each have two lead burnup 17x17 fuel assemblies. One of the lead assemblies in each unit has removable rods These assemblies were carefully measured prior to insertion and will be examined between cycles for dimensional changes, fretting corrosion near the spacer grids, fuel rod bowing, axial gamma distribution, cladding defects and surface deposits. Inspection after two cycles in Unit 1 and after the first cycle in Unit 2 has revealed no anomalies.

The other four reactors included in the surveillance program will each have an initial core loading of 17x17 fuel assemblies (Trojan, Beaver Valley Unit 1, Farley Unit 2, and Salem Unit 1). Each core will include one removable-rod assembly. Only two of the four, however, will be examined as part of the 17x17 fuel assembly surveillance program, and these will be selected on the basis of the first two to actually reload fuel. The surveillance program includes visual examination (100 percent television scanning) of the initial loaded (first core) fuel assemblies to be removed during the first three refueling outages. If any anomalies are detected, further examination will be performed using the removable fuel rod assemblies. The first surveillance results obtained after Cycle 1 (15,700 megawatt days per metric ton) at Trojan indicate that the fuel is performing in a satisfactory manner.

Conclusions

On the basis of the Sequoyah 17x17 fuel design analyses, technical specifications that will limit off-gas and effluent activity, and the confirmatory results from irradiated assemblies as discussed above, we conclude that there is reasonable assurance that the cladding integrity of Sequoyah Units 1 and 2 fuel will be maintained, that significant amounts of radioactivity will not be released, and that neither accidents nor earthquake-induced loads will result in either an inability to cool the fuel or interference with control rod insertion.

4.2.2 Reactor Vessel Internals

The materials for construction of components of the reactor internals have been included in specifications and found to be in conformance with the requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The materials for reactor internals exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials is expected to be negligible.

The controls imposed on reactor coolant chemistry have been reviewed and provide reasonable assurance that the reactor vessel internals will be adequately protected during operation from an environment which could lead to stress corrosion of the materials and loss of component structural integrity.

The requirements and controls on welding processes provide reasonable assurance that no deleterious hot cracking will be present during the assembly of austenitic stainless steel components. All weld filler metal was to be of selected composition to produce welds with at least five percent delta ferrite. Tests and examinations in accordance with Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code assure that adequate delta ferrite levels are met. The controls imposed to avoid sensitization during fabrication and processing of austenitic stainless steels satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

We have concluded that the material selection, fabrication practices, examination procedures, and protection procedures described by the applicant provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. The use of materials proven to be satisfactory by actual service experience and conformance with the recommendations of Regulatory Guide 1.44 constitutes an acceptable basis for meeting the applicable requirements of General Design Criteria 1 and 14 of Appendix A to 10 CFR Part 50.

The mechanical properties of structural materials selected for the control rod system components exposed to the reactor coolant satisfy Appendix I of Section III of the American Society of Mechanical Engineers Code, or Part A of Section II of the Code, and also our position that the yield strength of cold worked austenitic stainless steel should not exceed 90,000 pounds per square inch. The requirements and controls on welding processes provide reasonable assurance that no deleterious hot cracking was to be present during the assembly of austenitic stainless steel components. All weld filler metal was to be of selected composition to produce welds with at least five percent delta ferrite; and tests and examination in accordance with Section III of the American Society of Mechanical Engineers Code were required to assure that the adequate delta ferrite levels are met. The controls imposed in the application and processing of austenitic stainless steels to avoid sensitization satisfy the recommendations of Regulatory Guide 1.44. Fabrication and heat treatment practices performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the components.

The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria for Articles NB-2160 and NB-3120 of Section III of the American Society of Mechanical Engineers Code. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with our positions.

Conformance with the codes and Regulatory Guide 1.44, and with our positions on the allowable maximum yield strength of cold worked austenitic stainless steel and minimum tempering or aging temperatures of martensitic and precipitation-hardened stainless steels, constitutes an acceptable basis for meeting the requirements of General Design Criterion 26 of Appendix A to 10 CFR Part 50.

The design procedures and criteria that the applicant has used for the reactor internals are in conformance with established technical procedures, positions, standards, and criteria as cited above which are acceptable to the staff.

4.3 Nuclear Design

The nuclear design of Sequoyah Units 1 and 2 is the same as that of Trojan and Salem Unit 1. These reactors have been previously reviewed and approved. Sequoyah has rated power of 3411 thermal megawatts and consists of 193 assemblies containing the Westinghouse 17x17 rod fuel assembly array. Our review was based on information supplied by the applicant in the Final Safety Analysis Report and amendments thereto, and referenced topical reports. Our review was conducted within the guidelines provided by the Standard Review Plan, Section 4.3.

4.3.1 Design Bases

Design bases are presented which comply with the applicable General Design Criteria. Fuel design limits are specified which meet the requirements of General Design Criterion 10. A negative prompt feedback coefficient is required which satisfies General Design Criterion 11, and power oscillation is required either to be not possible or to be detected and suppressed by the control system, which satisfies General Design Criterion 12. A monitoring and control system is provided which automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation and anticipated transients. This satisfies General Design Criteria 13 and 20. The control system is designed so that no single failure or single operator error will cause a violation of fuel design limits and so that shutdown is assured even when the single rod cluster control assembly (control rod) of highest worth is assumed to be stuck out of the core. Further a chemical shim system is provided which is capable of controlling normal power changes and bringing the reactor to cold shutdown. The control system, when combined with the engineered safety features, is required to control reactivity changes during accident conditions. Reactivity insertion rates and amounts are controlled so that limited damage occurs to the pressure boundary and the core stays in coolable geometry. The reactivity control system meets the requirements of General Design Criteria 25, 26, 27 and 28. On the basis of the above, we find the design bases presented in the Final Safety Analysis Report to be acceptable.

Design Description

The Final Safety Analysis Report contains the description of the first cycle fuel loading which consists of three different enrichments and has a first cycle of approximately one year. The enrichment distribution, burnable poison distribution, soluble poison concentration and higher isotope (Plutonium) content as a function of core exposure are presented. Values are given for the delayed neutron fraction and prompt neutron life-time at beginning and end of cycle. The values presented are consistent with those normally used meet design bases and satisfy applicable sections of the General Design Criteria, and are acceptable.

Power Distribution

The design bases affecting power distribution are:

- The peaking factor in the core will not be greater than 2.25 during normal operation at full power in order to meet the initial conditions assumed in the loss-of-coolant accident analysis.

- Under abnormal conditions (including maximum overpower) the peak fuel power will not produce melting.

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

The applicant has described the manner in which the core will be operated and power distributions monitored so as to assure that these limits are met. The core will be operated in the constant axial offset control mode which has been shown generically to result in peaking factors less than 2.32 for constant power and load-following operation.

In order to demonstrate that the peaking factor will not exceed 2.25 in Sequoyah, rather than the generic 2.32, the applicant has done two things. First, the maximum azimuthal plane unrodded peaking factor (F_{xy}) has been reduced from the generic value of 1.55 to 1.52. Adherence to the azimuthal plane peaking factor limit is ensured by the surveillance requirements in the peaking factor Technical Specification. Second, the results of 18 cases of load-following transients analyzed using the rules of constant axial offset control, an axial model specific to the first cycle of Sequoyah, and the above azimuthal plane peaking factor have been provided. We have approved the use of such plant specific load-following analyses in Branch Technical Position CPB 4.3-1, "Westinghouse Constant Axial Offset Control (CAOC)."

The 18 case load follow analysis shows that the peaking factor for normal operation of the Sequoyah reactor will not exceed 2.25. Because the analysis described uses the assumptions (required in the Technical Specifications) of constant axial offset control and azimuthal plane peaking factor, and uses an approved analysis technique, we conclude that the assumption of a 2.25 peaking factor as an initial condition for the loss-of-coolant accident is valid for full power operation of the Sequoyah reactor.

The reactor will be provided with two types of monitoring instrumentation systems to measure core power distributions: a system of movable incore fission chamber detectors and a system o' fixed ion chambers located symmetrically around the core outside the reactor pressure vessel. The movable incore detectors will be capable of measuring the fuel rod peaking factor to within five percent and will be used to make periodic incore maps of the power distribution. The ion chambers located outside the reactor pressure vessel will provide an indication of total power, relative power in each quadrant of the core, and the relative power in the top and bottom of the core. Limits placed on the axial power offset, as measured from the relative power in the top and bottom of the core, and the radial tilt will ensure that (1) the core peaking factor can be maintained below the design limit value, and (2) all power distributions produced will be conservative relative to the design power distribution used in the departure from nucleate boiling analyses.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. The applicant has presented values of the coefficients in the Final Safety Analysis Report and has evaluated the uncertainties in these values. We have reviewed the calculated values of reactivity coefficients and have concluded that they adequately represent the full range of expected values. We have reviewed the reactivity coefficients used in the transient and accident analyses and conclude that they conservatively bound the expected values, including uncertainties. Furth -, moderator and power Doppler coefficients along with boron worth are measured as a part of the startup physics testing to assure that actual values are within those used in the analyses.

Control

To allow for changes in reactivity due to reactor heatup, load following, and fuel burnup with consequent fission product buildup, a significa: amount of excess reactivity is built into the core. This excess reactivity is controlled by a combination of full length control rods and soluble boron. Soluble boron is used to control reactivity changes due to:

- Moderator defect from ambient to operating temperatures
- Equilibrium xenon and samarium buildup
- Fuel depletion and fission product buildup - that portion not controlled by lumped burnable poison
- Transient xenon resulting from load following.

Regulating rods are used to control reactivity change due to:

- Moderator defect from hot zero to full power
- Power level changes (Doppler).

Burnable poison rods are used for radial flux shaping and to control part of the reactivity change due to fuel depletion and fission product buildup.

The applicant has provided data to show that adequate control exists to satisfy the above requirements with enough additional control rod worth to provide a hot shutdown effective multiplication factor less than the design basis value of 0.984 during initial and equilibrium fuel cycles with the most reactive control rod stuck out of the core. In addition the chemical and volume control system will be capable of shutting down the system by adding soluble boron and maintaining it shut down in the cold, xenon- and samarium-free condition at any time in core life. These two systems satisfy the requirements of General Design Criterion 26 of Appendix A to 10 CFR Part 50.

Comparisons have been made between calculated and measured control rod bank worth in operating reactors and in critical experiments. These comparisons lead to the conclusion that bank worths may be calculated to within approximately ten percent. In addition, bank worth measurements are performed as part of the startup test program to assure that conservative values have been used in safety analyses.

Based on these comparisons, we conclude that the applicant has made suitably conservative assessments of reactivity control requirements and that adequate reactivity has been provided to assure shutdown capability.

Provision is made in the design for the use of part length control rods. However, Westinghouse has informed us that the use of part length rods has not been completely analyzed. Until the analysis is completed, use of these rods will be prohibited by the Technical Specifications.

Control Rod Parameters and Reactivity Worths

The full-length control rods are divided into two categories - shutdown rods and regulating rods. The shutdown rods are always completely out of the core when the reactor is at operating conditions. Core power changes are made with regulating rods which are nearly out of the core when it is operating at full power. Regulating rod insertion will be controlled by power-dependent insertion limits which will be established to assure that:

- There is sufficient negative reactivity available to permit rapid shutdown of the reactor with adequate margin

- The worth of a control rod that might be ejected is not greater than that which has been shown to have acceptable consequences in the safety analyses.

In accordance with Standard Review Plan Section 4.3, we have reviewed the calculated rod worths and the uncertainties in these worths, and conclude that rapid shutdown capability exists at all times in core life assuming the most reactive control rod assembly is stuck out of the core in accordance with General Design Criterion 26, and is acceptable.

Stability

The stability of the Sequoyah core to xenon-induced spatial oscillations is discussed in the Final Safety Analysis Report. The overall negative reactivity (power) coefficient provides assurance that the reactor will be stable against total power oscillation. It is also concluded that sustained radial or azimuthal oscillations are not possible. This conclusion is based on measurements on an operating reactor of the same dimensions which showed stability against these oscillations.

Unstable axial oscillations are predicted to occur after about 12,000 megawatt days per ton of exposure for this core. The applicant has provided sufficient information to show that axial oscillations may be controlled by the regulating rods to prevent reaching any fuel safety limits.

4.3.2 Analytical Methods

A summary description of the methods used in the nuclear design of the Sequoyah reactor is presented in the Final Safety Analysis Report. Comparisons between calculation and experiment are also given which permit evaluation of uncertainties in the calculations. Based on the summary description and the reports referenced in the Final Safety Analysis Report, we find that the methods used are state-of-the-art and can calculate adequately the reactor physics characteristics of the Sequoyah cores, and are acceptable.

4.3.3 Summary of Evaluation Findings

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of the analyses to predict reactivity and physics characteristics of the Sequoyah Nuclear Plant.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to make the reactor subcritical wth an effective multiplication factor no greater than 0.984 in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position.

On the basis of our review, we conclude that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27 and 28 of Appendix A to 10 CFR Part 50.

4.4 Thermal-Hydraulic Design

The reactor is designed to operate at a core power of 3411 megawatts thermal which is the basis of the thermal-hydraulic design evaluation. The principal criterion for the thermal-hydraulic design of the reactor is to prevent fuel rod damage by providing adequate heat transfer for the various core heat generation patterns occurring during normal operation and operational transients (Condition I), and transient conditions resulting from faults of moderate frequency (Condition II). The following design bases are used to satisfy the above criterion:

(1) Departure from nucleate boiling will not occur on at least 95 percent of the limiting fuel rods at a 95 percent confidence level.

(2) There shall be no fuel melting.

(3) At least 95.5 percent of the flow is effective for core heat transfer.

(4) Permitted modes of operation shall not lead to hydrodynamic instability.

In order to show compliance with these design bases, the applicant performed departure from nucleate boiling and fuel temperature calculations as well as flow distribution and flow stability analyses. Departure from nucleate boiling calculations, performed with the THINC-IV code, are based on the W-3 departure from nucleate boiling correlation approved for application to 15x15 fuel designs. The departure from nucleate boiling ratios reported in the Final Safety Analysis Report are based on the W-3 correlation reduced by 15 percent for the 17x17 fuel design to allow for uncertainties in the extrapolation.

The first part of the departure from nucleate boiling tests, utilizing uniformly heated rods, was completed and reported in WCAP-8296, "Effect of 17x17 Fuel Assembly Geometry on DNB," which we accepted in our letter to Westinghouse dated

December 31, 1974. The results indicate that: (1) the previously used departure from nucleate boiling correlation (W-3 correlation with modified spacer factor) must be multiplied by 0.88 in order to show agreement with the 17x17 data; (2) the use of a thermal diffusivity coefficient of 0.038 is conservative; and (3) a departure from nucleate boiling ratio value of 1.275 corresponds to the 95/95 criterion. Further tests with simulated fuel rods employing non-uniform heat generation verified the acceptability of the W-3 correlation when multiplied by 0.88 and by the R-grid spacer factor.

There is additional margin in the departure from nucleate boiling model as calculated for the 17x17 fuel assembly design as follows:

Source	Margin(percent)
Departure from nucleate boiling calculations used as multiplier of 0.86 while data justify a multiplier of 0.88	2.3
A departure from nucleate boiling ratio of 1.3 was used in lieu of the 95/95 criterion. Data justify a ratio of 1.275	2.0
17x17 Pitch Reduction	1.7
A thermal diffusivity value of 0.051 was used in the data reduction while a value of 0.038 was applied in the analysis	1.2
Extra Grid	2.9

Thus, the initial Sequoyah design calculations offer a total departure from nucleate boiling margin of approximately 10 percent beyond the criteria.

Data have been reported (see our letter to Westinghouse dated June 19, 1978) which indicate that methods that account for the effect of fuel rod bowing on departure from nucleate boiling in a pressurized water reactor may not contain adequate margins when unheated rods, such as instrument tubes, are present. Further experimental verification of these data is being reviewed by the staff. As an interim measure, the staff requires that a burnup-dependent penalty factor be applied to the reactor operating limits to reflect reduced departure from nucleate boiling conditions caused by increasing rod bow. The enthalpy hot channel factor ($F^N_{\Delta H}$), a parameter which varies inversely with departure from nucleate boiling, is used to account for this penalty.

We defined interim methods for evaluating the effects of fuel rod bowing on thermal margin calculations. These methods conservatively consider burnup-dependent factors not included in initial design calculations. The table below shows the departure from nucleate boiling ratio reduction factor after credit is taken for initial design margin and the enthalpy hot channel factor correction to account for rod bowing effects.

Burnup	Net Reduction in departure from nucleate boiling ratio, percent	Net Correction in enthalpy hot channel factor, percent
0	0	0
15000	17.37	9.5
24000	21.9	12.2
33000	21.9	12.2

The rod bow penalty will be implemented in the technical specifications for Sequoyah Units 1 and 2.

The Sequoyah reactors were designed to operate at a higher reactor core heat output and a higher coolant nominal inlet temperature than Zion. This performance increase was based on the use of the THINC-IV Code which permitted a more detailed analysis of the thermal-hydraulic characteristics of the core. The THINC-IV Code was developed to consider crossflow between adjacent assemblies in the core and thermal diffusion between adjacent subchannels in the assembly. The THINC-IV Code is described in WCAP-7956, "THINC-IV: An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," June 1973, and WCAP-8195, "Application of the THINC-IV Program to PWR Design," October 1973, which we accepted in our letter to Westinghouse dated April 19, 1978.

The staff has reviewed the THINC-IV code and found it acceptable for performing steady-state core hydraulic calculations which are limited to conditions of single phase or homogeneous two-phase flow. Comparisons between tests results, which included severe flow blockage and the conservative omission of interassembly thermal mixing, to THINC-IV calculations were acceptable.

The applicant has used the HYDNA digital computer code to predict the hydrodynamic stability of parallel, closed channels. The program was compared to experimental results with good predictability.

The applicant calculated that flow instability would occur at 185 percent of normal power, using core coolant conditions typical of a Westinghouse four loop pressurized water reactor with a power rating of 3250 megawatts thermal similar to the Sequoyah units. The analysis considered the plant to be made of discrete, parallel flow channels.

The applicant contends that an open core, permitting cross flow between hotter and cooler channels at all points along the channel length, is more stable because of this cross-coupling than are cores with discrete, parallel channels, which do not permit cross coupling. In order to demonstrate this effect, a test was conducted consisting of three vertical parallel channels, two of which were heated. The two heated channels were connected by valves at six elevations along the channel. Upon detection of instability in the hot channel the valves were opened and the instability disappeared.

The total reactor design flow rate is based upon conservative estimates of the pump characteristics and hydraulic resistances in the primary loop. Tests during plant startup will verify the flow rates used in the design analyses. As a limiting condition of operation (to be specified in the Technical Specifications), the flow in any one loop must equal or exceed the prorated design value.

The applicant has reduced the inlet flow to the hot assembly by five percent in order to account for flow maldistribution. Information presented indicates that the departure from nucleate boiling ratio is relatively insensitive to inlet flow maldistribution.

Conclusions

The thermal-hydraulic design of the core for the Sequoyah plant was reviewed. The scope of review included the design criteria, core design, and steady state analysis of the core thermal-hydraulic performance. The review concentrated on the differences between the proposed core design and criteria and those designs and criteria that have been previously reviewed and found acceptable by the staff.

We conclude that the thermal-hydraulic design of the core conforms to applicable General Design Criteria and to applicable Regulatory Guides and staff technical positions as cited above, and is acceptable, based on implementation of the technical specifications discussed above.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

Each reactor coolant system consists of four similar heat transport loops connected to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator, and associated piping. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control. All of these components are located within the containment building.

During operation, the reactor coolant system transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine-generator. Borated demineralized water is circulated in the reactor coolant system at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The coolant also acts as a neutron moderator and reflector, and as a solvent for the neutron absorbing boric acid used for chemical shim control.

The reactor coolant system pressure boundary provides a second barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the plant.

The reactor coolant system pressure changes during normal operation are controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water spray. Spring-loaded safety valves and power-operated relief valves are mounted on the pressurizer and discharge to the pressurizer relief tank where steam is condensed and cooled by mixing with water.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

The design loading combinations specified for ASME Code Class 1 reactor coolant pressure boundary components have been appropriately categorized with respect to the plant condition identified as normal, upset, emergency or faulted. The design limits proposed by the applicant for these plant conditions are consistent with the criteria recommended in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." Use of these criteria recommended in Regulatory Guide 1.48 for the design of the reactor coolant pressure boundary components will provide reasonable assurance that in the event an earthquake or other system upset should occur at the site, and emergency or faulted conditions should develop, the resulting combined stresses imposed on the system components will not exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. We conclude that the design load combinations and associated stress and deformation limits specified for ASME Code Class 1 components constitute an acceptable basis for design in satisfying the related requirements of General Design Criteria 1, 2, and 4 of Appendix A to 10 CFR Part 50. See Section 3.9.2 of this report for a discussion of ASME Class 1 bolted connections.

Quality Group A reactor coolant pressure boundary components have been constructed to the maximum extent practical in accordance with the codes and addenda described in 10 CFR Part 50, Section 50.55a. The codes and addenda used for construction of the Quality Group A components are those that were required at the time of procurement of the components.

We reviewed the American Society of Mechanical Engineers codes and addenda used in the construction of the reactor coolant pressure boundary components and identified no major differences between these codes and addenda and those described in the codes and standards rule, 10 CFR 50.55a. We conclude that the American Society of Mechanical Engineers codes and addenda used in the construction of the Quality Group A reactor coolant pressure boundary components are designed in compliance with the codes and addenda described in 10 CFR 50.55a to the maximum extent practical; and that in accordance with considerations of 10 CFR 50.55a(a)(2), the Quality Group A components meet the requirements of 10 CFR 50.55a which provide adequate assurance that component quality is commensurate with the safety function of the reactor coolant pressure boundary.

The ASME Code Cases specified in Table Q3.21-1 of the Final Safety Analysis Report, whose requirements have been applied in the construction of Quality Group A components within the reactor coolant pressure boundary, are acceptable to the Commission. We conclude that compliance with these code cases, in conformance with the Commission's regulations, is expected to result in a component quality level commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

The applicant has identified the active valves within the reactor coolant pressure boundary whose operation is relied upon to safely shut down the plant and maintain it in a safe condition in the unlikely event of a safe shutdown earthquake or a design basis accident. The applicant has also stated that a component operability test program, supplemented by analytical methods, have been developed to provide additional assurance that these active components will (1) withstand the imposed loads associated with normal, upset, emergency and faulted plant conditions without loss of structural integrity, and (2) perform their "active" function (i.e., valve closure or opening), under conditions and combinations of conditions comparable to those expected when a safe plant shutdown is to be effected or the consequences of an accident are to be mitigated. We conclude that the program proposed by the applicant will provide reasonable assurance of valve operability.

5.2.2 Overpressurization Protection

Overpressurization protection in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000 (1971) is provided by three pressurizer safety valves and two power-operated relief valves. These valves discharge to the pressurizer relief tank.

The relief valves are designed to limit the pressurizer pressure to a value below the high pressure trip set point for all transients up to a 50-percent step load decrease with steam dump, but without reactor trip.

The safety valves are designed to limit the reactor coolant system pressure to less than 110 percent of the 2485 pounds per square inch gauge design pressure. The adequacy of the relief capacity is based on a complete loss of steam flow to the turbine with main feedwater flow being maintained and no reactor trip. Each of the three safety valves is rated at 420,000 pounds per hour steam flow with a set point of 2500 pounds per square inch absolute plus three percent accumulation.

The pressurizer safety valves and steam generator safety valves are sized to protect the reactor coolant system and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control assembly control, or direct reactor trip or turbine trip.

A loss of load transient was also analyzed for the case where the main feedwater flow is lost simultaneously with the loss of steam flow. For this transient, the system is protected against overpressurization by the reactor coolant system and steam generator safety valves in conjunction with the reactor protection system.

The staff concludes that the design of the pressure relief system for protection against the worst transient from hot operating conditions is in conformance with General Design Criteeria 15 and the ASME Code Section III.

A number of transients have occurred in operating pressurized water reactors in which the limits of 10 CFR Part 50, Appendix G have been exceeded during startup and shutdown operations. We require that the applicant provide acceptable protection equipment that will preclude violating these limits considering the various modes for initiating adverse pressure transients when the pressure vessel is cooled. If protection equipment is not installed prior to initial fuel load, the applicant must provide adequate justification for operation until such equipment is installed. In this event, the operating license will be appropriately conditioned to require equipment installation at a later date. We will report further on this matter in a supplement to this report. This protection equipment must be capable of meeting at least the following requirements:

1. Credit for operator action. No credit can be taken for operator action until 10 minutes after the operator is made aware that a transient is in progress.

2. Single failure criteria. The pressure protection system shall be designed to protect the vessel, given any event initiating a pressure transient. Redundant or diverse pressure protection systems will be considered as meeting the single failure criteria.

3. Testability. Provisions for periodic testing of the overpressure protection system(s) and components shall be provided. The program of tests and frequency or schedule thereof will be selected to assure functional capability when required.

4. Seismic Design and IEEE-279 criteria. The pressure protection system(s) should remain functional during and after an operating basis earthquake and meet IEEE-279 criteria. The basic objective is that the system(s) shall not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

The design and installation criteria for pressure relief devices on the reactor coolant pressure boundary are in accordance with the appropriate rules of Subsection NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III. The maximum full discharge loads resulting from the opening of ASME Code Class 1 safety and relief valves were calculated by either an equivalent static analysis or a time response dynamic analysis of the system. In the case of open safety or relief valves mounted on a common header and full discharge occurring concurrently, the additional stresses induced in the header were combined with previously computed local and primary membrane stresses to obtain the maximum stress intensity.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Code Class 1 systems provide adequate assurance that, under discharging conditions, the resulting stresses are expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function. The criteria used for the design and installation of overpressure relief devices in ASME Code

Class 1 Systems constitute an acceptable design basis in meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 of Appendix A to 10 CFR Part 50.

5.2.3 Materials

General Material Considerations

The materials used for components of the reactor coolant pressure boundary, including the reactor vessel and its appurtenances, have been identified by specification and found to be in conformance with the requirements of Section III of the American Society of Mechanical Engineers Code.

We have reviewed the materials of construction for the reactor coolant pressure boundary to ensure that the possibility of serious corrosion or stress corrosion is minimized. All materials used are compatible with the expected environment, as proven by extensive testing and satisfactory service performance. The applicant has shown that the possibility of intergranular stress corrosion in austenitic stainless steel used for components of the reactor coolant pressure boundary was minimized because sensitization was avoided and adequate precautions were taken to prevent contamination during manufacture, shipping, storage, and construction. Sensitization was avoided through appropriate controls on welding processes.

The use of materials with satisfactory service experience in other reactors and the controls placed on welding procedures provide reasonable assurance that austenitic stainless steel components will be compatible with the expected service environments and that the possibility of loss of structural integrity is minimized.

Further protection against corrosion problems will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled and the proposed maximum contaminant levels as well as pH, hydrogen overpressure and boric acid concentrations, have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems.

The possibility that serious corrosion or stress corrosion problems would occur in the unlikely event that emergency core cooling system or containment spray system operation is required will be minimized because the predicted pH of the circulating coolant will be between 8.0 and 8.5.

The requirements imposed upon the external insulation used on austenitic stainless steel conform acceptably with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

The controls on chemical composition that will be imposed on the reactor coolant and the use of external thermal insulation in conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," provide reasonable assurance that the reactor coolant boundary materials will be acceptably protected from conditions that would lead to loss of integrity from stress corrosion.

We have reviewed the controls used to prevent hot cracking (fissuring) of austenitic stainless steel welds. These precautions include control of weld metal compositions and welding processes to assure adequate delta ferrite content in the weld metal.

The methods compiled with Section III of the ASME Code and were in conformance with Regulatory Guide 1.31, "Control of Stainless Steel Welding." The use of materials, processes, and test methods that were in accordance with these requirements and recommendations provides reasonable assurance that loss of integrity of austenitic stainless steel welds caused by hot cracking during welding will not occur.

Fracture Toughness

We have reviewed the materials selection, toughness requirements, and extent of materials testing proposed by the applicant to provide assurance that the ferritic materials used for pressure-retaining components of the reactor coolant boundary will have adequate toughness under test, normal operation, and transient conditions. The ferritic materials are specified to meet the toughness requirements of the 1968 ASME Code, Section III.

The ferritic pressure boundary material of the reactor pressure vessel was qualified by impact testing in accordance with the ASME Code, Section III, 1968 Edition and evaluated in accordance with Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. We have evaluated the applicant's compliance with the fracture toughness requirements of Appendix G to 10 CFR 50. The results of our evaluation to date indicate that the applicant meets the requirements of Appendix G to 10 CFR 50 except that the orientation of the specimens used to determine upper shelf energy does not meet exactly the requirements in Paragraph IV.B. We have requested additional information from the applicant concerning other possible areas where the requirements of Appendix G may not have been met exactly. When the areas of noncompliance with 10 CFR Part 50 Appendix G are identified, we will determine whether, pursuant to 10 CFR Section 50.12, specific exemptions can be granted. If we grant such exemption for Sequoyah, our safety evaluation supporting this matter will accompany the granting documents.

The fracture toughness tests and procedures required for the reactor vessel by Section III of the ASME Code, as augmented by Appendix G, 10 CFR Part 50, with any appropriate exemptions, provide reasonable assurances that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant boundary.

Operating Limitations

The reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure, in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code, Summer 1972 Addenda, and Appendix G, 10 CFR Part 50. Additional conservatism in the pressure-temperature limits used for heatup, cooldown, testing, and core operation will be provided since their pressure-temperature limits will be determined assuming that the belting region of the reactor vessel has already been irradiated.

Although Westinghouse Topical Report WCAP-7924, "Basis for Heatup and Cooldown Limit Curves," has been accepted by the staff (see our letter to Westinghouse dated January 8, 1975), the method or determing the nil ductility transition reference temperature shift was not accepted. We have requested the applicant to provide information confirming that the pressure-temperature limits for reactor vessel heat-up and cooldown will be constructed using the prediction for temperature shift contained in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." When we receive this confirmation we will verify that the oprating limitations are such that the possibility of rapidly propagating failure are minimal, and will report on this matter in a supplement to this report.

The use of Appendix G of the Code as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the Code and Commission regulations, will ensure adequate safety margins during operating, testing, and maintenance conditions. Compliance with these Code provisions and 10 CFR Part 50, Appendix G, and Regulatory Guide 1.99 constitute an acceptable basis for satisfying the requiremen.. of General Design Criterion 31.

5.2.4 Leakage Detection System

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

The leakage detection system includes diverse leak detection methods. These will have sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practical, and will be provided with suitable control room alarms and readouts.

The major components of the system are the containment radiogas and atmosphere particulate radioactivity monitors, containment sump and pump operation system, and the condensate measuring system. Indirect indication of leakage will be obtained from the containment humidity, pressure, and temperature indicators. Intersystem leakage will be detected by abnormal readings from radioactivity monitors in the secondary system.

The applicant has complied with the requirements of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," except that the airborne particulate radioactivity monitoring system has not been specifically designed to remain functional when subjected to a safe shutdown earthquake, which requirement was subsequent to the purchase date of this equipment.

The diverse leakage detection systems used to detect leakage from components and piping of the reactor coolant pressure boundary in accordance with Regulatory Guide 1.45 provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. This degree of compliance with the recommendations of Regulatory Guide 1.45 constitutes an acceptable basis for satisfying the requirements of General Design Criterion 30.

5.2.5 Reactor Vessel Integrity

General

We have reviewed all factors contributing to the structural integrity of the reactor vessel and we conclude there are no special considerations that make it necessary to consider potential vessel failure for the Sequoyah Nuclear Plant, Units 1 and 2.

The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements conform to the rules of the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, all addenda through Winter 1968, and all applicable Code Cases. The fracture toughness requirements of the ASME Code, Section III, 1968 Edition have been met. We have identified one area where the applicant has not meet the exact requirements of Appendix G to 10 CFR 50. As noted in Section 5.2.3 above, we have requested additional information from the applicant concerning other areas which may not meet the exact requirements of Appendix G.

Operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer Addenda of the ASME Boiler and Pressure Vessel Code, Section III, and Appendix G, 10 CFR Part 50.

The integrity of the reactor vessel is assured because the vessel:

1. Was designed and fabricated to the standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent Code Cases listed above.

2. Was made from materials of controlled and demonstrated quality.

3. Was extensively inspected and tested to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.

4. Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation.

5. Will be subjected to monitoring and periodic inspection to demonstrate that the initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

6. May be annealed to restore the material toughness properties if this becomes necessary.

Reactor Vessel Material Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a materials surveillance program that meets the requirements of American Society for Testing Materals Standard E 185-73 and Appendix H, 10 CFR Part 50 (July 17, 1973).

The program identifies proper methods of assessment of changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation. Adequate safety margins against the possibility of vessel failure will be provided through conformance with the essential material surveillance requirements of American Society for Testing Materials Standard E 185-73 and Appendix H, 10 CFR Part 50. The applicant has stated that should results of tests indicate that the toughness is not adequate, the reactor vessel can be annealed to restore the toughness to acceptable levels. We agree that the methods proposed are feasible and would be effective if needed. The surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of General Design Criterion 31.

5.2.6 Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically.

Since Sequoyah received its construction permit prior to the first issuance of ASME Code, Section XI, its design does not necessarily provide accessibility for all inspections required by this Code. However, the applicant has stated that its inspection program will comply with the 1974 Edition of Section XI, including Addenda through Summer 1975, to the extent practical. The program will include examinations of Code Class 1, 2 , and 3 components.

We require that the inservice inspection program for Sequoyah Nuclear Plant for ASME Code Class 1, 2 and 3 components be in accordance with the revised rules in 10 CFR 50, Section 50.55a, paragraph (g). We have requested additional information concerning the applicant's inservice inspection program. When this information is submitted, we will evaluate the results to assure compliance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g). This matter will be satisfactorily resolved prior to issuance of an operating license.

The conduct of periodic inspections and hydrostatic testing of ASME Code Class 1, 2 and 3 components in accordance with the requirments of ASME Code, Section XI, specified in 10 CFR 50, Section 50.55a, paragraph (g), provides reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of General Design Criterion 32, Appendix A of 10 CFR Part 50.

We have reviewed the information submitted by the applicant and find that inservice inspection of the steam generator tubes is not included at this time. We require that the inservice inspection program for the steam generators be in accordance with the recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" and ASME Section XI with respect to the inspection methods to be used, provisions for a baseline inspection, selection and sampling of tubes, inspection interval and actions to be taken in the event defects are identified. Conformance with Regulatory Guide 1.83 and ASME Code Section XI constitutes an acceptable basis for meeting the applicable requirements of General Design Criteria 1 and 32.

We have requested additional information from the applicant regarding the inservice inspection of the steam generators, which the applicant has indicated will be provided shortly. When we receive this information we will verify that an adequate steam generator inservice inspection is performed to ensure the integrity of the reactor coolant pressure boundary is maintained. We will report further on this matter in a supplement to this report.

5.2.7 Reactor Coolant Pump Flywheel Integrity

Because flywheels have large masses and rotate at speeds of about 1200 revolutions per minute during normal reactor operation, a loss of integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features.

The potential for the reactor coolant pump flywheel to become a missile in the event of a rupture in the pump suction or discharge sections of reactor coolant system piping, is under generic study by Westinghouse and the staff. The electrical Power Research Institute has contracted Combustion Engineering, Incorporated to perform a 1/5 scale reactor coolant pump research program. The objective of the program will be, in part, to obtain empirical data to substantiate or modify current mathematical models used in predicting pump performance during a postulated loss-of-coolant accident. We will be following the development and performance of this program as well as other industry analytical and experimental programs on a generic basis.

If the results of the generic investigations of this matter indicate that additional protective measures are warranted to prevent excessive pump overspeed or to limit potential consequences to safety related equipment, we will determine what modifications, if any, are necessary to assure that an acceptable level of safety is maintained.

The probability of a loss of pump flywheel integrity can be minimized by the use of suitable material, adequate design, and inservice inspection. The use of suitable material and adequate design and inservice inspection for the flywheels of reactor coolant pump motors as specified in the Final Safety Analysis Report provides reasonable assurance (a) that the structural integrity of flywheels is adequate to withstand the forces imposed in the event of design overspeed transient without loss of their function, and (b) that their integrity will be verified periodically in service to assure that the required level of soundness of the flywheel material is adequate to preclude failure. The applicant states and we concur that they are in compliance with the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." Compliance with the recommendations of that Regulatory Guide constitutes an acceptable basis for satisfying the requirements of General Design Criterion 4, Appendix A, 10 CFR Part 50.

5.2.8 Loose Parts Monitor

The applicant had previously stated that he would evaluate the loose parts monitoring systems available, would keep abreast of development and experience in this field, and has incorporated in the Sequoyah design the flexibility to install a practical loose parts monitoring system.

Recently, prototype loose parts monitoring systems have been developed and are in operation and being installed at several plants.

We require that an acceptable loose parts monitoring system be installed at Sequoyah before initiation of startup testing after initial fuel load, and that the applicant provide appropriate descriptive information on the selected system for our review.

5.3 Component and Subsystem Design

5.3.1 Steam Generator Tube Integrity

We have evaluated the factors that affect the integrity of the steam generator tubes for Sequoyah Units 1 and 2. We conclude that reasonable measures have been taken to ensure that the tubing will not be subjected to conditions that will cause degradation of integrity. Our conclusion is based on the following:

1. The steam generators are of advanced design with improved secondary water flow characteristics. This will provide more tolerance for occasional lack of control of the secondary water chemistry.

2. All volatile treatment will be used for secondary water chemistry control, thereby minimizing the probability of tube degradation.

3. To further control impurities in the secondary water to very low levels, Sequoyah Units 1 and 2 will use steam generator secondary blowdown and makeup demineralization.

4. The condenser tubing is made of 90-10 copper-nickel, thus minimizing the probability of condenser leakage contributing to contamination of the secondary water.

5. The design and layout of the steam generator allows sufficient access to perform adequate inservice inspection. We require Sequoyah Units 1 and 2 to perform inservice inspection of the steam generator tubes in accordance with Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactors Steam Generator Tubes." See Section 5.2.6 herein for further information on this subject.

5.3.2 Residual Heat Removal System

The residual heat removal system is designed to remove heat from the reactor coolant system after the primary system is cooled down, approximately four hours after shutdown, to 350 degrees Fahrenheit and 425 pounds per square inch gauge. Cooldown to the point at which the residual heat removal system is operable is performed by the steam generators.

The residual heat removal system operates in several modes. These are:

1. Cold shutdown - removing decay heat.

2. Startup - connected to chemical and volume control system, acting as an alternate letdown path to control reactor pressure.

3. Cooldown - removing sensible heat and decay heat from reactor and core.

4. Refueling - used for refilling the refueling canal.

5. Emergency core cooling system - the residual heat removal system is aligned during power operation and hot shutdown for low pressure coolant injection into the reactor coolant system as an integral part of the emergency core cooling system.

The residual heat removal system is capable of removing residual heat from the reactor in accordance with the requirements of General Design Criterion 34, with only one residual heat removal train in operation.

The residual heat removal system has two parallel lines which discharge to the reactor coolant system cold legs. When the residual heat removal system is in operation, heat removal is controlled by regulating primary coolant flow through the residual heat removal exchangers by means of butterfly valves.

The residual heat removal system is housed within a structure that is designed to withstand tornadoes, floods, and seismic phenomena in accordance with General Design Criterion 2.

The system meets the seismic requirements of Regulatory Guide 1.29, "Seismic Design Classification," and the quality standards of 10 CFR Part 50.55a having been designed to meet Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

The residual heat removal system has been designed to withstand pipe whip inside containment as required by General Design Criterion 4. It is protected against piping failures outside of containment in accordance with General Design Criterion 4. See Section 3.6 for further information.

As noted above, the residual heat removal system also serves as the low pressure cooling system in the emergency core cooling system. This function and that of shutdown cooling are mutually exclusive since residual heat removal alignment to emergency core cooling system operation is maintained only while the reactor is in power operation and hot shutdown. A separate residual heat removal system is provided for each unit, thus satisfying General Design Criterion 5.

The residual heat removal system is designed to provide an adequate isolation between the reactor coolant system and residual heat removal when the reactor coolant system is above the design pressure of the residual heat removal system (600 pounds per square inch absolute) as follows:

1. There are two separate and redundant motor-operated isolation valves between the residual heat removal pump suction line and the reactor coolant system. These valves are interlocked with one of the two independent reactor coolant system pressure signals. Valve opening is prevented until the reactor coolant system pressure falls to a value of 425 pound per square inch gauge and already opened valves are closed when the reactor coolant system pressure rises to 600 pounds per square inch gauge.

2. There are two check valves and an open motor operated valve on each residual heat removal discharge line to protect the system from the reactor coolant system pressure during operation. The applicant has provided design features to permit leak testing of each valve separately during plant operation to fulfill the staff requirements for high/low pressure isolation with two check valves.

Overpressure protection of the residual heat removal system is provided by relief valves on the suction line and each of the discharge lines. The suction line valve has a capacity sufficient to discharge the flow from both charging pumps. The valves in the dischage lines have a capacity which protects the system from leakage past the check valves (estimated to be much less than one gallon per minute). These relief valves are adequate to protect the residual heat removal from overpressurization.

The staff has reviewed the description of the residual heat removal system and the piping and instrumentation drawings to determine whether the system can be operated with or without offsite power and assuming a single failure. The two residual heat removal pumps are connected to separate buses which can be powered by separate diesel generators in the event of loss of offsite power.

The staff noted that there is only a single suction line with isolation valves in series. A mechanical valve failure could preclude actuating the residual removal system and an inadvertent closure of one of these isolation valves during residual heat removal operation would result in loss of suction and potential failure of the residual heat removal pumps. The applicant has indicated that prior to start-up following the first refueling outage, an alarm will be provided to indicate loss of flow to the residual heat removal pumps. Until such installation, the applicant has indicated that an operator will be dedicated to monitor this flow whenever the residual heat removal system is in operation. We will review confirmatory documentation, including recovery procedures, and report further on this matter in a supplement to this report. Also, the operating license will be conditioned to require installation of an acceptable flow alarm prior to startup following the first regularly scheduded refueling outage.

The planned preoperational and startup test program provides for demonstrating the operation of residual heat removal system in conformance with Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

The requirement that a nuclear power generating station be able to go to cold shutdown using only safety-grade equipment is discussed in Branch Technical Position RSB 5-1, "Design Requirement of the Residual Heat Removal Systems." The applicant has discussed with us the capability of the system in this regard, indicating that the existing equipment and appropriate operating procedures will satisfy our position, and will provide appropriate confirmatory documentation. We will report further on this matter in a supplement to this report.

Subsequent to resolution of the staff concerns over potential failure of the residual heat removal pump following inadvertent closure of a residual heat removal suction valve and compliance with Branch Technical Position RSB 5-1, we conclude that the residual heat removal design meets all General Design Criteria and Regulatory Guides as discussed above, and is therefore acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.1 Design Considerations

The purpose of the various engineered safety features is to provide a complete and consistent means of assuring that the public will be protected from excessive exposure to radioactive materials should a major accident occur in the facility. The reactor containment systems and the emergency core cooling systems are described in this section. Certain of these systems or parts of these systems will have functions for normal facility operations as well as serving engineered safety features.

We have reviewed the proposed systems and components designated as engineered safety features. These systems and components are designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15.0 of this report. These systems are designed, therefore, to seismic Category I requirements and must function even with complete loss of offsite power.

Components and systems are provided with sufficient redundancy so that a single failure of any component or system will not result in the loss of the capability to achieve and maintain safe shutdown of the reactor. The instrumentation systems and emergency power systems for the engineered safety features are designed to the same seismic and redundance requirements as the systems they serve. These systems are described in Section 7.0 and 8.0 of this report, respectively.

6.1.1 Engineered Safety Features Materials

The mechanical properties of materials selected for the engineered safety features satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, or Parts A, B and C of Section II of the Code, and our position that the yield strength of cold worked stainless steels shall be less than 90,000 pounds per square inch.

The requirements and controls used on welding processes provide reasonable assurance that no deleterious hot cracking would be present during the assembly of austenitic stainless steel components. All weld filler metal was of selected composition to produce welds with at least five percent delta ferrite. Tests and examinations were made in accordance with Section III of the ASME Code to assure that adequate delta ferrite levels were met.

Controls imposed in the application and processing of austenitic stainless steels for engineered safety features components to avoid sensitization satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that stress corrosion cracking will not occur during the postulated accident time interval. The control of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, are in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident," and provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution by corrosion of containment metal or cause serious deterioration of the containment. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the engineered safety features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Conformance with the codes and regulatory guides mentioned above, and with our positions on the allowable maximum yield strength of cold worked austenitic stainless steel and the minimum pH of containment sprays and emergency core cooling water, constitute an acceptable basis for meeting applicable material requirements of General Design Criteria 35, 38, and 41 of Appendix A to 10 CFR Part 50.

6.1.2 Organic Materials Inside Containment

Organic coating systems (paints) used inside the containment may decompose under the condition of a design basis loss-of-coolant accident. Decomposition products consist of combustible gases (such as hydrogen and methane), and water-insoluble residues which may fall into the containment sumps. Radiolytic decomposition of coating systems can be effectively reduced if the paints and the application procedures are qualified according to recommendations of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." The applicant has stated that the design complies with recommendations in this Regulatory Guide. We therefore conclude that the organic coating materials used in the containment have been qualified under conditions up to and including the design basis loss-of-coolant accident and are acceptable.

6.2 Containment Systems

The containment systems for the Sequoyah Nuclear Plant, Units 1 and 2, include dual containment structures, containment heat removal systems, containment isolation systems, containment combustible gas control systems, secondary containment (annulus) emergency gas treatment systems, and the auxilary building gas treatment system. The plant will utilize an ice condenser-type pressure suppression containment similar to the Donald C. Cook Plant.

The design of the Sequoyah Nuclear Plant containment is very similar to the containment design for the McGuire Nuclear Station, which has been previously reviewed by us. Both plants utilize the dual containment concept, with a free-standing steel primary containment within a reinforced concrete shield building. Volumes and plant arrangement within the primary containment are quite similar. Basic differences between McGuire and the Sequoyah Nuclear Plant are slight and are limited to the design of the reactor coolant system, containment spray system, emergency gas treatment system for the annulus and auxiliary building gas treatment system, and containment internal structures. Table 6-1 indicates the differences in principal containment parameters between the two plants. These differences have been evaluated and the results of the evaluation are addressed in the appropriate sections of this report.

The primary reactor containment, which has a net free volume of about 1,192,000 cubic feet, is divided into three major subvolumes, including a 383,000 cubic foot compartment enclosing the reactor system, a 111,000 cubic foot ice condenser compartment enclosing the ice condenser, and a 698,000 cubic foot upper compartment.

The basic performance and design evaluation of the ice condenser system have been the subject of both analyses and experimental programs. These efforts are described in the Staff Evaluation of Tests Conducted to Demonstrate the Functional Adequacy of the Ice Condenser Design, dated April 25, 1974, and provide the basis for our evaluation of the containment functional design.

6.2.1 Containment Functional Design

The containment for each unit of the Sequoyah Nuclear Plant consists of a primary containment vessel and a shield building, and the common auxiliary building. The primary containment vessel is a free-standing, welded steel structure consisting of a vertical cylinder, a hemispherical dome, and a concrete base mat with steel membrane. The shield building is a medium-leakage concrete structure enclosing the containment vessel and is designed to provide for the collection, mixing, holdup, and controlled release of containment vessel fission product leakage following an accident. The interior of the primary containment vessel is divided into three compartments: (a) a lower compartment which houses the reactor and reactor coolant system; (b) the ice condenser compartment housing the energy-absorbing ice bed in which steam is condensed; and (c) the upper compartment which accommodates the air displaced from the other two volumes during postulated loss-of-coolant and steam line break accidents.

The intermediate, or ice condenser compartment, is an enclosed annular compartment encompassing most of the perimeter of the containment structure. Borated flake ice is stored within the ice condenser compartment in 48-foot long cylindrical perforated metal baskets. The ice contained in the baskets is provided to condense the steam released in the event of a loss-of-coolant accident or a steam line break accident.

TABLE 6-1

CONTAINMENT DESIGN PARAMETERS

	McGuire	Sequoyah
Reactor Containment Volumes (net free volume)		
Upper Compartment (cubic feet)	717,000	698,000
Ice Condenser (cubic feet)	111,000	110,500
Lower Compartment (cubic feet)	360,000	383,000
Total Containment Volume (cubic feet)	1,196,000	1,191,500
Reactor Containment Air Compression Ratio	1.41	1.43
Reactor Power (megawatts thermal)	3,579	3,582
Design Energy Release to Containment		
Initial Blowdown Mass Release (pounds)	493,210	543,330
Initial Blowdown Energy Release (British thermal units)	318.4 x 106	334.6×10^6
Ice Condenser Parameters		
Weight of Ice in Condenser (pounds)	2.45×10^6	2.45×10^6
Vent Flow Areas (Lower Compartment)		
Vent Flow Area Past Steam Generators (total) (square feet)	2,724	2,372
Vent Flow Area Past Pressurizer (square feet)	679	632
Vent Flow Area Past Ice Condenser Lattice Frames (square feet)	1,344	1,344
Vent Flow Area Through Lower Inlet Doors (square feet)	1,064	1,064
Containment Spray Flow (loss-of-coolant accident analysis)		
One Spray Train Inoperable		
Upper Compartment (gallons per minute)	3,432	4,750
Lower Compartment (gallons per minute)	0	0
One RHR Pump Inoperable		
Upper Compartment (gallons per minute)	1,623	2,000
Lower Compartment (gallons per minute)	0	0
Total Spray (gallons per minute)	5,055	6,750

We will require the applicant to weigh the ice in a large statistical sample (approximately 50 percent) of the 1944 ice baskets in each unit, following their initial ice loading. We will require that this information be used in statistical analyses to determine; (1) the initial distribution of ice in the ice condenser, (2) the minimum amount of ice loaded into the ice condenser at a 95 percent level of confidence, and (3) appropriate subdivision of the ice condenser into groups of bays to be utilized in the periodic ice weight surveillance program.

In an effort to provide the earliest possible indication of the actual sublimation rate for the D.C. Cook, Unit 1 ice condenser, the American Electric Power Company implemented a program to periodically measure the weight of selected ice baskets in the ice condenser, and has weighed a sample of ice baskets on numerous occasions. The results of the ice basket weighing program have indicated that the average sublimation rate of two to three percent per year is significantly greater than the expected rate of about 0.5 percent per year, and slightly greater than the maximum design sublimation rate of two percent per year. The results have also shown that ice sublimation does not occur uniformly over the cross-sectional area of the ice condenser. Baskets adjacent to the crane and containment wall cooling ducts lose ice at a greater rate than baskets located in the interior of the ice condenser. Interpretation of the ice basket weighing program data has been complicated by the fact that variations in original ice loading techniques resulted in three distinct groups of ice baskets within the ice condenser having significantly different mean basket weights. The frequent weighing program conducted at the D.C. Cook, Unit 1 has provided early identification of the ice condenser loss rates and patterns, the opportunity to develop corrective modifications and procedures, and has assured the safety of continued operation of the plant.

Based on the above discussion of current ice condenser operating experience, we require TVA to institute a periodic ice basket weighing program for each unit at the Sequoyah Nuclear Plant similar to the program being conducted at the D.C. Cook Plants. We have recommended that the applicant continue to evaluate the equipment and techniques available for ice loading in order to achieve an initial ice inventory which is uniformly distributed. We will pursue the development of a suitable periodic ice weighing program with the applicant during the development of technical specifications for the operation of the plant and will include appropriate operating limits to assure an acceptable margin of safety.

During normal plant operation, the ice bed is maintained at about 15 degrees Fahrenheit by a redundant refrigeration system. Refrigeration ducts and insulation on the ice condenser walls serve to minimize heat losses from the ice. Thirty chiller units are provided in the containment but only 21 of the units are required to operate at any time to maintain the design temperature of 15 degrees Fahrenheit within the ice bed. In the unlikely event that a complete loss of the refrigeration system occurs, the insulation within the ice condenser is sufficient to prevent the ice from melting for a minimum period of seven days which allows adequate time for safe plant shutdown.

Inlet and outlet doors are provided at the top and bottom of the ice condenser compartment. In the event of a loss-of-coolant accident, the lower inlet doors will open due to the pressure rise in the lower compartment caused by the release of the reactor coolant to the lower compartment. The differential pressure will then cause air, entrained water, and steam to flow from the lower compartment into the ice condenser. The resulting pressure rise, due principally to the air mass in the ice condenser, will cause the doors at the top of the ice condenser to open and allow the air to flow from the ice condenser into the upper compartment. Steam will be condensed as it contacts the ice contained in the baskets in the ice condenser compartment and therefore does not appear in the upper compartment. Complete steam condensation is assured because of the ice mass and geometrical arrangements of the ice columns. Developmental testing by the Westinghouse Electric Corporation has confirmed this phenomenon. Our evaluation of the test programs was completed in conjunction with our review of the Donald C. Cook Nuclear Plant, and was reported in our Safety Evaluation Report for that plant and in the report "Staff Evaluation of Tests Conducted to Demonstrate the Functional Adequacy of the Ice Condenser Design," dated April 25, 1974.

An operating deck separates the upper and lower compartment and ensures that steam and air flow resulting from a loss-of-coolant accident is directed through the ice condenser to the upper compartment rather than through uncontrolled bypass paths. Following initial blowdown, 1,790,000 pounds of ice (or 73 percent of the initial mass of ice) remains in the ice condenser. Condensation of the steam in the ice limits the containment pressure to approximately 8.0 psig between the time reactor blowdown is complete and the time that meltout of the ice bed occurs. Ice meltout is predicted to occur about 66 minutes after a design basis loss-of-coolant accident Following ice meltout, the rise in the containment pressure due to the release of decay energy from the core is limited by the containment spray system. Figure 6-1 illustrates the containment pressure response as a function of time following a design basis loss-of-coolant accident.

The lower compartment is divided into a number of subcompartments formed by internal equipment, structures, and components. The pressure responses within these subcompartments were analyzed by the applicant using the TMD (Transient Mass Distribution) computer code developed by Westinghouse Electric Corporation. The code is described in a non-proprietary topical report, "Ice Condenser Containment Pressure Transient Analysis Methods", WCAP 8078. The code provides a means for computing pressure, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. We have reviewed the mathematical description of this code during our review of the Donald C. Cook plant and have found it to be acceptable for calculating the short-term pressure response in subcompartments. The pressure response within the subcompartments is different from the overall pressure response of the containment only during the early blowdown phase of the accident; i.e., up to about 10 seconds following the occurrence of the break.

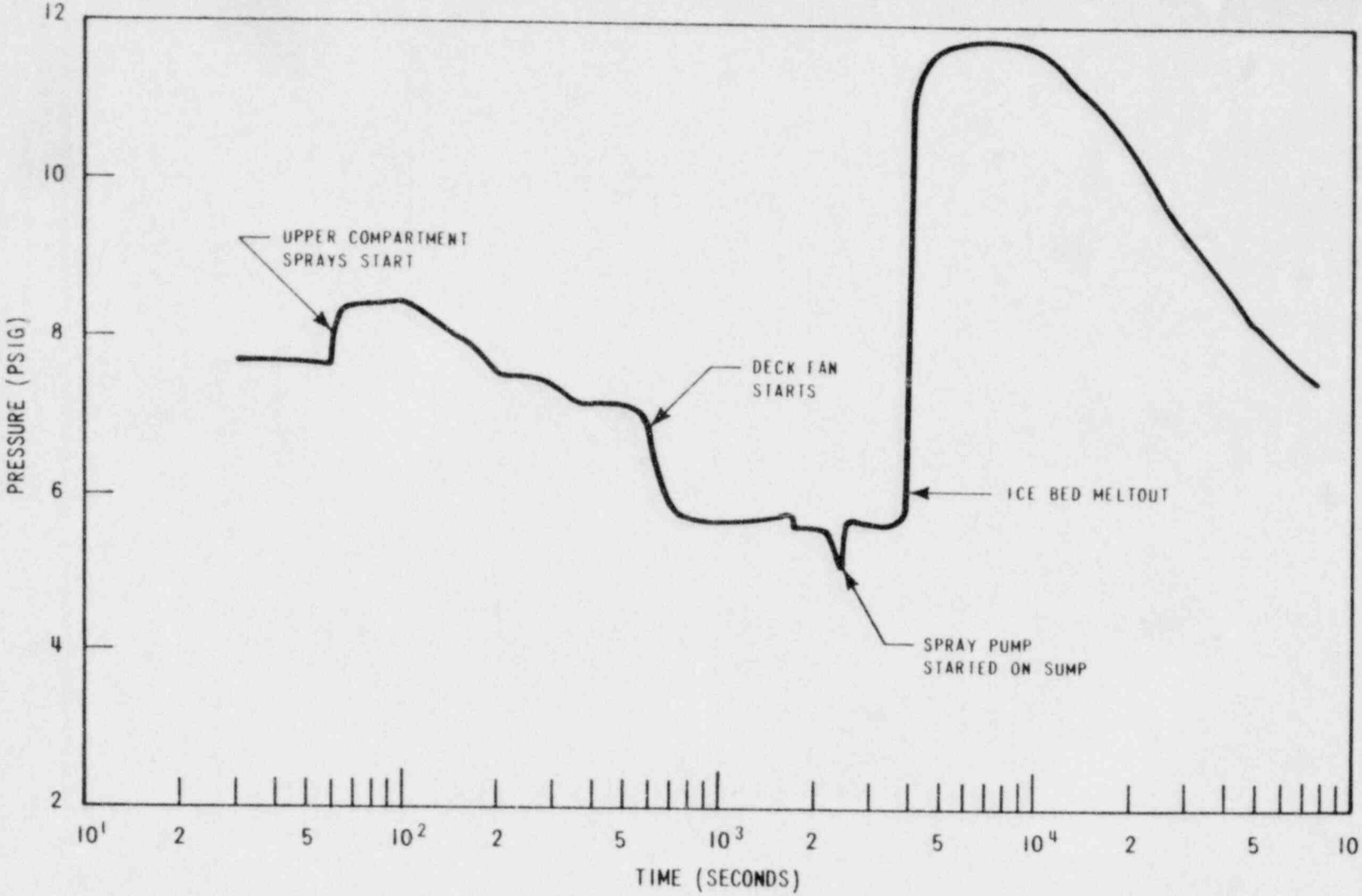

Figure 6-1 Containment Pressure - Double Ended Pump Suction Break

Following the early blowdown phase of the accident, the pressure and temperature responses of the upper and lower compartments are analyzed with the Westinghouse Electric Corporation LOTIC-1 computer program. This program has been described in Westinghouse Topical Reports WCAP-8354 "Long Term Ice Condenser Containment Code - LOTIC Code." We have completed a generic review of the LOTIC-1 computer program through the our topical report evaluation program and have concluded that the LOTIC-1 code is acceptable for the calculation of the long term ice condenser containment response to postulated loss-of-coolant accidents (see our letter to Westinghouse dated May 3, 1978).

Containment Short-term Pressure Response

Following a postulated reactor coolant pipe rupture, differential and local pressures build up in the subcompartments of the lower containment compartment as high-energy fluid is released and transported throughout the various regions. The pressure magnitudes depend upon the volumes of the subcompartments, interconnecting vent flow paths, mass flow behavior and the thermodynamic behavior within the pressure nodes. During this phase of the transient, flow to the upper containment compartment is not significant, and the upper containment compartment pressure is still near its initial pressure. It is during this time that the peak operating deck differential pressure and peak subcompartment differential pressures would be experienced. As the blowdown continues, the pressure in the upper compartment rises, and about 10 seconds after the start of blowdown the upper compartment reaches a peak pressure approximately equal to the lower compartment pressure; i.e., about 8.0 pounds per square inch gauge. The primary factor in producing this upper compartment pressure peak is the displacement of air from the lower compartment through the ice columns into the upper compartment.

Westinghouse uses the SATAN-V computer code to determine the mass and energy addition rates to the containment during the blowdown phase of the accident. The SATAN-V computer code has been accepted for the calculation of blowdown mass and energy release rate during a loss-of-coolant accident as stated in our letter to Westinghouse dated March 12, 1975. We have found that the SATAN-V code is reasonably conservative; e.g., the applicant has increased the energy release rate to the containment during the blowdown phase by extending the time that the core would remain in nucleate boiling so that the energy release rate from the core is maximized. Due to this assumption, the core would transfer more heat to the containment than would be calculated in an analysis suitable for emergency core cooling performance evaluation. This additional energy release from the core will increase the calculated containment pressure and therefore assures a margin of conservatism in the calculation.

The applicant used the TMD computer program to calculate the short-term pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment, including the containment compartments, following either a loss-of-coolant or a main steam line break. The model includes a nodalization scheme of 50 elements representing the containment to analyze the pressure response of each of the subcompartments within the lower compartment, including dead-ended compartments, the ice condenser compartment, the upper compartment and the steam generator enclosures.

TMD was developed specifically to analyze the short-term pressure response of the ice condenser system. The mathematical modeling in TMD is similar to that of the SATAN-V blowdown code in that the analytical solution is developed by considering the conservation equations of mass, momentum and energy, and the equation of state, and uses the control volume technique for simulating spatial variation. The governing equations for TMD are somewhat different from those in SATAN-V in that a two-phase (liquid water droplets and steam-air vapor), two-component (air-water) system is considered. We have reviewed these mathematical differences between SATAN-V and TMD and concur that TMD has maintained the conservatisms incorporated in SATAN-V. The TMD calculates the critical flow of a two-component, two-phhase fluid (air, steam, and water) assuming a thermal equilibrium condition. However, a correction factor, which was determined by Westinghouse to account for experimental data on applicable flow regimes, is then applied to the calculated critical flow. The correction factor as used in the code increases the critical flow up to 20 percent through the compartments as the quality of the fluid decreases. The applicant refers to this increased critical flow as "augmented" flow. The net effect results in a lower compartment differential pressure when compared to a nonaugmented flow regime. The use of the augmented flow factor results in less conservatism than use of the thermal equilibrium correlation.

Following our review of the experimental data and analysis performed during our review of the Donald C. Cook Nuclear Plant, we informed TVA that we could not justify the use of this correlation factor for the type of analysis being performed. In response to our request, TVA repeated the short-term containment pressure response using the latest version of the TMD code with a non-augmented or unity flow correlation.

The applicant's heat transfer model of the ice condenser used in the TMD code was based on the results of full-scale testing done by Westinghouse during 1968 and 1969. The test program used in an ice basket design which had the equivalent of an 81 percent basket open surface area (i.e., exposed ice for condensing). Structural problems with the initial basket design required a change in basket design to accommodate the postulated loads. The redesign resulted in a decrease in exposed surface area from the previous design. In 1973, Westinghouse reactivated the ice condenser full-scale test facility to provide final verification of the ice condenser functional performance with the redesigned ice basket and final designs of all the ice condenser internal structures. Results of the staff review of the 1973-1974 full-scale ice condenser tests have been presented in our April 1974 report, "Staff Evaluation of Tests Conducted to Demonstrate the Functional

Adequacy of the Ice Condenser Design." The test program resulted in a further redesign of the basket to achieve acceptable heat transfer between steam and the ice in the baskets. A new heat transfer correlation was derived for the TMD code to conservatively predict the ice condenser performance with the new basket design. The latest version of the TMD code which was used for the reanalysis of the short-term containment transient uses the new heat transfer correlation and a compressibility factor which is used with the subsonic incompressible flow equations to include the effects of compressible fluid flow.

Table 6-2 presents a comparison of the maximum calculated differential pressures from the short-term ice condenser transient analysis to the design for the operating deck, ice condenser crane wall, and the portion of the containment vessel wall which forms the back wall of the ice condenser inlet plenum.

We have found the applicant's method of analysis and containment modeling to be acceptable. We therefore conclude that the calculated short-term pressure transients are acceptable for the structural evaluation of the containment interior compartments, except the steam generator and pressurizer enclosures as discussed below.

The applicant has also used the TMD code to analyze the response of the steam generator enclosures to a double-ended steam line rupture and the pressurizer enclosure to a double-ended rupture of the pressurizer spray line. The applicant has performed nodalization sensitivity studies and has developed a 10-node model for the steam generator enclosure and a four-node model for the pressurizer enclosure. These models are similar to the models developed for the analyses of steam generator and pressurizer enclosures at the D. C. Cook, Unit 2 and McGuire, Units 1 & 2 nuclear plants. However, the information provided by the applicant regarding the analyses and results does not confirm that the pressure response of these subcompartments has been utilized in evaluating the adequacy of the design of the steam generator and pressurizer supports. We have therefore requested additional information to provide such confirmation, and we will report further on this matter in a supplement to this Report.

The applicant used the TMD code without the augmented critical flow correlation to analyze the reactor cavity response to a loss-of-coolant accident. The reactor cavity was modeled by 50 nodes within the cavity structures and 11 nodes external to the structures. The annulus between the reactor vessel and the shield wall was broken into axial and circumferential nodes. The applicant has calculated the reactor cavity response to an assumed 100 square inches break of the reactor cold leg pipe at the pressure vessel nozzle to pipe weld. The maximum credible break sizes, considering the dynamic pressure calculated for the assumed 100 square inch break, the effects of piping restraints in the penetrations, and reactor vessel movement, are 58 square inches for a cold leg break and 32 square inches for a hot leg break. We find the applicant's method of analysis, modeling assumptions, and

TABLE 6-2

DATA ON INTERNAL COMPARTMENT PRESSURES

OPERATING DECK*

Control Volumes	1	2	3	4	5	6
Maximum Calculated Δ Pressure, pounds per square inch	14.9	11.2	8.6	8.4	11.4	14.7
Minimum Design Pressure, pounds per square inch	20.0	15.0	11.8	11.6	15.6	19.6
Margin, percent	34	34	37	38	37	33

CONTAINMENT WALL IN ICE CONDENSER INLET PLENUM

Control Volumes	1	2	3	4	5	6
Maximum Calculated Δ Pressure, pounds per square inch	11.7	9.8	9.0	9.0	9.6	11.4
Minimum Design Pressure, pounds per square inch	16.4	13.0	12.5	12.6	13.6	16.0
Margin, percent	40	33	39	40	42	40

ICE CONDENSER CRANE WALL IN UPPER COMPARTMENT*

Control Volumes	1	2	3	4	5	6
Maximum Calculated Δ Pressure, pounds per square inch	8.2	6.9	6.2	6.2	7.0	8.2
Minimum Design Pressure, pounds per square inch	11.3	9.5	8.7	8.7	9.5	11.5
Margin, percent	38	38	40	40	36	40

* See Figures 6.2.9 through 6.2.13 of the Final Safety Analysis Report for reference to plant arrangement.

results acceptable for the evaluation of both the reactor cavity structures and the reactor vessel supports.

Containment Long-Term Pressure Response

As seen in Figure 6-1, the containment spray system is activated after the completion of blowdown; i.e., about 30 seconds after accident initiation, and causes a slight reduction in the containment pressure. After about 10 minutes, the return air fans are started and the containment pressure is reduced to approximately 5.5 pounds per square inch gauge as air is returned from the upper volume to the lower volumes. Steam from the reactor coolant system is still being removed almost entirely by the stored ice at this time. After ice meltout, which occurs about 66 minutes after the accident, steam from the reactor coolant system is removed by the containment spray system. The containment pressure will again peak about two hours after the accident, at which time the energy input equals the minimum heat removal capability of the sprays. The magnitude of this peak pressure is determined by the heat input rate to the containment and heat removal rate of the containment spray system.

The applicant used the LOTIC-1 computer program to calculate the long term containment pressure response. LOTIC-1 is a computer program similar to COCO which has been used to analyze the containment pressure transients for other types of containments. The main differences between these computer codes lies in the methods by which the heat removal systems are modeled. LOTIC-1 includes features for modeling the heat removal capabilities of the ice and has provisions to calculate the pressure response of the containment. The containment upper and lower compartments and the ice condenser are modeled as control volumes in the code to represent the physical geometry of the containment. Conservation of mass and energy are applied and equations are solved by appropriate numerical procedures.

We have reviewed the LOTIC-1 computer program and have accepted the code for the calculation of long term ice condenser containment response to loss-of-coolant accidents. Using the LOTIC-1 code, the applicant has calculated a peak containment pressure of 11.8 pounds per square inch gauge compared to a containment design pressure of 12.0 pounds per square inch gauge.

The mass and energy release rates to the containment were calculated by the applicant during the reflood phase of the accident following blowdown, using the computer code REFLOOD. Proper analysis of the reflood phase of the event is important since it models the manner in which additional energy is removed from the secondary system during core refill. This is particularly true with regard to pipe ruptures in the reactor coolant system pump suction cold leg because the steam and entrained liquid carried out of the core for these break locations passes through the steam generators, which become an additional energy source. The water leaving the core and passing through the steam generators is assumed to be superheated to the temperature of the steam generator secondary fluid. Results of the FLECHT experiments indicate that the carryout fraction of fluid leaving the core during reflood is about 80 percent of the incoming flow to the core. Therefore, the rate and amount of energy release to the containment during this phase becomes proportional to the reflood flow into the core. We have found that the rupture of the cold leg at the pump suction results in the highest mass flow through the core, and thus through the steam generators, because of the low resistance flow paths between the steam generators and the broken pipe. Therefore, such a break location leads to calculation of the highest containment pressure. To determine the mass and energy release to the containment during the reflood phase of the accident, we have compared the results using our FLOOD code with those predicted by the applicant using the Westinghouse REFLOOD computer program. The results of this comparison indicate essentially equivalent predictions of energy release.

The applicant uses the Westinghouse FROTH code to calculate the mass and energy release to the containment following the reflood phase of the accident. The applicant assumes that, following reflood of the core, a two-phase mixture of steam and water is displaced from the core by the cooler water in the downcomer such that the two-phase mixture reaches the steam generator tubes. The mixture is assumed to enter all four steam generators, until the entire inventory of additional energy available in the steam generators is transferred through the tubes, boiling the liquid of the two-phase mixture. The applicant assumes that the steam exiting the steam generators is mixed with the emergency core cooling system injection water in the intact loop cold legs because the steam in the loops must flow through the points of safety injection in the cold legs of the broken and unbroken loops before reaching the containment. The amounts of steam that would be quenched in the intact loops by the available safety injection flow before reaching the containment, is determined from an energy balance between the injection water and steam flow. We have reviewed the applicant's analysis and supporting data and conclude that the calculation of mass and energy release to the containment for the period from the end of reflood until all steam generator sensible energy has been removed is conservative.

Following the removal of all steam generator sensible energy, the applicant has assumed that all residual heat from the reactor is released to the containment through the broken loop as saturated steam (i.e., no flow split, with no reduction of mass and energy release to the containment by quenching in the cold leg injection points). The mass and energy calculations discussed above are described in the Westinghouse Electric Corporation topical reports WCAP-8264-P-A, "Westinghouse Mass and Energy Release Data for Containment Design." This topical report has been reviewed under our topical report evaluation program and has been accepted by the staff for use in licensing applications in our letter to Westinghouse March .2, 1975. We have also reviewed the plant containment parameters and have found them to be conservative for the evaluation of the long term containment response to postulated loss-of-coolant accidents. Using the above mass and energy release rate data, containment input parameters and the LOTIC-1 ice condenser containment analysis program, the applicant has identified the double-ended rupture of the reactor coolant system pump suction cold leg as the controlling reactor coolant system pipe break accident. Since the peak calculated containment pressure of 11.8 pounds per square inch gauge is less than the 12.0 pounds per square inch gauge containment design pressure, we find the applicant's long term containment response calculations for loss-of-coolant accidents acceptable.

The applicant has also analyzed the effect of steam bypassing the ice condenser on the containment pressure response. Drain lines in the floor of the refueling canal are provided to allow water sprayed into the upper compartment to return to the containment sump. These drains represent a bypass path. The applicant has included in the containment analysis the effect of this bypass area (2.2 square feet). The applicant has also provided analyses which indicate that about 40 square feet of bypass area can be accommodated in the design without the design pressure of the containment being exceeded.

The containment vessel is designed for an external pressure of 0.5 pounds per square inch gauge. Three 24-inch vacuum relief valves are provided. Inadvertent operation of the spray system and/or return air fan systems would cause a reduction in the containment pressure. The applicant has therefore conservatively analyzed the effect that inadvertent operation of the containment spray systems, return air fan systems, or simultaneous operation of both systems would have on the containment response. The applicant's analysis shows that operation of two of the three vacuum relief valves will prevent the inadvertent operation of containment sprays and/or return air fans from exceeding the containment external design pressure of 0.5 pounds per square inch gauge.

The applicant has calculated the containment response to a postulated double-ended circumferential steam line break using the LOTIC-3 computer program. This program has been described in Supplement 2 to the Westinghouse Electric Corporation topical report WCAP-8354, "Long Term Ice Condenser Containment Code - LOTIC Code." We have completed a generic review of the LOTIC-3 code and have concluded that the LOTIC-3 code is acceptable for the calculation of long term ice condenser containment response to postulated secondary system pipe break accidents (see our letter to Westinghouse dated May 3, 1978). The applicant has also presented information to show that the calculated temperature transient inside the Sequoyah containment following a small postulated main steam line break accident is conservatively predicted by the analyses presented in Supplement 2 to WCAP-8354. These analyses were performed for a "generic" ice condenser plant using the LOTIC-3 computer code to demonstrate the adequacy of the code for ice condenser long term transient analyses for secondary system ruptures. While we have accepted Supplement 2 to WCAP-8354 and approved the LOTIC-3 code, we do not believe that a sufficient spectrum of small split breaks was analyzed in the topical report to permit us to conclude that the most severe temperature transient for the "generic" ice condenser plants has been determined. Westinghouse has indicated that temperature response for the small break analyzed in WCAP-8354 will bound the expected temperature responses for the spectrum of small breaks for which we have requested the applicant to provide results. We will report further on this matter in a supplement to this Report.

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the effect of operation of all the containment installed pressure reducing systems and processes be included in the emergency core cooling system evaluation. For the purpose of this evaluation, it is conservative to minimize the containment pressure. The reflood rate in the core will then be reduced because of the resistance to steam flow in the reactor coolant loops.

Following a loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the primary reactor system to the containment atmosphere. After initial blowdown, heat transfer from the core, primary metal structure, and steam generators to the emergency core cooling water will produce additional steam. This steam, together with any emergency core cooling water spilled from the primary system, will flow through the postulated break into the containment. This energy will be released to the containment during both the blowdown and later operational phases, i.e., the reflood and post-reflood phases.

Energy removal occurs within the containment by several means. Steam condensation on the containment walls and on internal structures serves as a passive energy heat sink that becomes effective early in the blowdown transient. Subsequently, the operation of the containment heat removal systems such as containment sprays will remove steam from the containment atmosphere. In an ice-condenser-type containment, energy is removed as the mixture of steam, air, and water passes through the ice condenser; i.e., when the mixture is forced from the containment lower compartment to the upper compartment.

The emergency core cooling system containment pressure calculations for the Sequoyah Nuclear Plant were done using the Westinghouse emergency core cooling system evaluation model. The containment response calculations were performed using the Westinghouse Electric Corporation's LOTIC-2 containment code. We have reviewed the LOTIC-2 code and have concluded that the LOTIC-2 code is acceptable for the calculation of minimum containment pressure response for ice condenser plants. Although we have accepted the methods used to calculate containment pressure response, we require that justification of the plant dependent input parameters used in the analysis of containment pressure response be submitted for our review on a plant-by-plant basis. This information was submitted in Amendments 50 and 57 to the Sequoyah Nuclear Plant Final Safety Analysis Report. The applicant has reevaluated the containment net-free volume, the passive heat sinks, operation of the containment heat removal systems and containment initial conditions with regard to the conservatism for the emergency core cooling system analysis. The containment heat removal systems were assumed to operate at their maximum capacities, and minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant dependent information used for the emergency core cooling system containment pressure analysis submitted in Amendment 50 and 57 to the Final Safety Analysis Report for the Sequoyah Nuclear Plant is reasonably conservative; and therefore that the containment pressures calculated in Amendment 50 and 57 to the Final Safety Analysis Report are in accordance with the applicable provisions of Appendix K to 10 CFR Part 50 of the Commission's regulations.

We have reviewed the applicant's analysis of the maximum differential pressures which could exist in the reverse direction (i.e., upper compartment to lower compartment) during a loss-of-coolant accident. The applicant's methods of analysis and assumptions are conservative. The applicant has calculated a maximum reverse differential pressure of 1.2 pounds per square inch gauge. The design reverse differential pressures of the operating deck and ice condenser lower inlet doors are 6.8 and 8.6 pounds per square inch gauge respectively.

Summary and Conclusions

We have evaluated the containment capability with respect to General Design Criteria 16 and 50 of, Appendix A, 10 CFR Part 50. We have found the applicant's analyses of the dynamic pressure loads which would act upon the containment vessel and some of its internal structures (namely the reactor cavity structures, operating deck, crane wall and loop compartment structures) to be acceptable. We have also found the minimum containment pressure response calculations for the emergency core cooling system performance evaluation to be acceptable. We require further information from the applicant concerning the maximum calculated dynamic loads for the steam generator and pressurizer enclosures, and the maximum containment temperature and pressure response for a postulated main steam line break.

6.2.2 Containment Heat Removal System

The energy released to the containment following a design basis loss-of-coolant accident is absorbed by the ice condenser. However, after the ice bed has melted, mass and energy will continue to be released to the containment. The containment spray systems are designed to maintain the containment pressure in the long term below the containment design pressure, and eventually reduce the containment pressure to about atmospheric pressure.

The containment spray for the Sequoyah Nuclear Plant is provided by two redundant spray systems, each designed to provide the cooling capacity required to maintain the peak pressure at less than design pressure for the full spectrum of break sizes. Each of the redundant spray systems delivers 4750 gallons per minute to the containment from one containment spray pump and heat exchanger and 2000 gallons per minute from one residual heat removal pump and heat exchanger. The containment spray pump is started by a containment pressure signal set at three pounds per square inch gauge, and containment spray starts at about 30 seconds after the accident. Containment spray from the residual heat removal pump is manually initiated one hour after the accident.

The containment is equipped with redundant return air fan systems. Each of the two 100-percent capacity return air fan systems uses a 4000 cubic feet per minute fan to force air from the upper compartment back to the lower compartment after the reactor coolant system blowdown and subsequent reactor reflooding are completed.

The return air fans are utilized to return air from the upper compartment to the lower compartment after the compression peak is reached and thus provide a homogeneous mixture of steam and air throughout the containment during the long-term pressure peak. The return air fans are also started by the containment pressure signal, but the fan startup is delayed for 10 minutes to provide an increased backpressure during the core reflood.

The applicant has provided a malfunction analysis and other information which demonstrates independence of the redundant spray trains and return air fan systems. Each spray train has its own recirculation piping suction inlet from a common sump. The sump is constructed with a grating in the inlet to prevent debris from passing. The spray nozzles are the limiting component in the containment spray systems and are not subject to clogging by particles less than one-fourth inch. Spray droplet size at design conditions, spray pattern, and header locations are acceptable. The applicant has provided and we have reviewed analyses to assure that adequate net positive suction head is available at both containment spray pump and residual heat removal pump inlets during flow from both the refueling water storage tank and the containment sump without taking credit for increased containment pressure following a loss-of-coolant accident, as required by Regulatory Guide 1.1.

The applicant used the LOTIC-1 code to demonstrate the long term capability of minimum containment heat removal systems (one complete train of spray and return air systems) to maintain the containment pressure below design pressure for the design basis loss-of-coolant accident. We have reviewed the applicant's containment pressure and temperature response as calculated by the LOTIC-1 code and conclude that the design of the containment heat removal systems is acceptable.

Provisions are made in the containment spray system and the return air system to permit in-service inspection of the system component and functional testing of active components in both systems.

We therefore conclude that the design of the containment heat removal system conforms to General Design Criteria 38, 39, 40, and 50 and Regulatory Guides 1.26, 1.29 and 1.82, and is acceptable.

6.2.3 Containment Air Purification and Cleanup Systems

Emergency Gas Treatment Systems

The containment of each unit of the Sequoyah Nuclear Plant consists of a primary containment structure and a shield building that encloses the primary containment of each unit. An emergency gas treatment system is provided for the annulus formed by these structures. The emergency gas treatment system is comprised of two 100-percent redundant trains and is common to both units. The system collects and filters radioactive airborne fission products that may leak from the primary containment during normal operation and following a loss-of-coolant accident.

The emergency gas treatment system consists of two redundant 100-percent capacity fan filter trains. Each train consists of the following components, designed to Quality Group C and seismic Category I requirements: demister, high efficiency particulate air filters, charcoal filter, ducting, valves, fan (4000 cubic foot per minute), and instrumentation and controls. In addition to the post-loss-of-coolant accident air fan filtration trains described above there are two 100-percent capacity (1000 cubic foot per minute)annulus vacuum control trains for each unit which are used to maintain the annulus space of each unit at a vacuum of five inches water gauge during normal plant operation.

In the event of a loss-of-coolant accident, annulus vacuum control system operation is terminated, and the system is isolated from the annulus of the affected unit. Simultaneously, the annulus space of the affected unit is communicated with the emergency gas treatment system. These actions are accomplished either in response to Phase A containment isolation signal or manually. The emergency gas treatment system is automatically aligned to exhaust up to 4000 cubic foot per minute in response to a static pressure controller set at a negative pressure of 0.5 inch water gauge. The applicant assumes a 30-second time delay before the emergency gas treatment system fan reaches full speed and flow. After attaining full fan speed, the emergency gas treatment system operates continuously to the end of the accident.

Auxiliary Building Gas Treatment System

The containment systems of Sequoyah Nuclear Plant also include the auxiliary building gas treatment system. The auxiliary building gas treatment system is used to maintain portions of the auxiliary building which contain emergency safeguards systems and fuel handling systems at a negative pressure of 0.25 inch of water following a loss-of-coolant accident. Exhaust from the auxiliary building gas treatment system is filtered prior to release to the atmosphere.

The auxiliary building gas treatment system is a redundant (two trains) air cleanup and exhaust network provided to reduce the radioactive nuclide release from the auxiliary building secondary containment enclosure during accidents. The rated capacity of each redundant air cleanup unit is 9000 cubic feet minute. The system includes a pre-filter, high efficiency particulate air filters, carbon adsorbers, and fans, designed to Quality Group C and seismic Category I requirements. We had also required the installation of heaters on the upstream side of the filters to reduce the humidity of the incoming air stream.

During normal operations the auxiliary building is maintained at a negative pressure of 0.25 inch water gauge by redundant non-safety grade exhaust systems. Operation of the auxiliary building gas treatment system begins automatically upon receipt of a Phase A containment isolation signal from either unit. The auxiliary building gas treatment system may also be started automatically by high radiation signals from either the fuel handling area radiation monitors or the auxiliary building exhaust vent radiation monitors.

The applicant's transient analysis of the annulus response to a loss-of-coolant accident demonstrates that the annulus will not exceed a 0.25 inch water gauge negative pressure throughout the transient with a minimum exhaust flow of 3,600 cubic feet per minute, compared to the 4,000 cubic feet per minute capacity of a single emergency gas treatment system fan.

Reactor Building Purge Ventilation System

The function of the reactor building purge ventilation system is to assure that activity released inside containment from a refueling accident or fuel-handling accident is treated prior to discharge to the environment. The engineered safety feature portions of the reactor building purge ventilation system are redundant. Each train has a design capacity of 14,000 cubic feet per minute of air and includes the following components: pre-filter, high efficiency particulate air filter, carbon adsorber, and fan. The equipment and components are designed to Quality Group C and seismic Category I and are located in a seismic Category I structure.

Conclusions

Based on the above, we conclude that the annulus emergency system, the auxiliary building gas treatment system, and the reactor building purge ventilation system are designed to meet General Design Criteria 41, 42, and 43 and are acceptable relative to these criteria.

We have also evaluated these cleanup systems with respect to the positions stated in Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Rev. 1)." We find that the the emergency gas treatment system, the auxiliary building gas treatment system, and the reactor building purge ventilation system are capable of controlling the release of radioactive materials in gaseous effluents after a postulated design basis acr ent in accordance with these positions, and are acceptable.

Ice Condenser

The ice condenser is designed to remove iodine from the post-accident atmosphere passing through the ice beds. Sodium tetraborate will be added to the ice makeup solution to enhance the iodine adsorption characteristics of the ice. Technical specifications will require a minimum ice pH of 9.0 whenever the reactor is critical.

The ice condenser iodine removal effectiveness is a function of the flow rate through the alkaline ice beds and the mole-fractions of air and steam in the flow. Based on the expected conditions following a postulated loss-of-coolant accident, the ice condenser iodine removal effectiveness is expected to be high from the initiation of the accident until meltout of the ice beds has occurred. However, it is difficult to establish assured minimum values for the flow rate and mole-fraction of air prior to startup of the recirculation fans. Therefore, in our model of ice condenser effectiveness, we have assumed that the alkaline ice will remove iodine from the 40,000 cubic feet per minute flow through the ice beds established by the recirculation fans, commencing at fan startup at 10 minutes and ceasing at the earliest ice bed melt out at 60 minutes after the design basis loss-of-coolant accident. A minimum value of 30-percent efficiency for the removal of the elemental form of iodine was assumed during this period.

6.2.4 Containment Isolation Systems

There are at least two barriers between the atmosphere outside the containment and the reactor coolant system or the containment atmosphere. No manual operation is required for immediate isolation of the containment. Automatic isolation valves are provided in those lines which must be isolated immediately following an accident. Lines which must remain in service following an accident for safety reasons are provided with at least one remote manual valve. Each automatic trip valve is provided with a manual switch and its positions displayed in the main control room. All air-operated isolation valves assume the position of greater safety upon loss of air or control power. Isolation valves inside the containment are located between the crane wall and the containment wall.

Isolation valves outside the containment are protected by missile shields. The containment isolation systems have been designed to the ASME code, Section III, Class 2, and have been classified as seismic Category I systems.

We have also reviewed the containment isolation signals and the closure times for the isolation valves, particularly the containment purge system isolation valves. Containment isolation will automatically occur upon receipt of safety injection and high containment pressure signals. Valve closure will occur within 60 seconds with most valves closing in 10 seconds or less; the containment purge system isolation valves are designed to close in 4 seconds. We conclude that the containment isolation signals provide acceptable diversity and that the valve closure times are also acceptable.

The applicant has addressed the recommendations of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." As recommended in the Branch Technical Position, the applicant has provided systems within the reactor containment building to control the temperature in the containment and filter the containment air to reduce the airborne activity in the containment. Thus the frequency and duration of containment purging to permit personnel access will be reduced.

The purge system's containment isolation valves are closed by the containment phase A isolation signal which is generated by either a safety injection signal or an internal containment pressure of 1.2 pounds per square inch gauge. The purge system's isolation valves are also closed by high airborne activity levels. We therefore conclude that there is acceptable diversity in the parameters sensed to initiate valve closure.

Purge systems are provided for the containment upper compartment, lower compartment, and instrument room. We have performed a dose consequence analysis for an assumed loss-of-coolant accident while the containment is being purged. In performing the analysis we have assumed the largest pair of lines (a 24-inch inlet line and a 24-inch outlet line) to be open. A pre-existing iodine spike in the reactor coolant system fluid and four-second valve closure times were also assumed. The results of our analysis show that in the event of a loss-of-coolant accident during purge operation, site boundary doses would not exceed the dose guidelines of 10 CFR Part 100 (see Section 15.4.1). The applicant has provided debris screens inboard of the inside containment isolation valves. These design provisions will assure that the purge system containment isolation valves will not be prevented from closing by debris following a loss-of-coolant accident. The applicant has stated that the purge system containment isolation valves meet the Sequoyah operability program for loads which would be experienced following a loss-of-coolant accident. Additional information has been provided on containment conditions which would exist at the time of valve closing (pressure of 8 pounds per square inch gauge) and on the design of the valve which assures the ability of the valve to close against flows induced by the containment pressure response. Based on the above, we find that the operability of the valves under accident conditions has been demonstrated. We therefore conclude that the containment purge system may be used as frequently as necessary during the normal plant operating modes of startup, power, hot standby and hot shutdown, but in a manner consistent with the above dose consequence analysis; i.e., with only one pair of purge system lines open at a time. In the cold shutdown and refueling modes all purge systems may be used simultaneously. The technical specifications will reflect this requirement.

The applicant has provided the necessary test connections to permit leak testing of the containment isolation valves in the purge system piping. We require that the containment isolation valves in the affected system be local (type C) leak rate tested following each use of a system. This is consistent with the action taken regarding leak testing of the purge system containment isolation vavles for the Donald C. Cook Nuclear Plant, Unit 2 and McGuire Nuclear Station, Units 1 & 2.

We have reviewed the containment isolation systems with respect to General Design Criteria 54, 55, 56, and 57 and Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," and on the basis of conformance with these criteria, we conclude that the design of the containment isolation system is acceptable.

6.2.5 Combustible Gas Control Systems

Following a loss-of-coolant accident, hydrogen may accumulate within the containment as a result of (1) metal-water reaction between the fuel cladding and the reactor coolant, (2) radiolytic decomposition of the post-accident emergency cooling water and (3) corrosion of metals by emergency core coolant and containment spray solutions. The applicant has analyzed the production and accumulation of hydrogen within containment from the above sources using the guidelines of Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident". The applicant will provide redundant Westinghouse electrical thermal hydrogen recombiners to limit the hydrogen concentration within the containment to below the Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Rev. 1)" limit of four volume percent. The applicant has used the same assumptions as Regulatory Guide 1.7 to calculate the rate of hydrogen released by radiolysis and corrosion of metals, and a 1.5 percent zirconium-water reactor in the reactor core. The 1.5 percent zirconium-water reaction was determined by assuming a maximum total reaction of 0.3 percent of the zircaloy clad in the reactor core; preliminary emergency core cooling system analyses have indicated a maximum total reaction of 0.27 percent. The 1.5 percent zirconium-water reaction was determined by assuming a maximum total reaction of 0.3 percent of the Zircaloy clad in the reactor core. Emergency core cooling system analyses have indicated a maximum total reaction of less than 0.3 percent. The use of zirconium-water reaction of 1.5 percent is consistent with our position regarding metal-water reactions as stated in Branch Technical Position CSB 6-2.

With the foregoing assumptions, and considering that the return air fan recirculation system is operated at 10 minutes following the accident, mixing the upper and lower compartment volumes, Regulatory Guide 1.7 hydrogen flammability limits (four percent) would not be reached in the containment volume until about eight days following the accident. The applicant proposes to operate the hydrogen recombiners well in advance of this time; i.e., after 24 hours, to maintain hydrogen concentration to below this limit. We find the applicants method of analysis to be acceptable.

The Westinghouse electric thermal hydrogen recombiner system incorporates several design features which are intended to ensure the capability of the system to be operated in the event of an accident. Notably among these are: (1) seismic Category I design, (2) IEEE requirements for the wiring and electrical equipment, (3) protection from missile and jet impingement from broken pipes, (4) redundancy to the extent that no single component failure can disable both recombiners, and (5) separate power supplies for each heater.

Each of the two 100-percent capacity electric recombiners is capable of processing 100 scfm of containment atmosphere for post-accident hydrogen control. We have reviewed tests that have been conducted for a full-scale prototype and a production recombiner. The tests consisted of proof-of-principle tests, testing on a prototype recombiner, environmental qualification testing and functional tests for a production recombiner. (These tests are described in WCAP-7820 and its supplements 1-4). The results of these tests demonstrated that the recombiner should be capable of controlling the hydrogen in a post-loss-of-coolant accident containment environment. The recombiner system is designed to seismic Category I criteria and to the IEEE requirements for an engineered safety feature.

Two redundant hydrogen collection systems are provided to prevent the accumulation of hydrogen within the lower compartment, subcompartments and containment dome. These areas are continuously vented by diverting a portion of the return air fan flow through the collection system and therefore limit the potential for local hydrogen pocketing.

In accordance with Regulatory Guide 1.7 and Branch Technical Position CSB 6-2, the applicant has also provided a containment purge system as backup to the recombiner system.

We conclude that the combustible gas control system satisfies the design and performance requirements of Section 50.44 of 10 CFR Part 50, "Standards For Combustible Gas Control Systems in Light Water Cooled Power Reactors", Regulatory Guide 1.7 (Revision 2), "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident", and the provisions of General Design Criteria 41, 42, and 43, and is therefore acceptable.

6.2.6 Containment Leakage Testing Program

The Sequoyah Nuclear Plant containment design includes the provisions and features required to satisfy the testing requirements of Appendix J, 10 CFR Part 50. The design of the containment penetration and isolation valves permits periodic leakage rate testing at the pressure specified in Appendix J, 10 CFR Part 50. Included are those penetrations that have resilient seals and expansion bellows; i.e., airlocks, emergency hatches, refueling tube blind flanges, and electrical penetrations.

Section III.D.2 of Appendix J to 10 CFR Part 50 requires airlocks to be leak tested at six-month intervals, and after each opening during the intervals. Section III.B.2 of Appendix J requires all penetrations to be leak tested at the calculated peak containment internal pressure corresponding to the design basis accident (Pa).

Based on plant operating experience, requiring an airlock to be leak tested after each opening is an impractical requirement when frequent airlock usage is necessary over a short period of time. Testing an airlock for leakage within a limited time period following the initial opening is more practical, and still provides the desired confidence that the leak tightness of the airlock is within acceptable limits.

The airlock design for the Sequoyah Nuclear Plant includes dual seals on the air-lock doors with the capability to apply a pressure between the seals. This will permit door seal integrity to be demonstrated without pressurizing the total airlock. This is an acceptable approach for tests other than the six-month test.

The applicant proposes to leak test the airlock door seals within three days after an airlock is opened; the volume between the door seals will be pressurized to Pa, the peak calculated containment pressure. The six-month total airlock leak test will be retained.

Based on our review of the applicant's proposed testing of a containment airlock, we conclude that the commitment to leak test the airlock door seals within three days after being opened at a test pressure of Pa is an acceptable alternative to the requirement of Section III.D.2 of Appendix J pertaining to airlock leak testing after each opening. In our reviews of other operating license applications we have determined that an exemption to Appendix J with regard to containment personnel access hatch testing is required and justified. If we grant a similar exemption for Sequoyah, our safety evaluation supporting the matter will accompany the granting documents.

The applicant has designed the Sequoyah Nuclear Plant containments such that there is no potential path by which containment leakage could bypass both the emergency gas treatment system and the auxiliary building gas treatment system and reach the environs untreated. The applicant has identified systems for which through-line or penetration leakage could bypass the annulus and be released within the areas of the auxiliary building which are treated by the auxiliary building gas treatment system. The applicant has committed to perform local leak rate tests in accordance with the requirements of Appendix J to 10 CFR Part 50 and limit the total potential leakage which could bypass the emergency gas treatment system and be treated by the auxiliary building gas treatment system to 10 percent of the containment design leakage rate (0.25 percent per day by weight of the containment atmosphere) at 12.0 pounds per square inch gauge. We have identified twenty-one additional fluid lines which we believe are also potential paths for through-line leakage from the containment to the auxiliary building. We will complete our review of these lines with the applicant and will include them as necessary in the tabulation of potential bypass leakage paths to the auxiliary building gas treatment system during development of the Technical Specifications for the operation of the plant.

With the exception of the airlock testing, for which we may grant an exemption as discussed above, the proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that containment leak-tight integrity can be verified periodically throughout service lifetime on a timely basis to maintain such leakage within the limits of the technical specifications.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J and any exemptions constitutes an acceptable basis for satisfying the requirements of General Design Criteria 52, 53, and 54.

6.3 Emergency Core Cooling System

6.3.1 Design Basis

The emergency core cooling system is designed to provide core cooling as well as additional shutdown capability for accidents that result in significant depressurization of the reactor coolant system. These accidents include mechanical failure of the reactor coolant system piping up to and including the double ended break of the largest pipe, rupture of a control rod drive, spurious relief valve operation in the primary and secondary fluid systems, and breaks in the steam piping.

The design basis is to limit clad damage due to excessive temperatures and clad-water reactions. The applicant states that the requirements will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus, with offsite power unavailable.

6.3.2 System Design

The emergency core cooling system consists of both passive and active systems. The upper head injection and low pressure accumulator tanks are passive systems that are actuated when the reactor coolant pressure falls below preset values. The active components of the emergency core cooling system are high head, medium, and low pressure pumps that are actuated by the safety injection signal. Following a postulated loss-of-coolant accident, the passive and active injection systems will operate, and after the water inventory in the refueling water storage tank has been depleted, the long-term recirculation mode will be activated.

The upper head injection system consists of a borated water-filled tank connected to a nitrogen tank that is pressurized. When the reactor system pressure falls below the charging pressure, water will be injected into the top of the reactor vessel.

Each of the four low pressure accumulator tanks contains borated water pressurized with nitrogen gas to approximately 400 pounds per square inch absolute. When the reactor coolant system pressure falls below that in the tanks, water is forced into the four cold legs. The high pressure injection mode consists of the operation of two high head centrifugal pumps which provide high pressure injection of boric acid solution into the reactor coolant system, upon actuation by a safety injection signal. Also part of the high pressure injection mode are two safety injection pumps which take suction from the refueling water storage tank with a boron concentration of 2,000 parts per million.

Low pressure injection consists of two residual heat removal pumps which take suction from the refueling water storage tank.

6.3.3 Evaluation

Single Failures

We have reviewed the system description and drawings to assure that abundant core cooling will be provided during the initial injection phase with and without offsite power and assuming a single failure. The upper head injection subsystem is aligned for injection, through two parallel lines with normally open isolation valves, when the primary pressure drops below the upper head injection set pressure. An inadvertent valve closure in either discharge line will not preclude upper head injection. Each discharge line has two isolation valves in series which are closed automatically when a low level in the upper head injection accumulator is reached. Failure of a single valve to close will not prevent isolation of the upper head injection accumulat... The cold leg accumulators have normally open isolation valves in their discharge lines. These valves will have control logic design to preclude inadvertent closure during the emergency core cooling injection phase. There are two pumps in each of the three different active injection systems. The pumps in each system are connected to separate power buses and would be powered from separate diesel generators in the event of loss of offsite power, as required by General Design Criterion 17. Thus, at least one pump in each injection train would be actuated. The high and intermediate head injection systems contain parallel valves in the suction and discharge lines, ensuring system function even if one valve fails to open. The low head injection system is normally aligned so as to not require any valve actuation during the injection phase.

We have identified those emergency core cooling system valves, which if inadvertently mispositioned due to a single failure or operator error, could seriously degrade the peformance of the system. See Section 7.3.2 for a discussion of the design changes made to ensure proper positioning of these valves.

We have reviewed emergency procedures which may permit early manual reset of the safety injection signal during the injection phase. As a result, we require that the emergency operating procedures preclude manual reset of the safety injection signal for at least 10 minutes following a safety injection signal. The applicant has committed to conform with this requirement.

We have reviewed the procedures for loading emergency core cooling system equipment on the emergency power bus following an offsite power failure after manual reset of the safety injection signal as discussed in Section 7.3.2.

The applicant has stated that several valves may be under water following an assumed loss-of-coolant accident. However, we have verified these valves are not required to function after being flooded. See Section 7.3.4 for additional information.

The applicant has proposed both manual and automatic valve actuation designs to switch from the injection mode to the recirculation mode. These procedures, which effectively switch the residual heat removal pump suction from the refueling water storage tank to the containment sump, would be initiated about 15 to 30 minutes after an assumed loss-of-coolant accident. We find the design modifications acceptable.

We have reviewed the procedures for initiating hot leg injection during the recirculation phase to preclude excessive buildup of boron concentration in the pressure vessel. We have concluded that there is sufficient redundancy in injection lines and pumps to ensure adequate hot leg injection when required.

We have reviewed the applicant's submittal on emergency core cooling passive failures following a loss-of-coolant accident. Passive failures considered were limited to pump and valve seal leakage. Leakage is detected by conductivity-type water level detectors and sump pump operation. Each emergency core cooling system pump and heat exchanger compartment is monitored by a water level detection device. The staff finds the passive failure leakage detection system acceptable.

The design features discussed above demonstrate that the emergency core cooling system complies with the single failure criterion of General Design Criterion 35.

Qualification

The emergency core cooling system is designed to Seismic Category I requirements in compliance with Regulatory Guide 1.29, "Seismic Design Classification," and is housed in structures designed to withstand a safe shutdown earthquake and other natural phenomena as required by General Design Criterion 2. The equipment is designed to Quality Group B in compliance with Regulatory Guide 1.26, "Seismic Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

The emergency core cooling system is protected against missiles inside and outside containment by the design of suitable reinforced concrete barriers which include reinforced concrete walls and slabs. Barriers are designed to be three times as thick as the penetration depth of a possible missile; when a steel barrier is used, the required thickness of this barrier is one-twelfth of that required for a reinforced concrete barrier. This constitutes partial fulfillment of General Design Criterion 4. The protection of the system from pipe whip inside and outside of containment is discussed in Section 3.6.

The active components of the system have been designed to function under the most severe duty loads including the safe shutdown earthquake as discussed in Section 3.9. The system is designed to permit periodic inspection as discussed in Section 5.4 in accordance with ASME Code, Section XI, which constitutes compliance with General Design Criterion 36.

The emergency core cooling system incorporates two subsystems which serve other functions. The residual heat removal system provides for decay heat removal during reactor shutdown. At other times the residual heat removal system is aligned for emergency core cooling operation. The centrifugal charging pumps are utilized for maintaining the required volume of primary fluid in the reactor coolant system; on emergency core cooling system actuation signal, the system is aligned to emergency core cooling operation and the chemical and volume control system function isolated. In neither case does the normal system use impair its capability to function as an integral portion of the emergency core cooling system.

Each reactor unit has a separate emergency core cooling system, however, portions are housed in a common auxiliary building. The individual components within the building are separated by barriers and the installation has been reviewed for possible flooding as discussed in Section 2.4. The design constitutes a demonstration that the emergency core cooling system is not shared by the two units, in compliance with General Design Criterion 5.

Instrumentation and Control

The emergency core cooling system is initiated automatically on: (1) low pressurizer pressure, (2) high containment pressure, (3) high differential pressure between any two steam generators, (4) high steam flow coincident with low T_{AVG} or low steam pressure. As noted above, the cold leg accumulator and upper head injection subsystems actuate automatically when the reactor coolant pressure decreases to a value below that at which the subsystems are maintained. This meets the requirements of General Design Criterion 20.

Equipment status indication is provided in accordance with the requirements of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems". Automatic actuation is provided by redundant signals whose diversity is noted above. The emergency core cooling system may also be manually actuated, monitored, and controlled from the control room as required by General Design Criterion 19.

The instrumentation needed to monitor and control the system equipment following a loss-of-coolant accident has been reviewed. This instrumentation provides sufficient information so that the operator can maintain adequate core cooling following an assumed loss-of-coolant accident.

Functional Design

The available net positive suction head for all the pumps in the emergency core cooling system (the safety injection, centrifugal charging, and residual heat removal pumps), has been shown to provide adequate margin in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps."

Boron injection tank pressure is indicated in the control room and has a high pressure alarm. All accumulators have pressure indication in the control room with high and low pressure alarms.

We have reviewed the refueling water storage tank vent line design to determine the potential for vent blockage due to freezing or other causes. The staff finds the vent design of the refueling water storage tank acceptable.

The valve arrangements on the emergency core cooling system discharge lines have been reviewed with respect to adequate isolation between the reactor coolant system and the lower pressure emergency core cooling system. In some lines, this isolation is provided by two check valves in series with a closed isolation valve which complies with the staff position.

Other discharge lines have only two check valves in series. This arrangement is acceptable provided periodic leak detection across each check valve is performed during plant operation. Test lines are provided for periodic checks of leakage of reactor coolant past the check valves forming the reactor coolant system accumulator boundaries. For the upper head injection system there are test vents between the two isolation valves in series in each train to ascertain leakage past the first valve forming the reactor coolant system/upper head injection boundary.

All emergency core cooling system lines, including instrument lines, have suitable containment isolation features that meet the requirements of General Design Criterion 56 and Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment", as discussed in Section 6.2.

We have reviewed the capability of the Sequoyah plants to detect intersystem leakage between the reactor coolant system and other water systems. The staff finds acceptable the capability to detect intersystem leakage into the cold leg accumulators, the upper head injection system, the chemical and volume control system, the safety injection system, and the residual heat removal system from the reactor coolant system.

Switchover from the injection to recirculation phase is accomplished manually witn automatic backup, i.e., automatic switching of residual heat removal pump suction from the refueling water storage tank to contaiment sump at a level 40,000 gallons below the low level setpoint.

The applicant has proposed to divert emergency core cooling flow from the reactor coolant system to the auxiliary spray headers in the containment if only one containment spray train is operational. This would be accomplished by isolating the direct injection of the residual heat removal pumps into the cold legs and diverting the residual heat removal discharge to a safety injection train and the auxiliary spray line. The applicant has shown that this procedure would not be required for at least one hour after an assumed loss-of-coolant accident. After one hour, the core flow from one safety injection pump would be greater than 1.8 times the required core cooling flow. The applicant has justified that no pump runout will occur should the residual heat removal cold leg injection line fail to be closed. On the basis of the above, we find the proposed diversion acceptable, provided that operating procedures preclude initiation of containment spray flow diversion before the emergency core cooling system has been aligned for the recirculation phase or one hour after the start of a postulated loss-of-coolant accident, whichever is later. The applicant will incorporate this requirement in the plant operating procedures.

To minimize the potential for water hammer occurring due to emergency core cooling injection into dry lines, the applicant has stated that during normal operation the emergency core cooling system lines will be maintained full. The capability to maintain these lines full of water will be verified prior to startup and will constitute a periodic surveillance requirement in the technical specifications.

6.3.4 Tests and Inspection

The applicant will demonstrate the operability of the emergency core cooling system by subjecting all components to preoperational tests and periodic testing, as required by Regulatory Guides 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," and 1.79, "Preoperational Testing of Emergency Core Cooling System for Pressurized Water Reactors," and General Design Criterion 37. The tests performed fall into two categories.

(1) Preoperational Tests

One of these tests is to verify system actuation, namely the operability of all emergency core cooling system valves initiated by the safety injection signal, the operability of all safeguard pump circuitry down through the pump breaker control circuits; and the proper operation of all valve interlocks.

Another test is to check the cold leg accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The applicant will perform a low pressure blowdown of each accumulator to confirm the line is clear and check the operation of the check valves.

We also require preoperational tests to be performed with the upper head injection system to demonstrate hydraulic resistances, absence of nitrogen entrainment (to include vortexing phenomena), level setpoints, and isolation valve function. These tests have been performed; confirmatory documentation will be reviewed by the staff. We will report further on this matter in a supplement to this report.

Operational tests of all the major pumps comprise the last category. These pumps consist of the high head injection pumps, the residual heat removal pumps, and the safety injection pumps. The applicant will use the results of these tests to evaluate the hydraulic and mechanical performance of these pumps delivering through the flow paths for emergency core cooling. The pumps will be operated under both miniflow (through test lines) and full flow (through the actual piping) conditions.

By measuring the flow in each pipe, the applicant will make the adjustments necessary to assure that no one branch has an unacceptably low or high resistance. They will also check the system to assure there is sufficient total line resistance to prevent excessive runout of the pump. The applicant must show that the minimum acceptable flows used in the loss-of-coolant accident analysis are met by the measured total pump flow and relative flow between the branch lines, and that the maximum flow rate predicted from the test results confirms the maximum flow rate used in the net positive suction head calculations under the most limiting conditions.

The applicant has indicated a commitment to implement Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1)," which covers testing of the emergency core cooling system. The applicant has run scale model tests of the containment emergency sump design.

We have reviewed results of that testing and has requested additional information to verify the capability of the sump to perform adequately in the event of certain postulated line breaks in the vicinity of the sump. We will report further on these matters in a supplement to this report.

The systems will be accepted only after demonstration of proper actuation of all components and after demonstration of flow delivery to all components within design requirements.

(2) Periodic Component Tests

Routine periodic testing of the emergency core cooling system components and all necessary support systems at power will be performed. Valves which operate after a loss-of-coolant accident are operated through a complete cycle, and pumps are operated individually in this test on their miniflow lines, except the charging pumps which are tested by their normal charging function. These tests will be performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, as discussed in Section 5.4. We also require that the low head safety injection system be tested in the injection mode at the end of each refueling outage. This requirement will be reflected in the technical specifications.

We conclude that these tests conform to the requirements of General Design Criterion 37 and are acceptable.

6.3.5 Perform_ce Evaluation

The emergency core cooling system must provide abundant core cooling to minimize fuel and clad damage in accordance with the requirements of 10 CFR Part 50.46. An upper head injection/loss-of-coolant accident evaluation model has been approved by the staff. The applicant has recently submitted a loss-of-coolant accident analysis which the staff has reviewed. We require additional information from the applicant to confirm that the most limiting case has been analyzed. We will report further in a supplement to this report.

The applicant has presented acceptable procedures to preclude excessive boron concentrations in the pressure vessel during long-term cooling. The applicant assumed conservative boron contributions from the boron injection tank, cold leg and upper head injection accumulators, the reactor coolant system, the refueling water storage tank, and the ice bed.

We are reviewing postulated moderate energy line breaks in the residual heat removal system when in the normal shutdown cooling mode. Under these conditions, the safety injection signal is blocked and much of the emergency core cooling system equipment is bypassed. The applicant has indicated that sufficient time is available for operator action to respond to the postulated break, and will provide information verifying actions required and time available. We will report further on this matter in a supplement to this report.

6.3.6 Conclusions

As noted above, we have reviewed the emergency core cooling system design and functional capability to assure that there are suitable redundancy in components and ,eatures, and suitable interconnections, leak detection, isolation, and containment capabilities so that the system will be capable of performing its safety function assuming a single failure (with or without offsite power) as required by General Design Criterion 35. Based on our review and on conformance with criteria noted above, we conclude that the system is acceptable pending satisfactory confirmation of the issues noted.

The ability of the emergency core cooling system to provide abundant emergency core cooling as required by General Design Criterion 35 has been reviewed and found acceptable pending confirmation that the the loss-of-coolant accident analyses meet the requirements of 10 CFR 50.46.

6.4 Habitability Systems

The emergency protective provisions of the control room related to the accidental release of radioactivity or toxic gases are evaluated in this section. While relevant portions of the control room ventilation system are mentioned here, a more complete description and evaluation of the control room ventilation system is given in Section 9.4 of this report.

6.4.1 Radiation Protection Provisions

The applicant will meet General Design Criterion 19, Control Room, of Appendix A to 10 CFR Part 50, by use of concrete shielding and by installing redundant 4000 cubic feet per minute recirculating charcoal filters in the control room ventilation system. The control room will be isolated automatically and placed in the emergency mode by a safety injection system signal and/or by a radiation signal from beta detectors located in the outside air intake stream. In the emergency mode, the control room will be pressurized by introducing 200 cubic feet per minute of filtered outside air. We have calculated the potential radiation doses to control room personnel following a loss-of-coolant accident and have found that they are within the General Design Criterion 19 dose guidelines. Thus, we find that the control room is adequately protected against potential accidents involving airborne radioactivity.

6.4.2 Toxic Gas Protection Provisions

We have reviewed the question of toxic gas releases as they apply to control room habitability in accordance with the guidelines given in Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." We had identified a potential toxic gas hazard due to the storage of acrolein, which was to be used for Asiatic clam control in the condensers, near the control room; however, the applicant has agreed to eliminate the use and storage of this chemical.

The storage of four 150-pound chlorine cylinders within the turbine building is a safety concern due to the possibility of direct communication between the turbine building and the control room via several doors. To safeguard against chlorine entering the control room directly through potentially open doors, the applicant has committed to maintain a positive pressure inside the control room at all times, as well as to establish administrative procedures to assure that the doors remain closed at all times when not in use for necessary transit of personnel or materials. To guard against the possibility of chlorine entering the control room via the outside air intakes, the applicant has installed redundant, quick-acting chlorine detectors in the intakes. The detectors will isolate the control room ventilation system automatically upon detection of chlorine. In addition, the control room is equipped with portable self-contained breathing apparatus for the control room operators. We conclude that the control room habitability system is adequate to protect the plant operators against an accidental release of chlorine.

7.0 INSTRUMENTATION AND CONTROL

7.1 General

We have evaluated the adequacy of the protection and control systems of Sequoyah using as bases (1) the Commission's General Design Criteria, (2) the Institute of Electrical and Electronic Engineers (IEEE) standards covering systems and equipment for nuclear powered generating stations, and (3) the applicable regulatory guides for light water reactors, as now included in Table 7-1 of the Standard Review Plan.

The design of the instrumentation and control systems of the Sequoyah plant, excluding the secondary cycle, is similar to that of Donald C. Cook and Trojan. The design of the instrumentation and control systems for the secondary cycle is similar to that of Diablo Canyon, Donald C. Cook, and Zion stations. Items of difference are identified and evaluated in this report. The evaluation o this design concentrated on equipment qualification, system implementation, and applicability of previous generic evaluations to the Sequoyah systems.

A major design change from that presented in the Preliminary Safety Analysis Report was the incorporation of a newer reactor trip and engineered safety features actuation system. This will be discussed further in Sections 7.2 and 7.3 of this report.

The results of our review of selected logic and schematic diagrams and drawings are reflected in this report. However, the review of additional drawings, outstanding issues, and an additional site visit will be reported in a supplement to this report.

7.2 Reactor Trip System

7.2.1 General

Our review examined selected aspects of the protective system that initiates, monitors, bypasses, controls, and accomplishes trip of the reactor. This review includes review of field implementation performed during the first site visit.

The reactor trip system is subdivided into (1) the process analog system, and (2) the reactor trip system actuation logic. Several aspects of the review of these subdivisions will be discussed in the appropriate subsections.

7.2.2 Process Analog System

Sequoyah is provided with the Foxboro Process Analog System which supplies power to and receives and processess analog signals from the trip system sensors. This system is essentially that described and reviewed for Donald C. Cook Units 1 and 2. We conclude that the functional design and implementation of the process analog system is acceptable for these plants with exceptions noted below.

Seismic Qualification of Westinghouse-Supplied Class 1E Equipment

We concluded from previous reviews (e.g., Diablo Canyon, Trojan, D.C. Cook) that the results of the seismic qualification tests of the Westinghouse-supplied Class 1E equipment, including the solid state protection system as reported in WCAP-7817, "Seismic Testing of Electrical and Control Equipment," were not acceptable, primarily because of deficient test methods and procedures. We are currently reviewing the acceptability of the seismic qualification of this equipment with the applicant. When the review is completed, the results will be reported in a supplement to this report.

Environmental Qualification of Westinghouse-Supplied Class 1E Equipment

Topical report WCAP-7744, "Environmental Testing of Engineered Safety Features Related Equipment," is still under review for use as the basis to support the Westinghouse program of environmental qualification of safety-related Class 1E instrumentation and control equipment. The staff is currently determining the acceptability of the environmental qualification of this equipment with the applicant. The results of this determination will be reported in a supplement to this report.

The applicant is analyzing the steamline break accident to determine the environmental extremes to which Class 1E sensors and equipment located inside the containment could be exposed as the result of a steam line break occurring inside the containment. As required by the staff, the applicant is taking steps to provide equipment qualified to function in the worst-case environment that could occur within the containment. The staff will review this additional information when it becomes available and report on its acceptability in a supplement to this report. (See Section 6.2.1 for additional information.)

Response Time Testing

The applicant had originally proposed to measure response times of the reactor trip system and engineered safety features activation system channels without including the sensor response times. This proposal was unacceptable. In Amendment 37, the applicant committed to measure the response times of these channels, including the sensors, to ensure that the actual response times remain conservative with respect to those assumed for the safety analyses. The detailed test procedures are now being developed by the applicant. The staff will review the general procedures and selected detailed procedures and will assure their acceptability. We will report on our confirmatory review of the program in a supplement to this report.

Anticipatory Trips

The anticipatory trips had previously included the reactor coolant pump underfrequency, undervoltage and breaker open trips together with the turbine trip. We took the position that all such input to the reactor trip system must fulfill all the requirements for Class 1E circuits. At this point, the reactor coolant pump breaker open trip was deleted because its function would be performed by the reactor coolant low flow trip. The underfrequency and undervoltage trips were reclassified as essential reactor trips and made to fulfill all the requirements for Class 1E circuits, including relocation to the seismic Category I auxiliary building from their original location in the non-seismic Category I turbine building. We find these actions acceptable in accordance with the guidance described in Table 7-1 of the Standard Review Plan.

The turbine trip sensors and their cabling remain in the non-seismic Category I turbine building. However, the sensors and wiring of this trip input meet the requirements of IEEE Standards 279-1971 and 308-1971, except that those portions located in the turbine building are not seismically qualified. On the basis that these sensors are similar to others which have been seismically qualified and that other inputs are available to trip the reactor in the case of a seismic event, we find this acceptable.

Neutron Detector Seismic Test

The ex-core neutron detectors were subjected to a seismic test which initially failed to cover the required frequency range of from one to 33 Hertz. In particular, the range of from one through six Hertz where most resonances may occur was not covered. The applicant has since submitted a report showing that these detectors have been seismically tested over the full range of one to 33 Hertz. Based on our review of this report and on subsequent documentation of the test results in Amendment 39, we find the results acceptable.

7.2.3 Reactor Trip System Actuation Logic

Functionally the reactor trip system actuation logic has not changed from the earlier design. However, the equipment and implementation is materially different because of the provision of a solid state actuation logic system. This solid state protection logic system generates appropriate signals to open the reactor trip breakers whenever any required combination of input signals from the process analog system occurs.

The staff evaluation of this new protection system design is presented in Section 7.2 and 7.4 of the Donald C. Cook Safety Evaluation Report dated September 10, 1973, and in the evaluation of Topical Report WCAP-7488-L, "Solid State Logic Protection Description," which also described the operation of the system. This documentation showed that the design is acceptable for other plants provided that (1) it is seismically and environmentally qualified, (2) means are incorporated into the breaker trip circuitry to prevent negating the trip function by placing both trains in bypass or both trains in test or one train in bypass and one in test, (3) the protective functions provided meet the safety criteria for the plant, and (4) implementation of the system meets the requirements for preserving the independence of redundant portions of the protection system. It has been subsequently determined that item 2 has been successfully remedied as verified by review of the trip breakers and their interlocking system as implemented for the Sequoyah plant. An onsite review of the field installation for adequacy of separation and independence has been made and is reported below. Based on our review we have concluded that the solid state logic protection system is acceptable for this plant with the comments stated below.

Electrical Isolation

The photodiode isolators used to prevent noise and electrical interference that could be generated in the non-safety-related equipment from coupling back into the safety-related equipment as implemented in the solid state protection system, had not been previously qualified as acceptable isolation devices. On the basis of noise and interference immunity tests reported in Appendix A of "Westinghouse Report on Protection System Noise Tests" dated December 1974 which was included as part of the Diablo Canyon review, these devices were determined to qualify as acceptable isolation devices.

Conformance to IEEE Standard 379-1971 and Regulatory Guide 1.53

In response to our request for information showing how the reactor trip system design for independence met the requirements of IEEE Standard 379-1971 and Regulatory Guide 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems," the applicant provided additional information addressing these requirements. The staff has reviewed this additional information and found it acceptable. We conclude that the reactor trip system independence conforms with these standards.

Separation and Independence of Reactor Trip System Actuation Logic

During the site visit of June 21 to 23, 1977, we reviewed the field installation for adequacy of separation and independence of the redundant circuits. We found that the internal wiring of the Westinghouse analog process control racks appeared to provide no separation between the protection system inputs to the isolation amplifiers and the isolated outputs to the control system. The input wiring as well as other protection system wiring was bundled together in the same wireways with the redundant isolated outputs, which in some instances terminated in a common control rack. This appeared unacceptable because a failure in the control portion of the system could negate protective actions by propagating around the isolation amplifier through this lack of physical separation of the input and output wiring.

The applicant's response to our concern referred to noise and crosstalk tests that were performed on a similar type of process control system equipment installed at Diablo Canyon. We found these results to be acceptable in our Diablo Canyon review. We also determined that differences in wiring layouts between the Sequoyah and Diablo Canyon process control racks were not significant enough to affect the applicability of the test results to Sequoyah and that the installation of the external cabling at Sequoyah was adequately enveloped by the test installation. The applicant has also determined that these output cables are not routed with cables carrying voltages higher than those which the isolation devices and wiring are qualified. We further believe that the probability of occurrence of significant physical damage to this cabling inside these heavy cabinets was sufficiently small to be acceptable for this plant. Based on the above, we conclude that this response is acceptable.

Solid State Protection System General Warning Alarm Circuits

An Abnormal Occurrence Report from another operating reactor of similar design pointed out a defect in the solid state protection system that constituted an unacceptable compromise of the independence of the reactor trip system. Westinghouse has issued a field modification to eliminate this problem. We reviewed the proposed change and found it acceptable. The applicant has notified the staff by letter that this change has been implemented at Sequoyah, and therefore we find this action acceptable.

Physical Separation of Wiring In Solid State Protection System Multiplexer and Demultiplexer

During the first site visit, the concern was raised that spurious signals from the isolated (control) side could be coupled into the protection side through the close proximity of the input and output wiring bundles of the isolation boards because of inadequate separation. In October 1974, a series of tests was run on an identical solid state protection system rack installed at Diablo Canyon. The results were reported in the Westinghouse document referred to in the subsection entitled Electrical Isolation above. These test results indicated that insufficient signals or noise were coupled into the protection circuits to have any effect on the performance of the safety system. The results have been found to be applicable to the Sequoyah installation. We also found that the probability of significant damage occurring to this cabling inside those substantially constructed cabinets was sufficiently small as to be acceptable for this plant. Based on the above, we find the Sequoyah installation acceptable.

7.2.4 Testability of Protection System

The applicant has stated that the reactor trip system conforms to the testability requirements of IEEE Standards 279-1971, and 338-1971, and Regulatory Guide 1.22, "Periodic Testing of Protection System Actuati/ Functions." We have examined the measures provided to enable testing of the trip system and have concluded that with the exceptions noted in Section 7.2.2, the trip system's testability is acceptable.

7.2.5 Control Room Rack Wiring

The staff was concerned that the wiring and cabling in the control room racks, which is to be separated according to train and channel, could be intermixed and might lack adequate separation. As a result of our review some changes were made by the applicant and we concluded that the design as presented in the Final Safety Analysis Report was acceptable. On the first site visit we were unable to review the implementation of the design because the cabling and wiring was not sufficiently complete. In several areas we noted an apparent lack of separation between redundant circuit wiring, but were informed that barriers would be provided in these areas. We will verify the design implementation on the next site visit and report our findings in a supplement to this report.

7.2.6 Instrument Trip Setpoint Determination

For all plants, we have been concerned with the margins allowed for instrument drift, instrument accuracy, inaccuracies in the determination of limits of safe operation, and methods of determination of drift. We have requested that additional information addressing this subject be provided and suggested that Regulatory Guide 1.105, "Instrument Spans and Setpoints," be considered as a guide for providing the requested information. TVA has stated that they are preparing the requested information and will submit it after our review of similar information on D.C. Cook, which is very similar to Sequoyah, is completed. We have concluded that this commitment is acceptable and that use of the present trip setpoints until the first refueling outage does not pose a threat to the public health and safety. We will condition the operating license to assure receipt of acceptable information on this matter prior to startup following the first refueling outage.

7.2.7 Removal of Power to Control Rod Drive Mechanisms on Scram

The staff developed a concern about whether all power to the control rod drive mechanism circuits was removed upon opening the reactor trip system breakers. Our drawing review showed that the 70 and 125 volt direct current power to the hold, lift and gripper coils is removed; however, about five kilovolt-amperes of 120 volt alternating current power is tapped off ahead of the scram breakers that form the source for 24 volt direct current control rod drive mechanism control circuit power. Analysis by the applicant shows that there is very little likelihood that this power could prevent the rods from dropping in the event of a demand for scram. We agree and find the control rod drive mechanism circuits to be acceptable in this respect.

7.2.8 Radiation Instrumentation Saturation Effects

The staff questioned whether saturation or "foldover" testing had been performed and would be performed periodically on all safety-related radiation detectors, sensors, and signal handling equipment. The applicant stated that such testing had been performed only on certain radiation monitors. Subsequent response to a request for information identified two G-M tube monitors in the fuel handling area that initiate isolation of the auxilairy building when high radiation levels are detected. The applicant's commitment to preoperational and periodic testing of these monitors for saturation effects is acceptable, and the requirement for periodic testing will be incorporated in the technical specifications.

Two scintillation monitors in the main control room ventilation intake duct that initiate isolation of the control room when high radiation levels occur were also identified as safety-related. These monitors were stated not to be subject to saturation effects because testing showed that operation at 100 times the full scale reading of 10^6 counts per minute had no detectable effect. The trip point at which isolation of the control room ventilation system is initiated is 400 counts per minute. We believe that the margin between this level and the maximum test level is large enough to ensure that saturation effects will not prevent the monitor from accomplishing timely isolation of the control room ventilation system. The signal handling equipment will undergo preoperational and periodic testing for saturation effects. We find this response and the commitment to periodic testing acceptable and will incorporate the requirement for such testing in the technical specifications.

The ex-core neutron detectors are designed to limit the current demand from the power supply to avoid damage to the detector electronics that might result in "foldover." This limit is set at 10 times full nuclear power. No postulated reactor excursion could exceed this limit, therefore "foldover" should not occur in these detectors. Preoperational and periodic testing of the signal handling equipment will be done to ensure that this ability to resist saturation is not degraded. We find this acceptable and will incorporate requirements for periodic testing in the technical specifications.

7.2.9 Conclusions

Provided the outstanding items identified in item 7.2.2 are acceptably resolved, we can conclude that the design of the reactor trip system meets the Commission's requirements as discussed in that section and is acceptable.

7.3 Engineered Safety Features Actuation Systems

7.3.1 General

Our review examined all aspects of the protective system that initiates, monitors, bypasses, controls, and actuates the engineered safety features systems and their essential supporting systems.

7.3.2 Engineered Safety Features Actuation Systems Logic

The engineered safety features actuation systems logic is part of the solid state protection system that is discussed in Section 7.2 of this report. This logic is similar to that evaluated in Section 7.5 of the Safety Evaluation Report for Donald C. Cook Units 1 and 2. An identified difference is that increased on-line testability for engineered safety features systems is provided for Sequoyah.

Provided that items in Section 7.2.2 and the additional items below are acceptably resolved, and on the basis of previous evaluations on the D.C. Cook and Trojan plants and of conformance with requirements now included in Table 7-1 of the Standard Review Plan, we find that the engineered safety feature actuation system is acceptable for these plants.

Upper Head Injection

In our review, we requested and reviewed additional information describing the provision and operation of the instrumentation that senses low water level in the upperhead injection accumulator and initiates the closure and gagging of the hydraulically actuated isolation valves as well as the interlocks that control the reopening of those valves. Our review showed that spurious closure of one isolation valve will not prevent performance of the system safety function because two redundant 100 percent capacity injection lines are provided. Two valves in series in each injection line ensure that failure to close one isolation valve will not prevent the required isolation of the accumulator after its charge is delivered.

The isolation valves are required to gag automatically only during the accident mitigation sequence. The safety injection signal must be present to initiate the gagging, therefore it must not be reset before this action has occurred since resetting removes the signal and prevents automatic initiation of isolation valve closure and gagging. This is acceptable because we have required that the safety injection signal not be reset prior to ten minutes after initiation of safety injection.

During normal cooldown, closure of these valves must be manually initiated before the reactor coolant system pressure falls below the upper head injection accumulator pressure. The gag insertion must then be manually initiated when the valve reaches its fully closed position. The staff requires that these valves remain closed during the time the reactor coolant system is cooled and depressurized to prevent the inadvertent discharge of the upper head injection accumulator into the reactor coolant system thus causing its overpressurization.

Removal of power from the isolation valves, their gag motors, and the hydraulic service panel to meet the requirements of the staff position is not needed because of the valve arrangement discussed above and the fact that two separate, deliberate, manual actions must be taken to initiate valve reopening. Since no single random failure can cause the opening of these valves and the consequent discharge of the accumulators, the staff position is acceptably met.

During startup, the operator must manually initiate gag removal and then manually initiate the opening of each of the four upper head injection isolation valves when the reactor coolant system pressure rises above the safety injection system unblock pressure. This manual restoration procedure conflicts with part (1) of Branch Technical Position 4 (ICSB) "Requirements on Motor-Operated Valves in the ECCS Accumulator Lines," which requires automatic opening of these valves to restore the upper head injection system to operable status. However, our review has shown that (1) since no single failure can prevent operation of the upper head injection system, power need not be removed from the valves or their controls; (2) since power is not removed, position indication and out-of-position alarms remain energized and functional for each isolation valve as required by Branch Technical Positions 4 and 18 (ICSB), "Application of Single Failure Criteria to Manually-Controlled Electrically-Operated Valves", (3) the technical specifications will require that surveillance of the valve position be performed once every twelve hours whenever the reactor coolant system is pressurized; and (4) the staff has determined that these valves need not be opened immediately upon reaching the safety injection system unblock point during the startup. Based on these considerations we have concluded that operator action to open the upper head injection system isolation valves during startup is acceptable.

Engineered Safety Features Final Actuated Device Testing

We have reviewed the applicant's plans for testing the engineered safety feature final actuated devices and have concluded that the procedures identified and plans described for these tests meet the recommendations of Regulatory Guide 1.22, "Periodic Testion of Protection System Actuation Functions" and IEEE-279, and are acceptable.

Isolation Valve Interlocks and Position Indication

To prevent spurious closure at power or opening at shutdown of the cold leg accumulator isolation valves and the block valve in the suction line from the refueling water storage tank to the emergency core cooling pumps, the staff requires that power be removed from these valves during these two states of operation. However, removal of power from the motor control centers for these valves results in loss of power to the position indication for these valves. This loss of position indication does not meet the requirements of Branch Technical Positions 4 and 18 (ICSB), which require that operable, redundant position indication be provided in the main control room for these valves. We required that information be furnished to show how the requirements of these two Branch Technical Positions will be met. The applicant indicates that the design will be modified so that control power, and therefore position indication, will be retained when motor power is removed. We will review forthcoming confirmatory information when submitted and report further on this matter in a supplement to this report.

Spurious Operation of Manually Controlled, Electrically Actuated Valves

During our review, we were concerned that single failures in the electrical control circuits of these valves could result in spurious operation of certain valves in the emergency core cooling system which could have unacceptable consequences. The applicant has proposed circuit modifications using additional contacts on the main control board switches that in effect open the control circuit to both sides of the "open" and "close" coils, thus electrically isolating them. Plastic covers over the switch handles prevent inadvertent switch actuation. This modification as implemented at Sequoyah is effective only for the main control board switches. However, the remote control switches are disconnected by the transfer switches, which are located according to unit and train in locked closed side rooms off the auxiliary control room. Whenever access is gained to these rooms and the transfer switches operated, an alarm sounds in the main control room. We have reviewed this proposed modification and found that it provides acceptable protection against spurious valve operation that could result from electrical malfunctions including that caused by mechanical or fire damage to wiring.

Automatic Switchover of Emergency Core Cooling

We took the position that the originally proposed manual switchover from the injection phase to the recirculation phase of post-loss-of-coolant accident core cooling was not sufficient and required the applicant to provide automatic action to back up the manual action. The applicant responded by providing a scheme of automatic switchover with manual backup. We completed our review of the applicant's design of the instrumentation, logic, indicating, and interlock circuits and equipment provided by the applicant to initiate, monitor and control this automatic switchover. We have concluded that there is reasonable assurance that it will perform its safety function in accordance with the Commission's requirements as now included in Table 7-1 of the Standard Review Plan, and is acceptable.

Emergency Core Cooling System and Engineered Safety Feature Interlocks

In some previous pressurized water reactor designs, emergency core cooling system and engineered safety feature valves and pumps were interlocked in such a manner as to compromise the independence of the redundant trains by having interlocks from one train perform functions in the redundant train. Because of this interdependence, power, instrumentation, or equipment failure in one train could cause valves to fail to operate or pumps to start in the redundant train.

The design of the emergency core cooling and some of the engineered safety feature systems for Sequoyah were reviewed to determine if this type of problem existed in these systems. Our review found several valves that were interlocked with valves in the redundant train. Isolating relays were interposed between the redundant trains to provide electrical isolation. Functional isolation was not provided by these relays. However the majority of these inter-train interlocks were to valves in the minflow lines for the safety injection pumps. Analysis of these cases showed that performance of the system safety function could still be accomplished even if a failure of the interlock occurred. We conclude that this is acceptable.

Effect of Power Transients on Safety-Related Equipment

A. Voltage Degradation on Engineered Safety Feature Buses.

This concern arises from the experience of Millstone Point, Unit 2 with power system transients during July 1976 and the effects of these transients on the safety-related systems that are energized from these engineered safety feature buses. The staff's generic review of this matter resulted in a request that the applicant review his power system design to ascertain that it complied with these four generic staff positions.

Position I: Provision of an Additional Level of Under- and/or Overvoltage Protection With a Time Delay;

Position II: Interaction of Onsite (Standby) Power Sources with Load Shed Feature;

Position III: Standby Power Source Testing;

Position IV: Optimization of Transformer Tap Settings.

We have reviewed the applicant's response to these four positions and have reached the following conclusions.

Position I: The applicant has taken exception to this position by stating (a) that his electric distribution systems have been designed to use the voltage limits specified in ANSI Standard C-84-1 (1970), (b) that his safety-related equipment has been designed to function satisfactorily throughout this range of voltages, (c) that his grid stability studies have shown that these voltage limits will not be exceeded, and (d) that, since the safety-related equipment is not supplied from the offsite power but from the turbine-generator through the unit auxiliary transformers, it is safer to operate the plant than to immediately shut down and depend on the onsite diesel generators.

We agree with items (a) and (b), but for item (c) we do not have sufficient information to conclude that for all possible combinations of events, loadings, and generating station outages, the grid will not suffer voltage degradations beyond the ANSI C-84-1 limits over the life of the plant. We therefore require that, whenever any change in grid conditions that could affect its stability is made or found, its stability shall be reanalyzed and the results of this reanalysis used to determine that the voltages on Sequoyah's engineered safety feature buses will not degrade beyond the ANSI C-84-1 limits or that other unacceptable conditions affecting these buses will not occur. In its response to our positions the applicant has stated that such analyses will be frequently made in any event. With regard to item (d), there are several sets of plant conditions under which the engineered safety feature buses are necessarily supplied from the grid. These include startup, hot standby, generator trip, turbine trip, and shutdown. The implementation of Staff Position I will protect against unacceptable effects occurring on the buses during these times.

In view of the above, we conclude that insufficient basis has been presented to justify exception to Position I for the life of the plant, and we require compliance with that position. The applicant has indicated that he will conform fully prior to startup following the first refueling outage, and the operating license will be so conditioned. Based on conformance with Positions II, III, and IV (see below), and agreement to conform to Position I and the justification presented above, we conclude that operation of the plant until that time without implementation of the full requirements of Position I is acceptable.

Position II: The applicant has stated that his design complies with this position. Our review has confirmed this and we find this response acceptable.

Position III: We have concluded that the applicant's justification for exception to this position is inadequate. A program of periodic testing that conforms to the provisions of the staff position will be incorporated into the plant Technical Specifications. This program will confirm the overall operability of the standby power system including its source.

Position IV: We conclude that the commitment to make verification measurements during the preoperational tests is acceptable subject to documentation of the results.

B. Loss of Offsite Power Following Safety Injection Reset

This concern arises from the fact that should a loss of offsite power occur following manual reset of safety injection signal during an accident recovery sequence, many of the engineered safety feature loads would not be automatically sequenced onto the diesel generators. The staff has reviewed this concern and requested that the applicant provide specific operating procedures that set forth the necessary operator action required to reinstate the loads in the event of this occurrence. The applicant has provided and we have accepted an emergency operating procedure that calls for the operator to manually reload the engineered safety feature equipment onto the diesel generators should this event occur.

We questioned whether the design of the manual reset of safety injection at Sequoyah used two push buttons, one for each train, that must be depressed simultaneously to initiate reset. The applicant has shown that each pushbutton independently initiates reset of its own train. There is, therefore, no interdependence between trains for the manual reset. We find this acceptable.

In response to the concern that Sequoyah's design of the safety injection reset not have any modes whereby reset would cause safety-related equipment to change status, the applicant stated that there were no such modes. We find this acceptable.

The Sequoyah design automatically starts the diesel generator on initiation of a safety injection signal thereby avoiding an unnecessary delay in the restoration of emergency cooling water flow to the core should loss of offsite power interrupt it. We find this acceptable.

<u>Air Supply for Auxiliary Feedwater Control Valves</u>

The auxiliary feedwater control valve system located at the hot shutdown panel is air operated. We inquired about the assured availability of the air supply to this system when it is required for accident mitigation. The applicant has stated that an auxiliary seismic Category I air supply is provided as documented in Final Safety Analysis Report Section 10.4.7. We find this acceptable.

<u>Maintenance of Independence Through Common Testing Arrangements</u>

We questioned the possibility of negating the independence of redundant channels through the use of testing arrangements and equipment that is common to more than one redundant channel. We have verified that there are interlocks that prevent placing more than one redundant channel in test at a time and the applicant states that verification of channel trip upon placing a channel in test is an integral part of the test procedure. This is satisfactory; however, we were concerned that single failures occurring in the test apparatus or arrangements could compromise the independence of redundant channels or trains. The applicant provided sufficient information to assure us that no common failure in the testing or interlock arrangements will cause loss of more than one channel. We find this acceptable.

Periodic Test Procedures

The staff was concerned that the design of the safety-related instrumentation and control systems not rely on the use of jumpers, fuse removals, or breaker openings as part of the procedure of any routine periodic test of a safety-related system or component. Further, if such procedures are used, description of how restoration to normal operating status is accomplished and confirmed should be provided. The applicant has indicated that such procedures are used for tests that are made at semiannual or longer intervals. Administrative controls are relied on to confirm return to normal operating status. We have concluded that since the use of tests that require the use of jumpers, fuse removals, or breaker openings as part of the test procedure is limited to these few cases, the use of administrative controls to confirm the return to operable status is acceptable.

7.3.3 Turbine Stop and Downstream Valve Control Circuits

The resolution of the steam line break accident coincident with failure of a main steam isolation valve to close included reliance on prompt closure of the turbine stop valves and other downstream isolation valves to prevent the uncontrolled blowdown of more than one steam generator. We reviewed the controls that initiate and control the closure of these valves to determine their adequacy to serve as backup for the main steam isolation valves. This review confirmed that all the large aperture valves in the main steam system with the exception of the heating steam supply to the moisture separator/reheaters receive signals to close that are related to the steam line break. Isolation of the heating steam supply to the reheaters is not needed because the self regulating action automatically closes the flow path.

Our review also found that the valves are of the good quality normally associated with turbine stop valves which have previously been accepted as backups to main steam isolation valves. They are actuated periodically during normal operation. On these bases we find them acceptable for use as backups to the main steam isolation valves.

7.3.4 Submerged Sensors, Equipment and Actuators

We previously requested that any essential emergency core cooling system valves or operators that could be submerged due to the occurrence of any design basis events or natural phenomena be identified and measures for their protection described. The original response in Amendment 46 stated that there were no essential valves or operators that could be submerged by these events.

We required the applicant to perform the same analyses for all engineered safety features and emergency core cooling system sensors, signal handling equipment, logic and actuators. The applicant responded as follows: (1) for natural phenomena, Final Safety Analysis Report Appendix 2.4A provides a summary description of detailed procedures and measures that will be followed to protect essential safety-related systems in the event of flooding caused by natural phenomena. This appendix states that the safety-related equipment needed to maintain the plant in a safe condition during and after a design basis flood is located either above the flood level, inside a non-flooded structure, or is designed for submerged operation. Unneeded circuits and equipment that are located below the design basis flood level will be de-energized and disconnected as part of the preflood preparation to avoid short circuits and consequent undesirable interactions with essential safety-related equipment. A listing of essential equipment will be given in Section 8.3 of the Final Safety Analysis Report. (2) In case of flooding in emergency core cooling pump rooms and valve galleries due to design basis events, the equipment is protected by breakaway panels that lead to large passive sumps. (3) Protection against flooding caused by firefighting activities is provided by adequate floor drains in areas where fixed sprinklers are installed and in areas where standpipes and manual hose stations are located as required by Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants." (4) The applicant has identified major non-safety-related equipment items and classes of minor non-safety-related equipment that are submerged in the event of a loss-of-coolant accident or steam line break occurring inside the containment.

The applicant has determined that flooding of this equipment will not affect the plant safety. TVA has also determined that no safety-related equipment that is required to remain operable during or after a loss-of-coolant accident or steam line break inside the containment will be submerged.

We have concluded that these measures will give reasonable assurance that sufficent equipment will remain operable in these cases to safely shut the plant down, and are therefore acceptable.

7.3.5 Conclusion

Provided that the outstanding items identified above in Section 7.3.2 are acceptably resolved, we can conclude that the design of the instrumentation and control systems for the engineered safety features and their actuation systems meet the Commission's requirements as described above, and are acceptable.

7.4 Systems Required for Safe Shutdown

We have reviewed the instrumentation and control systems that are provided to effect safe shutdown of the plant. The Sequoyah design provides sufficient instruments and controls located in the auxiliary control room to reach and maintain hot shutdown. This design has transfer switches located in closed, locked, and alarmed rooms that can be used to transfer control of certain essential shutdown systems from the main control room to the auxiliary control room, so that if control equipment located in the main control room were damaged or destroyed, control could still be exercised from the auxiliary control room to safely shut down the plant. We have also determined that suitable procedures have been prepared for modifying the necessary equipment to achieve and maintain cold shutdown from outside the main control room in the event its use was lost. We conclude that these provisions comply with the Commission's requirements as now included in Table 7-1 of the Standard Review Plan, and are acceptable.

7.5 Safety-Related Display Instrumentation

The design of this plant provides systems for displaying safety-related information that enables the operator to manually initiate any necessary safety actions and provides the equipment and circuits to accomplish post-accident monitoring. The instrumentation provided is stated to be similar to that provided for D.C. Cook and Trojan except for the physical configuration. Our review has examined the items presented below.

In the issuing transmittal letter for Regulatory Guide 1.97, Revision 1, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," we stated that the guide would be implemented on all nuclear plant applications and operating reactors with the scope of implementation to be determined by the staff on a case-by-case basis.

In order to develop guidelines to be used by applicants, licensees, and the staff in the implementation of Revision 1 to Regulatory Guide 1.97, the Office of Nuclear Reactor Regulation's Technical Activities Steering Committee has approved generic Task Action Plan A-34, "Instruments for Monitoring Radiation and Process Variables During Accidents." The Task Action Plan calls for developing specific guidance for implementing Revision 1 of Regulatory Guide 1.97. We are currently revising the Task Action Plan to more accurately reflect the process to be utilized and the difficulties that have been encountered in defining specific requirements.

We plan to implement Revision 1 of Regulatory Guide 1.97 on all plants on a schedule and to the degree determined in carrying out Task A-34. Any requirements for additional instrumentation will be considered for incorporation on Sequoyah.

7.5.1 Engineered Safety Feature and Reactor Protection System Status Monitoring System

The status monitoring system uses a computer-based system that provides alarms for each major system when it becomes unavailable. The system further allows the operator to obtain on a catrode ray tube readout a map-type display of the system in question that shows each major component and identifies those responsible for the unavailable status. We find that this system satisfies the recommendations of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," and is acceptable.

7.5.2 Post-Accident Monitoring Separation Criteria

The applicant originally included separation criteria for post-accident monitoring channels that conflicted with the provisions of Branch Technical Position 23 (ICSB), "Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown," which requires all post-accident monitoring channels to be independent as well as redundant. In response to our review, the applicant has committed to providing separation and independence between redundant channels. These revised design criteria required running one set of circuits in rigid conduit with the redundant circuits in armored cable within the main control room panels and the use of barriers between meters or indicators that are not physically separated by six inches or more. We find these revised design criteria acceptable and will report on the implementation of these criteria in a supplement to this report.

7.5.3 Conclusions

On the basis of our review of these items and of previous evaluations of similar systems, and assuming acceptable implementation of item 7.5.2, we conclude that the systems provided to display safety-related information meet the Commission's requirements and are acceptable.

7.6 All Other Systems Required for Safety

These systems are stated to be similar to those of D.C. Cook and Trojan. As a result of our review, the following two items have been satisfactorily resolved.

7.6.1 Residual Heat Removal Isolation Valve Interlocks

The applicant's original design for the residual heat removal suction line isolation valve interlocks was unacceptable because it did not use diversity in the selection of signals or sensors to actuate these interlocks as required by Branch Technical Position 3 (ICSB), "Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System." The applicant has documented in Amendment 48 that diversity is provided by using pressure sensors from different manufacturers in the redundant channels. We conclude that this response meets the requirements of our position and is acceptable.

7.6.2 Level Instrumentation for Essential Raw Cooling Water Intake Structure

The staff questioned the adequacy and redundancy of the level instrumentation for the essential raw cooling water system as installed in the intake structure. This instrumentation shuts down the condenser circulating water pumps in case of a downstream dam failure to help maintain an adequate water level for the essential raw cooling water system. Our review indicated that there is sufficient redundancy to provide adequate assurance that the design will perform its safety function on demand. Further, when the new essential raw cooling water structure is placed in service, the water level will no longer represent a concern since the lower, original river level has been shown to provide an adequate head for the pumps at the new location. We find this acceptable.

7.6.3 Conclusions

On the basis of our review of this application and previous reviews of similar applications, we have concluded that the design of these systems required for safety meets the Commission's requirements as now included in Table 7-1 of the Standard Review Plan, and is acceptable.

7.7 Control Systems Not Required For Safety

The design of the non-safety-related control systems provided for Sequoyah, Units 1 and 2, is similar to those employed for the D.C. Cook and Trojan plants with these exceptions: (1) D.C. Cook has 100 percent load rejection capability but Sequoyah has only 50 percent; (2) Trojan uses a digital rod position indication system whereas Sequoyah and D.C. Cook both use an analog system. This design has previously been shown to meet our requirements. Our review has determined that the design differences noted do not significantly affect the safety of the plants, and we have concluded that the design of these systems is acceptable for Sequoyah.

7.8 Seismic and Environmental Qualification of Balance-of-Plant Safety-Related Equipment

7.8.1 Seismic Qualification of Balance-of-Plant Class 1E Instrumentation, Control and Electrical Equipment

The applicant has indicated in Section 3.10 of the Final Safety Analysis Report that the seismic design and testing of all the balance-of-plant Class IE instrumentation, control and electrical equipment has been accomplished according to the principles contained in IEEE Standard 344-1971 as supplemented by Branch Technical Position 10 (ICSB), "Electrical and Mechanical Equipment Seismic Qualification Program." This equipment is stated to have been qualified by type testing and the results of these tests were stated to indicate that the equipment so tested would perform its safety function during and after a design basis seismic event as required by General Design Criteria 2.

On its site visit, the Seismic Qualification Review Team requested documentation on the seismic qualification of selected items. Information submitted by the applicant permitted us to conclude that function of the following items was acceptable during and after seismic testing: GE voltage relays, 120 volt alternating current vital inverters, motor control centers and switchgear, diesel generator excitation system, 125 volt vital batteries and chargers, radiation monitors, and the 6.9 kilovolt shutdown board. Additional information will be furnished by the applicant on motor operators for outboard containment isolation valves to confirm their seismic qualification. We will review this information and report further in a supplement to this report.

7.8.2 Environmental Qualification of Balance-of-Plant Class 1E Equipment

The applicant has not fully demonstrated that it has a satisfactory program for environmentally qualifying balance-of-plant Class 1E instrumentation, control and electrical equipment. The following two concerns require confirmation.

(1) We required that test plans and digested test results for typical, representative items of balance-of-plant Class 1E equipment be submitted for our review. This documentation was to show the bases for qualification and show that the equipment meets the requirements of these bases.

In response to this request the applicant supplied environmental qualification information based on then current industry standards for (1) safety-related radiation monitors, (2) 6.9 kV shutdown Board, (3) 125 volt direct current vital battery chargers and (4) outboard containment isolation valves. These industry standards (ANSI and NEMA) are based on operating experience and do consider the effects of environment on electrical, control and instrumentation equipment. The standards allow tests based on temperature rise limitations to be performed in the environment existing at the test facility location. Equipment tested in accordance with these temperature rise limitations has demonstrated through operating experience, its satisfactory performance over a normal range of environment. The applicant proposes the use of environmental control systems to establish such normal environments. We do not believe, however, that testing in conformance with the industry standards is sufficient for all areas of the nuclear power plant where electrical, instrumentation, and control equipment may be located. We are concerned that the environmental systems which are used to maintain this normal environment may not have sufficient capacity during extreme outside climatic conditions or may not be available at all times during the plant lifetime. As a result, this equipment may be exposed to a more severe environment than the normal range of environments encompassed by the industry standards.

In response to this concern the applicant has committed to making the environmental control systems that service areas containing safety-related electrical, instrumentation, and control equipment redundant and has stated that these environmental control systems will function to continuously maintain those environments within these qualified ranges. The applicant is also designing and installing an environmental monitoring system that will meet our requirements. We have found this action acceptable and will report further on verification of its implementation in a supplement to this report.

(2) We have reviewed information regarding equipment identification, environmental conditions and environmental design criteria supplied in Final Safety Analysis Report Tables 3.11-1 through 3.11-3 and have found omissions, discrepancies and, in some cases, lack of justification for entries made. We required that these tables be revised to correct and clarify all deficiencies. The applicant has agreed to revise these tables to remove these deficiencies. We accept this commitment and will verify its implementation in a supplement to this report.

A concern arose that the stem-mounted and the gear driven switches used on power-actuated valves for control and for position indication were not qualified for the worst case environment they would be required to operate in, particularly those used inside containment that would have to survive the steam line break or loss-of-coolant accident environments. The applicant received and responded to I&E Bulletin 78-04 and resolved this concern by replacing these switches with NAMCO switches that have been qualified for the worst case environment. We found this acceptable.

The applicant has stated that all balance-of plant valve motor operators for use inside the containment are Limitorque operators of the same type as those supplied and previously qualified through type test for this application by Westinghouse. Tables 3.11-4 and 5 of the Final Safety Analysis Report have been revised to show these operators and their qualification documentation. We have found this acceptable conditioned on our acceptance of the Westinghouse environmental test results for these valves. In the event that the Westinghouse documentation is found unacceptable, the applicant will be required to submit acceptable qualification test results for these motor operators. We will report further on this matter in a supplement to this report.

7.9 Conclusion

Provided that the outstanding items identified in the various sections of this report are acceptably resolved, we can conclude that the design of the instrumentation and control systems meets the Commission's requirements as described above and is therefore acceptable.

8.0 ELECTRICAL POWER SYSTEMS

8.1 General

The evaluation of the adequacy of the design of the safety-related electrical power system of the Sequoyah plant was carried out using as bases (1) the Commission's General Design Criteria 17 and 18, (2) the applicable IEEE standards covering the design of electrical power systems for nuclear fueled generating stations, and (3) the applicable Regulatory Guides and technical positions for light water reactors, are now included in Table 8-1 of the Standard Review Plan.

8.2 Offsite Power System

The Sequoyah Nuclear Plant will be connected to the TVA grid system via four 500 kilovolt and nine 161 kilovolt transmission lines emanating from a 500 kilovolt and 161 kilovolt switchyard. Unit 1 feeds directly into the 500 kilovolt switch-yard, Unit 2 feeds the 161 kilovolt switchyard, and the two yards are tied together through a 1200 megavolt ampere 500-161 kilovolt intertie transformer bank. The switchyards are of the main and transfer bus type with sections arranged in a zig-zag pattern. The preferred power source is from the 161 kilovolt switchyard with backup from the 500 kilovolt yard through the intertie bank. Separate buses and lines feed each common station service transformer to provide the two immediate access sources of offsite power required by General Design Criterion 17. The 161 and 500 kilovolt switchyards and transformer banks are protected by extensive primary and backup relaying and fault isolation arrangements. These protective relaying controls, switchyard breakers, and motor-operated disconnect switches are provided with direct current control power supplied from separate non-class 1E 250 volt batteries.

Each common station service transformer has two 6.9 kilovolt secondary windings. One of these windings serves as the normal feed for a start bus while the other winding serves as an alternate feed for the redundant start bus. The other common station service transformer has its secondary windings connected in opposite order to the start buses. Transfer from normal to alternate feed for each 6.9 kilovolt start bus is automatic on loss of voltage to the bus. The transformers are adequately sized to handle the bus loadings. These two start buses supply 6.9 kilovolt power to the two common boards and reserve power to the four unit boards of each unit. These unit boards normally receive power from the 6.9 kilovolt windings of the two unit station service transformers for each unit. Each unit transformer supplies two unit boards which in turn supply normal and alternate power to their associated shutdown board (engineered safety features bus).

The design of the offsite power system has provision for periodic inspections and testing to demonstrate its capability of fulfilling its safety function as required by General Design Criterion 18 and Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions".

8.2.1 Grid Stability

TVA has conducted grid stability analyses that indicate that for the assumed conditions of (a) loss of the largest generating unit on the system, (b) loss of one 161 kilovolt bus section, (c) loss of the 500 kilovolt unit and two 500 kilovolt lines, (d) loss of a 161 kilovolt line, and (e) 3 phase faults on both 161 and 500 kilovolt lines including stuck breaker conditions, the 161 kilovolt power supply to the engineered safety features buses will remain stable and serviceable. Calculations indicate that the 161 kilovolt system will remain stable even for the loss of both Sequoyah units. We find these analyses acceptable.

8.2.2 Crossovers

A 500 kilovolt line crosses over a 161 kilovolt transmission line coming into the Sequoyah switchyards. Assuming that the 500 kilovolt line drops into the 161 kilovolt line resulting in the loss of both lines, then eight 161 kilovolt and three 500 kilovolt lines into the Sequoyah switchyards are left intact. There is also a crossover where the lines connecting the 161 kilovolt switchyard to common station service transformer "A" passes over interyard buses connecting the 500-161 kilovolt intertie bank and the 161 kilovolt Unit 2 bank to the 161 kilovolt switchyard. If this line should fall onto the two interyard buses, it would result in the loss of common station service transformer "A" and the feed from the Unit 2 turbine generator. In this case, Unit 2 shutdown loads can be supplied from common station service transformer "B" along with Unit 1 loads or Unit 1 can be fed from the 500 kilovolt yard through its main transformer and the unit service transformer. We conclude that neither of the crossovers will prevent supplying both units with sufficient power to achieve safe shutdown, and that the design is acceptable.

8.2.3 Effect of Power Transients on Safety-Related Equipment

This concern is discussed in Section 7.3.2 of this report.

8.2.4 Unit Start Buses

Because of the close proximity to one another of both start buses and the common supporting structures used, we were concerned that failure of a support or damage by external factors, e.g., wind-driven missile or falling structures, would cause loss of both offsite power feeds to the onsite alternating current system. Analyses performed by TVA showed that loss of a single supporting structure would not cause loss of both buses. However, on the first site visit, the staff identified a shield wire support tower that was located in close proximity to the two unit start buses and that could fall in a way that would damage both buses. We required that this tower either be relocated away from the buses to preclude its falling into and damaging the buses, or that it be braced and supported to prevent its failure from damaging these buses. Alternatively, we required that an analysis be made to demonstrate that the failure of the shield wire support tower will not damage the unit start buses to the extent of jeopardizing the flow of offsite power to the onsite system. In Amendment 59 of the Final Safety Analysis Report, the applicant states that the tower will be removed and the shielding supported on building walls. On the basis that the response satisfies our requirements, we find this acceptable.

8.2.5 Uses of Fuses in Switchyard Breaker Control Circuits

During the first site visit, we reviewed the use of fuses in the switchyard breaker control circuits and the method of keeping the operator informed of the status of these fuses and the resulting availability of these breaker control circuits. We found that the switchyard breakers and their controls are arranged in two redundant groups with each such group being fed from an independent 250 volt direct current non-class 1E battery supply. These circuits are fused at the switchyard breaker control panels which are located in the main control room. Indicator lamps on the control panel for these breakers go dark on loss of control power, and annunciation of control power failure is provided. We have concluded that this design is acceptable.

8.2.6 Conclusion

On the basis of our review we conclude that the offiste power system will deliver power to the plant reliably and the system design meets the requirements of General Design Criteria 17 and 18. Therefore, we conclude that the design is acceptable.

8.3 Onsite Power Systems

8.3.1 Alternating Current System

A major design change made since the construction permit review is the addition of a fourth diesel generator to the plant thus providing two independent and redundant diesel generators for each unit. The capacity of each diesel generator was increased to 4000 kilowatts and each generator now powers its own independent engineered safety features bus.

The engineered safety feature and other essential safety-related loads are divided between two 6.9 kilovolt buses per unit. Thus all engineered safety features loads are now arranged in a redundant two-train system with each train completely independent. The operability of any one of these buses and its attendant distribution system will provide power to operate sufficient engineered safety features to safely shut the plant down or mitigate the effects of an accident. The two sources of offsite power for these buses are described in Section 8.2 above.

These diesel generators are started by an undervoltage signal from the bus, by a safety injection signal, or by manual initiation. The generators are located in separate rooms of the diesel generator building which is a seismic Category I structure. Each diesel is started by redundant air-starting motors supplied from redundant air supplies. The diesel generator control and instrument power is supplied by a Class IE battery and charger system for each machine. Each machine is supplied from a day tank which in turn is supplied from its own underground tank which contains a seven-day fuel supply. Additional fuel oil supply is contained in two large above-ground tanks located on the site.

The 120-volt alternating current vital instrumentation bus system consists of four separate and electrically independent buses and distribution systems that supply power to four redundant instrumentation, control, protection and annunciation load groups. Each of the buses is served by a separate inverter that is supplied by either (a) one of the four 125-volt direct current distribution systems, or (b) one of the 480-volt engineered safety features buses. The inverters are each normally fed from the 480-volt supply through individual direct current power supplies and upon loss of the 480-volt alternating current supply, transfer automatically to its appropriate 125-volt direct current battery bus. Additionally 120-volt alternating current is available through a manual transfer to supply the vital bus in event the inverter is out of service.

The Sequoyah diesel generators use a new, tandem engine design that had not been previously qualified for use as a standby power source in nuclear generating stations. TVA had therefore proposed conducting qualification tests on these machines.

The staff had reviewed and approved this proposed diesel generator qualification test program as presented in Appendix 8.3A to the Final Safety Analysis Report. However, we questioned the provision of means for determining that each engine of the pair was carrying its share of the load. Information was subsequently reviewed that showed that the position of the actuator lever for the fuel rack on each engine of the pair is monitored and an alarm actuated if the difference in position between engines exceeds three degrees. Also the exhaust gas temperature between engines and between each cylinder is monitored and alarms are actuated for differences of 220 degrees Fahrenheit or more. Thus, governor or injector malfunctions can be detected. We consider this to be satisfactory.

The applicant has provided documentation showing the performance of 306 start and load cycles with no failures. We have reviewed this test data and found that seven of these test cycles were voided because of operator error or due to failure of test stand components. No failed test cycles occurred. In each test cycle, the generator was loaded to 56 percent of its capacity in a single step within 15 seconds after the time the start signal was applied. The observed fluctuations in voltage, frequency, current, and engine speed stabilized within the 15 second interval.

The diesel generators successfully passed a margin test where, for a period of 72 hours, the unit was loaded to 118 percent of capacity. Also every tenth start and load test was a full-load test.

We have concluded that this program has demonstrated the margin and reliability of these units and, hence, their suitability for use as standby power sources for nuclear generating stations.

The applicant has provided several protective trip functions for the diesel generators. However, in accordance with the recommendations of Branch Technical Position 17 (ICSB), "Diesel-Generator Protective Trip Circuit Bypasses," all such protective trips, with the exception of the generator differential protection and the diesel engine overspeed protection, are bypassed by the emergency start signal. We find this to be acceptable.

On the basis of our review, we conclude that the design of the onsite alternating current power system meets the requirements of General Design Criteria 17 and 18 and IEEE Standards 308-1971, and is acceptable.

8.3.2 Direct Current Power System

A major design change in the direct current power supply system since the construction permit review is the replacement of the two 250-volt battery systems by four 125-volt battery systems that are shared between the two nuclear units. Battery boards I through IV are also shared by the two units.

The sharing is as follows: Battery I through Battery Board I supplies 125-volt direct current to inverters 1-I of Unit-1 and 2-I of Unit 2 which in turn supply power to Unit 1 120-volt alternating current vital instrument board 1-I and to Unit 2 120-volt alternating current vital instrument board 2-I. Batteries II through IV supply their respective inverters in a similar manner. 125 volt direct current loads for Unit 1, Trains A and B, are supplied from Batteries I and III respectively. Similar loads for Unit 2, Trains A and B, are supplied from Batteries II and IV respectively. Such sharing of vital batteries between units is permitted by Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants" for plants for which construction permit application was made prior to June 1, 1973. Sequoyah falls into this category; therefore, this sharing of batteries is permissible. On the basis of adequate battery capacity and the ability to automatically isolate faulted sections of the system as discussed below, we find such sharing acceptable.

The 125 volt direct current systems provide uninterruptible power for control, instrumentation, annunciation and emergency lighting. Each 125 volt battery has a charger associated with it. Each pair of batteries, e.g., I and II, has a spare charger assigned that can be manually connected to either (but not both) battery bus of the pair. The battery chargers are supplied with standby onsite alternating current power in event of loss of station and offsite power. Each charger is capable of maintaining the battery at full charge and of recharging the battery from its discharged state in approximately twelve hours while also supplying accident loads.

Each 125 volt battery is able to supply its connected loads for a minimum of two hours in the event of loss of all alternating current power when starting from a fully charged state. Each battery is located in a separate room with its individual heating and ventilating system. Its distribution system is supplied with fuses and circuit breakers that automatically isolate faulted sections from the remainder of the system.

Two independent 250 volt direct current non-safety related batteries have been installed to provide reliable power for loads such as turbine auxiliaries, computer, switchyard control, and relaying equipment.

On the first site visit, we found that the normal and auxiliary 125 volt direct current control power feeds for the 6900 volt shutdown boards came from different battery boards but had not been properly separated at the transfer switches. This was pointed out to the applicant. This situation was corrected and we conclude that the separation is now acceptable.

During our first site visit, we found that no monitor for hydrogen concentration was provided because each battery room exhaust fan runs continuously, fan failure is annunciated, and an alternate fan starts automatically. We concluded that this arrangement for keeping the hydrogen concentration below the flammable limit removes the need for monitoring and is acceptable.

Our review determined that the applicant had not provided performance or service test requirements and criteria for the Class 1E batteries. As a result, the applicant has agreed to load test these batteries according to the requirements of IEEE Standard 450-1972. He has further stated that, in the performance of these tests, the actual loads supplied by the battery under the worst-case discharge conditions will be used. We found this acceptable.

The applicant had initially provided no information as to whether or not the 125-volt direct current vital batteries remained connected to the direct current supply buses during the equalizing charges. Further, no information was provided regarding the qualification of the direct current equipment and loads to operate at the equalizing charge voltage if the batteries remain connected during the charge. In response to our request, we were informed that the batteries remain connected to the vital bus during the equalizing charge, that the equalizing charge voltage is 140 volt direct current and all direct current equipment has been qualified to operate at this voltage. We found this acceptable.

On the basis of our review, we conclude that the design of the onsite direct current power systems meets the requirements of General Design Criteria 17 and 18, and IEEE Std 308-1971, and therefore is acceptable.

8.4 Environmental Qualification of Electrical Equipment

The majority of the staff concerns with the environmental qualification for Class 1E electrical equipment has been covered in Section 7.8.2 of this report. As a result of our review, the following additional items have been identified.

8.4.1 Nuclear Instrument Penetrations

A revision of subparagraph 8.3.1.2.3 of the Final Safety Analysis Report had made it appear that the nuclear instrument triaxial penetration assemblies did not need to be qualified for hostile environments. We took the position that all penetration assemblies must be qualified to maintain their mechanical and electrical integrity when subject to the worst-case environment associated with design basis accidents. The applicant responded by amending the Final Safety Analysis Report to state that these penetrations are qualified. We find this acceptable.

8.4.2 Radiation Damage Testing

The applicant's position on testing to determine dose rate dependent radiation damage effects for balance-of-plant materials and equipment located in the containment stated that the materials were exposed at a dose rate of 10^6 rads per hour and that this dose rate was comparable to the maximum expected during a loss-of-coolant accident. However, calculations for pressurized water reactors have indicated that peak dose rates in containment of 4 to 5 x 10^6 rads per hour may be experienced during design basis accidents. In response to our position, the applicant stated that there were no significant radiation damage mechanisms that had thresholds in the region of 10^6 to 10^7 rads per hour for materials and equipment in the balance-of-plant scope. We agree and found this acceptable.

8.4.3 Class 1E Cabling to Outlying Structures

The staff requested that the applicant describe the facilities used for routing Class 1E electric and control cables between the main reactor building and remote installations such as the diesel generator building and the essential raw cooling water pumping station. In response the applicant stated that these circuits were run in underground Category I duct banks and cable vaults, and described various qualification methods that were used to test the cables and splices. On the first site visit, we viewed the duct banks, observed that splices were made in manholes, learned that the duct banks could be sealed, and observed the splicing methods used. The splicing methods used produce splices that have a history of being reliable under wet or dry conditions. We have found this implementation acceptable.

8.4.4 Protection of Containment Penetrations Against Physical Damage from Electrical Faults

We questioned whether the penetrations are designed to carry without damage the maximum possible fault currents, or, if circuit breakers are relied upon, whether the breakers are both redundant and qualified as Class 1E. The applicant submitted additional representative qualification and other supporting data regarding these penetrations.

This information shows that, for large loads such as reactor coolant pumps, the penetrations are protected by two circuit breakers in series in each line; for small loads, they are protected by a breaker and a fuse type device in series in each line. For non-Class 1E circuits, these protective devices are not Class 1E; however, they are redundant and can meet the single failure criterion. The penetrations have been designed to carry without damage the maximum available fault current for a limited time. We have concluded that this combination of design features meets the recommendations of Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants," and is therefore acceptable.

The applicant states that short-circuit tests were conducted on representative samples of each type of penetration. The short circuit test was done prior to the leakage testing to confirm the penetration's ability to withstand the mechanical stress imposed by the subtransient currents. Thermal heating of the conductors by sustained overcurrents has been determined by the manufacturer to be the worst case event and therefore formed the basis for the time-current damage curves presented in the Final Safety Analysis Report. We have reviewed the short-circuit test description and results for the 6.9-kilovolt and low voltage penetrations and found them acceptable.

We questioned the lack of seismic qualification and the location in non-seismic Category I structures of circuit breakers used to protect penetrations carrying power into the containment for non-Class 1E loads located in the containment. Seismically induced damage to such wiring and equipment could result in faults that could damage the containment penetrations if the breakers failed to open on demand following the seismic event. The applicant has stated that all wiring and equipment located inside the containment is seismically qualified and supported and will remain operable following a design basis seismic event. This response satisfies the concern and is acceptable.

We questioned the qualification of connectors and terminal blocks used to connect circuits and equipment located inside the containment. The applicant responded as follows. (1) Only the neutron detectors use connectors. The connectors are not qualified for steam line break conditions because they are not required to function in that environment. (2) Terminal blocks are qualified as part of the equipment in which they are included. (3) Circuits other than the neutron detectors are connected using splices that have been qualified for the steam line break environment. We find this response acceptable.

We have reviewed documentation describing the environmental qualification of the containment electrical penetrations. We had found some instances where necessary information had not been presented. These were discussed with the applicant and the results are reported below.

(1) The applicant submitted reports showing that the low voltage, teflon-epoxy seal penetrations meet the worst case containment accident environmental requirements, with the exception of radiation which is discussed in (2) below.

(2) TVA submitted documentation on the Brunswick Nuclear Plant penetrations showing that the three classes of electrical and instrumentation penetrations were qualified for the postulated worst case containment radiation environment. We reviewed this documentation and determined that the Sequoyah penetrations are of the same design and use the same materials as those on Brunswick, except for the connecting cables on the nuclear instrument penetrations. We have determined that the Sequoyah cables have also been qualified for the worst case radiation environment. Based on the above, we find this response acceptable.

(3) The circuit breakers that protect the containment penetrations against unacceptable overcurrents must be periodically tested to verify their continued ability to provide this protection. These periodic tests shall include the breakers, their sensors, and their trip relays in an overlapping set of tests of the system to ensure that this total protective capability has not been lost or become degraded. We will incorporate in the technical specifications requirements for such periodic testing that includes simulating overcurrent conditions through the breaker's fault-sensing devices.

Based on all the above, we conclude that the containment electrical and instrumentation penetrations are capable of performing their safety function in the environment and under the operating conditions likely to be encountered, and are therefore acceptable.

8.4.5 Conclusion

Assuming the satisfactory resolution of the concerns stated in Section 7.8.2 of this report, we can conclude that the environmental qualification of the electrical equipment for this plant meets the Commission's requirements and is acceptable.

8.5 Physical and Electrical Independence of Electrical Equipment and Circuits

The provisions for physical independence made in the applicant's design of the plant electrical system is detailed in Final Safety Analysis Report Section 8.3.1.4 for the alternating current onsite system and 8.3.2.4 for the direct current system. The stated criteria for separation and independence are essentially identical to those set forth in Regulatory Guide 1.75, "Physical Independence of Electric Systems," even though the plant was laid out prior to the development of that Regulatory Guide.

The identification of these circuits and equipment is discussed in Final Safety Analysis Report Sections 8.3.1.4.5 and 8.3.1.5 for the onsite alternating current system and in Section 8.3.2.5 for the onsite direct current system. Equipment, cables and trays are identified by assigning to each item a coded alphanumeric label and suffix. Classification by train and channel is indicated through a color coding scheme.

8.5.1 Power Cables In Cable Spreading Area

On the first site visit, we verified that, in accordance with our criteria, no alternating current power cables carrying voltages of 600 volts or more were run through the cable spreading area or control room. However, we did find an exception to the criteria in that many switchyard control cables operating at 250-volt direct current originate in the control room and run through the cable spreading area. These cables were separated from other redundant cables by grouping and by being run in conduit through the area. We have concluded that this exception is acceptable.

8.5.2 Piping in Class 1E Battery, Switchgear and Equipment Rooms

We had questioned the piping that runs through Class 1E battery, switchgear and equipment rooms, particularly its divisional assignment and whether it was of seismic Category I design. An example was a line running through the switchgear rooms directly behind the shutdown boards of both Unit 1 redundant trains. This was identified as a seismic Category I air line that was properly supported and analyzed. Fire protection piping in this area, not yet installed at the time of the site visit, is of the dry-pipe type, will be seismically supported, and has been analyzed in both the moderate energy line break and fire hazards analyses. Based on the above, we find the piping in these rooms to be acceptable.

8.5.3 Implementation of Separation Criteria

On the initial site visit, we questioned the implementation of separation criteria in instances where non-divisional cable trays were routed between redundant divisional cable tray stacks, unsupported cable bundles run vertically between cable trays in the same stack, and trays of different divisions cross. The applicant has (1) committed to coat all cabling in the spreading room with fire retardent coating, (2) provided adequate sprinkler coverage, (3) supported and will encase the vertical runs in fire retardent coating, and (4) committed to provide barriers at interdivisional crossings. We conclude that these provisions adequately address our concerns and are acceptable.

8.5.4 Circuit Breakers as Isolation Devices

We confirmed that circuit breakers actuated by fault current are used as isolation devices in this plant. This usage conflicts with the current recommendations of Regulatory Guide 1.75, "Physical Independence of Electric Systems." However, because this design preceded the issuance of Regulatory Guide 1.75, this use is permitted. In addition, the breakers are qualified as Class 1E equipment and will be periodically tested to verify that the originally designed coordination continues to be available. We conclude that this is acceptable, and requirements for such testing will be included in the technical specifications.

8.5.5 Separation of Alternating and Direct Current Instrument Power

The independence of the normal and alternate 125 volt direct current supplies to the 120 volt alternating current vital inverters and the normal and alternate feeds to the 120 volt alternating current vital instrument power board is maintained through the use of appropriate barriers. The breakers and transfer switches are also interlocked to prevent redundant or independent sources from being inadvertently paralleled. We find this acceptable.

8.5.6 Conclusion

We conclude that the applicant's implementation of his design for independence of the safety-related electrical, control and intrumentation systems meet the Commission's requirements as described above and are acceptable.

9.0 AUXILIARY SYSTEMS

In the course of our review, we have focused our attention on the design of the auxiliary systems, including any safety-related objectives of the respective systems, and the manner in which these objectives are achieved.

The systems necessary to assure safe handling of fuel and adequate cooling of the spent fuel include the new and spent fuel storage systems, the fuel pool cooling system, and the fuel handling system.

The auxiliary systems necessary to ensure safe plant shutdown include portions of the chemical and volume control system, the essential raw cooling water system, the component cooling water system, the ultimate heat sink, the control room ventilation system, portions of the auxiliary building ventilation system, the emergency diesel generator auxiliary systems and the auxiliary air compressor system, and the fire protection system.

We have reviewed the equipment and floor drainage system whose failure would not prevent safe shutdown but could be a potential source of a radiological release to the environment.

We have also reviewed other auxiliary systems and those non-seismically designed systems whose failure would neither prevent safe shutdown nor result in potential radioactive releases. These include the demineralized water makeup system, the raw water cooling system, the potable and sanitary water system, the station control and service compressed air system, and the turbine building ventilation system.

Where systems or portions of systems are to be shared by both units of each facility, the applicant has stated that such sharing will not impair their ability to perform their safety functions. We have reviewed those systems and components to be shared and find that the design meets the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," and is acceptable.

From our review of the proposed design of the auxiliary systems for Sequoyah Units 1 and 2, we find they are similar in design and function to other pressurized water reactor facilities that have been previously reviewed, approved and are operating.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

In Amendment 40 to the Final Safety Analysis Report, the applicant proposed to increase the new fuel storage capacity from 129 fuel assemblies to 180 fuel assemblies.

The storage facility is shared between the two units. The new fuel will be stored dry. The new fuel storage racks are constructed so that it is impossible to insert fuel assemblies except in prescribed locations having a minimum center-to-center spacing of 21 inches in both directions. The spacing is sufficient to assure an effective multiplication factor $\leq$ 0.95 even if immersed in unborated water, or effective multiplication factor $\leq$ 0.98 with optimum moderator. The racks are designed to seismic Category I requirements and are capable of withstanding loads imposed by the dead load of the fuel assemblies, safe shutdown earthquake, and uplifting force of 3000 pounds without causing the effective multiplication factor to increase above 0.98.

We have reviewed the design of the new fuel storage facility and conclude that it meets the requirements of General Design Criterion 62 as regards prevention of criticality, General Design Criterion 5 as regards to shared facilities, and the positions of Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," including seismic design and missile protection guidelines, and is therefore acceptable.

9.1.2 Spent Fuel Storage

In Amendment 40, the applicant proposed to increase the spent fuel storage capacity from 322 fuel assemblies to 800 fuel assemblies (approximately four and one-third cores). The spent fuel pool is a shared facility and is designed to seismic Category I requirements. The spent fuel storage racks have a minimum spacing of 13 inches between fuel assemblies. The spacing is such that effective multiplication factor is below 0.95 if the racks are filled with fuel assemblies having the highest anticipated enrichment even when flooded with unborated water. The racks are designed so that fuel can be inserted only in the designated spaces, and include provisions for storage of spent control rods and burnable poison rods. The racks are designed to withstand loads imposed by the dead load of the fuel assemblies, loads resulting from the impact and handling of fuel assemblies, safe shutdown earthquake, and the maximum uplift force from the spent fuel bridge hoist. A minimum of 24 feet of water above the fuel assemblies can be maintained at all times. This water level is adequate for shielding purposes.

The facility is designed to prevent the cask handling crane from traveling over, or in the vicinity of, the spent fuel storage areas, thereby precluding damage to the stored fuel due to a dropped cask.

Based on our review, we conclude that the design of the spent fuel storage facilities is in conformance with the requirements of General Design Criterion 62, as regards prevention of criticality, General Design Criterion 5 as regards to shared facilities, and the positions of Regulatory Guides 1.29, "Seismic Design Classification," and 1.13, including the positions on seismic and missile protection design and compatability with the handling of the fuel cask in the fuel pool areas, and are therefore acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the spent fuel elements stored in the fuel pool. A secondary function is to clarify and purify the spent fuel pool water, the transfer canal water, and the refueling water. Two cooling trains are provided, each containing a pump and heat exchanger. A spare pump capable of operation in either train is also provided. The system piping and components are designed to seismic Category I equipments. When the spent fuel pit contains the spent fuel resulting from back-to-back refueling of both units, the system can maintain the spent fuel pit water temperature at or below 120 degrees Fahrenheit when two pumps and two heat exchangers are in operation. If it is necessary to remove a complete core from one unit subsequent to the back-to-back refueling of both units, a 49.1 day delay would be required before placing the spent fuel assemblies of the full core in the spent fuel pit. With this delay, the spent fuel pool cooling system can maintain the spent fuel pit water at or below 150 degrees Fahrenheit with only one cooling train in operation. These temperature maintenance capabilities of the spent fuel pool cooling system are within the limits stated in Standard Review Plan Section 9.1.3 and are acceptable.

Normal makeup water to the fuel pool is provided from the demineralized water system. A backup water source for filling the fuel pool is available from the seismic Category I fire protection system. Alarms for high and low water level and high temperature in the fuel pool are provided in the control room to alert the operator. The piping layout and the use of anti-siphon holes assure that the fuel pool cannot be drained below an unacceptable level.

We reviewed the adequacy of the applicant's design for the fuel pool cooling and purification systems necessary for continuous cooling during normal, abnormal, and accident conditions. We conclude that the design is in conformance with General Design Criterion 61, General Design Criterion 5, and the positions of Regulatory Guides 1.13 and 1.29, including the positions on seismic design, missile protection, and availability of assured makeup water systems, and are therefore acceptable.

9.1.4 Fuel Handling System

The fuel handling system is designed to safely handle and store fuel assemblies from receipt of new fuel to shipping of spent fuel. The system is designed to conduct all spent fuel transfer and storage operations underwater to limit radiation dose levels. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper under the safe shutdown earthquake. The manipulator crane will remove each spent fuel assembly from the core and load it into a transfer carriage, which transports the spent fuel assembly to the fuel pool via the refueling canal and the transfer tube. When the spent fuel is ready to be shipped offsite the spent fuel cask will be placed in the cask loading area, and the spent fuel assemblies will be removed from the fuel pool and placed in the spent fuel cask by the spent fuel bridge hoist. After installation of the cask head, the cask will be removed from the cask storage area and placed in the cask decontamination room for washdown. The cask is then placed on the shipping conveyance and moved out of the building.

The spent fuel cask is handled with the auxiliary building crane which is shared by both units. The main hoist is rated at 125 tons. Movements of the bridge and trolley in the vicinity of the spent fuel pool are restricted by limit switches in order to prevent the crane from transporting a load over the irradiated fuel in the pool. Trolley movement is also restricted by mechanical stops. The separating wall between the fuel cask loading pit and the spent fuel storage area is designed to withstand loads by a cask dropped in a tipped position and, therefore, this event will have no subsequent adverse effects on the spent fuel storage pool.

We have reviewed the adequacy of the applicant's design necessary for safe operation of the fuel handling system during normal, abnormal and accident conditions. We conclude that the design is in conformance with the positions of Regulatory Guide 1.13, including protection of spent fuel storage facility from the impact of heavy loads carried by overhead cranes, and is therefore acceptable.

9.2 Water Systems

9.2.1 Component Cooling System

The component cooling system is designed to remove residual and sensible heat from the reactor coolant system, residual heat removal system, chemical and volume control system, waste disposal system, sampling system, safety injection system, and the spent fuel pool cooling system.

Cooling water to the above systems will be provided by the component cooling system during all modes of plant operation and shutdown. The heat energy transferred to the component cooling system is then dissipated to the river or atmosphere via the essential raw cooling water system which is discussed below. The system consists of five component cooling system pumps, four thermal barrier booster pumps, three heat exchangers and two surge tanks serving both units. Each unit may be aligned with two completely independent cooling trains from the control room. Two component cooling system pumps and one heat exchanger are included in each cooling train having the capacity to provide the maximum cooling water requirement for safe shutdown of both units. The spare component heat exchanger and component cooling system pump are aligned to the accident unit during the safety injection phase of a loss-of-coolant accident. Since portions of the component cooling system are required for post accident removal of decay heat from the reactor, these portions are considered engineered safety features and are, therefore, designed to seismic Category I requirements. In addition, these equipment trains are sufficiently independent to ensure the availability of at least one train at any time. Should a single failure result in the loss of a train, the other train is available for handling all required heat loads.

Based on our review, we conclude that the component cooling system design is in conformance with the requirements of General Design Criterion 44 regarding the single failure criterion and the ability to transfer heat from safety related components to the ultimate heat sink via the essential raw cooling water system. It is also in conformance with the requirements of General Design Criteria 45 and 46 regarding the system design for periodic tests and inspections, including functional testing and confirmation of heat transfer capabilities. We conclude that the system is acceptable.

9.2.2 Essential Raw Cooling Water

The essential raw cooling water system supplies cooling water to meet plant cooling requirements during normal operations or under accident conditions. Prior to operation of Unit 2, the essential raw cooling water system will consist of four essential raw cooling water pumps, four mechanical draft cooling towers and two auxiliary essential raw cooling water pumps.

The essential raw cooling water system piping is arranged in two headers and fitted with isolation valves such that a failure in either header will not jeopardize the safety function of the system. The operation of two pumps is sufficient to supply all cooling water requirements for any operation or accident condition of Unit 1.

In the event of flood above the elevation where the four essential raw cooling water pumps are mounted, i.e., elevation 705.5, or upon loss of the downstream dam, the auxiliary essential raw cooling water system would be put into operation. The auxiliary essential raw cooling water pumps are arranged so that either pump may serve in conjunction with any three of the four mechanical draft cooling towers. Three of the four cooling towers will be sufficient to provide the required cooling water for safe plant shutdown.

Both the essential raw cooling water system and the auxiliary essential raw cooling water system are designed to seismic Category I requirements. In addition, means of protection against tornado missiles will be provided for the essential raw cooling water system (see Section 3.5.1 of this report). The auxiliary essential raw cooling water towers are not designed to withstand tornado effects since they are not required for any plant conditions associated with this event.

Prior to fuel loading of Unit 2, a separate essential raw cooling water pumping station, designed to seismic Category I requirements, is to be constructed and placed in operation serving both units. The new essential raw cooling water pumping station is designed and located to eliminate any dependence upon the intake forebay, the portion of the essential raw cooling water system located in the forebay pumping station, and the entire auxiliary essential raw cooling water system. The new two-unit essential raw cooling water system will consist of eight essential raw cooling water pumps connecting to two supply headers arranged and fitted with isolation valves to form two completely independent cooling trains. Thus the system safety functions will not be compromised in the event of pipe failure in either train. The new essential raw cooling water pump sump is at elevation 625 feet above mean sea level which is 58.0 feet below minimum summer river elevation and 50.0 feet below minimum winter river elevation, and 11.0 feet below elevation 636 feet above mean sea level which is the minimum possible elevation of the river. Therefore, sufficient pump submergence is always available for the essential raw cooling water pumps. The minimum combined requirement for one "accident" unit and one "non-accident" unit, or two "non-accident" units are met by only two pumps. Since the shared essential raw cooling water headers each have two pumps that are operable and each is assigned to an emergency diesel generator on loss of offsite power, total loss of either header will not prevent safe shutdown of either unit under any credible plant condition.

The essential raw cooling water pumps were originally protected against tornado missiles only in the horizontal direction. We found the proposed design unacceptable. In Amendment 36, the applicant submitted a revised tornado missile protection barrier design for the essential raw cooling water pumps. This barrier will guard againts tornado missiles from all directions, and as noted in Section 3.5.1, we found this design acceptable.

Based on our review, we conclude that the design criteria and bases for the emergency raw cooling water system meet the requirements of General Design Criterion 2 in regard to protection against natural phenomena, General Design Criterion 44 regarding their ability to transfer heat from safety related components to the ultimate heat sink, and General Design Criteria 45 and 46 regarding testing and inspection. We therefore conclude that the design of the essential raw cooling water system is acceptable.

9.2.3 Ultimate Heat Sink

The ultimate heat sink used in the initial operation of Sequoyah Nuclear Plant will be modified early in plant life when the new independent pumping station is made effective prior to operation of Unit 2. Initially, the ultimate heat sink will consist of the Tennessee River as the water source including the complex of TVA-controlled dams upstream of the plant intake, TVA's Chickamauga Dam, the pumping station intake forebay, and the auxiliary emergency raw cooling water cooling towers, pumps and basins. Additional information is contained in Section 2.1 of this report.

After the new essential raw cooling water station begins operation, the composition of the ultimate heat sink will no longer include the auxiliary essential raw cooling water system or the original pumping station forebay. For the interim operation, i.e., preceeding the operation of the new pumping station, cooling water from Chickamauga Reservoir flows under the intake skimmer wall into the intake forebay. Water is pumped to the plant from the forebay via the essential raw cooling water pumps and then discharged to the discharge pond. Under extreme flood condition, that is flooding above elevation 705.5 mean sea level, or loss of the downstream dam, the auxiliary essential raw cooling water portion of the essential raw cooling water system is utilized. This closed-cycle water circulating operation rejects heat to the atmosphere via the mechanical draft cooling towers. In this mode, water is circulated by the auxiliary essential raw cooling water pumps through the appropriate plant heat exchangers, the mechanical draft cooling towers, and tower basins.

The seismic Category I forebay pool is sized to store 2,540,000 gallons of water, and the cooling tower basins provide an additional 118,000 gallons of water. The amount of water will be sufficient for this mode of operation for a period of six days without makeup water. Beyond this six-day period, makeup water for the auxiliary essential raw cooling water towers will be required. The applicant indicates that portable diesel-engine-driven pumps and a portable piping system will be provided for this purpose. See Section 2.4.3 for further description of this feature. The applicant has provided a summary of procedures for the deployment, installation, and operation of the portable makeup system which we find acceptable. The applicant is committed to complete and commence operation of the new emergency raw cooling water pumping station prior to fuel loading of Unit 2 (approximately 8 months after fuel loading of Unit 1). We consider use of a portable makeup system prior to the initial fuel loading of Unit 2 to be acceptable because the probability of requiring the use of the portable system before fuel loading of Unit 2 is small, and six days is adequate time for operator action in accordance with prepared procedures.

With the new essential raw cooling water pumping station in operation, the water intake to the pumping station will be at an elevation below the Tennessee River bed elevation and the area outside the intake will be dredged to form a channel that provides free access to the river. The channel will be monitored and dredged as required to maintain a clear passage to the river. Therefore, adequate water will be available to the essential raw cooling water pumps for plant safe shutdown purposes at all times. The ultimate heat sink will be designed to withstand a 95 miles per hour basic wind or the most severe tornado including the associated missile spectrum. (See Section 9.2.2 above.) In addition, the ultimate heat sink is also designed to accommodate the most severe combination of events considered credible to occur such as simultaneous occurrence of safe shutdown earthquake, a loss-of-coolant accident in one unit and shutdown of the other, loss of offsite power, and loss of upstream and/or downstream dams either individually or concurrently.

The applicant has demonstrated to our satisfaction that the ultimate heat sink will be designed in accordance with Position 2 of Regulatory Guide 1.27, "Ultimate Heat Sink," namely, the capability of the system to withstand the most severe natural phenomena expected and a single failure of man-made structural features. Based on our review, we conclude that the ultimate heat sink design is compatible with the positions of Regulatory Guide 1.27 and are therefore acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

The compressed air system is shared by both units and the system is divided into three subsystems: the station control and service air system and two auxiliary control air systems for emergency use. The two auxiliary control air systems are designed to seismic Category I requirements and powered from separate emergency electrical power sources, and are completely separated and independent from each other. Therefore a single failure cannot render both trains inoperable. The station control and service air system has no safety-related requirement. It normally supplies air to both trains of the auxiliary control air system but is automatically disconnected when the output pressure falls below 80 pounds per square inch gauge. At this compressed air system pressure, the auxiliary compressors will start automatically to supply compressed air to safety-related equipment.

We have reviewed the compressed air system and find that the design for those safety-related portions of the system is in conformance with Regulatory Guide 1.26 and 1.29 with regard to Quality Group and seismic classification. We conclude that the system is adequately designed to protect the safety function of plant safety related systems, and is therefore acceptable.

9.3.2 Equipment and Floor Drainage System

The equipment and floor drainage system is designed to handle nontritiated liquids separate from tritiated liquids as discussed in Section 11 of this report.

The liquid drains are segregated into three basic systems. The first system collects all tritiated water. This system is further divided into aerated liquids which are collected in the tritiated drain collector tank and deaerated liquids which are collected from reactor coolant drain or the chemical and volume control system holdup tank. The second system collects nontritiated water and normally nonchromated water which is collected in the the floor drain collector tank, or laundry and hot shower tank. The third system provides for returning all chromated water from equipment drains to the component cooling water surge tank.

The equipment and floor drain system is a series of sumps located in the auxiliary building which drains to a sump which is equipped with two sump pumps with separated suctions and high level alarms. Therefore, failure of any one pump will not result in flooding due to equipment and floor drainage system overflow.

Based on our review, we find that the equipment and floor drainage system design is adequate to preclude any damage to safety related systems or components due to system failure and to prevent inadvertent release of radioactive liquids to the environment due to piping or tank failure. We conclude that the system is acceptable.

9.3.3 Chemical and Volume Control System

The chemical and volume control system is designed to control and maintain reactor coolant inventory and also to control the boron neutron absorber concentration in the reactor coolant through the process of makeup and letdown. The chemical and volume control system also provides seal-water injection flow to the reactor coolant pumps, controls the primary water chemistry by ion exchange and chemical addition, and processes the effluent reactor coolant to recover the chemical neutron absorber and makeup water. The chemical and volume control system charging pumps and associated valves and piping are utilized for high pressure injection of borated water for emergency cooling in the event of a postulated accident. A boron recovery system collects borated water that results from plant operation for both units. Such plant operations include dilution of reactor coolant to compensate for core burnup, load follow, hot shutdown and startup, cold shutdown and startup, and refueling shutdown and startup. The reactor makeup control system is used to vary the reactor coolant boron concentration to compensate for xenon transients occurring when reactor power level is changed.

The safety-related portion of the chemical and volume control system is designed to seismic Category I requirements. The chemical and volume control system is capable of borating the reactor through either one of two flow paths and from either one of two boric acid sources. The amount of boric acid stored in the chemical and volume control system exceeds that required to borate the reactor coolant system to cold shutdown concentration, assuming that the control assembly with the highest reactivity worth is stuck in its fully withdrawn position. This amount also exceeds that required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

All portions of the chemical and volume control system that contain boric acid solution are located in a heated area or heat traced to assure the solubility of boric acid in water.

Based on the review of the applicant's design and safety classification for the chemical and volume control system, and the requirements for system performance of necessary functions during normal, abnormal, and accident conditions, the staff has determined that the design of the chemical and volume control system and supporting systems is in conformance with the Commission's regulations as set forth in General Design Criteria 2, 4, and 33, and meets the guidelines contained in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Guide 1.29, "Seismic Design Classification," and therefore is acceptable.

9.4 Heating, Ventilating and Air Conditioning

9.4.1 Control Building

The control building heating, ventilating, air-conditioning and air cleanup systems are designed to maintain the temperature and humidity conditions throughout the building to assure the protection and operation of plant controls and the safe, uninterrupted occupancy of the main control room during an accident and the subsequent recovery period. The control building air-conditioning system is an engineered safety feature and is designed to Quality Group C and seismic Category I requirements. Each pair of full-capacity air compressors and air handling units is served from separate trains of emergency electric power system and from a coordinated separated loop of the essential raw cooling water system. The building pressurized air supply system is capable of maintaining the control building at a positive pressure relative to outdoor or the adjoining buildings at all times. The control building outside air intakes are provided with radiation monitors and smoke detectors that indicate and annunciate in the control room.

Isolation of the main control room occurs automatically upon the actuation of a safety injection signal from either unit or upon indication of high radiation, temperature, chlorine, or smoke concentration in the outside air supply stream to the building.

Upon the signal for main control room isolation, the emergency air cleanup fans will be started to recirculate a portion of the control room air-conditioning system return air through a cleanup train composed of high efficiency particulate air filters and charcoal adsorbers. The emergency pressurized air supply fans will operate to keep the control room pressurized.

As stated in Section 6.4, Habitability Systems, the heating, ventilating, air-conditioning and air cleanup system for the main control room is adequately designed to protect the control room operator from radiation and other airborne hazards.

Based on our review, we conclude the control building heating, ventilating, air-conditioning and air cleanup system design meets the requirements set forth in General Design Criterion 19, "Control Room," and is therefore acceptable.

9.4.2 Auxiliary Building Ventilation System

The safety-related auxiliary building ventilation systems are the engineered safety feature pump room coolers, the shutdown board room air conditioning system, and the auxiliary board rooms air conditioning system.

The engineered safety feature pump room coolers are designed to maintain the ambient temperature in each of the charging, residual heat removal, containment spray, component cooling and auxiliary feedwater pump rooms at or below 104 degrees Fahrenheit during pump operation. Cooling water is supplied from the essential raw cooling water system train associated with the individual pump. The coolers and all associated components are designed to seismic Category I requirements. The coolers and their power supplies satisfy the single failure criterion.

The shutdown board room air conditioning system is designed to maintain the ambient temperature at or below 75 degrees Fahrenheit. In the event of a temperature exceeding 87 degrees Fahrenheit, an alarm in the control room will alert the operator to energize the standby air conditioning assembly. The system is designed to meet seismic Category I requirements. Power supplies and essential raw cooling water cooling supplies to these units meet the single failure criterion.

The auxiliary board room's air conditioning system are separated into two trains serving two subareas per plant unit. The equipment and attendant air conditioning system for each subarea are redundant and designed to seismic Category I criteria. Only one train is needed for the safe shutdown of the unit.

The battery rooms for each unit are supplied with conditioned air from one of the auxiliary board room's air conditioning system on a coordinated train basis. Each battery room is continuously ventilated and monitored to prevent the possibility of accumulation of hydrogen gas.

We have reviewed the design of the auxiliary building ventilation system and conclude that it meets the guidelines of Regulatory Guide 1.29 as well as the single failure criterion. We further conclude that the system design is capable of providing adequate protection under normal and postulated accident conditions, and is therefore acceptable.

9.4.3 Diesel Generator Building Ventilation Systems

The diesel generator building ventilating systems are designed to maintain an acceptable building environment for the protection of the diesel generators, electrical board rooms, batteries, and the safety of the operating personnel. Each diesel generator room is separately ventilated to limit the maximum ambient temperature to 120 degrees Fahrenheit when the entering air is 97 degrees Fahrenheit and the diesel generator is operating. Battery areas are ventilated at all times to prevent accumulation of hydrogen gas, and the electrical board rooms are ventilated to limit the ambient temperatures to 104 degrees Fahrenheit when the entering air is 97 degrees Fahrenheit.

For each of the diesel generators, two 100-percent capacity diesel generator room exhaust fans, one battery hood exhaust fan, and one electrical board room exhaust fan are provided. Each pair of diesel generator room exhaust fans is connected to its own respective diesel generator engineered safety power supply, and one fan will automatically start upon diesel generator start.

The diesel generator building ventilation system is required to operate to maintain plant safety in the event of natural phenomena or plant accidents and, is therefore, designed to seismic Category I requirements.

We have also reviewed the applicant's analysis regarding the maximum concentration of noxious gases in the diesel engine combustion air inlet. The maximum calculated carbon dioxide concentration is about 15 percent. The diesel manufacturer has specified that there will be no reduction in emergency power for carbon dioxide concentrations up to 15 percent by volume in the air intake stream.

Based on our review, we conclude that the diesel generator building ventilation systems meet the guidelines of Regulatory Guide 1.29 and the single failure criterion, and are therefore acceptable.

9.5 Fire Protection System

Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1976," documents our position on fire protection for such plants as the Sequoyah Nuclear Station Units 1 and 2. We transmitted Appendix A to the applicant and requested performance of a fire hazard analysis and an evaluation of the fire protection program for this plant, including a comparison with Appendix A. The applicant has submitted the required information, and our review, which included an inspection of the plant, is in process. We will complete our review prior to issuance of an operating license and if necessary we will appropriately condition the operating license to assure timely implementation of any required improvements, modification and performance of analyses and evaluations.

The Technical Specifications will include limiting conditions for operation and surveillance requirements for the fire protection systems and administrative controls.

We will report our findings in a supplement to this report.

9.6 Diesel Generator Auxiliary Systems

9.6.1 Diesel Generator Fuel Oil System

The diesel generator fuel oil system has storage and transfer capacity to supply diesel fuel to all four diesel generator sets operating at full load for a period of no less than seven days with the ability to replenish the supply from an offsite source during that period. The system consists of four storage tank assemblies, one for each tandem diesel generator unit, and associated pumps, valves, and piping. The tanks are embedded in the seismic Category I diesel generator building separated by 18 inches of concrete. Two motordriven pumps are provided for each diesel generato unit to transfer fuel oil from its storage tank to its 550-gallon engine-mounted day tank.

9.6.2 Diesel Generator Cooling Water System

A closed cooling water system is provided for each diesel engine. The system includes two closed engine cooling water loops. Each loop includes a pump, heat exchanger, expansion tank and all accessories required for a cooling loop. The heat sink for the engine cooling water system is provided by the essential raw water system.

9.6.3 Diesel Generator Starting System

Two independent air start trains are provided for each diesel engine complete with valves, piping and controls. For each diesel engine, two full capacity starting air motors and two accumulators are provided. Each accumulator has the capability for five engine starts without recharging. Each train has a compressor of sufficient capacity to recharge one set of accumulators in 30 minutes. Except for the compressors the diesel generator starting system is designed to seismic Category I requirements.

9.6.4 Diesel Generator Lubrication System

Each diesel engine is provided with a full pressure lubrication system which rejects heat to the diesel generators cooling water system. When the engine is not running, an electric heating system maintains the lube oil at temperature to enhance "first try" starting. The system is designed to satisfy single failure criterion.

9.6.5 Conclusions

We have reviewed the adequacy of the applicant's design and safety classification for the diesel generator fuel oil system, cooling water system, starting system, and lubrication system, and conclude that these systems are designed to perform their designated safety functions in accordance with the Commission's regulations as set forth in General Design Criteria 2, 4, 44, 45 and 46, and meet the guidelines of Regulatory Guides 1.26 and 1.29, and therefore are acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion systems are of conventional design, similar to those of previously approved plants. The systems are designed to remove thermal energy from the reactor coolant by four steam generators and convert it to electrical energy by the turbine generator. The systems are designed for the maximum expected thermal output from the nuclear steam supply system.

In the event of a turbine trip or a large load reduction, the heat transferred from the reactor coolant to the steam generators is dissipated via the turbine bypass system to the condenser, or through the power-operated relief valves and safety valves to the atmosphere if the condenser is not available.

10.2 Turbine Generator

The turbine generator is the tandem compound type with a double-flow high pressure turbine and three double-flow low pressure turbines. The rotational speed is 1800 revolutions per minute.

The turbine utilizes an electrohydraulic control system for control of both speed and load. The electrohydraulic control system, composed of solid state electronic devices coupled through suitable electrohydraulic transducers to a high-pressure hydraulic fluid system, provides control of the main stop, governing, intercept, and reheat stop valves of the turbine. Overspeed protection is provided by an electrical overspeed governor, backed up by a mechanical overspeed governor. When a turbine trip is initiated, the extraction system nonreturn valves are tripped to close by a pilot dump valve connected to the turbine trip system.

Additional turbine protection is provided that will trip the turbine on evidence of low condenser vacuum, excessive shaft vibration, abnormal thrust bearing wear, or low bearing oil pressure. Turbine governor and turbine controls electrical are covered more fully in Section 7 of this report.

For overpressure protection of the turbine exhaust hoods and the condenser, four diaphragms which rupture at approximately five pounds per square inch gauge are provided on each turbine exhaust hood. Additional protective devices include exhaust hood high temperature alarm and trip.

Based on our review, we conclude that the turbine generator protection system is acceptable.

10.3 Main Steam Supply System

The steam generated in each of four steam generators is routed to the turbine by means of four main steam lines. Each main steam line contains five safety valves, one air-operated relief valve, one main steam isolation valve, and a check valve. The main steam supply system is designed to seismic Category I requirements up to and including the main steam isolation valve and check valve.

The main steam isolation valves are globe wye type, air to open and spring to close within five seconds of receipt of high containment pressure signal or high steam flow rate signal. In series with and downstream of the isolation valve is a check valve to prevent reverse flow of steam.

We have reviewed the main steam isolation valve arrangement to determine that it will prevent blowdown of more than one steam generator in the event of a main steam line break inside or outside containment, assuming a single active failure.

Based on our review, we conclude that the main steam supply system design is acceptable.

10.4 Other Features

10.4.1 Circulating Water System

The circulating water system furnishes the main steam condenser for each unit with cooling water from the intake pumping station at a flow rate of approximately 535,000 gallons per minute.

We have reviewed the consequences of flooding as a result of failure of this system affecting safety-related equipment that is required for safe plant shutdown. There is no safety-related equipment in the turbine building. Assuming that the total volume of the circulating water system should be discharged to the turbine building, there will be no communication between the flooded space and safety related areas via passageways, pipe chases, cableways or other flow paths.

Based on our review, we conclude that the circulating water system design is acceptable.

10.4.2 Auxiliary Feedwater System

The auxiliary feedwater system is designed to supply water to the steam generators for reactor coolant system sensible and decay heat removal. This need would occur when the normal feedwater system is not available. Therefore, the auxiliary feedwater system will be utilized during certain periods of normal startup and shutdown in the event of malfunctions such as loss of offsite power, and also, in the event of accidents. The auxiliary feedwater system is an engineered safety feature system and is designed to seismic Category I requirements, with the exception of the condensate storage supply system.

The auxiliary feedwater system contains two motor-driven pumps and one turbine-driven pump. Each motor driven pump has a capacity of 440 gallons per minute, which is sufficient for safe cooldown. The motor-driven pumps are connected to separate emergency power buses. The turbine-driven pump has a capacity of 880 gallons per minute. Steam supply to the turbine is taken from two of four main steam lines at a point upstream of the main steam isolation valves. Separate remote operated isolation valves are provided for these connections.

Normally, the pumps take suction from two condensate storage tanks located in the plant yard adjacent to the south wall of the turbine building. Each tank has a capacity of 397,700 gallons of which 190,000 gallons is reserved for the auxiliary feedwater system by means of a standpipe in the tank. The condensate storage tanks are not designed to seismic Category I requirements; however, the essential raw cooling water system provides an alternate assured source of water.

All three pumps will start automatically in the event of a safety injection signal, loss of offsite power, or tripping of both steam generator feed pumps. The turbine driven pump also starts automatically in the event of a two-out-of-three low-low water level signal in any steam generator. Auxiliary feedwater flow will be adjusted by remote-operated flow control valves.

Separate engineered safety feature-quality power subsystems and control air subsystems serve each electric-driven auxiliary feedwater pump and its associated valves. The valves associated with the turbine-driven pump are served by both electric and control air subsystems, with appropriate measures precluding any interaction between the two subsystems. The turbine-driven pump receives control power from a third direct current electrical channel that is distinct from the channel serving the electric pumps.

Except for the common supply line from the condensate tanks, the two reactor units have separate auxiliary feedwater systems.

Damage to the feedwater system piping could originate as a consequence of uncovering of the feedwater sparger in the steam generator or uncovering of the steam generator feedwater or auxiliary feedwater inlet nozzles. Subsequent events in turn can lead to the generation of a pressure wave that is propogated through the pipes and could result in unacceptable damage.

We are currently evaluating this problem on a generic basis for all pressurized water reactors. In their letter dated May 25, 1976, the applicant committed to install J-tubes in the steam generators and minimize the length of that horizontal portion of feedwater piping entering the steam generator. In addition, we require the applicant to conduct a test program on the modified system to demonstrate that unacceptable feedwater hammer damage will not result from anticipated transients.

We have reviewed the adequacy of the proposed design criteria and bases of the auxiliary feedwater system necessary for safe operation of the plant during normal, abnormal, and accident conditions. We conclude that the system design conforms with the diversity requirements of our Branch Technical Position APCSB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants," that the system has sufficient flexibility and redundancy including the capability of the system to withstand the combination of a single active failure and high energy line break, and that, subject to confirmation during the preoperational test program, the feedwater system will not be subject to water hammer damage, and is therefore acceptable.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste management systems are designed to provide for controlled handling and treatment of liquid, gaseous and solid wastes. The liquid radioactive waste system processes wastes from equipment and floor drains, sample waste, decontamination and laboratory wastes, regenerant chemical wastes, and laundry and shower wastes. The gaseous radioactive waste system provides holdup capacity to allow decay of short lived noble gases stripped from the primary coolant and treatment of ventilation exhausts through high efficiency particulate air filters and charcoal adsorbers as necessary to reduce releases of radioactive materials to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 20 and 10 CFR Part 50.34a. The solid radioactive waste system provides for the solidification, packaging, and storage of radioactive wastes generated during station operation prior to shipment offsite to a licensed facility for burial.

In our evaluation of the liquid and gaseous radioactive waste systems, we have considered: (1) the capability of the systems for keeping the levels of radioactivity in effluents "as low as is reasonably achievable" based on expected radwaste inputs over the life of the plant, (2) the capability of the systems to maintain releases below the limits in 10 CFR Part 20 during periods of fission product leakage at design levels from the fuel, (3) the capability of the systems to meet the processing demands of the station during anticipated operational occurrences, (4) the quality group and seismic design classification applied to the equipment and components and structures housing these systems, (5) the design features that will be incorporated to control the releases of radioactive materials in accordance with General Design Criterion 60 and (6) the potential for gaseous release due to hydrogen explosions in the gaseous radwaste system.

In our evaluation of the solid radioactive waste treatment system, we have considered: (1) system design objectives in terms of expected types, volumes and activities of waste processed for offsite shipment, (2) waste packaging and conformance to applicable Federal packaging regulations, and provisions for controlling potentially radioactive airborne dusts during baling operation, and (3) provisions for onsite storage prior to shipping.

In our evaluation of the process and effluent radiological monitoring and sampling systems we have considered the system's capability: (1) to monitor all normal and potential pathways for release of radioactive materials to the environment, (2) to control the release of radioactive materials to the environment, and (3) to monitor performance of process equipment and detect radioactive material leakage between systems.

In our evaluations, we have determined the quantities of radioactive materials that will be released in liquid and gaseous effluents and the quantity of radioactive waste that will be shipped offsite to a licensed burial facility. In making these determinations, we have considered waste flows, activity levels, and equipment performance, consistent with expected normal plant operation, including anticipated operational occurrences, over the projected 30 year operating life of the plant.

The estimated releases of radioactive materials in liquid and gaseous effluents were calculated using the PWR GALE Code described in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR GALE Code)", April 1976. The principal parameters used in these calculations are given in Table 11-1. The liquid and gaseous source terms are given in Tables 11-2 and 11-3, respectively.

The source terms given in Tables 11-2 and 11-3 were used to calculate the individual and population doses in accordance with the mathematical models and guidance contained in Regulatory Guide 1.109, "Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" Rev. 1, November 1977. Meteorologic factors in the dose calculations were determined using the guidance in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents from Routine Releases from Light-Water-Cooled Reactors", March 1976. The calculated individual doses are given in Table 11-6.

Our dose assessment considered the following three effluent categories: 1) pathways associated with radioactive materials released in liquid effluents to the Chickamauga Reservoir; 2) pathways associated with noble gases released to the atmosphere; and 3) pathways associated with radioiodines, particulates, carbon-14 and tritium released to the atmosphere. The mathematical models used to perform the dose calculations to the maximum exposed individual are described in Regulatory Guide 1.109.

In conformance with the requirements of Section V.B of Appendix I, the Tennessee Valley Authority filed with the Commission on June 4, 1976, July 14, 1976, and September 10, 1976, the necessary information to permit an evaluation of the Sequoyah Nuclear Plant with respect to the requirements of Sections II.A, II.B, and II.C of Appendix I. In this submittal TVA provided the necessary information to show conformance with the Commissions' September 4, 1975 amendment to Appendix I rather than perform a detailed cost-benefit analysis required by Section II.D of Appendix I.

Based on the following evaluation, we conclude that the liquid and gaseous radioactive waste treatment systems for Sequoyah Nuclear Plant are capable of maintaining releases of radioactive materials in liquid and gaseous effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 50.34a, and with Sections II.A, II.B, II.C, and II.D of Appendix I to 10 CFR Part 50.

Based on our evaluation, as described below, we find the proposed liquid, gaseous and solid radioactive waste systems and associated process and effluent radiological monitoring and sampling systems to be acceptable.

11.2 System Description and Evaluation

11.2.1 Liquid Waste Processing System

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

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

Chemical and Volume Control System

A letdown stream of approximately 75 gallons per minute of primary coolant is removed from the primary reactor coolant system for processing through the chemical and volume control system. The letdown stream is cooled through the letdown heat exchangers, reduced in pressure, filtered, and processed through one of two mixed-bed demineralizers. For cation control, a cation bed demineralizer is valved into the process stream approximately 10 percent of the time. The processed letdown stream is collected in the volume control tank and reused in the primary coolant system. The chemical and volume control system is used to control the primary coolant boron concentration by diverting a portion of the treated letdown stream to the boron recovery subsystem of the chemical and volume control system as shim bleed. We estimated the boron recovery system input from the chemical and volume control system letdown stream to be approximately 2900 gallons per day per reactor.

Primary coolant-grade water from equipment drains, equipment leakage, and from relief valves inside containment is collected in the reactor drain tank and equipment drain tank. We estimated the boron recovery system input from the reactor and equipment drain tanks to be approximately 300 gallons per day per reactor.

The 2900 gallon per day shim bleed and the 300 gallon per day input from the reactor and equipment drain tanks is collected in two 256,000 gallon holdup tanks. Liquid collected in the holdup tank is processed batchwise through one of two shared process trains, each consisting of a mixed bed demineralizer in series with a cation demineralizer, resin filters, a 30 gallon per minute boric acid evaporator, a condenser, and an anion bed condensate polishing demineralizer. A stripper column removes dissolved gases from the vapor body zone of the evaporator. The processed liquid is returned to the primary coolant system, stored in a holdup tank, or released to the Chickamauga Reservoir via the liquid waste processing system discharge header. In our evaluation, we assumed that approximately 10 percent of the treated process stream from the boron recovery system is released to the Chickamauga Reservoir via the liquid waste processing system discharge header.

Tritiated Waste Subsystem

Tritiated wastes are processed through the tritiated waste subsystem and recycled to the chemical and volume control system monitor tank for reuse. The tritiated waste subsystem consist of a 24,700 gallon tritiated drain collector tank, a two gallon per minute waste evaporator, and a mixed-bed polishing demineralizer.

Tritiated wastes from valve and pump leakoffs, floor drains, equipment drains, and plant samples are collected in a 24,700 gallon collection tank at an input flow rate of approximately 300 gallons per day. This waste is processed through an evaporator and a polishing demineralizer, and the distillate is collected in a 1500 gallon chemical and volume control system monitoring tank for sampling and analysis. The distillate collected in the monitoring tank will normally be recycled to the primary coolant system for reuse.

The decontamination factors listed in Table 11-1 were applied for radionuclide removal in the tritiated waste subsystem. In our evaluation we assumed that 10 percent of the tritiated waste distillate is discharged to the Chickamauga Reservoir via the cooling tower blowdown line.

NonTritiated Waste Subsystem

Aerated wastes and nontritiated wastes are processed through the nontritiated waste subsystem for discharge to the environment. Nontritiated wastes consisting of floor drains, nontritiated equipment drains and other waste sources containing less than 10 percent of the tritium concentration in the reactor coolant will be collected in a 23,500 gallon floor drain collector tank at an input flow rate of approximately 1,100 gallons per day per reactor. This waste will be processed through a 15 gallon per minute auxiliary evaporator and the distillate collected in one of three 2,000 gallon test tanks where samples are taken and analyzed to determine suitability for release to the environment.

Laundry, Hot Shower, Laboratory, and Decontamination Drains

Laundry and hot shower drains are normally released without treatment after filtration and monitoring for radioactivity. The nontritiated waste subsystem can process these wastes should radioactivity measurements indicate activity levels above a predetermined value. Decontamination liquid wastes and laboratory chemical liquid wastes will normally drain to a 600 gallon chemical drain tank. The wastes are than transferred to the solid radwaste system for solidification. Other laboratory wastes are transferred to the floor drain collector tank for processing through the nontritiated waste subsystem. Shipping cask decontamination wastes are collected in a 15,000 gallon cask decontamination tank. The waste is sampled and analyzed and if the radioactivity level is below a predetermined value, the waste is filtered and discharged to the environment. Waste not suitable for release in this manner is transferred to the floor drain collector tank for processing in the nontritiated waste subsystem.

Steam Generator Blowdown Treatment System

A steam generator blowdown treatment system is provided for each reactor unit. This system consists of a flash tank and the necessary piping to distribute the flashed steam to the number seven heaters and the water from the flash tank to the inlet header to the condensate demineralizers, where the impurities are removed. Water from the flash tank may also be directed to the condenser hotwell for mixing with condensate prior to treatment in the condensate demineralizers.

Condensate Demineralizer System

A system of six deep-bed regenerable condensate demineralizers is provided for cleanup of turbine condensate. Each reactor unit has its own condensate demineralizer system. In the full flow polishing mode, each system has a maximum capacity of 17,000 gallons per minute. In our evaluation, we assumed that 45 percent of the condensate bypasses the condensate demineralizers, that one demineralizer is regenerated every 3.5 days, and that an average of 3400 gallons per day of regenerant solutions is produced.

Condensate Demineralizer Waste Evaporator Package

Condensate demineralizer regenerant solutions may be released to the environment, processed through the nontritiated waste system, or processed through the condensate demineralizer waste evaporator package. The latter system consists of a 30 gallon per minute evaporator, together with pumps, coolers, condensers, and two distillate test tanks. An average of 3400 gallons per day of condensate demineralizer regenerant solutions is processed through the evaporator package, with the condensed bottoms going to the solid waste system for packaging and the condensate being collected in the distillate test tanks for sampling and analysis prior to release to the Chickamauga Reservoir. The condensate demineralizer waste evaporator package can also process wastes from the nontritiated waste subsystem in the event that the auxiliary waste evaporator is out of service.

We consider the system capacity and system design to be adequate for meeting the demands of the station during anticipated operational occurrences.

Conformance with Federal Regulations and Branch Technical Positions

The liquid waste processing system is located in the auxiliary building which is designed to seismic Category I criteria. The seismic design and quality group classification and capacities of principal components considered in the liquid waste processing system evaluations are listed in Table 11-4. We find the applicant's liquid radioactive waste treatment system design to be acceptable in accordance with the guidelines of Regulatory Guide 1.140, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

The system design includes measures intended to control the release of radioactive materials due to potential overflows from indoor and outdoor storage tanks. Tank levels are monitored either locally or in the control room, and high level alarms are activated should preset levels be exceeded. Overflow provisions such as sumps, dikes and overflow lines permit the collection and subsequent processing of tank overflow. We consider these provisions to be capable of controlling the release of radioactive materials to the environment.

We have determined that during normal operation including anticipated operational occurrences, the liquid radioactive waste processing system is capable of reducing the release of radioactive materials in liquid effluents to approximately 0.22 curies per year per reactor, excluding tritium and dissolved gases, and 460 curies per year per reactor for tritium. The calculated annual releases for radionuclides in liquid effluents from each unit are given in Table 11-2.

Using the source terms given in Table 11-2, we calculate the total body dose to an individual in an unrestricted area to be less than 5 mrem from the combined liquid effluents from Units 1 and 2.

Based on our evaluation, we conclude that the liquid radioactive waste processing system for the Sequoyah Nuclear Plant is capable of maintaining releases of radioactive materials in liquid effluents during normal operation, including anticipated operational occurrences, such that the calculated doses are less than the numerical design objectives of Section II.A of Appendix I to 10 CFR Part 50. Our evaluation also shows that the applicant's design of the liquid waste processing systems for Units 1 and 2 satisfies the design objectives specified in the option provided by the Commission's September 4, 1975 amendment to Appendix I, and therefore meets the requirements of Section II.D of Appendix I of 10 CFR Part 50.

We conclude that the liquid waste processing system is capable of reducing the releases of radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 50.34a and Appendix I to 10 CFR Part 50.

Based on our calculations, we have determined that the liquid waste processing system is capable of reducing the release of radioactive materials in liquid effluents to concentrations below the limits in 10 CFR Part 20.

11.2.2 Gaseous Waste Processing System and Plant Ventilation System

The gaseous radioactive waste processing system and plant ventilation system are designed to collect, store, process, monitor, recycle, and/or discharge potentially radioactive gaseous wastes which are generated during normal operation of the plant. The systems consist of equipment and instrumentation necessary to reduce releases of radioactive gases and particulates to the environment. The principal sources of gaseous waste are the effluents from the gaseous waste processing system, condenser vacuum pumps, and ventilation exhausts from the auxiliary building, reactor containment, and turbine building.

The gaseous water processing system for the Sequoyah Nuclear Plant is shared between Unit Nos. 1 and 2. The system collects and processes the hydrogenated fission product gases stripped from the primary coolant letdown, the volume control tanks, and the reactor drain tanks. The gases are compressed into pressurized storage tanks for decay. Redundant 40 standard cubic feet per minute capacity compressors are provided for this purpose. Releases from the gas decay tanks are processed through high efficiency particulate air filters and charcoal adsorbers installed in the reactor building vent, prior to release to the environment. Ventilation exhaust air from the containment is processed through high efficiency particulate air filters and charcoal adsorbers prior to release to the environment. Ventilation exhaust air from the auxiliary building and turbine building are released without treatment. Condenser vacuum pump exhaust is processed through high efficiency particulate air filters and charcoal adsorbers prior to release to the environment. There are nine storage tanks included in this system with a design pressure of 150 pounds per square inch gauge and a 600 cubic foot volume in each.

In our evaluation, we observed that the nine tanks provided have the capacity to store the radioactive waste gases approximately 90 days for decay. Based on our calculations, we consider the system capacity and design to be adequate for meeting the demands of the station during normal operation including anticipated operational occurrences.

Containment Ventilation System

Radioactive gases are released inside the containment when primary system components are opened or when primary system leakage occurs. During normal operation, the gaseous activity is sealed within the containment but will be released during containment purges. Based on information submitted by the applicant we assumed six-24 hour purges per year through high efficiency particulate air filters and charcoal adsorbers and subsequent release to the environment. Four purges per year are assumed to occur after shutdown for the purpose of reducing radioactivity concentrations prior to operator access. Two purges per year are assumed to occur while the reactor is operating at full power to control the containment pressure, temperature, humidity, and airborne radioactivity levels. The containment purge exhaust monitors will automatically isolate the purge system upon receipt of a radioactivity level above a predetermined value. In our evaluation, we assumed a particulate decontamination factor of 100 for filters and an iodine decontamination factor of 10 for charcoal adsorbers.

Ventilation Releases from Other Buildings

Radioactive materials are introduced into the plant atmosphere due to leakage from equipment transporting or handling radioactive materials. We estimated that 160 pounds per day of primary coolant will leak to the auxiliary building with an iodine partition factor of 0.0075. Small quantities of radionuclides are released to the turbine building atmosphere based on an estimated 1700 pounds per hour of steam leakage. The plant ventilation systems are designed to induce air flows from potentially less radioactive contaminated areas to areas having a greater potential for radioactive contamination. Our calculations assumed that effluents from the auxiliary building and turbine building are released directly to the environment without treatment.

Main Condenser Evacuation Systems

The main condenser evacuation system, one for each unit, is designed to establish and maintain main condenser vacuum by removing noncondensable gases from the condenser and discharging the gases through a turbine building roof ventilator. The system is designed to Quality Group D and to a nonseismic design classification. Each main condenser evacuation system consists of three mechanical vacuum pumps, an electrical heating coil, a high efficiency particulate air filter, and a carbon adsorber. Air and noncondensables from the filtered vacuum pump exhaust are continuously monitored by a radiation monitor prior to release to the environment. Offgas from the main condenser vacuum pumps contains radioactive gases as a result of primary to secondary leakage. In our evaluation, we assumed a primary to secondary leak rate of 100 pounds per day. Noble gases and iodine are contained in the steam generator leakage and released to the environment through the main condenser vacuum pumps in accordance with the partition factors listed in Table 11-1.

The scope of our review included the system capability to process radioactive gases and the design provisions incorporated to monitor and control releases of radioactive materials in gaseous effluents in accordance with General Design Criteria 60 and 64. Based on our evaluation, we find the main condenser evacuation system to be acceptable. The basis for our acceptance was conformance of the applicant's design, design criteria, and design bases for the main condenser evacuation system to the applicable regulations.

Conformance with Federal Regulations and Branch Technical Positions

The seismic design and quality group classification and capacities of the principal equipment in the gaseous waste processign system are referenced in Table 11-4. We find the applicant's design for the gaseous waste processing system and structure housing this system to be in conformance with the guidelines given in Regulatory Guide 1.140, and therefore acceptable.

The gaseous waste processing system provides for monitoring hydrogen and oxygen upstream of the waste gas compressor and the contents of the gas storage tanks. If the oxygen content exceeds a predetermined level, an alarm will sound in the reactor control room, alerting the operator to the condition. Flow will then be diverted from the gas decay tank being filled to a standby tank and a nitrogen diluent introduced into the system to reduce the potential for a hydrogen explosion. The hydrogen and oxygen monitoring system does not meet our current acceptance criteria because redundant monitors are not provided and because the system is not designed to automatically initiate action to mitigate the potential for explosion in the event of a high oxygen content. Therefore, we will provide a technical specification which will require sampling and analysis every four hours during gas monitor outages and will require that the reactor be shutdown in the event that the gas monitor outage exceeds seven days. With the inclusion of the technical specification described above, we find the sytem to be acceptable.

We calculated that the proposed gaseous radwaste treatment and plant ventilation systems are capable of reducing the release of radioactive materials in gaseous effluents to approximately 2900 curies per year per reactor for noble gases, 0.053 curies per year per reactor for iodine-131, 970 curies per year per reactor for tritium, and eight curies per year per reactor for carbon-14.

Using the calculated releases of radioactive materials in gaseous effluents from Units 1 and 2 given in Table 11-3, we calculated the annual gamma and beta air doses at or beyond the site boundary to be less than 10 mrad and 20 mrad, respectively, as shown in Table 11-6. As shown in Table 11-3, we calculate the release of iodine-131 to be less than one curie per year per reactor. Using the calculated releases for iodine-131 given in Table 11-3, we calculate the dose or dose commitment to any organ of an individual in an unrestricted area to be less than 15 mrem per year as shown in Table 11-6.

Based on our evaluation, we conclude that the gaseous waste processing systems and ventilation treatment systems for Sequoyah Nuclear Plant, Units 1 and 2, are capable of maintaining releases of radioactive materials in gaseous effluents during normal operation, including anticipated operational occurrences, such that the calculated doses are less than the numerical design objectives of Sections II.B, and C of Appendix I of 10 CFR Part 50. Our evaluation also shows that the applicant's design of the gaseous waste treatment systems for Units 1 and 2 conforms to the numerical design objectives specified in the option provided by the Commission's September 4, 1975 amendment to Appendix I and, therefore, meets the requirements of Section II.D of Appendix I of 10 CFR Part 50. We conclude that the gaseous radwaste treatment systems are capable of reducing radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 50.34a and therefore are acceptable.

11.2.3 Solid Radioactive Waste Treatment System

The solid waste system for Sequoyah Nuclear Plant is shared between Units 1 and 2, and is designed to process two general types of solid wastes: "wet" solid wastes which require solidification prior to shipment, and "dry" solid wastes which requires packaging and, in some cases, compaction prior to shipment to a licensed burial facility. "Wet" solid wastes, consisting of waste evaporator bottoms and chemical drain tank effluents are injected into a vermiculite-cement and mixture contained in 55-gallon drums. Spent resin slurries are encapsulated in 30 or 55 gallon drums for offsite shipment. Each resin drum will contain a resin cage assembly enclosed in a vermiculite-cement mixture. Bulk resins are packaged for shipment in 150 cubic foot and 180 cubic feet disposable steel liners. The steel liners are placed inside returnable steel or steel and lead shipping containers for offsite shipment. The applicant will be required by the Technical Specifications to submit a process control program to assure complete solidification of all "wet" solid waste.

The principal radionuclides in the solid wastes are long-lived fission and corrosion products, namely, Cs-134, Cs-137, Co-58, Co-60, Mn-54, and Fe-55.

"Dry" solid wastes, consisting mainly of ventilation air filters, contaminated clothing, paper, laboratory glassware, and tools, are compacted in 55 gallon drums by a waste baler. The baler is equipped with a shroud to prevent the escape of radioactive materials during the compaction process. During the baling operation, the air flow in the vicinity of the baler is exhausted by a fan through a high efficiency particulate air filter to the auxiliary building ventilation system to reduce the potential for airborne radioactive dusts. We estimate the dry solid waste total to be 4100 cubic feet per year per reactor with a total activity content of less than five curies.

Wastes are packaged in containers designed to meet the requirements of 49 CFR Parts 170-189. Shielding is provided to maintain acceptable contact dose rates to meet the provisions of 10 CFR Part 71.

We have evaluated the solid radwaste treatment for normal operation including anticipated operational occurrences. We estimate that the solid waste volumes and activities shipped offsite will be 17,000 cubic feet per year per reactor of solidified wet waste containing 22,000 curies and 4,100 cubic feet per year per reactor for dry solid waste containing not more than 5 curies per reactor total.

Conformance with Federal Regulations and Branch Technical Positions

The solid radwaste system is housed in the auxiliary building which is a seismic Category I structure and therefore meets the guidance given in Regulatory Guide 1.140. Storage facilities for solid waste include an area in the auxiliary building for approximately 70-55 gallon drums in the drum storage area. We find the storage capacity adequate for meeting the demands of the station for normal operation.

On the basis of our evaluation of the solid radwaste system, we conclude that the system design can accommodate the radwastes expected during normal operations, including anticipated operational occurrences. The packaging and shipping of all wastes are in accordance with the applicable requirements of 10 CFR Parts 20 and and 71 and 49 CFR Parts 170-178.

From these findings, we conclude that the solid radwaste system is acceptable.

Process and Effluent Radiological Monitoring Systems

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

Table 11-5 provides the proposed locations of continuous monitors. Monitors on certain effluent release lines automatically terminate discharges should radiation levels exceed a predetermined value. Systems which are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, are periodically sampled and analyzed in the plant laboratory.

We have reviewed the locations and types of effluent and process monitoring provided. Based on the plant design and on continuous monitoring locations and intermittent sampling locations, we have concluded that all normal and potential release pathways are monitored. We have also determined that the sampling and monitoring provisions are adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which could affect radioactivity releases. On this basis, we consider the monitoring and sampling provisions to meet the requirements of General Design Criteria 60, 63 and 64 and guidelines of Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

11.3 Evaluation Findings

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

In our evaluation we determined that the applicant's design of the liquid and gaseous waste treatment systems satisfies the design objectives of Appendix I to 10 CFR Part 50.

We conclude that the liquid and gaseous radwaste treatment systems will reduce radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 50.34a and therefore are acceptable.

We have considered the potential consequences resulting from reactor operation with a one percent operating power fission product source term and determined that under these conditions, the concentrations of radioactive materials in liquid and gaseous effluents in unrestricted areas will be a small fraction of the limits specified in 10 CFR Part 20.

We have considered the capabilities of the radwaste systems to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the liquid, gaseous, and solid waste system capacities and design flexibilities are adequate to meet the anticipated needs of the plant.

We have reviewed the applicant's quality assurance provisions for the radwaste systems, the quality group classifications used for system components, the seismic design applied to the gaseous waste processing system, and the seismic classification applied to the design of structures housing the radwaste systems. The design of the radwaste systems and structures housing these systems meet the guidelines as set forth in Regulatory Guide 1.140.

We have reviewed the provisions incorporated in the applicant's design to control the releases of radioactive materials in liquids ue to inadvertent tank overflows and conclude that the measures proposed by the a cant are consistent with our acceptance criteria as set forth in Regulatory G. 1.140.

Our review of the radiological process and effluent monitoring system included the provisions for sampling and monitoring all normal and potential effluent discharge paths in conformance with General Design Criterion 64, for providing automatic termination of effluent releases and assuring control over releases of radioactive materials in effluents in conformance with General Design Criterion 60 and Regulatory Guide 1.21, for sampling and monitoring plant waste process streams for process control in conformance with General Design Criterion 63, for conducting sampling and analytical programs in conformance with the guidelines in Regulatory Guide 1.21, and for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points. We conclude that the applicant's radiological process and effluent monitoring systems are acceptable.

Based on the foregoing evaluation, we conclude that the radwaste treatment and monitoring systems are acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the radwaste treatment and monitoring systems to the applicable regulations and guides referenced above, as well as to staff technical positions and industry standards.

TABLE 11-1

PRINCIPAL PARAMETERS AND CONDITIONS USED IN CALCULATING RELEASES OF RADIOACTIVE MATERIAL IN LIQUID AND GASEOUS EFFLUENTS FROM SEQUOYAH NUCLEAR POWER PLANT UNITS 1 AND 2

Reactor Power Level (Megawatts thermal)	3582
Plant Capacity Factor	0.80
Failed Fuel	0.12 percent[a]
Primary System	
Mass of Coolant (pounds)	5.4×10^5
Letdown Rate (gallons per minute)	75
Shim Bleed Rate (gallons per day)	2.9×10^3
Leakage to Secondary System (pounds per day)	100
Leakage to Containment Building	b
Leakage to Auxiliary Building (pounds per day)	160
Frequency of Degassing for Cold Shutdown (per year)	2
Secondary System	
Steam Flow Rate (pounds per hour)	1.5×10^7
Mass of Steam/Steam Generator (pounds)	7.8×10^3
Mass of Liquid/Steam Generator (pounds)	8.7×10^4
Number of Steam Generators	4
Secondary Coolant Mass (pounds)	9.8×10^5
Rate of Steam Leakage to Turbine Bldg. (pounds per hour)	1.7×10^3
Fraction of Feedwater Processed Through Condensate Demins	0.55
Containment Building Volume (cubic feet)	1.3×10^6
Annual Frequency of Containment Purges	6-24 hour
Iodine Partition Factors (gas/liquid)	
Leakage to Auxiliary Building	0.0075
Steam Generator	0.01
Leakage to Turbine Building	1.0
Main Condenser/Air Ejector (volatile species)	0.15
Liquid Waste Processing Systems	

System	Input Flow Rate Gallons Per Day	Decontamination Factors I	Cs, Rb	Others
Nontritiated Waste	1100	10^3	10^4	[illegible]
Condensate Regenerate Waste	3400	10^3	10^4	10^4
Tritiated Waste	300	10^4	10^5	10^5
Laundry Waste	450	1	1	1
Boron Recovery	3200	10^5	2×10^4	10^5

[a]This value is constant and corresponds to 0.12 percent of the operating power fission product source term as given in NUREG-0017.

[b]One percent per day of the primary coolant noble gas inventory and 0.001 percent per day of the primary coolant iodine inventory.

TABLE 11-2

CALCULATED RELEASES OF RADIOACTIVE MATERIAL IN LIQUID EFFLUENTS FROM SEQUOYAH, UNITS 1 AND 2 RELEASE (CURIES PER YEAR REACTOR)

Nuclide	Curies Per Year	Nuclide	Curies Per Year
Corrosion & Activation Products		Fission Products	
Cr-51	1.3(-4)[a, b]	Te-129m	9(-5)
Mn-54	1(-3)	Te-129	6(-5)
Fe-55	1.3(-4)	I-130	1.2(-4)
Fe-59	8(-5)	Te-131m	5(-5)
Co-58	5.2(-3)	I-131	9.2(-2)
Co-60	8.5(-3)	Te-132	7.8(-4)
Zr-95	1.4([illegible])	I-132	1.8(-3)
Nb-95	2(-3)	I-133	3.3(-2)
Np-239	4(-5)	Cs-134	2(-2)
Fission Products		I-135	5.9(-3)
		Cs-136	1.9(-3)
Br-83	3(-5)	Cs-137	2.9(-2)
Rb-86	1(-5)	Cs-137m	4.9(-3)
Sr-89	3(-5)	Ba-140	1(-5)
Mo-99	2.2(-3)	La-140	1(-5)
Tc-99m	2.3(-3)	Ce-144	5.2(-3)
Pu-103	1.4(-4)	All Others	6(-5)
Ru-106	2.4(-3)	Total (except H-3)	2.2(-1)
Ag-110m	4.4(-4)	H-3	460
Te-127m	2(-5)		
Te-127	2(-5)		

[a]Exponential notation; $1.3(-4) = 1.3 \times 10^{-4}$

[b]Nuclides whose release rates are less than 10^{-5} Curies per year per reactor are not listed individually, but are included in the category "All Others".

TABLE 11-3

CALCULATED RELEASES OF RADIOACTIVE MATERIAL IN GASEOUS EFFLUENTS FROM SEQUOYAH, UNITS 1 AND 2 RELEASE (CURIES PER YEAR PER REACTOR)

RADIO-NUCLIDE	REACTOR BUILDING	AUXILIARY BUILDING	TURBINE BUILDING	DECAY TANKS	AIR EJECTOR OFF-GAS	TOTAL
Kr-83m	a	a	a	a	a	a
Kr-85m	a	2	a	a	2	4
Kr-85	57	2	a	310	a	370
Kr-87	a	1	a	a	a	1
Kr-88	a	5	a	a	3	8
Kr-89	a	a	a	a	a	a
Xe-131m	22	2	a	1	1	26
Xe-133m	11	5	a	a	3	19
Xe-133	1900	360	a	a	220	2500
Xe-135m	a	a	a	a	a	a
Xe-135	3	8	a	a	5	16
Xe-137	a	a	a	a	a	a
Xe-138	a	a	a	a	a	a
TOTAL NOBLE GASES						2900
I-131	4.7(-3)[b]	4.5(-2)	6.1(-4)	a	2.8(-3)	5.3(-2)
I-133	7.4(-4)	6.3(-2)	8(-4)	a	4(-3)	6.9(-2)
Mn-54	2.2(-4)	1.8(-2)	c	4.5(-5)	c	1.9(-2)
Fe-59	7.5(-5)	6.0(-3)	c	1.5(-5)	c	6.1(-3)
Co-58	7.5(-4)	6.0(-2)	c	1.5(-4)	c	6.1(-2)
Co-60	3.4(-4)	2.7(-2)	c	7.0(-5)	c	2.7(-2)
Sr-89	1.7(-5)	1.3(-3)	c	3.3(-6)	c	1.3(-3)
Sr-90	3(-6)	2.4(-4)	c	6.0(-7)	c	2.4(-4)
Cs-134	2.2(-4)	1.8(-2)	c	4.5(-5)	c	1.8(-2)
Cs-137	3.8(-4)	3.0(-2)	c	7.5(-5)	c	3.0(-2)
TOTAL PARTICULATES						1.6(-1)
C-14	1	a	a	7	a	8
H-3	c	970	c	c	c	970
Ar-41	25	a	a	a	a	25

[a]less than 1.0 curie per year for noble gases and carbon-14, less than 10^{-4} curies per year for iodine.

[b]exponential notation; $1.2(-3) = 1.2 \times 10^{-3}$

[c]less than one percent of total for this nuclide.

TABLE 11-4

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN THE EVALUATION OF LIQUID AND GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEMS

COMPONENT	NUMBER	CAPACITY EACH
Liquid Systems		
Tritiated Waste Processing System[a]		
Tritiated Drain Collector Tank	1	24,700 gallons
Waste Evaporator Package	1	2 gallons per minute
Condensate Demineralizer	1	100 gallons per minute
Nontritiated Waste Processing System[a]		
Floor Drain Collector Tank	1	22,800 gallons
Auxiliary Waste Evaporator Package	1	15 gallons per minute
Waste Condensate Tank	3	2 - 1,[illegible]00 gallons 1 - 2,000 gallons
Laundry, Hot Shower, Chemical Waste and Decontamination Waste Processing System		
Laundry and Hot Shower Collector Tanks	2	600 gallons
Cask Decontamination Tank	1	15,000 gallons
Chemical Waste Collection Tank	1	600 gallons
Condensate Demineralizer Regenerant Waste Processing System		
Nonreclaimable Waste Collection T[illegible]	1	11,000 gallons
Condensate Demineralizer Waste [illegible]ator Package	1	30 gallons per minute
Gaseous Systems		
Gaseous Waste Processign System		
Compressor	2	40 scfm
Decay Tanks	9	600 ft^3

[a]Quality Group and Seismic design in accordance with Regulatory Guide 1.140.

TABLE 11-5

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM

Stream Monitored*	No.	Monitor Classification	Monitor Sensitivity
LIQUID			
Component Cooling Water	3/plant	Gamma-Scint.	3×10^{-7} (Co-60)
Service Water	2/plant	Gamma-Scint.	3×10^{-7} (Co-60)
Steam Generator Blowdown (Process)	2/plant	Gamma-Scint.	3×10^{-7} (Co-60)
Waste Disposal System**	1/plant	Gamma-Scint.	3×10^{-7} (Co-60)
Steam Generator Blowdown Liquid Discharge**	2/plant	Gamma-Scint.	3×10^{-7} (Co-60)
GASEOUS			
Auxiliary Building Exhaust Vent**			
Particulate	1/plant	Beta-Scint.	5.7×10^{-11} (Co-60)
I-131	1/plant	Gamma-Scint.	7.4×10^{-10} (I-131)
Noble Gas	1/plant	Beta-Scint.	4.1×10^{-7} (Kr-85)
Gaseous Waste Process System Discharge**	1/plant	Beta-Scint.	1×10^{-6} (Kr-85)
Condenser Vacuum Pump, Low Range Exhaust	2/plant	Beta-Scint.	1.4×10^{-7} (Kr-85)
Condenser Vacuum Pump, High Range Exhaust	2/plant	Beta-Scint.	1×10^{-3} (Gross)
Containment Purge Exhaust**	4/plant	Beta-Scint	5.7×10^{-4} (Gross)
Shield Building Vent Exhaust	2/plant	Beta-Scint	5.7×10^{-11} (Co-60)

*All monitors have in[illegible]ment malfunction and high radiation visual and audible alarms in the main control ro[illegible]

**These monitors terminate the release when the radiation level exceeds a predetermined level.

TABLE 11-6

COMPARISON OF SEQUOYAH NUCLEAR POWER STATION, UNITS 1 AND 2, WITH APPENDIX I TO 10 CFR PART 50, SECTIONS II.A, II.B, AND II.C (MAY 5, 1975)[b]

Criterion	Appendix I[a] Design Objectives	Annex Design Objectives[c]	Calculated Doses
Liquid Effluents	(per unit)	(per site)	(per unit)
Dose to total body from all pathways	3 mrem/yr	5 mrem/yr	0.19 mrem/yr
Dose to any organ from all pathways	10 mrem/yr	5 mrem/yr	0.26 mrem/yr
Noble Gas Effluents[c]			
Gamma Dose in air	10 mrad/yr	10 mrad/yr	2.3 mrad/yr
Beta dose in air	20 mrad/yr	20 mrad/yr	6.8 mrad/yr
Dose to total body of an individual	5 mrem/yr	5 mrem/yr	1.4 mrem/yr
Dose to skin of an individual	15 mrem/yr	15 mrem/yr	4.4 mrem/yr
Radioiodines and Other Radionuclides Released to the Atmosphere[d]			
Dose to any organ from all pathways	15 mrem/yr	15 mrem/yr	5.4 mrem/yr

[a]Federal Register, V. 40, p. 19442, May 5, 1975.

[b]Federal Register, V. 40, p. 40816, September 4, 1975.

[c]Limited to noble gases only.

[d]Carbon-14 and Tritium have been added to this category.

12.0 RADIATION PROTECTION

This section presents an evaluation of the adequacy of the radiation protection design features and the health physics program at the Sequoyah Nuclear Plant to control radiation exposures within the limits of 10 CFR Parts 20 and 50. The review emphasis centered around the applicant's program to maintain occupational radiation exposure as low as reasonably achievable.

We have reviewed the Sequoyah radiation protection program for assuring that occcupational radiation exposures will be as low as reasonably achievable. Towards this end, the review covered management's policies and organizational structure relating to radiation protection, a description of the applicant's design considerations and features and methods used for developing plans and procedures. The review considered the manner in which the applicant's policies, plans, and organization conform to the guidelines of Regulatory Guides 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable," and 8.10, "Operating Philosophy for Maintaining Radiation Exposures as Low as is Reasonably Achievable (Nuclear Power Reactors)."

The basis for acceptance has been conformance of management's policies, and design and operational considerations, with the aforementioned Regulatory Guides as well as established criteria and practices at licensed nuclear plants.

We conclude that policy, design, and operational considerations relating to occupational exposures conform to the Commission's regulations, appropriate Regulatory Guides, and industry standards, as identified below.

12.1 Assuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable

Management's commitment to the philosophy of maintaining occupational radiation exposures as low as reasonably achievable is provided through TVA's management policies, administrative guides and instructions, and organizational structure, all relating to radiation protection. Additionally, responsibility for implementing Sections C.3 and C.4 of Regulatory Guide 8.8 has been assigned to various divisions within TVA. The coordinator for implementing this philosophy for Sequoyah is the health physics staff supervisor who is a member of the TVA Radiological Hygiene Branch which is located in Muscle Shoals, Alabama. A committee will audit each of TVA's nuclear facilities at least once a year to determine that criteria are being met regarding as low as reasonably achievable exposures.

12.2 Radiation Sources

The sources of airborne and contained radioactivity are described as necessary for input into the shielding and ventilation design and the dose assessment. Inside the containment during operation nitrogen-16, noble gases, and neutrons present the greatest potential for personnel dose. Inside the containment during shutdown and in the other parts of the plant, fission and activation products present the greatest dose potential. The source terms for shielding design are based on a core thermal power of 3582 megawatts thermal with one percent failed fuel and other parameters which are presented in Chapter 11 of the Final Safety Analysis Report. Airborne radioactivity source terms are based on one quarter of one percent failed fuel with the exception of certain rooms in the auxiliary building which are based on one percent failed fuel. Estimates of coolant concentrations of activated corrosion products are based on measurements from operating reactors and are not based on failed fuel percentage.

Operating experience shows that one quarter of one percent failed fuel was exceeded in less than ten percent of pressurized water reactor cycle averages in the early 1970's. Improvements in fuel design and production and in operating methods since then should ensure an even lower incidence in the future. On this basis, the use of one quarter of one percent failed fuel leads to reasonable estimates of the airborne source terms.

The assumptions and procedures used by the applicant in estimating radiation source terms, and the estimates themselves, have been evaluated. The methods are consistent with the acceptance criteria of our Standard Review Plan with one exception. As indicated above, a lower failed fuel percentage than that stated in the Standard Review Plan was used for airborne source terms for most of the plant. We find this approach to be acceptable for the reasons discussed above. The estimates are reasonable, and they are comparable to estimates by other applicants with similar designs. Therefore, we conclude that the source term information presented is acceptable.

12.3 Radiation Protection Design Features

Radiation shielding at Sequoyah is designed to ensure that the criteria of 10 CFR Parts 20 and 50 will be met during normal operations or anticipated operational occurrences and that design features to achieve exposures to operating personnel as low as reasonably achievable during refueling, maintenance, and inservice inspection, and similar operations have been included in the plant design. Our review showed that the applicant has designed his shielding and shield penetrations to minimize doses outside shield enclosures. Manually-operated valves such as those in valve galleries and reach rods through shield walls are used to control process equipment. Actual doses at valves depend upon anticpated occupancy at valve locations and valve service and maintenance requirements. Shield walls or portions of shield walls that are subject to removal for equipment repair or replacement are made of removable solid concrete blocks. There is no field-run process piping. Piping is run in shielded pipe chases where possible. One pipe chase has a concrete partition between pipes from the two units to minimize the spread of contamination between units should a pipe rupture occur. Areas where continuous occupancy may occur have been shielded to give dose rates of less than 0.1 millirem per hour.

The ventilation system is designed to meet the applicable regulations of 10 CFR Parts 20, 50 and 100. Air flows are designed to go from clean to low potential airborne radioactivity areas to areas having higher potentials for airborne radioactivity. Routine checks are to be made to assure that negative pressures are always maintained in those areas where the potential for surface or airborne contamination exists. Exhaust fan failures are indicated and alarmed in the control room thus giving an early indication of potential air flow reversal. With respect to maintenance and inplace depth of penetration testing of high efficiency particulate air filters, the ventilation system is designed in accordance with applicable sections of Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

We conclude that the ventilation system is based on design criteria which provide reasonable assurance that the system has the capability to maintain concentrations of airborne activity in areas normally occupied in accordance with 10 CFR Part 20.

Twenty area radiation monitors are provided throughout the plant in areas in which personnel may routinely work without direct health physics supervision and in areas where there is a possibility of noble gas activity in concentrations that are a significant fraction of those given in 10 CFR Part 20, Appendix B, Table I. Additionally, two monitors provide for monitoring near the containment air locks and two monitors provide for personnel safety during fuel loading and refueling. The monitors are of sufficient sensitivity to detect minor changes in radiation levels. Each monitor has local and control room ratemeters and local and control room alarms. Local alarms are audible and visible. Instrumentation calibration checks will be performed, and dose rate levels will be recorded, in the control room.

Inplant airborne radioactivity will be monitored by seven gaseous radioactive effluent monitors and ten air particulate monitors stationed throughout the plant, each having local indication ratemeters and alarms with redundant control room ratemeter display and alarm. Areas with a high potential for airborne radioactivity have restricted entrance requirements, such as locked doors, that require special work permits from the health physics staff for access. Sample ports are available within the ventilation system for additional surveillance with continuous air monitors during work within these areas.

Based on the location of area monitors, their sensitivity and range, and their alarm annunciation and recording devices, we conclude that the area monitoring program will provide satisfactory radiological protection to inplant personnel.

12.4 Dose Assessment

The applicant has provided an assessment of the dose which will be received by workers at the plant. TVA estimates that nonmaintenance personnel will receive 120 man-rem per year. It is estimated that maintenance personnel will receive 70-140 man-rem per year in the early plant life; as the plant ages the dose which maintenance personnel will receive will increase to 210-280 man-rem per year.

An acceptable dose assessment should be based on occupancy factors, expected dose rates, expected airborne radioactivity concentration, and estimates of the time and manpower necessary to perform the tasks involved in plant operation. The tasks involved in operations, maintenance, technical services, inservice inspection, refueling, surveillance, calibration, and radwaste handling should be included in the assessment. Also, the assessment should include nonroutine tasks when possible even though such estimates probably are not as accurate as estimates for routine tasks. Finally, the assessment should include experience from operating power reactors where applicable.

The applicant's dose assessment has adequately considered these factors, and his estimates are reasonable in light of the operating experience reported by the Commission in NUREG-0109, "Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1975." Therefore, we conclude that the dose assessment is acceptable.

12.5 Health Physics Program

We have reviewed the Final Safety Analysis Report to determine that the health physics program will assure that occupational exposures will be as low as reasonably achievable. The review covered management policies and organizational structure and program for maintaining exposures as low as reasonably achievable, the health physics facilities and monitoring equipment related to the program, and procedures related to contamination control and occupational radiation exposures.

The applicant's stated policy for radiation protection is based on appropriate Commission regulations. Consistent with this policy, programs and procedures will be adapted conforming to the positions of Regulatory Guide 8.8. Health physics coverage will be supplied by a health physicist (whose qualifications conform to Regulatory Guide 8.8) and his staff who will provide coverage on a round-the-clock basis as part of the effort to maintain exposures as low as reasonably achievable. The health physics staff has a laboratory for routine health physics use such as equipment for counting smear and air samples and storage of health physics equipment. Additionally, the radiochemistry laboratory has equipment such as a 4000 channel pulse height analyzer with Ge(Li) and sodium iodide detectors, and liquid scintillation counters, all of which will be available for health physics staff operations. Portable survey instruments are calibrated monthly at the central laboratory in Muscle Shoals, Alabama. Protective clothing and respirators are available for personnel. A mobile whole-body counter will be used in the bioassay program to check all employees on a routine basis and is available for emergency use. To supplement the whole-body counting program, urinalysis bioassay programs will be conducted for tritium and strontium body burden evaluation. Personnel dosimetry equipment is provided by the TVA film badge service. The health physics staff also provides radiation protection training and has developed a Health Physics Manual and Handbook of Health Physics for TVA employees. Access control to radiation areas is such that personnel must pass through a health physics control point during entrance and exit. On the basis of the applicant's description of the design and the operating philosophy of the Sequoyah Nuclear plant, we conclude that sufficient consideration has been given to the design and health physics program of the facility to keep exposures to operating personnel as low as reasonably achievable in accordance with Regulatory Guide 8.8.

13.0 CONDUCT OF OPERATIONS

13.1 Plant Organization and Staff Qualifications

Operational activities are conducted under the onsite supervision of the Power Plant Superintendent. He reports to the Chief, Nuclear Generation Branch, who in turn reports to the Director, Division of Power Production of the Office of Power. The Power Plant Superintendent is responsible for the safe and reliable operation of the plant. The plant staff of approximately 175 full time employees, includes organizational units responsible for power plant results (approximately 24 people), power plant operations (approximately 44 people), power plant maintenance (approximately 59 people), health physics, engineering quality assurance, and other non-technical supporting services. This is a conventional type of plant organization to provide onsite technical support for plant operation.

The Plant Operations Supervisor directs the day-to-day operation of the station and is responsible to the Power Plant Superintendent for the safe and efficient operation of the station in accordance with the operating license, Technical Specifications, and approved procedures. Reporting to him are an assistant operations supervisor and the plant operating shifts. The minimum shift composition for single unit operation is one shift engineer licensed as a senior operator, one reactor operator licensed as an operator, one unit operator licensed as an operator, and four assistant unit operators. For two-unit operation, one assistant shift engineer licensed as a senior operator, and one unit operator licensed as an operator, will be added to each operating shift.

The Power Plant Results Supervisor is responsible to the Power Plant Superintendent for analysis of the performance of the reactor and turbine cycle and associated equipment during the test, startup, and operation of the plant. Included in the staff reporting to him are a nuclear engineer, instrument engineer, and a chemical engineer. The plant maintenance supervisor is responsible to the Power Plant Superintendent for mechanical and electrical maintenance work and inspections in the plant. The health physicist is responsible for the direction of the onsite radiological health program. The engineering unit is responsible for the maintenance and testing of the relaying associated with the transmission system. The Supervisor, QA Staff, is responsible for implementing the plant quality assurance program

The applicant has stated that all Department of Power Production personnel at the Sequoyah plant will meet the qualifications described in ANSI N18.1-1971, "Standard for the Selection and Training of Personnel for Nuclear Power Plants." This meets the regulatory position described in Regulatory Guide 1.8, "Personnel

Selection and Training." We have reviewed the qualifications of key supervisory personnel assigned to the Sequoyah plant staff. We find these acceptable, since the qualifications of key supervisory personnel with regard to educational background, experience, and technical specialties are in accord with those defined in ANSI N18.1.

Offsite technical support for the plant staff for the operation of the facility is provided by the various technical branches of the Division of Power Production. In addition, technical support for the operation of the facility is available from TVA's Division of Power System Operations, Division of Transmission, Planning and Engineering, Division of Power Resource Planning, Division of Engineering Design, Division of Chemical Development and the Division of Environmental Planning.

Based on: (1) our review of TVA's corporate and technical organization; (2) the technical resources as embodied in the numbers and technical experience of personnel assigned and available to the project; (3) the Quality Assurance Program as described in Section 17.0 of this report; (4) the exchange of technical information experienced in our meetings and correspondence during the course of the review; we conclude that TVA is technically qualified to operate the Sequoyah Nuclear Plant.

We have concluded that the organizational structure and qualifications of the plant organization meet Regulatory Guide 1.8 and are satisfactory to provide an acceptable operating staff. We further conclude that the applicant has the necessary resources to provide offsite technical support for the operation of the facility. Additional technical support during the startup test program will be provided by Westinghouse Electric Corporation.

13.2 Training Program

The overall training program for the Sequoyah Units 1 and 2 plant staff is the responsibility of the Plant Superintendent. The Training Supervisor will be responsible for organizing and conducting the licensed and nonlicensed operator training program. The program for formal education and training of the Sequoyah Nuclear Plant staff has been designed to meet the individual needs of the participants, depending upon their background, previous training and expected job assignment. The program conforms to the requirements set forth in ANSI-N18.1, 1971, Section 5 and 10 CFR Part 55.

Personnel in training for NRC licenses will be trained in the following subject areas: principles of reactor operation, design features and general operating characteristics of the nuclear power plant, instrumentation and control systems, safety and emergency systems, standard and emergency operating procedures, and radiation control and safety provisions. In addition to the above subjects, technical training for candidates for NRC senior reactor licenses will cover the following subjects: reactor theory, handling and disposal of radioactive materials, specific operating characteristics of the nuclear power plant, fuel handling and core parameters, and administrative procedures, conditions and limitations. Mechanical and electrical personnel, chemists and health physicists will receive training in their particular specialty. Sequoyah personnel received reactor training at Oak Ridge National Laboratory, observation training at R. E. Ginna, Point Beach, and the Zion Nuclear Power Plants. Simulator training will be provided at the Westinghouse Nuclear Training Center in Zion, Illinois. Onsite operator training and audit examination will be provided by the Westinghouse training staff.

Plans for requalification training and replacement training conforms to 10 CFR Parts 50 and 55, Appendix A, and follows the guidance given in ANSI-N18.1.

On the basis of our review, we conclude that the training programs and schedule for all staff members are acceptable for the preoperational test program, for operator licensing examinations, and for fuel loading.

13.3 Emergency Planning

The applicant has formulated and submitted a Radiological Emergency Plan which describes the program for coping with emergencies within and beyond the site boundary. The applicant has established a formal organization that includes written agreements, liaison, and communication with appropriate local, State and Federal agencies that have responsibilities for coping with emergencies. The plan includes a spectrum of accidents including criteria for determining when protective measures should be considered as indicated by defined accident assessment techniques. The plan also describes arrangements made for providing necessary medical attention for persons with contaminated injuries, and provisions for maintaining an adequate emergency preparedness posture throughout the expected lifetime of the plant through training, exercises, and drills. The plan has been coordinated with the radiological response planning function of the State of Tennessee Department of Public Health.

Our initial review of the Radiological Emergency Plan was conducted prior to issuance of Standard Review Plan Section 13.3 and Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants." TVA subsequently submitted a Radiological Emergency Plan for the Watts Bar Nuclear Plant, which, except for site specific information, is very similar to that of Sequoyah. We conducted our review of the Watts Bar plan using the guidelines of the Standard Review Plan and Regulatory Guide 1.101 and informed TVA of our concerns with some aspects of that plan. We asked that the applicant commit to amending the Sequoyah plan to rectify those deficiencies found during the Watts Bar review which are also applicable to Sequoyah. The applicant has made this commitment and expects to submit the requested additional information on Sequoyah in early 1979. We will then review this new information and report our findings in a supplement to this Report.

13.4 Review and Audit

The applicant has agreed to a review and audit program as described in Section 6.5 of the Standard Technical Specifications for Pressurized Water Reactors. This consists of an onsite review group that will provide a continuing review of plant operations, and an offsite group that will provide an independent review and audit of plant operations. We find that these provisions for review and audit meet those described in Section 4 of ANSI N18.7-1976, "Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants." This meets the regulatory position described in Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation) (Revision 1, 1/77)," and is acceptable.

13.5 Plant Procedures & Records

All safety-related operations are to be performed in accordance with written and approved operating and emergency precedures. Areas covered include administrative procedure, operating instructions, off-normal instructions, emergency instructions, fuel handling and maintenance procedures. The applicant's provisions meet the requirements of 50.54(i), (j), (k), (l) and (m) of 10 CFR Part 50. Plant procedures related to nuclear safety follow the guidance of ANSI-N18.7, 1972, "Administrative Controls for Nuclear Power Plants" and Regulatory Guide 1.33, Appendix A, and are reviewed by the Plant Operations Review Committee and approved by the Power Plant Superintendent before initial use. The significant provisions will be included in the administration controls section of the plant's technical specifications. We conclude that the provisions for preparation, review, approval, and use of written procedures are adequate.

The applicant has described his program for maintaining plant records and has committed to maintaining records according to N45.2.9-1974. Specific records and their retention periods will be delineated in the facility technical specifications.

Based on our review, we conclude that the applicant's provisions for maintaining records meet the position described in ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants," and are satisfactory.

13.6 Industrial Security

TVA filed with the Commission an Amended Security Plan for Sequoyah dated May 16, 1977, pursuant to 10 CFR 73.55. The Commission's staff conducted a review of the Amended Security Plan. This review included a site visit to the facility. As a result of our review, a number of changes were made in the security design and proposed security plan. The changes are incorporated into a Modified Amended Security Plan dated August 25, 1978, as amended.

Based on our review of the Modified Amended Security Plan and our visit to the site, we have concluded that the plan is satisfactory, that the protection provided by TVA is adequate to deter and defend the Sequoyah Nuclear Plant from attempts of sabotage directed from within or outside the facility, and that the plan meets the requirements of 10 CFR 73.55(a). Accordingly, we conclude that the Modified Amended Security Plans will ensure that the health and safety of the public will not be endangered.

14.0 INITIAL TESTS AND OPERATION

The TVA Division of Power Production has responsibility for the overall preoperational and startup test program administration. The Seouoyah Nuclear Plant Superintendent is the onsite representative of the Division of Power Production and is responsible for the conduct of the test program and direction of the onsite implementation of the program through the Power Plant Results Supervisor who acts as the program coordinator.

Test instructions are prepared for each test with input as applicable from Westinghouse and TVA's Division of Engineering Design. These test instructions are reviewed by the Division of Power Production Plant Engineering Branch, the Sequoyah plant staff, and Plant Operation Review Committee, and are approved by the Power Plant Superintendent. Test results are approved by the Power Plant Superintendent and submitted to the Division of Engineering Design which has final responsibility for acceptance to test results.

The plant staff for Sequoyah is augmented by staff engineers from the Plant Engineering and Plant Maintenance Branches of the Division of Power Production. These engineers act as test directors for specific tests and function under the direction of the test program coordinator. Additional technical support is available from TVA's Division of Engineering Design, Division of Environmental Planning, and Westinghouse.

We have reviewed the preoperational and startup test program and conclude that it meets Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," November 1973, except as noted in item 1 below. We conclude, with the following exceptions, that the program described by the applicant is acceptable:

1. Plant Trip from 100 Percent Power Startup Test - The applicant has not provided a sufficiently detailed description of the test to enable the staff to conclude that a meaningful test will be performed. We have requested the applicant to provide additional information on the acceptance criteria for both the turbine trip and generator load reject portions of the test. We will review this information and report further in a supplement to this Report.

2. Regulatory Guide 1.68.2, Revision 1, July 1978, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants" and Regulatory Guide 1.108, Revision 1, August 1977, "Periodic Testing of Diesel Generator Units Used as Onsite Power Systems at Nuclear Power

Plants" - The applicant has not addressed these Regulatory Guides in its description of the initial test program. We have concluded that these guides are applicable to the Sequoyah test program. We require the applicant to address these guides and modify its test description to show that testing will be conducted in accordance with the guides, or provide technical justification for exceptions. We will report further on this matter in a supplement to this Report.

15.0 ACCIDENT ANALYSES

15.1 General

We and the applicant have evaluated the ability of the Sequoyah Nuclear Station to withstand normal and abnormal transients and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The core thermal power level is 3411 megawatts thermal, and the expected ultimate core power level is 3582 megawatts thermal. The applicant has based all core physics and core thermal-hydraulic information on the core thermal power level of 3411 megawatts. However, the applicant has designed the containment and other engineered safety features for operation at a core thermal power level of 3582 megawatts and has used this power level in analyzing certain postulated accidents in conformance with the siting guidelines of Section 100.0 of 10 CFR Part 100.

15.2 Normal Operation and Anticipated Operational Transients

The applicant has analyzed several events expected to occur one or more times in the life of the plant. It is to be demonstrated that all of these events are satisfactorily terminated without exceeding specified acceptable fuel design limits (departure from nucleate boiling ratio remains greater than 1.30 using the W-3 correlation) and the reactor coolant pressure stays below 110 percent of design.

The effect of rod bow on the departure from nucleate boiling heat flux was not included in the safety analyses. As discussed in Section 4.4, a penalty on the allowable enthalphy hot channel factors will be included in the technical specifications to correct for the rod bow effect on departure from nuclear boiling as a function of burnup. This penalty factor provides assurance that the minimum departure from nucleate boiling values predicted for the anticipated transients will not violate the fuel design limit of 1.30.

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

(1) Core power, ± 2 percent for calorimetric error

(2) Average reactor coolant system temperature (T_{AV}), ± four degrees Fahrenheit for deadband and measurement error

(3) Pressure (at pressurizer), ± 30 pounds per square inch for steady-state fluctuations and measurement error

These assumptions for initial conditions are acceptable because they are conservatively applied to produce the most adverse effects.

The transients analyzed are protected by the following reactor trips in accordance with General Design Criterion 20.

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

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

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control insertion, accounts for a struck rod in accordance with General Design Criterion 27.

The analysis methods for postulated transients and accidents are normally reviewed in a generic sense. In this regard, we have received submittals from Westinghouse for the loss-of-coolant accident, main steamline break accident, feedwater line accident, and rod ejection accident. The descriptions of the computer programs used in the analysis of these accidents have also been submitted.

The loss-of-coolant accident and rod ejection accident reviews have been completed and analysis methods were found acceptable. Our safety evaluation is documented in Letters dated August 28, 1973 and May 30, 1975. The steamline and feedline break reviews are presently underway. The status of the code reviews as well as the ongoing steamline break and feedline break reviews are discussed below:

1. The following topical reports have been approved:

    (a) WIT-6 (WCAP-7980) - Approved 8/30/76
    (b) THINC IV (WCAP-7956) - Approved 4/19/78
    (c) PHOENIX (WCAP-7973) - Approved 3/31/76

2. The LOFTRAN, FACTRAN, MARVEL and BLKOUT code topical reports are currently under review by the staff. These analysis methods are described in WCAPs-7907, 7908, 7909 and 7898, respectively. Our review of these topicals has progressed to the point that there is reasonable assurance that the

conclusions based on these analyses will not be appreciably altered by completion of the analytical review, and therefore that there will be no effect on the decision to issue an operating license. If the final approval of these topical reports indicates that any revisions to the analyses are required, Sequoyah will be required to implement the results of such changes.

3. Main Steamline and Feedline Breaks - Westinghouse has recently submitted topical reports which present their analysis methods and sensitivity studies for postulated main steamline and feedline breaks. This information is contained in WCAP-9226 for steamline breaks and WCAP-9320 for feedline breaks. In addition, WCAP-9236 was submitted which discusses the NOTRUMP computer program. This code is used in the analyses of the postulated feedline breaks. The review of these topical reports is scheduled for completion in late 1979.

The staff is currently reviewing the analysis methods for steam generator tube rupture and the various transients analyzed as Condition II and III events in Chapter 15. Our review at this time indicates that there is reasonable assurance that the conclusions based on these analyses will not be appreciably altered by completion of the analytical review, and therefore that there will be no effect on the decision to issue an operating license. If the final approval of the methods indicates that any revisions to the analyses are required, Sequoyah will be required to implement the results of such changes.

Based on previous acceptable analyses for Westinghouse plants, on comparison with other industry models, on independent staff audit calculations, and on previous startup testing experience, we conclude that with the exceptions noted above, the analytical methods for Sequoyah are acceptable for the operating license stage.

Analyses of abnormal transients have been submitted and show that the integrity of the reactor coolant system boundary is maintained and that the minimum departure from nucleate boiling ratio is about 1.3. These transients can be classified as reactivity insertions, loss-of-flow, system depressurization, and spurious operation of the emergency core cooling system. The results of these analyses indicate that the minimum margins occur for a rod withdrawal transient from 100 percent power (minimum departure from nucleate boiling ratio = 1.32) and a loss of load transient (peak pressure of 2525 pounds per square inch gauge).

For postulated boron dilution events during refueling, the applicant had relied upon an audible rate count to alert the operator. We find this unsatisfactory and require that the demineralized water source have its isolation valves locked closed with power removed during refueling to preclude a boron dilution event. In addition, the applicant has indicated that it will provide for staff review the procedures associated with generating the alarm set points for the high flux alarm which provides protection against a boron dilution event when starting up or shutting down. We will report further on the applicant's confirmatory information in a supplement to this report.

The applicant has analyzed postulated loss-of-flow transients accounting for flow coastdown due to pump flywheel inertia. The applicant has also demonstrated that the frequency decay during a grid collapse will not result in a more rapid flow coastdown than calculated.

We have reviewed the results of these analyses in accordance with Standard Review Plan Section 4.4, and find them acceptable because the fuel design limits and primary system pressure limits are not violated in any of these transients. We will report on the resolution of the boron dilution items noted above in a supplement to this report.

15.3 Accidents and Infrequent Transients

15.3.1 General

There are a number of transients and postulated accidents which are not expected during the life of any one plant. Within this group are events which form the design basis for the various barriers and safety systems. The acceptance criteria for accidents are provided in 10 CFR Part 50 and 10 CFR Part 100. Infrequent events are judged primarily on the basis of 10 CFR Part 100.

In the analysis of these events, the applicant must investigate a broad spectrum of related events to determine the bounding case, including the worst single active failure. Sensitivity studies are performed to identify parameters for initial conditions and appropriate credit for systems and their performance during the limiting events in terms of protection of various barriers.

The emergency core cooling system includes an upperhead injection subsystem intended to produce acceptable thermal hydraulic conditions in the core following a postulated loss-of-cooling accident. A proposed evaluation model has been under staff review since 1974 for compliance with 10 CFR 50, Appendix K requirements. An acceptable evaluation model has been defined by the staff.

15.3.2 Loss-of-Coolant Accident

The applicant has submitted an analysis of a spectrum of postulated primary system pipe breaks including the identification and justification of the worst single failure. See Section 6.3 for a discussion of the loss-of-coolant accident analysis.

In addition to the matters discussed in Section 6.3, the applicant has submitted an evaluation of pressure vessel integrity following repressurization after a small-break loss-of-coolant accident which indicates that faulted vessel stress criteria may not be satisfied after 27 years of reactor operation. We have requested additional information to confirm the suitability of the analysis and will report further on this matter in a supplement to this report. If necessary, the operating license will be conditioned appropriately.

15.3.3 Steam Line Break Accident

The applicant has submitted analyses of postulated steam line breaks that show no additional fuel failures attributed to the accident. These results are similar to those obtained for previously reviewed Westinghouse four-loop plants. The staff has requested additional information to verify operator actions. We will review the requested confirmatory information when submitted and report further in a supplement to this report.

15.3.4 Feed Line Break Accident

A major rupture of a main feedwater pipe between the steam generator and check valve was analyzed as a reactor coolant system heatup transient. Since the feedwater line rupture has the potential of reducing the ability of secondary systems to remove the heat generated by the core, the auxiliary feedwater system is provided to assure that adequate feedwater will be available such that no substantial overpressurization of the reactor coolant system shall occur and the reactor core shall remain covered at all times. The analysis indicates that the reactor core will remain covered throughout the accident, that sufficient auxiliary feedwater capacity is provided, assuming the worst single failure, to remove the decay of heat from the core, and that the relief capacity of the pressurizer safety valves are sufficient to prevent reactor coolant system overpressurization.

15.3.5 Locked Rotor Accident

The locked rotor accident was analyzed by postulating an instantaneous seizure of one reactor coolant system pump rotor. The reactor flow would decrease rapidly and a reactor trip would occur as a result of a low flow signal. A thermal analysis of the hot rod in the core was performed and revealed a maximum cladding temperature of 2017 degrees Fahrenheit. The peak reactor coolant system pressure (corresponding to this cladding temperature) during the locked rotor accident (2632 pounds per square inch absolute) did not exceed the code pressure limits (110 percent of reactor coolant system design pressure, 2750 pounds per square inch absolute) and the integrity of the reactor coolant system pressure boundary was maintained.

15.3.6 Control Rod Misalignment

Rod cluster control assembly misalignment accidents including a dropped full length rod cluster control assembly, dropped full length rod cluster control assembly bank, and a misaligned full length rod cluster control assembly, and the withdrawal of a single assembly while at power, have been analyzed by the applicant. The analysis was performed using the TURTLE code to determine X-Y peaking factors. The THINC IV code was then used to calculate departure from nucleate boiling ratio. For the transient response to a dropped rod cluster control assembly or rod cluster control assembly bank, the LOFTRAN code is used.

Misaligned rods are detectable by the following means: (1) asymmetric power distributions sensed by external nuclear instrumentation or core exit thermocouples, (2) rod deviation alarm, and (3) rod position indicators. A deviation of a rod from its bank by about 15 inches, or twice the resolution of the rod position indicator, will not cause power distributions to exceed design limits. Administrative controls are provided to assure rod alignment if one or more rod position channels are out of service. In the event of a dropped rod cluster control assembly, the automatic controller may return the reactor to full power. Analysis indicates that a departure from nucleate boiling will not occur during this event. For the case of dropped rod cluster control assembly groups, the reactor is tripped by the power range negative neutron flux trip and the reactor is shut down without violating fuel integrity. For cases where an rod cluster control assembly group is inserted to its insertion limit with a single rod cluster control assembly in the group fully withdrawn, analysis indicates that departure from nucleate boiling will not occur.

We have reviewed the calculated estimates of the expected reactivity and power distribution changes that accompany postulated misalignments of representative assemblies. We conclude that the values used in this analysis conservatively bound the expected values including calculational uncertainties.

The inadvertent withdrawal of a single assembly requires multiple failures in the control system, multiple operator errors, or deliberate operator actions combined within a single failure of the control system. As a result the single assembly withdrawal is classified as an infrequent occurrence. The resulting transient is similar to that due to a bank withdrawal but the increased peaking factor may cause departure from nucleate boiling to occur in the region surrounding the withdrawn assembly. Less than five percent of the rods in the core experience departure from nucleate boiling.

Comparisons of calculations of the power distributions for the normal fuel loading pattern and five cases of fuel assembly and burnable poison misloadings are presented by the applicant. These represent the spectrum of probable inadvertent improper loadings. With the exception of a case involving an interchange of region 1 and 2 assemblies near to the center of the core, the resultant distortion of the power distribution would be detectable by the instrumentation provided. In the excepted case, the distortion of power distribution is sufficiently small that the increase in the total peaking factor would be approximately the uncertainty in the measurement of that value and hence cause no safety problems.

Incore instrumentation using movable fission chamber detectors is provided that would detect the loading mistake. A power distribution measurement using this system is required by the technical specifications to determine if misloadings exist. Thermocouples in approximately one-third of the fuel assemblies would also provide an indication of a loading mistake. In most cases, an improperly loaded fuel assembly could cause a quadrant power tilt that would be detected by the excore nuclear instrumentation. In addition to the instrumentation system to detect misloading, strict administrative controls are provided to prevent such an event.

We conclude that an improperly loaded fuel assembly or burnable poison cluster that would cause a significant safety problem could be detected by the instrumentation provided.

15.3.7 Control Rod Ejection

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a rod cluster control assembly. For assemblies initially inserted, the consequences of this would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made so that this accident would be extremely unlikely, the applicant has analyzed the consequences of such an event.

Methods used in the analysis are reported in WCAP-7588, Revision 1, "An Evaluation of the Rod Ejection Accident in Westinghouse Reactors Using Spatial Kinetics Methods," which has been reviewed and accepted by the staff in our letter to Westinghouse dated August 28, 1973. This report demonstrates that the model used in the accident analysis is conservative relative to a three-dimensional kinetics calculation.

The applicant's criteria for gross damage of fuel are a maximum clad temperature of 2700 degrees Fahrenheit and an energy deposition of 200 calories per gram in the hottest pellet. These criteria are more conservative than those proposed in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." Regulatory Guide 1.77 has an acceptance criterion of 280 calories per gram energy deposition and no criterion for clad temperature other than that implicit in requirements for fuel and pressure vessel damage. Therefore, we conclude the criteria are acceptable.

Four cases were analyzed: beginning-of-cycle at 102 percent and zero power and end-of-cycle at 102 percent and zero power. The highest clad temperatures, 2420 degrees Fahrenheit, was reached in the zero power end-of-cycle case and the highest fuel enthalpy, 177 calories per gram was reached in the beginning-of-life full power case. The analysis also shows that less than 10 percent of the fuel experiences departure from nucleate boiling and less than 10 percent of the hot pellet melts.

Analyses have been performed to show that the pressure pulse produced by the rod ejection will not stress the reactor coolant system boundary beyond emergency limits. Further analyses have shown that a cascade effect is not credible.

The ejected rod worths and reactivity coefficients used in the analysis have been reviewed and have been judged to be conservative. Also the assumptions and methods of analysis used by the applicant are in accordance with or are more conservative than those recommended in the Regulatory Guide 1.77. Therefore, we conclude that this analysis is acceptable.

15.3.8 Anticipated Transients Without Scram

A number of plant transients can be affected by a failure of the scram system to function. For a pressurized water reactor, the most important transients affected include loss of normal feedwater, loss of electrical load, inadvertent control rod withdrawal, and loss of normal electrical power. In September 1973, we issued WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," establishing acceptance criteria for anticipated transients without scram. In conformance with the requirements of Appendix A to WASH-1270, Westinghouse submitted an evaluation of anticipated transients without scram in Topical Report WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis." On December 9, 1975, we issued our "Status Report on Anticipated Transients Without Scram for Westinghouse Reactors." Since the publication of the 1975 status report, Westinghouse has submitted additional anticipated transients without scram (ATWS) analyses.

Based on review of these reports and discussions with Westinghouse and other vendors, the Nuclear Regulatory Commission Division of Systems Safety has published a report on "Anticipated Transients Without Scram for Light-Water Reactors," NUREG-0460, April 1978 (Vol. I and II) and December 1978 (Vol. III).

In a pressurized water reactor, the anticipated transients which require prompt action to shut down the reactor in order to avoid plant damage and possible offsite effects can be classified in two groups: those that isolate the reactor from the heat sink, and those that do not. (A list of these transients is included in Appendix IV of Volume II of NUREG-0460.) In general, the consequences of both of these types of events are an increase in reactor power or system pressure, or both. In Section 6.3 of NUREG-0460, Volume I, potentially unacceptable consequences of ATWS events for pressurized water reactors of designs like Sequoyah are indicated to include (1) pressure rises that could threaten the integrity of the reactor coolant pressure boundary, (2) loss of core cooling, and (3) leakage of radioactive material from the facility.

In NUREG-0460, the staff concluded that for plants which fall within the envelope of the Westinghouse generic ATWS analyses, the ATWS acceptance criteria will not be violated if the actuation circuitry of turbine trip and auxiliary feedwater systems which are relied upon to mitigate ATWS consequences are sufficiently reliable and are separate and diverse from the reactor protection system. Additionally, the functionability of valves required for long-term cooling following the postulated ATWS events has to be demonstrated. This has been essentially completed for Westinghouse plants in the course of earlier generic ATWS reviews by the staff.

The turbine trip and auxiliary feedwater actuation for Sequoyah are both tied into the reactor protection system circuitry and could both be affected by assumed common mode failures in portions of the reactor protection system. For this reason, the staff believes that the actuation circuitry for these systems should be diverse from the circuitry of the reactor protection system. We believe these changes can be accommodated during any scheduled shutdown. The staff's review of Westinghouse analyses suggests that these are the only plant modifications that may be needed to satisfy the NUREG-0460 criteria.

As discussed below, the turbine trip and the auxiliary feedwater system can also be actuated manually from the control room.

The Commission considers ATWS to be an unresolved safety issue. However, the staff has proposed the type of plant modifications which, if provided, would reduce ATWS risk to an acceptable level. Volume 3 of NUREG-0460, which describes the rationale for specifying these plant modifications, is currently being reviewed by the Advisory Committee on Reactor Safeguards. The NRC's Regulatory Requirements Review Committee has completed its review and concurred with the staff approach described in Volume 3 of NUREG-0460 insofar as it applies to Sequoyah. The staff has issued requests for the industry to supply generic analyses to confirm the ATWS mitigation capability described in Volume 3 of NUREG-0460.

The staff plans to present its recommendations for rulemaking on ATWS to the Commission in May 1979, including the recommendations for modifications contained in Volume 3 of NUREG-0460. The Commission would by rulemaking determine required modifications to resolve ATWS concerns as well as the required schedule for implementation of such modifications. Sequoyah would, of course, be subject to the Commission decision in this matter.

The following discusses the bases for operation of Sequoyah in the interim period while final resolution of ATWS is before the Commission.

In NUREG-0460, Volume 3, the staff states:

> "The staff has maintained since 1973 (for example, see pages 69 and 70 of WASH-1270) and reaffirms today that the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is

no undue risk to the public from ATWS. This conclusion is based on engineering judgment in view of: (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of scram failure; (b) the favorable operating experience with current scram systems; and (c) the limited number of operating reactors."

In view of these considerations and our expectation that the necessary plant modifications will be implemented in 2 to 5 years following Commission rulemaking on ATWS, the staff has generally concluded that pressurized water reactor plants can continue to operate because the risk from ATWS events in this time period is acceptably small. As a prudent course, in order to further reduce the risk from ATWS events during the interim period before completing the plant modifications determined by the Commission to be necessary, the staff requires that the following steps be taken:

(1) Emergency procedures be developed to train operators to recognize an ATWS event, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicators, pressurizer relief valve and safety valve indicators, coolant average temperature, containment temperature and pressure indicators, steam generator level, pressure and flow indicators, and any other alarms annunciated in the control room with emphasis on alarms not processed through the electrical portion of the reactor scram system.

(2) Operators be trained to take actions in the event of an ATWS, including consideration of manually scramming the reactor by using the manual scram button, prompt actuation of the auxiliary feedwater system to assure delivery of the full capacity of this system, and initiation of turbine trip. The operator should also be trained to initiate boration by actuation of the high pressure safety injection system to bring the plant to a safe shutdown condition.

The staff considers these procedural requirements an acceptable basis for interim operation of the Sequoyah plant based on our understanding of the plant response to postulated ATWS events. We will require that the applicant implement in a timely fashion any additional modifications that may be determined as a result of the Commission's future rulemaking proceedings on ATWS.

15.3.9 Summary

In summary, we reviewed the safety analyses for the Sequoyah plant. The scope of review included the description of the event, the methods used in the analyses, the values of the parameters used as input to the analyses, and the design criteria. The review concentrated on the differences between the safety analyses and those previously reviewed and found acceptable by the staff.

We conclude that the safety analyses presented are acceptable for demonstrating conformance of the core and reactor coolant system transient response with the Commission's regulations and staff technical positions, subject to acceptable resolution of the issues in Sections 15.3.2 and 15.3.3 dealing with loss-of-coolant accident and steam line break. A staff review of these events and a resolution of any issues arising from these reviews will be discussed in a supplement to this report.

15.4 Radiological Consequences Of Accidents

The postulated design basis accidents analyzed by the applicant to determine the offsite radiological consequences are the same as those analyzed for previously licensed pressurized water reactor plants. To evaluate the effectiveness of the engineered safety features proposed for the Sequoyah Nuclear Plant and to assure that the radiological consequences of these accidents meet the applicable dose criteria, we have analyzed the loss-of-coolant accident, the fuel handling accident, the steam line break accident, the steam generator tube rupture accident, and the control rod ejection accident. The calculated doses for these accidents are shown in Table 15-1.

On the basis of our evaluation of the secondary side accidents (steamline break, steam generator tube rupture and rod ejection accident), using the values of primary and secondary coolant activity concentrations and primary-to-secondary leakage as given by the Standard Technical Specifications, we have concluded that the consequences of these events will be acceptably controlled and mitigated by these limits. We will, therefore, include in the Technical Specifications limits on values of primary and secondary coolant activity concentrations and primary-to-secondary leak rate.

15.4.1 Loss-of-Coolant Accident

The Sequoyah Nuclear Plant includes a double containment design to collect and filter the leakage of fission products from a postulated design basis loss-of-coolant accident. The double containment consists of a free-standing steel primary containment vessel surrounded by a reinforced concrete shield building. The reinforced concrete auxiliary building is also a part of the secondary containment barrier. Leakage which enters the secondary containment is treated by either the emergency gas treatment system or the auxiliary building gas treatment system prior to release to the atmosphere. Both of these systems are engineered safety features. Another engineered safety feature is the ice condenser with a sodium tetraborate additive to the ice to enhance the removal of iodine in the containment following a loss-of-coolant accident. The principal assumptions employed in our analysis are listed in Table 15-2. The dose model and dose conversion parameters are consistent with those given in Regultory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

TABLE 15-1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Accident	Exclusion Area 2-Hour Dose, Rem Thyroid	Exclusion Area 2-Hour Dose, Rem Whole Body	Low Population Zone** 30-Day Dose, Rem Thyroid	Low Population Zone** 30-Day Dose, Rem Whole Body
Loss of Coolant	32	6	10	<1
Fuel Handling	20	1	<1	<1
Steam Line Break				
1) I-131 at 1 microcurie per gram	13	<0.1	<1	<0.1
2) I-131 at 60 microcurie per gram	26	<0.1	1	<0.1
Steam Generator Tube Rupture				
1) I-131 at 1 microcurie per gram	19	<0.1	1	<0.1
2) I-131 at 60 microcurie per gram	214	<0.1	10	<0.1
Control Rod Ejection				
1) Leakage through secondary side	42	<0.1	2	<0.1
2) Leakage through containment	97	<0.1	4	<0.1

Exclusion area minimum boundary distance = 556 meters

**Low population zone distance = 4828 meters

TABLE 15-2

ASSUMPTIONS USED IN THE CALCULATION OF LOSS-OF-COOLANT ACCIDENT DOSES

Power Level	3582 Megawatts thermal
Operating Time	3 years
Fraction of Core Inventory Available for Leakage	
Iodines	25 percent
Noble Gases	100 percent
Initial Iodine Composition in Containment	
Elemental	91 percent
Organic	4 percent
Particulate	5 percent
Primary Containment Volumes	
Upper Containment	7.16×10^5 cubic feet
Lower compartment (including ice condenser)	5.25×10^5 cubic feet
Shield Building Annulus Volume	3.75×10^5 cubic feet
Mixing Fraction in Annulus	50 percent

Annulus Ventilation Flow Distribution

Time Step	Recirculation Flow cubic feet per minute	Exhaust Flow, cubic feet per minute
0-46 seconds	0	0
46-200 seconds	500	3500
200-400 seconds	1500	2500
400-1000 seconds	3000	1000
1000 seconds - 30 days	3900	100

Filter Efficiencies	
Elemental Iodine	95 percent
Organic Iodine	95 percent
Particulate Iodine	95 percent
Ice Condenser Removal Efficiency	
Elemental Iodine	30 percent
Flow Rate thru Ice Condenser	40,000 cubic feet per minute
Period of Ice Condenser Effectiveness	10-60 minutes

TABLE 15-2 (Cont'd)

Primary Containment Leak Rates	
0 - 24 Hours	0.25 percent per day
> 24 Hours	0.125 percent per day
Bypassing Leakage Fraction	0 percent
Minimum Exclusion Area Boundary Distance	556 meters
Low Population Zone Distance	4828 meters
Atmospheric Diffusion (X/Q) Values	
0-2 hours	1.4×10^{-3} seconds per cubic meter
0-8 hours	6.4×10^{-5} seconds per cubic meter
8-24 hours	4.5×10^{-5} seconds per cubic meter
1-4 days	2.1×10^{-5} seconds per cubic meter
4-30 days	6.9×10^{-6} seconds per cubic meter

In the analysis of the design basis loss-of-coolant accident, the primary containment was assumed to leak at the design leak rate of 0.25 percent per day for the first 24 hours following the accident and at 0.125 percent per day thereafter. The applicant established to our satisfaction that the shield building annulus and auxiliary building pressure would not exceed -0.25 inch water gauge pressure and that no leakage would bypass the gas treatment systems throughout the course of the accident (see Section 6.2 of this report for further discussion of these items). Ten percent of the leakage from the primary containment enters the auxiliary building following the accident and we assumed that this leakage was exhausted to the atmosphere through the auxiliary building gas treatment system without credit for holdup or mixing in the auxiliary building. Ninety percent of the leakage from the primary containment enters the shield building annulus where we assumed that it went directly to the intake of the shield building annulus recirculation/ exhaust system. Following passage through the emergency gas treatment system filters, a fraction of this leakage was assumed in our analysis to be exhuasted to the atmosphere with the remainder recirculated to the shield building annulus where credit was given for mixing in 50 percent of the annulus free volume. The split between the exhaust and recirculation fractions was assumed to be proportional to the air flow rates in the exhaust and recirculation paths of the systems.

The doses we calculate for the postulated design basis loss-of-coolant accident for the Sequoyah Nuclear Plant, shown in Table 15-1 are well within the exposure guidelines of 10 CFR Part 100.

As part of the loss-of-coolant accident, we have also evaluated the consequences of leakage of containment sump water which is circulated by the emergency core cooling system after that postulated accident. We have assumed the sump water contains a mixture of iodine fission products in agreement with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." During the recirculation mode of operation the sump water is circulated outside of the containment to the auxiliary building. If a source of leakage should develop, such as from a pump seal failure, a fraction of the iodine in the water could become airborne in the auxiliary building and exit to the atmosphere. Since the emergency core cooling system area in the auxiliary building is served by an engineered safety feature air filtration system (the auxiliary building gas treatment system), we conclude that the doses resulting from the postulated leakage of recirculation water would be low and, when added to the direct leakage loss-of-coolant accident doses, would result in total doses that are within the guideline values of 10 CFR Part 100.

The applicant may purge the containment periodically during reactor operation. Should a loss-of-coolant accident occur when the purge lines are open, a portion of the containment atmosphere plus a portion of any flashed reactor coolant containing radioactive iodine fission products would be released to the environment in the short interval before the purge isolation valves close and isolate the containment. We have estimated the radiological consequences of this event using conservative assumptions regarding the radioactive iodine concentration in the primary coolant, the amount of reactor coolant inventory released, and the flow rate through the valves. We conclude that the consequences are such, that even when added to the calculated doses from containment leakage, the total is within the guideline values of 10 CFR Part 100.

The applicant has provided redundant hydrogen recombiners for the purpose of controlling any accumulation of hydrogen within the primary containment following a loss-of-coolant accident. In the event of failure of both recombiners, the applicant has provided a backup system. The purged containment effluent would flow to the shield building annulus where it would be subsequently discharged to the atmosphere through the emergency gas treatment system filters. We find the combination of redundant recombiners plus a back-up purge capability to be an acceptable method for controlling the potential contribution to the offsite doses from hydrogen purging following a loss-of-coolant accident.

15.4.2 Fuel Handling Accident

For the analysis of the fuel handling accident, we have assumed that a fuel assembly was dropped in the fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged thereby releasing the volatile fission gases from the fuel rod gaps into the pool. The radioactive material that escaped from the fuel pool was assumed to be released to the environment over a two-hour time period with the iodine activity reduced by filtration through the auxiliary building gas treatment system. The dose results are shown in Table 15-1 and the assumptions and parameters used in the analysis are shown in Table 15-3. The dose model and dose conversion factors employed in the analysis were the same as those given in Regulatory Guide 1.25, 'Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors".

We have also evaluated the consequences of a fuel handling accident inside primary containment. The applicant states that at all times during refueling operations the containment will either be isolated or ventilated to the atmosphere through the reactor building purge ventilation system. This is an engineered safety feature system with filter efficiencies equivalent to those in the auxiliary building gas treatment system. The assumptions regarding the reactor shutdown time, the number of fuel rods damaged, the iodine decontamination efficiency of the water in the refueling cavity, and the atmospheric dispersion factors are the same for the fuel handling accident inside containment as those assumed for the fuel handling accident in the fuel pool area in the auxiliary building. Therefore, we conclude that the doses resulting from a postulated fuel handling accident inside containment during ventilation through the reactor building purge ventilation system will be identical to the doses calculated for the fuel handling accident in the fuel pool and are well within the dose guidelines of 10 CFR Part 100.

TABLE 15-3

ASSUMPTIONS USED IN THE FUEL HANDLING ACCIDENT ANALYSIS

Power Level	3582 megawatts thermal
Number of Fuel Rods Damaged	264
Total Number of Fuel Rods in Core	50,952
Radial Peaking Factor of Damaged Rods	1.65
Shutdown Time	100 hours
Inventory Released From Damaged Rods (Iodines and Noble Gases)	10 percent
Pool Decontamination Factors	
Iodines	100
Noble gases	1
Iodine Fractions Released From Pool	
Elemental	75 percent
Organic	25 percent
Filter Efficiencies for Iodine Removal	
Elemental	95 percent
Organic	95 percent
0-2 hour X/Q Value at 556 meters	1.4×10^{-3} seconds per cubic meter
0-8 hour X/Q Value of 4828 meters	6.4×10^{-5} seconds per cubic meter

15.4.3 Steam Line Break Accident

Both we and the applicant have evaluated the radiological consequences of a postulated steamline break accident occurring outside containment and upstream of the main steam isolation valve. Although the contents of the secondary side of the affected steam generator would be vented initially to the atmosphere as an elevated release, we have conservatively assumed that the entire release throughout the course of the accident is released under ground level conditions.

The applicant has indicated that no departure from nucleate boiling is exected to occur and, therefore, no cladding failure is assumed in our calculation. However, as a result of the power and pressure transient, we assumed that an iodine spike occurred in which the iodine release rate from fuel to coolant is increased by a factor of 500.

During the course of the accident, the shell side of the steam generator was assumed to stay dry since auxiliary feedwater flow to the affected steam generator would be blocked off under the conditions of this accident. Due to the dry-out condition, all iodine transported to the secondary side by leakage was assumed available for release to the atmosphere with no reduction due to holdup or attenuation. Other assumptions on listed in Table 15-4.

We performed an evaluation using the acceptance criteria and procedures given in Standard Review Plan Section 15.1.5, Appendix (Rev. 1). The doses we calculted are shown in Table 15-1, and are well within the guideline values of 10 CFR Part 100.

15.4.4 Steam Generator Tube Rupture Accident

A non-mechanistic guillotine break of a steam generator tube is postulated to occur when the reactor is at power. Because of the rupture, the primary coolant boundary is breached. The initial leak rate of primary coolant through each end of the broken tube is 32 pounds per second (using a discharge coefficient of 0.61) and gradually decreases as the pressure difference between the reactor vessel and the steam generator is reduced. This leak rate is larger than the maximum capacity of the charging pumps to maintain inventory and of the pressurizer heaters to maintain pressure. Thus both pressure and level in the pressurizer would decrease. At about 15 minutes post-accident, either the low pressurizer pressure trip or the low pressurizer level set-point is reached. The resultant reactor and turbine trip immediately terminate power output to the grid. This disturbance to the grid is assumed to cause loss of offsite power to the plant.

With loss of offsite power, plant cool down is effected by a combination of the operation of automatic safety valves and manual atmospheric relief valves. Diagnosis of the accident is achieved by observing annunciations of condenser high radiation alarm, steam generator feedwater/steam flow mismatch, decreasing

TABLE 15-4

ASSUMPTIONS USED FOR STEAMLINE BREAK ACCIDENT

1. Power = 3582 Megawatts thermal

2. Pre-accident dose-equivalent I-131 in primary coolant = 1.0 microcurie per gram and 60.0 microcurie per gram (two cases analyzed).

3. Primary-to-secondary leak rate, as limited by Technical Specifications, 1.0 gallon per minute.

4. All of the one gallon per minute leak occurs in the affected steam generator.

5. All the iodine transported to the shell side of the steam generator by the leakage is lost to the environment without delay.

6. Iodine release rate from fuel increases by a factor of 500 as a result of the accident.

7. X/Q values:

   0-2 hours at 556 meters = 1.4×10^{-3} seconds per cubic meter
   0-8 hours at 4828 meters = 6.4×10^{-5} seconds per cubic meter

pressurizer pressure and level, and increasing level in one steam generator. We assumed, conservatively, that it would take the operator 30 minutes from the onset of tube rupture to diagnose the accident and isolate the affected steam generator. The steam generator is isolated when its emergency feedwater is terminated and its steam relief valves are closed. Meanwhile, the primary system is expected to have been depressurized such that the leak through the broken tube has terminated.

Before reactor trip and steam venting, some iodine is removed from the condensate by the air-ejector and released to the environment. However, radioactivity released via this route is small due to preferential retention of iodine in the condensate. At 15 minutes post-accident, reactor trip and loss of offsite power terminate availability of the condenser and steam is vented through the safety/ relief valves as mentioned earlier. At this time, leakage through the broken tube is down to 23 pounds per second through each end. Only the jet of coolant pointing upward at the steam outlet contributes directly to offsite doses. This jet of water is assumed atomized into droplets having diameters in the micron range, and carried by steam through the safety/relief valves to the environment. Some of the droplets, carrying iodine at the same level as that of the primary coolant, would be captured by the surrounding water in the steam generator, by the steam separator/ dryer and by other internal hardware (see NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident"). We conservatively assumed that 50 percent of the iodine in this jet of water is released to the environment. The doses we calculated, using the criteria and procedure described in Standard Review Plan Section 15.6.3, Revision 1, are within the guideline values of 10 CFR Part 100. These doses are presented in Table 15-1. Our other assumptions are given in Table 15-5.

15.4.5 Control Rod Ejection Accident

A non-mechanistic rupture of a control rod drive housing is postulated. Because of the resultant opening in the pressure vessel, primary coolant is lost to the containment with concurrent rapid depressurization of the reactor pressure vessel. Reactor trip, initiated by one of several trip signals, occurs rapidly.

Ejection of a control rod results in rapid reactivity insertion. The applicant has calculated and conservatively assumed that 10 percent of the fuel elements will experience cladding failure, releasing all their gap radioactivity. The released radioactivity is mixed immediately with the primary coolant. Activity release to the environment may occur via each of two pathways. The first pathway involves a release of activity of the primary containment which is then assumed to leak to the atmosphere as in the design basis loss-of-coolant accident, except that the containment pressure is considered insufficient to effect iodine removal from the ice condensers. In the second pathway, activity is transferred from the primary to the secondary coolant via an assumed one gallon per minute primary-to-secondary

TABLE 15-5

ASSUMPTIONS USED FOR STEAM GENERATOR TUBE RUPTURE ACCIDENT

1. Power = 3582 Megawatts thermal
   Initial primary system pressure = 2100 pounds per square inch
   Initial primary system temeprature = 577 degrees Fahrenheit

2. Pre-accident dose equivalent I-131 in primary coolant = 1.0 microcuries per gram and 60 microcuries per gram (two cases analyzed).

3. Initial leak rate through each end of broken tube = 32 pounds per second.

4. Leak rate through each end of broken tube at 15 minutes = 23 pounds per second.

5. Isolation of affected steam generator at 30 minutes
   (Steam venting occurs from 15 to 30 minutes)

6. Iodine release rate from fuel increases by a factor of 500 at reactor trip.

7. X/Q value:
   0-2 hours at 556 meters = 1.4×10^{-3} seconds per cubic meter
   0-8 hours at 4828 meters = 6.4×10^{-5} seconds per cubic meter

leak rate. With loss of offsite ; wer and subsequent steam venting (see Section 15.4.4), some of the iodine transferred to the shell side is available for leakage to the environment.

In calculating the consequences of this postulated event, we calculated the doses as if all the activity was released via each of the above pathways. We would expect the actual consequences to be some combination of these pathways. Our other assumptions are shown in Table 15-6. We conclude that the calculated doses, as shown in Table 15-1, are well within the guideline values of 10 CFR Part 100.

15.4.6 Waste Gas Decay Tank Accident

The radioactive waste gas decay tanks are designed to seismic Category I requirements. Therefore, the total failure of these tanks is sufficiently improbable that 10 CFR Part 100 guideline doses are applicable. Our calculations indicate tht the doses for failure of these tanks would be small fraction of the 10 CFR Part 100 guidelines. Appropriate technical specifications will be placed on the maximum activity that can be stored in any one tank at any time such that single failure of active components, including the lifting or sticking of a safety or relief valve, will not result in radiological consequences that exceed small fractions of 10 CFR Part 100 guideline doses.

15.4.7 Liquid Tank Failure Accident

We evaluated the consequences of tank failures for tanks located outside the reactor containment which could result in releases of liquids containing radioactive materials to the environs. Considered in our evalution were (1) the radionuclide inventory in each tank assuming a one percent operating power fission product source term, (2) a tank liquid inventory equal to 80 percent of its design capacity, (3) the mitigating effects of plant design including overflow lines and the location of indoor and outdoor storage tanks in curbed areas designed to retain spillage, and (4) the effects of site geology and hydrology.

The applicant has incorporated provisions in the design to retain releases from liquid overflows as discussed in Section 11 of this Report. In our evaluation we assumed the flow of ground water will move in the direction of the Chickamauga Reservoir. The water supply for the metropolitan Chattanooga area is taken from the Nickajack Reservoir, at a point approximately 19 miles downstream from the site and six miles downstream from the Chickamauga Dam. We calculated the liquid transit time for radwaste leakage to the Chickamauga Reservoir to be 303 days. The combined dilution factor resulting from mixing with the ground water and mixing with the reservoir water prior to reaching the nearest potable water supply is approximately 1×10^6.

TABLE 15-6

ASSUMPTION USED FOR CONTROL ROD EJECTION ACCIDENT

1. Power = 3582 Megawatts thermal
   Volume of primary coolant = 11800 cubic feet

2. Primary-to-secondary leak rate is 1.0 gallons per minute as limited by Technical Specifications.

3. Ten percent of the fuel rods experience cladding failure, releasing all their gap radioactivity. The released activity is mixed immediately with the primary coolant.

4. A fraction of the iodine transported to the shell side of steam generators is lost to the environment. This fraction is equal to the maximum flash fraction of the leakage.

5. Ten percent of the iodine transported to and mixed with the secondary coolant is lost during the course of he accident.

6. Primary system depressurized in 20 minutes, terminatng primary-to-secondary leak.

7. Fifty percent of the iodine released into the containment is plated out instantaneously.

8. Containment leak rate = 0.25 percent per day.

9. X/Q values:
   0-2 hours at 556 meters = 1.4×10^{-3} seconds per cubic meter
   0-8 hours at 4828 meters = 6.4×10^{-5} seconds per cubic meter

Based on our evaluation, we conclude that a rupture of the tank will give a concentration at the intake of less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2. Based on the foregoing evaluation, we conclude that the provisions incorporated in the applicant's design to mitigate the effects of component failures involving contaminated liquids are acceptable.

16.0 TECHNICAL SPECIFICATIONS

The technical specifications of a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. The finally-approved technical specifications will be made a part of the operating license. Included will be sections covering safety limits limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

At the time of submittal of the Final Safety Analysis Report, the applicant had proposed technical specifications in Chapter 16. Shortly thereafter, we informed the applicant that we intended to use the Standard Technical Specifications for Westinghouse Pressurized Water Reactors as the basis for development of the final technical specifications for the Sequoyah plant.

The Westinghouse Standard Technical Specifications to be used as the basis for the Sequoyah technical specifications have been updated as a result of their application to technical specifications for other plants and of continued extensive discussion with Westinghouse and applicants with Westinghouse reactors.

We have worked with the applicant and have prepared a draft of the technical specifications for the Sequoyah plant. On the basis of our review to date, we conclude that normal plant operation within the limits of the technical specifications will not result in offsite exposures in excess of the 10 CFR Part 20 limits. Furthermore, the limiting conditions for operations and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

17.0 QUALITY ASSURANCE

17.1 General

The description of the quality assurance program for the operations phase of the Sequoyah Nuclear Plant is contained in Section 17.2 of the Final Safety Analysis Report through Amendment 58. Section 17.2 references the Tennessee Valley Authority topical report TVA-TR-75-1A, "Quality Assurance Program Description for Design, Construction, and Operation for Nuclear Power Plants." Our evaluation of this quality assurance program is based on a detailed review of this information and discussions with representatives of the Tennessee Valley Authority to determine how the quality assurance program for the operations phase complies with the requirements of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the applicable regulatory guidance which is listed in Table 17-1.

17.2 Organization

The organization responsible for the operation of the Sequoyah Nuclear Plant and for the establishment and execution of the operations phase quality assurance program is shown in Figure 17-1. During the operations phase, the Office of Engineering Design and Construction provides design, procurement, construction, and installation support. This office includes a quality assurance organization at the "Office" level as shown in Figure 17-1.

The Office of Power has overall responsibility for the Tennessee Valley Authority power program, including operation of the Sequoyah Nuclear Plant; and the Manager of Power has overall responsibility for quality assurance during startup and operation of the plant. The quality assurance program is implemented under the direction of the Assistant Manager of Power who has responsibility for establishing quality policies, goals, and objectives. This responsibility is carried out through the Office of Power Quality Assurance Manager and the Directors of other involved divisions. The Quality Assurance Manager is responsible to develop, coordinate, monitor, audit, and evaluate the quality assurance program to meet regulatory requirements and guidance as well as licensing commitments.

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

TABLE 17-1

REGULATORY GUIDANCE APPLICABLE TO

QUALITY ASSURANCE PROGRAMS

1. Regulatory Guide 1.8 (Revision 1 - September 1975), "Personnel Selection and Training."

2. Regulatory Guide 1.30 (Revision 0 - August 1972), "Quality Assurance Requirements for Installation, Inspection, and Testing of Instrumentation and Electric Equipment."

3. Regulatory Guide 1.33 (Revision 1 - January 1977), "Quality Assurance Program Requirements (Operation)."

4. Regulatory Guide 1.37 (Revision 0 - March 1973), "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

5. Regulatory Guide 1.38 (Revision 2 - May 1977), "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants."

6. Regulatory Guide 1.39 (Revision 1 - October 1976), "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

7. Regulatory Guide 1.58 (Revision 0 - August 1973), "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel."

8. Regulatory Guide 1.64 (Revision 2 - June 1976), "Quality Assurance Requirements for the Design of Nuclear Power Plants."

9. Regulatory Guide 1.74 (Revision 0 - February 1974), "Quality Assurance Terms and Definitions."

10. Regulatory Guide 1.94 (Revision 1 - April 1976), "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

11. Regulatory Guide 1.116 (Revision 0-R - May 1977), "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems."

12. Regulatory Guide 1.123 (Revision 1 - July 1977), "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants."

13. ANSI N45.2.9 (Draft 11, Revision 0 - January 1973), "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants."

14. ANSI N45.2.12 (Draft 3, Revision 4 - February 1974), "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants."

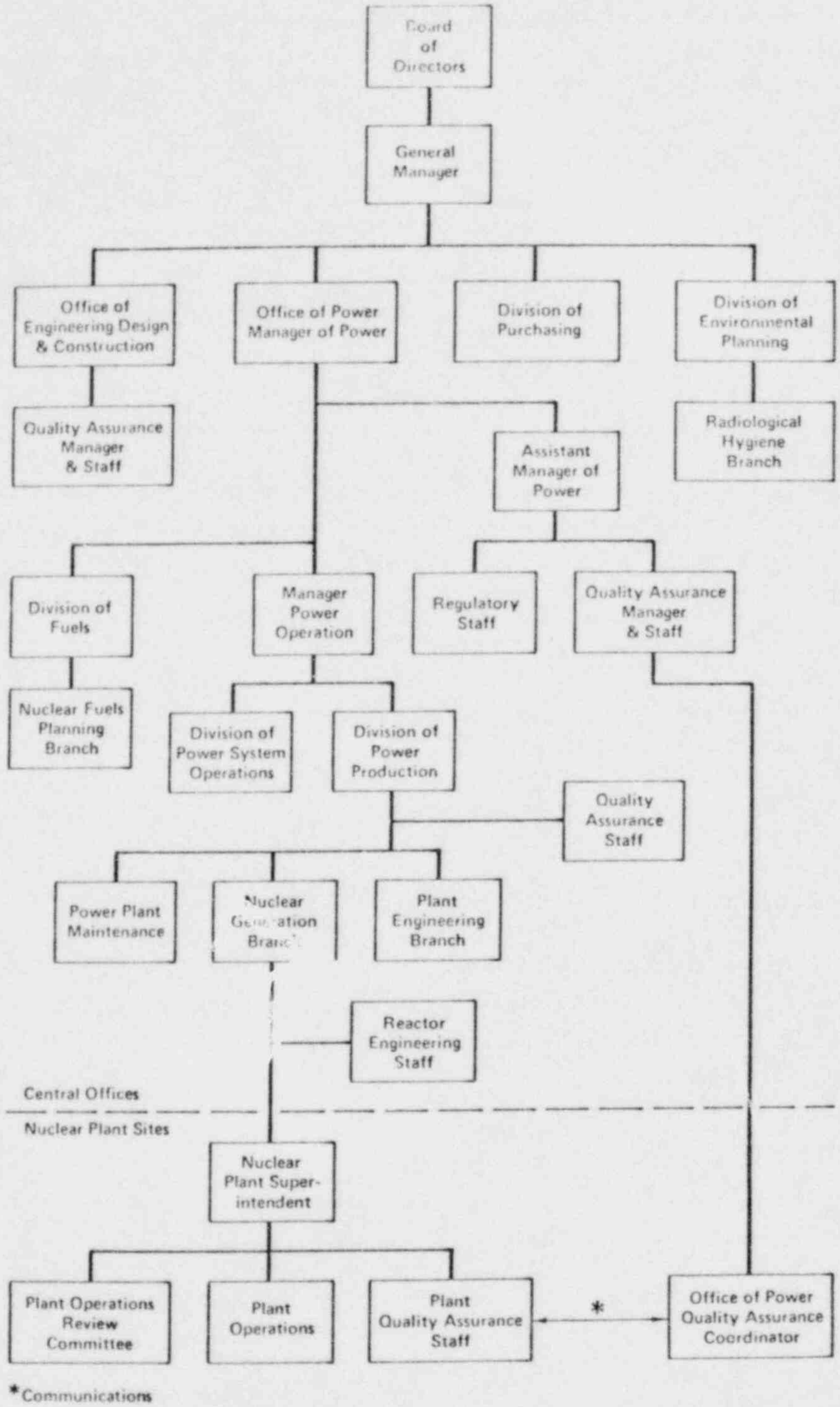

Figure 17-1 Organization For Operation Quality Assurance

The Director of the Division of Power Production is responsible for the operation and maintenance of the Sequoyah Nuclear Plant during the operations phase. He has delegated the responsibility of the day-to-day operation and maintenance activities for the Sequoyah Nuclear Plant to the Chief of the Nuclear Generation Branch. The Plant Superintendent, who reports to the Chief of the Nuclear Generation Branch, has primary responsibility for operating and maintaining the Sequoyah Nuclear Plant in compliance with the requirements of the operating license and the plant Operational Quality Assurance Manual. The resolution of any disputes on quality assurance program requirements arising between quality assurance personnel and other department personnel which cannot be resolved locally are referred to higher management for resolution, with eventual resolution by the Manager of Power is necessary.

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

The Plant Operations Review Committee, chaired by the Plant Superintendent, serves in an advisory capacity to the Plant Superintendent in operational matters related to safety. This committee reviews all standard and emergency operation and maintenance instructions and changes thereto, proposed tests and experiments, proposed changes to technical specifications, and proposed modifications to plant systems that affect nuclear safety. The Supervisor of the plant quality assurance staff is a member of this committee.

17.3 Quality Assurance Program

The quality assurance program for the operation of the Sequoyah Nuclear Plant implements the requirements of the Tennessee Valley Authority quality assurance policies via the Office of Power Quality Assurance Manual, the Operational Quality Assurance Manual, the Nuclear Fuel Quality Assurance Manual, and other operating procedures and standard practices. These documents control quality-related activities involving safety-related items so that they comply with the requirements of Appendix B to 10 CFR Part 50 and other commitments in the Final Safety Analysis Report. The quality assurance program requires that implementing documents include detailed controls for 1) translating codes, standards, regulatory requirements, technical specifications, engineering requirements, and process requirements into drawings, specifications, procedures, and instructions; 2) developing, reviewing, and approving procurement documents, including changes; 3) prescribing all quality-related activities in documented instructions, procedures, and drawings; 4) issuing and distributing approved documents; 5) purchasing items and services; 6) identifying materials, parts, and components; 7) performing special processes; 8) inspecting and/or testing materials, equipment, processes, and services; 9) calibrating and maintaining measuring and test equipment; 10) handling, storing, and shipping items; 11) identifying the inspection, test, and operating status of items; 12) identifying and dispositioning nonconforming items; 13) correcting conditions adverse to quality; 14) preparing and maintaining records; and 15) auditing activities which affect quality.

An indoctrination and training program is established to assure that personnel performing activities affecting quality are knowledgeable in quality assurance requirements, implementing procedures and instructions, and that they have competence and skill in the performance of their quality-related activities.

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

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

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

17.4 Conclusions

Our review of the Sequoyah Nuclear Plant quality assurance program description for the operations phase has verified that the criteria of Appendix B to 10 CFR Part 50 have been adequately addressed in Section 17.2 of the Final Safety Analysis Report.

Based on our detailed review and evaluation of the quality assurance program description contained in Section 17.2 of the Final Safety Analysis Report through Amendment 58 for the Sequoyah Nuclear Plant, we conclude that:

1. The quality assurance organizations of the Tennessee Valley Authority are provided sufficient independence from cost and schedule (when opposed to safety considerations), sufficient authority to effectively carry out the operations quality assurance program, and sufficient access to management at a level necessary to perform their quality assurance functions.

2. The quality assurance program description contains adequate quality assurance requirements and a comprehensive system of planned and systematic controls which address each of the criteria of Appendix B to 10 CFR Part 50 in an acceptable manner. This quality assurance program description, therefore, can serve as an adequate basis for the development of specific policies and procedures to implement the quality assurance responsibilities of TVA for the operation of the Sequoyah Nuclear Plant.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In its letter of February 11, 1970, the Advisory Committee on Reactor Safeguards indicated that certain matters would require further attention and resolution during construction of the Sequoyah Nuclear Plant. These items were addressed in our Safety Evaluation Report dated March 24, 1970.

Certain of these matters are addressed further in this Safety Evaluation Report, as identified below. References are given to sections in this report for further discussion.

Measurements to justify core power level	Sections 4.3, 4.4
Independent check of containment divider barrier	Section 6.2.1
Design of decay heat removal during flooding and recovery after flooding	Sections 2.4.5, 5.3.2
Study of dose reduction methods	Sections 15.3, 6.2.3
Study of common failure modes and anticipated transients without scram	Section 15.3.8
Review of hydrogen control methods	Section 6.2.5
Study of fuel failure mechanisms	Sections 4.2.1, 4.4

The Sequoyah Nuclear Station operating license for the proposed facility is being reviewed by the Advisory Committee on Reactor Safeguards. We intend to issue a supplement to this Safety Evaluation Report after 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.

19.0 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 citizens of the United States. The applicant, being an agency of the United States Government, 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 that 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 find that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility construction permit are Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. To assure that we have the latest information to make a determination of the financial qualifications of an applicant, it is our current practice to review this information during the later stages of our review of an application. We are continuing our review of the financial qualifications of the applicant and will report the results of our evaluations in a supplement to this report.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 General

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors under 10 CFR Part 50.

21.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in 10 CFR Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The applicant will furnish the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association Policy (Nuclear Energy Liability Policy, facility form No. NF-226). Further, the applicant will execute an Indemnity Agreement with the Commission effective as of the date of its preoperational fuel storage license. The applicant will pay the annual indemnity fee applicable to preoperational fuel storage.

21.3 Operating Licenses

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for Sequoyah Nuclear Plant, Units 1 and 2 (which have a rated capacity in excess of 100,000 electrical kilowatts), is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $140 million.

Accordingly, licenses authorizing operation of Sequoyah Nuclear Plant, Units 1 and 2, will not be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended so that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. Similarly, operating licenses will not be issued until an appropriate amendment to the present indemnity agreement has been executed. The applicant will be required to pay an annual fee for operating license indemnity as provided in our regulations.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating licenses, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, it is our position that, upon favorable resolution of the outstanding matters described herein, we will be able to conclude that:

1. The application for facility licenses filed by Tennessee Valley Authority dated October 15, 1968, as amended complies with the requirements of the Atomic Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and

2. Construction of Sequoyah Nuclear Plant, Units 1 and 2, has proceeded and there is reasonable assurance that it will be substantially completed, in conformity with Construction Permits Nos. CPPR-72 and CPPR-73, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

3. The facilities will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (a) that the activities authorized by the operating licenses can be conducted without endangering the health and safety of the public, and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

5. The applicant is technically and financially qualified to engage in the activities authorized by these licenses, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

6. The issuance of these licenses will not be inimical to the common defense and security or to the health and safety of the public.

Before operating licenses will be issued to Tennessee Valley Authority, for operation of Sequoyah Nuclear Plant, Units 1 and 2, the units must be completed in conformity with the provisional construction permits, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power levels must be verified by the Commission's Office of Inspection and Enforcement prior to issuance of the licenses.

Further, before operating licenses are issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

APPENDIX A

CHRONOLOGY
FOR
RADIOLOGICAL SAFETY REVIEW
SEQUOYAH NUCLEAR PLANT

December 3, 1973	Final Safety Analysis Report tendered for acceptance review.
December 3, 1973	Letter to applicant acknowledging receipt of application and advising that review will be performed.
December 20, 1973	Letter from applicant advising that ATWS analysis will be submitted by October 1, 1974.
December 20, 1973	Letter from applicant transmitting Report No. 72-21 concerning stability analysis of containment vessels.
January 10, 1974	Letter to applicant accepting application for review and requesting that additional information be finished.
January 29, 1974	Letter from applicant transmitting their "Handbook of Health Physics."
January 31, 1974	Amendment '4 from the applicant submitting required copies of application documents.
January 31, 1974	Letter from applicant transmitting the Physical Security Plan (withheld from public disclosure).
February 1, 1974	Letter from applicant transmitting Amendment 15 to the Final Safety Analysis Report.
February 11, 1974	Received Amendment No. 1 to SNP-1 Instrumentation drawings.
February 11, 1974	Letter from applicant transmitting Amendment 16 to the Final Safety Analysis Report.

February 19, 1974	Letter from applicant transmitting Quality Assurance Program.
February 27, 1974	Letter to applicant establishing review schedule.
March 4, 1974	Letter from applicant transmitting supplemental information in Report No. 72-21.
March 8, 1974	Letter from applicant transmitting Amendment 17 to the Final Safety Analysis Report.
March 8, 1974	Letter to applicant commenting on Report No. 72-21.
March 20, 1974	Letter to applicant transmitting Federal Register notices and display ads.
April 12, 1974	Letter from applicant transmitting Amendment 18 to the Final Safety Analysis Report.
May 3, 1974	Letter from applicant transmitting Amendment 18 to the Final Safety Analysis Report.
May 7, 1974	NRC Site Visit.
May 8, 1974	Letter from applicant transmitting additional information on Report No. 72-21.
May 14, 1974	Letter from applicant advising of fuel loading dates.
May 15, 1974	Letter from applicant transmitting Report No. 72-22, "Evaluation of the Effects of Postulated Pipe Failure Outside of Containment."
May 31, 1974	Letter from applicant transmitting Amendment 20 to the Final Safety Analysis Report.
June 7, 1974	Letter from applicant transmitting Amendment 21 to the Final Safety Analysis Report.
June 7, 1974	Letter to applicant transmitting NRC requests for additional information.

June 14, 1974	Letter to applicant transmitting requests for additional information on Industrial Security Plan.
June 25, 1974	Letter from applicant transmitting Amendment No. 22 to the Final Safety Analysis Report.
July 1, 1974	Letter from applicant transmitting Amendment No. 23 to the Final Safety Analysis Report.
July 1, 1974	Letter to applicant accepting freedom of Quality Assurance personnel to perform critical functions.
July 12, 1974	Letter from applicant transmitting additional information on Industrial Security Plan.
July 16, 1974	Letter from applicant indicating minimum response to staff questions on seismology and geology.
July 22, 1974	Letter to applicant requesting additional information on UHI.
July 23, 1974	Letter from applicant transmitting Amendment No. 24 to the Final Safety Analysis Report.
July 31, 1974	Letter from applicant transmitting revised Industrial Security Plan.
August 1, 1974	Letter to applicant requiring further response to questions on seismology and geology.
August 6, 1974	Letter from applicant transmitting additional supplementary information on Report No. 72-21.
August 8, 1974	Letter to applicant noting revised review schedule.
August 9, 1974	Meeting to discuss seismology and geology items.
August 9, 1974	Letter from applicant transmitting Amendment No. 25 to the Final Safety Analysis Report.
August 21, 1974	Letter to applicant requesting additional information.

APPENDIX A (Continued)

September 3, 1974	Letter from applicant transmitting Amendment No. 26 to the Final Safety Analysis Report.
September 18, 1974	Letter from applicant transmitting Amendment No. 27 to the Final Safety Analysis Report.
September 23, 1974	Received revision to Radiological Emergency Plan.
September 25, 1974	Letter from applicant transmitting photographs of site excavations and foundations.
September 19, 1974	Letter from applicant transmitting additional supplementary information on Report No. 72-21.
September 30, 1974	Letter from applicant furnishing information on anticipated transients without scram schedule.
October 3, 1974	Letter from applicant notifying use of change in schedule for fuel loading.
October 10, 1974	Letter to applicant requesting additional information and taking staff positions.
October 10, 1974	Letter to applicant transmitting positions on the Industrial Security Plan.
October 25, 1974	Letter to applicant accepting proposed anticipated transients without scram submittal date of January 1, 1975.
October 31, 1974	Letter from applicant requesting meeting on tornado missiles.
November 1, 1974	Letter to applicant requesting additional information.
November 6, 1974	Letter from applicant transmitting Amendment No. 28 to the Final Safety Analysis Report.
November 12, 1974	Meeting with applicant on tornado missiles.
November 26, 1974	Letter from applicant transmitting Amendment No. 28 to the Final Safety Analysis Report.

APPENDIX A (Continued)

November 27, 1974	Letter to applicant requesting additional information on tornado missiles.
November 27, 1974	Letter to applicant revising the review schedule.
December 6, 1974	Letter from applicant transmitting replies to staff positions on the Industrial Security Plan.
December 6, 1974	Letter from applicant transmitting additional information on Report No. 72-22.
December 13, 1974	Letter from applicant transmitting Amendment No. 30 to the Final Safety Analysis Report.
December 16, 1974	Meeting with applicant, Portland General Electric, and diesel generator vendor to discuss qualification testing of diesel generators for Sequoyah, Watts Bar, and Trojan.
December 31, 1974	Letter from applicant providing anticipated transients without scram information.
January 3, 1975	Letter from applicant transmitting Amendment No. 31 to the Final Safety Analysis Report.
January 9, 1975	Letter to applicant requesting additional information.
January 31, 1975	Letter from applicant transmitting Amendment No. 32 to the Final Safety Analysis Report.
February 13, 1975	Site visit to Sequoyah.
February 24, 1975	Letter to applicant requesting additional information on Report No. 72-22.
March 7, 1975	Letter from applicant responding in part to our letter of January 9, 1975.
March 18, 1975	Letter to applicant requesting additional financial information.
March 24, 1975	Letter from applicant transmitting information in response to our letter of February 24, 1975.

APPENDIX A (Continued)

April 2, 1975	Letter to applicant transmitting copies of the Westinghouse Standard Technical Specifications.
April 4, 1975	Letter from applicant transmitting Amendment No. 33 to the Final Safety Analysis Report.
April 4, 1975	Letter from applicant covering the Sequoyah emergency core cooling system.
April 9, 1975	Letter to applicant transmitting staff positions and requests for additional information.
April 22, 1975	Letter to applicant transmitting staff positions and requests for additional information.
April 22, 1975	Letter from applicant transmitting additional financial material.
May 13, 1975	Letter from applicant transmitting Amendment No. 34 to the Final Safety Analysis Report.
May 27, 1975	Letter from applicant responding in part to our letter of April 9, 1975.
June 6, 1975	Letter from applicant transmitting Amendment No. 35 to the Final Safety Analysis Report.
June 9, 1975	Letter from applicant providing additional information on pipe failures both inside and outside containment.
June 12, 1975	Letter to applicant requesting additional information on the emergency core cooling system.
June 17, 1975	Letter to applicant transmitting power distribution section of the Westinghouse Standard Technical Specifications.
June 18, 1975	Meeting with applicant to discuss outstanding electrical items.
June 26, 1975	Letter from applicant advising us of revised expected fuel loading dates.

APPENDIX A (Continued)

June 26, 1975	Letter from applicant responding in part to our letter of April 22, 1975.
June 30, 1975	Meeting with applicant to discuss additional outstanding electrical items.
July 2, 1975	Letter from applicant describing settlement of east steam valve room.
July 3, 1975	Letter from applicant transmitting Amendment No. 36 to the Final Safety Analysis Report.
July 16, 1975	Letter from applicant responding to some items discussed in meeting of June 18, 1975.
July 30, 1975	Letter from applicant responding to our letter of June 12, 1975 on the emergency core cooling system.
July 31, 1975	Site visit to discuss settlement of east steam valve room.
August 12, 1975	Meeting with applicant to discuss preliminary draft of Technical Specifications.
September 4, 1975	Letter to applicant requesting additional information on UHI.
September 8, 1975	Letter to applicant identifying major operations and requesting schedule for applicant action.
September 30, 1975	Letter from applicant transmitting Amendment No. 37 to the Final Safety Analysis Report.
October 6, 1975	Letter from applicant transmitting Revision 2 to Report No. 75-22.
October 15, 1975	Meeting with applicant on the emergency core cooling system testing through containment sump.
October 28, 1975	Letter from applicant requesting extension of latest completion date of construction permits.

APPENDIX A (Continued)

October 31, 1975	Letter from applicant responding in part to our letter of September 8, 1975.
November 20, 1975	Letter from applicant transmitting Amendment No. 38 to the Final Safety Analysis Report.
December 4, 1975	Letter to applicant advising of potential safety question regarding design of reactor pressure vessel support systems for pressurized water reactors.
December 8, 1975	Letter from applicant responding to questions on the Final Safety Analysis Report.
December 9, 1975	Letter from applicant responding to staff questions concerning bypass leakage and containment purge.
December 15, 1975	Letter from applicant providing additional information concerning the emergency core cooling system evaluation.
December 23, 1975	Letter from applicant responding to major outstanding items.
December 29, 1975	Summary of meeting on mass and energy release to containment - Westinghouse.
December 30, 1975	Letter from applicant furnishing information on the design of the reactor vessel support system.
December 30, 1975	Letter from applicant responding to questions on the Final Safety Analysis Report.
December 30, 1975	Letter from applicant furnishing information concerning test results of Watts Bar diesel generator as they apply to Sequoyah.
January 5, 1976	Letter to applicant requesting additional information to continue review of the emergency core cooling system analysis.
January 15, 1976	Letter from applicant transmitting response on automatic switchover from injection to recirculation mode.

APPENDIX A (Continued)

January 16, 1976	Letter to applicant transmitting draft Safety Evaluation Report Section 1.8.
January 30, 1976	Letter from applicant transmitting list of current status of responses to 15 outstanding items.
March 24, 1976	Letter from applicant transmitting Amendment No. 39 to the Final Safety Analysis Report.
March 26, 1976	Letter from applicant furnishing information concerning Appendix I.
April 26, 1976	Letter from applicant regarding our February 13, 1976 meeting regarding the flood protection plan.
May 6, 1976	Letter from applicant concerning zero containment bypass leakage and reactor cavity pressure analysis.
May 7, 1976	Letter from applicant responding to questions regarding UHI isolation valves.
May 25, 1976	Letter from applicant addressing outstanding items on residual heat removal isolation valve interlock and feedwater flow instabilities.
May 27, 1976	Letter from applicant listing major items on which NRC has caused unavoidable delays in schedule.
June 1, 1976	Letter from applicant transmitting Revision 3 to Tennessee Valley Authority Report No. 72-22.
June 1, 1976	Letter from applicant concerning review of scoping document for residual heat removal sump vortex test.
August 11, 1976	Letter from applicant concerning revised scoping documents for residual heat removal sump vortex test.
September 7, 1976	Letter from applicant furnishing Tables 5-1 through 5-6.
September 23, 1976	Letter from applicant transmitting Amendment No. 40 to the Final Safety Analysis Report.

APPENDIX A (Continued)

October 5, 1976	Letter from applicant transmitting Amendment No. 41 to the Final Safety Analysis Report.
October 19, 1976	Letter from applicant transmitting Amendment No. 42 to the Final Safety Analysis Report.
November 16, 1976	Letter to applicant regarding trip report on seismic audit of Tennessee Valley Authority equipment.
December 20, 1976	Letter to applicant regarding fire protection evaluation.
December 29, 1976	Letter from applicant transmitting Amendment No. 43 to the Final Safety Analysis Report.
January 24, 1977	Letter from applicant transmitting information on the fire prevention and protection program.
January 26, 1977	Summary of January 3, 1977 meeting with applicant on pipe restraints.
February 21, 1977	Letter from applicant furnishing fuel load dates.
February 25, 1977	Letter to applicant regarding guidance on implementing new rule regarding physical security plan.
March 7, 1977	Letter from applicant furnishing information regarding decay heat curve.
March 15, 1977	Letter from applicant trasmitting Amendment No. 44 to the Final Safety Analysis Report.
March 15, 1977	Summary of February 3, 1977 meeting on pipe restraints.
March 18, 1977	Letter to applicant regarding fuel handling accident.
March 24, 1977	Letter to applicant requesting additional information on selection of instrumentation trip on setpoint values.
April 7, 1977	Letter from applicant transmitting change to technical specifications.

APPENDIX A (Continued)

April 8, 1977	Letter from applicant transmitting automatic switchover scheme from injection to recirculation mode.
April 29, 1977	Letter from applicant transmitting Amendment No. 45 to the Final Safety Analysis Report.
May 4, 1977	Letter to applicant transmitting Intrusion Detection Systems Handbook.
May 5, 1977	Letter from applicant furnishing information related to instrument trip setpoint values.
May 10, 1977	Letter to applicant requesting additional information.
May 13, 1977	Letter from applicant furnishing information regarding fuel handling accident analysis.
May 23, 1977	Letter from applicant transmitting Detailed Physical Security Plan.
May 31, 1977	Letter from applicant transmitting Amendment No. 46 to the Final Safety Analysis Report.
June 13, 1977	Letter to applicant requesting additional information concerning purge system containment isolation valves.
June 17, 1977	Memorandum, Silver to Varga, regarding forthcoming electrical site visit of June 21, 1977.
June 20, 1977	Electrical site visit.
July 15, 1977	Letter from applicant transmitting Amendment No. 47 to the Final Safety Analysis Report.
July 29, 1977	Letter from applicant requesting extension of the latest completion dates.
August 2, 1977	Letter to applicant requesting additional information concerning analysis of main steam line break accident.
August 15, 1977	Letter from applicant responding to additional Appendix I questions.

APPENDIX A (Continued)

August 18, 1977	Letter to applicant concerning emergency core cooling system upper head temperature verification.
August 22, 1977	Letter from applicant transmitting model study for containment sump performance.
August 23, 1977	Letter from applicant transmitting Preservice Baseline Inspection Program.
August 29, 1977	Letter to applicant transmitting fire protection information.
August 30, 1977	Letter from applicant transmitting revised proposed environmental technical specifications.
September 1, 1977	Letter from applicant concerning the review and analysis of flood levels.
September 15, 1977	Letter from applicant transmitting Amendment No. 48 to the Final Safety Analysis Report.
September 15, 1977	Letter from applicant transmitting interim report concerning reactor vessel support and nozzle loads.
October 3, 1977	Letter to applicant regarding model study for plant containment sump performance.
October 27, 1977	Memorandum, Lasher to Ippolito, regarding site visit of June 20, 1977.
October 31, 1977	Letter to applicant concerning financial information.
November 8, 1977	Letter to applicant requesting additional information.
November 16, 1977	Letter from applicant transmitting Appendix II to Tennessee Disaster Assistance Plan.
December 14, 1977	Letter from applicant consisting of responses regarding planned containment sump model test.
December 27, 1977	Letter to applicant concerning seismic design basis.

December 30, 1977	Letter from applicant concerning reactor vessel support.
January 5, 1978	Letter from applicant responding to questions.
January 6, 1978	Letter from applicant transmitting Amendment No. 49 to the Final Safety Analysis Report.
January 13, 1978	Summary of December 21, 1977 meeting on seismic design.
January 17, 1978	Letter to applicant concerning seismic design basis.
January 27, 1978	Letter from applicant transmitting Amendment No. 50 to the Final Safety Analysis Report.
January 30, 1978	Letter from applicant transmitting certain Westinghouse reports.
February 10, 1978	Letter from applicant transmitting Amendment No. 51 to the Final Safety Analysis Report.
February 10, 1978	Memorandum, Watt and Rubin to Novak, regarding December 16, 1977 trip report on containment sump test.
February 14, 1978	Letter from applicant requesting extension of latest completion dates.
March 6, 1978	Letter from applicant transmitting Amendment No. 52 to the Final Safety Analysis Report.
March 7, 1978	Letter to applicant concerning review of inservice testing program for pumps and valves.
March 17, 1978	Letter from applicant furnishing response to questions.
March 28, 1978	Summary of meeting on seismic design basis.
March 30, 1978	Letter to applicant concerning containment sump tests.
April 5, 1978	Letter from applicant transmitting Amendment No. 53 to the Final Safety Analysis Report.
April 5, 1978	Meeting with TVA on seismic matters.

APPENDIX A (Continued)

April 14, 1978	Meeting with TVA on seismic matters.
May 1, 1978	Letter from applicant forwarding report on Phase I activities.
May 5, 1978	Letter to applicant transmitting NUREG-0219.
May 16, 1978	Meeting with Tennessee Valley Authority on seismic matters.
May 19, 1978	Letter from applicant forwarding response to our concerns regarding sump scale model tests.
May 26, 1978	Letter from applicant furnishing information concerning development of planned Phase II submittal concerning specific site response spectra.
May 26, 1978	Letter from applicant transmitting Amendment No. 54 to the Final Safety Analysis Report.
May 30, 1978	Letter to applicant concerning seismic design.
June 19, 1978	Summary of meetings with Tennessee Valley Authority.
June 26, 1978	Letter from applicant furnishing responses concerning geology and seismology issues.
June 28, 1978	Letter to applicant requesting additional information.
July 3, 1978	Letter from applicant furnishing responses to questions.
July 5, 1978	Memorandum, Silver to Varga, regarding July 18, 1978 fire protection site visit.
July 5, 1978	Letter from applicant forwarding Revision 3 of preservice baseline inspection program.
July 27, 1978	Letter from applicant furnishing response to Containment Systems Branch questions.
August 2, 1978	Letter from applicant informing NRC of scheduled fuel load dates.

APPENDIX A (Continued)

August 4, 1978	Summary of meeting to discuss review schedule matters.
August 7, 1978	Letter from applicant forwarding information concerning outstanding issues.
August 11, 1978	Letter from applicant responding to seven hydrological engineering questions.
August 14, 1978	Letter to applicant transmitting amendment to indemnity agreement.
August 31, 1978	Letter from applicant transmitting justification of seismic design criteria.
September 1, 1978	Letter to applicant concerning fire protection review.
September 11, 1978	Summary of meeting on August 17, 1978.
September 12, 1978	Letter from applicant forwarding responses to questions.
September 28, 1978	Letter from applicant transmitting Amendment #56 to Final Safety Analysis Report.
September 29, 1978	Letter to applicant transmitting Order Extending Construction Completion Dates.
October 4, 1978	Letter to applicant concerning seismic design basis.
October 6, 1978	Letter from applicant providing additional justification of seismic design criteria.
October 19, 1978	Letter from applicant transmitting Physical Security Plan.
October 25, 1978	Memo, Lasher to Satterfield, re record of conference call.
October 26, 1978	Meeting with applicant on fire protection.
November 2, 1978	Memo, Silver to Varga, re forthcoming meeting on November 9.

November 7, 1978	Letter from applicant responding to request for information on sump demonstraton test.
November 9, 1978	Letter from applicant responding to request re fire protection.
November 9, 1978	Letter from applicant re Radiological Emergency Plan.
November 9, 1978	Letter from applicant responding to questions.
November 14, 1978	Letter from applicant responding to questions.
November 14, 1978	Letter from applicant responding to questions.
November 15, 1978	Letter from applicant transmitting Amendment #57 to Final Safety Analysis Report.
November 15, 1978	Letter to applicant concerning review schedule.
November 16, 1978	Letter to applicant concerning guidance and submittal schedule for effluent technical specifications.
November 30, 1978	Letter to applicant concerning clarificaton of seismic issues.
December 4, 1978	Letter from applicant responding to questions.
December 6, 1978	Letter from applicant responding to questions.
December 7, 1978	Letter from applicant forwarding marked-up drawings of control room.
December 7, 1978	Letter from applicant forwarding material on balance of plant Class IE equipment qualification.
December 8, 1978	Summary of meeting on October 26, 1978.
December 8, 1978	Letter to applicant requesting additional information.
December 19, 1978	Letter from applicant responding to fire protection questions.

December 19, 1978	Letter from applicant responding to questions.
December 19, 1978	Letter from applicant responding to fire protection questions.
December 19, 1978	Letter from applicant responding to questions.
December 21, 1978	Letter from applicant transmitting Amendment #58 to Final Safety Analysis Report.
December 29, 1978	Letter from applicant responding to questions.
December 29, 1978	Letter to applicant requesting additional financial information.
January 3, 1979	Letter from applicant responding to questions.
January 4, 1979	Letter from applicant responding to questions.
January 5, 1979	Letter from applicant responding to questions.
January 5, 1979	Letter to applicant requesting additional information.
January 15, 1979	Letter from applicant forwarding questions.
January 18, 1979	Letter from applicant responding to questions re local leak rate testing.
January 19, 1979	Letter from applicant responding to fire protection review questions.
January 19, 1979	Letter to applicant requesting additional information.
January 19, 1979	Letter from applicant forwarding photos of containment sump.
January 19, 1979	Meeting with applicant to discuss cold shutdown.
January 26, 1979	Letter from applicant forwarding Amendment #59 to Final Safety Analysis Report.

APPENDIX B

BIBLIOGRAPHY FOR SEQUOYAH SAFETY EVALUATION REPORT

NOTE: Documents referenced in or used to prepare this Safety Evaluation Report may be obtained at the source stated in the Bibliography or, where no specific source is given, at most major public libraries. Correspondence between the Commission and the applicant and Commission's rules and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. Correspondence between the Commission and the applicant may also be inspected at the Chattanooga - Hamilton County Bicentennial Library, 1001 Broad Street, Chattanooga, Tennessee 37402. Specific documents other than Commission's rules and regulations, Regulatory Guides, the various standards, and those listed in the Final Safety Analysis Report relied upon by the Commission's staff and referenced in this Safety Evaluation Report are listed as follows:

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APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - GENERIC MATTERS

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic matters applicable to all large light water reactors. We believe each of these matters should be carefully considered and, as conclusions are drawn, each facility or reference design application should be evaluated with respect to those issues appropriate to that application. We recognize that this could result in a necessity for modification of a facility even after the facility is completed. This is consistent with our continuing efforts toward reducing still further the already small risk to the public health and safety from nuclear power plants. The most recent such report concerning these generic items was issued in a letter dated November 15, 1977 to Commission Chairman Hendrie from (then) Committee Chairman M. Bender.

The status of staff efforts leading to resolution of all unresolved generic matters identified by the Committee is contained in our Status Report on Generic Items periodically transmitted to the Committee. The latest such Status Report is contained in a letter from H. R. Denton to S. Lawroski dated December 4, 1978.

Each of the ACRS' generic items have been considered for inclusion in the Office of Nuclear Reactor Regulation's (NRR) Program for the Resolution of Generic Issues to be described in a supplement to this report. A cross index of the ACRS generic items and the generic tasks in the NRR program is provided in Table C-1. The supplement will provide discussions of those issues in the NRR program applicable to the Sequoyah facility that have been identified as "Unresolved Safety Issues" for the purpose of reporting to the Congress pursuant to Section 210 of the Energy Reorganization Act of 1974 as amended. Each discussion will describe how the issue relates to the Sequoyah facility and the basis for the staff's conclusion that the Sequoyah facility may be operated prior to the ultimate resolution of the issue without endangering the health and safety of the public.

For several of the items, we have provided in this report specific discussions particularizing for the Sequoyah facility the generic status in the Status Report. For those items, reference to the appropriate section numbers of this report are provided below. For those matters applicable to the Sequoyah design, but for which specific conclusions or actions have not been identified on this application, the status of our efforts are provided below and in the December 4, 1978 Status Report.

Group II - Resolution Pending

(1) Turbine Missiles. This item is resolved for the Sequoyah facility by the turbine orientation and turbine overspeed protection system. (Section 3.5.1 and 10.2 of this report)

(2) Effective Operation of Containment Sprays in a LOCA. This item is not applicable to Sequoyah since no credit is given in accident analyses for fission product removal by the containment sprays.

(3) Possible Failure of Pressure Vessel Post-Loss-of-Coolant Accident by Thermal Shock. This item is resolved for this facility by conformance to or exemptions from the requirements of Appendix G to 10 CFR Part 50. (See Section 5.2.3)

(4) Instruments to Detect (Severe) Fuel Failures. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. Instruments that can detect fuel failures are included in the Sequoyah design (see Section 9.3.5 of the Sequoyah FSAR). Revision 1 of Section 4.2 of the Standard Review Plan addresses the use of such instruments. When these instruments are used with other plant instruments that sense coolant pressure, temperature, flow, etc., all events or accidents that lead to fuel failures are detectable.

(5A) Monitoring for Loose Parts Inside the Reactor Pressure Vessel. This item is resolved for this facility by our requirement for the installation of a loose parts monitoring system. (See Section 5.2.8)

(5B) Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel. We are developing a task action plan for this item as indicated in our status report to the Committee dated December 4, 1978. This subject has been approved by our Technical Activities Steering Committee as a Category B Task. Development of a task action plan and efforts toward resolution will commence as necessary resources become available.

(6A) Common Mode Failures: Reactor Scram Systems. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. We published reports on anticipated transients without scram in December 1975 in which we identified the portions of reactor scram systems that needed modifications to improve the reliability of scram systems. In addition, these reports provided guidelines on evaluation models, analysis assumptions, system reliability requirements, and acceptance limits. We published a report NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors" dated

April 1978. A supplement to NUREG-0460 was issued in December 1978. This report is currently being reviewed by the ACRS. (See Section 15.3.8)

(6B) Common Mode Failures: Alternating Current Sources Onsite and Offsite. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. As part of the scope of Technical Activity No. A-35, "Adequacy of Offsite Power Systems," we are evaluating the need, if any, to upgrade the offsite power source and its interface with the onsite power system for license applications. The results of this task will serve as the input and bases for any modifications that may be required to our existing licensing criteria. We will prepare a report in the form of a NUREG document which will provide complete documentation of the details, conclusions and any new or augmented criteria developed as the result of the staff's implementation of this task action plan relating to offsite power. The NUREG report is currently scheduled for completion by July 15, 1980.

If an effort to improve the reliability of the diesel generators, we contracted with an experienced qualified outside consultant (University of Dayton) (1) to perform a study of Licensee Event Reports related to diesel generator malfunctions (2) to make a limited number of visits at operating facilities, (3) to obtain the manufacturers' recommendations regarding operations, maintenance and repair of their equipment, and to survey comparable industrial experience with standby emergency power supplies. This generic item has been considered during the development of the staff's technical activities program. It is included in the scope of Technical Activity No. B-56, "Diesel Reliability." (See Section 8.2)

(6C) Common Mode Failures: Direct Current Systems. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. A full description of this problem and the plan for resolution are contained in our Task Action Plan A-30. Under this plan the first four tasks are essentially complete. These tasks have expanded on NUREG-0305 in terms of data base, recalculation of allowable times for manual actions, and finer definition of the spectrum of concerns that accompany total loss of D.C. power. The next phase of the plan includes quantifying D.C. power system reliability in relationship to assuring adequate decay heat removal capacibility. These analyses will be based on reliability data for various systems and components, using Reactor Safety Study methodology.

The end product of this program will be a NUREG report which will provide complete documentation of the analyses performed and develop a staff position regarding the adequacy of the existing acceptance criteria for D.C. power systems. Completion is scheduled in mid-1979. (See Section 8.3.2)

(7) Behavior of Reactor Fuel Under Abnormal Conditions. As stated in our status report dated December 4, 1978, we believe that item II-7 should no longer be carried as an unresolved generic item.

(8) BWR Recirculation Pump Overspeed During a LOCA. This item is not applicable to Sequoyah which is a pressurized water reactor facility.

(9) The Advisability of Seismic Scram. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. The final report by Lawrence Livermore Laboratory, UCRL-52156, "Advisability of Seismic Scram," was reviewed by the staff and on May 19, 1977, a letter from E. G. Case to Myer Bender advised the ACRS of the staff's conclusion that we do not propose to require installation of seismic trip systems on commercial nuclear power plants. The letter further indicated that the staff considered this matter to be adequately resolved.

Staff members met with the Regulatory Activities Subcommittee on June 8, 1977 to discuss this matter. The Subcommittee comments, later documented in a letter from M. Bender to E. G. Case dated June 14, 1977, were that perhaps the selected seismic trip level should be set at about one-half the SSE, which could change the conclusions of UCRL-52156. Further, the Subcommittee expressed interest in what the Japanese are doing in regard to seismic scrams.

Based upon the Committee's June 14, 1977 letter, the staff has attempted to ascertain the positon of the Japanese regarding automatic seismic scram systems. In July of 1977, the staff requested information from the Japanese regarding their requirements for seismic scram and the bases for these requirements. We also requested the views of the Japanese on the UCRL-52156 study. To date we have received no formal response to this request. However, we learned during a staff visit to Japan in November 1977, and confirmed during a visit to the U.S. by a Japanese delegation in June 1978, that the Japanese do require the installation of seismic scram systems. Trip levels are set at what corresponds to 1/2 to 2/3 of the SSE design level.

This generic item has been considered during the development of the staff's technical activities program. It is included in the scope of Task No. D-1.

(10) Emergency Core Cooling System Capability for Future Plants. This item is now included as one of the research topics in the Commission's long-range safety research plan for improved safety system concepts.

Group IIA - Resolution Pending - Items Added Since December 18, 1972

(1) Ice Condenser Containments. As noted in our status report on this item, the staff has developed independent analytical capability for determining short term ice condenser performance. Preliminary results using our independent analysis indicate a favorable comparison with Westinghouse calculations on other ice condenser plants very similar to Sequoyah.

The staff is also developing an independent capability for the long term analysis of ice condenser performance. For Sequoyah, this item is resolved through use of conservative requirements in the technical specifications.

(2) PWR Pump Overspeed During a LOCA. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. A topical report on pump overspeed has been submitted by Combustion Engineering and is being reviewed by the NRC staff. The initial review of this submittal indicated a need for experimental verification of analytical calculations. The staff has asked each PWR vendor to submit its most recent prediction of pump overspeed during a LOCA in order to reassess the potential for pump flywheel failure and the necessity, practicality, and validity of electrical braking or other means of controlling pump speed. Combustion Engineering has stated that their more recent analyses use an Idaho National Engineering Laboratory model two-phase flow homologous head degradation which was not included in their original topical report. Based on the approved emergency core cooling system evaluation model, Combustion Engineering predicted approximately 400 percent of rated pump speed with a flow discharge coefficient of 1.0. However, for a more mechanistic calculation which limited the break offset, Combustion Eigneering predicted a significant decrease in pump overspeed.

The staff is performing some independent reactor coolant pump overspeed calculations during a LOCA using the RELAP 4/MOD5 computer code. The study results will be obtained during 1979.

Staff efforts on this problem are included within the scope of Task Action Plan B-68. (Also see Section 5.2.7)

(3) Steam Generator Tube Leakage. This item is discussed for Sequoyah in Section 5.2.6 of this report.

This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. Nuclear steam supply system vendors are currently conducting research programs designed to determine the structural integrity of steam generator tubes which are subjected to various degradaton mechanisms. In addition, the NRC is funding a confirmatory experimental research program at Pacific Northwest Laboratory to verify the burst and cyclic strengths of degraded steam generator tubes and to obtain leakage rate data. Results of these programs will be used to establish steam generator tube leakage rate limits and tube plugging criteria which will be incorporated into NRC Regulatory Guides and Standard Technical Specifications.

We are currently reviewing and evaluating nuclear steam supply system vendor's analyses of the probability and consequences of postulated main steam line break and loss-of-coolant accidents concurrent with steam generator tube failures. The purpose of these efforts is to determine (1) the maximum number of tube failures which can be tolerated without undue risk to the public health and safety and, (2) the probability of degraded tubes failing during nomal operation or during postulated accidents. These efforts include evaluations of the effects of steam generator tube failures on offsite doses and on safety related systems. Several NRC sponsored programs related to these issues are currently in progress. Idaho National Engineering Laboratory is developing a computer code to aid in the evaluation of the effects of tube plugging on the predicted peak clad temperatures and on emergency core cooling system performance following a postulated loss-of-coolant. Brookhaven National Laboratory is in the process of evaluating the impact of steam generator tube failures on the consequences of a main steam line break accident. Results of these programs will be utilized to establish improved inservice inspection criteria for NRC Regulatory Guides and Standard Technical Specifications.

Periodic inservice inspections of a statistically significant number of steam generator tubes in conjunction with a tube leak detection system provides reasonable assurance that a critical number of tubes will not fail during normal operating and postulated accident conditions. Statistical studies, including an NRC sponsored program at Sandia National Laboratory, are being conducted to confirm the adequacy of the existing inservice inspection criteria and to optimize sampling schemes in accordance with results from the above mentioned consequences analyses. Regulatory Guide 1.83, "Inservice Inspection of PWR Steam Generator Tubes" will incorporate results of these programs. Study and development of eddy current testing techniques will be incorporated into the statistical studies and should result in improved confidence levels and in improved techniques for steam generator tube inspection.

The effects of water chemistry and corrosion on steam generator tube degradation are being studied. Improved rquirements for secondary water chemistry, which greatly affects steam generator integrity, are being developed, but may be dependent on steam purity requirements.

This generic item has been considered during the development of the staff's technical activities program. It is included in the scope of Task Action Plan A-4 which addresses Combustion Engineering steam generators.

(4) ACRS/NRC Periodic 10-Year Review of All Power Reactors. This matter is concerned with the development of a program of periodic comprehensive reviews to be conducted by the staff for operating licensed power reactors. This matter does not apply to designs or facilities for which our review for design approval or operating licenses has yet to be completed. Therefore this matter is not applicable to Sequoyah.

Group IIB - Resolution Pending - Items Added Since February 13, 1974

(1) Computer Reactor Protection System. This item is not applicable to Sequoyah which will not utilize a computer reactor protection system (Section 7.2).

(2) Qualification of New Fuel Geometries. This item is resolved for Sequoyah. (See Section 4.2.1)

(3) Behavior of BWR Mark III Containments. This item is not applicable to Sequoyah which is a pressurized water reactor facility.

(4) Stress Corrosion Cracking in BWR Piping. This item is not applicable to Sequoyah which is a pressurized water reactor facility.

Group IIC - Resolution Pending - Items Added Since March 12, 1975

(1) Locking Out of ECCS Power-Operated Valves. This item is discussed in Section 7.3.2 of the Safety Evaluation Report. This generic item has been considered during the development of the staff's technical activities program. It is included in the scope of Task No. B-8. A Task Action Plan for this activity is currently under development.

(2) Design Features to Control Sabotage. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978.

(3A) Decontamination of Reactors. To date there has beer little experience with primary system decontaminaton of operating U.S. commercial power reactors. Th Hanford N reactor primary coolant system has been periodically decontaminated. In Canada, successful decontaminations have been accomplished at Gentilly I and Douglas Point. Also contractors to the Division of Naval Reactors have reported the decontamination of many reactor systems under their cognizance. More recently the Electric Power Research Institute has initiated research programs on decontamination of operating power reactors and Commonwealth Edison is conducting an extensive test program in preparation for decontamination of the Dresden Unit 1 primary system. The staff is evaluating the results of the test program and the effects of the proposed decontamination on system integrity prior to our approval.

This generic item has been considered during the development of the staff's technical activities program. It is included in the scope of Task Action Plan A-15.

(3B) Decommissioning of Reactors. Access control for radiation areas, exposure control, concentration limits for release of radioactive material, personnel monitoring requirements and radiation survey are among the requirements specified in 10 CFR Parts 20 and 50 for all phases of reactor operation, including decommissioning. Regulatory Guide 1.86, published in June of 1974, was developed to provide specific guidance on reactor decommissioning and includes a discussion of the steps required to assure adequate decontamination prior to termination of a reactor license.

The Atomic Industrial Forum and Battelle Northwest are now engaged in studies of reactor decommissioning alternatives, including protective storage or mothballing, entombment, dismantlement and combinations of these alternatives. Both studies will evaluate safety, environmental aspects and costs of each decommissioning alternative. The Atomic Industrial Forum report was published in November 1976. Battelle completed their report on pressurized water reactor decommissioning in June 1978. An addendum is being prepared that relates the costs of and exposures from decommissioning facilities of various power levels and addresses entombment in more detail. In addition, the NRC Office of Standards Development requested proposals for conducting a study to evaluate the dose commitment for radioactive material released to unrestricted areas from decommissioning of nuclear reactor facilities. Experience has been gained in the decommissioning of Elk River, Hallam, Fermi 1, Saxton, Peach Bottom 1 and numerous smaller test and research reactors.

The decommissioning experience and reports will be used as background information in the modification of existing regulations and guides on reactor decommissioning and in the development of any new standards or guides on reactor decommissioning.

This generic item has been considered during the development of the staff's technical activities program. It is included in the scope of Task Number B-64. It is anticipated this program will be completed in approximately two years.

(4) Vessel Support Structures. This item is resolved for Sequoyah as described in Sections 3.9.1 and 6.2.1.

(5) Water Hammer. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. Sections 6.3.3 and 10.4.2 of this Safety Evaluation Report discuss this matter.

The generic consideration of water hammers was incorporated in Task Action Plan A-1, "Water Hammers." Work on the task is proceeding as described below.

Under Task 1.1 a report of a review of water hammer in nuclear power plants was completed and is currently in the final review stage prior to publication.

Under Task 4.1 work has started at Idaho National Engineering Laboratory on a review and evaluation of specific waterhammer problems identified in the Task 1.1 review. A draft of the final report on this work is scheduled to be issued in February 1979.

Under Task 4.2 work has started on a state-of-the-art review of analytical, experimental and design work pertinent to water hammers in nuclear power plants. A draft of the final report on this work is scheduled to be issued in May 1979.

Under Task 4.3, work is continuing at Battelle Northwest Laboratory on potential water hammer problems in preheater type steam generators. The final report is scheduled to be issued in May 1979.

Under Task 4.4, work is continuing at Idaho National Engineering Laboratory on preparation of calculational methods to be used in the analysis of hydraulic and structural consequences of water hammers in operating plants. Reports on this work are scheduled to be issued in February and May 1979.

(6) Maintenance and Inspection of Plants. This item is resolved for Sequoyah by compliance with our current requirements (Section 12.0). This item is also under generic review as indicated in our status report to the Committee dated December 4, 1977.

(7) Behavior of BWR Mark I Containments. This item is not applicable to Sequoyah which is a pressurized water reactor facility.

Group IID - Resolution Pending - Items Added Since April 16, 1976

(1A) Safety-Related Interfaces Between Reactor Island and Balance-of-Plant. This item is not applicable to Sequoyah which has no standardized portions.

(1B) Systems Interactions in Nuclear Power Plants. This item is under generic review as indicated in our status report to the Committee dated December 4, 1978. This matter has been incorporated as a part of our technical activities program. It is included in the scope of Task Action Plan A-17.

The Task Action Plan was approved on November 15, 1977 and has been revised to accomplish the task with contract assistance. This is made necessary because of the workload impact on technical branches that are critical to accomplishment of the task. A combined effort involving contract assistance, technical personnel from the Office of Nuclear Reactor Regulation, and personnel from the Office of Standards Development will perform the task. The contract assistance group will develop an independent methodology for conducting a review for systems interaction and will asssess the Standard Review Plan against this methodology to identify whether any changes are necessary. The contract assistance group will be guided and assisted by our personnel. The performance will be evaluated by a selected group within NRC and will be reviewed by all of our cognizant technical branches.

The contract effort was initiated in May 1978. The first phase of the task will be completed within sixteen months and will identify any corrective procedures. The second phase will be completed in an additional twelve months and will implement the corrective procedures.

(2) Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment. This matter is addressed in this report only as a general requirement for environmental qualification of equipment (Sections 7.2.2 and 7.8.2). However, this item is under generic review as indicated in our status report to the Committee dated December 4, 1978. This generic item has been considered during

the development of our technical activities program. It is included in the scope of Task Number C-1. We have established a plan of action which is pending management approval. The plan includes a schedule for accomplishing the needed investigation into:

a. field experience,
b. adequacy of current designs and quality assurance practices,
c. practicability of testable designs, and
d. the need for the development of guidance criteria.

Group IIE - Resolution Pending - Items Added Since February 24, 1977

(1) Soil-Structure Interactions. This item is not applicable to Sequoyah since the principal seismic Category I structures are founded on rock.

TABLE C-1

CROSS INDEX OF ACRS GENERIC ITEMS VS NRR GENERIC TASKS

ACRS GENERIC ITEM		NRR GENERIC ITEM	
II-1	Turbine Missiles	A-32 A-37	Missile Effects Turbine Missiles
II-2	Effective Operation of Contaiment Sprays in a LOCA	C-10	Effective Operation of Containment Sprays in a LOCA
II-3	Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock	A-11	Reactor Vessel Materials Toughness
II-4	Instruments to Detect (severe) Fuel Failures	Not yet considered by NRR. Will be considered as a Category C proposal.	
II-5A	Loose Parts Monitoring	B-60	Loose Parts Monitoring Systems
II-5B	Monitoring for Excessive Vibration	B-73	Monitoring for Excessive Vibration
II-6	Common Mode Failures	C-13	Non-Random Failures
II-6A	Scram Systems	A-9	ATWS
II-6B	Alternating Current Systems	A-24 A-25 A-35 A-44 B-56	Qualification of Class IE Safety Related Equipment Non-Safety Loads on Class IE Power Sources Adequacy of Offsite Power Systems Station Blackout Diesel Reliability
II-6C	Direct Current Systems	A-24 A-25 A-30 A-44	Same as above Same as above Adequacy of Safety Related DC Power Supplies Same as above
II-7	Behavior of Reactor Fuel Under Abnormal Conditions	B-22	LWR Fuel
II-8	BWR Recirculation Pump Overspeed During LOCA	B-68	Pump Overspeed during a LOCA
II-9	The Advisability of Seismic Scram	D-1	Advisability of Seismic Scram
II-10	ECCS Capability for Future Plants	D-2	ECCS Capability for Future Plants
II A-1	Ice Condenser Containments	B-54	Ice Condenser Containments
II A-2	PWR Pump Overspeed During a LOCA	B-68	PWR Pump Overspeed During a LOCA
II A-3	Steam Generator Tube Leakage	A-3 W A-4 CE A-5 B&W	Steam Generator Tube Integrity
II A-4	ACRS/NRC Periodic 10-year Review of All Power Reactors	Not a generic technical task. Is being treated as a policy matter.	

TABLE C-1 (Continued)

ACRS GENERIC ITEM		NRR GENERIC ITEM	
II B-1	Computer Reactor Protection System	A-19	Digital Computer Protection System
II B-2	Qualification of New Fuel Geometries	B-22	LWR Fuel
II B-3	Behavior of BWR Mark III Containments	B-10	Behavior of BWR Mark III Containments
II B-4	Stress Corrosion Cracking in BWR Piping	A-42	Pipe Cracks in Boiling Water Reactors
II C-1	Locking Out of ECCS Power Operated Valves	B-8	Locking Out of ECCS Power Operated Valves
II C-2	Design Features to Control Sabotage	A-29	Design Features to Control Sabotage
II C-3A	Decontamination of Reactors	A-15	Chemical Decontamination
II C-3B	Decommissioning of Reactors	B-64	Decommissioning of Reactors
II C-4	Vessel Support Structures	A-2	Asymmetric Blowdown Loads on the Reactor Vessel
II C-5	Water Hammer	A-1	Water Hammer
II C-6	Maintenance and Inspection of Plants	B-34	Occupational Radiation Exposure Reduction
II C-7	Behavior of BWR Mark I Containments	A-6 A-7	Mark I Short Term Program Mark I Long Term Program
II D-1A	Safety Related Interfaces Between Reactor Island and Balance-of-Plant	Not a generic technical task. Is being treated as a policy matter.	
II D-1B	Systems Interactions in Nuclear Power Plants	A-17	Systems Interactions in Nuclear Power Plants
II D-2	Assurance of Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	C-1	Assurance of Continuous Long-Term of Seals on Instrumentation and Electrical Equipment
I E-1	Control Rod Drop Accident (BWRs)	D-3	Control Rod Drop Accident (BWRs)
I E-2	Rupture of High Pressure Lines Outside Containment	B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
I E-3	Isolation of Low Pressure From High Pressure Systems	B-63	Isolation of Low Pressure Systems Connected to RCPB

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER *(Assigned by DDC)* NUREG-0011
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15. SUPPLEMENTARY NOTES Pertains to Docket Nos. 50-327 and 50-328	14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

A safety evaluation of the Tennessee Valley Authority's application for a license to operate its Sequoyah Nuclear Plant, Units 1 and 2, located in Hamilton County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. It consists of a technical review and staff evaluation of applicant information on: (1) population density, land use, and physical characteristics of the site area; (2) design, fabrication, construction, testing criteria, and performance characteristics of plant structures, systems, and components important to safety; (3) expected response of the facility to anticipated operating transients, and to postulated design basis accidents; (4) applicant engineering and construction organization, and plans for the conduct of plant operations; and (5) design criteria for a system to control the plant's radiological effluents. The staff has concluded that the plant can be operated by the Tennessee Valley Authority without endangering the health and safety of the public provided that the outstanding matters discussed in the report are favorably resolved.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
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