SAFETY EVALUATION OF THE TENNESSEE VALLEY AUTHORITY BROWNS FERRY NUCLEAR PLANT UNITS 1, 2 & 3

Docket Nos: 50-259
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U.S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF LICENSING
WASHINGTON, D.C.

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SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

UNITS 1, 2 AND 3

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APPENDICES

ABBREVIATIONS

a-c	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AEC	United States Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BNL	Brookhaven National Laboratory
$Btu/hr\text{-}ft^2$	British thermal units per hour per square foot
Btu/lb	British thermal units per pound
BWR	Boiling Water Reactor
cfm	Cubic feet per minute
cfs	Cubic feet per second
Ci/sec	Curies per second
CSS	Core Spray System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
d-c	direct current
ECCS	Emergency Core Cooling System

ESF	Engineered Safety Feature
EECW	Emergency Equipment Cooling Water System
°F	degrees Fahrenheit
ft^2	square feet
ft^3	cubic feet
FSAR	Final Safety Analysis Report
g	acceleration, 32.2 feet per second per second
GDC	AEC General Design Criteria for Nuclear Power Plant Construction Permits
GE	General Electric Company
gpm	gallons per minute
HEPA	High Efficiency, Particulate, Air
HPCIS	High Pressure Cooling Injection System
IEEE	Institute of Electrical and Electronics Engineers
in	inch
$\Delta k/k$	reactivity change
kV	kilovolts
kW/ft	kilowatts per foot
lb	pound
lb/hr	pounds/hr
LOCA	Loss-of-Coolant Accident
LPCIS	Low Pressure Coolant Injection System
LPCS	Low Pressure Core Spray

MCHFR	Minimum Critical Heat Flux Ratio
m	meters
mph	miles per hour
mem	Millirem
MSL	mean sea level
MSLIV	Main Steam Line Isolation Valve
MWD/ton	megawatt days per ton
MWe	megawatts electrical
MWt	megawatts thermal
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
OBE	Operating Basis Earthquake
PHS	Public Health Service
PMF	probable maximum flood
PORC	Plant Operations Review Committee
PSAR	Preliminary Safety Analysis Report
psi	pounds per square inch
psid	pounds per square inch differential
psig	pounds per square inch gauge
QA	Quality Assurance
QC	Quality Control
Rem	Roentgen equivalent man
SRB	Safety Review Board
TVA	Tennessee Valley Authority

R&D	Research and Development
RCICS	Reactor Core Isolation Cooling System
RCPB	Reactor Coolant Pressure Boundary
RHRS	Residual Heat Removal System
sec	second
SGTS	Standby Gas Treatment System
χ/Q	atmospheric diffusion factor (sec/m^3)
w/o	weight percent
10 CFR	AEC, Title 10, Code of Federal Regulations
Part 2	AEC Rules of Practice
Part 20	AEC Standards for Protection Against Radiation
Part 50	AEC Licensing of Production and Utilization Facilities
Part 100	AEC Reactor Site Criteria

1.0 INTRODUCTION

This report is the Atomic Energy Commission's safety evaluation of the application by the Tennessee Valley Authority (TVA) for a license to operate the Browns Ferry Nuclear Plant (herein referred to as the plant or facility) and is applicable to the three reactor units, Unit 1, Unit 2, and Unit 3 which comprise the facility. The Browns Ferry Nuclear Plant is being constructed on a 840 acre site located in Limestone County, Alabama. The site, located approximately 10 miles southwest of Athens, Alabama, and 10 miles northwest of Decatur, Alabama, is on the north shore of Wheeler Lake formed by Wheeler Dam on the Tennessee River. The estimated completion date for Unit 2 is approximately eight months after Unit 1. Unit 3 is expected to be completed about seven months after Unit 2.

On July 7, 1966, TVA filed an application for permits to construct Browns Ferry Nuclear Plant Units 1 and 2. A review of this application was made by the AEC's regulatory staff and by the Advisory Committee on Reactor Safeguards (ACRS). Both concluded that the facility could be constructed without undue risk to the health and safety of the public. On May 10, 1967, Construction Permit Nos. CPPR-29 and CPPR-30 were issued for Units 1 and 2 respectively. Construction was started on May 17, 1967. After a similar review, Construction Permit No. CPPR-48 was issued for Unit 3 on July 31, 1968. Construction for Unit 3 began on August 1, 1968. The applicant's Final Safety Analysis Report and a request for an operating license for all three units were submitted to the AEC on September 25, 1970 as Amendment 9 to the application. The information submitted in the FSAR was subsequently supplemented by Amendments 10 through 31. Due to delays in construction the applicant requested and was granted extensions of CPPR-29 to December 31, 1972 and of CPPR-30 to July 15, 1973 in order to complete construction. Because of additional delays, the applicant is expected to request an extension of CPPR-48 in order to complete construction of Unit 3.

TVA, like other Federal agencies, is subject to the requirements of Section 102 of the National Environmental Policy Act of 1969 which became effective on January 1, 1970. Accordingly, TVA will prepare and issue a final environmental statement for the Browns Ferry Nuclear Plant. In accordance with guidelines promulgated by the Council on Environmental Quality, a draft environmental statement for the Browns Ferry Nuclear Plant was submitted in July 1971 to state and Federal agencies including the AEC for review and comment. A draft supplement and additions to this statement were submitted to the AEC for comment on November 8, 1971. The AEC has reviewed and commented on the draft environmental statement including its additions and supplement with respect to radiological environmental matters. TVA will also issue the final environmental statement which will be used to satisfy the AEC requirements for an environmental statement as specified in Appendix D to 10 CFR 50.

This report summarizes the results of our safety evaluation of the Browns Ferry Nuclear Plant, Units 1, 2, and 3 performed by the Commission's regulatory staff. Our evaluation included a technical review of the information submitted by TVA with regard to the following principal matters:

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

3. We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.
4. We evaluated the applicant's plans for the conduct of plant operations, the organizational structure, the technical qualifications of operating and technical support personnel, the measures taken for industrial security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that TVA is technically qualified to operate the plant and has established effective organizations and plans for continuing safe operation of the facility.
5. We evaluated the design of the systems provided for control of the radiological effluents from the plant to determine that these systems will be able to control the release of radioactive wastes from the station within the limits of the Commission's regulations (10 CFR 20) and that TVA will operate the facility

in such a manner as to reduce radioactive releases to levels that are as low as practicable within the contemplation of the Commission's regulations in 10 CFR Part 50.

6. We evaluated the financial qualifications of the applicant, and the protection and indemnity agreements made for the plant.

Our technical evaluation of the facility was accomplished partly with the assistance of consultants. The reports of our consultants on meteorology, structural design and the environmental monitoring program are appended to this report as Appendices B, C, and D.

During our review of the information submitted in the FSAR we requested that TVA provide additional information needed to complete our evaluation. This additional information was provided in subsequent amendments to the application. In the course of our review we held meetings with TVA and its representatives to discuss and clarify the technical information submitted. As a result of our review, we requested a number of changes to be made in the facility design; these changes are described in the applicant's amendments and are discussed in appropriate sections of this report. A chronology of principal milestones related to our review of this application is attached as Appendix A. The chronology continues the review milestones from those previously listed in the Safety Evaluation Report on our review of the application to construct Unit 3.

Many features of the [illegible] design are similar to those we have evaluated and approved previously for other reactors now under construction or in operation. To the extent feasible and appropriate, we have made use of our previous evaluations to expedite our review of those features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our Safety Evaluation reports for these other facilities have been published and together with the FSAR, as amended, are available for public inspection at the U.S. Atomic Energy Commission, Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama. The applicant has submitted a document entitled "Browns Ferry Nuclear Plant Protection Against Industrial Sabotage" and some detailed information on fuel design as proprietary information. We have determined that this information is of the type that may properly be withheld from public disclosure pursuant to sections 2.790(d) and 9.5(a)(4) of the Commission's Rules and Regulations, and accordingly have decided not to make the information available for public inspection.

Although our conclusion is addressed only to the proposed operation of Browns Ferry Unit 1 at a rated thermal power level of 3293 MW, our evaluation included all three units, each operating at a thermal power level of 3293 MW, which will be each unit's licensed power level. Our evaluation included multiunit operation since systems are shared among the three units and the units are similar. We intend to issue a supplemental safety evaluation for Units 2 and 3 at the time construction for each of these units is nearing com pletion. Our evaluation of the plant safety systems and radiological consequences of design basis accidents were based on the plant design power level of 3440 MWt.

Based on our evaluation of the application to operate the facility, with the conditions presented in subsequent sections of this report, we have concluded that the Browns Ferry Nuclear Plant, Unit 1 can be operated as proposed at power levels up to 3293 MWt without endangering the health and safety of the public.

2.0 SITE

2.1 Site Description, Land Use and Population

The site for the Browns Ferry Nuclear Plant is a 840 acre tract of land in Limestone County, Alabama, on the north shore of Wheeler Lake at Tennessee River Mile 294. The dominant character of the land is small scattered villages and homes in an agricultural area.

The population concentration in the vicinity of the plant is low and averaged 39 persons per square mile within a 4-mile radius of the site in 1970 and it is expected to increase slightly to the year

1990. The 1970 population within a 10-mile radius of the site is expected to increase from 19,580 to 26,130 by the year 1990 with a corresponding increase in population density from 79 to 98 persons per square mile. Within a 20-mile radius of the plant site, there are only three towns that have a 1970 population greater than approximately 1,800 persons. The population of Athens, 10 miles northeast of the site, is expected to increase from 14,360 in 1970 to approximately 25,000 in 1990. Decatur, 10 miles southwest of the site, is the nearest city with a population of 25,000 or greater and is expected to increase from a 1970 population of 38,044 to a 1990 population of 50,000. Within a 60-mile radius, the largest city is Huntsville, located about 30 miles due east from the site, with a 1970 population of 137,802 and an estimated 1990 population of 215,000 persons. Few population centers exist within a 60-mile radius of the site. Approximately 70% of the land in the counties nearest the site is still agricultural, but increasingly greater amounts of land are gradually being transferred to industrial use, especially along the Tennessee River and primarily at large centers of population. The nearest site boundary is approximately 4000 feet northeast of the reactor building, and the closest house is about 4500 feet from the reactor building. The applicant has proposed a minimum exclusion distance of 4,000 feet from the stack and the low population zone radius of 2.0 miles (3218 m). The population center distance is approximately 7 miles. The exclusion zone extends into

Wheeler Lake but does not intercept any land area on the shore opposite the plant site.

TVA has evaluated the consequences of marine accidents such as barge impact on the intake structure, oil spills in the intake structure, chlorine gas release from a pressurized cylinder and explosive forces from a barge located in the river channel and determined that plant safety would not be impaired. We concur with TVA's findings.

There are no airports within 5 miles of the site. The nearest commercial airport is at Huntsville about 25 miles from the site. Bay Airport, 8-1/2 miles west of the plant handles small aircraft with light traffic. Pryor Field in Decatur, Alabama, 10 miles east of the site also handles only small planes. There are no missile sites within 100 miles of the plant. We conclude that the site location with respect to these facilities is acceptable.

Based on the population surrounding the site and our independent analyses of the radiological effects from the design basis accidents described in Section 9.0 herein, we conclude that the TVA's proposed low population zone, population center distance and exclusion zone for the plant meets the AEC guidelines specified in 10 CFR Part 100.

2.2 Meteorology

The terrain in the vicinity of the plant is relatively flat. The airflow at low levels over the region is governed primarily by the large scale meteorological features rather than locally induced topographical flow.

The diffusion meteorology of the site has been described by data from a 300-foot onsite meteorological tower. Twenty-three months (February 1967 - December 1968) of hourly wind direction and speed at the 300-foot level and vertical temperature difference between the 300 and 75-foot levels were used to evaluate the atmospheric diffusion characteristics of the site for accidental and routine airborne releases of radioactive material. The data recovery during the 23 month period was 94%.

Our calculated values of atmospheric diffusion factors (χ/Q) were used in the evaluation of radiological doses for accidental and routine releases of airborne effluents. The meteorological data presented in the FSAR were considered adequate by the AEC regulatory staff and our consultant from the National Oceanic and Atmospheric Administration (NOAA) to make these evaluations with reasonable assurance that there would not be significant changes in the values as additional data are obtained. The report from NOAA is attached as Appendix B.

For evaluation of accidental releases from the plant vents, the joint frequency of wind speed at the 300-foot level was extrapolated to the 30-foot level above plant grade using the power law relationships suggested in a 1968 ASME Guide[1/] and vertical temperature difference was used. Assuming a ground level release with a building

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[1/] ASME Recommended Guide for the Prediction of the Dispersion of Airborne Effluents. (M. E. Smith, Editor, 1968) wake factor cA of $1200m^2$, the relative concentration which is exceeded 5% of the time at the minimum exclusion distance from the turbine building of 1200m is 3.4×10^{-4} sec/m^3 and at the outer boundary of the LPZ from the turbine building of 3218m is 1.3×10^{-4} sec/m^3. These values are equivalent to Pasquill Type F diffusion with a wind speed of 0.9 meters/second.

For evaluation of accidental releases from the 183m stack the joint frequency of unadjusted wind speed at the 300-foot level and vertical temperature difference was utilized. Assuming an elevated point release 183m above the ground the maximum relative concentration which is exceeded 5% of the time at or beyond the site boundary was found to be 9.8×10^{-7} sec/m^3 at the shortest site boundary distance from the stack (1465m). This value is the equivalent of Pasquill Type B diffusion with a wind speed of 5 m/sec and was used as the basis of our calculations of accident doses released from the stack described in Section 9.0 herein.

Although the AEC Safety Guide No. 3, which gives acceptable assumptions for evaluating the potential radiological consequences for a loss-of-coolant accident for BWRs, assumes fumigation conditions for a one half hour period during the initial two hour period of the accidental release of radioactive effluents, the probability of having fumigation conditions persisting for this period is less than 1%, and therefore was not considered in our calculations.

For longer time period accidental release calculations at the outer boundary of the LPZ (3218m), onsite meteorological data were used to modify Safety Guide 3 relative concentrations. Generally, atmospheric diffusion conditions for the stack release are better than those indicated in AEC Safety Guide 3 by a factor of five for the 0-8 hour and 8-24 hour time periods and by a factor of 2.5 for the 1-4 day and 4-30 day time periods with less than a 5% chance that these values may be exceeded.

Computations of annual average relative concentration for the stack release considering plume rise as a function of wind speed showed a maximum offsite value of 1.3×10^{-8} sec/m^3 east southwest of the stack at the site boundary 1600m from the stack. This value is lower than TVA's value by a factor of 2.3 primarily due to TVA's use of a different set of curves to determine the vertical plume spread with distance.

The limiting annual average relative concentration of 2.7×10^{-6} sec/m^3 for vent releases was found at the 1600 meter site boundary distance north northwest of the plant. This value is higher than TVA's value by a factor of 4.8 due to TVA's use of different dispersion parameters.

We conclude that the meteorological characteristics of the site have been determined adequately and provide an acceptable basis for determining routine gaseous effluent release limits, and for establishing a conservative meteorological model for use in the accident evaluations described in Section 9.0 of this evaluation report.

2.3 Hydrology

The site is on the north side of the Tennessee River Wheeler Reservoir about 19 miles upstream of Wheeler Dam, 55 miles downstream of Guntersville Dam, and about 30 miles due west of Huntsville, Alabama. Normal reservoir level is elevation 556 ft. MSL, average ground elevation at the site is 580 ft. MSL, and plant grade adjacent to the reservoir is elevation 565 ft. MSL. Cooling water for the three units is supplied from a river bank intake structure. A single trifurcated conduit supplies water for each unit. The intake structure pumps are mounted outdoors above plant grade, and will draw water from the intake structure sump which has a bottom elevation of 518 ft. MSL and an excavated 25 foot wide approach channel at elevation 523 ft. MSL to deep water in Wheeler Reservoir. Cooling water is discharged into the reservoir via three corrugated metal pipes, each of which is perforated for diffusion in an existing deep channel of the Tennessee River. The pipes extend 1010, 1610, and 2210 feet from the shoreline, respectively, with the last 600 feet of each used for diffusion.

The applicant has evaluated flooding from three sources, the Tennessee River, a local tributary west of the site, and from plant drainage. Each potential flooding source is discussed separately below:

a. Tennessee River

Historical streamflow recorded 40 miles upstream since 1937 indicates the maximum Tennessee streamflow after TVA dam construction occurred in February 1957 and was 293,000 cfs. The minimum recorded streamflow was 400 cfs in July 1966 and is attributed primarily to upstream regulation. The maximum flood of record in the region occurred in 1897 prior to construction of TVA dams with an estimated local maximum runoff rate of about 470,000 cfs.

TVA's evaluation initially assumed a hypothetical Tennessee River Flood, which we consider inadequate, being only about half as severe as the probable maximum flood (PMF). Subsequently, TVA performed an evaluation for a PMF based on the Weather Bureau's latest hydrometeorological estimates of probable maximum precipitation for the region, and determined that the peak runoff rate at the site would be about 1,200,000 cfs resulting in a river level elevation of about 572.5 ft. MSL.

This PMF determination included an extensive study of the runoff capability of the upstream 27,130 square mile drainage area and was greatly complicated by the necessity for determining the effects of more than 22 major TVA and 6 privately owned reservoirs. TVA found that the reservoir and outlet capacity of the four Tennessee River dams immediately upstream of the site would be insufficient to pass a PMF and, therefore, included the effects of their potential failure in the PMF estimate. TVA also assumed a sustained wind speed of 14 mph coincident with the maximum PMF river level, and has estimated the maximum corresponding wind wave runup level could reach elevations as high as 574 ft MSL. TVA assumed that the most likely month for a PMF was in March and used the mean March wind speed (14 mph) as the coincident wind. However, we have independently estimated the wind wave effects using the guidance provided by the Corps of Engineers for the plant area and accordingly estimate that a reasonably severe windstorm producing 45 mph sustained wind speeds could occur coincidentally with a PMF and produce a maximum wind wave runup level as high as an elevation of 580 ft. MSL.

TVA has verbally proposed procedures and design changes for shutting down the plant and maintaining it in a safe shutdown condition for floods which would exceed plant grade (elevation

565 ft. MSL) up to elevation 574 ft. MSL, corresponding to TVA's maximum PMF coincident with wind wave runup. The proposed procedures call for shutting down the reactor, sealing the diesel generator building, filling and/or closing radwaste tanks, filling the tori to prevent floatation, making pipe connections for raw water makeup and venting, and providing raw makeup water to the reactors and the spent fuel pools by pumps that are to be added to the present system.

However, we recommend as a condition for licensing that TVA's procedures provide for shutting down the plant and maintaining it in a safe shutdown condition considering our estimates for PMF and the effects of a concurrent wind speed of 45 mph with protection against water levels as high as elevation 580 ft. MSL.

Subject to meeting these guidelines, a final design together with emergency procedures, when implemented, will be adequate to protect the health and safety of the public. TVA has agreed to submit a final design for our review by September 1972.

b. Local Tributary

During the construction of the plant a local tributary was diverted into Wheeler Reservoir west of the site. The applicant was requested to provide an analysis of the capability of the tributary to flood safety related plant facilities as a result of a local PMF. TVA found the existing diversion channel and bridge incapable of passing floods up to the severity of local PMF (with a maximum runoff rate of about 14,000 cubic feet per second) without inundating the plant. Consequently, TVA has proposed modifying the diversion channel to safely pass a local PMF and has provided details of the changes. TVA has not, however, provided sufficient details to allow a review to be made of the adequacy of the proposed channel changes. TVA has stated that the final selection of the changes to be made will depend on the location of cooling towers, which are presently under construction on the site. Accordingly, our review of the adequacy of this channel will be resolved prior to licensing in a supplemental safety evaluation.

c. Plant Drainage

The applicant has also evaluated the flooding potential from surface drainage and the extensive roof surface area of the facility. The applicant has determined that the roof and its drainage are adequate for severe storms, but has indicated that modifications will be required to three service building doors and their seals to prevent flooding of the radwaste building. We conclude that these watertight seals can be made and that when these are installed and tested, protection against surface and roof drainage will be satisfactory.

Ground water at the site is derived from local precipitation, part of which percolates into the residuum. Deep regional ground water movement is prevented from reaching the site by local anticlinal and synclinal bedrock structures. All local ground water, as reported by the applicant, flows directly into Wheeler Reservoir. The 32 public ground water supplies within 20 miles of the site are not expected to be affected by plant operation. Since the onsite liquid radioactive waste storage is contained entirely within the radwaste building concrete structure which will be watertight and capable of the requirements of a Class I (seismic) structure, we conclude that there is little likelihood of accidental release of liquid radwastes to the ground. The eight private wells within one mile of the site have been surveyed and the applicant has stated that special local monitoring will be carried out in the event of any unusual release, even though there is also little likelihood of their contamination.

Cooling water is to be taken directly from Wheeler Reservoir. Adequate water supply is available for normal operation. However, we considered the limiting water supply condition that would occur following the effects of an assumed failure of the downstream Wheeler Dam. The applicant has estimated under these assumptions that the volume of water available in a large natural depression in the river bottom, coincident with minimum runoff, would still provide an adequate source of cooling water for safe shutdown cooling water requirements (45 cfs) for all three units. We conclude that adequate shutdown cooling water is available.

The Tennessee River from 12 miles upstream of the site to 49 miles downstream serves five public water supplies. Four intakes are downstream of the site, three of which are owned, operated, and controlled by TVA. TVA has stated that it will monitor both public and private supplies periodically. We concur with the applicant that there is little likelihood of contaminating public or private surface or ground water supplies based on conditions of storage and control of radioactive liquid effluent discussed in Section 8.2 herein, and that a suitable monitoring program is (as indicated by the applicant) a desirable safeguard for warning potable water users in the unlikely event of a spill.

In summary, we concur with the applicant's estimates of the magnitude of a PMF for the Tennessee River at the site, for the local unnamed tributary, and for plant site drainage. We disagree, however, with the applicant's wind assumptions on wave action, which could occur coincidentally with a maximum PMF level on the Tennessee River and conclude that adequate protection is available if the guidelines previously discussed are met by the applicant. We have also concluded that a seismically induced failure of any upstream dam would not impose flood conditions on the plant worse than a PMF.

We concur with the applicant that adequate water supply is available for both normal and emergency operation from the Tennessee River and Wheeler Reservoir.

We intend to submit our final evaluation on the outstanding hydrology concerns discussed in this section in a supplemental safety evaluation.

2.4 Geology, Seismology and Foundations

2.4.1 General

Previous conclusions derived from the regional geological and seismological analyses accomplished during the construction permit review remain valid, therefore, the earthquake design bases (.10 for the OBE and .20 for the DBE) remain appropriately conservative. The engineering design bases for the foundations of specific Category I structures, which are based on the physical properties of the subsurface materials, and the modifications imposed after excavation, appear to be reasonable and to represent standard engineering and construction practices for this type of geologic terrain. We consider these to be conservative for the safety of the Category I structures during conditions caused by a safe shutdown earthquake.

2.4.2 Geology

The site is located on the southern margin of the Highland Rim section of the Interior Low Plateaus according to Fenneman.

Structurally it is situated on the eastern flank of the Nashville structural dome. There are no known geologic structures that could be expected to localize earthquakes in the area. Relief in the area is moderate, varying in elevation from below +600 at Wheeler Lake to +800 northeast of Athens. Drainage is into Wheeler Lake by means of southeast flowing streams. The site is underlain by from 41 to 69 feet of clay residual soil which becomes gravelly as bedrock is approached. Bedrock consists of limestone of the Tuscumbia formation underlain by essentially dolomite of the Fort Payne formation. The surface of the Tuscumbia is pinnacled and the formation contains numerous cavities. The Fort Payne formation is essentially solid with calcite and quartz-filled vugs up to 1 inch diameter in the upper portion. The bedding within these materials is nearly horizontal with no evidence of displacements or structural deformation. Minor shears and thin brecciated zones were found and were interpreted as being caused by adjustments related to regional uplift at the end of the Paleozoic.

2.4.3 Seismology

The applicant's seismological evaluation consisted of a study of areas which have had or could produce earthquakes of significance to the site:

1. the Mississippi Valley located 170 miles to the northwest;
2. the Lower Wabash Valley in Indiana and Illinois, about 225 miles north northwest of the site;
3. the Southern Appalachian region 200 miles to the east;
4. the Charleston, South Carolina area, 420 miles east; and
5. an area of minor activity north and east of Huntsville, 35 to 45 miles from the site.

Based on this study, the applicant concluded that a Modified Mercalli Intensity of VII generated by an earthquake at an undetermined location was the maximum earthquake intensity that could possibly occur at the site within the lifetime of the plant. Our consultant, the U. S. Coast and Geodetic Survey (now NOAA), evaluated the seismicity of the area around the site and their report was attached as Appendix D of our previous Safety Analysis, dated March 31, 1967. In that report our consultant stated: "As indicated in both lists of earthquakes, Modified Mercalli Intensity VII attributed to the New Madrid, Missouri earthquakes of 1811-1812, is the greatest experienced in the region. Our estimate, based on the seismic history and the geology of the site, is that during the lifetime of the facility, the area will be subjected to MM Intensity VII with an acceleration of 0.10g for granite or massive limestone bedrock. In view of this, the Coast and Geodetic Survey agrees with the applicant that the assumption of ground acceleration of 0.20g will be adequate for the maximum potential earthquake." NOAA has orally acknowledged that this conclusion remains valid and we concur.

2.4.4 Foundations

Extensive subsurface investigations and design analyses were performed regarding site foundation conditions. Modifications were accomplished during construction to improve existing subsurface conditions.

The FSAR indicates that the reactor building and the pumping house structure are founded on bedrock of the Fort Payne formation. Where the base of the mat foundations of Units 2 and 3 were above the Fort Payne formation, over-excavation was accomplished and concrete backfill was placed up to foundation grade level. Unit 1, the foundation mat of which, lies within the Tuscumbia formation has been underpinned by means of concrete filled trenches resting directly on the Fort Payne formation. We believe that these are sufficiently conservative techniques of foundation treatment are sufficiently conservative for the specific subsurface conditions beneath the reactor buildings.

The stack foundation slab has been placed on and keyed into, the Tuscumbia formation. The applicant has stated orally that investigations were made and there were no solution channels or large cavities beneath this structure. Based on this statement and data in the FSAR we concluded that this design approach was adequate.

The foundation for the Diesel Generator Building and Standby Gas Treatment Building are located adjacent to the reactor complex on less than 10 feet of soil backfill, compacted to 95% Standard Proctor, overlying about 30 feet of crushed rock, which has been compacted in 4 to 6 inch lifts by a vibratory roller. The gradation of the rock was reported to range from 3/4" to fines. Conversation with TVA officials confirm that although a formal settlement monitoring program has not been carried out, there has been no evidence of excessive settlement since construction began. Dynamic analyses were accomplished using the earthquake design bases. We have concluded that the design approach is adequate.

The intake channel slopes were analyzed under the most severe adverse seismic conditions that have reasonable occurrence probabilities. Based on these analyses, the intake channels were originally constructed with a 3 horizontal to 1 vertical slopes. About 1-1/2 years after excavation a slide occurred along a portion of one of the channel slopes. This slide was caused by horizontal movement of a wedge of soil along a previously undetected layer of fat clay on top of bedrock. The displacement was initiated by full hydrostatic pressure buildup in a local vertical crack behind the slope. To prevent any possible reoccurences the channel section between the pumping station and construction dike was subsequently excavated to bedrock for a distance of 52 feet on both sides of the canal centerline and backfilled with rock fill. The rock forming the canal walls was placed at a 3:1 slope. Beyond the construction dike, in-situ slopes were flattened to 6 horizontal on 1 vertical. Dynamic and static analyses were performed by the applicant using standard techniques and TVA concluded that movement of the intake channel slopes due to the Design Basis Earthquake following a rapid drawdown of the reservoir would not pose a threat to safe shutdown of the reactor. The investigations and analyses were independently reviewed by the Corps of Engineers Waterways Experiment Station which stated that the investigations and analyses were reasonable and sufficient to provide a basis on which to initiate repair and construction of the slopes. The applicant stated orally on May 24, 1972 that since the fix there have been no indications of a resumption of sliding. Based on the information presented, we have concluded that the intake channel slopes are adequately designed to resist the effects of a Design Basis Earthquake.

2.5 Environmental Surveillance

The applicant has provided a three year preoperational environmental monitoring program to determine background radioactivity levels in the area of the site and will continue this program throughout the operation at the plant. This surveillance program has included the comments of the State of Alabama and the Southeastern

Radiological Health Laboratory (Public Health Service). The applicant has used thermoluminescent dosimeters to record the integrated gamma radiation background exposure at appropriate locations around the site. Radioactive particulates · the air are monitored by three local air monitors within the plant boundaries, four perimeter air monitors located at distances ten miles from the plant and five remote air monitors located at distances up to 45 miles from the plant. The particulate filters are removed weekly from each monitoring station and analyzed for gross beta activity. In addition, the filters for each station are composited monthly and quantitatively and qualitatively analyzed for specific gamma emitting radioisotopes. The charcoal filters are removed bi-weekly from each station and analyzed for Iodine 131. The applicant has included the sampling of vegetation, milk, rainwater, water, and marine samples which include sediment, plankton, fish, and clams.

Our consultant, the Fish and Wildlife Service of the United States Department of the Interior, whose report is attached as Appendix D, has reviewed the Browns Ferry Environmental Monitoring Program and has determined that it is adequate to safeguard the fish and wildlife resources in the project area.

We have reviewed the radiological monitoring aspects of this program and have concluded that they are adequate. We have reviewed TVA's Draft Environmental Statement which also includes nonradiological matters and have commented accordingly. The Technical Specifications for the Browns Ferry Plant as presently proposed consider only the radiological monitoring requirements of the environmental monitoring program. The applicant's proposed program either meets or exceeds the effluent measuring and reporting programs which are acceptable to the AEC regulatory staff as described in AEC Safety Guide 21, Measuring and Reporting of Effluents from Nuclear Power Plants.

We conclude that the applicant's program is adequate to monitor the radiological impact of plant operation on the environment and to assess the health and safety aspects of the release of radioactivity during plant operation.

3.0 REACTOR

3.1 General

Each nuclear steam supply system includes a General Electric Company (GE) boiling-water reactor (BWR) which generates steam for direct use in the steam-driven turbine-generator. The reactor core, containing nuclear fuel elements and control rods, is supported in a domed, cylindrical shroud inside the reactor vessel. Steam separators are mounted on the shroud dome. Two external, motor-driven recirculating pumps inject high-velocity water into 20 jet pumps which are located in the annulus between the shroud and the and the reactor vessel. The high velocity water from the jet nozzles entrains and imparts energy to additional water from the annular region. The combined liquid flow (about 3 times that of the high-velocity water flow) enters the bottom of the reactor core. This fluid becomes a steam-water mixture as it passes through and cools the reactor core. The steam emerges from the steam separators and dryers and enters four 26-inch diameter pipes leading to the turbine-generator.

Reactor power is controlled either by movement of control rods or by changing the speed of the two external recirculating pumps. Reactor power operation is terminated (reactor shutdown) by inserting control rods into the core. A standby liquid control system is provided as a backup system for reactor shutdown and operates by pumping a sodium pentaborate solution into the reactor.

3.2 Nuclear Design

The initial core to be used for the reactors will consist of three types of fuel assemblies. Two of the three types contain gadolinia in some of the assembly fuel rods which acts as a burnable poison to control the core excess reactivity throughout the operating cycle. The reference design described for the construction permit safety evaluation employed boron-steel curtains which now have been replaced by gadolinia. Type I fuel assemblies will contain an average uranium-235 enrichment of 1.1% and the fuel pins for this assembly will contain no gadolinia-uranium fuel pins. After approximately 10,000 MWD/ton average exposure the type I fuel elements will be removed from the core and replaced with type II and III fuel assemblies. After the first refueling cycle the type I fuel assemblies will no longer be used in the core. The type II and III fuel assemblies will contain an average uranium-235 enrichment of 2.5%. Type II and III fuel assemblies will contain fuel pins with five different uranium-235 enrichments to reduce the local power peaking factors. Three fuel pins in each type II and III fuel assemblies contain full length gadolinia-uranium fuel pins. In addition to the full length gadolinia-uranium fuel pins the type II and III fuel assemblies will contain one and two partial length gadolinia bearing fuel pins which provide axial flux shaping throughout the fuel cycle.

The end fittings of each fuel assembly are designed to assure that a higher enrichment fuel pin cannot be positioned in a lower enrichment fuel pin location. Following fabrication of the fuel pins each fuel pin will be gamma scanned to assure that the proper enriched uranium fuel pellets have been loaded. In addition five of the high enriched fuel assemblies will have two removable fuel pins to facilitate interim fuel pin inspection during their expected core life.

The unit cell of the core consists of a repeating array of three type II or III fuel assemblies and one type I fuel assembly. The type I assemblies have flow restricting orifices to provide proper flow distribution to meet thermal hydraulic limits. Details of the changes in the physical dimensions of the fuel pin design are given in Section 3.0 of the FSAR.

Our evaluation of the nuclear design indicates the characteristics of this gadolinia controlled core will be similar to the gadolinia cores used in the Quad-Cities and Dresden 2 reactors, which were extensively reviewed and evaluated during our operating license review of these facilities. The adequacy of the calculations to predict gadolinia controlled initial core reactivities, have been confirmed by critical experiments, Dresden I tests, and Quad-Cities start-up tests. The design criteria used in the Quad-Cities core design have been applied to the Browns Ferry core design. Even after burn-out of the highly absorbing gadolinium isotope, the power density in a gadolinia fuel pin is sufficiently lower than the power density in a peak uranium-oxide fuel pin so that thermal-mechanical limits such as MCHFR, center line melting and one percent clad strain, are limited by the uranium oxide fuel pins. The extent of reduction of these limits in the gadolinia fuel pins resulting from changes in the thermal conductivities and lower melting point are the same as those evaluated in our review of the Quad-Cities core. The principal difference in the use of gadolinia between the Quad-Cities and the Browns Ferry designs is a stronger degree of axial power shaping for the Browns Ferry cores. This increased axial flux shaping capability is expected to improve the end of cycle power distribution.

The design limits for normal operation, MCHFR $\leq$1.9 and maximum power density of 18.5 kW/ft, are unchanged from the referenced design evaluated for the plant construction permit applications. The peak power densities and peak assembly power distributions used in the LOCA analysis remain within the range studied in General Electric Company Report, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, Supplement 1 (April 1971). From our review of the power distribution information we conclude that power distribution, as monitored by incore instrumentation, will maintain adequate safety margins.

Other nuclear design parameters which are important for abnormal operational transient analysis such as moderator and Doppler reactivity coefficients and the scram reactivity function, have undergone relatively minor changes as a direct result of the changes in the gadolinia cores.

In Amendment 21 to the FSAR, TVA has presented revised transient analyses which take into account the improvements in core design involving control augmentation using gadolinia and enrichment distribution, changes in fuel design, nuclear parameters, scram reactivity function, and calculational models. Our review of these revised transient analyses indicates the results still meet the thermal limit criteria such as MCHFR >1.0 and maximum linear heat generation rate less than that which would result in a clad strain greater than 1% and have been found to be adequate in previously reviewed plants. The major effect on transient results occurred from the change in the scram reactivity function. The nuclear steam system supplier, General Electric Company, performed parametric studies on the moderator void and Doppler coefficients for the more important transients. For these transients the changes in the nuclear parameters produced relatively minor changes in the results. We have concluded from our review, that in general, satisfactory parameters are being used in the normal and transient analysis of the core design.

3.3 Core Thermal and Hydraulic Design

The core design power level (3293 MWt) for each reactor is the same as reviewed during the construction permit review. The design core power density is 18.35 kW/ft and is the same as for Vermont Yankee, Peach Bottom, and Cooper. Our evaluation of the thermal and hydraulic design criteria of Browns Ferry is on the same bases that we reviewed Vermont Yankee.

Our review of the applicant's analyses of the various transients that can be expected to occur during the lifetime of the plant indicated that the analyses are the same as those previously approved for Vermont Yankee. The core thermal and hydraulic design basis is to control the local power density within the core to levels that assure that the fuel heat flux is maintained within acceptable limits so that the fuel rods do not overheat during normal plant operation including operational transients.

The controlling mechanism that could cause fuel damage in reactor transients is severe overheating of the fuel cladding caused by inadequate cooling if critical heat flux conditions in the core are exceeded. The critical heat flux is defined as that which occurs on the fuel cladding at the onset of the transition from nucleate boiling to film boiling and below which fuel damage does not occur. For design purposes the critical heat flux is conservatively used as a fuel thermal limit although actual fuel damage may not occur until well into the film boiling regime. The present critical heat flux limits are calculated using the correlation reported in the GE topical report APED-5286, "Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors," issued in 1966. This correlation is based on experimental data taken over the range of conditions representative of BWRs. The minimum critical heat flux ratio (MCHFR) is defined as the ratio of the critical heat flux correlation value at the corresponding fluid conditions to the actual maximum calculated heat flux occurring at a given point in the fuel assembly at any time during operation including reactor anticipated transients. A MCHFR > 1.0 conservatively assures that cooling of the fuel is maintained through nucleate boiling heat transfer.

The current design basis for normal operation is that the MCHFR calculated for any point is greater than 1.9 during normal operation and greater than 1.0 during anticipated transients. These limits provide considerable margin between expected conditions and those required to cause fuel clad damage since the critical heat flux correlation presented in APED-5286 is conservatively based on a limit line drawn below all of the available experimental data points.

We have reviewed the methods used to calculate the MCHFR, the experimental basis for the calculation, its validity as a damage limit and the applicant's analyses of normal operation and anticipated transients for this station and previously reviewed reactors, and conclude that the design provides adequate margin to protect the core against fuel damage.

3.4 Reactivity Control

Reactor power can be controlled by either movement of control rods or variation in reactor coolant recirculation system flow rate. A standby liquid control system is also provided as a backup shutdown system.

There are 185 control rods which are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each control rod drive has separate control and rapid insertion (scram) devices. A common hydraulic pressure source for normal operation and a common dump volume for scram operation are used for the drives. On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in other General Electric reactors, we conclude that the installed system will meet the functional performance requirements for each reactor in a safe manner.

During operation at power levels between zero to 10% of the rated power, control rod reactivity worths are limited by the rod worth minimizer (RWM), a device which utilizes a computer to restrict control rod patterns such that the total worth of any insequence rod that can be moved will be no more than 1% Δk. For reactor power levels in excess of 10% of the rated power, when the RWM is inoperable, the maximum worth of any control rod that could be established is limited to less than 2.0% Δk. Under these limiting conditions of operation, the calculations of the consequences of a control-rod-drop accident (discussed in section 9.0) indicate that the peak fuel enthalpy is well below the threshold value (280 cal/gm) assumed to cause fuel failure and damaging pressure pulses to the reactor core and that the radiological doses at the site boundary from the estimated fuel cladding failures are well within the guidelines of 10 CFR Part 100. Accordingly, we have concluded that use of the RWM is not required at power levels above 10%.

A control-rod-ejection accident is precluded by a control rod housing support structure located below the reactor pressure vessel, similar to that installed on the other large General Electric reactors. This structure limits the distance that a ruptured control rod drive housing could be displaced. The applicant concluded and we agree, that the control rod displacement would be so small in this event that any resulting nuclear transient could not be sufficient to cause fuel rod failure.

Reactor power can also be controlled through changes in the primary coolant recirculation flow rate. The recirculation flow control system is the normal control method used to adjust reactor power level to station load demand whenever the reactor is operating between approximately 60% to 100% rated power. The recirculation flow control system is designed to allow either manual or automatic control of reactor power. This method of reactor power control has been demonstrated to be acceptable in the Dresden Units 2 and 3, Monticello and Millstone I facilities.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life, independent of the control rod system capabilities. The injection rate of the system is adequate to compensate for the effects of xenon burnup and decay.

Each of the foregoing design features is similar to the corresponding features provided in plants we have previously reviewed. On the basis of our previous review of similar designs and of satisfactory operating experience with similar systems in other operating BWRs, we conclude that the mechanical, thermal and hydraulic, and reactivity control features of each reactor is acceptable.

3.5 Reactor Internals

3.5.1 Design Criteria

The reactor vessel internals have been designed to function within the acceptable stress limit criteria of Article 4, Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition for all design loading conditions of mechanical, hydraulic and thermal origin, including anticipated plant transients and the operational basis earthquake.

The reactor internals have also been designed to provide for the maintenance of a coolable core configuration and for safe shutdown of the plant under the loads from the Design Basis Accident, the Design Basis Earthquake and a combination of these events. The DBA load calculations considered both a steamline break and a recirculation line break. The break which resulted in the most severe loading condition was used as the DBA loading for each component analyzed.

Calculated primary stresses in the reactor internals under the above loading combinations are within the acceptable emergency and faulted stress limits specified in the current component codes. Deflections of the fuel channels, control rod housings and core support structure under the above loading combinations were limited to assure control rod operation and the preservation of core cooling geometry.

The highest peak stresses in the internals assembly and its supporting structure occur during the thermal transients resulting from the DBA and subsequent startup and operation of the emergency core cooling systems. Thermal stress analyses were conducted for the shroud at the point of highest predicted irradiation, the baffle plate which supports the jet pump diffusers, the diffusers, and the various welded joints that connect these components. In some instances the elastically calculated upper bounds for the peak strain ranges resulting from these thermal stresses exceed slightly the design fatigue limits for 10 cycles as specified in ASME Section III Code. In these instances a combination of elastic and plastic analyses were applied to verify that structural integrity of these members can be maintained during and after the postulated loading conditions and that the resulting distortions do not impair core flooding and cooling capabilities.

We have concluded that the design loading conditions, design stress limits, deflection limits, and design fatigue analyses as applied to the reactor vessel internals are acceptable and that adequate margins of safety are available to provide reasonable assurance that core flooding and cooling capabilities will not be impaired under the most severe loading conditions.

3.5.2 Dynamic System Seismic, Operating and LOCA Analysis

Seismic loading on the core support structure has been determined by means of a multimass dynamic analysis using a lumped mass mathematical model of the reactor pressure vessel, internals and core support structure coupled with the containment building soil-structure model. We find this procedure acceptable.

Design loadings for the postulated Loss-of-Coolant Accident (LOCA) have been determined by computing the response of each structural member to the calculated peak pressure differential applied as an equivalent static load. In response to our concerns regarding the validity of this static analysis, the applicant has stated that the natural frequency of the BWR internal structures is more than ten times the calculated forced frequency of the LOCA loads thus assuring no significant dynamic amplification. On the basis of the information submitted by the applicant we find this analytical method acceptable.

3.5.3 Vibration Control

The applicant has planned for vibration tests of reactor internals in Units 1, 2 and 3 during plant start-up. During these tests, for Unit 1, the displacement of the shroud and a jet pump relative to the reactor pressure vessel wall will be monitored, the separator motions will be recorded with accelerometers, strain levels will be recorded from a jet pump riser brace and the guide tube. Measurements will also be provided for Units 2 and 3 based on the results of tests made on Unit 1. The data obtained should be sufficient to verify that the steady state and cyclic stresses in the components, as determined by analyses, are within the acceptable design limits set forth in the design specifications and code requirements and that the results meet the acceptance criteria of the vibration test program.

At the present time, the vibration test program for Browns Ferry Unit 1 does not qualify as a prototype test program since the applicant has not submitted quasi analytical predictions of the dynamic response loadings that will be encountered during the vibration test program. We will also require that visual and nondestructive surface examinations of the reactor internals for all three units be performed following the vibration testing program to meet the intent of AEC Safety Guide 20, Vibration Measurements of Reactor Internals, which provides an acceptable method for evaluating by test whether flow-induced vibrations similar in nature to those expected during operation will not cause damage to reactor internals important to safety. The adequacy of TVA's vibration monitoring program will be resolved prior to licensing.

4.0 REACTOR COOLANT SYSTEM

4.1 General

The principal components of the reactor coolant system include the reactor vessel, the reactor vessel internals, the two recirculation pumps and lines, the main steam and feedwater lines, the pressure relief system, and portions of the primary coolant auxiliary systems, i.e., the reactor core isolation system (RCIC), the residual heat removal system (RHR), and the reactor water cleanup system. Portions of these systems as well as other piping extend from the reactor vessel up to the second isolation valve. All components of the system were designed to applicable codes in effect at the time the components were ordered.

4.2 Reactor Coolant Pressure Boundary - Design

The reactor coolant system was designed as a Class 1 (seismic) system to withstand normal design loads of mechanical, hydraulic and thermal origin, including anticipated transients and the operational basis earthquake within the acceptable stress limits of the applicable codes specified below.

Additional analyses of the reactor coolant system have confirmed that the stress levels calculated under loads from the Design Basis Accident, the Design Basis Earthquake and the combination of these events are within the acceptable emergency and faulted stress limits, respectively, of current component codes.

The reactor pressure vessel was designed, fabricated, and inspected to the Class A requirements of Section III of the ASME Boiler and Pressure Vessel Code (B & PV code), 1965 Edition including published addenda through and including 1965 summer addenda for Units 1 & 2 vessels and the 1966 summer addenda for Unit 3 vessel.

Reactor coolant system piping was designed, fabricated and inspected in accordance with the USAS B 31.1.0 - 1967 Power Piping Code. Additional nondestructive inspection requirements were applied in accordance with the requirements of the Power Piping Code Cases N2, N7, N9 and N10. The recirculation lines have been provided with a system of pipe restraints designed to limit pipe motion in the event of either a circumferential break or a longitudinal split. These motion restraints have been designed within acceptable stress limits to permit normal and necessary pipe movements due to pressure and thermal expansion.

Reactor coolant system valves were designed, fabricated and inspected under the rules of the USAS B31.1.0 - 1967 Code. Additional level 2 nondestructive inspection requirements were applied in accordance with the requirements of Power Piping Code Cases N2, N7, N9 and N10.

The recirculation pump casing was designed in accordance with the Class C requirements of Section III of the ASME B & PV Code 1965 Edition, including the winter 1965 Addenda.

We have found that the codes and published addenda used in the design of the reactor coolant system are acceptable.

In accordance with Paragraph 101.5.4 of USAS B31.1, "Power Piping," which requires that piping be arranged and supported with consideration of vibration, a vibration operational test program will be performed during startup and initial operating conditions. These tests will be conducted to verify that the piping and piping restraints within the RCPB have been designed to withstand dynamic effects due to valve closures and pump trips. The tests will develop loads similar to those experienced during reactor operation and provide an acceptable basis for conducting the vibration operational test program.

4.3 Fracture Toughness Criteria

The reactor vessel has been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Recent fracture toughness test data, however, indicate that the current ASME Code rules do not always assure adequate fracture toughness of ferritic materials. The fracture toughness data submitted by the applicant meet the current requirements of Section III of the ASME Code, but are not adequate to establish compliance with the proposed Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," (36 Fed. Reg. 12697, July 3, 1971).

We have reviewed the available fracture toughness data for the reactor vessel and have applied proposed Appendix G to arrive at a lowest pressurization temperature of 185°F.

We intend to specify the following limits in the Technical Specifications, to be applicable during the first five years of operation, or until the first material surveillance specimens are withdrawn, whichever occurs first.

1. The reactor coolant system shall be operated in such a manner that at temperatures below 185°F, the pressure does not exceed 255 psig, i.e., 25% of the normal operating pressure.
2. Operation of the reactor coolant system at full pressure will be acceptable at temperatures above 185°F.
3. The reactor coolant system may be subjected to isothermal hydrostatic tests at temperatures below 185°F provided that the test pressure does not exceed 510 psig i. e. 50% of the normal operating pressure.

4.4 Reactor Vessel Material Surveillance Program

The proposed material surveillance program was planned in accordance with ASTM-E-185-66 and meets the requirements of that specification.

The proposed program is consistent with programs which we have accepted for previous BWR plants and is acceptable.

The proposed material surveillance program also complies with proposed Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," (36 Fed. Reg. 12697, July 3, 1971) except for the requirements for material chemistry documentation.

We have concluded that the proposed program will adequately monitor neutron radiation induced changes in the fracture toughness properties of the ferritic materials of the plant's reactor vessels during their service life.

4.5 Sensitized Stainless Steel

Stainless steel that has been sensitized has an increased susceptibility to stress corrosion cracking. All sensitized stainless steel has been replaced on the Browns Ferry pressure vessels except the recirculation system safe-ends on Unit 1 and the jet pump riser brace pads on all units. These components instead have been clad with stainless steel weld overlay. Austenitic stainless steel used in other component parts of the reactor coolant pressure boundary including relief and safety valves has been fully annealed prior to installation to preclude sensitization. All welding processes have been limited to 110,000 joules per inch and the interpass temperature has been limited to 350°F to avoid local sensitization of stainless steel.

Stainless steel with deliberate additions of nitrogen for enhancing the material strength has not been used. To prevent gas entrapment, all high points on non-flowing parts of the reactor coolant system have been vented to prevent gas entrapment.

We have concluded that the planning to avoid sensitization of austenitic stainless steel during the fabrication period is acceptable and that the sensitized has been either removed or adequately clad.

4.6 Electroslag Welding

Electroslag welding has been used in the fabrication of the Browns Ferry pressure vessels. The electroslag welding process variables, quality control procedures, and technical details were the same as those used in the fabrication of Dresden Nuclear Power Station, Units 2 and 3, and based on our review of that facility are acceptable.

4.7 Leakage Detection System

The leakage detection system provided for the reactor coolant pressure boundary is sensitive, includes diverse leak detection methods, and is equipped with suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate, gaseous and halogen radioactivity monitors and the level and flow rate indicators and recorders on the containment sump. Indirect indication of leakage can be obtained from the containment pressure, temperature and humidity indicators.

We have concluded that the leakage detection system provides acceptable redundancy and diversity and provides detection sensitivities capable of detecting small leakage in the reactor coolant system, and consistent with other approved BWR plants is acceptable.

4.8 Inservice Inspection Program

An inservice inspection program for the reactor coolant system is described in the proposed Technical Specifications. This program complies with section XI of the ASME Code "Inservice Inspection of Nuclear Reactor Coolant Systems" (January 1, 1970) to the extent permitted by the existing design. Accessibility to the reactor coolant system has been provided within the limits of the plant design as of that date. Access has been provided for critical areas such as vessel nozzle welds and dissimilar metal welds. The specific components to be inspected, the frequency of inspection, and the type of inspection to be made for each item listed under Examination Categories, and Components Parts and Methods of Inspection of Section XI have been identified.

The applicant is participating in a development program for inservice inspection systems. Reliable processes or systems developed by this program will be incorporated into the inspection program.

The proposed inservice inspection program satisfies the provisions of the January 31, 1969 AEC document, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for

Inservice Inspection." We therefore conclude that the program is acceptable.

4.9 Pressure Relief System

Overpressure protection for the reactor vessel is provided by safety valves and relief valves similar to those used at other BWR plants. The safety valves, two in number, are a balanced, spring-loaded type. The eleven relief valves for the reactor coolant system are dual-purpose valves to provide pressure relieving capability for the reactor coolant system and have the capability to provide reactor depressurization in conjunction with the emergency core cooling systems (i.e., ADS function described in Section 6.4). However, only six of the relief valves are assigned to the ADS system. The safety valves and the relief valves when acting as pressure relief devices are sized to limit reactor pressure to below design if the main steam isolation valves are closed, which is the most severe reactor overpressure transient.

The applicant has submitted a Summary Technical Report, Reactor Vessel Overpressure Protection, which has been prepared in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code. This report describes the design of the valves and an analysis of the reactor pressure limiting capabilities of the valves. We have reviewed this report and conclude that the specified relieving capacity of the safety valves provided for overpressure protection of the reactor coolant pressure boundary meets the intent of the ASME Code and is therefore acceptable.

4.10 Reactor Coolant Auxiliary Systems

The reactor coolant auxiliary systems consist of the reactor core isolation cooling systems (RCIC), the reactor shutdown and torus water cooling modes of the residual heat removal system (RHR), the reactor water cleanup system, and the main steamline and feedwater piping. The reactor coolant auxiliary systems are designed as Class I (seismic) systems except for the reactor water cleanup system external to the primary containment and the main steamlines and feedwater lines outside the primary containment. All piping in these systems is designed and fabricated to the requirements of the Power Piping Code, USAS B31.1.0 - 1967. We have reviewed the design, fabrication and inspection requirements used for those piping systems and find them acceptable. We have also reviewed the various codes and specifications used for the design, fabrication and inspection of the various tanks and heat exchangers included in these systems and found them acceptable for their respective applications.

The Reactor Core Isolation Cooling System (RCIC) is of the same design and serves the same function as in the Monticello, Quad-Cities and Vermont Yankee designs. The function is to supply about 600 gpm of water to the reactor vessel so that the core does not become uncovered in the event that the vessel is isolated from the feedwater system. This condition would occur in the event of a loss of all offsite power. In this case, upon isolation of the reactor, the relief valves and the RCIC system would be actuated so as to remove the core decay heat through blowdown of steam and concurrently to maintain water level.

The Residual Heat Removal System (RHR) consists of two interconnected low pressure cooling loops connected to the primary coolant recirculation loops by a single suction line and return lines to the reactor inlet side of each recirculation loop. Each loop contains two pumps and two heat exchangers in parallel. In addition to its function as a Core Standby Cooling System, the RHR provides a means of removing residual heat produced in the core by radioactive decay so that refueling and servicing can be performed. The RHR may also provide cooling to the suppression pool for RCIC operation, and may be used to supplement the fuel pool cooling system when necessary.

The Reactor Water Cleanup System provides a means to maintain high reactor water purity to limit chemical and corrosive action within the primary coolant system, to remove corrosion products in the reactor coolant and thereby limit impurities available to neutron flux activation and for decreasing reactor water inventory during heatup.

The main steamline and feedwater piping systems provide for the routing of the reactor steam to the main turbine generator and the supply of reactor makeup water respectively. The main steamlines are fitted with flow-restrictors to reduce the rate of coolant loss in the event of a main steamline rupture outside of the primary containment. Main steamline isolation valves both inside and outside the containment provide a redundant means of quickly terminating steam blowdown during this accident.

We have reviewed the above features and systems of the Reactor Coolant Auxiliary Systems on the basis of their similarity to those we have previously reviewed and have found acceptable for other reactors now in operation and we have concluded that they are acceptable.

4.11 Main Steam Line Isolation Valve Leakage

Leakage through the closed Main Steam Line Isolation Valves following a postulated LOCA results in an uncertainty in calculating potential radiological doses. The applicant will study systems which could be added to the present design to further limit potential leakage through the Main Steam Line Isolation Valve following a postulated LOCA.

5.0 CONTAINMENT SYSTEMS

5.1 General

The containment systems include the primary containment which utilizes the pressure suppression concept and the secondary containment which is formed by the low-leakage reactor building that surrounds the primary containment. The reactor building has an air recirculation system and a Standby Gas Treatment System (SGTS) to mix and filter primary containment leakage prior to its discharge to the environment.

5.2 Primary Containment

5.2.1 Design

The primary containment is a typical "lightbulb" pressure suppression system consisting of a drywell, pressure suppression chamber (torus), and a connecting vent system. The drywell has a steel spherical lower portion 67 feet in diameter, and a steel cylindrical upper portion 38 feet 6 inches in diameter. Overall height of the drywell is about 115 feet. The pressure suppression chamber is a steel torus located below and encircling the drywell, with a centerline diameter of approximately 111 feet and a cross-sectional diameter of 31 feet. Eight vent pipes lead from the drywell to a header inside the torus, and 96 downcomer 24 inch diameter pipes project downward from the header and terminate approximately 4 feet below the surface of the torus pool. The free air volumes in the drywell and torus are approximately 159,000 ft^3 and 119,000 ft^3 respectively. The torus pool contains about 135,000 ft^3 of water. In the event of a reactor coolant system pipe rupture within the drywell, the released steam passes through the vent pipes, torus header, and downcomer pipes into the torus pool water where it is condensed. This transfer of energy into the pool water reduces the peak accident pressure that otherwise would be experienced by the primary containment.

The applicant has calculated that the peak pressures that might be reached as a result of the design basis loss-of-coolant accident are 49.1 psig in the drywell and 27 psig in the torus. These pressures were calculated assuming a hypothetical instantaneous break of one recirculation loop pipe. The analytical methods used are similar to those used on other recently reviewed BWR plants and have been verified by comparison with the results of tests performed at the Moss Landing test facility.

The primary containment is designed for an internal pressure of 56 psig coincident with a temperature of 281°F. The design leak rate for the containment is 0.5% per day. In accordance with Section III of the ASME Boiler and Pressure Vessel Code, maximum drywell pressures up to 62 psig are permissible for this design. Combinations of live, dead, and seismic loads in conjunction with thermal stresses have been considered in the design analysis. The design also considered the jet forces that might act on the containment consequent to a pipe severance. Adequate strength has been provided to prevent failure of the containment wall as a result of direct jet impingement, and all pressurized penetrations have been supported with anchors and limit stops to limit pipe movement and prevent failure of the containment.

The primary containment was designed to sustain the combination of loads resulting from the design basis loss-of-coolant accident, the Operational Basis Earthquake, and the conventional live and dead loads within the stress limits defined in Subsection III B of the ASME Boiler and Pressure Vessel Code (1965) and applicable addenda in effect as of April 1967. We find the design stress limits for the primary containment system to be acceptable.

Containment piping penetrations which must sustain large thermal movements utilize a multiple flued fitting to accommodate the use of bellows expansion joints and a guard pipe concentric to the process line, e.g., steam piping. The function of the guard pipe is to protect the bellows in the event of a rupture of the process line and maintain the leak tight integrity of the containment. Jet force deflector plates are also included in large penetration assembly designs. All two-ply bellows expansion joints utilized meet the provisions of ASME Boiler and Pressure Vessel Code Cases 1177-5 and 1330-1 and will be capable of being tested for leakage. We find these design provisions for containment penetrations to be acceptable.

Based on our review of the information contained in this application and similar designs, we conclude that the primary containment design basis is acceptable.

5.2.2 Missile and Pipe Whip Protection

The applicant has considered the effect of missiles ranging in size from nuts and bolts to valve bonnets, and concludes that no missile would have sufficient energy to penetrate the drywell wall. In addition, where possible, components are arranged so that the direction of flight of potential missiles is away from the containment wall.

If a high pressure pipe were to rupture within the drywell, the containment shell might be damaged in three different ways. These are direct impingement on the wall of the jet of fluid issuing from the broken pipe, the reaction forces of the jet acting on containment penetrations, and impact of a pipe that is moved by jet forces (pipe whipping). The plant design includes provisions in the design to reduce the possibility of containment failure as a result of these effects.

The direct impingement of a jet on the containment wall has been considered in the design of the containment, and adequate strength has been provided to prevent failure as a result of such impingement. Reaction loads acting on containment penetrations have also been considered in the design, and anchors and limit stops located outside the containment have been provided to limit pipe movement and prevent failure of the containment. To prevent pipe whip from causing failure of the containment, two design approaches have been taken. In the first approach the reactor coolant system recirculation lines have been provided with restraints which will prevent these lines from whipping in the event one ruptures. This design approach was not applied to the other lines within the drywell, such as the steam and feedwater lines. However, the applicant is protecting the lower spherical portion of the drywell wall with energy adsorbing material. The material is a corrugated steel plate sandwich which can plastically deform to absorb the energy of a whipping pipe and is the same material previously proposed for and used in Vermont Yankee. This material provides protection to the entrainment against the effects of whipping of the main steam, feedwater, and RHR pipelines. In addition, TVA will inspect the critical welds of this unrestrained piping inside a drywell at a more frequent interval than that required by the inservice inspection program. The probability of failure of these lines is therefore minimized because of the accelerated inservice inspection program and because at the leak detection capabilities at the units. We therefore conclude that since the majority of the piping in the containment is either restrained or the containment is protected against its failure, and the remainder of the piping is of high quality, frequently inspected and continuously monitored for leakage, the probability of violating the integrity of the containment is acceptably low.

5.2.3 Leakage Testing Program

The primary containment and components which will be subjected to containment test conditions were designed so that periodic integrated leakage rate testing can be conducted at peak accident pressure. We have reviewed the proposed test procedures for determination of the primary containment overall leakage, as well as penetration and isolation valve leakage, for both preservice and inservice containment leakage tests.

Penetrations, including personnel and equipment hatches and airlocks, and isolation valves, have and are being designed with the capability of being individually leak tested at peak accident pressure.

We conclude that design of the primary containment system will permit the conduct of a containment leakage testing program in compliance with the requirements set forth in proposed Appendix J to 10 CFR Part 50, "Reactor Containment Leakage Testing for Water Cooled Power Reactors" (36 Fed. Reg. 17053, Aug. 27, 1971).

In addition to agreeing to meet the requirements of proposed Appendix J, the applicant has agreed to perform a leak test of each unit's drywell to suppression chamber interconnecting vent pipes, headers and appurtenances at each refueling outage. Leakage in excess of the equivalent of a one-inch plate orifice which could cause the pressure resulting from blowdown into theccontainment to exceed the containment design pressure will be corrected prior to the resumption of power operation. We conclude that this test together with frequent surveillance testing of the vacuum breakers will be adequate to detect possible excessive bypassing of the suppression pool during postulated accidents.

5.2.4 Containment Atmosphere Control

Following a loss-of-coolant accident (LOCA), (a) hydrogen gas could be generated inside the primary containment from a chemical reaction between the fuel rod cladding and steam (metal-water reaction), and (b) both hydrogen and oxygen would be generated as a result of radiolytic decomposition of recirculating coolant solutions. If a sufficient amount of the hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent reaction of hydrogen with oxygen can occur at rates rapid enough to lead to a significant pressure increase in the containment. This could cause damage to the containment and could lead to failure of the containment to maintain low leakage integrity.

General Design Criterion 41 of Appendix A to 10 CFR Part 50 requires that systems to control hydrogen, oxygen and other substances which may be released into the primary containment be provided as necessary to control their concentrations following postulated accidents to ensure that containment integrity is maintained. In accordance with guidelines of the supplement to Safety Guide 7 "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," the applicant has proposed a Containment Atmospheric Dilution System (CAD). Presently this system is only a conceptual design of a system using nitrogen dilution for the control of combustible gases and is being considered as a backfit item in accordance with the supplement to Safety Guide 7.

Basically the CAD concept involves the maintenance of an oxygen deficient (inert) containment atmosphere in the post-LOCA period. This would be accomplished by addition of nitrogen gas from an external nitrogen makeup and supply system. As nitrogen is added, the containment pressure would rise in the post-LOCA period. However, even assuming a zero containment leakage rate in the post-LOCA period, the containment pressure would reach about 40 psig within 30 days following the accident. Assuming that no accident recovery actions were to be undertaken after the 30-day period, it would take about 2 months before the containment design pressure (56 psig) could be reached. Under this condition, containment purging under long-term controlled conditions would be necessary to prevent excess pressure rise and to allow the introduction of nitrogen to maintain the hydrogen-oxygen balance below the flammable limits and resultant radiological doses would not exceed the 10 CFR Part 100 guideline values. If the containment is assumed to have a leakage rate greater than zero,or leak at a rate of 2 w/o per day, as is the situation postulated for analyses of the radiological consequences of a LOCA, the containment pressure would not exceed about 35 psig at any time during the post-LOCA period. Consequently, use of a CAD as conceived should allow the control of combustible gases to be accomplished in the post-LOCA period, while at the same time its usage should not increase the presently predicted radiological consequences of a LOCA.

The applicant has not provided us with final design information or answers to our questions, consequently our review is not yet complete. The applicant will provide this information by about September 1972 so that we may complete our review. In order to reasonably assure system effectiveness as an engineered safety feature, considerable upgrading of the existing inerting and purge systems will be required. Some of our concerns on the system are related to the conservativeness of the applicant's analysis, the mixing capability of the gases within containment, and the structural and leak-tightness capability of the containment now needed for a long period of time following a loss-of-coolant accident.

Pending the satisfactory completion of the review of the system, we will consider this matter as a condition to the license of Unit 1 and will be resolved prior to the licensing of Unit 2.

5.2.5 Isolation Valves

The basic function of all primary containment isolation valves is to provide containment integrity between the primary coolant system pressure boundary or the containment atmosphere and the environs in the event of accidents or similar equipment failures. Where necessary the valves are provided with valve operators, and these valves are automatically closed when the sensors detect certain accident or faulted conditions. The consequences of postulated pipe failures both inside and outside the containment have been evaluated. For example, the operational aspects of the main steam line isolation valves for a steamline break outside the containment are described in the accident analysis given in Section 9.

As a safety system, the isolation valves and their control systems have been reviewed to assure that no single accident or failure can result in a loss of containment integrity. An exception occurs in the case of the instrument lines that connect to the reactor primary coolant system, penetrate the containment and dead-end in instrument transducers located in the reactor building. These 1-inch lines have only two isolation valves, both of which are outside the containment.

The inboard valve nearest the containment is a hand-operated globe valve. The second valve, immediately adjacent, is a spring-loaded excess flow check valve with position indication. A break in the portion of the instrument line between the containment and the excess flow check valve would result in a blowdown directly to the reactor building. We have reviewed these isolation provisions consistent with the guidelines of the Supplement to Safety Guide 11, "Instrument Lines Penetrating Primary Reactor Containment Backfitting Considerations."

The applicant has installed orifices in each of these lines inside the primary containment. The orifice size (1/4-inch diameter) selected is sufficiently small that the quantity of coolant that would be discharged from the reactor into the reactor building in the event of a rupture of an instrument line is limited, would not result in a failure of the secondary containment, and would not affect the operability of the standby gas treatment system. The potential offsite exposure would be substantially below the guideline values of 10 CFR Part 100. The applicant has also proposed an adequate method of verifying the status (open or closed) of each isolation valve. Based on our review of the design, we conclude that the instrument lines penetrating the primary containment are adequately designed and meet the intent of the Supplement to Safety Guide 11.

5.3 Secondary Containment

The secondary containment (reactor building) is a Class I (seismic) building common to the three units and encloses the primary containment vessel of each reactor, and contains the refueling facilities and other equipment provided to support the operation of each reactor. Up to the refueling floor, the reactor building is a reinforced concrete structure and, above this, a structural steel frame covered with insulated metal siding.

The reactor building is designed as a low pressure low leakage building and provides for the control of any radioactive gases that might be released into the building during a refueling accident or by leakage from the primary containment following a loss-of-coolant accident. During normal operations, the reactor building atmosphere is monitored and exhausted to the environs through the reactor building stack. In the event of an accident, the reactor building isolation system would isolate the reactor building and the reactor building atmosphere and any leakage from the primary containment into the reactor building would be processed through the Standby Gas Treatment System (SGTS) prior to being discharged to the plant stack. The SGTS, which is shared by the three units, consists of two parallel, redundant trains, each with a full capacity exhaust fan, filters and charcoal beds. Each train is designed to treat a gas flow rate of 9,000 cubic feet per minute (cfm). With the reactor building isolated, each train has the necessary capacity to reduce the building pressure and maintain it at a negative pressure of 1/4 inch of water (under neutral wind conditions).

The filters will be tested to demonstrate a removal efficiency for particulates of not less than 99%. The charcoal beds will also be tested to demonstrate that their iodine removal efficiency is not less than 99%. A test program will be conducted before reactor operation and periodically during the life of the plant to demonstrate the design capability and operability of the secondary containment and SGTS. Because the secondary containment and SGTS are shared among the three units, provisions will be made to isolate Units 2 and 3 from Unit 1 while construction proceeds on Units 2 and 3.

Based on our review of this and other similar systems, we conclude that the design and testing of the reactor building and SGTS are acceptable.

6.0 EMERGENCY CORE COOLING SYSTEMS (ECCS)

6.1 General

The emergency core cooling systems consist of two high pressure systems (the high pressure coolant injection systems [HPCI] and the auto-depressurization system [ADS]; and two low pressure systems (the low pressure coolant injection system (LPCI) and the core spray system). The emergency core cooling systems for the Browns Ferry Nuclear Plant are the same systems, except for flow capacity, as the designs previously reviewed and accepted for the Monticello, Quad-Cities, and Vermont Yankee Plants. Certain of the systems are similar in design and equipment to the corresponding systems on Millstone 1 and Dresden 2 and 3.

The emergency core cooling systems are designed as Class I (seismic) systems. All piping within these systems is designed and fabricated to the requirements of thy Power Piping Code, USAS B31.1.0-1967. We have reviewed the design, fabrication and inspection requirements proposed and find them acceptable.

6.2 ECCS Objectives

The ECCS subsystems provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the primary coolant system piping resulting in a loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The ECCS subsystems are provided of such number, diversity, reliability, and redundancy that, even if any active component of the ECCS fails during a loss-of-coolant accident, inadequate cooling of the reactor core will not result.

As with other plants, all systems in the ECCS are initiated by a low water level condition in the pressure vessel. As a backup to the low water level signal, an initiation signal for the HPCI and starting of diesels is provided from a high drywell pressure signal. The initiating signal for starting the core spray pumps and RHR pumps has been modified, however, to require a coincident high drywell pressure and low reactor pressure signal as a backup to low water level. This coincident high drywell pressure and low reactor pressure signal prevents starting the core spray and RHR pumps due to high pressure in the drywell caused by non-accident transients. The reactor vessel pressure set point (500 psig) for starting the core spray and RHR pumps is the same pressure permissive set point for opening of the injection valves for the core spray and low pressure injection coolant. Because the pumps reach speed faster than the valves can be opened, there is no resultant change in the effectiveness of the core cooling systems. We, therefore, conclude that this modification is acceptable.

Each of the ECCS subsystems is designed to function over a specific range of primary coolant piping system break sizes. For small breaks in liquid line, up to about 0.10 ft^2 in area, the high pressure coolant injection (HPCI) subsystem is capable of delivering 5,000 gpm at 1120 psi and supply sufficient coolant to depressurize the vessel and cool the core. Delayed initiation of the core spray subsystem consisting of 2 (100%) loops capable of delivering 6,250 gpm per loop and/or the LPCI mode of the RHRS which can deliver 40,000 gpm would provide long-term core cooling. For breaks between 0.10 ft^2 and 0.2 ft^2 in area in liquid lines, the depressurizing function of the HPCI and the large volume coolant makeup capability of either the core spray subsystem or the LPCI mode of the RHR would act in combination to provide effective core cooling. In the event of a loss-of-coolant accident without high pressure coolant injection capability (i.e., the normal feedwater and HPCI are assumed to be unavailable), the ADS would cause the reactor vessel blowdown to occur in a time interval sufficiently short to permit core spray and/or LPCI mode operation with rapid vessel reflooding before excessive fuel clad heating occurs.

For breaks in liquid lines larger than about 0.2 ft^2, depressurization assistance is not required. The core spray subsystem by itself and in conjunction with the LPCI mode of the RHRS is capable of cooling the core independently of the HPCI or ADS for a range of break areas from approximately 0.2 ft^2 up to and including 4.9 ft^2, the latter corresponding to the double-ended break of the largest primary coolant (recirculation) pipe. Both the LPCI mode of the RHRS or core spray subsystem are designed to respond quickly to the larger break sizes with large volumes of coolant water in flooding and spraying modes respectively.

In the case of steamline breaks within the drywell, the ECCS objectives are satisfied more easily for breaks in steam lines than for breaks in liquid lines because the reactor primary system depressurizes more rapidly with less coolant mass loss for steam breaks than for liquid breaks. For example, the HPCI system is capable of providing short term core cooling for steamline break sizes up to about 1.3 ft^2.

6.3 High Pressure Coolant Injection System (HPCI)

The HPCI system is substantially the same as the system provided on Vermont Yankee, and similar except for sizes and capacities to the systems provided on Monticello, Quad-Cities, and Dresden 2 and 3. The HPCI system includes one steam-turbine-driven pump injecting 5000 gpm of high pressure cooling water through one of the feedwater lines into the reactor vessel. Steam for the turbine is drawn from one of the main steam lines within the drywell and turbine exhaust steam is discharged into the torus water through a submerged pipe. The pump takes suction first from the 135,000 gallon capacity condensate storage water tank with an automatic transfer of suction to the torus water if additional water is required.

6.4 Auto-Depressurization System (ADS)

The ADS system utilizes six of the eleven dual purpose relief and safety valves which are also part of the Pressure Relief System described in Section 4.9. The ADS system is similar to the systems provided on Millstone 1 and Monticello. Actuation of the ADS requires coincident indication of reactor low water and high drywell pressure.

The design includes an interlock to prevent automatic actuation of the automatic pressure relief system unless one of the LPCI or core spray pumps is operating which is consistent with the designs previously approved and satisfies the ACRS concern identified during the construction permit review. We have reviewed the ADS and conclude that it is acceptable.

6.5 Low Pressure Coolant Injection System (LPCIS)

The LPCI mode of the RHR system provides rapid flooding of the reactor vessel in the event of a large break loss-of-coolant accident. Protection provided by the LPCI mode also extends to a small break, in which the feedwater, control rod drive water pumps, RCIC, and HPCI are all unable to maintain the reactor vessel level and the ADS has operated to lower the reactor vessel pressure so that LPCI and the core spray system start to provide core cooling.

The containment spray mode of the RHR system provides spray cooling to the drywell and suppression chamber after the reactor core has been reflooded following a loss-of-coolant accident. The design and equipment for these portions of the RHR system performing these two functions are similar to the sub-systems provided on Quad Cities, Millstone 1, Monticello, and Vermont Yankee. The major equipment of the RHRS consists of four main system pumps and four heat exchangers for long-term core and containment cooling. The equipment is connected by associated valves and piping and the controls and instrumentation are provided for proper system operation. Each RHR pump is rated at a flow of 10,000 gpm at 20 psid.

6.6 Core Spray System

The core spray system provides high volume spray to the reactor core in the event of a large break loss-of-coolant accident. It consists of two independent subsystems drawing water from the suppression chamber, and pumping directly into the reactor vessel and onto the core through the two core spray headers. Each of the core spray pumps are designed to deliver 3125 gpm at 122 psid. System design is based on the assumption that only 1 of the 2 core spray loops are required to deliver core spray flow. The core spray system provided for this plant is similar to the system provided on Millstone 1, Monticello, Quad-Cities, Vermont Yankee, and Dresden 2 and 3.

6.7 Net Positive Suction Head (NPSH) to RHR and Core Spray Pumps

Safety Guide 1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps", requires that these systems not rely on calculated increases in containment pressure. The safety guide was published after the construction permits were issued and the plant design is not consistent with the requirements of the safety guide. TVA's analysis based on conservative design assumptions demonstrates that a positive NPSH margin would be available following a loss-of-coolant accident but would require a containment overpressure. The applicant's analysis shows that a containment overpressure of about three psi is needed for about 15 hours following a loss-of-coolant accident to assure adequate NPSH to the RHR and core spray pumps. A satisfactory containment overpressure margin (three psi) is available even if the containment spray were operating following a design basis loss-of-coolant accident (LOCA). Although the design does not meet the guidelines of the safety guide we have concluded that the applicant has conservatively predicted containment overpressure and that there should be adequate and sufficient NPSH to the emergency core cooling system pumps to ensure their operability in the unlikely event of a LOCA.

6.8 Discussion of ECCS Review

The performance of the ECCS was analyzed using the assumptions of the AEC interim acceptance criteria adopted on June 29, 1971. The analysis applied the AEC assumptions with no deviations. Break sizes from 0.02 ft^2 to 4.9 ft^2 were treated in the analysis. Various single failure assumptions were made to determine the situation that resulted in the maximum fuel clad temperature.

For the LOCA with the largest break size, (the design basis accident), the calculated peak clad temperature is 2090°F, assuming a failure of the LPCI injection valve that renders both LPCIS inoperative. This was the single failure that results in the maximum peak clad temperature. In this case only the four core spray pumps are operable. The corr ponding metal-water reaction was calculated to be less than 0.12%. An assumed failure of one of the four station diesel generators resulted in a calculated peak clad temperature of 1930°F. In this latter analysis, three of the four LPCIS and two of the four core spray pumps would be operable.

Analyses for the entire break spectrum, up to and including a double ended severence of the largest pipe of the reactor coolant system (the DBA) showed a continuous decrease in the peak clad temperature and percentage of metal-water reaction as the break size was decreased from the largest break size to about 0.2 ft^2 below which, the peak clad temperature and percentage metal-water reaction increased, reaching a peak at 0.05 ft^2. For breaks smaller than 0.05 ft^2 the peak clad temperature and percentage metal-water reaction again decreased. These analyses neglect the operation of the HPCIS. Thus, the 0.05 ft^2 break size was found to result in the maximum clad temperature and metal-water reaction for a break in the intermediate break range.

The calculated peak clad temperature for a 0.05 ft^2 break was 1830°F with a corresponding percentage of metal-water reaction of less than 0.1%. The limiting assumption for this break size was the failure of one diesel generator to operate. An assumed failure of the LPCIS injection valve resulted in a lower maximum clad temperature of 1750°F.

6.9 Conclusion

We conclude that the design of the Browns Ferry Nuclear Plant emergency core cooling system is acceptable based on analyses using the evaluation model of Part 2 of the Interim Policy Statement which shows that the consequences of the loss-of-coolant accident are such that (a) the calculated maximum fuel rod cladding temperature does not exceed 2300°F, (b) the amount of fuel rod cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor, (c) the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching, and (d) the core temperature is reduced and decay heat is removed for an extended period of time. Based on the above, we conclude that the emergency core cooling system meets the requirements the AEC interim acceptance criteria and is therefore acceptable.

7.0 INSTRUMENTATION, CONTROL & EMERGENCY AUXILIARY POWER SYSTEMS

7.1 General

Our review encompassed the reactor protection and control systems, the engineered safety feature circuits, and the emergency auxiliary electric power systems. The Commission's General Design Criteria outlined in Appendix A to 10 CFR Part 50, as published in the Federal Register on February 20, 1971, and the Institute of Electrical and Electronic Engineers (IEEE) "Criteria for Protection Systems For Nuclear Power Generating Stations" IEEE Standard 279-1971 (IEEE-279), IEEE "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations" IEEE Standard 308-1971 (IEEE-308), and AEC Safety Guides for Water-Cooled Power Plants served, where applicable, as the bases for evaluating the adequacy of these designs.

7.2 Instrumentation and Control Systems

The review of the reactor protection and control systems and the engineered safety feature circuits was based on a comparison of the designs with those of the Pilgrim Nuclear Power Station which were previously reviewed and found acceptable. Features that differed from the Pilgrim plant were identified and given special consideration during our review of the Browns Ferry Plant. Features for which new information has been received, or which have remained as continuing areas of concern during this and prior reviews of similarly designed plants were also identified and were reviewed. Specifically, the areas identified were:

a. Protection System Generic Items

   (1) Incident Surveillance Instrumentation

   (2) Addition of a High Reactor Vessel Water Level Isolation Signal

   (3) Addition of an Average Power Range Monitor (APRM) Reactor Trip in Startup

   (4) Annunciation of Engineered Safety Feature (ESF) Bypasses

   (5) Standby Gas Treatment System (SGTS) Actuation

   (6) Anticipated Transient Without Scram (ATWS)

   (7) Backup Control Center

   (8) Operational Bypasses

b. Radiation, and Environmental Testing

c. Separation Criteria

7.2.1 Protection Systems Generic Items

(1) Incident and Accident Surveillance Instrumentation

The BWR reactor protection and engineered safety feature instrumentation channels generally use blind sensors and, therefore, do not provide continuous readout in the control room of the parameters being monitored. The neutron monitoring and main steam line radiation monitoring systems are exceptions. The other vital parameters, however, are monitored by instrument channels associated with control systems. As such, these information readout channels are not designed to satisfy protection system criteria and availability and testing requirements are not included in the Technical Specifications.

Information readout channels are required by the operator to assess plant conditions during and subsequent to an anticipated operational occurrence or accident in order that he may determine whether to intervene in the operation of the Automatic Depressurization System (ADS) or to initiate containment spray. The applicant has provided a list of redundant channels that readout and, in some cases are recorded in the control room. This listing is consistent with that of the Pilgrim design except that the applicant has not proposed redundant surveillance instrumentation for monitoring primary containment pressure. We will require that a second drywell monitoring instrument be installed before licensing of each unit.

(2) Reactor High Water Level Isolation Signal

The primary containment isolation system instrumentation for Browns Ferry has been modified to include a reactor high water level signal to trip the main steam line isolation valves for all operating modes except the "Run" mode. The stated objective is to prevent exceeding the design rate of change of vessel temperature resulting from rapid depressurization caused by pressure regulator failures during startup. A high level would result from water level "swell" caused by the rapid depressurization. Depressurization protection is provided in the "Run" mode by the low steam pressure signals that trip the main steam line isolation valves. The applicant has documented that this instrumentation meets the requirements of IEEE-279. We have reviewed this aspect of the design and conclude that the circuit design meets IEEE-279 and is acceptable.

(3) Average Power Range Monitor (APRM) Reactor Trip in Startup Range

The applicant has documented changes necessary to modify the APRM channels to extend their effectiveness into the startup range and to include an APRM trip at 15% power. In previous BWR designs, the APRM channels were effective only in the "Run" mode. We have reviewed this design change and have concluded that it meets the requirements of IEEE-279 and is acceptable.

(4) Annunciation of Engineered Safety Feature Bypass

Our review of the design revealed that annunciation of bypass of engineered safety features resulting from a deliberate operator action was not included. We do not consider administrative controls as an effective and adequate means to identify these bypasses nor do we consider administrative controls to satisfy the requirements of IEEE-279. The applicant has agreed to provide for the capability of initiating control room annunciators whenever operator action results in the loss of an ESF function or a reduction in system redundancy. The details of this design has not as yet been completed. We will review this design during the preparation of the supplemental safety evaluation for Unit 2. We have concluded that the design need not be completed prior to issuance of an operating license for Unit 1.

(5) Standby Gas Treatment System (SGTS)

The SGTS consists of two separate and redundant full capacity filter/absorber/fan units. The major components are shown in Figure 5.3-3 of the FSAR. This system is provided to maintain a small negative pressure (0.25 inches of water) in the reactor building under isolation conditions to minimize ground level release of airborne radioactivity.

Although the system is redundant with respect to filter/absorber/fan units, the applicant's design requires sequential operation of the fans. This is not consistent with our previous requirements for concurrent starting of engineered safety features. We will require the applicant to modify the design of the system to provide for concurrent starting of the fans. Resolution of this item is necessary prior to licensing of Unit 1.

(6) Anticipated Transient Without Scram (ATWS)

As further confirmation of the adequacy of design, we and the ACRS have requested the reactor supplier, General Electric, to study means for preventing common mode failures from negating scram action and design additional features to mitigate the consequences of failures to scram during anticipated transients. GE has submitted the results of these studies in two topical reports, NEDO-10189, "An Analysis of the Functional Common Mode Failures in GE BWR Protection and Control Instrumentation" dated July 1970 (submitted October 26, 1970), and NEDO-10349, "Analysis of Anticipated Transients Without Scram" dated March 1971 (submitted May 4, 1971). These reports are now under review by the regulatory staff and the applicant has agreed to install these systems when our review and the system design is complete. We have concluded that this commitment by the applicant is adequate.

(7) Backup Control Center

Backup control panels are being provided for each unit to permit the control of shutdown functions from outside the control room and to bring the reactor to a cold condition in an orderly fashion in the event habitability of a control room is lost. The backup control panel will serve as an information and transfer center to permit and direct operations of essential shutdown equipment from local backup controls. We reviewed this design to confirm (1) that independence and separation of redundant ESF were mainta[illegible] that failures in the panel will not affect manual or automa[illegible]eration of redundant ESF; and (3) that the transfer switche[illegible]d local manual backup controls are protected against inadvertent operation and their operation is annunciated in the main control room. The applicant has documented satisfactory design bases and our review of the applicant's design shows conformance to these design bases. We conclude that the design is acceptable.

(8) Operational Bypasses

The applicant has proposed that circuitry be included to provide a means for manually bypassing one of the initating signals for the core spray and low pressure coolant injection system (i.e., high drywell pressure coincident with low reactor pressure). The purpose of this bypass is to reduce the possibility of an operational transient resulting in initiation of the core spray and LPCI system. The design of this operational bypass circuitry is not complete. We will review the design for conformance to IEEE-279 and determine its adequacy prior to licensing of Unit 1.

7.2.2 Radiation, and Environmental Testing

(1) Radiation Testing

The applicant has included in its equipment specifications the doses to which items important to safety are expected to sustain without loss of function. The equipment manufacturers are expected to use and be able to substantiate that the materials used are capable of sustaining this environment. The applicant has documented in Amendment 15 that materials with no history of successful radiation experience or testing will not be used. We conclude that these criteria are heavily dependent on a quality control program for which the applicant is required to maintain appropriate records to substantiate suitability of the materials used in these equipments. The applicant has documented that his procedures require the cognizant design engineer to review vendor drawings including lists of materials to determine the suitability of materials used. The applicant will use authoritative published data in making this determination. We conclude that the applicant's plan when suitably implemented is acceptable.

(2) Environmental Testing

In response to our request for test results establishing the suitability of electrical equipment and components within the containment to sustain accident or anticipated operational occurrence environments, the applicant stated that these equipment and components are identical to those used and found acceptable in Millstone 1. The applicant has in addition proposed to add circuitry to alert the operator to high primary containment temperatures. The operator will be instructed by operating procedures to initiate containment spray in order to ensure that primary containment for certain plant conditions does not exceed 280°F for 30 minutes or 35 psig high drywell pressure. The need for this operator action and the additional circuitry is to ensure, with margin, that the environmental capability of the instrumentation in containment is not exceeded. We will require that the circuitry meet the single failure criterion. This item will be resolved prior to licensing of Unit 1 and will be included in the supplemental report for Unit 1.

7.2.3 Separation Criteria

The applicant's separation criteria are incomplete in some areas. One of these areas concerns the separation of redundant devices and the connection of redundant circuits to single devices in control room panels, boards, and racks. Consistent with our position in Pilgrim and Vermont Yankee, we require that redundant protection system circuits not be connected to a single device (switch) and that a minimum separation of 6 inches or physical barrier be provided between such devices. Our review has revealed that the applicant's separation criteria do not commit to a minimum of 6 inch separation or equivalent physical barriers for redundant components in panel and control boards.

We have notified the applicant that this criterion must be met and that we will require that it be demonstrated that all designs are consistent with it prior to issuance of the operating license of Unit 1. Our review has also revealed that the main steam line and the HPCI and RCIC steam supply line redundant high flow sensors are mounted on common racks. We have informed the applicant that these sensors must be separated unless the applicant can demonstrate acceptability on the bases that diverse instrumentation provide equal protection.

Another area where the applicant's separation criteria were incomplete concerned cable routing. We identified the criteria which had been omitted and the applicant has responded by including these criteria with a minimum of exceptions. The exceptions are concerned with the degree of separation (9 vs 12 inches between cable trays). We do not consider this to be sufficiently significant to safety to warrant backfit and have determined that the applicant's design is acceptable.

7.3 Emergency Electrical Power Systems

7.3.1 Offsite Power

Units 1, 2, and 3 of the Browns Ferry Nuclear Plant will be interconnected to the transmission system through 500 kV circuits. Power from each unit generator will be fed via separate circuit containing a step-up transformer to the 500 kV switchyard. The 500 kV switchyard will be arranged in a modified breaker-and-a-half configuration.

Future connections to the 230 kV switchyard via an auto transformer is being planned. Six transmission circuits will emanate from the plant. These circuits are routed on separate rights-of-way.

Offsite power for plant startup, shutdown, and engineered safety features is supplied from a separate 161 kV switchyard. This switchyard is connected to the 161 kV grid by two circuits each of which is mounted on separate towers. While these two circuits share a common right-of-way for a short distance, there is sufficient separation to preclude one tower or line failure from affecting the other. The 161 kV switchyard is arranged in a simple two bus configuration interconnected with a single circuit breaker and motor operated switch to disconnect these buses in the event of circuit or bus faults.

The failure of this circuit breaker could result in the loss of offsite power to the plant. In response to our concern, the applicant has stated in Amendment 14 that his design provides protection against the most probable causes of failure with the following features:

(1) Two trip coils are provided; one coil tripped from normal relays, the other tripped by backup relays.

(2) The trip coils are continuously monitored from the control room.

(3) The circuit breaker is provided with manual mechanical trip device.

(4) Fail-to-trip relaying has been added to trip incoming (supply) breakers at their source if the breaker does not trip.

The applicant has provided, in conjunction with "(4)" above, an analysis which shows that the control room operator can activate the motor operated disconnect switches and isolate the fault in sufficient time to re-energize one bus in the switchyard to ensure that the system meets the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50 as published in the Federal Register on February 20, 1971. On the basis of our review we conclude that the design of the offsite power to the 161 kV switchyard satisfies General Design Criterion 17 in this regard and is acceptable.

Two circuits interconnect the 161 kV switchyard to the plant emergency distribution system. Each circuit is routed on separate towers through a redundant 161/4.16 kV common station service transformer to the 4160 volt distribution system shutdown buses. There is one tower immediately adjacent to the plant whose failure could result in the loss of the redundant 161 kV circuit. The applicant has agreed to increase the separation but the design and the applicant's schedule for installing this modification has not been provided. We will address this item in our supplement to this report.

The offsite power available to the shutdown boards is limited by the size of the circuit breakers. Less than half of the installed cooling equipment can be operated with offsite power sources. This results in an inordinate amount of operator action to provide for safe shutdown of the facility. The applicant has agreed to increase the shutdown capability of the plant with offsite power sources however the designs are not complete. The unacceptable aspects of the design are only related to multiple facility operation. The system is acceptable for Unit 1 operation only. This item will be considered as a condition to the license and will be resolved prior to licensing of Unit 2.

Our review of the offsite power system design reveals that the design pending satisfactory resolution of the above mentioned matters meets the requirements of General Design Criterion 17 and IEEE-308 and is acceptable.

7.3.2 Onsite Power

The emergency standby a-c power system for the plant consists of four diesel generator sets each assigned to power one 4160 volt shutdown board. The engineered safety feature (ESF) and shutdown loads for all three units are distributed among these shutdown boards and attendant distribution systems. The intent of this arrangement is to ensure that any three of the four diesel generator sets or shutdown boards will supply minimum ESF loads in one unit and safe shutdown loads in the remaining two units.

The applicant has attempted to respond to the concerns of the ACRS as expressed in the Committee's letter dated May 15, 1968 issued in connection with its review of the application for a construction permit for Brown's Ferry Unit 3. These concerns were with regard to the improvement of the marginally acceptable onsite power system with respect to capacity of diesel generator sets and the need for paralleling of these generators. The applicant attempted to improve the design by eliminating the need for paralleling the diesel generators. However, these attempted design improvements have resulted in the development of a more complex design that requires extensive interrelationship among the units' control circuits, requires automatic transfer of load groups, results in excessive diesel generator loadings and requires an excessive amount of operator coordination.

The applicant has been advised that the standby a-c power supply should be modified to improve the independence of the redundant power sources by reducing the need for automatic bus transfer features. This matter will be resolved prior to licensing of Unit 1.

Our review of the system revealed that single circuit failures, maintenance operations or testing operations in one unit will affect all or at least half of the ESF in the remaining two units. This is due to the need to shed and lockout non-essential loads in the accident unit and ESF of the non-accident units made necessary because of the limited capacity of the totally shared standby a.c. power supply.

The control circuits which accomplish this shedding and lockout are initiated by the accident signals and effect the block or lockout in the ECCS circuits of each unit. Therefore with regard to this control scheme, the ECCS circuits of each unit are interconnected. This interrelationship is such that the testing of a channel of one unit and another channel in another unit could disable automatic ECCS actuation in all three units. This design interrelationship is not consistent with our requirements for independence in the design of engineered safety feature control circuits. We, therefore, have concluded that the controls need to be modified to provide additional independence prior to issuance of an operating license for Unit 2.

We cannot conclude that the capacity of the onsite a-c power system is adequate to provide safe and orderly shutdown of the plant as required by General Design Criterion 5 of Appendix A to 10 CFR Part 50. The diesel generators do not have the capacity to power a sufficient number of Class I seismically qualified cooling components to allow safe and orderly shutdown of the plant without exceeding the guidelines of Safety Guide 9, "Selection of Diesel Generator Set Capacity For Standby Power Supplies." Further, the associated electrical distribution design of the onsite a-c power system is extensively shared among the three units which results in a complex design requiring extensive electrical interlocks and an excessive amount of operator control. We have concluded that the design of the onsite a-c power system although acceptable for operation of Unit 1 is unacceptable for multiple unit operation.

The standby d-c system consists of three 250 voltage batteries each housed in an individual Class I room which is separately ventilated. Our review reveals that this system is dependent upon automatic transfer between batteries to meet the single failure criterion. We have required that the applicant modify the 250 volt d-c system to provide greater independence between redundant sources by eliminating automatic bus transfers. This would result in an additional battery. Resolution of this matter is necessary prior to licensing of Unit 1.

7.3.3 Conclusion

Our conclusions are separately grouped below regarding system improvements needed to make the plant acceptable for single unit operation (Unit 1) and additional system improvements needed for multiunit operation.

(1) Operation of Unit 1 only

a. The two transmission lines that supply offsite power to the plant emergency distribution system are routed so that at one point close to the buildings, the location of the support towers could permit one to fall on the other. This separation should be increased or other provisions made to make the design acceptable with the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

b. An additional battery should be added with associated changes to the d-c system to eliminate the need for automatic bus transfer of d-c loads as expressed in Safety Guide 6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems."

c. Automatic bus transfer of a-c loads should be limited to only the low pressure coolant injection system valves to make the design more consistent with the guidelines of Safety Guide 6.

(2) Multi-unit Operation

a. Our review of the onsite and offsite systems for multi-unit operation indicates that modifications in addition to the above will be necessary to make the system acceptable. Our concerns are related to the capability of the system to power sufficient cooling equipment to allow safe and orderly shutdown of the three units and the extensive interrelationship of the systems among the three units.

The onsite or offsite power systems do not have sufficient capability to power safety grade cooling equipment (i.e., equipment designed to Class I seismic criteria, controlled by the quality assurance program and designed to IEEE-279 criteria) without requiring a pressurized primary containment or prompt and fast blowdown of each reactor. Further, an unrealistic amount of operator coordination is necessary to preclude overloading of the onsite or offsite power sources. We do not believe that the proposed operation of these systems meets the intent of General Design Criteria 5 with respect to sharing of systems and components.

b. In addition, the extensive sharing of the power systems could result in possible anticipated operational occurrences, single failures, maintenance operations or testing operations of one unit affecting the operations of all units. This is exemplified by the electrical control system of the engineered safety features. We recognized during the review that as a consequence of sharing, expected operational transients of one unit could prevent activation of the emergency core cooling systems of another unit which might require emergency core cooling. This kind of interrelationship is not consistent with the concept of independence in the design of emergency power and control systems.

At a meeting held on June 7, 1972, the applicant indicated that system modifications are being planned with respect to separating the transmission lines and minimizing the need for electrical load transfers. These changes, when acceptably implemented, would appear to make the system acceptable for Unit 1 operation only. However, we need a detailed description of the modifications to review their adequacy to meet our concerns and a schedule for their subsequent installation.

With respect to our concerns related to multi-unit operation, the applicant indicated that additional system modifications are being considered, e.g., adding a fifth diesel-generator, improving the offsite power capability and improving the performance capability of the high pressure injection system and reactor core isolation cooling system. These modifications, as we understand them, would improve the shutdown capability for the three units but do not appear to address all our concerns outlined in items (2)a and (2)b above. We will require the applicant to give further consideration to meeting our concerns related to reducing the extent of sharing the power sources and significantly reducing the amount of operator coordination now required. These items which are related to multi-unit operation will be resolved prior to the licensing of Unit 2.

8.0 AUXILIARY SYSTEMS

8.1 General

The auxiliary systems are described in Section 10 of the FSAR. These process systems normally provide plant services auxiliary to the production of power. In the course of our review, we have directed our attention to the safety related objectives of the respective systems and the manner in which these objectives are achieved. We have reviewed the safety related auxiliary systems for redundancy, independence, physical separation, and sharing among units and for those criteria that establish the quality of the systems.

The latter review considered the appropriateness of the seismic design classification, and the use of suitable codes, standards and speciifcations for the design, fabrication and inspection of the piping and other components within each system. The safety-related items that received special attention in the course of our review are discussed in the following sections.

8.2 Radioactive Waste Systems

8.2.1 General

The original radioactive waste systems were designed to comply with the AEC regulations (10 CFR Part 20) in effect at the time the construction permit was issued. During our review of their application for an operating license, TVA modified the gaseous and liquid radioactive waste treatment systems so that the design and operation will comply with the requirements of 10 CFR Parts 20 and 50 that releases of radioactivity be reduced to the lowest practicable level and that all installed equipment be used to achieve these release levels. This entailed design changes which included the addition of a 30 gpm evaporator for liquid sources of activity and for each unit, the addition of 6 ambient temperature charcoal tanks (containing 18 tons of charcoal) to reduce the gaseous activity of Xe and Kr released to the environment. The capacity of the system for treating liquids from each source is considerably greater than that provided in the radioactive waste treatment systems of earlier boiling water reactors.

The applicant estimates that the annual total quantity of radioactive material except tritium to be released from the plant in liquid effluents is less than 5 curies. Our review of the revised design of the liquid radwaste treatment system indicates that these estimates can be achieved and that with proper operation of the system releases will be as low as practicable and acceptable.

8.2.2 Liquid Radwaste System

The liquid radioactive waste system which is common to the three units collects, processes, stores and disposes of all radioactive liquid wastes. The present system components consist of storage tanks, demineralizers, filters and evaporators similar to those used on other boiling water reactor facilities. The liquid radwaste system is divided into four main subsystems, i.e., high purity, low purity, chemical wastes and detergent wastes, so that wastes from various sources can be kept segregated by chemical purity for separate processing. The applicant has provided cross connections between sub-systems to provide additional flexibility for processing the wastes by alternate methods.

High purity (low conductivity) liquid wastes are primarily collected from equipment drain sumps. Liquids from these sources are processed by filtration and ion exchange through the waste filter and demineralizer and then transferred to the condensate storage tank for reuse as makeup water if the liquid meets the conductivity or radioactivity requirements. In the event additional reprocessing is required, liquids may be recycled through the demineralizer train or evaporator. Floor drain and chemical wastes will be processed through the floor drain filter and sent to the evaporator feed tank. Evaporator distillate will be sent to the waste sample tank if it is to be recycled to condensate storage or to the floor drain sample tank if it is to be released to the discharge canal.

The low activity detergent wastes will be collected in the laundry drain tanks. These are processed through the laundry drain filter, monitored and released at a controlled rate to the circulating water discharge canal. Operation of the radwaste system is on a batch basis with a manual start and an automatic stop if the radwaste monitors detect a high radiation signal. Liquid batches are held in a sample monitor tank for sampling and analyzing before discharge to the environs.

Liquid radioactive waste effluents are diluted with condenser cooling water (average flow rate for 3 units of 1,800,000 gpm available 80% of the time) in the discharge canal prior to entering the Tennessee River. The applicant expects that the releases of liquid radwaste except tritium from the system will be less than 5 Ci/yr.

The applicant estimates that 51 Ci/yr of tritium will be released in liquid effluent from all three units. The concentration in the discharge canal is expected to be an insignificant fraction (6.0×10^{-6}) of 10 CFR Part 20 limits.

Based on the performance of similar operating plants, we conclude that the estimates of activity that will be released from the liquid radwaste system can be met and therefore, we conclude it meets the as low as practicable regulations.

8.2.3 Gaseous Radwaste System

During the operation of the plant, radioactive materials released to the atmosphere in gaseous effluents will include fission product noble gases (krypton and xenon) and halogens (mostly iodine); activated corrosion products. Fission products are released to the coolant and carried to the turbines by the steam.

The major source of gaseous waste activity during normal station operation will be the offgases from the main steam air ejectors. Other sources of gaseous waste include purging of the drywell and suppression chamber, offgases from the mechanical vacuum pump and ventilation air released from the radwaste, reactor and turbine building exhaust systems.

A decay system has been incorporated into the revised Browns Ferry gas treatment system in addition to the normal 30-minute holdup system.

The original condenser air-ejector offgas system consisted of a 30-minute holdup pipe, HEPA filters and release from a 600-foot high stack, which is roughly twice the height of any previously constructed boiling water reactor stack.

The modifications for each unit will consist of the addition of hydrogen-oxygen recombiners between the air ejector and the holdup pipes, and the addition of 6 charcoal beds (18 tons of charcoal) in series between the holdup pipe and the HEPA filter.

Gas from the redundant recombiner systems will flow through the holdup pipe, then will pass through a cooler-condenser, moisture separator, the reheater and the prefilter. At this stage, the gas will be about 74°F and 1.4 psig with a dew point of about 45°F. The gas will then flow through the six charcoal beds and a HEPA filter prior to release from the 600-foot stack.

The charcoal delay system will provide additional delay times of 3.75 hours and 2.18 days for kryptons and xenons respectively. With the original gaseous radwaste treatment system, the annual average release estimate was 1.11 Ci/sec from the common stack for 3 units. The modified radwaste system will reduce this to about 0.039 Ci/sec for the three units. With the modified system, the annual average whole body dose at the maximum dose point on the site boundary from the stack (1600 m ESW) for this release rate will be about 3.0 mRem/yr for all three units. Essentially all of the radioactive iodines which may be present in the offgases from the condensers will be removed by the charcoal beds.

Releases of activity from the mechanical vacuum pump, used to supplement the main condenser gas removal system during start up is estimated by the applicant to be 0.07 mRem/year, based on 40 hours/year pump operation. The whole body dose from gland seal leakage is estimated to be 0.06 mRem/year. Iodine releases from the building vent result in doses of about 10^{-3} of 10 CFR Part 20 limits at the site boundary. Our independent calculations confirm these values. Our review of the revised design of the gaseous radwaste treatment system indicates that these estimates can be achieved and that with proper operation of the system, releases will be acceptably low. The applicant has agreed to complete installation of this system for Unit 1 by September 1973.

8.2.4 Solid Radwaste System

The solid radwaste system is housed in the radwaste building and is designed for processing wet waste from water treatment and cleanup systems and radwaste processing, liquid concentrate from the waste evaporator, and dry wastes such as filters, rags, clothing, and equipment parts. Radioactive materials in these solid forms will be properly protected and packaged for shipment to an authorized disposal site. The solid radwaste system is similar to those used satisfactorily in other BWR facilities. We have concluded that the concept of design and system operation is acceptable.

8.3 Fuel Handling and Storage

Fuel handling and storage facilities common to the three units are provided for storage and transfer of new and spent fuel. New fuel is stored dry in racks which are spaced to preclude attaining criticality. Spent fuel is stored underwater in the spent fuel storage pool which is located adjacent to the reactor. During refueling, the drywell and reactor vessel heads are removed and the cavity over the reactor is filled with water. Spent fuel is then transferred underwater to the fuel pool. Subsequently, spent fuel is transferred to a fuel shipping cask which is submerged in the fuel pool. After loading, the cask is removed from the pool and shipped to a fuel reprocessing plant. As with previous BWR designs, we have considered the capability of the fuel pool to withstand an inadvertent dropping of the fuel shipping cask into the pool without causing pool damage that might result in a sudden loss of water. Failure of the reactor building crane or handling slings has been precluded by the design of the crane with redundant components such that failure of any single component would not result in dropping of the cask. This crane design modification was made because the dropped cask impact could result in a loss of the pool water. The applicant has described these provisions in Amendment 24. We conclude that the method of preventing a sudden loss of fuel pool water due to a fuel handling accident is acceptable.

8.4 Control Room Ventilation Systems

The applicant proposes to meet the 30 rem accidental exposure thyroid dose limit of General Design Criterion No. 19, Control Room, of Appendix A of 10 CFR Part 50, by using automatic isolation of the control room roof intake vent along with self contained breathing devices to protect the control room operators during the course of the design basis accidents. Consequently, the applicant does not propose to utilize charcoal filters in the air intake system for the control rooms.

We believe that charcoal filters must be added to the control rooms' air intake system to supply a source of filtered air to the control rooms and to maintain the control rooms at a positive pressure with respect to the outside atmosphere (i.e., prevent inleakage through doors, line and cable penetrations and the control room isolation damper). We are however, continuing discussions with the applicant and will resolve this matter prior to licensing of Unit 1.

8.5 Emergency Equipment Cooling Water System

The emergency equipment cooling water (EECW) system is a common system to the three units. The system provides cooling water to the residual heat removal heat exchanger, diesel-generator, compartments housing ECCS equipment and the reactor building closed cooling water system. The EECW system is designed to Class I seismic criteria and has adequate redundancy to provide safe shutdown of Unit 1 in the event of single active or passive failures. However, the capability of the EECW system to provide safe shutdown capability for the three units simultaneously remains to be resolved in our discussions related to improvements in the emergency power system. Additional power sources would require modifications to the system.

We have reviewed the EECW system for Unit 1 operation and conclude that there is adequate redundancy in passive and active components. Two redundant supply headers -e provided to supply cooling water to essential equipment and six of eight pumps will provide adequate cooling for safe shutdown.

9.0 ACCIDENT ANALYSIS

9.1 General

We have evaluated the applicant's analyses of various anticipated operating transients. The events that characterize abnormal operating transients have been described in Section 14 of the FSAR and include such events as process system control malfunctions, inadvertent control rod withdrawal, turbine trip, and variations in operating parameters. We have reviewed the results of the applicant's analyses of these events and conclude that the design of the facility, including the protection and control systems, is such that the occurrence of such transients would not result in damage either to the fuel or to the primary coolant boundary. Consequently, the occurrence of these abnormal transients would not lead to a significant release of fission products to the environs.

We also have evaluated a broad spectrum of accidents that might result from postulated failures of equipment, or maloperation. We have selected four highly unlikely accidents (design basis accidents) that are representative of the spectrum of types and physical locations of postulated causes and that involve the various engineered safety feature systems provided. The calculated potential consequences of the design basis accidents exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to assess the adequacy of the engineered safety features to control and minimize the possible escape of fission products from the facility. The design basis accidents analyzed were: (1) loss-of-coolant accidents, (2) refueling, (3) control-rod-drop, and (4) steam-line-break.

Our evaluation of these accidents shows that the calculated doses resulting from these postulated accidents are well within the 10 CFR Part 100 guideline values. The results of our analyses, and the analytical method and assumptions used in each analysis are described in the following sections.

9.2 Loss-of-Coolant Accident

In calculating the potential consequences of the postulated loss-of-coolant accident, to provide a conservative assessment we have arbitrarily assumed that in spite of the operation of the emergency core cooling system, large amounts of fission products would be released from the reactor fuel. The fractions of the total core fission product inventory we assumed to be released from the core are those given in AEC Safety Guide 3, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, which was published on November 2, 1970, i.e., 100% of the noble gases, and 50% of the halogens. In addition, 50% of the halogens released from the core are assumed to plate out onto internal surfaces of the containment building or onto internal components. Of the remaining 25% of halogens assumed available for leakage, we used Safety Guide 3 assumptions of 87%, 8%, and 5% for the elemental, organic, and particulate forms respectively. The primary containment was assumed to leak at the Technical Specification limit rate of 2.0 weight percent of the containment volume per day at accident conditions for the duration of the accident (i.e., 30 days) without consideration of the mitigating effects of decreasing pressure during the post-accident interval.

We have assumed a 90% halogen removal efficiency for the elemental and particulate forms of iodine, and 70% for the organic forms of iodine in the HEPA filters and charcoal absorbers of the standby gas treatment (SGTS) in the secondary containment building. In our analysis, we adopted the conservative assumption that leakage from the drywell goes directly to the standby gas treatment system without mixing in the reactor building and then passes through the SGTS to the environment via the 183 meter stack.

We have calculated two hour nearest site boundary (1465m) doses of 42 Rem and 2.8 Rem to the thyroid and whole body respectively.

This is based on the 183 meter stack height and the use of onsite meteorological measurements.

In calculating the course of the accident doses at the low population zone radius of 2 miles, we have again used the actual stack height since the terrain elevation at that distance is approximately equal to the elevation at the base of the stack.

The diffusion factors ($^X/Q$ values) used in our calculations are based on the results of the onsite meteorological program which is discussed in the section on meteorology. The calculated 30 day thyroid and whole body doses based on this data are 27 Rem and less than one Rem respectively.

9.3 Refueling Accident

In evaluating the postulated refueling accident, we assumed that during fuel handling operations, a fuel bundle falls with sufficient force to damage (perforate the cladding) 111 fuel rods. We also assumed that 10% of the noble gases and 10% of the halogens from the damaged rods are released to the refueling pool water. Ninety-nine percent of the halogens released to the refueling pool water from the perforated fuel rods are assumed to remain in the refueling pool water. Halogens released from the pool water are assumed to be 25% organic and 75% elemental. The airborne fission products within the building are assumed to pass through the standby gas treatment system (with a charcoal adsorber iodine removal efficiency of 90% for elemental forms and 70% for organic forms) and be discharged from the 183 meter stack over a two-hour period. It is assumed that the accident occurs 24 hours after shutdown. The meteorological conditions assumed are the same as described for the 0-2 hour period following a loss-of-coolant accident.

The calculated radiation dose for exposure for two-hours, the assumed duration of the accident, is less than one Rem to both the thyroid and whole body.

9.4 Control Rod Drop Accident

For the postulated design basis control rod drop accident, it is assumed that a bottom entry control rod has been fully inserted and has stuck in this position, the drive becomes uncoupled and withdrawn from the rod. Subsequently, it is assumed that the rod falls out of the core inserting an amount of reactivity corresponding to the worth of the rod.

The reactor is designed to reduce the probability of this accident and engineered safety features are provided to limit the consequences of the accident. For example, the control rod worth minimizer is designed to limit the reactivity worth of any control rod during the startup phase of reactor operation. The control rod velocity limiter will limit the velocity during free fall to less than five feet per second. The steam line radiation monitor will detect excessive radioactivity and isolate the main turbine and condenser by closing isolation valves in the condenser mechanical vacuum pump system before the radioactive steam can travel from the detector to these isolation valves. Because of the operation of these engineered safety features, the fission products that escape to the environment would be only those which leak from the isolated turbine and condenser.

In evaluating the radiological consequences of this accident, we have made assumptions based upon the applicant's analytical model as presented in the Final Safety Analysis Report. As discussed in the subsequent paragraphs, the analyses techniques for this particular accident are being revised by General Electric and, depending on the results of these analyses, we may require modifications, in addition to those presently provided, to mitigate the potential consequences.

From the standpoint of radiological consequences, when the reactor is in the hot standby condition at zero power is the worst situation at which a rod drop accident could occur because a high energy release is calculated for this condition and because a path for the unfiltered release of fission products could exist through the mechanical vacuum pump. However, to mitigate the consequences, the main steam line radiation monitoring system, upon detection of high activity in the steam line, would provide signals to circuits that close an isolation valve in the suction of the mechanical vacuum pump and also electrically de-energize the pump. This isolation is designed to occur before the radioactivity reaches the vacuum pump.

For this accident, the most reactive control rod assembly was assumed to drop out of the core 30 minutes after shutdown, causing 330 fuel rods to exceed a calculated energy input of 170 cals/gm. These rods were assumed to perforate, releasing 100% of the contained noble gases and 50% of the contained halogens to the reactor coolant system. Of the halogens released from the affected rods, 90% are assumed to be retained in the primary system and one-half of the remaining halogens are assumed to be removed by plate-out. All of the noble gases and 2.5% of the halogens are assumed to be released from the primary system through the condenser vacuum pump system to the atmosphere. A conservative ground level release from the turbine building was assumed. A wake factor of 0.5, a turbine building area of $2400m^2$, and LOCA meteorology are used for diffusion calculations.

Exposure doses calculated for the whole body and for the thyroid at the Exclusion Area Boundary are less than one Rem and 3.6 Rem, respectively for the assumed two hours exposure, and at the Low Population Zone Boundary are less than one Rem and 5.9 Rem for 24 hours exposure assumed as the duration of the accident. The exposure doses for this accident are well within 10 CFR Part 100 guidelines.

The Atomic Energy Commission has for some time utilized Brookhaven National Laboratory (BNL) as its consultant as part of the regulatory assistance program. For the past few months, personnel at BNL have been performing independent calculations of boiling water reactor control rod worths and potential consequences of a design basis control rod drop accident. As a consequence of the work performed to date at BNL,* it appears that the model used by General Electric to evaluate the design basis control rod drop accident should be revised. Specifically, the assumed rate of negative reactivity insertion from control rod scram is not suitably conservative since it uses insertion characteristics now considered to be not readily attainable in large boiling water reactors. In addition, the actual reactivity insertion rates are not linear as assumed.

The General Electric Company has now revised the analysis of the effects of a control rod drop accident and has submitted a topical report** to the regulatory staff. The analysis presented in the report applies to those reactor plants using control curtains in the core for initial reactivity control. We expect a supplementary report soon from GE regarding a similar analysis for plants using gadolinia poison in the fuel. The regulatory staff with the assistance of BNL is currently evaluating the adequacy of the revised model and the resultant consequences of this postulated accident. Included in the revised analyses are, among other features, a change in the method for modeling the rate of negative reactivity insertion

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* BNL 16717-RP1021, "Rod Drop and Scram in Boiling Water Reactors," dated April 1972

** NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," dated March 1972.

from a control rod scram. A description of the revised analyses and the results of the new analyses are expected to be available within the next few months. Upon receipt of this information, the staff will evaluate the adequacy of the revised model and the resultant consequences of the postulated accident situation. If the consequences of any of the analyzed transients exceed a peak calculated enthalpy of 280 calories per gram, or the radiological consequences approach the guideline values of 10 CFR Part 100, we will require modifications to limit the consequences within acceptable values.

In the interim period we will require restrictions on control rod worths described in Section 3.4 herein and included as limiting condition of operation (LCO) in the Technical Specifications based on our current evaluation of the revised rod drop analysis in the GE Topical Report NEDO-10527, to avoid unsafe fuel damage or radiological consequences if a rod drop accident should occur.

9.5 Main Steamline Break Accident

The break of a main steamline outside of the drywell represents a potential escape route for reactor coolant from the vessel to the atmosphere without passage through the standby gas treatment system.

This escape route would exist only for the few seconds required for the isolation control instrumentation to sense the break and close the main steamline isolation valves.

The occurrence of a main steamline break outside the containment would be sensed by either high steam flow or increased temperature in the steamline tunnel area. The steamline isolation valves would start to close within 0.5 seconds after the steamline break is sensed. The applicant has provided analyses to show that fuel rod cladding perforations would not occur as a result of a steamline break if the isolation valve closure times, including instrument delay, are less than 10.5 seconds. To provide additional margin to assure that cladding perforations will not occur during the transient before the valves are closed and to reduce the amount of radioactivity released, the Technical Specifications require a valve closure time of not greater than 5 seconds.

The meteorological considerations assumed for this accident are those given in AEC Safety Guide No. 5, "Assumptions Used For Evaluating The Potential Radiological Consequences of a Steam Line Break Accident For Boiling Water Reactors." In our analyses, the mass of primary coolant released (185,000 lbs in 10.5 seconds) is assumed to have a total iodi e fission product specific activity of 20 microcuries per cubic centimeter, which is the maximum coolant activity permitted by the Technical Specifications.

The calculated thyroid dose resulting from exposure for two hours at the exclusion distance of 1465 meters is 38 Rem. The whole body dose from noble gases would be negligible since noble gases are continuously removed from the coolant by the condenser air ejector.

9.6 Conclusion

On the basis of our evaluation, the calculated potential radiological doses that might result from any of the design basis accidents are well within the guidelines given in 10 CFR Part 100.

10.0 DESIGN BASES FOR STRUCTURES AND EQUIPMENT

The applicant has classified the plant structures and equipment into two categories dependent upon their relationship to safety.

Structures (e.g., primary containment vessel, reactor building and plant stack) and equipment (e.g., reactor pressure vessel and internals, primary coolant system and the emergency core cooling system) whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the facility and the removal of decay heat have been classified as Class I. Class II structures and equipment are defined as those which are necessary for station operation but are not essential to a safe shutdown. We have reviewed the applicant's classification of structures and equipment and we conclude that they have been appropriately classified.

The Class I reactor building, concrete chimney and pumping station structures are founded on mats on bedrock. The Class I diesel generator building is founded on about 3 feet of earth backfill on top of 32 feet of crushed stone backfill. The Class I equipment access lock rests on a row of steel bearing piles to rock under each vertical wall and another row at the mid-point of the ground level slab. The Class I standby gas treatment structure bears on about 10 feet of earth backfill over the same crushed stone backfill as for the diesel generator building. The Class II turbine building is supported on steel H-piles to bedrock. As a result of some weathered rock in the foundation material the Unit 1 reactor building was underpinned, while under the Unit 2 and 3 portions fill concrete was placed. Seam grouting was utilized under the turbine building for the bearing pile clusters. The foundations as designed are acceptable, and it can be concluded that their construction was in accordance with the design criteria.

Class I structures, as defined in Appendix C of the FSAR and listed in Section 12 of the FSAR are designed for normal dead and live loads, 100 mph wind, 300 mph tornado wind and 3 psi pressure drop, operating and design basis earthquakes of 0.1 g and 0.2 g maximum ground accelerations respectively. Soil, hydrostatic and missile loads have also been included.

For tornado design, the upper 320 feet of the chimney is designed to fail well before the lower 280 feet reaches its ultimate load capacity. Therefore, the chimney fall line under tornado winds does not reach any Class I structures, the nearest of which is 365 feet from the chimney. Pieces of concrete and an aircraft warning beacon are considered as potential missiles originating from the chimney in the spectrum of missiles for which Class I structures are analyzed.

The Radwaste Building, although not defined as a Class I (Seismic) structure, meets Class I (Seismic) structural design criteria under tornado or earthquake loading, and it can be concluded that it will satisfactorily perform its function under these loads.

The reactor vessel concrete support pedestal is capable of withstanding, within acceptable stress limits, either design basis accident, earthquake (OBE or DBE) or design basis accident combined with earthquake (OBE or DBE).

The applicant has described the consequences of a short duration peak temperature on the drywell steel shell of 340°F. No buckling is anticipated, and stresses remain within the allowable stress intensity value of the ASME code. Direct jet impingement on the drywell plate has been analyzed by the applicant and a determination made that containment integrity would not be endangered. We have reviewed the applicant's findings with respect to the effects on the containment of local or general high temperatures and find them acceptable.

The design loads and their application to Class I structures are acceptable. Missile impact, structural interactions and design procedures have been reviewed and found to be acceptable.

Splicing of reinforcing bars by the Cadweld process, where used, was carried out with an acceptable testing program to ensure quality control.

The design strength of the concrete is generally 3000 psi with some 4000 psi. The reinforcing used conforms to ASTM A432 and nas a yield point of 60,000 psi.

No unresolved construction items are under review, and the materials used in construction are considered to be acceptable.

The secondary containment building will be leak tested after construction to verify a minimum of 0.25 inch water gauge negative pressure at calm wind conditions at a flow rate of 9000 cfm (1.5 secondary containment volumes per day). Surveillance will be carried out as charted in Table 5.3-1 of the FSAR. Penetration testability has been reviewed and found to be acceptable.

The secondary containment leak testing and surveillance criteria are similar to past applications and are acceptable.

Amendment 24 presented structural revisions to the intake and discharge structures which will be completed prior to Unit 1 operation. These are structural modifications for the future use of cooling towers, in place of once-through river cooling water. The only changes reviewed are those which will be carried out prior to construction, in order not to interfere with plant operation at a later date if cooling towers are to be used.

The Class I intake structure will have a cellular cofferdam installed, with an opening left in the center for continued flow of river water, but which can be closed off if cooling towers are to be installed. The design criteria have been reviewed and are acceptable.

The discharge structure (not Class I) will have future connection openings installed in the conduits, and gates placed and provided for in order to make it possible to reroute the discharge water when future cooling tower connections are made.

In evaluating the structural design of the Class I structures, systems, and equipment, our seismic design consultant (Nathan M. Newmark Consulting Engineering Services), whose report is enclosed as Appendix C, concluded that the design incorporates an acceptable range of margins of safety for the hazards considered and that the design could be considered adequate in terms of provision for safe shutdown for the design basis earthquake and capable otherwise of withstanding the effects of an operating basis earthquake.

Class I components for the mechanical fluid systems exclusive of the reactor coolant pressure boundary have been designed, fabricated and inspected in accordance with the following codes:

(a) Piping conforms to the requirements of the USAS B31.1.0-1967 Code for Pressure Piping.

(b) Pumps conform to the Class C requirements of Section III of the ASME Boiler and Pressure Vessel Code.

(c) Valves conform to the B31.1.0-1967 Code for Pressure Piping.

We find the codes and standards specified for Category I mechanical fluid systems provide an acceptable quality level and are consistent with recently reviewed plants of this type.

All Class I systems, components, and equipment outside of the reactor coolant pressure boundary were designed to sustain the Operational Basis Earthquake within the appropriate code allowable stress limits and the Design Basis Earthquake within stress limits which are comparable to those associated with the emergency operating condition category of current component codes. We consider that these stress criteria provide an adequate margin of safety for Category I systems and components which may be subjected to seismic loadings.

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods are used for the analysis of all Class I structures, systems and components. Governing response parameters have been combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. The absolute sum of responses is used for closely spaced frequencies. Concurrently applied horizontal and vertical floor spectra inputs used for design and test verification of structures, systems and components were generated by semi-empirical methods and confirmed by the normal mode-time history method. Vertical ground accelerations were assumed to be 2/3 of the horizontal ground accelerations for items rigidly attached to structures. Constant vertical load factors were employed only where analysis show sufficient vertical rigidity to preclude significant vertical amplifications in the seismic system being analyzed. We and our seismic consultant conclude that the seismic-system dynamic methods and procedures proposed by the applicant provide an acceptable basis for the seismic design.

The basic seismic instrumentation program proposed for this facility corresponds to the recommendation of Safety Guide 12 with respect to 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.

The proposed program does not call for the provision of supporting instrumentation such as peak recording accelerographs and peak deflection recorders which would be of great assistance in determining the validity of the design analyses relied upon to assess the effects of a seismic event on equipment and components of the reactor systems. We recommend in accordance with the regulatory position taken in Safety Guide 12, that such instrumentation be installed to determine the accumulative damage fraction even though the applicant will perform a controlled shutdown for earthquakes greater than the OBE followed by a thorough investigation and extensive tests of all safety related equipment to insure all component damage has been located and repaired.

11.0 CONDUCT OF OPERATIONS

11.1 Station Organization and Staff Qualification

Approximately 188 full-time employees will be assigned to the station during commercial operation of all three units under the supervision of the Plant Superintendent who has onsite responsibility for the safe operation of the facility. The Plant Superintendent reports to the Nuclear Operations Coordinator who reports to the Director of the Division of Power Production.

Five technical groups (Operations, Results, Electrical Maintenance, Mechanic Maintenance, and Health Physics) report through the Assistant Plant Superintendent to the Plant Superintendent. Minimum shift staffing will include one Senior Licensed Operator and two Licensed Operators for single unit operation, two Senior Licensed Operators and three Licensed Operators for two unit operation and three Senior Licensed Operators and four Licensed Operators for three unit operation. Minimum plant staff for conduct of operations is specified in Section 6 of the Technical Specifications.

The qualifications of the management and operating staff will meet the minimum acceptable levels as described in Safety Guide 8 and ANSI N18.1, Selection and Training of Nuclear Power Plant Personnel, dated March 8, 1971. The Plant Superintendent or the Assistant Plant Superintendent and the Operations Supervisor or the Assistant Operations Supervisor will hold Senior Reactor Operators Licenses.

Preoperational tests, initial fuel loading and startup of the facility is the responsibility of the Tennessee Valley Authority with General Electric providing technical direction and guidance including, as a minimum: a Site Operations Manager, Operations Superintendent and Shift Supervisors; in addition to the Brown's Ferry operating staff.

We have concluded that the organization structure and the qualifications of the staff for the Brown's Ferry Units 1, 2, and 3 facility is satisfactory to provide an operations staff and engineering support capable of operating the proposed facility during normal and abnormal conditions.

11.2 Emergency Planning

The applicant has established a formal organization for coping with emergencies that includes written agreements, liaison and communication with appropriate local, State, and Federal agencies that have responsibilities for coping with emergencies. The applicant has defined a spectrum of accidents including criteria for determining when protective measures should be considered. Formal arrangements have been made by the applicant to provide for extensive medical support in the event of a radiological or other emergency. Provisions for periodic training, and drills have been included in the emergency plan. Revisions to dose rate curves and estimated total thyroid dose curves are in process.

We have reviewed the details of the radiological emergency plan and conclude that it is in substantial conformance with Appendix E to 10 CFR Part 50, Emergency Planning Requirements. We conclude that the arrangements made by the applicant to cope with the possible consequences of accidents at the site are both reasonable and prudent, and that there is adequate assurance that such arrangements will be implemented in the unlikely event that they are needed.

11.3 Review and Audit

The review and audit functions for Browns Ferry Units 1, 2 and 3 will be conducted by a Plant Operations Review Committee (PORC), a Safety Review Board (SRB), and a Nuclear Engineering Branch. The PORC is a plant level committee and acts in an advisory capacity to the Plant Superintendent. The Nuclear Engineering Branch of the Division of Power Resource Planning in providing technical support conducts an audit of the operation of each nuclear plant at least once each year. The SRB, the majority of the members of which are independent of the Division of Power Production, review the minutes of the meetings of the PORC and the reports of the annual audits performed by the Nuclear Engineering Branch. The details of committee membership, quorum, meeting frequency, responsibilities and authorities of these committees are delineated in the Technical Specifications. We conclude that the review and audit structure proposed by the applicant is acceptable.

11.4 Plant Procedures

Safety-related plant operations will be conducted in accordance with detailed written procedures. These procedures will be reviewed by the Plant Operations Review Committee and approved for use by the Plant Superintendent.

We conclude that the provisions for the use of written procedures and their review and approval prior to use is satisfactory.

11.5 Industrial Security

The plant site and its structures will be protected by security fencing, lighting, physical barriers, and guard force. A system of personnel identification and access control to various areas within the plant site boundary (including areas designated for visitor use) has been established. The applicant has also established administrative arrangements within its security program to effect liaison with law enforcement agencies, in addition to the availability of additional TVA public safety officers in the event of a security emergency.

We have reviewed the details of the applicant's Industrial Security Program and have concluded that a suitable program for protecting against industrial sabotage as described in AEC Safety Guide No. 17, Protection Against Industrial Sabotage has been provided.

11.6 Test and Startup Program

The test and startup program implementation is the responsibility of the Tennessee Valley Authority. The program consists of preoperational tests, fuel loading and shutdown power level tests, initial heatup to rated temperature and pressure, power testing and warranty testing. The program has been prepared by TVA and the General Electric Company and approved by the station staff and by TVA's Division of Engineering Design and Division of Construction. The objective of the program is to confirm that system components are properly installed, calibrated, and adjusted; that the systems are operational and fulfill design criteria; that the nuclear characteristics of the reactor are verified; and that the station can operate safely and reliably in conformity with design values.

TVA has the responsibility for planning, scheduling, carrying out, and documenting the plant startup program. The startup program is intended to conform to the requirements of the operational quality assurance program as described in Appendix D of the FSAR. We have reviewed the test program and conclude that TVA's program generally meets the AEC's publications "Guide For The Planning of Preoperational Testing Programs" and the "Guide For The Planning of Initial Startup Programs" and is acceptable.

We have concluded that the applicant's organization and plans for testing will provide an adequate basis to confirm the safe operation of the plant, and is therefore acceptable.

12.0 QUALITY ASSURANCE

The applicant has described the quality assurance program plan for the Brown's Ferry Nuclear Plant in Appendix D and Amendment 30 of the FSAR. After construction of the plant was started, the Commission issued Appendix B of 10 CFR Part 50 which established quality assurance criteria and guidelines for nuclear plants. Tennessee Valley Authority has developed a quality assurance plan which is intended to meet the criteria set forth in Appendix B of 10 CFR Part 50. The quality assurance program developed by TVA has been in effect during the major portions of construction, procurement of equipment and includes an operational plan for quality assurance to be followed after each unit becomes operational.

The operational quality assurance plan meets all the requirements set forth in Appendix B of 10 CFR Part 50. The applicant has described in Amendment 30 of the FSAR, those design and construction activities that are not in compliance with the Quality Assurance Criteria of Appendix B. The applicant does not have complete procurement and quality assurance documentation as required by Criterion II and Criterion XVII of Appendix B for those components and systems installed or procured prior to the issuance of Appendix B or before TVA's program was improved to meet the requirements of Appendix B. The program now in effect meets the requirements of Appendix B and will be used for the remaining design and construction of Units 1, 2 and 3.

We conclude that the actions and plans when satisfactorily implemented and subject to verified by the Division of Compliance, will provide reasonable assurance that the quality of the Browns Ferry Nuclear Plant is adequate and acceptable.

13.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. We have reviewed the proposed Technical Specifications for the plant in detail and have held numerous meetings with the applicants to discuss their contents.

Modifications to the proposed Technical Specifications have been suggested by the staff to more clearly describe the allowed conditions for plant operation.

The finally approved Technical Specifications for Unit 1 will be included as part of the operating license. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. On the basis of our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features for continued operation will be available in the event of malfunctions within the plant.

14.0 CONFORMANCE WITH GENERAL DESIGN CRITERIA

Based on our evaluation of the preliminary design and design criteria for the proposed Browns Ferry Nuclear Plant, we have concluded that there is reasonable assurance that the intent of the General Design Criteria for Nuclear Power Plants, published in the Federal Register on May 21, 1971, as Appendix A to 10 CFR Part 50 in the final design of the station will be met.

15.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The report of the ACRS on this project will be placed in the Commission's Public Document Room and can be discussed by the regulatory staff at the public hearing in the event a public hearing is required.

16.0 COMMON DEFENSE AND SECURITY

The applicant states that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are United States citizens. TVA is a corporate agency of the Federal Government. We find nothing in the application to suggest that the applicant is owned, controlled, or dominated by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will obtain fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

17.0 FINANCIAL QUALIFICATIONS

We have reviewed the financial information presented in the application. The funds necessary to meet operating costs of the facility will come from operating revenues of the applicant as more fully set forth in its application. Information contained in the application indicates that such revenues will be ample to cover the estimated cost of operating this reactor as well as the safe decommissioning of the unit if such should become necessary.

We conclude that the applicant is financially qualified to engage in the activities authorized by the operating license. Our detailed evaluation of the applicant's financial qualifications is presented in the attached Appendix E.

17.1 Financial Protection and Indemnity Requirements

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 licensed under 10 CFR Part 50.

17.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be 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 Tennessee Valley Authority is, with respect to Browns Ferry Nuclear Plant, subject to the foregoing requirements. Accordingly no license 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 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

17.3 Operating License

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, for example, preoperational fuel storage only) has been executed.

Accordingly, no license authorizing operation of the Browns Ferry Nuclear Plant will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

18.0 CONCLUSIONS

Based on our evaluation of the application as set forth above and assuming favorable resolution of outstanding unresolved items described above, we have concluded that:

1. The application for an operating license filed by the Tennessee Valley Authority, as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and
2. The construction of the Browns Ferry Nuclear Plant Unit 1 has proceeded, and there is reasonable assurance that it will be completed, in conformity with Construction Permit No. CPPR-29, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facility 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 (i) that the activities to be authorized by an operating license can be conducted without endangering the health and safety of the public, and (ii) 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 an operating license in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

6. The issuance of an operating license for Unit 1 will not be inimical to the common defense and security or to the health and safety of the public.

Prior to final consideration of the matter of the issuance of an operating license to Tennessee Valley Authority for Browns Ferry Nuclear Plant Unit 1, the Commission's Directorate of Regulatory Operations will prepare a supplement to this Safety Evaluation which will deal with those matters relating to the status of construction completion and conformance of that construction to the construction permit and the application; and if such a license is authorized the Commission's Directorate of Regulatory Operations will verify, prior to issuance of the license that construction required for safe operation at the authorized power level has been completed. Further, before either a license for the preoperational storage of nuclear fuel or an operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

APPENDIX A

CHRONOLOGY

Regulatory Review of Tennessee Valley Authority Browns Ferry Nuclear Power Station Units 1, 2 and 3

September 9, 1969	Meeting with Tennessee Valley Authority representatives to discuss organization and quality levels for the completion of fabrication of Units 2 and 3 reactor vessels.
July 6, 1970	AEC letter requests environmental impact information to be submitted when filing the Final Safety Analysis Report.
July 17, 1970	Meeting with TVA representatives to discuss conformance to proposed rule on codes and standards and use of furnace-sensitized stainless steel.
September 25, 1970	TVA submits Final Safety Analysis Report and financial information as Amendment No. 9 to the application.
September 28, 1970	TVA letter containing proposed procedures of environmental statement.
August 26, 1970	Meeting with TVA representatives to discuss procedures for preparation of environmental statement.
October 1, 1970	Receive TVA proposed procedures for preparation of environmental statement.
October 9, 1970	Meeting with TVA representative to discuss onsite electrical power system.
November 12, 1970	AEC letter commenting on proposed procedures for the preparation of environmental statement.
November 24, 1970	TVA submits Amendment 10 containing Proposed Technical Specifications and reactor thermal-hydraulic information.
January 29, 1971	Meeting with TVA representatives to discuss review schedule and items requiring additional information.
March 1, 1971	TVA submits Amendment 11 containing Unit 1 Reactor Pressure Vessel Report and revised fuel design information.

March 25, 1971	AEC letter requests additional information.
April 16, 1971	Meeting with TVA representatives to discuss preparation of environmental statement.
May 7, 1971	TVA submits Amendment 12 containing proprietary information on fuel design.
May 22, 1971	AEC letter requests additional information.
June 17, 1971	Meeting at site to discuss hydrology and controls and instrumentation.
June 30, 1971	AEC letter confirming procedure for the preparation and issuance of the environmental statement.
July 15, 1971	TVA issues Draft Environmental Statement for comment.
July 30, 1971	AEC letter requests additional analyses consistent with AEC interim criteria for the performance of emergency core cooling system.
August 3, 1971	TVA submits Amendment 13 containing revised and supplementary information in response to 3-25-71 DRL letter.
October 12, 1971	AEC letter requests additional information relative to the requirements of Safety Guide 7.
October 18, 1971	TVA submits show cause information.
November 2, 1971	TVA submits Amendment 14 containing partial responses to AEC letters dated 3-25-71 and 5-22-71 and all the information requested in DRL letter dated 7-30-71.
November 8, 1971	TVA submits supplements and additions to Draft Environmental Statement for comment.
November 11, 1971	TVA submits Amendment 15 containing revised and supplementary information in response to 3-25-71 and 5-22-71 AEC letter.
November 24, 1971	AEC publishes Show Cause Determination and discussion and findings not to suspend construction of the Browns Ferry Nuclear Plant.

December 1, 1971	TVA submits Amendment 16 containing revised Proposed Technical Specifications and information in response to 3-25-71 DRL letter.
December 6, 1971	AEC letter requests additional information.
December 14 & 15, 1971	Meeting with applicant to discuss hydrology, conduct of operations, radwaste systems, electrical power and instrumentation and control system and review schedule.
January 4 and 5, 1972	Meeting to discuss onsite electrical power system, instrumentation, and control system and future proposed fuel modification.
January 19, 20 & 21, 1972	Meeting with applicant to discuss controls and instrumentation systems.
January 26 & 27, 1972	Meeting with applicant to discuss Proposed Technical Specifications.
January 26, 1972	TVA submits Amendment 17 containing proprietary information on fuel design.
February 1, 1972	TVA submits Amendment 18 containing supplemental and revised information.
February 3, 1972	Meeting with TVA to discuss quality assurance program, emergency operating procedures, hydrology, pipe whip protection; and the standby gas treatment system.
February 10, 1972	AEC letter commenting on radiological matters of the TVA Draft Environmental Statement.
February 10, 1972	Meeting with TVA to discuss review schedule and affects of stem bypassing of suppression pool.
February 11, 1972	Meeting with TVA to discuss Proposed Technical Specification.
February 14, 1972	TVA submits Amendment 19 containing supplemental and revised information.
February 23, 1972	TVA submits Amendment 20 containing proprietary information related to revised fuel design.

February 28, 1972	TVA submits Amendment 21 containing revised information related to fuel design.
February 28, 1972	TVA submits Amendment 22 containing revised information; responses to AEC letter dated 10-12-71 and partial responses to AEC letter dated 12-6-71.
March 2 & 3, 1972	Meeting with TVA to discuss Proposed Technical Specifications.
March 6, 1972.	TVA submits Amendment 23 containing proprietary information related to industrial security plans.
March 13, 1972	AEC letter requesting additional analyses of tower structural components.
March 14, 1972	Meeting with TVA to discuss onsite and offsite electrical power system.
March 20, 1972	TVA submits Amendment 24 revising information.
March 22, 23 & 24, 1972	Meeting with TVA to discuss Proposed Technical Specifications.
March 27, 1972	TVA submits Amendment 25 revising information.
March 29, 1972	AEC letter requests additional information on conbustible gas control system.
April 18, 1972	TVA submits response to AEC letter dated 3-13-72.
April 26, 1972	AEC letter requesting additional financial information.
April 26, 1972	TVA submits Amendment 26 containing revised Proposed Technical Specifications, Unit 2 Reactor Pressure Vessel Report and revised responses to DRL requests for information.
	TVA submits Amendment 27 containing revised proprietary information related to industrial security.

May 10, 1972	Meeting with TVA to discuss fuel design.
May 11, 1972	TVA submits Amendment 28 containing revised information to the Proposed Technical Specifications, revised answers to previous DRL requested information.
May 12, 1972	Meeting at site to discuss emergency power system.
May 19, 1972	TVA submits Amendment 29 containing revised information and additional financial information requested by AEC letter dated April 26, 1972.
May 25, 1972	TVA submits Amendment 30 containing revised proprietary information related to "Protection Against Industrial Sabotage" report.
May 25, 1972	TVA submits Amendment 31 containing revised responses to AEC questions and additional emergency plan information.
June 2, 1972	AEC letter requesting additional information on fuel design.
June 7, 1972	Meeting with applicant on outstanding review items needed to complete Safety Evaluation Report, including electrical power system concerns.

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

January 21, 1972

R323

APPENDIX B

50-259
50-260
50-296

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Morris:

This refers to the letter of June 28, 1971, from Roger S. Boyd, Assistant Director, Boiling Water Reactors of the Division of Reactor Licensing, requesting comments on the following:

Browns Ferry Nuclear Plant
Tennessee Valley Authority
Final Safety Analysis Report
Volumes 1 through 5 dated September 25, 1970

These comments are attached.

Sincerely,

Isaac Van der Hoven

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

Attachment

cc: E. H. Markee, USAEC

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

Comments on

Browns Ferry Nuclear Plant
Tennessee Valley Authority
Final Safety Analysis Report
Volumes 1 through 5 dated September 25, 1970

Prepared by

Air Resources Environmental Laboratory
National Oceanic and Atmospheric Administration
January 21, 1972

For short-term (0 to 2 hours) effluent emissions, two types of releases are postulated, namely 1) a release from the turbine building due to a steam line break and 2) a release from the 183 m stack. In the former a slow atmospheric diffusion rate would cause higher ground concentrations, while in the latter, because of the elevated release, a rapid diffusion to the ground would cause higher concentrations.

For the turbine building emission we have assumed a ground release and therefore have adjusted the applicant's wind speeds measured at 300 feet to the equivalent speed at 30 feet by means of the power law suggested in the ASME Guide (1968) for stable conditions. Using the adjusted speeds and the direction and stability frequencies in Tables E.0-7 and E.0-8 we have determined that 5 percent of the time diffusion rates will be slower than that equivalent to Pasquill Type F and a speed of 1 m/sec. At the nearest exclusion distance of 1200 m and assuming a building wake correction of $cA = 1500$ m, the resulting short-term concentration would be 2.9×10^{-4} sec m^{-3}.

For the elevated release we have used the unadjusted speed and direction at 300 ft under unstable conditions as shown in Table E.0-10 and determined that concentrations higher than that equivalent to Type B and a speed of 5 m/sec with an 183 m effective stack height would occur 5 percent of the time. No additional stack rise above the physical height of the stack was used because of the high wind speed and the lack of plume buoyancy. The resulting concentration at the nearest exclusion distance of 1200 m is 9×10^{-7} sec m^{-3}.

On an annual basis downwind concentrations from routine emissions from the stack would be dominated by the occurrence of unstable conditions. Thus, from Table E.0-10 we have assumed a 1.5 percent frequency of unstable (Type B) conditions at an average speed of

4 m/sec and a stack height of 183 m. The resulting maximum concentration averaged over a sector would be 2×10^{-8} sec m^{-3} at the site boundary.

In summary, our computation of the short-term concentration at the exclusion distance of 1200 m is 2.9×10^{-4} and 9×10^{-7} sec m^{-3}, respectively, for a ground and elevated release. The former is a factor of 4 higher than the applicant shows in Table 14.8-5, primarily because of the applicant's assumption of a 30 m release height. The maximum annual concentration agrees closely with the applicant's value of 2.7×10^{-8} sec m^{-3} as found in Table E.0-21.

Reference

ASME (1968), Recommended Guide for the Prediction of the Dispersion of Airborne Effluents. M. E. Smith, Editor, 85 pp.

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

APPENDIX C

11 May 1972

FINAL SAFETY ANALYSIS REPORT

FOR

BROWNS FERRY NUCLEAR PLANT UNITS 1, 2 AND 3

TENNESSEE VALLEY AUTHORITY

AEC Docket Nos. 50-259, 50-260 and 50-296

by

N. M. Newmark and W. J. Hall

After our review of the FSAR, including Amendments through No. 25, we believe that the design of the Browns Ferry Nuclear Plant Units 1, 2 and 3 can be considered adequate in terms of provision for safe shutdown for a Design Basis Earthquake of 0.20g maximum horizontal ground acceleration, and capable otherwise of withstanding the effects of an Operating Basis Earthquake of half this intensity.

Our review was based on consideration, among other things, of the design criteria and results of analyses presented by the applicant for Foundations; Intake Channel; Seismic Design of Reactor Building, Reinforced Concrete Chimney, Diesel Generator Building and Standby Gas Treatment Building, Torus Header, Piping Systems, Equipment, and Vertical Earthquake Effects; Buried Piping; and Controls for Reactor Protection System.

We believe the procedures used and the designs developed are in accord with the state of the art. We conclude that the design incorporates an acceptable range of margins of safety for the hazards considered.

N. M. Newmark

ADDRESS ONLY THE DIRECTOR,
BUREAU OF SPORT FISHERIES
AND WILDLIFE

United States Department of the Interior
FISH AND WILDLIFE SERVICE
BUREAU OF SPORT FISHERIES AND WILDLIFE
WASHINGTON, D.C. 20240

MAY 19 1971

APPENDIX D

50-259
50-260
50-296

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

Dear Mr. Price:

This letter is in response to Mr. Boyd's letter of October 14, 1970, requesting our comments on Volumes 1 through 5 of the Final Safety Analysis Report for the Tennessee Valley Authority's proposed Brown's Ferry Nuclear Powerplant, Units 1, 2, and 3. The proposed reactors are currently under construction in Limestone County, Alabama.

The Fish and Wildlife Service, by letter dated February 3, 1967, commented on the application of the Tennessee Valley Authority for a license to construct and operate the Brown's Ferry plant. Included in the letter were a number of recommendations for the conservation and development of fish and wildlife resources. The recommendations were recognized by the applicant in planning their environmental monitoring program.

Our review of Volumes 1 through 5 of the Final Safety Analysis Report indicate that the environmental monitoring program is adequate to safeguard the fish and wildlife resources in the project area.

The opportunity to present our views on this project is appreciated.

Sincerely yours,

Associate Director

APPENDIX E
FINANCIAL QUALIFICATIONS

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

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

The TVA submissions contain the estimated annual operating costs of the Browns Ferry Nuclear Power Plant, Units 1, 2 and 3, plus the estimated costs of permanently shutting down the facility and maintaining it in a safe shutdown condition. The estimated annual operating costs for the first years of operation for each unit are, in millions of dollars:

Fiscal Year	Unit 1	Unit 2	Unit 3	Total
1973	$ 30.9	-	-	$ 30.9
1974	40.3	$ 32.8	$ 14.8	87.9
1975	39.8	32.3	36.7	108.8
1976	39.4	32.0	36.3	107.7
1977	38.9	31.7	35.9	106.5
1978	38.5	31.5	35.6	105.6
1979	38.0	31.1	35.2	104.3

These estimates include the following costs: interest on investment, payments in lieu of taxes, production - operation and maintenance, insurance, overhead, depreciation, and fuel. The applicant's estimate of the cost of permanently shutting down each unit as specified in Amendment 29 to the application is $7.5 million and $63,000 annually thereafter for maintaining each unit in a safe shutdown condition.

We have examined the certified financial statements of TVA to determine whether it is financially qualified to meet these estimated costs. The information contained in TVA's fiscal year 1971 financial report indicates that operating revenues for 1971 totaled $598.0 million; operating expenses were $449.5 million, of which $80.0 million represented depreciation. The interest on long-term debt was earned 4.0 times; and the net income for the year was $119.0 million, of which $65.1 million was repaid to the U. S. Treasury as a return on the net appropriation investment and the remainder of $53.8 million was transferred to retained earnings. As of December 31, 1971, the TVA's assets totaled $3,352.4 million, most of which was invested in utility plant ($3,183.8 million); retained earnings amounted to $714.7 million. Financial ratios computed from the 1971 statements indicate a sound financial condition, e.g., long-term debt to total capitalization - .27, and to net utility plant - .46; net plant to capitalization - 1.29; the operating ratio - .75; and the rates of return on proprietary capital - 6.7%, and on total investment - 5.9%. The record of TVA's operations over the past 5 years reflects that operating revenues increased from $326.8 million in 1966 to $598.0 million in

1971; net income increased from \$47.9 million to \$119.0 million; and net investment in plant from \$2,166.6 million to \$3,183.8 million. Moody's Investors Service rates the TVA's first mortgage bonds as Aaa (gilt-edge).

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) with respect to the operation of the Brown's Ferry Nuclear Power Plant, Units 1, 2 and 3. A copy of the staff's financial analysis of the TVA is attached as an appendix.

TENNESSEE VALLEY AUTHORITY (POWER PROGRAM)

DOCKET NOS. 50-259, 50-260 AND 50-296

FINANCIAL ANALYSIS

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

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