SAFETY EVALUATION REPORT

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

TENNESSEE VALLEY AUTHORITY

BELLEFONTE NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-438 AND 50-439

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ABBREVIATIONS

a-c	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AEC	United States Atomic Energy Commission
AISC	American Institute of Steel Construction
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Material
$Btu/hr\text{-}ft^2$	British thermal units per hour per square foot
B&W	Babcock & Wilcox Company
BWST	borated water storage tank
CCWS	component cooling water system
cfs	cubic feet per second
CHF	critical heat flux
CFR	Code of Federal Regulations
Ci/sec	Curies per second
Ci/yr/unit	Curies per year per unit
DBA	design basis accident
d-c	direct current
DF	decontamination factor
DHRS	decay heat removal system
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio

ECCS	emergency core cooling system
ESF	engineered safety features
°F	degrees Fahrenheit
FSAR	Final Safety Analysis Report
ft^2	square feet
ft^3	cubic feet
ft/sec	feet per second
g	gravitational acceleration, 32.2 feet per second per second
GDC	AEC General Design Criteria for Nuclear Power Plant Construction Permits
gpd	gallons per day
gpm	gallons per minute
HEPA	high efficiency particulate air
HPIS	high pressure injection system
hr.	hour
Hz	Hertz (cycles per second)
I-131	Iodine 131
IEEE	Institute of Electrical and Electronics Engineers
in	inch
IPS	Interim Policy Statement
kV	kilovolt
kVA	kilovolt amperes
kW	kilowatt
kW/ft	kilowatts per foot

ABBREVIATIONS (Cont'd)

lb	pound
LOCA	loss-of-coolant accident
LPIS	low pressure injection system
LPZ	low population zone
m	meter
m^2	square meters
MM	modified Mercalli
mph	miles per hour
m/sec	meters per second
MSL	mean sea level
MWe	megawatts thermal
mrem	one thousandth of a roentgen equivalent man
NFPA	National Fire Protection Association
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
NSSS	nuclear steam supply system
OBE	operating basis earthquake
OSHA	Occupational Safety and Health Act
PMF	probable maximum flood
PMP	probable maximum precipitation
ppm	parts per million

PSAR	Preliminary Safety Analysis Report
psi	pounds per square inch
psig	pounds per square inch gauge
psia	pounds per square inch absolute
PWR	pressurized water reactor
QA	quality assurance
rad	radiation absorbed dose
RCPB	reactor coolant pressure boundary
rem	roentgen equivalent man
RCS	reactor coolant system
sec/m^3	seconds per cubic meter
SSE	safe shutdown earthquake
TVA	Tennessee Valley Authority
U-235	uranium 235
UO_2	uranium dioxide
USGS	United States Geological Survey
v/o	volume 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 Siting Criteria

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The Tennessee Valley Authority (hereinafter referred to as TVA or the applicant) filed with the Atomic Energy Commission (AEC or Commission) an application, docketed on June 21, 1973, for licenses to construct and operate its proposed Bellefonte Nuclear Plant, Units 1 and 2 (Bellefonte plant, Bellefonte 1 and 2, or the facility). The facility will be located six miles northeast of Scottsboro, Alabama, at the Bellefonte site in Jackson County, Alabama.

A Preliminary Safety Analysis Report (PSAR) was submitted with the application. The information in the PSAR was supplemented by Amendments 1 through 10. The PSAR and copies of these amendments are available for public inspection at the U.S. Atomic Energy Commission, Public Document Room, 1717 H Street, NW, Washington, D. C. and at the Scottsboro Public Library, 1002 South Bend Street, Scottsboro, Alabama.

This Safety Evaluation Report (SER) summarizes the results of the technical evaluation of Bellefonte 1 and 2 performed by the Commission's Regulatory staff (Regulatory staff or staff) and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the proposed facility. An assessment of the impact on the environment of this proposed facility, in accordance with Appendix D to 10 CFR Part 50 of the Commission's Regulations, implementation of the National Environmental Policy Act of 1969, is discussed in the Commission's Draft Environmental Statement, issued in February 1974.

Based on our evaluation of TVA's application to construct and operate the facility, we conclude that the Bellefonte Nuclear Plant, Units 1 and 2 can be constructed and operated as proposed without endangering the health and safety of the public. Our detailed conclusions are presented in Section 21.0 of this SER.

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

1.2 General Plant Description

The Bellefonte plant consists of two individual units sharing certain common structures, systems and components. Each of the proposed reactors will be designed to operate at a thermal power of 3600 megawatts (MWt) with an expected ultimate capability of producing 3763 MWt. The nuclear steam supply system for each unit will consist of a pressurized water reactor using two heat transport loops. The reactor core will be composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The fuel tubes will be grouped and supported in assemblies. The reactor core will be loaded initially in regions consisting of three different enrichments of U-235. Water will serve as both the moderator and the coolant and will be circulated through the reactor vessel core by four coolant pumps. The water, heated by the reactor, will flow through two steam generators where heat will be transferred to the secondary (steam) system. The water will then flow back to the pumps to repeat the cycle. An electrically heated pressurizer attached to one of the coolant loops will establish and maintain the reactor coolant pressure and will provide a surge chamber and a water reservoir to accommodate reactor coolant volume changes during operation.

The nuclear steam supply system for each unit will be housed inside a steel-lined, pre-stressed concrete, cylindrical containment structure which, in turn, will be completely enclosed by a reinforced concrete structure called the secondary containment building. The containment, including its penetrations, will be designed to safely confine the radioactive material that could be released in the event of an accident. Upon receipt of an accident signal, the secondary containment building filtration and vent system will start and maintain the pressure within the secondary containment building at a negative value. Nearly all leakage from the containment will be collected within the secondary containment building volume, subjected to multipass filtration through either of the redundant fan-filter systems, and exhausted to the environment at a rate sufficient to maintain the negative pressure within the building.

A preliminary layout of the Bellefonte plant site is shown on Figure 1.1. An auxiliary building, to be located adjacent to the pair of containment structures, will house the waste treatment facilities (portions of which are shared), components of the engineered safety features equipment, the spent fuel pools, auxiliary control room, and various related auxiliary systems for each of the two reactor units. A control building, to be located between each pair of containment structures and adjacent to the turbine building will house the control consoles and panels for the two units in a common control room. The chemical addition and boron recovery system, spent fuel cooling system, essential raw cooling water system, raw cooling water system, fire protection, control building ventilation systems, fuel oil storage tanks, offsite electric power system and various labs and maintenance shops will be shared by the two units. A turbine generator building, shared by both units, will be located on the opposite side of the containment structures and adjacent to the auxiliary building. Diesel-engine generators for each unit will be housed in separate buildings adjacent to but on opposite sides of the control building.

The steam and power conversion system for each unit will be designed to remove heat energy from the reactor coolant in the two steam generators and convert it to electrical energy. The heat rejected to the condensers will be discharged through the circulating water system to the atmosphere utilizing hyperbolic, natural draft cooling towers. Makeup water to the cooling towers will be from the Guntersville Reservoir.

The reactor will be controlled by control rod movement and by regulation of the boric acid concentration in the reactor coolant. The control rods, whose drive shafts penetrate the top head of the reactor vessel, will be moved vertically within the core by individual control rod drives. A reactor protection system, that automatically initiates appropriate corrective action whenever a plant condition monitored by the system approaches pre-established limits, will be provided. The reactor protection system and an engineered safety features actuation system will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The Essential Raw Cooling Water System and the Component Cooling Water System will operate together to provide cooling for all components necessary for safe operation. The plant will be provided with two 100% redundant and independent cooling water flow trains to maintain reactor cooling and to provide containment cooling in the unlikely event of an accident.

The two units of the Bellefonte plant will be interconnected to offsite A-C power via four 500 kV lines and two 161 kV lines to different portions of TVA's transmission system. The normal preferred source of power for each unit will be from the 500 kV system via two unit station transformers to the safety feature buses. In the event that the 500 kV system is not available, the redundant safety feature buses of each unit are powered by two separate reserve auxiliary transformers from the 161 kV switchyard. Either of two fast starting diesel generators and its associated safety features bus will be capable of providing adequate power for a safe shutdown under accident conditions with a concurrent loss of offsite power. A constant supply of d-c power to vital instruments and controls of each unit will be assured through the redundant 125-volt buses and their associated battery banks and battery chargers.

1.3 Comparison with Similar Facility Designs

The principal features of the design of the Bellefonte plant are similar to those we have evaluated and approved previously for other nuclear power plants now under construction or in operation, especially the North Anna Power Station, Units 3 and 4 (Docket Nos. 50-404 and 50-405). To the extent feasible and appropriate we have made use of our previous evaluations of these plants in conducting our review of the Bellefonte plant. Where this has been done, the appropriate sections of this report identify the other facilities involved. Comparisons to these facilities are given in Tables 4.1 and 4.2 of this report. Our safety evaluations for these other facilities have been published and are available for public inspection at the Atomic Energy Commission's Public Document Room at 1717 H Street, NW, Washington, D.C.

1.4 Identification of Agents and Contractors

The Tennessee Valley Authority will own and operate the Bellefonte plant and is the sole applicant for the facility license. TVA will specify and procure all systems, components, and elements of the plant except those supplied by Babcock & Wilcox. TVA will design, fabricate and construct the integrated plant.

The Nuclear Steam Supply System (NSSS) including the initial cores will be supplied by Babcock & Wilcox (B & W). B & W will be responsible for the design, manufacture, and delivery to the site of all items within its scope of supply. Besides the NSSS, this includes the Reactor Building Spray System, the Reactor Building Cooling System, the Spent Fuel Pool Cooling and Cleanup

System, the Fuel Handling System, the Component Cooling Water System, the Decay Heat Removal System, the Solid, Liquid and Gaseous Radioactive Waste Systems, the Turbine Bypass Valves, the Chemical Addition and Boron Recovery System, the Makeup and Purification System and the Instrumentation and Control System. The turbine generators will be purchased from the Brown Boveri Corporation.

1.5 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by the applicant considered the principal matters summarized below.

We reviewed the population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to determine that these characteristics have been determined adequately and have been given appropriate consideration in the plant design, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100) taking into consideration the design of the facility including the engineered safety features provided.

We reviewed 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, Regulatory Guides, and other appropriate codes and standards, and to determine that any departures from these criteria, guides, codes and standards have been identified and justified.

During the course of our review, we considered the response of the facility to certain anticipated operating transients and postulated accidents. We judged that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered credible. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite doses that might result in the highly unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

We evaluated the applicant's plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel), the measures to be 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 the applicant will be technically qualified to operate the plant and will have established effective organizations and plans for the continuing safe operation of the facility.

We evaluated the design of the systems provided for control of the radioactive effluents from the plant to determine that these systems can control the release of radioactive wastes from the plant within the limits of the Commission's regulations (10 CFR Part 20) and that the equipment to be provided will be capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as practicable within the contemplation of the Commission's regulations (10 CFR Part 50).

We evaluated the applicant's Quality Assurance Program for the design and construction of the plant to assure that the program complies with the requirements of the Commission's regulations (10 CFR Part 50) and that the applicant will have proper control over facility design and construction such that there will be a high degree of assurance that when completed that plant can be operated safely and reliably.

We evaluated the financial data and information provided by the applicant as required by the Commission's regulations (Section 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50) to determine that the applicant is financially qualified to design and construct the proposed facility.

1.6 Facility Modifications as a Result of Regulatory Staff Review

During the review of the Bellefonte application several formal meetings were held with representatives of the applicant, its contractors, and its consultants to discuss the facility and the technical material submitted. A chronological listing of the meetings and other significant events is given in Appendix A to this report. During the course of the review the applicant proposed or we requested a number of technical and administrative changes. These are described in various amendments to the original application. We have listed below the more significant modifications that have been or will be required to be made as a result of our review. The sections of this report where these matters are discussed more fully are noted in parenthesis.

- Upgrading of the meteorological measurement program to Regulatory Guide 1.23 (Section 2.3)
- Additional and more closely spaced core borings under all Category I structures (Section 2.5.1)
- Additional core borings and stability analysis for the intake canal slopes (Section 2.5.4)
- Consideration of additional tornado missiles (Section 3.5)
- Incorporation of additional seismic instrumentation (Section 3.7)
- Change in design requirements of containment to accomodate staff conditions related to use of the proposed ACI 359 code (Sections 3.9.1, 3.9.2 and 3.9.3)
- Provision for minimizing manual actions in mitigating the consequences of a LOCA (Section 6.3.2)

- Addition of instrumentation to detect leakage past the high pressure check valves in the CFT lines (Section 6.3.2)
- Single failure consideration of inadvertent actuation of all electrically operated passive and active components in safety related fluid systems (Section 7.3.4)
- Addition of two interlocked valves on the Decay Heat Removal letdown line to allow for cold shutdown assuming a single failure (Section 7.4)
- Provision of system level automatic bypass indication (Section 7.4)
- Additional instrumentation to follow the course of an accident (Section 7.5)
- Performance of diesel-engine generator qualification tests (Section 8.3.1)
- Supplementation of criteria for physical independence of electrical systems (Section 8.4)

1.7 Requirements for Future Technical Information

The applicant has identified in Section 1.5 of the PSAR the research and development (R&D) programs applicable to the Bellefonte plant. Those programs to be conducted by B&W are to verify the new 17 x 17 (Mark C) fuel assembly design and confirm the design margins of the NSSS. The R&D programs and their objectives are summarized in Table 1.1. The results of these programs will be reviewed generically by the staff as progress in these experimental programs is reported. All tests directed toward the verification of the 17 x 17 design are scheduled for completion during 1975, well in advance of the proposed fuel loading dates for this facility.

We have reviewed the programs and conclude that they represent the test requirements needed to evaluate the safety related performance characteristics of the Mark C fuel design. Our conclusion is based on our review of the safety related mechanical and thermal-hydraulic differences between the Mark C fuel design and the Mark B fuel design which we have reviewed and approved for previous plants using B&W NSSS such as Oconee Units 1, 2 and 3 and North Anna Units 3 and 4.

The scheduled critical heat flux (CHF) and incore flow mixing tests will use bundles which are much shorter than the Bellefonte fuel assemblies. These tests are intended to verify the applicability of the B&W-2 CHF correlation to the Mark C fuel assembly design by demonstrating that the correlation conservatively predicts the test data for the Mark C geometry and grid design. This correlation also contains an axial flux shape factor based on tests of short length bundles. The applicability of the B&W-2 correlation with the flux shape factor to actual reactor conditions can be verified by non-uniform axial heat flux CHF tests with full length fuel assemblies. We will require the applicant to demonstrate the applicability of the CHF correlation during the operating license review.

We have concluded that: (1) the test program outlined in the PSAR will provide the information necessary for the design and safe operation of this facility, (2) in the event that these R&D programs provide unexpected results, appropriate restrictions on operation

TABLE 1.1

Babcock & Wilcox Research and Development Program

TESTS	PURPOSE*
Assembly Flow Tests	Assembly Pressure Drop Hydraulic Loads Dynamics of holddown springs Fuel Rod Vibrations Verify scram times Control Rod, Guide tube, Orifice Rod Fretting and Wear
Reactor Vessel Flow Tests	Incore flow mixing/Distribution Vessel pressure drop
Assembly Mechanical Tests	Vibration and Damping characteristics Load Response
Component Mechanical Tests	Spacer Grid Spring Characteristics Seismic capability of spacer grids End-fitting characterisitics
Critical Heat Flux Tests	Verify applicability of B&W-2 correlation
Fuel Densification Tests	Refine models to be used to analyse the effects of fuel densification

* All tests provide input data for Seismic and LOCA analyses.

can be imposed or proven alternate designs such as B&W's Mark B (15x15) fuel assembly design can be utilized to protect the health and safety of the public, and (3) the applicant has met the requirements of 10 CFR Part 50.35(a) in regard to needed research and development programs.

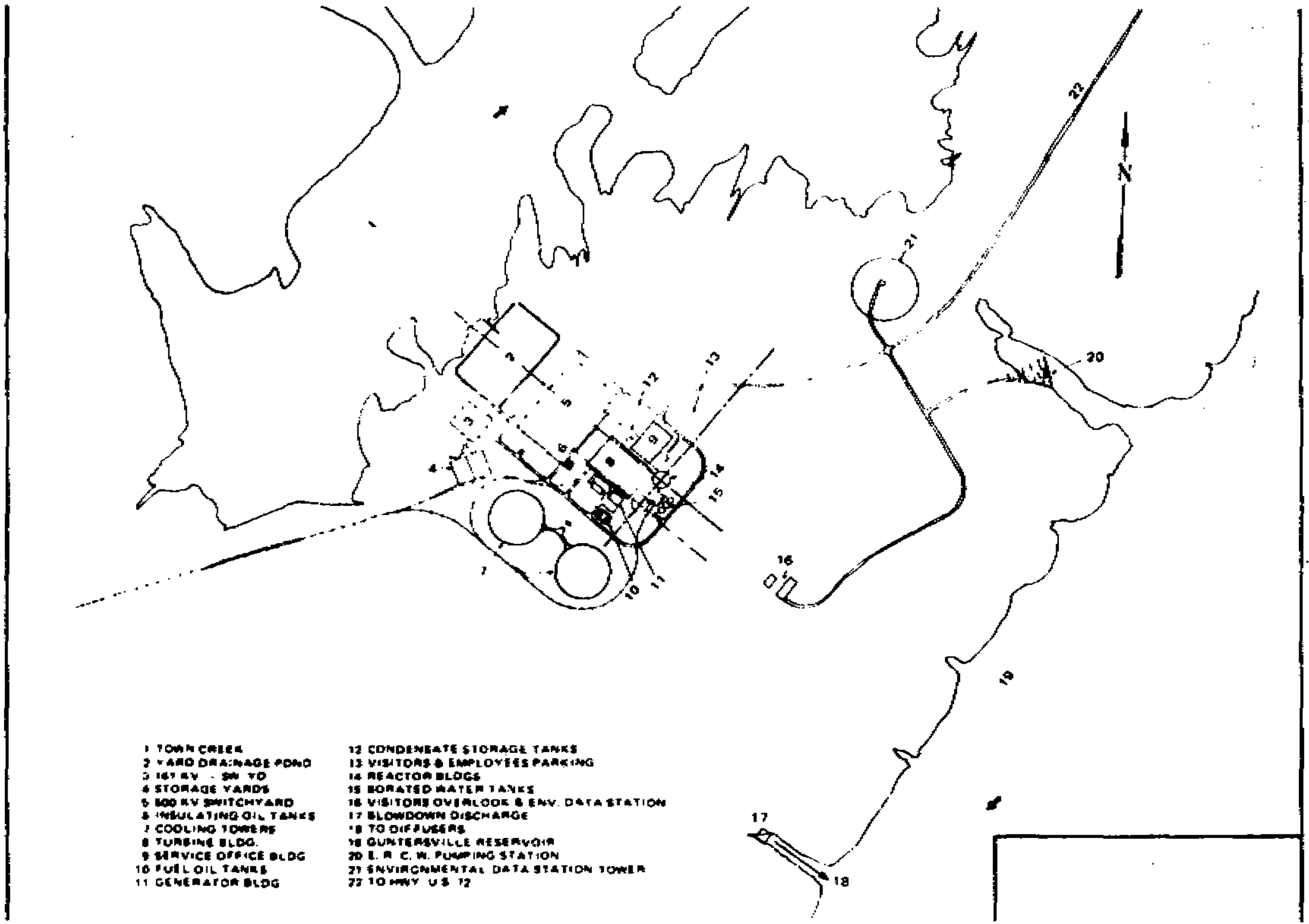


Figure.1.1. Preliminary Layout of Bellefonte Plant.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

The site for the Bellefonte Nuclear Plant is a 1500 acre tract of land located in Jackson County, Alabama, approximately 38 miles east of Huntsville, Alabama. The site location is indicated in Figure 2.1 and located inland along the banks of the Tennessee River.

The topography of the site is generally wooded with steep hills on the eastern portion. The plant will be located west of the hills. Two prominent features which characterize the site are the Tennessee River and Town Creek. Figure 2.2 represents the site in relation to these features. All land and mineral rights within the site boundary are owned by the United States Government and are in the custody of the applicant.

The applicant has indicated that the exclusion area (dashed line in Figure 2.2) will include the area within the site boundary, an area covered by the Town Creek embayment and some additional land along the shore. The minimum exclusion radius will be 914 meters. The additional land within the exclusion area is also owned by the U. S. Government in the custody of TVA. The staff has concluded that TVA meets the requirements of 10 CFR Part 100 with respect to its authority to control all activities within the exclusion area.

The nearest occupied structure is 1218 meters from the site. Table 2.1 shows the 1970 census cumulative resident population as a function of distance out to 5 miles.

TABLE 2.1

Distance (Miles)	1	2	3	4	5
Population	15	460	4095	6930	11,570

Figure 2.3 shows the 1970 cumulative resident population as as function of distance from 0-50 miles. For reference, the cumulative population corresponding to a moderately populated area of 500 people per square mile has been drawn.

The 1970 resident population within 50 miles was 847,855. The PSAR projects that this will increase to 1,650,855 in the year 2020. This corresponds to a population increase of about 14% per decade and is in substantial agreement with the population projections of the Bureau of Economic Analysis for Economic Area No. 48 which includes Jackson County.

The population center distance as defined by 10 CFR Part 100 is Huntsville, Alabama, which had a 1970 population of about 146,000. However, population projections for the cities of Scottsboro and Hollywood, Alabama, indicate that their combined populations will characterize a densely populated area by about 1990. Consequently, the applicant has identified the combined cities of Scottsboro and Hollywood, Alabama, 4 miles west of the site as the population center for the purposes of 10 CFR Part 100. Scottsboro-Hollywood had a combined 1970 population of about 9600 people. The applicant has defined the low population zone to be a circle with a 2 mile radius surrounding the plant.

On the basis of the 10 CFR Part 100 definitions of the population center distance, the exclusion area and low population zone outer boundary, our analysis of the onsite meteorological data from which dilution factors were calculated (Section 2.3 of this report), and the calculated potential radiological dose consequences of design basis accidents (see Section 15.0 of this report), we conclude that the exclusion area radius and the low population zone distance are acceptable.

2.2 Nearby Industrial, Transportation and Military Facilities

There are no gas lines, military facilities or significant industries located within five miles of the site which might present a hazard to the safe operation of the Bellefonte plant. No public roads or railways will traverse the site boundary.

The Volunteer Army Ammunition Plant (VAA) located in Chattanooga, Tennessee, approximately 50 miles from the site ships explosives by truck and rail. The closest approach over which explosives can be transported passes the Bellefonte plant at a distance of about

3 miles for rail and 2 miles for road, over U.S. 72. Because of the distances separating the plant from the explosives shipments, postulated explosions involving explosive material transported from VAA would not adversely affect the safe operation of the facility.

Because the Bellefonte plant takes water necessary for safe operation and shutdown from the Tennessee River, the staff and the applicant have investigated the possibility of river traffic interferring with the facility's intake structure. The structure itself is located at the mouth of a narrow river channel (Figure 1.1) which is narrow enough to afford natural protection against the possibility of a drifting barge, or debris impacting the structure.

The U.S. Army Corps of Engineers operates locks associated with TVA dams. Their records show that no shipment of explosive chemicals or munitions have ever passed through the oldest locks (Wheeler and Gunterville) on the TVA system.

The Scottsboro Municipal Airport is the only activity within five miles of the Bellefonte site which might have a significant effect on the safe operation of the facility. The airport is located about 4.3 miles west southwest of the facility and has one paved runway, 4000 feet in length. Scottsboro can handle small aircraft. There are approximately ten light planes based there with an estimated total of 7500 annual airport operations. A study is currently underway to determine the future disposition of this airport with respect to future air traffic in the Scottsboro vicinity. If the results of the study indicate that Scottsboro Municipal Airport will be expanded to a degree more than a reasonable growth projection, the applicant has committed to reevaluate the Bellefonte plant with respect to the probability of aircraft collisions.The applicant has conducted a probability analysis of aircraft collisions with the facility assuming a reasonable growth projection for this airport (15,000 movements per year in the year 2000.) We concur with the applicant that the probability of damaging aircraft impacts on the plant is remote on the basis of estimated future usage and conclude that an aircraft strike need not be used as a design basis event for the plant.

2.3 Meteorology

2.3.1 Regional Climatology

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

2.3.2 Local Meteorology

Based on meteorological measurements at Scottsboro, Alabama, and Chattanooga, Tennessee (7 miles west-southwest and 45 miles east-northeast of the site, respectively) mean monthly temperatures at the site may be expected to range from about 43°F in January to about 80°F in July. Precipitation is primarily associated with the winter and spring seasons, with December through May accounting for nearly 60% of the normal annual precipitation of about 54 inches. Average annual snowfall in the area ranges from 2.8 inches at Scottsboro to 4.5 inches at Chattanooga.

Wind data from the 130-ft level of a tower, located about 2.2 miles north-northeast of the plant site, for the period August, 1972 to July, 1973 indicates the distinctive bimodal wind characteristics of the river valley site location. Winds from the north-northeast and northeast directions occur about 27% of the time, while winds from the south-southwest and southwest occur about 21% of the time. Wind data from the 33 foot onsite tower for the period November, 1972 to July, 1973 indicates winds from the north-northeast and northeast directions occur about 24% of the time, and winds from the south-southwest and southwest occur about 23% of the time.

The primary cause of severe weather conditions at the site is warm, moist unstable air masses from the Gulf of Mexico contacting cold, dry continental air pressing southward and eastward. Thunderstorms are most frequent in June, July, and August accounting for about 56% of the 55 thunderstorm days expected annually. During the period 1955-1967, 38 tornadoes were reported in the one-degree latitude-longitude square west of the site while 18 were reported in the one-degree square containing the site, a mean annual tornado frequency of 1.6 representative of a one-degree square containing the site was determined. The computed recurrence interval for a tornado at the site is 870 years. The "fastest mile" of wind recorded at Chattanooga was 82 mph. The potential for high air pollution (atmospheric stagnation) exists on about 30 days every 5 years. In the period 1936-1970, there were about 70 cases of atmospheric stagnation lasting a total of about 300 days.

2.3.3 Onsite Meteorological Measurements Program

There are several phases of the applicant's pre-operational meteorological measurements program.

1. A 130-ft tower, 2.2 miles NNE of the plant site, began operation May 12, 1972. Instrumentation on this tower consists of wind speed and direction sensors at 130-ft and 33-ft elevations, although the 33-ft sensor was not installed until September 1973. Ambient temperature is measured at both elevations.
2. A 33-ft tower, erected on the proposed site of the reactor structures, became operational October 20, 1972. Only wind speed and direction are measured at the 33-ft level. This tower is to be removed when construction begins.
3. A permanent tower, 300-ft high, is scheduled for installation when either a limited work authorization or a construction permit is obtained. Instrumentation on this tower is to include wind speed and direction sensors at 33-ft and 300-ft elevation temperature and dewpoint temperature at 4-ft, 33-ft, 150-ft

and 300-ft elevations and solar radiation, total radiation, rainfall, and atmospheric pressure all at the 4-ft elevation.

The wind speed and direction instruments meet the accuracy specifications recommended in Regulatory Guide 1.23 (Onsite Meteorological Programs - February 1972). However, the accuracy of the delta-T measurements (obtained by subtracting two measured temperatures) may not fully meet the recommended accuracy specifications. The applicant is currently engaged in an investigation of the accuracy of the delta-T method presently employed, and will inform the staff of the results of this investigation prior to issuance of a construction permit. The applicant has also converted from 5-minute averaging times to 15-minute averaging times as recommended in Regulatory Guide 1.23.

Joint frequency distributions of wind speed and direction by atmospheric stability (as defined by vertical temperature gradient) were submitted for the offsite tower for the period August, 1972 to July, 1973 in accordance with the recommendations of Regulatory Guide 1.23. Wind data were measured at 130 ft and the speeds were reduced to represent conditions at 33 ft by the power law for wind profiles. Vertical temperature gradient was measured between 33 ft and 130 ft. Data recovery for this period was 90%.

Similar distributions were also submitted for the onsite tower for the period November, 1972 to July, 1973. Wind data were measured at 33 ft and simultaneous stability conditions were defined using the delta-T measurements on the offsite tower. Data recovery exceeded the recommended value of 90%.

After examination of all data submitted, the expected accident and annual average dispersion conditions for the site have been evaluated by the staff using the 9 months of data from the onsite tower. These data provide the most conservative initial estimates of relative concentration values. These relative concentration values will be verified once one full year of onsite data are made available to the staff. The applicant has agreed to provide the additional data needed to permit this verification.

2.3.4 Short-Term (Accident) Diffusion Estimates

In the evaluation of short-term (0-2 hours at the exclusion distance and 0-8 hours at the LPZ distance) accidental releases from the buildings and vents, a ground level release with a building wake factor, cA, of 1225 m^2 was assumed. The relative concentration value (X/Q) for the 0-2 hour time period which is exceeded 5% of the time was calculated by the staff using the model described in Regulatory Guide 1.4 (Assumptions Used for Evaluating the Potential Radiological Consequences of Loss of Coolant Accident for Pressurized Water Reactors - November 1970), to be 1.8×10^{-3} sec/m^3 at the exclusion distance of 914m. This relative concentration is equivalent to dispersion conditions produced by moderately stable atmospheric conditions accompanied by a wind speed of 0.2 m/sec. The relative concentration for the 0-8 hour time period at the outer boundary of the low population zone (3218m) was estimated by the staff to be 1.8×10^{-4} sec/m^3. The staff estimated relative concentration at the LPZ for the 8-24 hour time period was 1.2×10^{-4} sec/m^3; for the 1-4 day time period was 4.8×10^{-5} sec/m^3; and for the 4-30 day time period was 1.3×10^{-5} sec/m^3.

These relative concentration values exceed the applicant's design bases relative concentration values presented in Table 2.3-87 of the PSAR. The 0-2 hour value calculated by the staff is 50% higher than the design basis value proposed by the applicant. The greatest variation in values is for the 8-24 hour period where the staff's value is about a factor of 3 higher than the applicant's value. The design basis values presented by the applicant were based on meteorological observations at other TVA reactor sites and compared with 7.6 months of offsite tower data available at the time of the PSAR submittal. The applicant has not made any calculations using the onsite data.

In Section 15 of this report we have used our more conservative values in computing the offsite doses.

2.3.5 Long-Term (Routine) Diffusion Estimates

The highest offsite annual average relative concentration value of 1.1×10^{-5} sec/m^3 for vent releases occurred at the site boundary (1290m) southwest of the reactor structures. The applicant has used 7.6 months of offsite tower data in the calculations and determined a relative concentration value of 3.0×10^{-6} sec/m^3 at a distance of 1314m southwest of the reactor structures. This difference in values can be attributed to a different data base for the analysis.

2.3.6 Conclusions

The staff concludes that the 9 months of wind data from the onsite tower coupled with stability determinations from the offsite tower are the most conservative data available and, therefore, have been used in our analyses of atmospheric dispersion characteristics at the site. These data were obtained using a 5-minute averaging technique. Our concern over the accuracy of the delta-T measurement used for the determination of atmospheric stability will be resolved prior to issuance of a construction permit. Therefore, the relative concentration values presented may change somewhat with the use of a full year of data in the analyses, and with resolution of the differences to Regulatory Guide 1.23. The staff will examine and evaluate the results of the applicant's investigation of these matters.

At this time, the staff believes that resulting changes in the relative concentration values will not be significant. The staff will verify the relative concentration values when additional data are made available and report our evaluation in a supplement to this SER. The staff will also evaluate the permanent onsite tower once the location and instrumentation are finalized during our post-construction permit review.

2.4 Hydrology

2.4.1 Hydrologic Description

The site extends along the west bank of Guntersville Lake at Tennessee River Mile (TRM) 391.5, on a peninsula between Town Creek Embayment and the lake. The drainage area of the Tennessee River below Nickajack Dam and above the site is 23,340 square miles. The immediate downstream dam (TRM 349) is Guntersville with a drainage area of 24,450 square miles and upstream, about 30 miles is Nickajack Dam with a drainage area of 21,870 square miles. There are 21 major reservoirs in the TVA system upstream of the site, 13 of which have substantial reserved flood detention capacity during the main flood season. In addition, there are six major dams upstream owned by the Aluminum Company of America (ALCOA). Although the ALCOA dams often contribute to flood reduction, they do not have specific flood detention capacity.

Major flood producing storms are of the cool season, winter type and of the warm season, hurricane derived tropical storm type. Although snowfall occurs in the watershed, individual snowfalls are normally light and snowmelt is not a factor in maximum flood determinations. Most major floods in the vicinity of the site have been produced by winter-type storms in the flood season months of January through early April.

Water supply is to be taken from Guntersville Lake for cooling tower make-up at about 161 cubic feet per second (cfs). Average daily streamflow at the site is estimated to be about 38,300 cfs, based on the stream gage upstream at South Pittsburg, Tennessee (TRM 418.1). Since regulation of the river by TVA has been in effect, the maximum daily discharge was 223,000 cfs on February 2, 1957, and the minimum daily discharge was 2,900 cfs on November 1 and 15, 1953. The minimum discharge was the result of regulation by Chickamauga and Hales Bar Resevoir (which have been replaced by the Nickajack Resevoir). However, under normal operating conditions there may be periods of several hours a day when there are no releases from either or both Nickajack and Guntersville Dams. This results in surges that develop reversals of flow in the reservoir, and for short periods, flow at the site can be in an upstream direction.

Minimum plant floor elevation for all safety related structures except the intake pumping station is proposed at elevation 629.0 feet MSL (mean sea level). Normal full pool elevation of Guntersville Reservoir is 595.0 feet MSL.

There are 15 public ground water supplies within 20-mile radius of the site, 9 public surface water supplies and 4 industrial users between Nickajack and Guntersville Dams. Springs and shallow wells in the general vicinity of the site are known to supply local domestic water users.

2.4.2 Flood Potential

The applicant has estimated a probable maximum flood (PMF) having a peak flow rate at the site of 1,160,000 cfs which would reach a maximum stillwater elevation of 624.7 feet MSL. Coincident wind wave activity could raise the lake level to 628.4 feet MSL. This flood is based on the estimated probable maximum precipation for the region as determined by the Hydrometeorological Branch of the National Weather Bureau, and the suggested rainfall has been applied to a verified runoff model of the basin. The analysis is complicated by the conclusion that dams both upstream and downstream of the site would be incapable of safely passing such a severe flood and could fail. For this analysis, the flood crest at the site would be augmented by the failure due to overtopping of the earth embankments at Watts Bar, Chickamauga, and Nickajack Dams, upstream. However, the flood level at the site would be lower by about 2.7 feet due to the overtopping failure of Guntersville Dam downstream prior to the flood crest reaching the site.

The applicant proposes to protect safety-related structures, systems, and components from site drainage flooding caused by a local probable maximum precipitation (PMP) condition by grading the plant yard away from all such facilities and providing adequate capacity drainage structures. The roofs of the safety-related structures will be designed to withstand the maximum loading of the site PMP (the loading is a function of roof drainage design).

The applicant also examined the 21 major dams above the site, both individually and in groups, to determine if failure could result from a seismic event concurrent with storm runoff and create floods levels at the site higher than the PMF. The most critical condition at the site would result from the assumed simultaneous failure of Cherokee and Douglas Dams due to an earthquake equal to about one-half of a Safe Shutdown Earthquake (SSE) concurrent with a flood equal to about half a PMF. The flood resulting from this event would cause the overtopping failure of Fort Loudoun, Watts Bar and Nickajack Dams. The flood level at the site was estimated to be elevation 615 feet MSL, some 9.7 feet below the PMF level.

2.4.3 Low Water Considerations

Safety-related water requirements are estimated by the applicant to be about 134 cfs. To obtain this amount of water at the intake pumping station, the minimum Guntersville Lake level would have to be at least at elevation 568.5 feet MSL. If Guntersville Dam is assumed to fail, a flow rate of 900 cfs must be available in the river at the entrance of the intake channel to maintain a water surface level of 570.3 feet MSL and allow the required withdrawal. At the staff's request the applicant analyzed severe droughts that could occur in the Tennessee Valley and concluded that under even the most severe conditions, Guntersville Lake could be maintained about elevation 568.5 feet MSL. For the postulated failure of Guntersville Dam, the applicant concluded that the minimum flow rate at the entrance of the intake channel under severe drought conditions and without flow augmentation by the upstream reservoirs that might not be available during a severe drought, would be sufficient to meet safety-related plant requirements. If consideration is given to the applicant's upstream reservoirs, there is added assurance that plant requirements for safe shutdown will be met under all conditions. Stored water at prescribed minimum pool levels in these reservoirs could provide 1,000 cfs at the site for 1-1/2 years with no rainfall in the watershed.

Due to the operating procedures for Nickajack and Guntersville Dams, there may be periods of several hours a day when no releases are made from either dam. This results in surges that develop reversals of flow in the reservoir, and for short periods, flow at the site can be in an upstream direction. At the request of the staff, the applicant evaluated the recirculation potential between the intake and discharge for this condition. The applicant concluded, and we concur, that recirculation to the extent of adversely affecting the safety of the plant is highly unlikely. The conclusion is based on the fact that since a closed-loop cooling system will be used, the intake and discharge are very small in relation to the flow past the structures.

Because of the site location in a temperate climate, significant amounts of ice have not been observed to form on lakes in this area. This surface ice which may occur is prevented from entering or blocking the intake pumping station by the location of the openings in the front face of the structure. Additional assurance that icing will not adversely affect the plant is provided since the openings are 21 feet below the normal winter reservoir level.

2.4.4 Ground Water

Ground water at the site is derived principally from precipation. There is no distinct aquifer in the Chickamauga limestone at the site. The majority of the ground water flow moves through the residual soil overlying rock paralleling the topographic surface. Only minor amounts of water penetrate the argillaceous limestone. The applicant's observations of water levels in exploratory holes indicate a piezometric surface slightly above the top of bedrock which slopes generally with the topography toward the Town Creek embayment (northerly) of Gunterville Lake. During the subsurface investigation of the site, the applicant found no indication in any of the exploratory holes of major solution channels in the Chicamauga limestone. No ground water will be used in the construction and operation of the plant. Based on the exploratory holes and other subsurface investigations, the design basis ground water levels for all safety-related structures except the intake pumping station will be two feet above bedrock. For the intake pumping station, the design basis ground water level will be 595.0 feet MSL; the full pool level for Guntersville Lake.

The rate of ground water usage within a 20-mile radius of the site is small and in any event, is not primarily obtained from the belt of Chickamauga Limestone in which the site is located because of the poor water-bearing characteristics of the formation in this area. This, along with the hydraulic isolation of the site due to Town Creek embayment to the north and east and Guntersville Lake to the south, indicates that the effect of present or future regional ground water development will be minimal to nonexistent at the site.

Although evidence tends to indicate a very low probability that the accidental release of radioactive liquids could reach any ground water users, the applicant will install a series of six ground water observation wells down gradient from the site. Water levels and radioactivity will be measured regularly and a pump will be installed in one well to provide continuous water flow sampling.

2.4.5 Technical Specifications

The safety-related facilities that will be located where flood levels could pose a threat will be designed with flood protection features (water-tight penetrations, etc). Therefore, no special technical specifications or emergency operation requirements are anticipated to be needed.

2.4.6 Conclusions

The staff concludes that adequate flood design bases have been provided, an adequate water supply can be assured for safety-related purposes, and plant construction and operation will not adversely affect, or be affected by, regional ground water supplies.

2.5 Geology and Seismology

The staff has completed its review of the geological and foundation engineering aspects of the Bellefonte site. We have not received a formal report from our geological advisor, the U. S. Geological Survey. Based on our review of the available material, including the results of TVA's site investigations, we conclude that there are no geological structures, including faults, in the immediate site vicinity that would tend to localize earthquakes or cause near surface displacement at the site. We conclude that the foundation bedrock is sound, of high quality, and capable of supporting the facility structures with acceptable margins of safety. There is no significant solutioning nor are there significant zones of deformation beneath the foundations of major structures. Confirmatory evaluation by our advisor, the Corps of Engineers, of the stability of Category I intake channel slopes is still underway pending the results of additional investigations. Our further conclusions and those of our advisors will be presented in a supplement to this report. The following is a summary of the geology and foundation engineering aspects of the site.

2.5.1 Regional and Site Geology

The site is located on the southeast side of the Brown-Sequatchie Valley of the Cumberland Plateau Section of the Appalachian

Plateau Province. The valley follows the approximate axis of the breached Sequatchie anticline. The valley was formed after the upper, more resistant sandstones were eroded from the crest of the anticline exposing the less resistant underlying carbonate rocks. The Tennessee River has entrenched itself to elevation +570 feet near the site with the general elevation of the valley being about +630 feet. Low monoclinal ridges reaching elevation +800 feet rise above the valley floor. Those ridges are formed by more resistant rock strata. One of these ridges lies between the site and Guntersville Lake, the shore of which is about 3000 feet southeast of the site. The Cumberland Plateau bounding the Sequatchie Valley, reaches elevations of +1400 feet on both sides.

The bedrock underlying the region in which the site is located consists of limestones, dolomites, shales and sandstones that were formed throughout the Paleozoic era in alternating environments ranging from long periods of submergence and deposition to extended periods of uplift and erosion. The Chicamauga limestone, which is the only formation involved in the site foundations was formed during the middle Ordovician era. The site region is believed to have been physiographically high and therefore exposed to erosion since the end of the Paleozoic era.

Although there were several periods of structural deformation during the Paleozoic era the period of greatest development of the folds and faults comprising the Valley and Ridge Province (including the outlying Sequatchie Anticline and fault) probably took place near the end of the Paleozoic era. There is no evidence of major tectonic activity since that time.

The Valley and Ridge structure, of which the Sequatchie Anticline and fault complex is an outlier, is characterized by asymmetrical folding and thrust faulting to the west. There features are believed to have been produced by forces acting from the southeast which produced large bedding plane thrust plates without involvement of the basement. Movement of these plates produced a series of inbricate thrust faults and rootless folds.

The fault bounded blocks strike northeast and dip to the southeast. The Sequatchie fault is the western most of these thrust faults. Dominant regional bedding attitudes as reflected at the site strike North 40°E and dip 17° to the southeast. Prominent joint sets range from North 30° East to North 50° East and dips range from 70° to 80° to the northwest.

The stratigraphy of the Sequatchie Valley and the bounding Cumberland Plateau consists of nearly flat lying sandstone, shale, limestone and dolomite representing normal stratigraphic sequences. A major exception is along the Sequatchie thrust fault, which in the site area places the Ordovician Chicamauga formation in contact with the Mississippian Fort Payne formation. The Sequatchie fault lies about 2 1/2 miles northwest of the site and dips steeply beneath the site so that it is at a depth of several thousand feet beneath the Bellefonte site. Structurally the fault is located on the northwest, asymetrical limb of the Sequatchie Anticline.

The topography in the immediate vicinity of the site slopes northwest to the Town Creek Embayment, then rises in a series of low knobs and ridges. To the southwest and northeast lie Dry Creek Embayment and Mud Creek Embayment, respectively. These lows were formed by erosion along more soluable belts of the lower Chicamauga and Upper Knox Groups. To the southeast, betwcen the site and Guntersville Lake lies the monoclinal ridge discussed earlier, rising to elevation +800 feet. The site is underlain by from 2.6 to 35.8 feet of residual soil over about 1400 feet of Chicamauga limestones and shales. Five hundred feet of the Chicamauga was penetrated by borings at the site. Several marker beds (six) consisting of meta-bentonite volcanic ash showed good continuity between borings indicating absence of deformation beneath the site since before the middle Ordovician era. Prior to submission of the PSAR, the site had been investigated by borings drilled on a 100 foot grid in the facility site area. The boring logs indicated that although there was some evidence of solution activity in the upper 10 feet of rock, and an occasional small void in the upper 10 to 20 feet, solution activity was non-existent below 20 feet. However, as units of the Chicamauga are known to be subject to solutioning in other areas, the applicant, at the staff's request, drilled additional holes on a 50 foot grid in the area of the major safety related structures. The additional investigations confirmed the original conclusion that there was no solution activity below the foundation levels.

2.5.2 Vibratory Ground Motion

A discussion of the geology of the site region is contained in Section 2.5.1 above and that characterization is followed for the tectonic provinces of the region. This section describes the determination of the Safe Shutdown Earthquake (SSE) based on the seismicity and tectonic characteristics of the Southern Appalachian Mountains and the Appalachian Plateau.

The historical record of earthquakes in the Southern Appalachian Mountain region reveals significant differences in the seismic characteristics among its three tectonic provinces. The Valley and Ridge Province shows the greatest rate of earthquake occurrence. Areally, this activity is confined to the complexly thrust-faulted southern Valley and Ridge. On the basis of historical activity and the lack of a demonstrated relationship between earthquakes and geologic structure, we consider an intensity VIII, measured on the Modified Mercalli (MM) scale, equally probable (a low order of probability) for any place in the southern Valley and Ridge Province. Earthquakes have occurred with less frequency in the Piedmont Province than in the Valley and Ridge. Geographically, there appears to have been a tendency to clustering of activity in an east-west zone in central Virginia. Similar clustering has been noted in an east-west zone in South Carolina (Bollinger, 1973). On the basis of historical activity, we consider the occurrence of an intensity VII equally probable (a low order of proability) anywhere in the Piedmont Province. The Piedmont Province is about 100 miles east of the Bellefonte site. Thus, the maximum intensity VII earthquakes have been the characteristic maximum event in that area and are no hazard to the site. Historical earthquakes in the Atlantic Coastal Plain have occurred in recognizable geographic clusters. One cluster in the vicinity of Charleston, South Carolina, is associated with the Northeast Georgia (Raritan) embayment, although it has no recognized association with geologic structure. This is also the epicenter region of the large intensity X MM earthquake of 1886. Because of the correlation with the embayment and the spatial clustering exhibited by historical events (more than 400 quakes in the Charleston area), we have accepted that near future earthquakes will also occur in the same region. Thus, an earthquake in the Coastal Plain Province is not expected to cause an intensity at this site that will exceed an intensity of approximately VI to VII.

The SSE intensity of this site is based upon the following considerations:

a. Near future earthquakes in the Atlantic Coastal Plain would occur in geographic clusters near Charleston, South Carolina, following the pattern that has shown stability in more than 200 years of historical record.

b. The maximum earthquake in the Piedmont Province would be a maximum intensity VII and would be at least 100 miles to the site.

c. The maximum earthquake in the Valley and Ridge Province will not exceed the intensity VIII event of May 31, 1897 centered in Giles County, Virginia.

d. The maximum earthquake in the Appalachian Plateau will not exceed intensity VI.

Consideration a. above would result in a site intensity of VI-VII by postulating a repeated occurrence of the 1886 Charleston, South Carolina, earthquake of maximum intensity X. Consideration c. would result in a site intensity VIII by postulating a repeat of the 1895 Giles County, Virginia, earthquake of maximum intensity VIII. Based on the above considerations the staff views the SSE acceleration of 0.18g proposed to be used for this facility to be an adequately conservative value.

There is no evidence of fault movement in the site region since the close of the Paleozoic era. Therefore, we conclude that surface faulting need not be a consideration in the site evaluation.

2.5.3 Foundation Engineering

All Category I structures except for the borated water storage tanks will be founded on bedrock. The rock is adequate to support the structures. The limestone has an average density of 160 lbs/ft^3 and unconfined compressive strengths range from 8010 psi to 36,000 psi. Dynamic properties of the rock include Poissons ratio of 0.31, Youngs Modulus of 7.73×10^6, and bulk modulus of 6.64×10^6. There are no deformation zones that would tend to weaken the foundation. The applicant has committed to geologically map and photograph in detail the walls and floors of the excavations for Category I structures. The six beds of meta-bentonite will not be involved in the foundation. The borated water storage tanks will be founded on Class 1 backfill placed after excavation into bedrock.

Class 1 backfill, which will be placed around all Category I structures, will be select material placed in 6 inch layers and will be compacted to a minimum of 95% of the maximum standard proctor density at optimum moisture content. A minimum of at least one test for each 2000 cubic yards placed will be performed.

We have evaluated the investigations performed and conclude that the foundation materials are adequate to support the facility structures with sufficient margin of safety.

2.5.4 Intake Channel Slope Stability

Cooling water, including emergency cooling water will be drawn from Guntersville Lake by way of a Category I channel, excavated within a natural draw, extending about 1200 feet from the reservoir to the intake structure. Soil will be excavated down to bedrock surface 100 feet on both sides of the channel centerline. The soil side slopes will then be cut back on a three horizontal to one vertical ratio. A mid-channel trench will be excavated into bedrock to insure an emergency cooling water supply in the event of the loss of Guntersville Dam. The slopes will be designed for the following conditions: sudden drawdown; sudden drawdown plus 1/2 SSE; and SSE plus normal pool. The slopes were analyzed statically by the Modified Swedish Slip Circle Technique. Slope behavior under earthquake conditions (SSE and 1/2 SSE plus sudden drawdown) were evaluated using pseudo-static slip circle analyses. The minimum factor of safety derived from the analyses was 1.8 for the SSE and normal pool condition. To preclude loss of the channel resulting from a mechanistic type failure, the applicant has committed to construct a 75 foot wide beram on each side of the channel which is excavation in bedrock.

The review of the slope stability of the intake channel is not complete pending the results of additional investigations and analysis of laboratory test data being carried out by the applicant. These investigations are primarily confirmatory and we believe that the design of the slopes is adequate. Final conclusions of the staff and our advisor, Corps of Engineers, on these investigations will be reported in a supplement to this report.

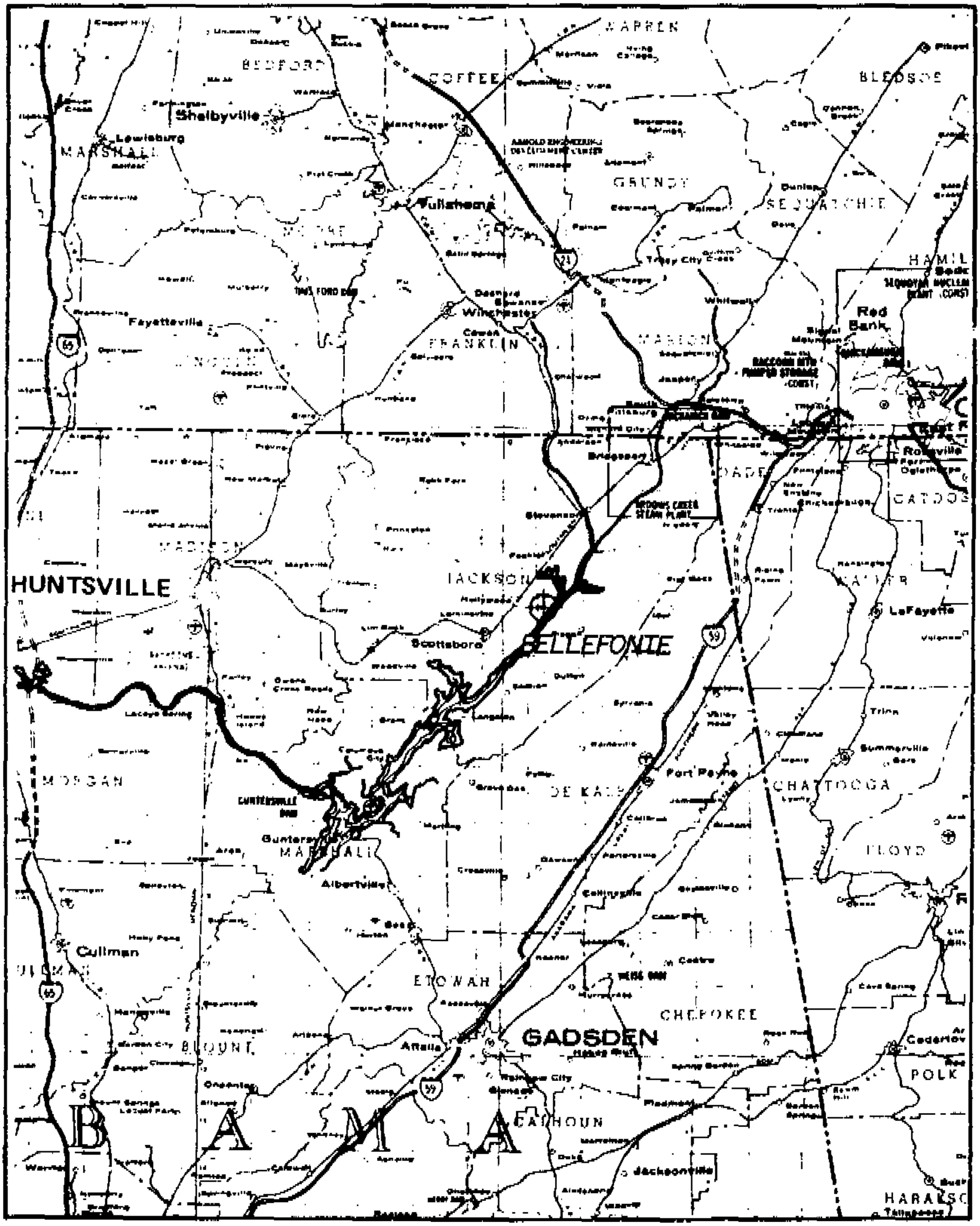


Figure. 2.1. Bellefonte Site Location.

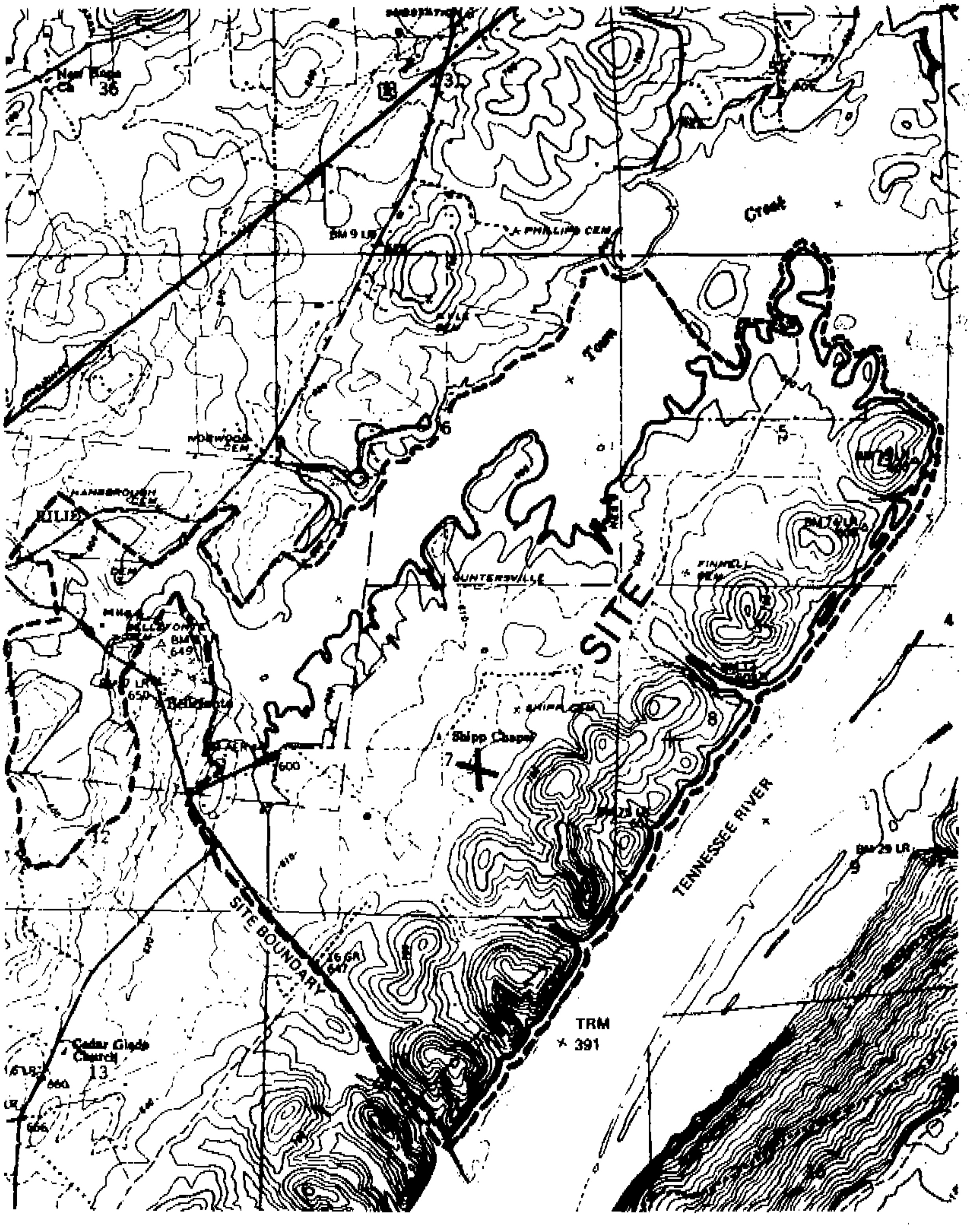
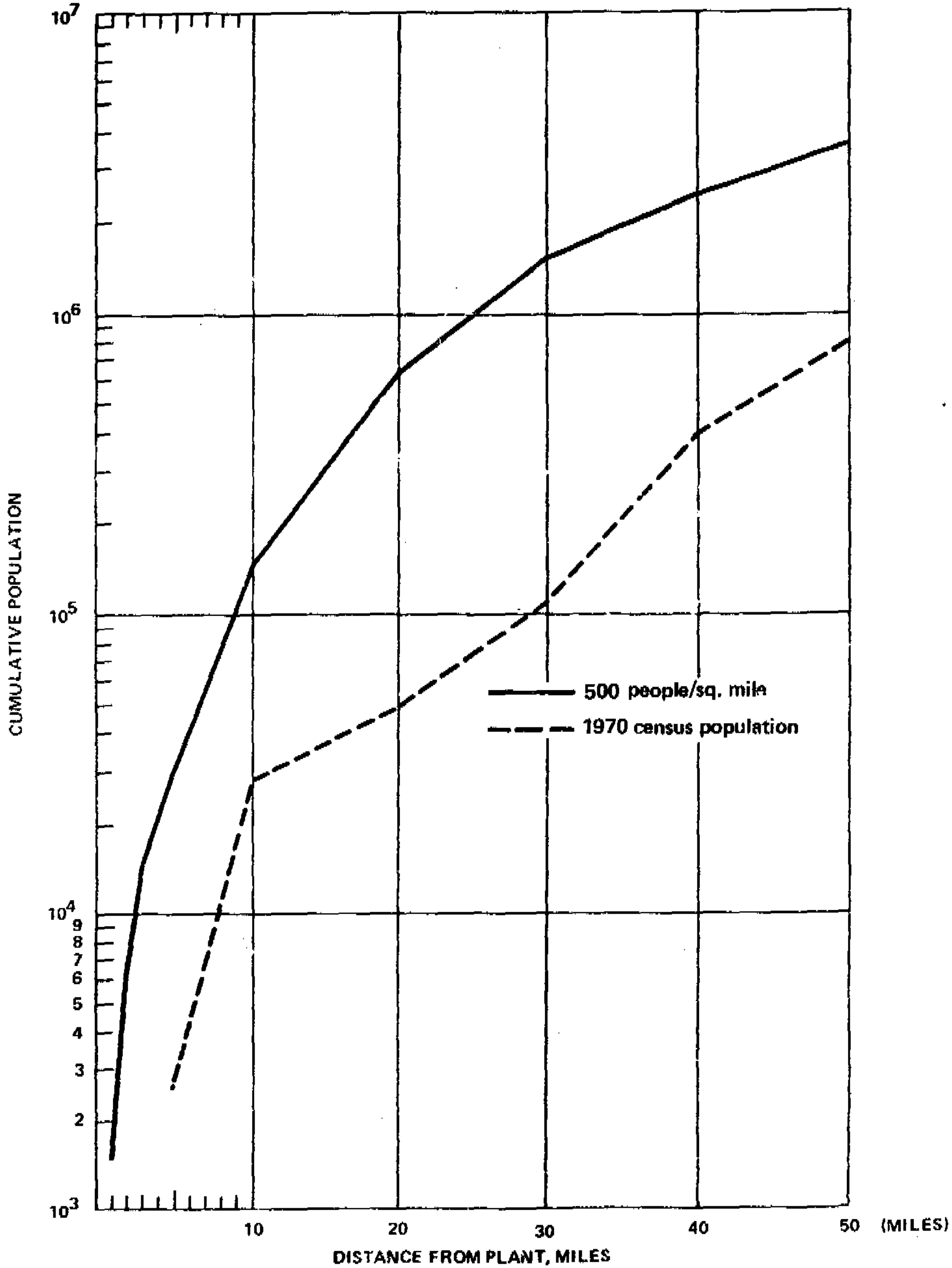


Figure. 2.2. Topographical Map of the Site Area.

BELLEFONTE POPULATION DISTRIBUTION



3.0 DESIGN CRITERIA FOR STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with AEC General Design Criteria

The applicant has stated that the Bellefonte plant will be designed, constructed, and operated in accordance with the Commission's General Design Criteria for Nuclear Power Plants (GDC) (Appendix A to 10 CFR Part 50). Each criterion is presented in Section 3.1 of the PSAR. On the basis of our review of the documentation supporting this commitment, we have concluded that this facility can be designed, constructed and operated to meet the GDC requirements.

3.2 Classification of Structures, Systems and Components

Table 3.2.1.2 of the PSAR identifies those structures, systems and components important to safety that are designed to withstand the effects of the Safe Shutdown Earthquake (SSE) and remain functional. These features (Seismic Category I) are required to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

All other structures, systems and components that may be required for operation of the facility are designed to other than

Seismic Category I requirements. Included in this classification are those portions of Category I systems which are not required to perform a safety function. Seismic Category I structures, systems and components, those items important to safety, are designed to withstand the effects of a SSE and remain functional, have been identified in an acceptable manner.

The applicant has applied the American Nuclear Society (ANS) classification system to those water and steam containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and its maintenance within safe shutdown conditions, and (3) to contain radioactive material. ANS Safety Classes 1, 2 and 3, correspond to Quality Group A, B and C in Regulatory Guide 1.26 (Quality Group Classification, and Standards, March 23, 1972).

For those fluid systems identified in Regulatory Guide 1.26, we and the applicant are in agreement on the application of the Quality Group Classification System. The applicant has identified in Table 3.2.2 of the PSAR those fluid systems important to safety and the industry codes and standards applicable to each pressure-containing component in the systems.

Piping and instrumentation diagrams identify the boundary limits of each classification group within the fluid systems. Pressure retaining components in fluid systems within the boundaries of the applicant's Safety Classes 1, 2 and 3 will be built to meet the requirements of the applicable codes. Conformance with such codes is an acceptable basis for meeting the requirements of General Design Criterion 1 and provides reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Wind and Tornado Design Criteria

All facility Category I structures exposed to wind will be designed for a 95 mph basic wind 30 feet above grade with a 100-year period of recurrence. In addition, these structures will be designed to resist a tornado with a maximum rotational plus translational wind velocity of 360 mph and a maximum depressurization loading of 3 psi in a period of 3 seconds. Appropriate tornado-generated missiles have also been postulated in the design. ASCE Paper No. 3269, "Wind Forces on Structures" (Reference 20) is being utilized to determine the loads resulting from these wind and tornado effects.

Also, structures are to be arranged on the plant site and protected in such a manner that a collapse of structures not designed for tornados will not affect those designed for tornados.

The use of these loading criteria provides reasonable assurance that, in the event of wind or tornados, the structural integrity and safety function of Seismic Category I structures will not be impaired by the specified environmental forces. Conformance with these criteria is an acceptable basis for satisfying the requirements of General Design Criterion 2. We conclude that the applicant's wind and tornado design criteria are acceptable.

3.4 Water Level (Flood) Design Criteria

All Category I structures will be designed for bouyancy and static water force effects associated with the probable maximum flood (PMF) water level discussed in Section 2.4, Hydrology. The additional forces due to wave runup will be included in the design of the Intake structure, which is the only Category I structure subject to such conditions .

Conformance with these criteria is an acceptable basis for satisfying the requirements of General Design Criteria 2 and 4 as related to environmental design basis for structures.

We conclude that the use of these design loading criteria provides reasonable assurance that, in the event of flooding, the Category I structures can be expected to withstand the specified environmental forces without impairment of their structural integrity and safety function.

3.5 Missile Protection

The effects of a spectrum of tornado generated missiles and those generated by rotational machinery or pressurized components, have been considered in the design of essential structures and vital equipment and systems. The applicant has assumed that a tornado having a rotational velocity of 300 mph and translational velocity of 60 mph could generate the following tornado missiles:

(a) A 2-inch x 4-inch x 12 foot board having a density of 40 pounds per cubic foot at a velocity of 300 mph end-on;

(b) A 7-inch x 9-inch by 8 1/2 foot crosstie having a density of 50 pounds per cubic foot at 300 mph end-on;

(c) A steel pipe 2 inches in diameter by 7 feet long, end-on at 100 mph;

(d) An automobile weighing 4000 pounds at a velocity of 50 mph and no higher than 25 feet off the ground.

In response to our request, the applicant has expanded the missiles spectrum to include the following items:

(a) a utility pole of 13.5-inch diameter x 35 feet long with density of 43 pounds per cubic feet;

(b) a 1-inch diameter steel rod x 3 foot long with a density of 490 pounds per cubic foot;

(c) a 3-inch schedule 40 x 15 foot long pipe with a density of 490 pounds per cubic foot;

(d) a 6-inch schedule 40 x 15 foot long pipe with a density of 490 pounds per cubic foot;

(e) a 12-inch schedule 40 x 15 foot long pipe with a density of 490 pounds per cubic foot.

The applicant's method of analysis given in Section 3.3 of the PSAR is based on the mathematical model presented by Bates, et al. which in turn utilizes Hoecker's studies of the tornado which occurred at Dallas, Texas on April 2, 1957. The resultant missile velocities and heights determined will be used to design missile protection for essential systems and components exposed to the damaging effects of tornado missiles. The assumptions used and subsequently the results of the calculation are similar to those for recently reviewed and approved reactor plants. We conclude that the facility will be adequately protected against tornado missiles.

The applicant has proposed design criteria in Section 3.5 of the PSAR that will be used to assure protection of safety related systems from missile damage due to failure of pressurized liquid or gas storage systems. We conclude from our review that these criteria are adequate to protect safety related systems from missiles that could be generated due to the failure of components containing stored energy.

With regard to the potential of missiles originating from the main steam turbine during overspeed conditions, the applicant indicates that the following control systems will be provided to preclude excessive overspeed of the turbine.

1. A conventional mechanical hydraulic control (MHC) system utilizing hydraulic-servo-valve controlled power pistons as the final control elements for positions of the steam flow control valves. The command signal to the servo-valves is generated at the speed governor. In addition, an initial pressure limiter and a pair of vacuum limiting devices are also provided. The former will prevent the opening of the control valves on loss of steam pressure, while the latter will provide full closure of all control valves at high turbine back pressure.
2. A turbine protection system (TPS) will utilize two overspeed governors consisting of spring-loaded bolts with eccentric centers of gravity, to protect the turbine against excessive overspeed. The overspeed governors are sequentially set to operate at 110 and 112% of rated speed, and can be on-line tested periodically to ensure system operability.

Irrespective of the above discussed overspeed protection system, the applicant has also indicated that the facility arrangement provides a low probability that the essential safety related components and structures would be struck by a turbine missile.

The potential damage that could be caused by a missile in the immediate vicinity of impact on concrete targets will be determined by the use of the Modified Petry Formula (Reference 24). In the case of steel targets, formulas developed by the Stanford Research Institute (Reference 25) for estimation of penetration of missiles will be used. The overall structural response of the target when impacted by a missile will be evaluated by methods presented by Williamson and Alvy in a paper, "Impact Effects of Fragments Striking Structural Elements." (NP-6515, 1957).

The use of these design bases and criteria for missile protection as discussed above provides reasonable assurance that, in the event of the generation of the postulated missiles, the resulting loads and effects will not impair the structural integrity of Category I structures, or result in any loss of required protection of Category I systems and components contained by such structures. We conclude that conformance with these design loading criteria is an acceptable basis for satisfying General Design Criteria 2 and 4 and that acceptable missile protection can be provided.

We are currently performing a generic study on the matter of turbine missiles. The results of this study are not yet available, however we expect that these results will confirm the adequacy of the facility design. When the results of the study are available we will evaluate the impact on this facility and if additional protection is required beyond that already afforded by the safety related structures, then we will require appropriate changes.

3.6 Postulated Pipe Breaks Outside Containment

The applicant states that the physical layout and arrangement of the facility will provide adequate physical separation of various systems to preclude such events as pipe rupture outside of containment from adversely affecting safe shutdown. The systems and components required to mitigate the consequences of each postulated pipe break, including safe shutdown to cold conditions, will be identified and will be separated into redundant trains and enclosed in suitably designed structural areas to protect the redundant trains from common failure modes due to a postulated pipe break. The identified redundant trains will be designed to ensure adequate safe shutdown capability in the event that the pipe break accident involves one of the redundant trains coincident with a single active failure in the remaining train.

The applicant's commitment regarding the design of piping systems will adequately conform to the criteria set forth in our July 12, 1973 letter to the applicant. We conclude that the design criteria and basis for the postulated pipe breaks outside containment are acceptable.

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

The applicant's criteria to be used for identifying high energy fluid piping and for postulating pipe break locations, break orientations and break flow areas will be consistent with the criteria set forth in Regulatory Guide 1.46 (Protection Against Pipe Whip Inside Containment, May, 1973).

The provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide acceptable assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the Safe Shutdown Earthquake (SSE) and a concurrent single pipe break of the largest pipe at any one of the design basis break locations, the following conditions and safety functions will be accomodated and assured:

1. The design basis loss-of-coolant accident will not lead to multiple failures of piping, that could aggravate the consequences of a pipe rupture.
2. The reactor emergency core cooling systems can be expected to perform their intended function.
3. Structures, systems and components important to safety will be appropriately protected.

The analytical methods and procedures that will be used to determine pipe motion subsequent to rupture and the pipe-whip restraint dynamic interaction appropriately consider the structural characteristics of the system. The pipe-whip restraints will be designed to withstand the resultant loadings in accordance with acceptable criteria.

On the basis of our review, we have concluded that the criteria that will be used for the identification, design and analysis of piping systems where postulated breaks may occur are acceptable and provide an adequate design basis in meeting the applicable requirements of General Design Criteria 1, 2, 4, 14 and 15.

3.8 Seismic Design

The input seismic design response spectra (Operating Basis Earthquake (OBE) and SSE) and the damping values applied in the design of Seismic Category I structures, systems and components are in accord with the positions in Regulatory Guide 1.60 (Design Response Spectra for Nuclear Power Plants, January 1973) and Regulatory Guide 1.61 (Damping Values for Seismic Design of Nuclear Power Plants, October, 1973) respectively.

The synthetic time history to be used for the design of Category I plant components and equipment is adjusted in amplitude and frequency to envelope the response spectra defined in the Regulatory Guide 1.60.

All Category I structures except the borated water storage tanks are founded on rock and the lumped-soil spring approach is used to account for the soil-structure interaction effects. For the soil-supported borated water tanks, the lumped-soil spring approach can be used provided that the applicant submits additional supporting information prior to issuance of a construction permit to demonstrate the similarity in soil properties, depth of soil medium, fundamental frequency, and other characteristics between the borated water tank structure and a diesel-generator building whose adoption of the lumped-soil spring method for soil-structure interaction analysis has been justified. The applicant will submit this additional supporting information prior to issuance of a construction permit.

Conformance with Regulatory Guides 1.60 and 1.61 will provide reasonable assurance that for an earthquake whose intensity is 0.09g for OBE, and 0.18g for SSE, the resulting accelerations and displacements imposed on Category I structures, systems, and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings. We conclude that compliance with these Regulatory Guides is an acceptable basis for satisfying the provisions of General Design Criterion 2.

Modal response spectrum multi-degree-of-freedon and time history methods form the bases for the analyses of all major Category I structure, systems, and components. Governing response parameters are 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 modal responses are used for closely-spaced modal frequencies. The square root of the sum of the squares of the maximum co-directional responses are used in accounting for the three components of the earthquake motion. Floor response spectra inputs to be used for design and test verification of structures, systems, and components are generated from the time history method. Vertical seismic-system dynamic analyses are employed for all structures, systems and components where analyses show significant structural amplifications in the vertical direction. The system and sub-system analyses are performed based on elastic theory .

We conclude that the seismic analysis methods and procedures proposed by the applicant provide an acceptable basis for system and subsystem seismic design.

The installation of seismic instruments in the reactor containment structure and at other Category I structures, systems and components as proposed by the applicant constitutes an acceptable program to adequately record data on seismic input of ground motion as well as data on the seismic responses of major structures and systems. The type, number, location and utilization of seismic instrumentation as defined by the program comply with the recommendations of the proposed Revision 1 of the Regulatory Guide 1.12 (Instrumentation for Earthquakes Draft 3 dated 1/10/74).

We conclude that the seismic instrumentation program proposed by the applicant is acceptable.

3.9 Design Category I Structures

3.9.1 Primary Concrete Containment

The primary containment structure will be a cylindrical structure with a shallow domed roof, and a foundation ring anchored with prestressed grouted rock anchors into a limestone rock foundation. A construction joint separates the primary containment foundation ring and the interior concrete base slab. The cylindrical portion (hoop and vertical) and the dome are prestressed by a post tensioning system with ungrouted tendons. The foundation ring is conventionally reinforced and also serves as the foundation for the secondary containment. A continuous access gallery is provided beneath the ring slab for access to the vertical tendons, which are anchored directly into the foundation rock.

The primary containment structure will be designed in accordance with the Proposed Standard Code for Concrete Reactor Vessels and Containments, ACI-ASME (ACI 359), 1973 edition as modified by conditions set forth in our September 14, 1973 letter to the applicant. Since the Code is presently issued for trial use, the applicant has included in the PSAR a list of deviations, corrections, modifications and clarifications to the Code which are acceptable and do not substantially change the level of conservatism of the proposed code.

Since the Code has not been formally adopted, the applicant will be permitted to use TVA General Construction Specification No. G2 for plain and reinforced concrete in lieu of the requirements of the ACI-ASME Code.

The static analysis for the containment shell will be based on thin shell theory with elastic material behavior. The finite element method will be utilized in analyzing the primary and secondary containment foundation ring, a portion of the cylinder walls, tendons and grout, and the surrounding rock.

The applicant has performed full scale tests of the prestressed grouted rock anchors at the plant site to confirm the feasibility of anchoring the vertical prestressing tendons from the containment structure directly into the rock foundation. These tests have been used to determine the minimum height of grout column, to verify that the assumptions used in the design are conservative, to determine that the anchorage is unaffected by cyclic loading and to determine the rock modulus.

The liner design for the containment is similar to those previously accepted. Tests, as outlined in Appendix 3.8A of the PSAR, will be conducted on simulated models of the liner plate and vertical stiffener assembly to determine the shear and pullout capacities of the angle anchorage.

Prior to operation, each containment will be subjected to an acceptance test in accordance with Regulatory Guide 1.18 (Structural

Acceptance Test for Concrete Primary Reactor Containment, 12/72) during which the internal pressure will be 1.15 times the containment design pressure. In the first containment structure tested, strain measurements in the concrete will be determined at the critical locations of the structure.

In addition to the documents mentioned above, the construction, testing and quality control will be based on Regulatory Guides 1.10, (Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures, 1/73); 1.15, (Testing of Reinforcing Bars for Category I Concrete Structures, 12/72); 1.19, (Non-destructive Examination of Primary Containment Liner Welds, 8/72); 1.35, (Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures, 2/73); and 1.55, (Concrete Placement in Category I Structures, 6/73).

The criteria used in the analysis, design and construction of concrete containment structures, to account for the loadings and conditions that are anticipated to be experienced by the structures during the service lifetime, are in conformance with acceptable codes, standards, Regulatory Guides and specifications.

The use of these design criteria defining the applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures, the structural acceptance criteria; the materials, quality controls and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of winds, tornados, earthquakes and various postulated accidents occurring within the containment, the Seismic Category I containment structures will withstand the specified conditions without impairment of their structural integrity and safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2, 4, 16 and 50.

3.9.2 Containment Internal Structure

The containment interior structure consists of a concrete shield wall surrounding the reactor, secondary shield walls surrounding the remainder of the nuclear steam supply system, a refueling canal and other structural elements such as floors, walls, columns, and equipment supports.

The internal structures will be designed in accordance with the ACI 318 Code, 1971 edition, for concrete and the AISC Code, 1969 edition, for structural steel.

The applicant has considered all the loads which may act on the structure during its lifetime, such as dead and live loads, accident induced loads including pressure and jet loads, and seismic loads. At the request of the Regulatory staff, the applicant has revised the load combinations and acceptance criteria used in the design to be in agreement with our position on this matter.

The design of the interior structure will be evolved through a series of stages. First, various structural components will be analyzed and designed individually for governing loading combinations with simplified assumptions of geometry and boundary conditions. Subsequently, the interaction effects of the structural components will be investigated.

The use of these design procedures and criteria provides reasonable assurance that the Category I containment internal structures will withstand all the specified design loads (including those due to earthquakes and various postulated accidents occurring within the containment) without impairment of the structural integrity and safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2, 4, 16 and 50.

3.9.3 Other Category I Structures

Category I structures other than primary containment and its interior will be built from structural steel and reinforced concrete members. The secondary containment structure will be a conventionally reinforced concrete shell. All other structural components will consist of slabs, walls, beams and columns. The design method for reinforced concrete will follow that specified in the ACI-318 Code. Structural steel components will be designed in accordance with the AISC specifications.

The various conditions used in the design of these Category I structures will include an appropriate combination of loads likely to occur during normal operation or shutdown, and during postulated accidents and earthquakes. At the request of the Regulatory staff, the applicant has revised the load combinations and acceptance criteria used in the design to be in agreement with the staff's position on this matter.

The use of these design criteria will provide reasonable assurance that the Category I structures will withstand all the specified loads without impairment of their structural integrity and safety functions. Conformance with these requirements constitute an acceptable basis for satisfying the requirements of General Design Criteria 2 and 4.

3.9.4 Foundations and Concrete Supports

The foundation for the containment is discussed in Section 3.9.1. The two borated water storage tanks are supported on reinforced concrete ring foundations on highly compacted backfill and placed on sound base rock. All other foundations for Category I structures consist of reinforced concrete slabs on rock.

The foundations will be designed in accordance with the ACI-318-71 Code.

With the reservation noted in Section 3.8 on soil-structure interaction, the use of the above design and analytical methods constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2 and 4.

3.10 Mechanical Systems and Components

3.10.1 Dynamic System Analysis and Testing

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

The tests will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and operating modes associated with the design operational transients. The tests, as planned, will develop loads similar to those experienced during reactor operation and are consistent with recent Regulatory staff positions concerning preoperational piping dynamics effects test programs. Compliance with this test program constitutes an acceptable basis for satisfying of the applicable requirements of General Design Criterion 2.

Dynamic testing and analysis procedures will be implemented to confirm that all Category I mechanical equipment will function during and after an earthquake of magnitude up to and including the SSE, and that all equipment support structures are adequately designed to withstand such seismic disturbances.

Subjecting the equipment and its supports to these dynamic testing and analysis procedures provides reasonable assurance that the Category I mechanical equipment as identified in the PSAR will continue to function during and after a seismic event, and the combined loading imposed on the equipment and its supports will not exceed applicable code allowable design stress and strain limits. Limiting the stresses of the supports under such loading combinations provides an acceptable basis for the design of the equipment supports to withstand the dynamic loads associated with seismic events and operational vibratory loading conditions without gross loss of structural integrity.

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

With regard to flow-induced vibrational testing of reactor internals for this facility, the applicant has stated that if one of the Bellefonte reactors is the first of the B&W 205 fuel assembly reactors to be ready for hot functional testing then it will be tested as a prototype reactor in accordance with Regulatory Guide 1.20 (Vibration Measurements on Reactor Internals, 12/71).

If the staff has accepted another 205 fuel assembly reactor as a satisfactory prototype before this facility is ready for hot functional testing, the applicant will perform additional confirmatory vibration testing and subsequent visual inspection on both Units 1 and 2 as part of the preoperational tests to provide added confirmation of the capability of the structural elements of the reactor internals to sustain flow-induced vibrations. The proposed program is consistent with Regulatory Guide 1.20.

We will review at the FSAR stage the preoperational vibration test program proposed by the applicant to verify the design adequacy of the reactor internals under loading conditions comparable to those experienced during operation. The combination of tests, predictive analysis, and post-test inspection will provide adequate assurance that the reactor internals can be expected to withstand flow-induced vibrations without loss of structural integrity during their service lifetime. The preoperational vibration test program will be performed in accordance with Regulatory Guide 1.20 and as such constitutes an acceptable basis for demonstrating the design adequacy of the reactor internals in satisfying the applicable requirements of General Design Criteria 2 and 14.

The applicant will perform a dynamic system analysis of the reactor internals and of the broken and unbroken piping loops. The dynamic system analysis will be performed to provide an acceptable basis for confirming the structural design adequacy of the reactor internals and the unbroken piping loops to withstand the combined dynamic effects of the postulated occurrence of a LOCA and a safe shutdown earthquake.

We have reviewed the analytical methods described in B&W Topical Report BAW-10008 (Part 1 - Reactor Internals Stress & Deflection Due to LOCA and Maximum Hypothetical Earthquake, 6/70). We find that an analysis using these methods will provide adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction as specified in Appendix F to the ASME Boiler and Pressure Vessel Code, Section III. In addition, the resulting deflections or displacements of any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling would be impaired.

The assurance of structural integrity of the reactor internals under the postulated SSE and the most severe LOCA conditions provides added confidence that the design can be expected to withstand a spectrum of lesser pipe breaks and seismic loading combinations.

We have concluded that the use of the proposed analytical techniques will result in an acceptable structural design for the Bellefonte 1 and 2 reactor internals.

3.10.2 ASME Code Class 2 and 3 Components

All Category I systems, components and equipment outside of the reactor coolant pressure boundary will be designed to sustain normal loads, anticipated transients, the Operating Basis Earthquake, and the Safe Shutdown Earthquake within design limits which are consistent with those outlined in Regulatory Guide 1.48 (Design Limits and Loading Conditions, 5/73). The specified design basis combinations of loading as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in Category I systems provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) other upset, emergency or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the system components are not expected to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity.

The applicant's design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components constitute an acceptable basis for design in satisfying General Design Criteria 1, 2 and 4.

The applicant will develop and conduct component test programs, supplemented by analytical predictive methods which will provide adequate assurance that ASME Code Class 2 and 3 active pumps and valves are capable of withstanding the imposed loads associated with Normal, Upset, Emergency and Faulted plant conditions without loss of structural integrity and of performing the "active" function (i.e. valve closure or opening) under conditions and combinations of conditions comparable to those expected when a safe plant shutdown is to be effected or the consequences of an accident are to be mitigated.

We have concluded that the program proposed by the applicant will provide reasonable assurance of pump and valve operability. In addition, the applicant, who is an active participant in current industry programs for codes for the design of pumps and valves, will remain cognizant of industry efforts to identify potential generic operability problems.

The design and installation criteria for ASME Class 2 pressure relief devices will be in accordance with the acceptable rules of Subsection NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III. The most severe discharge loads resulting from the opening of ASME Code Class safety relief valves will be calculated by either an equivalent static analysis or a time response dynamic analysis of the system. In the case of open safety or relief valves mounted on a common header and full discharge occurring concurrently, the additional stresses induced in the header will be combined with previously computed local and primary membrane stresses to obtain the maximum stress intensity. The criteria used in developing the design and mounting of the safety and relief valves of ASME Code Class 2 systems provides adequate assurance that, under discharging conditions, the resulting stresses are not expected to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function. The criteria used for the design and installation of overpressure relief devices in ASME Code Class 2 Systems constitute an acceptable design basis in meeting the applicable requirements of General Design Criteria 1, 2, 4, 14 and 15.

3.11 Seismic Qualification of Category I Instrumentation and Electrical Equipment

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

The applicant has proposed a seismic qualification program that will be implemented for Category I instrumentation and electrical equipment and the associated supports for this equipment to provide assurance that such equipment can be expected to function properly and that structural integrity of the supports will not be impaired during the excitation and vibratory forces imposed by the safe shutdown earthquake and the conditions of post-accident operation. The general program, as specified, constitutes an acceptable basis for satisfying staff requirements and the applicable requirements of General Design Criterion 2.

The applicant has referenced IEEE Standard 344, 1971 for seismic qualification of Category I electrical equipment and enclosure 5 (Electrical and Mechanical Equipment Seismic Qualification Program) to the staff letter to TVA dated September 14, 1973. Conformance with these documents will provide acceptable methods for seismic qualification.

A detailed presentation concerning the results of tests and analysis will be provided in the usual manner in the FSAR for evaluation during our review of the application for an operating license.

4.0 REACTOR

4.1 Summary Description

The reactor design for the Bellefonte plants will be geometrically similar to but larger than previously reviewed and approved B&W reactors including the North Anna facility. A significant difference is that the Bellefonte facility will use fuel assemblies with a 17 x 17 fuel rod array, whereas the North Anna and earlier core designs used 15 x 15 fuel rod assemblies. The proposed initial core power for each of the Bellefonte reactors is 3600 megawatts thermal, which is 27% higher than for the North Anna reactors. Bellefonte is presently considered to be the lead plant for the B&W 205-fuel-assembly class of reactor.

4.2 Mechanical Design

4.2.1 Fuel Mechanical Design

The proposed Bellefonte reactor fuel elements are to be provided by Babcock & Wilcox and will consist of Zircaloy-clad uranium dioxide fuel pellets. The fuel rod mechanical design is identical to that currently approved for use in North Anna Units 3 & 4, with the exception of those items listed in Table 4.1. All Bellefonte design items listed exhibit larger engineering safety margins compared to the approved North Anna Units 3 & 4 design.

All fuel rods will be internally prepressurized with helium during final welding to minimize cladding compressive stresses during service. The level of prepressurization is designed to preclude any cladding tensile stresses throughout operations due to total internal pres... .

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TABLE 4.1

FUEL MECHANICAL DESIGN COMPARISON
BETWEEN BELLEFONTE & NORTH ANNA UNITS 3 & 4

1.	Assemblys (No.)	Bellefonte (205)	North Anna (145)
	Rod Array	17 x 17	15 x 15
	Fuel Rods	264	208
	Rod-Rod Pitch	.501 inches	.568 inches
	Guide Tubes	24	16
2.	Fuel Rods		
	Outside Diameter	.379 inches	.430 inches
	Wall Thickness	.0235	.0265
	Average Specific Power	5.4 kw/ft	7.1 kw/ft
3.	Fuel Pellets		
	Diameter	.324	.370
	% Theoretical Density	94%	91%
	Stack Height	143 inches	144 inches
	Diametral Gap	.008 inches	.007 inches
	Max. Temp @ 100% Power	3760°F	4410°F

The staff requires that densification of uranium dioxide fuel pellets be assumed to occur during irradiation in power reactors. The initial density of the fuel pellets and the size, shape, and distribution of pores within the fuel pellet influence the densification phenomenon. The effects of densification on the fuel rod will increase the stored energy, increase the linear thermal output, increase the probability for local power spikes, and decrease the thermal conductance.

The primary effects of densification on the fuel rod mechanical design are manifested in calculations of time-to-collapse of the cladding and fuel-cladding gap conductance. Time-to-collapse calculations predict the time required for unsupported cladding to become dimensionally unstable and to flatten into an axial gap caused by fuel pellet densification. Gap conductance calculations predict the decrease in thermal conductance due to opening of the fuel-clad radial gap.

Babcock & Wilcox Topical Report BAW-10054 entitled, "Fuel Densification Report, May, 1973," is applicable to all B&W reactors beginning with Oconee Unit 1 and includes Bellefonte. The staff's review and acceptance with modifications of the B&W fuel densification model was presented in its report "Technical Report on Densification of Babcock & Wilcox Reactor Fuels," dated July 6, 1973. This model also applies to Bellefonte.

The fabrication of the Bellefonte fuel is not planned until late 1978. Thus, it is quite likely that the as-manufactured fuel will reflect significant improvements in design and manufacturing processes. The staff will remain cognizant of any B&W fuel design and manufacturing process changes in its continuing review of both standard and specific designs.

On the basis of our review of the current analytical models and their confirmatory test results we have concluded that the Bellefonte 17 x 17 fuel mechanical design provides for additional engineering safety margins compared to those provided in the approved design for North Anna Units 3 & 4.

4.2.2 Reactor Vessel Internals

We have reviewed the selection of materials for the reactor vessel internals required for reactor shutdown and components relied upon for adequate core cooling. All materials are compatible with the reactor coolant, and have performed satisfactorily in similar applications.

The use of materials proven to be satisfactory by actual service experience will provide reasonable assurance that the reactor vessel internals will not be susceptible to failure by chemical or stress corrosion cracking. Section 3.10.1 discusses the design, testing and performance of the reactor vessel internals for normal operation, seismic and LOCA conditions.

4.3 Nuclear Design

Our review of the nuclear design of the facility was based on the information provided by the applicant in the PSAR and revisions thereto, discussions with the applicant, and the results of independent calculations performed for us by the Brookhaven National Laboratory.

The proposed nuclear design of the Bellefonte reactors is the same as that reviewed and approved for the North Anna Units 3 and 4 reactors, except that Bellefonte will use additional fuel assemblies each with 17 x 17 fuel element array while most earlier B&W reactors were designed for a 15 x 15 fuel element array. The information available from the applicant concerning the 17 x 17 fuel assembly design indicates that the change in fuel design will improve overall reactor safety by lowering the average and maximum linear heat generation rate. For example, the original Bellefonte proposal used a 15 x 15 array which was designed to operate at 3414 MWt with an average linear power density of 6.49 kW/ft. The new 17 x 17 Bellefonte fuel assembly is designed for an average linear power density of 5.43 kW/ft at a thermal output of 3600 MWt.

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict the results of critical experiments using UO_2 and PuO_2 - UO_2 fuel.

We have concluded that the information presented adequately demonstrates the ability of these analyses to predict reactivity and the physics characteristics of the Bellefonte reactors.

Detailed three-dimensional power distribution measurements have been performed at the Babcock & Wilcox Critical Experiments Laboratory. The results of the applicant's calculations using PDQ07, a three-dimensional computer program, agree quite well with the measured power distribution. PDQ07 as used by B&W incorporates a thermal feedback in obtaining radial and axial power distributions for operations involving (1) changes in control rod positions, (2) various xenon stability and control conditions, and (3) various reactivity coefficients.

To allow for changes of reactivity due to reactor heatup, operating conditions, fuel burnup and fission product buildup, a significant amount of excess reactivity is built into the core. The applicant has provided sufficient information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to permit control of excess reactivity at all times. This will be done through the use of soluble boron in the reactor coolant, movable control rods, and fixed B_4C burnable poison rod assemblies (BPRA). The BPRA's will be used rather than increased soluble poison to prevent the beginning of life (BOL) moderator temperature coefficient from becoming more positive. The applicant has shown that sufficient control rod assembly (CRA) worth will be available to shut down the reactor with at least 1% $\Delta k/k$ subcritical margin in the hot condition at any time during the life cycle with the most reactive CRA stuck in the fully withdrawn position. Equipment will also be provided to add soluble boron to the reactor coolant to ensure a similar shutdown capability when the reactor is cooled to ambient temperatures. Control requirements for cycles beyond the first cycle will be established at the operating license stage as the design becomes more finalized.

On the basis of our review, we have concluded that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control rods, the soluble boron system, and the burnable poison rod assemblies to assure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available at the operating license stage.

The basic instrumentation for monitoring the nuclear power level and distribution in the Bellefonte reactors is the same in principle as for all PWR plants recently licensed for operation. Primarily reliance is placed on four axially split, out-of-core detectors. Also, 62 strings of self-powered incore neutron detectors are available for incore mapping. Each string can measure local neutron flux at seven elevations in the core. Test results showing that these incore detectors have a rated lifetime in excess of 5 years and a precision of $\pm$ 5% in determining relative power distribution are presented in B&W Topical Report 10001, "Incore Instrumentation Test Program," (August 1969).

We have concluded that the out-of-core detectors are adequate for detecting power maldistributions originating from axial xenon instability and misplaced control rods provided that the power distribution mapping capability of the incore detectors is used to calibrate the out-of-core detectors periodically and to investigate any power distribution anomalies detected by the out-of-core detectors.

We have reviewed the applicant's analyses of xenon-induced oscillations which are reported in three B&W Topical Reports BAW-10010, Part 1, "Stability Margin for Xenon Oscillations Model Analysis," August 1969, BAW-10010, Part 2, "Stability Margin for Xenon Oscillation - One Dimensional Digital Analysis," February 1970, and BAW-10010, Part 3, "Stability Margin for Xenon Oscillations - Two and Three Dimensional Analysis," April 1970. Those analyses indicated that while azimuthal and radial xenon oscillations will not be divergent, axial xenon oscillations could be divergent at the beginning of the fuel cycle. The analysis further indicated that axial xenon oscillations, which are slow changes taking place over several hours, can be controlled by having the reactor operator change the position of the eight part-length Axial Power Shaping Rods (APSR's). Results from induced axial xenon oscillation tests during the initial startup of the

Oconee Unit 1 have confirmed that good agreement exists between predicted and measured results and that APSR's are effective in damping axial oscillations.

4.4 Thermal-Hydraulic Design

The proposed Bellefonte reactors are each designed to operate at core power levels of up to 3600 MWt, which corresponds to a net electrical output of about 1200 MWe. We have reviewed the thermal-hydraulics on the basis of 3600 MWt. A comparison of the thermal and hydraulic design parameters for the Bellefonte and North Anna 3/4 plants is shown in Table 4.2.

The principal criterion for the thermal-hydraulic design of a reactor is to prevent fuel rod damage by providing adequate heat transfer for the various core heat generation patterns occurring during normal operation, operational transients, and accidents.

Maintenance of nucleate boiling is a basic objective of a thermal-hydraulic design. The applicant has demonstrated, through the use of the Babcock & Wilcox BAW-2 correlation, that a departure from nucleate boiling (DNB) heat flux can be avoided if the required DNB heat flux ratio (DNBR) greater than 1.32 is maintained for steady state and anticipated transient conditions. The values of minimum DNBR at design power and design overpower conditions, shown in table 4.2, are greater than the minimum DNBR design limit of 1.32; however, the hydraulics analysis was not based on vessel model flow tests which are completely applicable to the Bellefonte 205 fuel assembly

TABLE 4.2

Comparison of Thermal-Hydraulic Design Parameters
For the Bellefonte 1/2 and North Anna 3/4 Plants

	Bellefonte 1 & 2	North Anna 3 & 4
Reactor Core Heat Output (MWt)	3600	2631
System Pressure, Nominal (psia)	2250	2235
Minimum DNBR at Design Power	1.82	1.72
Minimum DNBR at Design Overpower	1.4	1.39
Minimum DNBR for Design Transients	>1.32	>1.32
Total Reactor Coolant Flow (10^6 lb/hr)	150.5	106.86
Effective Flow Rate for Heat Transfer (10^6 lb/hr)	142.4	103.0
Core Coolant Average Velocity	16.2	16.3
Average Mass Velocity (10^6 lb/hr-ft^2)	2.65	2.67
Coolant Temperature (°F)		
Design Nominal Inlet	572.5	566.3
Average Rise In Core	59.3	61.7
Total Heat Transfer Surface in Core (ft^2)	63,991	40,743
Average Heat Flux (BTU/hr-ft^2)	186,822	214,000
Maximum Heat Flux (BTU/hr-ft^2)	507,044	582,000
Maximum Thermal Output (kW/ft)	14.74	19.2
Maximum Thermal Output at Overpower (kW/ft)	16.51	21.5
Maximum Fuel Central Temperature (°F)		
100%	3760	4410
at 112% Overpower	4470	4720

design. Since the core inlet flow distribution is dependent on flow conditions in the inlet annulus and thermal shield regions, the 205 fuel assembly vessel model flow tests, applicable to the design of Bellefonte, will be reviewed at the operating license stage to confirm the acceptability of the thermal-hydraulics calculations. The applicant has indicated that the core flow distribution tests for the 205 fuel assembly plants are scheduled for completion in 1974.

The core power level and the peak linear power density of a PWR are controlling factors in the evaluation of various transients and accidents. For the Bellefonte reactors, the core power level used for the safety evaluation was 3600 MWt and the linear power density used was 14.74 kw/ft (3760 MWt and 15.36 kw/ft for LOCA analysis). With this assumed core power and linear power density, this facility complies with existing criteria. The maximum linear power density permitted during steady state operation and the maximum linear power density permitted to occur for certain plant operating maneuvers will be determined during the operating license review and will be required to be consistent with the criteria in effect at that time.

Preservation of nucleate boiling as the mode of heat transfer between the fuel cladding hot spot and coolant not only assures that the cladding temperature is only slightly greater than that of the coolant, but that the fuel centerline temperature will not reach the melting temperature. The applicant's criteria for overpower protection requires that the maximum fuel centerline temperature be less than that of the fuel melting temperature at a peak core power generation rate of 16.51 kW/ft during all modes of operation. In fulfillment of this objective, the applicant has calculated a fuel centerline temperature of 4470°F at 16.51 kW/ft compared to a fuel melt temperature of 5080°F at beginning of life which reduces linearly with burnup to 4800°F after 43,000 MWd/mtU.

On the basis of our review of the analytical techniques applied to the previously reviewed and approved 15 x 15 core designs, we have concluded that for the 17 x 17 core design, there is reasonable assurance (1) that the proposed thermal-hydraulic design accounts for DNB and fuel center line temperature limitation in a satisfactory manner, and (2) that the conservatism in the thermal-hydraulic design procedures can be verified. In the event that sufficient verification cannot be obtained from the test programs or that the analytical methods are not conservative, appropriate restrictions on operations can be established at the operating license stage.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The Reactor Coolant System (RCS) for Bellefonte will consist of two similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop will contain two reactor coolant pumps, and a once-through steam generator. In addition, the system will be provided with a pressurizer, a reactor coolant drain tank (pressurizer relief tank), interconnecting piping and sensing instrumentation necessary for operational control. All the above components will be located in the containment building.

During operation, the RCS will transfer the heat generated in the core to the steam generators where steam will be produced to drive the turbine generator. Borated demineralized water will be circulated in the RCS at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The water will also act as a neutron moderator and reflector, and as a vehicle for the neutron absorbing Boron to be used in reactivity control.

The reactor coolant system pressure boundary provides a second barrier against the release of radioactivity generated within the reactor (the fuel rod cladding is the primary barrier), and is designed to ensure a high degree of integrity throughout the life of the facility.

The RCS pressure changes caused by load transients will be controlled by the use of the pressurizer with water and steam maintained in equilibrium by electrical heaters or water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring-loaded safety valves and power operated relief valves are mounted on the pressurizer and discharge to the reactor coolant drain tank, where the steam is condensed and cooled by mixing with water.

The system concept is the same as reviewed and approved for North Anna Units 3 and 4.

5.2 Integrity of Reactor Coolant Pressure Boundary (RCPB)

The ASME Section III Code Class I components within the RCPB will be designed, fabricated, and inspected in accordance with the requirements of the applicable codes delineated in Section 3.2.2, System Quality Group Classifications and Table 5.2-1 of the PSAR. The applicable codes, code editions and addenda will comply with the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards.

No ASME Code Cases which are identified as unacceptable by the Commission will be applied in the construction of pressure retaining components within the RCPB. The applicant has specified those Code Cases that will be applied in the construction of ASME Section III Code Class 1 components. Compliance with the requirements of these Code Cases is expected to result in component quality level consistent with the acceptable level intended by the requirements of GDC 1.

5.2.1 <u>Design of Reactor Coolant Pressure Boundary Components</u>

The design loading combinations specified for ASME Code Class 1 RCPB components have been appropriately categorized with respect to the plant condition identified as Normal, Upset, Emergency or Faulted. The design limits proposed by the applicant for these plant conditions are consistent with the criteria recommended in Regulatory Guide 1.48 (Design Limits and Loading Combinations for Seismic Category I Fluid System Components, May 1973). Use of the criteria recommended in Regulatory Guide 1.48 for the design of the RCPB components will provide reasonable assurance that, (1) in the event an earthquake should occur at the site, or (2) other system upset, emergency or faulted conditions should develop, the resulting combined stresses imposed on the system components will not exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1 components constitute an acceptable basis for design in satisfying the related requirements of General Design Criteria 1, 2 and 4.

The applicant has identified the active valves within the RCPB whose operation is relied upon to safely shutdown the plant and maintain it in a safe condition in the unlikely event of a safe shutdown earthquake or a design basis accident. The applicant has also stated that component operability test programs supplemented by analytical methods, will be developed to provide additional assurance that the capability of these active components will, (1) withstand the imposed loads associated with Normal, Upset, Emergency and Faulted plant conditions without loss of structural integrity and (2) perform their "active" function (i.e. valve closure or opening), under conditions and combinations of conditions comparable to those expected when a safe plant shutdown is to be effected or the consequences of an accident are to be mitigated.

We have concluded that the program proposed by the applicants is acceptable and will provide reasonable assurance of valve operability.

The design and installation criteria for pressure relief devices on the RCPB will be in accordance with the acceptable rules of Subsection NB-3500 of the ASME Boiler and Pressure Vessel Code, Section III. The maximum full discharge loads resulting from the opening of ASME Code Class 1 safety and relief valves will be calculated by either an equivalent static analysis or a time response dynamic analysis of the system.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Code Class 1 systems provides adequate assurance that, under discharging conditions, the resulting stresses will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function. The criteria used for the design and installation of overpressure relief devices in ASME Code Class 1 Systems constitute an acceptable design basis applicable requirements of General Design Criteria 1, 2, 4, 14 and 15.

5.2.2 Overpressurization Protection

Overpressurization protection in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000 (1971) is provided by pressure relief of the RCS from two pressurizer safety valves. One electrically actuated relief valve mounted on nozzles on the pressurizer is also provided. The pressurizer safety valves discharge through a common header to the reactor coolant drain tank. The pressurizer safety valves are sized on the basis of the maximum pressure transient imposed on the RCS resulting from complete loss of main feedwater flow. The applicant's description of RCS overpressure protection references B&W Topical Report BAW-10043, Supplement 1. At present, this report is under review by the Staff and determination as to the adequacy of the overpressure protection for Bellefonte will be made as a part of the overall Anticipated Transients without Scram (ATWS) review (Section 15.3). The applicant has agreed to supply an analysis as required in the staff's report, WASH-1270, by October 1, 1974. The staff will evaluate this matter to the requirements of the licensing position in Appendix A to WASH-1270.

5.2.3 General Material Considerations

We have reviewed the materials of construction for the RCPB to ensure that the possibility of serious chemical or stress corrosion is minimized. All materials used are compatible with the expected environment, as proven by extensive testing and satisfactory service performance. The applicant has shown that the possibility of intergranular stress corrosion in austenitic stainless steel used for components of the reactor coolant pressure boundary will be minimized because sensitization will be avoided, and adequate precautions will be taken to prevent contamination during manufacture, shipping, storage, and construction. The plans to avoid sensitization are in conformance with Regulatory Guide 1.44, (Control of the Use of Sensitized Stainless Steel, May 1973) except for approved differences and include controls on heat treatments and welding processes.

The use of materials with satisfactory service experience, and the controls placed on list treatment and welding procedures used provide reasonable assurance that austenitic stainless steel components will be compatible with the expected service environments, and the probability of loss of structural integrity is minimized.

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

We have evaluated the proposed requirements for the external insulation used on austenitic stainless steel components, and conclude that the reflective metal insulation used will not lead to deterioration of the stainless steel when exposed to containment sprays.

The possibility that serious chemical or stress corrosion problems would occur in the unlikely event that ECCS or containment spray system are activated will be minimized because of the pH of the circulating coolant will be maintained above 7.0 by hydroxide additions.

The applicant has stated that the secondary water chemistry will be controlled using full flow demineralization of the condensate to prevent stress corrosion or wastage of the steam generator tubing, and that the adequacy of the compositional limits proposed has also been demonstrated by laboratory tests and service experience.

We conclude that the controls on chemical composition that will be imposed on the reactor coolant, secondary water, emergency core cooling water, and the use of all metal reflective external thermal insulation provide reasonable assurance that the reactor coolant boundary materials will be adequately protected from conditions that would lead to loss of integrity from chemical or stress corrosion.

5.2.4 Fracture Toughness

• We have reviewed the materials selection, toughness requirements, and extent of materials testing proposed by the applicant to provide assurance that the ferritic materials used for pressure retaining components of the RCPB will have adequate toughness under test, normal operation, and transient conditions. All ferritic materials will meet the toughness requirements of the ASME Boiler and Pressure Vessel Code, Section III (1971 Edition). In addition, materials for the reactor vessel will meet the additional testing and acceptance criteria of the Summer 1972 Addenda, and Appendix G, 10 CFR 50.

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

The applicant states that the reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure, in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code, Summer 1972 Addenda, and Appendix G, 10 CFR 50. The applicant also states that the method of determining pressure-temperature limitations will be described in a forthcoming B&W topical report. We expect to review this report or such other equivalent information as TVA provides to describe the method used to determine that the pressure-temperature limitations will be in conformance with the provisions of Appendix G, 10 CFR 50. We find this commitment acceptable for the construction permit stage of review. Conservatism will be required in the pressure-temperature limits used for heatup, cooldown, testing, and core operation because the applicant must assume that the beltline region of the reactor vessel has already been irradiated.

The use of Appendix G of the Code as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the Code and AEC Regulations, will ensure adequate safety margins during operating, testing maintenance, and postulated accident conditions. Compliance with these Code provisions and AEC regulations, constitute an acceptable basis for satisfying the requirements of General Design Criterion 31 at the construction permit stage of our review. As an additional requirement the toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program that will meet all the requirements of ASTM E 185-73 and Appendix H, 10 CFR 50 (July 17, 1973). The composition of reactor vessel beltline material, including welds, will be controlled during fabrication to minimize the copper and phosphorus content, thus ensuring thathe sensitivity to radiation damage will be low, but the number of capsules provided in the surveillance program is conservatively based on assuming high values of sensitivity.

Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation will be assessed properly, and assurance of adequate safety margins against the possibility of vessel failure can be verified through implementation of the material surveillance requirements of ASTM E 185-73 and Appendix H, 10 CFR Part 50. Compliance with these documents will ensure that the surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of General Design Criterion 31.

Although the use of controlled composition material for the reactor vessel beltline will minimize the possibility that radiation will cause serious degradation of the toughness properties, the applicant has stated that should results of tests indicate that the toughness is not adequate, the reactor vessel can be annealed to restore the toughness to acceptable levels. We conclude that adequate measures have been or will be taken to assure acceptable fracture toughness of the RCPB.

5.2.5 Control of Stainless Steel Welding

We have reviewed the controls proposed to prevent hot cracking (fissuring) of austenitic steel welds. These precautions include control of weld metal composition and welding processes to ensure adequate delta ferrite content in the weld metal. The proposed methods comply with Section III of the ASME Code, and are in conformance with Regualtory Guide 1.31, (Control of Stainless Steel Welding, 6/73) except for approved differences. The use of materials processes, and test methods that are in accordance with these requirements and recommendations will provide reasonable assurance that loss of integrity of austenitic stainless steel welds caused by hot cracking during welding will not occur.

5.2.6 Pump Flywheel

The probability of a loss of pump flywheel integrity can be minimized by the use of suitable material, adequate design, and inservice inspection. We have evaluated the integrity of the reactor coolant pump flywheel and have concluded that its integrity is provided by conformance with Regulatory Guide 1.14, (Reactor Coolant Pump Flywheel Integrity 10/71).

The use of suitable material, and adequate design and inservice inspection for the flywheels of reactor coolant pump motors as specified in the PSAR provides reasonable assurance (a) that the structural integrity of flywheels is adequate to withstand the forces imposed in the event of design overspeed transients without loss of their function, and (b) that their integrity will be verified periodically in service to assure that the required level of soundness of the flywheel material is adequate to preclude failure. Compliance with the recommendations of Regulatory Guide 1.14 constitutes an acceptable basis for satisfying the requirements of General Design Criterion 4 (See also Section 5.4).

5.2.7 Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically. The applicant has stated that the design of the reactor coolant system incorporates provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that a tool will be developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

The conduct of periodic inspections and hydrostatic testing of pressure-retaining components in the reactor coolant pressure boundary in accordance with the requirements of ASME Section XI Code provides reasonable assurance that loss of structural or leaktight-integrity occurring during service will be detected in time to permit corrective action before the required safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of General Design Criterion 32.

5.2.8 RCPB Leakage Detection System

Coolant leakage within the containment could be an indication, among other things, of a small throughwall flaw in the RCPB.

Leakage detection systems are proposed for leakage to the containment which will (1) include diverse leak detection methods, (2) will have sufficient sensitivity to measure small leaks, (3) will identify the leakage source to the extent practical, and (4) will be provided with suitable control room alarms and readouts. One system detects changes in containment activity (gaseous iodine and particulate), another monitors changes in containment sump liquid level while another measures vapor condensation. Indirect indications of leakage will also be obtained from the increase in containment humidity measured by pressure and temperature indicators. The leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary are in accordance with the positions of Regulatory Guide 1.45 (Reactor Coolant Pressure Boundary Leakage Detection Systems, 5/73) and provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. We conclude that RCPB leakage systems designed in accordance with the positions of Regulatory Guide 1.45 constitute an acceptable basis for satisfying the requirements of General Design Criterion 30.

5.3 Reactor Vessel Integrity

We have reviewed all factors contributing to the structural integrity of the reactor vessel and we conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for this facility.

The bases for our conclusion are that the design, material, fabrication, inspection and quality assurance requirements will conform to the rules of the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, all addenda through Summer 1972, and all applicable Code Cases.

The stringent fracture toughness requirements of the ASME Code, Section III, 1971 Edition, and the 1972 Summer Addenda will be met. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer

Addenda of the ASME Boiler and Pressure Vessel Code, Section III, and Appendix G, 10 CFR 50.

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

1. Will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent Code Cases.
2. Will be made from materials of controlled and demonstrated high quality.
3. Will be subjected to extensive inspection and testing to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.
4. Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation, and that the vessel will not fail under the conditions of any of the postulated accidents.
5. Will be subjected to monitoring and periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.
6. May be annealed to restore the material toughness properties if this becomes necessary.

5.4 Component and System Design

The Reactor Coolant Pump is designed to provide adequate core cooling flow and hence sufficient heat transfer to maintain a DNBR<1.32, within the parameters of operation.

Sufficient pump rotational inertia is provided by the flywheel to promote continued flow following a loss of forced flow resulting from mechanical or power failures to the pumps such that the reactor neutron power can be reduced before DNBR limits are exceeded.

A pump overspeed evaluation has not been submitted to the staff and the applicant has indicated that these studies, as well as the results of an investigation of overspeed protection devices, will be reported for our evaluation of the operating license application.

Analyses have been performed to hypothesize the effects and consequences of the loss of forced reactor coolant flow accident discussed in Section 15.

The steam generator is a vertical straight-tube-and-shell heat exchanger and produces superheated steam at constant turbine throttle pressure over the operating power range. The reactor coolant enters the steam generator upper hemispherical head, flows downward inside the tubes giving up heat to generate steam on the shell side secondary loop. The steam generator tubes and tubesheets as part of the RCPB are designed to withstand RCS design pressure and temperature to minimize the transfer of activity generated within the core to the secondary system. The steam generators provide a heat sink for the reactor coolant system. They are at a higher evaluation than the core to improve natural circulation for decay heat removal.

The reactor coolant piping is designed and fabricated to accomodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions.

The Decay Heat Removal System (DHRS), is designed to remove decay heat and sensible heat from the RCS and core during the latter, low pressure stages of cooldown. The system also provides auxiliary spray to the pressurizer for complete depressurization, maintaining the reactor coolant temperature during refueling, and provides the means for filling and draining the refueling cavity. In the event of a LOCA, the DHRS serves a part of the Emergency Core Cooling System by providing low pressure injection of borated water into the reactor vessel for emergency core cooling.

The DHRS is placed into operation approximately 6 hours after initiation of plant shutdown when the temperature and pressure of the RCS are below 305°F and 600 psig, respectively. Assuming that two pumps and coolers are in service, and that each cooler is supplied with component cooling water at design flow and temperature, the DHRS is designed to reduce the RCS temperature to 140°F within 14 hours.

The DHRS is provided with two DHR pumps and two DHR coolers arranged in redundant and independent flow paths. If one of the two pumps or one of the two coolers is not operable, safe cooldown of the plant is not compromised. The use of the DHRS as part of the ECCS is described in Section 6.3

The applicant has indicated that redundant instrumentation will be added to the DHR isolation system to detect check valve leakage and to prevent overpressurization of the DHRS. Details of this additional safety feature will be included in the FSAR. We find this commitment acceptable and will require it to be documented prior to issuance of a construction permit.

The two pressurizer safety valves are bellows sealed, balanced, spring-loaded safety valves equipped with a supplemental back pressure balancing piston for handling a bellows failure. The remaining pilot operated pressurizer relief valve is electrically actuated.

The combined capacity of the pressurizer safety valves is 10^6 lb/hr, and based on accepting the maximum surge resulting from a complete loss of main feedwater flow. This objective is met without reactor trip or any operator action provided that the steam safety valves open as designed when the steam pressure reaches the steam-side safety setpoints. The pressurizer safety valves prevent the reactor coolant system pressure from exceeding 110% of system design pressure.

The reactor internals vent valves are located on a common plane in the upper core support weldment above the outlet nozzles. These valves provide a direct flow path between the upper core region and inlet annulus in the event of a loss of coolant accident from an inlet line break. This flow path provides for pressure equalization by the venting of steam to the break and permits the emergency coolant water to reflood the core rapidly. There are 8 of these vent valves, each with an effective flow diameter of 14 inches. The seating face of each valve disc is inclined 5 degrees to the vertical to insure a positive seal (the differential pressure acting across the valve acts to seal it also). The vent valve design is essentially the same as used for the Oconee design except the exercising hook is a different shape and the valve is mounted by capture bolts through a flange.

In the thermal-hydraulic analysis of the Bellefonte plant for normal operation, the applicant assumed that there was no core bypass flow resulting from an open vent valve. At present, there is adequate instrumentation to detect the system flow change (approximately 5% reduction in core flow), which would result from an open valve. The staff position has not changed from that taken on the Oconee plant. Therefore, the staff requires that one valve less than the minimum detectable number of stuck open vent valves by assumed open and the corresponding core flow penalty be imposed for the thermal-hydraulic design of the RCS and core. This analysis should be incorporated into the FSAR and will be evaluated to determine the allowable operating characteristics of the system. Changes to this staff position will be considered as experience from operating plants for which vent valves have been installed becomes available.

We conclude, with the conditions as noted above, that the proposed reactor coolant system, subsystems and component designs are acceptable.

5.5 Loose Parts Monitor

Occasionally, miscellaneous items such as nuts, bolts, etc., have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For such reasons, the staff has encouraged applicants over the past several years to support programs designed to develop effective, on-line loose parts monitoring. For the past few years we have required each applicant to initiate a program, or to participate in an ongoing program, the objective of which is the development of a functional, loose parts monitoring system within a reasonable period of time.

As a result of such programs several prototype loose parts monitoring systems have been developed and are presently in operation or being installed at several plants. All recently approved plants utilizing a Babcock & Wilcox nuclear steam supply system have installed such systems.

The staff requested that such a system be installed on the Bellefonte reactors and the applicant has indicated that such an online monitoring system will be installed. The detailed description of the specific system will be reported at the operating license stage. We find this commitment acceptable and will require that it be documented prior to issuance of a construction permit.

6.0 ENGINEERED SAFETY FEATURES

6.1 Design Considerations

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

We have reviewed the proposed systems and components designated as engineered safety features. These systems and components will be designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15 of this report. They will be designed, therefore, to seismic Category I standards and must function even with complete loss of offsite power.

Components and systems will be provided in sufficient redundancy so that a single failure of any component or system will not result in loss of the capability to achieve safe shutdown of the reactor. These design requirements are in accordance with the AEC General Design Criteria of 10 CFR Part 50.

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We have reviewed the materials selection proposed for the containment heat removal and ECCS systems. The applicant has stated that the use of sensitized stainless steel will be avoided, and that the pH of the containment spray and the circulating coolant will be compatible with the reactor materials.

We have concluded that the controls on material and cooling water chemistry proposed will provide assurance that the integrity of components of these systems will not be impaired by chemical or stress corrosion .

The applicant has stated that welding of austenitic stainless steel for components of these systems will be controlled to prevent deleterious hot cracking. The proposed control of weld metal composition and welding procedures are in general conformance with the recommendations of Regulatory Guide 1.31 (Control of Stainless Steel Welding, 6/73) and will provide assurance that loss of function will not result from hot cracking of welds.

6.2 Containment Systems

6.2.1 Containment Functional Design

The containment system for the Bellefonte Nuclear Plant, Units 1 and 2 includes a dual reactor containment structure, containment heat removal systems, containment isolation systems, a combustible gas control system and a secondary containment air cleanup system.

The primary containment is a steel-lined, prestressed concrete structure with net free volume of 3,400,000 cubic feet. The containment structure houses the nuclear steam supply system, including the reactor, steam generator, reactor coolant pump and pressurizer, as well as certain components of the plant's engineered safety feature system. The containment is designed for an internal pressure of 50 psig and a temperature of 275°F.

The secondary containment building encloses the primary containment. The annulus formed by the secondary containment and the primary containment building is maintained at negative pressure conditions under both normal and accident conditions. The secondary containment system incorporates a cleanup system to provide for the cleanup and controlled release to the environment of fission product leakage from the primary containment following a postulated accident.

The applicant has described in the PSAR the methods and results used to analyze the containment pressure response for a number of design basis loss-of-coolant accidents (LOCA's). Various break locations and sizes were evaluated to determine that the double-ended pipe rupture at the pump suction of the reactor coolant system results in the highest containment pressure. Minimum containment cooling, assumed in the analysis, included one reactor building spray train and one reactor building fan cooler.

The applicant has analyzed the containment pressure response from postulated loss-of-coolant accidents in the following manner. The B&W CRAFT code was used to calculate the mass and energy release to the containment during the blowdown, core reflood, and post-reflood phases on the accident. The mass and energy addition rates calculated in this manner were then used as input by the CONTEMPT computer program, which is used by the applicant to calculate the containment pressure response.

As described above, the CRAFT program was used to calculate blowdown mass and energy releases. The blowdown phase of the accident is the phase during which most of the energy contained in the reactor coolant system including the coolant or water, metal and core stored energy is released to the containment. To obtain a conservatively high energy release rate to the containment the applicant assumed nucleate boiling in the core until the quality of the coolant was approximately 1.0. In addition, the analysis maximized the energy release to the containment by assuming full ECCS operation and neglecting the quenching action of the incoming ECCS fluid on the exiting steam.

The CRAFT program was also used by the applicant to predict mass and energy release to the containment during the core reflood phase of the accident. The analysis of the reflood phase of the accident is important with regard to pipe ruptures of the reactor coolant systems cold legs since the amount of steam and entrained liquid carried out of the core for these break locations passes through the steam generators can be evaporated and/or superheated to the temperature of the steam generator secondary fluid. During core reflood the carryout rate fraction which determines the amount of steam and entrained water leaving the core and therefore the amount of energy that can be transferred from the steam generator is calculated based on a correlation inherent in CRAFT. The CRAFT reflood calculations for the design basis accident included average carryout rate fractions in excess of 0.8. Results of the applicable FLECHT experiments indicate that the carryout fraction of fluid leaving the core during reflood is about 0.8 of the incoming flow to the core which confirms the assumptions of CRAFT. The rate of energy release to the containment during this phase is proportional to the flow rate into the core, and thus through the steam generators. The applicant has also presented the results of an analysis of the mass and energy release during this phase of the accident assuming quenching of the exit steam by ECCS injection flow and in addition without assuming quenching of the exit steam to the containment. Because the applicant has not, however, provided sufficient justification to demonstrate the extent of steam quenching assumed, we have based our review on the conservative assumption that no steam is quenched by ECCS water.

After the core has been completely covered with water, decay heat generation will produce boiling in the core and a two-phase mixture of steam and water will exist in the core. This mixture can enter the steam generators and provide an additional energy release to the containment. The applicant's analytical model accounts for this additional frothing energy. The calculated containment peak pressure as determined by the applicant, occurs at 200 seconds. At 500 seconds after the accident essentially all of the available sensible heat has been removed from the reactor coolant system and the steam generators.

The CRAFT computer program has been accepted by the staff for calculating mass and energy releases to the containment during the blowdown phase of the postulated accident. In addition, we have compared the CRAFT calculations of mass and energy release to the containment during the reflood and post-reflood phase of the accident to our FLOOD-2 calculations. This comparison indicates good agreement between CRAFT and FLOOD-2. We therefore conclude that the applicant's methods for calculating the mass and energy releases to the containment are reasonably conservative and acceptable if the quenching action of the ECCS fluid on the exiting steam is neglected following blowdown.

Using the analytical methods described above the applicant calculates a peak containment pressure of 43.5 psig for the cold leg pump suction double-ended rupture (equivalent area of 11.2 ft^2).

We have performed confirmatory containment pressure response calculations for the design basis accident using our CONTEMPT computer code. We calculate a peak containment pressure essentially the same as the applicant's (i.e., about 43 psi).

The applicant has also analyzed the containment pressure response to a postulated main steam line failure. The applicant has conservatively assumed that the energy associated with this accident is instantaneously released to the reactor building and has not taken credit for static heat sinks or reactor building cooling. The applicant calculates a peak containment pressure of 23 psig for this accident.

The applicant has analyzed the pressure response within the containment interior compartments, such as the reactor vessel cavity, the steam generator compartments and the primary shield pipe penetration annulus. The applicant used the CRAFT computer program to calculate the peak compartment pressure differentials. Consistent with our current practice the applicant has set compartment design pressure differentials using a 40% margin between the maximum differential pressures calculated by conservative methods and design values used in the structural loading equations. The applicant calculated a limiting design pressure differential of 200 psi for a single-ended rupture of the hot leg and 16.1 psi for a double-ended rupture of the hot leg in the reactor cavity and steam generator compartments, respectively. The design pressure of the pipe annulus was set at 2200 psia, which corresponds to the primary coolant pressure. We performed confirmatory analyses using the RELAP-3 computer program and predicted pressures in good agreement with the applicant's results.

We have evaluated the containment system functional design in accordance with the General Design Criteria stated in 10 CFR Part 50 and in particular, Criteria 16 and 50. We conclude that the applicant's containment design pressure of 50 psig provides adequate margin (about 15%) when compared to the maximum calculated containment pressure of 43.5 psig and is therefore acceptable. In addition, based on our review of the applicant's CRAFT subcompartment model, our confirmatory calculations and the 40% margin specified for the subcompartment design pressure differentials we find the subcompartment design pressure differentials acceptable. We therefore conclude that containment functional design meets the requirements of General Design Criteria 16 and 50 and is acceptable.

6.2.2 Containment Heat Removal Systems

The Reactor Building Spray System (RBSS) and the Reactor Building Cooling System (RBCS) are provided to remove heat from the containment following a loss-of-coolant accident. Any of the following combinations of equipment will provide adequate heat removal capability:

a. both spray trains of the RBSS,

b. two fan-cooler units of the RBCS, and

c. one spray train of the RBSS and one fan-cooler unit of the RBCS.

The RBSS serves as an engineered safety feature. It is a seismic Category I system consisting of redundant piping, valves, pumps and spray headers. All active components of the RBSS are located outside of the reactor building. Missile protection is provided by direct shielding or physical separation of equipment. The reactor building sump screen assembly is designed to prevent debris from entering the spray system that could clog the spray nozzles.

A high reactor building pressure signal from the engineered safety features actuation system will automatically place the RBSS in operation. The spray pumps will initially take suction from the borated water storage tank. When the water in the tank reaches a low level, the spray pump suction is manually transferred to the reactor building sump.

The Reactor Building Cooling System (RBCS) is used during both normal and accident conditions. Three equal capacity fan-coolers are provided. Each unit contains a cooling coil and a two-speed fan. Under post-accident emergency cooling conditions, upon receipt of a reactor building high pressure engineered safety feature signal the fans transfer from high to low speed operation. Cooling water flow is supplied by the essential raw water cooling system.

The Reactor Building Cooling System (RBCS) is a seismic Category I system. The RBCS units will be located inside the reactor building but outside the secondary shield, at an elevation above the water level in the bottom of the reactor building during post-LOCA conditions. This location protects the units from missiles and from flooding. We have reviewed the containment heat removal and Regulatory Guide 1.1 (Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps, 11/70) and we find them to be acceptable.

6.2.3 Containment Air Purification and Cleanup Systems

There are two engineered safety feature systems that provide a containment air purification and cleanup capability. These are the Reactor Building Spray System which reduces airborne iodine inside the primary containment, and the secondary containment air cleanup system which reduces the levels of airborne particulates and iodine in the secondary containment annulus. The Reactor Building Spray System was discussed in Section 6.2.2.

During normal power operation the annulus is maintained at approximately 1 inch negative water gauge pressure. In the post-LOCA condition the secondary containment cleanup system serves two functions. First, it maintains the secondary containment at a negative pressure relative to atmosphere to collect essentially all primary containment leakage. Second, it processes and recirculates the secondary containment atmosphere. The system consists of two separate redundant systems, either of which is capable of maintaining the pressure inside the annulus to a negative 0.8 inch water gauge and providing filtration of the air inside the annulus while discharging to the environment through filters, sufficient air to maintain the negative pressure differential. The system will be designed to seismic Category I criteria and each subsystem is provided with a separate source of emergency power. Each filter train consists of a demister, heater assembly, HEPA prefilters, two banks of carbon adsorber trays and a final bank of HEPA filters. The trains are started automatically following an accident signal.

The applicant has performed analyses to demonstrate that the annulus pressure will be maintained at a negative 0.8 inches water gauge following a LOCA. We have reviewed the applicant's analysis of the annulus pressure response and find it acceptable. The applicant will perform a series of preoperational tests to confirm the containment leakage and the performance of the filtration systems for the secondary containment. We will review the results of this testing program at the operating license review stage and will require periodic inservice inspection tests as part of the Technical Specifications.Based on our review of the proposed design and the predicted performance of the air purification and cleanup systems, we concludthat these systems will meet the intent of Regulatory Guide 1.52, General Design Criteria 41, 42, 43 and 64, and are acceptable.

There are four other engineered safety feature air cleanup systems at the Bellefonte Nuclear Plant. These are:

a. Main control room air cleanup units.

b. Auxiliary Building Fuel Handling Area Exhaust air cleanup units.

c. Auxiliary Building Unit 1 mechanical equipment zone exhaust air cleanup units.

d. Auxiliary Building Unit 2 mechanical equipment zone exhaust air cleanup units.

The staff has analyzed the engineered safety feature filtration systems designated by the applicant to operate in emergency situations with respect to the positions in Regulatory Guide 1.52 (Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, 6/73). We find the applicant's design in agreement with these positions. We have used an adsorption efficiency of 95% for iodine removal in our accident consequence computations (see Section 15).

6.2.4 Containment Isolation Systems

The containment isolation system will be designed to automatically isolate piping systems that penetrate the containment to prevent out-leakage of the containment atmosphere following a loss-of-coolant accident. Double barrier protection, in the form of closed systems and isolation valves, will be provided to assure than no single active failure will result in the loss of containment integrity. Containment penetration piping, including the isolation will automatically occur on a high reactor building pressure of 4 psig. All fluid penetrations not required for operation of the engineered safety features equipment will be isolated.

We have reviewed the containment isolation system for conformance to General Design Criteria 54, 55, 56 and 57 and conclude that the system design will meet the requirements of the General Design Criteria and is therefore acceptable.

6.2.5 Combustible Gas Control Systems

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment. The major sources of hydrogen generation include: (1) a chemical reaction between the fuel rod cladding and the steam resulting from vaporization of emergency core cooling water, (2) corrosion of aluminum by the spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and the containment sump.

The applicant's analysis of post-LOCA hydrogen generation following a loss-of-coolant accident is consistent with the guidelines of Regulatory Guide 1.7, (Control of Combustible Gas Concentrations, 3/71) and indicates that the concentration in the containment would not reach the lower flammability limits of 4.0 v/o until about 20 days after the accident even assuming no recombiner operation.

Two 100% capacity electric recombiners each with a capacity of 100 cfm, will be located inside containment for post-accident hydrogen control.

The proposed recombiner system incorporates several design features that are intended to assure the capability of the system to be operable in the event of an accident. Among these are: (1) seismic Category I design, (2) protection from missile and jet impingement and (3) redundancy to the extent that no single component failure disables both recombiners. A post-accident purge system will be provided in addition to the recombiners to serve as a backup to the redundant hydrogen recombiner units.

The applicant calculates that the hydrogen concentration will be limited to 3.0 v/o with operation of a single recombiner three days following an accident. We have performed a similar analysis of hydrogen generation and hydrogen concentration in the containment following a loss-of-coolant accident and our results are in agreement with the applicant's.

Redundant hydrogen sampling trains, outside the containment will be provided to allow periodic sampling and analysis of the hydrogen concentration in the containment.

Based on our review of the systems to be provided for combustible gas control following a postulated loss-of-coolant accident, we conclude that the systems will conform to the guidelines of Regulatory Guide 1.7, meet the requirements of General Design Criteria 41, 42 and 43 and are therefore acceptable.

6.2.6 Leakage Testing Program

The containment design includes the provisions and features planned which satisfy the testing requirements of Appendix J, 10 CFR Part 50. The design of the containment penetrations and isolation valves permits individual periodic leakage rate testing at the pressure specified in Appendix J, 10 CFR Part 50. Included are those penetrations that have resilient seals and expansion bellow; i.e., airlocks, emergency hatches, refueling tube blind flanges, hot process line penetrations, and electrical penetrations.

The proposed reactor containment leakage testing progran complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that containment leak-tight integrity can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limits.

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

6.3 Emergency Core Cooling System (ECCS)

6.3.1 Design Bases

The applicant has stated that the ECCS will be designed to provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the reactor coolant system piping resulting in loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The ECCS is also designed to protect against steam line break consequences.

The applicant's design bases are to prevent fuel and cladding damage that would interfere with adequate emergency core cooling and to mitigate the amount of clad-water reaction for any size break up to and including a double ended rupture of the largest primary coolant lines. The applicant states that these requirements will be met even with minimum engineered safeguards available, such as would occur with the loss of one emergency power bus together with the unavailability of offsite power.

The ECCS subsystems to be provided are of such diversity, reliability and redundance that no single failure of ECCS equipment, occurring during a LOCA, will result in inadequate cooling of the reactor core. Each of the ECCS subsystems are to be designed to function over a specific range of reactor coolant piping system break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe.

6.3.2 System Design

The ECCS is to consist of two core flooding tanks, two high pressure injection and low pressure injection systems, with provisions for recirculation of the borated coolant after the end of the injection phase. Various combinations of these systems will assure core cooling for the complete range of postulated break sizes.

Following a postulated LOCA, the ECCS will operate initially in the passive core flooding tank injection mode and the active high pressure injection mode, then in the active low pressure injection mode, and subsequently in the recirculation mode.

The high pressure injection system (HPIS) mode of operation, upon actuation of a safety injection signal, will consist of the operation of two of three centrifugal charging pumps (rated at 700 gpm each at a design head of 2600 ft) which provide high pressure injection of 1800 ppm of boric acid solution into the reactor coolant system cold legs. Suction is taken from the borated water storage tank (BWST) which has a nominal tank capacity of 570,000 gallons.

Low pressure injection system (LPIS) will consist of two decay heat removal pumps (rated at 5000 gpm each at a design heat of 350 ft) which will take their suction from the borated water storage tank for short term cooling. The low pressure lines terminate directly in the reactor vessel through the core flooding nozzles located in the vessel wall. Crossover lines containing cavitating venturies between the redundant low pressure lines are provided to ensure that sufficient flow will be available for core cooling if a rupture occurs in the core flooding piping. The staff has required that additional instrumentation be added to the LP piping after the first check valve in order to detect and prevent the potential for overpressurization of the low pressure system. The applicant has indicated that a detection system will be provided and its description will appear in the FSAR (see Section 5.4).

When a predetermined amount of water in the borated water storage tank has been injected, or receipt of a low-level alarm for the BWST, suction will be transferred manually to the containment sump for the recirculation mode of operation provided by the LPIS. The ECCS will then provide the long-term core cooling requirements by recirculating the spilled reactor coolant collected in the containment sump, back to the reactor vessel through the core flooding line nozzles. However, should the reactor coolant system pressure be higher than the LP pump head, the required flow is delivered by the HPIS by aligning the flow from the discharge of the LP pumps to the suction of the HP pumps.

suction of the HP pumps. Presently, the PSAR shows this alignment is accomplished by the operator manually opening one valve in each of the two crossover pipe lines located in the auxiliary building. The applicant has indicated that these valves will be motor operated with control and indication in the control room. We will require this commitment to be documented in the PSAR before a construction permit is issued.

The passive injection mode of operation is provided by the core flooding (CF) system, which protects the core in the event of intermediate and large-sized pipe breaks. The coolant is automatically injected when the RCS pressure drops below the core flooding tank pressure (600 psig). Each of the two core flooding tanks has a normal water volume of 1350 ft^3 with 450 ft^3 of nitrogen gas at a normal operating pressure of 600 psig. Each tank is connected by a core flooding line directly to a 9-inch reactor vessel core flooding nozzle. The driving force for injection of the 1800 ppm borated water is supplied by pressurized nitrogen. Each core flooding line will have an electric-motor-operated stop valve for isolation of the CFT during reduced reactor coolant pressure non-critical operation and two series inline check valves for isolation of the CFT during normal reactor coolant pressure operation.

6.3.3 Performance Evaluation

The applicant has stated that the emergency core cooling systems have been designed to deliver fluid to the reactor coolant system in order to control the predicted cladding temperature transient following a postulated pipe break and for removing decay heat in the long-term, recirculation mode.

On June 29, 1971, the AEC issued an Interim Policy Statement containing Interim Acceptance Criteria for the performance of the ECCS for light-water cooled nuclear power reactors. The Interim Policy Statement includes a set of conservative assumptions and procedures to be used in conjunction with computer codes to analyze and evaluate the ECCS performance for a pressurized water reactor. A public rule making hearing on the Interim Acceptance Criteria for ECCS for light-water cooled nuclear power reactors has been completed and new ECCS criteria issued which will be effective for construction permits issued after December 27, 1974.

In accordance with the Interim Policy Statement (IPS), the performance of the ECCS is judged to be acceptable if the course of the LOCA is limited as follows:

1. The calculated maximum fuel element cladding temperature does not exceed 2,300°F.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of cladding in the reactor.

3. 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.
4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining the core.

The applicant presented an evaluation of the LOCA in accordance with the requirements of the IPS in BAW-10065, BAW-10065 Supplement 1 and BAW-10074. This evaluation resulted in a peak clad temperature of 1929°F and showed compliance with the Interim Acceptance Criteria.

As part of the FSAR for the Bellefonte plant, the applicant shall submit a LOCA analysis performed by an acceptable evaluation model under the ECCS criteria published in the Federal Register on January 4, 1974, and show that this facility is in compliance with the same criteria. However, if a construction permit is to be issued after December 27, 1974, the applicant will be required to show compliance before the construction permit is issued.

6.3.4 Tests and Inspections

The applicant will demonstrate the operability of the ECCS by subjecting all components to preoperational tests, periodic testing, and in-service testing and inspections.

The preoperational tests performed fall into three categories. One of these categories consists of system actuation tests to verify the operability of all ECCS valves initiated by Engineered Safety Feature Actuation Signal (ESFAS), the operability of all safeguard pump circuitry down through the pump breaker control circuits, and the proper operation of all valve interlocks.

Another category is the core flooding system tests. The objective of this test is to check the core flooding system and injection line to verify that the lines are free of obstructions and that the core flooding line check valves and isolation valves operate correctly. The applicant will perform a low pressure blowdown of each core flooding tank to confirm the line is clear and check the operation of the check valves.

Operational test of all the major pumps comprises the last category of tests. These pumps consist of the makeup/high pressure injection pumps, the low pressure/decay heat removal pumps, and the containment recirculation pumps. The applicant will use the results of these tests to evaluate the hydraulic and mechanical performance of these pumps delivering through the flow paths for emergency core cooling. These pumps will operate under both miniflow (through test lines) and full flow (through the actual piping) conditions.

By measuring the flow in each pipe, the applicant will make the adjustments necessary to assure that no one branch has an unacceptably low or high resistance. They will also check the system to assure there is sufficient total line resistance to prevent excessive runout of the pump. The applicant must show that the minimum acceptable flows as determined for the FSAR analysis are met by the measured total pump flow and relative flow between the branch lines. In addition, preoperational flow tests will be conducted to verify the sizing of the cavitating venturies to confirm the as-built flow split performance of the LPI system. The system will be accepted only after demonstration of proper actuation of all components and after demonstration of flow delivery of all components within design requirements.

The applicant will perform routine periodic tests of the ECCS components and all necessary support systems at power. Valves which operate after a loss of coolant accident are operated through a complete cycle, and pumps are operated individually in this test.

The Staff is presently developing a generic position with regard to testing of the ECCS. During the post-construction permit stage, the Staff will require the applicant to provide equivalent testing capability to comply to this position.

6.3.5 Conclusions

On the basis of our evaluation, we have concluded that the performance of the ECCS is in accordance with the Commission's Interim Acceptance Criteria and is acceptable in regard to a decision concerning issuance of a construction permit if issued prior to December 28, 1974.

6.4 Control Room Habitability

The applicant proposes to meet General Design Criterion 19, Control Room, of Appendix A to CFR Part 50, by use of concrete shielding and a dual mode control room emergency ventilation system. An accident signal or a high radiation signal will initiate control room isolation accompanied by pressurization with 500 cfm of outside air for minimizing the inleakage of unfiltered air. We have calculated the potential doses to control room personnel following a LOCA. The resultant doses are within the guidelines of General Design Criterion 19.

The applicant has indicated that the chlorine biocide system originally proposed will be replaced with a hypochlorite system. This hypochlorite system eliminates the potential onsite chlorine hazard. The closest potential location for a toxic gas release is a railroad line 3 miles from the site. We have evaluated this hazard and have determined that a release of 55 tons of chlorine from a railcar would not pose a serious hazard to the control room operators. However, we will require that the applicant supply emergency breathing apparatus for operators and that the operators be able to manually isolate the control room in the unlikely event of a toxic gas release. The applicant has indicated that he will meet these requirements. This will be documented in a future amendment to the PSAR. We conclude that the proposed control room design except as noted above will provide adequate protection for the operators.

6.5 Auxiliary Feedwater System

The applicant has classified the auxiliary feedwater system as an engineered safety feature. A description of this system and our evaluation of the applicant's criteria are given in Section 10.6.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The Commission's General Design Criteria, IEEE Standards including IEEE Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE Std 279-1971), and applicable Regulatory Guides for Power Reactors have been utilized as the bases for evaluating the adequacy of the protection and control systems. Specific documents employed in the review are listed in the Bibliography to this report.

The review of the protection and control systems was accomplished by comparing the designs with those of a previously approved facility, the Rancho Seco Plant. Our review concentrated on those areas of design which are unique to the Bellefonte Plant, for which new information has been received, or which have remained as continuing areas of concern during this and prior reviews of similar designed plants.

7.2 Reactor Protection System (RPS)

The RPS of the Bellefonte Plant will be functionally identical to that of Rancho Seco except for the following features:

1. The Bellefonte Units have incorporated a high pressurizer level trip which prevents the pressurizer from being filled with liquid. It is also a back up trip for accidents that would normally be terminated by high reactor coolant pressure trip.

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2. Low pressurizer level trip was added to prevent the emptying of the pressurizer in the event of a small loss-of-coolant accident. It is also intended as a back up for the low reactor coolant pressure trip.
3. Power/flow trip was added to replace the overpower trip which was based on flow and power imbalance. The power/flow trip provides primary protection for the coastdown of one RC pump from a four-pump operation at maximum power and provides protection for the locked-rotor accident. It also protects against power excursions for all modes of pump operation.
4. A calculating module (part of a new RPS design, RPS-II, developed by B&W) was added which generates signal limits providing protection for DNBR and peak power density (kw/ft) limitations. It generates a power envelope trip and a power/delta T trip and utilizes a voltage reference check feature to provide continuous on line self-check of the validity of the generated signal.
5. The RPS for the Bellefonte Plant includes redundant manual trip switches at the systems level. This feature has been incorporated in order to comply with Section 4.17 of IEEE Std 279-1971.
6. The RPS does not supply any signals to the control system. The Rancho Seco plant RPS provided reactor coolant flow and reactor coolant pressure signals to the control system.

We have reviewed these changes and have concluded that these design changes provide an equivalent or better degree of safety than that of the Rancho Seco plant and are in conformance to IEEE Std 279-1971, and therefore find them acceptable for the construction permit review stage. However, in view of the unique design features and hardware utilized (i.e., use of integrated solid state logic and mini-computer technology in the calculating module), the total system acceptability is conditional pending our generic review of the new B&W RPS-II design. This design is described in B&W Topical Report BAW 10057 (Reactor Protection System, September, 1973) and will be reviewed by the Staff prior to the operating license stage.

The applicant has replaced the high reactor building pressure trip with a low pressurizer level trip in the new RPS-II design. (The new design retains the diverse low reactor coolant pressure trip.) The Staff's position is that since the analysis of the effectiveness of the ECCS performance assumes a reactor trip at ECCS actuation, we require that diverse signals be used to trip the reactor. At this point in our review there is insufficient information to evaluate the acceptability of the low pressurizer level trip as a diverse reactor trip for this purpose. Therefore, we may require that the Bellefonte RPS design be revised to provide a reactor trip on high building pressure for this diversity as is the case on plants previously reviewed and approved.

In response to the Staff's concern, the applicant has agreed to modify the reactor protection system to include a high reactor building pressure signal to trip the reactor if it cannot be demonstrated by the FSAR stage that the low pressurizer level signal will provide an equivalent or better degree of protection.

We consider this commitment acceptable for the construction permit application and intend to re-evaluate the applicant's analysis and design in the FSAR review.

7.3 Engineered Safety Features Actuation System (ESFAS)

The ESFAS is functionally identical to that of the Rancho Seco plant except that the Bellefonte system uses a five channel system to actuate the engineered safety subsystems as opposed to Rancho Seco's four channel system. We have reviewed the instrumentation, control and electrical systems being provided for the ESFAS and have concluded that the design criteria is in conformance to IEEE Std 279-1971 and the Commission's regulations and is therefore acceptable.

The following sections identify those aspects of the design that were not acceptable to us and were changed as a result of our review; and, those items of concern that have been identified during this and previous reviews of similar plants.

7.3.1 Transfer From the Injection Mode to the Recirculation and Cross-Over Modes of Operation

Changeover from injection to the recirculation mode and the cross-over mode (using LPI pumps as boosters for the HPI pumps) of operation following a loss of coolant accident is accomplished by the operator in accordance with established procedures which include a series of manual actions. The complexity of the proposed changeover procedures to be followed during worst possible operating conditions (LOCA) did not appear to provide adequate assurance that the operator would correctly perform the required actions (see Section 6.3 also).

The applicant was requested to modify the design to provide manual initiation at the systems level in accordance with Section 4.17 of IEEE Std 279-1971, or to demonstrate that the time required for manual initiation is available and that the procedures are of such simplicity that taking exceptions to Section 4.17 can be justified.

The applicant has indicated that this system will be modified to our requirements. We will require this commitment be documented in the PSAR prior to issuance of a construction permit.

7.3.2 Activation of Trip Setpoints

The applicant has identified the high reactor coolant outlet temperature trip point to be within 4.17% of the high end of the calibrated range of the transmitter. Since the transmitter saturates well above the calibrated range and provides an adequate safety margin in case of drift, we find that the trip point setting is adequate and therefore acceptable.

7.3.3 Containment Spray System

Manual initiation of the containment spray system is accomplished by two manually operated switches per train. One of the switches actuates the spray header valves and the other switch actuates the spray pumps. This arrangement, as opposed to a single switch for manual system level initiation minimizes inadvertent actuation of the containment spray system. We find this design feature acceptable. We believe that the intent of Section 4.17 of IEEE Std 279-1971 is satisfied and that the health and safety of the public is protected. We agree with the applicant's contention that no single operator error should cause containment spray actuation.

7.3.4 Safety Related Fluid Systems

The applicant was requested to address inadvertent actuation of all electrically-operated passive and active components in safety related fluid systems and evaluate the effects relative to the single-failure criterion, and to identify the degree of conformance with the staff position stated in our December 7, 1973 letter.

The applicant has committed to comply with the staff's position and provide appropriate systems analysis during the FSAR stage. This commitment is acceptable at this, the construction permit stage of our review.

7.3.5 **Periodic Testing of Reactor Protection and Engineered Safety Feature Systems**

The applicant's design criteria provide for testability of individual channels, logic and final actuating devices and satisfy the requirements of IEEE Std 279-1971 and Regulatory Guide 1.22 (Periodic Testing of Protection System Actuation Fucntions, 2/71). We conclude that the applicant's design criteria for testing of protection systems are acceptable.

7.4 Systems Required for Safe Shutdown

We have reviewed the instrumentation, control and electrical systems being provided for safe shutdown as well as the design provisions to place and keep the plant in a safe shutdown condition in the event that access to the main control room is restricted or lost. We have concluded that the design of these systems, as modified, will acceptably conform to our criteria. The following sections identify those aspects of the design that were or will be changed as a result of our review.

Our review of the DHR system design, as initially proposed, revealed that the reactor could not be brought to the cold shutdown condition given a single failure in either of the two serially connected isolation valves located in the suction line to the DHR pumps. The applicant has agreed to modify the design by providing two additionally serially connected valves in parallel with the two existing DHR inlet isolation valves.

Both valves in each serially connected set will be powered from a different emergency bus. Provisions will be made to permit manual transfer of power to the valves to affect opening in the event of loss of either emergency power bus. Also, the applicant has stated that the design of all valve interlocks will meet the requirements set forth in the position (December 7, 1973 letter to the applicant) for high pressure to low pressure interfaces. We find this design change acceptable.

Our review of the auxiliary feedwater control system and the main steamline break instrumentation and control system revealed that the design criteria did not conform to Section 4.12 of IEEE Std 279-1971. The applicant agreed to modify the design to be in full conformance to Section 4.12 and revised the PSAR accordingly. We conclude that the modified design is acceptable.

7.5 Safety Related Display Instrumentation

We have reviewed the design criteria for the instrumentation systems that provide information (1) to enable the operator to perform required safety manual functions and (2) for post-accident monitoring, and have concluded that they are acceptable except for the following:

The PSAR listing of the instrumentation channels for post-accident surveillance does not include provision for continuous control room recording of all parameters considered essential by the staff. Also, as proposed, the instrument channel wiring would pass through the non-nuclear instrumentation cabinets (which are not Class I equipment) where they are selected for recording. The applicant was advised by letter dated January 11, 1974 of the unacceptability of these features which must be modified to comply with the staff's position. We will require the applicant to commit to the necessary additions and modifications in this area prior to issuance of construction permits for this facility.

7.6 All Other Systems Required for Safety

We have reviewed the design criteria for all other systems required for safety and find that they conform to the applicable safety standards and Regulatory Guides and are therefore acceptable.

7.7 Control Systems

The applicant has stated that the Integrated Control System design is identical to that of the Rancho Seco plant except that the Auxiliary Feedwater Control, which is required for safe shutdown, has been separated from the Integrated Control System in order to meet the requirements of IEEE Std 279-1971. This design satisfies our evaluation requirements and is acceptable.

The applicant has stated that the design of the Nuclear Instrumentation system is essentially the same as that of the Rancho Seco plant and no differences have been identified. We conclude that this design is acceptable.

The applicant has stated that the design of the incore monitoring system is essentially the same as that for the Ranch Seco plant with the exception that this plant utilizes a greater number of detector strings (62) because of a larger core. We find this design acceptable.

7.8 Environmental and Seismic Qualification

The applicant has stated that all instrumentation and electrical equipment that is required to perform a safety function will be environmentally and seismically qualified by either test or analysis in accordance with the appropriate IEEE Standards.

We conclude that the environmental and seismic qualification criteria are acceptable.

8.0 ELECTRIC POWER

8.1 General

General Design Criteria 17 and 18, IEEE Standards including IEEE Criteria for Class IE Electric Systems for Nuclear Power Generating Stations (IEEE Std 308-1971), and Regulatory Guides for Power Reactors including Regulatory Guides 1.6 and 1.9 served as the bases for evaluating the adequacy of the electrical power system. Specific documents used in the review are listed in the Bibliography to this report.

8.2 Offsite Power System

Preferred (offsite) electrical power for all safety and non-safety related auxiliary loads of both units will be supplied from the TVA 161 kV and 500 kV power systems by three independent circuits. Each of the preferred power sources and associated circuits has sufficient capacity to supply all loads regardless of plant conditions. The applicant has conducted power system stability analyses showing that the loss of the largest generating unit, or the most critical transmission line, will not adversely affect the stability of the remainder of the transmission system or the ability to provide offsite power to this facility.

For each unit, the normal preferred power source is the TVA 500 kV power system via four 500 kV transmission lines, the 500 kV switchyard, the unit transformer, and two unit station service transformers. The use of the 500 kV system as the normal source of offsite power is made possible by the provision of a load break switch between the unit generator and the connection to the high voltage winding of the unit station service transformers. The load break switch may be operated either manually or automatically and will be open during startup and shutdown and closed during normal power operation.

Two sources of reserve preferred power are provided to each unit from the TVA 161 kV power system via two physically independent 161 kV transmission lines, the 161 kV switchyard, and two reserve station service transformers. Two physically independent circuits connect the switchyard to the reserve station service transformers. Each circuit serves one of the two transformers for each unit, assuring availability of reserve power to both units in event of loss of one 161 kV circuit. Each transformer has sufficient capacity to supply loads essential for safe shutdown or accident conditions in one unit.

Automatic switching is provided to transfer the auxiliary buses to reserve preferred power in event of failure of the normal preferred power source.

Each transmission line is protected with primary and backup relaying systems. Each power circuit breaker is equipped with two separate trip coils. These components are connected so that each protective function is redundant and loss of any single component in one relaying protective scheme does not affect the proper functioning of the other protective devices.

Two independent 250-volt batteries are provided in the powerhouse to supply substation requirements. Each of the two 250-volt batteries supplies portions of the protective devices independently, and for loss of one battery, a manual transfer to the other battery will be provided. The closing control circuits for the 161-kV power circuit breakers are supplied from the powerhouse batteries in such a manner that loss of either battery will not prevent closing breakers as required to energize a reserve auxiliary transformer.

We have concluded that the offsite power system satisfies the requirements of GDC 17 and 18 and IEEE Std 308-1971 and is acceptable.

8.3 Onsite Power Systems

8.3.1 A-C Power System

Onsite standby power will be supplied by two diesel generators (D-G's) approximately 6000 kw in size. Each D-G will supply one 6.9 kV emergency bus comprising one division of a two-division split-bus configuration. Interlocks will be provided to prevent paralleling the D-G's, and to prevent closure of more than one of the three power feed breakers (normal preferred reserve preferred, or standby) of each emergency bus. Each D-G will be automatically started by an undervoltage signal from its respective bus or by a safety features actuation signal. Only one of the two D-G's will be required to provide emergency power for accident conditions.

The redundant engineered safety features and vital instrumentation and control loads will be supplied, directly or indirectly, from the two 6.9 kv emergency buses through the two-division split-bus configuration. This configuration will be maintained throughout the a-c and d-c sybsystems. There will be no automatic switching of redundant buses or loads, and interlocks will be provided to prevent redundant buses from being paralleled.

Each D-G and its auxiliary systems will be separately housed in a Seismic Category I installation.

The starting and operation of any D-G is not to be conditioned by operation of the other.

Separate onsite fuel storage will be provided for each D-G sufficient for a minimum of seven days of operation at full rated load.

At the time of our review the applicant had not yet selected the standby D-G's for the plant. However, load studies indicate these D-G's will be of a size larger than any previously used in nuclear standby applications. The applicant has committed to a test program, similar to that performed for the Zion plant, to demonstrate that the reliability of the D-G's will be at least equivalent to that of the smaller "standard" D-G's previously approved for nuclear service. The test program will include the following:

1. At least two tests acceptable to the Regulatory Staff will be performed on each D-G to demonstrate the start and load capability of these units with some margin in excess of the design requirements.
2. Performance of at least 300 valid start and load tests will be required prior to initial reactor criticality of Bellefonte, with no more than three failures allowed. This must include all valid tests performed offsite. (A valid start and load test is defined as a start from design cold ambient conditions with loading to at least 50% of the continuous rating within the required time interval, and continued operation until temperature equilibrium is attained.)
3. A failure rate in excess of one per hundred will require further testing as well as a review of the system design adequacy.

For each unit, four independent channels of 120-volt a-c vital control power will be provided by static inverters and their associated distribution boards. Each inverter will receive normal 480 volt a-c power input from the a-c emergency power system (2 inverters supplied from each division) and backup d-c power system from a 120 volt vital battery through an auctioneering type circuit. Should the normal a-c power fail or be subjected to reduced voltage, the inverter will be supplied without interruption from the vital battery source.

We have concluded that the a-c emergency onsite power system satisfies the requirements of GDC 17 and 18, IEEE Std 308-1971 and Regulatory Guides 1.6, 1.9 an/ 1.22, and is acceptable.

8.3.2 D-C Power System

The vital d-c power system for each unit is comprised of four 125 volt batteries, each with an assigned static battery charger and distribution board. A spare installed charger is provided for each pair of batteries. The batteries are independent Class I installations housed in separate rooms.

The d-c system is compatible with the two-division split-bus configuration of the a-c system.

We have concluded that the design of the d-c system conforms to the stated criteria, safety guides and standard and is acceptable.

8.4 Physical Independence of Electric Equipment and Circuits

We have reviewed the applicant's criteria for physical separation of electric equipment and circuits to preserve the independence of redundant equipment. In addition, the applicant has supplemented its criteria, at our request, to include the staff's position on Physical Independence of Electric Systems as provided in our letter dated December 4, 1973.

We conclude that this commitment is satisfactory at this stage of our review. We will evaluate the applicants implementation of these criteria during the FSAR review of this facility.

9.0 AUXILIARY SYSTEMS

The safety related auxiliary systems have been evaluated for the appropriate reactor safety and radiological safety requirements. These systems are grouped in the following paragraphs to indicate the requirements that are applicable.

The auxiliary systems necessary to assure safe plant shutdown are: essential raw cooling water systems, component cooling water system, ultimate heat sink (in conjunction with the essential raw cooling water system,) portions of makeup and purification system, decay heat removal system, portions of the chemical addition and boron recovery system, air conditioning, heating, cooling and ventilation in the control building, the diesel generator building and portions of the auxiliary building, fuel oil system, cooling water system, starting system and lubricating system for diesel generators, and portions of the compressed air system.

We have also reviewed those auxiliary systems whose failure would not prevent safe reactor shutdown, but could, directly or indirectly, be a potential source of radiological release to the environment. These systems include raw water cooling, demineralized water makeup, portions of the compressed air system, process sampling system, and the miscellaneous air conditioning, heat and ventilation system.

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From our review of the design criteria and basis for these systems, we find they are comparable in design and function to other PWR facilities (eg., North Anna Units 3 and 4) that have been previously reviewed and approved. On this basis we have concluded that these auxiliary systems are acceptable.

9.1 Fuel Storage and Handling

9.1.1 New and Spent Fuel Storage

New fuel storage space will be provided for one refueling batch for each reactor or a total of at least 138 fuel assemblies. The new fuel storage space will be an integral part of the fuel handling building which will be a Category I structure.

The racks will be designed with sufficient spacing to limit the effective multiplication factor (K_{eff}) of the stored fuel to less than 0.90 even if immersed in unborated water. Based on our review, we conclude that the design criteria and basis for the new fuel storage space are acceptable.

The spent fuel storage pools are also located in the fuel handling building. Each reactor will be provided with its own spent fuel storage pool. Each pool will be seismic Category I, reinforced concrete, lined with stainless steel and of a size to store a total to 1 1/3 cores. The storage racks will be designed with sufficient spacing to limit the effective multiplication factor (K_{eff}) of the stored spent fuel to less than 0.90 even if immersed in unborated water.

A separate cask storage pit will be provided in the proximity of the spent fuel storage pool. The cask pit will be separated from the pool by a reinforced concrete wall with a gated slot which permits movement of fuel from the storage area of the loading area. The gate between the fuel pool and cask handling area will be kept closed whenever the cask is being loaded into or raised from the cask loading area, thus assuring that the water level of the fuel pool will not be affected in the event of a cask drop in the cask handling area. The shipping cask will be prevented from being moved over the stored fuel by arrangement of the fuel pool, cask storage and loading area. The applicant has evaluated the effects of the cask being dropped into the cask loading area even if it is tipped. The results indicate that the stored spent fuel and other safety related systems are protected from a postulated spent fuel cask drop accident.

We conclude the applicant has demonstrated that the design criteria and bases are in accordance with Regulatory Guide 1.13 (Fuel Storage Facility Design Basis, 3/71) and is, therefore, acceptable.

9.1.2 <u>Spent Fuel Cooling and Cleanup System</u>

The spent fuel pool cooling and cleanup system is designed to maintain the required quality and clarity of the pool water and to remove the decay heat generated by the stored spent fuel assemblies.

The energy release from the normal 2/3 core of spent fuel (1/3 core from each reactor) can be removed by one cooling train, maintaining a pool bulk fluid temperature of 150°F. With 1 1/3 cores of spent fuel stored in the pool, one cooling train will maintain the bulk fluid temperature of the pool at or below 200°F. Normal operation of the spent fuel cooling system (SFCS) utilizes one pump and one cooler. If the pump and/or cooler fails during operation, a standby pump and cooler are avaialble.

In the event that the standby pump or cooler would not be available the decay heat removal system (DHRS) will be initiated to cool the spent fuel pool. The DHRS will only be started to supplement the SFCS if the reactor of the associated DHRS is empty, being refueled, or being shutdown where only one DHRS string is required for decay heat removal. Locked valves will be provided to prevent the initiation of the corresponding DHRS for spent fuel cooling purposes.

The purity and clarity of the fuel pool water will be maintained by a purification loop and a fuel pool skimmer loop. The purification loop will also be used for cleanup of the reactor cavity and fuel transfer canal during refueling and the borated water storage tank contents following refueling. The cleanup system will be designed for non-nuclear service. This part of the system can be isolated from the spent fuel cooling system in the event of a malfunction.

All system connections to the fuel pool will penetrate the pool sufficiently above the top of the stored spent fuel assemblies to assure adequate cooling in the event of a leak or break in a pipe connection. The design of the piping system will be such as to prohibit siphoning below the minimum cooling level. The cooling loop piping and components will be designed to ANS Safety Class 3 and seismic Category I requirements.

Based on our review, we conclude that the design criteria and bases for the spent fuel pool cooling and cleanup system are acceptable.

9.1.3 Fuel Handling System

The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated state until it leaves after post-irradiation cooling. The system basically consists of cranes and handling equipment, fuel transfer system, fuel storage racks, refueling canal, fuel storage area and decontamination facilities. The components and structures that transport, position, raise, lower or house the fuel assembly during transportation will be designed to meet seismic Category I requirements. We conclude that the design criteria and bases are acceptable.

9.2 Water Systems

9.2.1 Essential Raw Cooling Water System

The essential raw cooling water system (ERCWS) will be designed to seismic Category I and ANS Safety Class 3 requirements, except that containment penetrations and piping inside the reactor building will be designed to seismic Category I and ANS Safety Class 2 standards. Cooling water will be provided by this system to all components essential for safe reactor shutdown, either directly or by way of the component cooling water system.

The ERCWS mainly consists of eight pumps, associated piping, strainers and valves. The system piping will be arranged in two independent header loops each providing a continuous flow of cooling water to those systems and components necessary for plant safety either during normal plant operation of under accident conditions. During any postulated combination of modes of units operation and adverse environmental occurrences, operation of three pumps on one header loop will be sufficient to supply all cooling water requirements for both units. During the hypothetical combined mode of loss-of-coolant-accident in one unit while the other unit is in cooldown with loss of offsite power and four diesel generators in operation, six pumps are required to operate.

We find that the system has adequate redundancy in addition to its ability to withstand a single failure without limiting the ability of the safety systems to perform their functions in the event of natural disasters or plant accidents. We, therefore, conclude that the design criteria and bases for the systems are acceptable.

9.2.2 Component Cooling Water System

The component cooling water system (CCWS) is safety related and provides cooling water to dissipate heat from various components and piping of the system will be designed according to ANS Safety Class 3 and seismic Category I requirements. It will act as an intermediate heat sink and barrier between the reactor coolant and the ERCWS which is open to the atmosphere. The system will be designed to operate at a higher pressure than the ERCWS to ensure that the cooling water quality can be maintained in the event of CCWS system leakage.

In order to preclude leakage of radioactivity to the environment, radiation monitors will be provided to detect potential radioactivity inleakage to the ERCWS with indication and alarms in the control room. Isolation valves are provided to facilitate the removal of leaking equipment and contain any radioactive leakage.

The system supplying cooling to safety related equipment will be designed to meet single failure criterion with two separate cooling loops having a physical separation barrier located between the loops.

The first subloop contains two pumps (one operating and the other on standby), one cooler, one surge tank, one filter; and will serve all the component cooling requirements during normal operation. The second subloop contains one pump, one cooler, one surge tank, one filter; and will be utilized as emergency standby. Under emergency conditions, cooling water flow to the non-safety components will be terminated with sufficient redundancy in order to ensure emergency heat removal capability of the system. We conclude that the design criteria and bases for the system are acceptable.

9.2.3 Ultimate Heat Sink

The ultimate heat sink design will be provided to ensure sufficient coolant flow for: (1) simultaneous safe shutdown of either unit coincident with a loss-of-coolant-accident in the other unit, and (2) maintain the two units in the safe shutdown condition, dissipating decay energy from both units for a minimum of 30 days without replenishment. A single water source and intake structure will be provided for the ultimate heat sink complex.

The intake structure and a 2000 foot channel which runs into the Guntersville Reservoir will be designed for the safe shutdown earthquake (SSE), tornado, severe storm and probable maximum flood. The applicant assumed a water surface slope of 0.1% over the entire channel length and an additional 1/2 foot for the transition from the intake forebay to the sump of the essential raw cooling water pump in order to ensure sufficient pump submergence with Guntersville Reservoir at its lowest level.

In the event of an accident condition causing either the main steam condenser or the cooling tower to malfunction, the plant would be shutdown and the Guntersville Reservoir would serve as the ultimate heat sink. In this regard, the occurrence of a SSE coincident with a loss-of-coolant accident and a loss of either the downstream or upstream dam is considered as a design basis event for this facility. We conclude that the design criteria and basis for the ultimate heat sink are acceptable.

9.2.4 Condensate Storage Facilities

The condensate storage facility provides water for makeup and surge capacity for the turbine plant, system inventory and a reserve water supply for the auxiliary feedwater pumps (its only safety related function).

Each unit will be provided with one condensate storage facility consisting of a 700,000 gallon capacity storage tank, transfer pump and associated piping. The tank internal piping arrangement will be such as to reserve a minimum of 300,000 gallons of condensate for auxiliary feedwater pumps. The storage tank will be designed to non-seismic Category I requirements and not protected from external missiles. Any loss of the tank would be compensated for by using essential raw cooling water for auxiliary feedwater supply. We conclude that the design criteria and bases for this system are acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

The compressed air system will be designed to supply compressed air to all pneumatically operated components during normal and/or emergency plant operations, and will be divided into three separate subsystems, the essential air system, the control air system, and the service air system.

These subsystems will be served by four seismically qualified, motor-driven, non-lubricated compressors located in the auxiliary building. Each of the compressors is sized to supply the total plant essential and control air requirements under normal conditions with sufficient additional capacity to handle minimal service air requirements.

The essential air system is safety related and will be designed to ANS Safety Class 3 and seismic Category I requirements. The system will have two separated trains consisting of an individual compressor and air receiver and will meet the single failure criterion. Each of the essential air headers will normally supply compressed air to components of the safety system within its domain. However, in the event of loss of a compressor in a particular train, the air receiver will have sufficient capacity to actuate containment isolation valves in that particular train. On loss of off-site power each of the compressors will automatically be provided power from the on-site emergency power supply.

The control air system and service air system are not safety related, nevertheless, all pipe hangers for piping located in seismic Category I structures will be designed for seismic requirement. Necessary valving will be provided, so that the service air system can be automatically isolated when air pressure in the header supplying control and essential air drops below a predetermined level. If the pressure drops to 80 psig, the control air system will be automatically isolated; thus conserving all the remaining air for the essential air system. We conclude that the design criteria and bases for this system are acceptable.

9.3.2 Equipment and Floor Drainage System

The equipment and floor drainage system will be designed to accommodate all auxiliary building and containment drains during plant operation. The equipment and floor drainage system will collect water from potentially radioactive sources and transfer it to the radioactive waste holdup tank.

The system will be designed to prevent the spread of radioactivity and the subsequent release of radioactive material to the outside atmosphere due to leakage during normal operation. In areas where safety related equipment are located leak detectors and emergency drains will be provided. This will enhance the operability of adjacent equipment in the event of equipment leakage during an accident. We conclude that the design criteria and bases for this system are acceptable.

9.3.3 Makeup and Purification System

Throughout the life of the reactor core the makeup and purification system compensates for core burnup by adjusting the concentration of boric acid in the reactor coolant and thereby assisting in the control of available reactivity to within acceptable limits. In addition it: (a) regulates the inventory of coolant in the reactor coolant system, (b) removes impurities from the coolant, (c) controls the concentration of corrosion inhibiting chemicals in the coolant, (d) supplies borated makeup water to the core flooding tanks, (e) provides injection water into the reactor coolant system following engineered safeguards actuation, and (f) provides injection water to the reactor coolant pump seals.

The system will be designed to ANS Safety Class 2 and seismic Category I requirements. Alternate fluid flow paths, redundant components and isolation valves, will be provided to ensure the system operability as an enginnered safeguard.

We conclude that the design criteria and bases for the system are acceptable.

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

9.4.1 Control Building

The control building will have two 100 percent redundant engineering safety feature quality air conditioning systems each containing a water-cooled condenser-compressor water chiller, an air handling unit, chilled water pump and piping, compressors, filter assemblies, duct work, dampers and necessary controls for automatic and/or manual operation. The operator will have remote manual control capability to ensure satisfactory control room conditions following an accident. The control room will be maintained at a positive pressure to prevent entry of dust, smoke, radioactivity and other contaminants.

Outdoor air to the control room will be taken from two separate locations each of which will be provided with duplicate radiation monitors, chlorine and smoke sensors. If the air to both inlets is radioactive or chlorine or smoke concentrations exceed established limits, the system would be automatically isolated and placed in emergency mode of operation. Electrical equipment essential for system operation will be powered by the onsite emergency power sources. Based on our review of this system, we conclude that the design criteria and bases are acceptable.

9.4.2 Auxiliary Ventilation System

The auxiliary building air-conditioning heating, ventilating and air cleanup system will be designed to maintain an acceptable building environment for protection of plant equipment and to limit the release of radioactivity to the atmosphere. The system will be required for shutdown of the reactor and to maintain it in a safe shutdown condition. Areas containing engineered safety-equipment will be provided with air cooling units in addition to the normal ventilating systems. Engineered safety equipment coolers will be provided with emergency power and essential raw cooling water sources. The fuel handling areas will be maintained at a slightly negative pressure to minimize communication of air with other portions of the building. Effluent from the fuel handling areas will be ventilated and discharged through the seismic Category I designed ducts and charcoal filters to the station vent. This system will be similar to that of other recently approved PWR plants. We conclude that the design criteria and bases for the system are acceptable.

9.4.3 Diesel Generator Buildings

Each diesel generator building will be provided with a seismic Category I heating and ventilating system consisting of two 100 percent capacity fans, filters, dampers, duct work and controls. Power for the redundant heating and ventilation equipment will be supplied from separate emergency power sources. The battery and electrical equipment rooms provided for each of the diesel generator buildings will be ventilated by a separate exhaust fan which will discharge to the atmosphere. Redundancy in components will ensure the system operability during accident conditions. We conclude that the design criteria and bases are acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The fire protection system will be designed to provide fire protection in the areas of essential systems or controls where a fire could prevent the operation of the safety related systems. It also provides fire protection to the safety related equipment in the auxiliary building, control building and reactor buildings which are designed to seismic Category I requirements.

The high pressure fire protection system will provide water to all points throughout the facility where water for fire fighting may be required. In areas where water could create a hazard due to the nature of equipment or the type of fire, a lower pressure carbon-dioxide fire protection system will be provided. All components, piping and water supply sources of the fire protection system necessary to maintain public safety with respect to nuclear hazards will comply with ANS Safety Class 3 and seismic Category I requirements. The remaining portion of the fire protection system is not associated with nuclear safety; this portion (including the turbine building, yard and other non-essential structures) will be designed in accordance with AWWA and Power Piping Code, ANSI B31.1-1967

A minimum of two fire detection devices will be installed in any given area of the facility not normally occupied by operators or where rapid fire detection is otherwise essential. All detectors which activate the system and alarms provided in the control room will be installed in accordance with the applicable standards of the latest edition of the National Fire Protection Association Codes. In addition, portable fire fighting equipment will be provided throughout the plant for fighting small fires. Based upon our review of this system, we conclude that the design criteria and bases are acceptable.

9.5.2 Diesel Generator Auxiliary Systems

The diesel generator fuel oil system will be designed to supply diesel fuel to each diesel engine operating at full load for a period of not less than seven days. The system will be composed of two redundant trains, each having a seven-day diesel oil supply tank assembly, motor driven day-tank fuel transfer pump and day tank. From the seven-day fuel tank assembly to day tank, the system will be designed to ANS Safety Class 3 and seismic Category I requirements; and will be protected from the effects of tornadoes, hurricanes, floods, snow or ice. We conclude that the system design criteria and bases are acceptable.

A closed circulating water cooling system will be provided for each diesel engine. The system will be designed to meet the single failure criteria and in accordance with seismic Category I requirements. Cooling water to each heat exchanger will be provided by duplicate headers from the essential raw cooling water system. A failure of one header will not jeopardize cooling water supply to the heat exchanger.

We conclude that the system design criteria and bases are acceptable.

Each diesel engine will be equipped with two full capacity starting air motors, and redundant starting systems. The diesel generator will be automatically started on the loss of off-site power or by the engineered safety features actuation system signals. Two air accumulators will be provided for each diesel engine. Each accumulator is sized for an air storage capacity sufficient to crank the engine five times without recharging. Two electric-motor driven air compressors will also be provided for each diesel generator power package. Each compressor is sized to recharge one set of accumulators in 30 minutes. The appropriate portions of the starting air supply system, as specified by IEEE Standard 387 (Trial Use Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations), will be designed to ANS Safety Class 3 requirements. We conclude that the design criteria and basis for the system are acceptable.

An individual lubricating oil system will be provided for each diesel generator engine. The system is internal to the diesel engine and meets the requirement of IEEE Standard 387. A failure in the lubricating oil system of one diesel generator will not effect the operation of any other diesel generator. Lubricating oil is cooled via the lubricating oil cooler by the closed circuit cooling water system, and will be heated to ensure rapid starting when the engine is not operating. We find the design criteria and bases for the system are acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam power conversion system will be of conventional design, similar to those of previously approved PWR plants using B&W NSSS such as North Anna 3 and 4. The system will generate electricity through a turbine generator which is supplied steam from the two steam generators described in Section 1.2. A condenser transfers unusable heat in the secondary cycle to the condenser cooling water and from there to the atmosphere through cooling towers. The entire system will be designed for the maximum expected energy from the nuclear steam supply system.

Immediately after loss of full load, this system will dissipate a portion of the stored energy in the reactor coolant system through safety valves and power relief valves to the atmosphere.

10.2 Main Steam Supply System

The steam from the two steam generators will be routed to the turbine by four main steam lines. Each line will contain one main steam isolation valve outside the containment. The four main steam lines will be joined to formed two separate steam supply headers. These two headers will conduct the total steam flow from the steam generators to a common header and then to the turbine. Branch piping from the main steam common header provides steam to the turbine bypass to the condensers. Steam for the turbine driven auxiliary pump will be taken from two main steam lines upstream of the main steam isolation valves. Portions of the main s...

supply system upstream of the main steam isolation valves and inside the main steam safety valve rooms, will be designed to ANS safety class 2 and seismic Category I requirements. The main steam isolation valves are located outside and as close as possible to the secondary containment building. In the event of a main steam line rupture accident the main steam isolation valves are capable of closing and isolating a steam generator within 7.5 seconds of the detection of high containment pressure or low steam generator pressure.

The main steam supply system is not normally radioactive, but in the event of reactor coolant leakage into the steam generators, radioactive material could be released to the secondary system. The condenser vacuum pump radiation monitoring system and the turbine building radiation monitoring system will provide redundant detection of radioactivity in the secondary system. The vacuum pump exhaust is passed through a high efficiency particulate air filter and charcoal absorber train before being discharged through the turbine building exhaust vent.

We conclude the design criteria and basis for the main steam system are acceptable.

10.3 Steam Generator Dump System

The steam generator dump system will be divided into two subsystems, the turbine bypass system and the power operated relief valve system. The turbine bypass system will be designed to maintain secondary system pressure within allowable ASME code limits and reduce the magnitude of nuclear system transients following large turbine load reduction. Inadvertent or accidental opening of any one steam dump valve during power operation will not subject the reactor coolant system to an uncontrolled depressurization and cooldown.

The power operated relief valve system will be designed with sufficient capacity to maintain secondary system pressure without lifting the ASME Code safety valve, and by such steam release, reduce the magnitude of nuclear system transient. The system consists of six power operated relief valves, four being on/off type and two are modulating types, located upstream of the main steam isolation valve and housed inside the safety valve rooms. Inadvertent or accidental opening of any one power operated relief valve during power operation will not subject the reactor coolant system to an uncontrolled depressurization and cooldown. We conclude that the design criteria and basis for these systems are acceptable.

10.4 Condenser Circulating Water System

The condenser circulating water design is such that system piping failures will not result in flooding of areas where safety related components are located. We conclude the design criteria and basis for the system are acceptable.

The condensate cleanup system acts to protect the steam generator tubes and tubesheets by removing corrosion products carried over from the steam lines, turbine, condenser, feedwater heaters, piping, as well as impurities which might enter the system in the makeup water. This system will also be able to remove condensate radioactivity produced in the event of steam generator tube leakage.

The cleanup system consists of five mixed bed demineralizers and accessories per reactor. The demineralizers will be shielded from the turbine building and valve galleries. Each of the five demineralizers is isolated in an individual concrete cell. During normal operation all five demineralizers will be in service with a spare resin charge in the resin storage tank; however, the components in the system can be separately isolated with no curtailment or interruption of power generation.

We conclude that the design criteria and bases for the system are adequate.

10.5 Feedwater System

Feedwater flow to the steam generators will be interrupted upon initiation of a feedwater isolation signal. This isolation (accompanied by a reactor trip) is accomplished by closure of redundant valves in the piping to each steam generator. High level in any one steam generator initiates rapid closure of the feedwater control valve associated with the steam generator in which the high level occurs. However, the remainder of the feedwater system, downstream of the feedwater isolation valve, will be designed according to safety class 2 requirements. We conclude the design criteria and bases for the system are acceptable.

10.6 Auxiliary Feedwater System

The auxiliary feedwater system will be designed to supply adequate secondary side cooling to remove system stored and residual reactor core energy in the event of a loss of main feedwater system. The system consists of one 1200 gpm capacity turbine driven pump and two 600 gpm capacity motor driven pumps, piping and valves. Either the two motor driven pumps or the one turbine driven pump will be sufficient to supply the required feedwater in the case of loss of all offsite A-C power, and a steam line or feedwater line break accient. The primary source of water supply for all auxiliary feedwater pumps is the condensate storage tank which will provide a minimum of 300,000 gallons for the system. As an unlimited backup water supply, a separate essential raw cooling water system header feeds each motor driven pump, and the turbine driven pump will receive backup water supply from either or both of the essential raw cooling water headers. The auxiliary feedwater system is a safety related system and will be designed to engineered safetyfeature requirements. It will remain functional following safe shutdown earthquake, a loss of off-site power, or a single failure in the system or its supporting systems. Redundant electrical power and air supplies will ensure reliable system initation and operation.

The motor driven pumps are powered by either off-site or on-site power sources; the turbine driven pump takes steam from two of the four main steam lines upstream of the main steam isolation valves. Essential parts of the system will be designed to seismic Category I requirements and protected against missiles by physical separation of all redundant components. We conclude that the criteria for the design of the auxiliary feedwater system are acceptable, and that the system design based on these criteria will satisfy the requirements set forth in the discussion concerning postulated pipe breaks outside containment.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste systems will consist of the liquid waste system, the gaseous waste system, and the solid waste system. These systems are shared by both reactors. The liquid waste system will process waste liquid streams such as equipment drains and leakage, condensate demineralizer regenerant liquids, decontamination and laboratory waste liquids, and laundry and shower waste water. The treated liquid waste will be recycled for reuse if the plant waste balance requires makeup and if the water quality is adequate. The liquid waste system will treat waste liquid utilizing evaporation, demineralization, and filtration for removal of radioactive material, chemical impurities, and particulates.

Gaseous wastes will be generated during the operation of the plant from degassing of primary coolant, vents from equipment handling radioactive materials, and leakage from systems and components containing radioactive material. The gaseous waste system will treat gaseous streams for radioactive materials removal by filtration, adsorption, and holdup for radioactivity decay. The treated gas streams will be released to the environment.

Solid wastes will be generated during plant operation, and the wastes will consist of such radioactive material as clothing, evaporator bottoms, demineralizer resins and discarded radioactive components and tools. Treatment will consist of solification. Disposal will consist of packaging and shipping to a licensed burial site. Units 1 and 2 share the radioactive waste management systems.

A listing of the liquid and gaseous source terms is given in Table 3.2 and 3.3 of the AEC's Draft Environmental Statement dated February, 1974.

11.2 Liquid Wastes

The liquid radioactive waste treatment system will be divided into the following four subsystems: (1) tritiated waste, (2) nontritiated waste, (3) chemical waste, and (4) detergent waste.

The tritiated waste subsystem will consist of a 25,000 gal tritiated waste holdup tank, a 2 gpm waste evaporator, two 2200 gal test tanks, and a mixed bed demineralizer. The liquid collected in the tritiated waste holdup tank, consisting of floor and equipment drains with tritium content and deborating demineralizer sluice water, will be processed through the system and stored in the distillate storage tank in the boron recovery system.

The non-tritiated waste subsystem will consist of a 55,000 gal spent regenerants tank, a 25,000 gal non-tritiated waste holdup tank, a 30 gpm auxiliary waste evaporator, two 22,000 gal test tanks, and a mixed bed demineralizer. Regenerant solution from the condensate demineralizers will be collected in the spent regenerant tank, and liquid floor and equipment drains with low tritium content will be collected in the nontritiated waste holdup tank. The liquid in these tanks will be processed through the nontritiated waste subsystem and discharged to the cooling towers blowdown stream.

The chemical waste subsystem will consist of a 600 gal chemical drain tank. Waste liquid collected in this tank will be transferred to the nontritiated waste subsystem for treatment. The wastes collected in the chemical drain tank will consist of laboratory drain and non-detergent decontamination liquids.

The detergent waste subsystem will consist of a 3700 gal laundry and hot shower drain tank, a 15,000 gal fuel cask decontamination waste collection tank, and two filters. Laundry and hot shower waste will be collected in the laundry and hot shower drain tank, filtered, and released to the cooling tower blowdown stream. Spent fuel cask decontamination wastes will be collected in the fuel decontamination waste collection tank, filtered, and released to the cooling tower blowdown stream.

The liquid effluent released to the cooling tower blowdown stream will be monitored. Upon a high radiation indication, the valve in the liquid waste discharge line will be automatically closed.

Based on the parameters given in WASH-1258, our evaluation assumed a tritiated waste subsystem flow rate of 895 gpd, a non-tritiated subsystem flow rate of 585 gpd, a chemical waste subsystem flow rate of 400 gpd, and a detergent waste subsystem flow rate of 450 gpd. We assumed that tanks will be filled to 80% capacity, and that the system should have the capacity to treat radioactive liquids generated from back-to-back refueling. We assumed the decontamination factor (DF) for evaporation to be 10^4 for all nuclides except iodine and 10^3 for iodine. For mixed bed demineralizers we assumed a DF of 10^2 for cations and anions, except cesium and rubidium for which we assumed a DF of 2. In our evaluation the quantity of radioactivity released in the liquid effluents during normal operation, including anticipated operational occurrences, we calculated using the ORIGEN Code described in WASH-1258. We calculate that the radioactive material in the liquid effluent exclusive of tritium and dissolved gases, will be 0.1 Ci/yr/unit. We estimate that there will also be an annual release of 350 Ci/yr/unit of tritium based on operating data of similar reactors.

The system will be designed for Quality Group C in accordance with Regulatory Guide 1.26. The tritiated waste holdup tank, waste evaporator feed tank, waste evaporator, and waste evaporator distillate test tanks will be designed for seismic Category I and the remaining components will be designed to nonseismic Category I in accordance with Regulatory Guide 1.29. The liquid waste system will be installed in a seismic Category I structure.

Based on our evaluation of liquid radwaste releases, we calculate that the whole body and critical organ doses will be less than 5 mrem/yr at or beyond the site boundary, and that the proposed systems will be capable of limiting the release of radioactive materials in liquid effluents to less than 5 Ci/yr/unit. We find that the proposed liquid radwaste system to be capable of reducing effluents to "as low as practicable" levels in accordance with 10 CFR Part 20 and 10 CFR Part 50.36a.

We also find that the liquid radwaste system will be designed in accordance with acceptable codes and standards that meet the guidelines of Regulatory Guides 1.26 and 1.29. Based on our findings, we conclude that the liquid waste system proposed for this facility is acceptable.

11.3 Gaseous Wastes

The gaseous waste treatment systems will consist of the waste gas system and the portions of the ventilation system that controls the release of radioactive effluents to the environment. The waste gas system will treat gases from the chemical addition and boron recovery system, primary system degassing, and equipment vents in systems handling tritiated liquid. The waste gases will be compressed into one of two 3000 cu/ft gas decay tanks at 85 psig. The gas decay tanks will be designed for 100 psig at 200 F. The gases, after a decay period of at least 60 days, will be vented through HEPA filters and charcoal adsorbers to the environment.

Exhaust gases from the main condenser mechanical vacuum pumps will be vented through HEPA filters and charcoal adsorbers to the environment. The turbine will be fitted with gland seals that will be supplied with sealing steam from an auxiliary boiler. Steam from the seal exhaust header will be condensed in the gland seal steam condenser. The noncondensibles from this condenser will exhaust to the atmosphere without treatment. The gaseous waste generated from this source is expected to be negligible.

The various areas of the reactor plant will be provided with ventilation systems. Effluents from these systems will contribute to the gaseous source term and the gaseous radioactive releases from the ventilation systems are included in this section. The evaluation of the ventilation systems from a source term point of view is given in Section 12.2.

Our evaluation of the gaseous waste and ventilation systems was based on the parameters, calculational techniques, and the STEFFEG computer code as described in WASH-1258. We calculate that a total of 2500 Ci/yr of noble gases and 0.007 Ci/yr of iodine-131 will be released form each unit. For reactor operation with up to 1% of the operating power fission product as the source term, the release of radioactive effluents are calculated to be less than 10 CFR Part 20 limits.

The waste gas decay tanks and compressors will be designed for Quality Group C in accordance with Regulatory Guide 1.26 and seismic Category I in accordance with Regulatory Guide 1.29. The gaseous waste treatment system will be installed in a seismic Category I structure.

Based on our evaluation of the applicant's proposed gaseous radwaste treatment system, we calculate that the annual air dose due to gamma radiation at or beyond the site boundary will not exceed 10 mrads, the annual air dose due to beta radiation at or beyond the site boundary will not exceed 20 mrads, the annual thyroid dose to an individual will not exceed 15 mrems considering the location of the nearest cow, 11 mi. SSW of the site, and the annual total quantity of iodine-131 released will not exceed 1 Ci for each reactor at the site. We find the proposed gaseous radwaste system to be capable of reducing effluents to "as low as practicable" levels in accordance with 10 CFR Part 20, 10 CFR Part 50.36a, and Regulatory Guide 1.42. We also find that the gaseous radwaste system will be designed in accordance with acceptable codes and standards and meets the guidelines of Regulatory Guides 1.26 and 1.29. Based on our findings, we conclude that the gaseous radwaste system propsed for this facility is acceptable.

11.4 Solid Waste

The solid radwaste system will be designed to collect, monitor, process, package, and provide temporary storage for radioactive solid waste prior to offsite shipment and disposal in accordance with applicable regulations.

Miscellaneous dry waste consisting of clothing, rags, paper, and air filters will be compacted into 55-gal drums by a baling machine.

Spent resins will be placed in 30-gal drums and mixed with a blend of vermiculite and cement for solidification. Evaporator bottoms will be placed in 55-gal drums and mixed with a blend of vermiculite and cement for solidification. If required by the activity level, the filled drums will be enclosed in steel-jacketed lead shields for shipment.

We estimate that for each reactor unit approximately 500 30-gal drums of spent resins, 200 55-gal drums of evaporator bottoms, and 600 55-gal drums of miscellaneous dry waste will be shipped offsite each year. We estimate that each drum of spent resins will contain approximately 10 Ci after 180 days of decay; each drum of evaporator bottom will contain approximately 2 Ci after 180 days decay; and the 600 drums of low activity waste will contain less than 5 Ci total.

Drums will be filled and sealed by remote means. Storage time will be provided depending on the curie content and number of drums generated.

Our evaluation indicates that the volumes and radionuclide concentrations estimated for this facility are consistent with operating experience for similar plants. The system for processing and packaging the solid wastes will be of adequate capacity. Solid waste will be packaged within the limits specified by Federal Regulations 10 CFR Part 71 and 49 CFR Parts 170-178 which are applicable to packaging, handling, and transportation of radioactive materials. The wastes will be shipped to a licensed burial site in accordance with AEC and DOT regulations.

Based on our evaluation of the solid waste system we conclude that the system design will accommodate the wastes expected during normal operations including anticipated operational occurrences in accordance with existing Federal and local regulations. The system design provides for waste handling in a manner consistent with maintaining as low as practicable (ALAP) exposures to operators. The wastes will be packaged and shipped to a licensed burial site in accordance with AEC and Department of Transportation Regulations. Based on these findings, we conclude that the solid waste system is acceptable.

11.5 Process and Effluent Radiological Monitoring Systems

The process radiation monitoring system will be designed to provide information on radioactivity levels of systems throughout the plant, on leakage from one system to another, and on levels of radioactivity released to the environment. The system will consist of radiation monitors for ventilation vent particulate and gas, elevated release particulate and gas, auxiliary building exhaust, fuel building exhaust, containment purge exhaust, leak collection area gas, component cooling water, condenser air ejector discharge, steam generator blowdown tank discharge and composite sample, reactor coolant letdown, auxiliary steam condensate, gaseous waste disposal system gas and particulate, waste gas tank vault ventilation, liquid waste contaminated drain monitor, liquid waste effluent monitor, reactor coolant plant component cooling system, and waste gas decay tanks.

Samples will be taken of all liquid and gaseous releases to the environment. Gaseous and liquid waste streams will be automatically terminated when radioactivity content is above a predetermined level.

Provisions to monitor all normal and potential pathways for release of radioactive materials to the environment will be made in conformance with General Design Criterion 64. Control of releases of radioactive effluents to the environment will be in accordance with General Design Criterion 60. Compositing of samples for low-level analyses and provisions for instrumentation and facilities to perform gross beta-gamma and alpha measurements and isotopic analyses will be in accordance with Regulatory Guide 1.21 (Measuring and Reporting of Effluents from Nuclear Power Plants, 12/7).

Based on our evaluation of the radiological process and effluent monitoring system for normal operation and anticipated operational occurrences, we conclude that the plant is adequately provided with process and effluent monitoring equipment and meets the requirements of General Design Criteria 60 and 64. Based on our findings, we conclude that the radiological process and effluent monitoring system is acceptable.

12.0 RADIATION PROTECTION

12.1 Shielding and Health Physics Program

The Staff has evaluated the radiation protection design features and health physics program presented in the PSAR for this facility. This review was conducted to determine that facility design and operational practices are such that exposures to personnel will meet the requirements of 10 CFR Part 20. The review included evaluation of the facility layout, radiation sources, shielding and ventilation, radiation monitoring, access control, expected radiation and airborne radioactivity levels, and the health physics organization, equipment and procedures.

This plant is similar to other licensed light water power reactors in terms of equipment layout, shielding, ventilation and health physics program. Based on past experience from operating nuclear reactor plants, it is estimated that the average collective dose to all on-site personnel will be approximately 450 man-rem per year per unit. Design measures described in the PSAR such as operating valves behind shield walls, provisions to drain equipment from behind shield walls prior to maintenance, and shielding of spent filters during removal should minimize the radiation exposures received by plant personnel.

On the basis of the above review, the staff has concluded that the proposed plant design and health physics program provides reasonable assurance that exposures to individuals will be in accordance with the requirements of 10 CFR Part 20 and are acceptable.

12.2 Ventilation

Ventilation systems will be provided for this facility to reduce airborne activity levels and induce air flows from potentially less radioactively contaminated areas to areas with potentially greater radioactivity levels.

The inner or primary reactor containment will be provided with a purge system and an air cleanup system. The air cleanup system will recirculate air through HEPA filters and charcoal adsorbers. The purge system will exhaust air from the primary containment through HEPA filters and charcoal adsorbers to the environment. The outer or secondary reactor containment will be provided with a ventilation purge system that will maintain the secondary containment at a slight negative pressure. This system will exhaust secondary containment air to the environment through HEPA filters and charcoal adsorbers.

The auxiliary building ventilation system will consist of the building air supply system and the air cleanup and exhaust system. The auxiliary building air exhaust system will consist of air clean-up filter trains for the mechanical equipment room, fuel handling area rooms, and common area rooms. Each air cleanup filter train will consist of banks of prefilters, HEPA filters, and charcoal adsorbers. The treated air will be exhausted to the environment.

The turbine building ventilation air will exhaust to the environment without treatment.

The primary reactor containment air cleanup system will have a capacity of 36,000 cfm. Charcoal adsorbers installed in the ventilation air cleanup and purge systems will provide a decontamination factor of 10. In our evaluation of the ventilation systems we used the parameters and the STEFFEG Code given in WASH-1258. We calculate that the iodine-131 releases for the reactor building, auxiliary building, and turbine building will be 0.0005 Ci/yr/reactor, 0.002 Ci/yr/reactor, and 0.003 Ci/yr/reactor, respectively.

The plant ventilation system will be designed to provide air flow from areas of low contamination to areas of progressively greater contamination before final exhausting. Building areas will be maintained at slightly negative pressure with respect to the exterior pressure to reduce exfiltration and contamination of outside areas. The ventilation system will have adequate capacity to limit radioactivity concentrations within plant areas to values listed in 10 CFR Part 20, Appendix b, Table I for normal plant operation and therefore we find it acceptable.

13.0 CONDUCT OF OPERATIONS

13.1 Organization and Qualifications

The plant staff proposed by the applicant will consist of approximately 170 persons under the direction of the Power Plant Superintendent. Reporting to the Power Plant Superintendent and Assistant Power Plant Superintendent will be a Power Plant Results Supervisor with a staff of approximately 25 persons, responsible for plant performance, reactor engineering, chemistry control and instrument maintenance; a Power Plant Operatings Supervisor, with a staff of approximately 43 persons, responsible for plant operations; a Power Plant Maintenance Supervisor, with a staff of approximately 58 persons, responsible for mechanical and electrical maintenance; a Health Physicist, with a staff of approximately 6 persons, responsible for plant health physics activities; an Engineering Unit with approximately 5 engineers and additional supporting groups such as administrative services, public safety officers and storekeepers. This is a customary type of organizational arrangement for two unit operation at the same site.

The shift complement for one unit operation will consist of seven men including one Senior Licensed Operator and two Licensed Operators and for two unit operation will consist of nine men including two Senior Licensed Operators and three Licensed Operators. These shift crew compositions are in accord with the established staff position.

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The applicant has stated that qualifications for plant employees will meet the criteria set forth in ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel.

Technical support for the plant staff will be provided by the TVA Division of Power Production (DPP). The main groups within the DPP supporting Bellefonte will be the Power Plant Maintenance Branch, the Plant Engineering Branch and the Nuclear Operations Coordinator. These groups are currently supporting the operation of the TVA's Browns Ferry Nuclear Station.

We conclude that the proposed organization, and the qualifications of the staff are adequate to provide an acceptable staff and technical support for the safe operation of the plant.

13.2 Training Program

A training program has been established to train the plant staff for the operation of Bellefonte Units 1 & 2. The formal training program for Senior Licensed Operators and Licensed Operators will consist basically of six phases; Basic Nuclear Courses, Plant Technology and Specialist Training, Reactor Operations, Simulator Training. Since it is expected that the applicant will be operating the Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants prior to the completion of the Bellefonte Nuclear Station, TVA expects to draw many previously nuclear trained and experienced personnel for the Bellefonte Nuclear

Plant Staff. Therefore, the specific training any one individual may receive will depend greatly on his previous training and experience. We will review the details of the training program at the operating license review stage.

We have analyzed the proposed training program and conclude that it will meet Regulatory Guide 1.8 (Personnel Selection and Training, which incorporates ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel) and will provide the basis for an adequately trained plant staff.

13.3 Emergency Planning

The applicant has described his preliminary plans for coping with emergencies, including the proposed contents of the TVA Radiological Emergency Plan for the Bellefonte Nuclear Plant. The preliminary plans describe his own organization including responsibilities and delegation of authority, protective measures for accidents affecting both onsite and offsite areas and a description of arrangements made or to be made with local, State and Federal agencies that may be needed in coping with emergencies occurring at the Bellefonte site. The proposed plan is similar to the plan in effect at the Browns Ferry site which includes arrangements currently in effect with the State of Alabama and Federal agencies.

We consider that the applicant's preliminary plans meet the requirements of 10 CFR Part 50, Appendix E, Part II and is, therefore, acceptable.

13.4 Administrative Controls

The applicant has described his intended administrative controls regarding review and audit of facility operation, and plant procedures. They are similar to those in effect for the operation of Browns Ferry Unit No. 1 which have been approved by the staff and are acceptable for this facility.

13.5 Industrial Security

The applicant has submitted a description of the elements to be included in his Industrial Security Plan. The plans include personnel selection policy and provisions that need be considered in the design of facility. We will review the detailed security plan for the Bellefonte Nuclear Station during the operation license review stage.

We conclude that the applicant has described those items important to industrial security at the CP review and meets the intent of Regulatory Guide 1.17 (Protection of Nuclear Plants Against Industrial Sabotage, 6/73).

14.0 INITIAL TESTS AND OPERATION

The applicant has committed to an initial test and startup program to assure that equipment and systems perform in accordance with design criteria. The TVA Division of Power Production has the responsibility for the overall test program administration including the conduct of tests, approval of results, and maintaining records of test results.

We conclude that satisfactory test and startup program can and will be implemented by the applicant.

15.0 ACCIDENT ANALYSIS

The postulated design basis accidents analyzed by the applicant for offsite radiological consequences are the same as those analyzed for previously licensed PWR plants (e.g. North Anna Units 3 and 4). These include a loss-of-coola accident, a steam line break accident, a steam generator tube rupture, a fuel handling accident, a control rod ejection accident, and a rupture of a radioactive gas storage tank in the gaseous radioactive waste treatment system.

We have reviewed these accidents and further evaluated the loss-of-coolant accident and the fuel handling accident. The offsite doses we calculated for these accidents are presented in Table 15.1, and the assumptions we used are listed in Tables 15-2 and 15-3 of this report. Further discussion of the loss-of-coolant accident dose modeling is given in Section 15.1 of this report. All potential doses calculated by the applicant and by us for the postulated accidents are within the 10 CFR Part 100 guideline values.

On the basis of our experience with the evaluation of the steam line break and the steam generator tube rupture accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible reactor coolant system (RCS) and secondary coolant system radioactivity concentrations so potential offsite doses are small.

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At the operating license stage, we will include limits in the Technical Specifications on the RCS and secondary coolant system activity concentrations such that the potential 2-hour doses at the exclusion radius, as calculated by the Regulatory staff for these accidents, will be small fractions of the guideline doses of 10 CFR Part 100. Similarly, we will include limits in the Technical Specifications on gas decay tank activity such that any single failure (such as a relief valve lifting and sticking open) does not result in doses that are more than a small fraction of the 10 CFR Part 100 guideline values.

The control rod ejection accident will be evaluated at the time of the operating license review. This may require the setting of an additional technical specification on the allowable operational RCS leakage into the steam generator secondary side to assure that radiological consequences of this accident are well within the 10 CFR Part 100 dose guidelines.

15.1 Loss-of-Coolant Accident Dose Model

The Bellefonte plant utilizes pressurized water reactors with a low leakage concrete primary containment and a concentric concrete secondary structure forming an enclosure building. The enclosure building along with the fuel handling area and the auxiliary building mechanical equipment zones for both units form an enclosure building region (EBR) which is maintained at a negative differential pressure by engineered safety feature filtration systems. The purpose of the EBR is to further reduce the amount of fission product leakage to the environment that is assumed to occur after the postulated design basis loss-of-coolant accident.

TVA has stated that the entire EBR will be normally maintained at a negative differential pressure. Upon receipt of a high radiation signal, ESF filtration units will be activated such that the EBR is always at a negative differential pressure. In addition, after a postulated loss-of-coolant accident, the annulus atmosphere will be recirculated and mixed before release to the environment. In analyzing the capability of the proposed enclosure building region to further minimize the direct outleakage of fission products, the staff considered three specific points: (1) the minimum negative differential pressure throughout the EBR, (2) the fraction of the primary containment leakage that could bypass the region filtered by the ESF filtration system and be released directly to the atmosphere, and (3) the fraction of the primary containment leakage that is processed by the annulus recirculation filter system as opposed to the once-through auxiliary building filtration system.

At the operating license review stage, the staff will require the applicant to show by appropriate tests that the ESF filtration system will maintain the enclosure building region to a minimum negative differential pressure of .25 inches water gauge under accident conditions following a containment isolation signal, assuming a single failure of an active component in the filtration or isolation systems. The applicant has indicated he will comply with this requirement.

The applicant has indicated that the Technical Specification for bypass leakage will be .5% of the containment leak rate. We will require that this specification include all potential bypass leak paths, including leakage through guard pipes which penetrate the annulus. In addition, we will require a Technical Specification to be set such that not more than 9.5% of the integrated containment leakage can enter the auxiliary building, as opposed to the annulus volume. The applicant has proposed similar Technical Specifications. With these restrictions, we can assume that at least 90% of the primary containment leakage is processed by the annulus recirculation system for the purposes of computing design basis accident doses.

The applicant has provided redundant recombiners for the purpose of controlling any formation of hydrogen after a design basis loss of coolant accident. In the event of failure of both recombiners, the applicant has provided a backup purging mode. Purging would be done through the Secondary Containment Cleanup System (SCCS) provided for the enclosure building region to minimize the radiological consequences of purging. We have evaluated the additional dose an individual might receive due to purging the containment after the design basis accident. Our assumptions are listed in Table 15-1. The combined LOCA and purge dose would be well within the guideline exposures given in 10 CFR Part 100 for the LPZ outer boundary and are acceptable.

15.2 Radioactive Spills

Tanks and equipment containing radioactive liquids located within the auxiliary, turbine,a nd reactor buildings will be serviced with an equipment and floor drainage system. This system will include instrumentation for leak detection, level control for drainage sumps and tanks, and system pump discharge pressures. Abnormal conditions will be annunciated in the control room to warn of need for operator action. The drain system will handle normal leakage, but further provisions will be made to accommodate major leaks. In the case of major spills, where liquid builds up in a room, emergency drain dropout panels will rupture to allow the liquid to drain to the emergency sump located in the lower portions of the building. In this manner radioacitve spills will be contained within the plant structure. Liquid dumped into the emergency sump will be pumped to the liquid radwaste system for treatment and disposal.

Two condensate storage tanks will be located outside. Water stored in these tanks will be makeup water and condensate that has been processed through demineralizers. A borated water storage tank will also be located outside. This tank will be designed to

TABLE 15-1

RADIOLOGICAL ACCIDENT CONSEQUENCES

(DOSE IN REM)

ACCIDENT	EXCLUSION AREA (0.568 Miles) Thyroid	Whole Body	LOW POPULATION ZONE (2 Miles) Thyroid	Whole Body
Loss of Coolant	89	10	76	6.5
Fuel Handling	30	10	3	1.0
Hydrogen Purge Dose			97	1.0

TABLE 15-2

LOCA DOSE CALCULATION INPUT PARAMETERS

Power Level	3763 MWt
Operating Time	3 years
Primary Containment Leak Rate	
(0-24 hours)	.2%/day
(>24 hours)	.1%/day
Iodine Composition	
Elemental	91%
Particulate	5%
Organic	4%
Filter Efficiencies	
Elemental	95%
Particulate	95%
Organic	95%
Minimum Site Boundary Distance (.568 miles)	914 meters
Low Population Zone (2.0 miles)	3218 meters
Accident Duration	30 days
X/Q Values (sec/m^3)	
0-2 hours @ .568 miles	1.8×10^{-3}
0-8 hours @ 2.0 miles	1.8×10^{-4}
8-24 hours @ 2.0 miles	1.2×10^{-4}
24-96 hours @ 2.0 miles	4.8×10^{-5}
96-720 hours @ 2.0 miles	1.3×10^{-5}
Volume of Secondary Building	1.36×10^{6} ft^3
Volume of Primary Containment	3.4×10^{6} ft^3

Recirculation System

Initial Exhaust Flow Rate	500 cfm
Final Exhaust Flow Rate	500 cfm
Total Recirculation Rate	6000 cfm

TABLE 15-3

REFUELING ACCIDENT CALCULATION INPUT PARAMETERS

Shutdown Time	72 hours
Total Number of Fuel Rods in the Core	54,120
Number of Fuel Rods Involved in the Refueling Accident	264
Power Peaking Factor	1.7
Iodine Fractions Released from pool	
Elemental	75%
Organic	25%
Filter Efficiencies	
Elemental	95%
Organic	95%
X/Q Values, Sec/M^3	
0-2 hours @ .568 miles	1.8×10^{-3}
0-2 hours @ 2 miles	1.8×10^{-4}

TABLE 15-4

HYDROGEN PURGE DOSE INPUT PARAMETERS

Power Level (MWt)	3763
Volume of Primary Containment (ft^3)	3.4×10^6
Purge Duration (days)	30
Holdup Time in Containment (days) Prior to Purge Initiation	20
Filter Efficiency for Iodine (%)	95
Purge Rate (SCFM)	25
4-30-day X/Q (sec/m^3)	1.3×10^{-5}

Quality Group B and seismic Category I according to Regulatory Guide 1.26 and 1.29, respectively. No provisions will be made to retain spills from these tanks due to the insignificant radioactivity content of the condensate storage tanks and the design standards for the borated water storage tanks.

Based on our evaluation of the applicant's design we conclude that provisions incorporated to monitor tank levels, retain liquid spillage and sample, process, and dispose of spillage, will be adequate to prevent potential uncontrolled releases of radioactive materials to the environs from this source.

15.3 Abnormal transients and Accidents

The applicant's safety analysis evaluates the ability of the Bellefonte plant to operate without undue hazards to the health and safety of the public. Two basic groups of events pertinent to safety are investigated by the applicant; abnormal transients and postulated accidents. All transients and accidents have been evaluated for a core power of 3600 MWt except the LOCA which has been evaluated at 3760 MWt. The environmental consequences of accidents have been evaluated at a core power level of 3763 MWt.

The criterion, adopted to assure that the reactor coolant pressure boundary integrity is maintained, is that the system pressure shall remain below the code pressure limits set forth in ASME code Section III (110% of RCS design pressure). The criterion adopted to ensure that no fuel damage has occured is that the DNBR must be greater than 1.32 throughout the transient.

The applicant has submitted analyses of abnormal transients and has shown that the integrity of the reactor coolant system pressure boundary has been maintained and that the minimum DNBR exceeded 1.32 for all transients. The maximum pressure transient was identified (B&W Topical Report BAW-10043, Supplement 1), as the complete loss of main feedwater from full power, resulting in a peak RCS pressure of approximately 2670 psia.

We conclude that the evaluation of abnormal transients indicates that the transients presented do not lead to unacceptableon-sequences and are therefore acceptable.

The applicant has evaluated a broad spectrum of accidents that might result from postulated failures of equipment, or their maloperation. These highly unlikely accidents (design basis accidents) that are representative of the spectrum of types and physical locations of postulated events and that involve the various engineered safety feature systems have been analyzed in detail.

The accidents reviewed in Chapter 15 of the SAR included the following:

(1) Locked Rotor

(2) Loss-of-Coolant (LOCA)

(3) Steamline Rupture

The locked rotor accident was analyzed by postulating an instantaneous seizure of one reactor coolant pump rotor. The reactor flow would decrease rapidly and a reactor trip would occur as a result of a high power-to-flow signal. The thermal analysis of the hot fuel element in the core was performed using 102% of rated power and nominal flow, pressure, and inlet temperature. The analysis revealed that 3% of the pins experienced a DNBR less than 1.32 and one percent experienced a DNBR less than 1.0. The applicant concluded that there was no fuel cladding failure since the maximum clad temperature was calculated to be 1060 F.

The loss-of-coolant accident analyses referenced B&W Topical Reports BAW-10065, BAW-10065 Supplement 1, "Multinode Analysis of B&W's 205-Fuel Assembly (Mark C Design) Nuclear Plant During a Loss-of-Coolant Accident", August, 1973 and BAW-10074, "Multinode Analysis of Small Breaks for B&W's 205 Fuel-Assembly Nuclear Plants with Internals Vent Valves", November, 1973. The evaluation model described in the AEC Interim Acceptance Criteria and Amendments for Emergency Core Cooling Systems was used in the break analyses except for the deviations noted in these topical reports. Pursuant to the latest Acceptance Criteria for ECCS published in the Federal Register on January 4, 1974, the applicant will be required to resubmit the LOCA analyses satisfying the requirements of the new Criteria.

For a core power of 3760 MWt, the largest double-ended cold leg break at the pump discharge resulted in the highest peak cladding temperature of 1929 F. The metal water reaction for the whole core was 0.112% which is well below the 1% limit specified in the Criteria.

Loss-of-secondary-coolant analyses have been performed to determine the effects and consequences due to a double-ended steam line rupture. A 29.53-inch I.D. steamline rutpure, between the steam generator and the main steam isolation valves, and a 42-inch steamline rupture downstream of the main steam isolation valves were analyzed. The analyses assumed that the reactor was operating at rated power prior to the accident. The reactor remained subcritical even with the maximum worth control rod assumed withdrawn.

In accordance with WASH-1270, "Anticipated transients Without Scram for Water-Cooled Power Reactors", the applicant will submit the required evaluation for Bellefonte by October 1, 1974. Any design changes required to make ATWS consequences acceptable for the Bellefonte plant will be delinated in this evaluation.

16.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. Final Technical Specifications will be developed and evaluated at the operating license stage. However, in accordance with Section 50.34 of 10 CFR Part 50, an application for a construction permit is required to include preliminary Technical Specifications. The regulations require an identification of and a justification for the selection of those variables, conditions, or other items that are determined, as a result of the preliminary safety analysis and evaluation, to be probable subjects of Technical Specifications for the facility, with special attention given to those items that may significantly influence the final design.

We have reviewed the proposed Technical Specifications presented in Section 16 of the PSAR with the objective of identifying those items that would require special attention at the construction permit stage, to preclude the necessity for any significant change in design to support the final Technical Specifications. The proposed Technical Specifications are similar to those being developed for or in use for plants of a design similar to Bellefonte.

On this basis we conclude that the proposed Technical Specifications are acceptable.

17.0 QUALITY ASSURANCE

The description of the Quality Assurance (QA) Program for the design and construction of Bellefonte Nuclear Plant, Units 1 and 2 is contained in Section 17 of the PSAR, as amended. Our evaluation of the QA Program is based on a review of this information, other applicable documentation, and related discussions with the applicant to determine the measures to be implemented by TVA and its principal contractor, B&W, to achieve compliance with the requirements of Appendix B to 10 CFR Part 50. Our review of the TVA QA Program includes discussions with Regulatory Operations, Region II, relating to their acceptance of the QA Manual and their inspection reports on TVA's compliance with Appendix B to 10 CFR Part 50.

This section reflects our review of the QA Program for this facility to ascertain:

a. that a QA organization for the facility design, procurement, and construction is established to develop and execute a QA Program in compliance with 10 CFR Part 50 Appendix B.

b. that this organization is structured such that the individuals responsible for QA can effectively manage and control the QA/QC functions.

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c. that personnel performing QA functions within the applicant's and contractor's organizations have sufficient authority, organizational freedom, and independence to perform their functions effectively without undue influence from project cost and schedule and without reservation, and

d. that the QA Program embodies sufficient policies and procedures to fully implement 10 CFR Part 50 Appendix B for safety related structures, systems, and components.

17.1 Tennessee Valley Authority

TVA is to perform the architect-engineering design, procurement, construction, and operation of this facility.

Three organizational elements of TVA will be involved. As shown in Figure 17.1, these are (1) the Office of Power, (2) the Division of Purchasing, and (3) the Office of Engineering Design and Construction (OEDC). The Manager of OEDC has overall responsibility for quality assurance during design and construction. Figure 17.2 shows the QA organization of OEDC. The management of the QA Program in engineering, design, and procurement is the responsibility of the Director of the Division of Engineering Design.

The management of the QA Program in construction is the responsibility of the Director of the Division of Construction. The Quality

Assurance Manager of OEDC has the responsibility of determining that the facility QA Program meets the requirements of Appendix B to 10 CFR 50. He has the responsibility of auditing the performance of the Division of Engineering Design and the Division of Construction. Figure 17.3 shows the staff supervised by the Quality Assurance Manager of OEDC.

The Division of Engineering Design has a Quality Assurance Staff (See Figure 17.2), managed by a Chief, which reviews and coordinates the department's QA Program within the department and as it interfaces with others. It performs internal audits within the department.

There is also an Inspection and Testing Branch within the Division of Engineering Design. This branch is responsible for assuring that suppliers of safety related structures, systems, and components for the facility meet the applicable criteria of Appendix B to 10 CFR Part 50. Inspectors from this branch perform inspections and surveillance at suppliers' facilities.

The staff concludes that these organizational arrangements will provide sufficient independence and are therefore satisfactory for engineering, design, and procurement QA functions.

Within the Division of Construction, the Construction Engineer is responsible for quality assurance in the receipt and storage of materials and in field fabrication, construction, erection, and preliminary tests and checks for Bellefonte 1 and 2. As shown in Figure 17.4, the staff of the Construction Engineer includes Engineers and Principal Engineers. The Engineers prepare detailed inspection and test procedures; and, in response to our questions, the applicant stated that these engineers are trained inspectors in their specific discipline, that they perform physical inspections, and that they make formal acceptance of material and equipment. In addition, TVA has stipulated that the Construction Engineer and his staff of field engineers will make no judgment of acceptance of inspection findings which are not within the literal limits of specifications and their tolerances as defined by the design engineering organizations. The Principal Engineers, on the staff of the Construction Engineer, perform independent checks and audits of these functions.

Figure 17.4 also shows that the Construction Engineer, responsible for construction quality assurance, and the Construction Superintendent, responsible for construction cost and schedule, are on the same organizational level; i.e. both report to the Project Manager. TVA has recently proposed organizational changes under the Division of Construction which include the addition of a QA Staff Supervisor reporting to the Director of Construction. The QA Staff Supervisor has reporting to him an onsite QA representative with responsibilities for auditing the QA activities on site.

We will require the details of TVA's organizational changes be documented in the PSAR prior to issuance of a construction permit.

The PSAR, as amended, states that TVA has implemented an overall QA Program for this facility. This program is documented in the TVA Quality Assurance Manuals and the B&W Quality Assurance Manuals, all of which provide details of how the design, procurement, and construction of this facility will comply with Appendix B of 10 CFR Part 50. TVA has listed the contents of its QA Manuals related to each of the criteria of Appendix B. Our review of this listing shows that the QA Program requirements of Appendix B have been addressed by implementing provisions in these manuals.

Based on our review of the QA Program as described in Section 17 of the PSAR and on its implementation in accordance with the contents of the QA Manuals, we conclude that except for the recent organizational changes noted above which have not been documented, an acceptable QA program has been documented and that this program is in compliance with Appendix B of 10 CFR 50.

TVA has established a three level program to obtain adequate quality control and quality assurance. The first level, quality control, is provided by supplier inspectors during manufacturing and by TVA inspectors (Engineers-Figure 17.4) of the Division of Construction during construction. This first level also includes the review of design drawings and procurement documents by O&MC.

The second level is surveillance conducted by the Inspection and Testing Branch of the Division of Engineering Design and by the Principal Engineers of the Division of Construction. The third level is the Audit Program which includes:

1. OEDC management audits performed under the direction of the Quality Assurance Manager.
2. Internal audits performed by the QA Staff of the Division of Engineering Design and by the QC and Records Supervisor of the Division of Construction.
3. Vendor audits performed by the QA Staff of the Division of Engineering Design and site contractor audits performed by the Construction Engineers Staff of the Division of Construction.

Based on our review of the QA Program description for this facility as contained in the PSAR and in other applicable documentation, we find that the program provides for sufficiently detailed procedures, requirements, and elements of control to assure that all safety related structures, systems, and components are designed, constructed, installed, inspected, and tested in accordance with the requirements of 10 CFR Part 50 Appendix B.

17.2 Babcock & Wilcox

TVA has purchased the Nuclear Steam Supply System (NSSS) from B&W, and B&W is responsible for developing quality control requirements and procedures for the NSSS and assuring that these requirements and procedures are followed. The B&W Power Generation Group (PGG) is comprised of a number of Operating Divisions (See Figure 17.5). Each PGG facility that is fabricating nuclear equipment is periodically audited by the Quality Assurance organization of Group Operation Services of PGG. The Nuclear Power Generation Division (NPGD) of PGG is responsible for supplying the nuclear plant. Organization elements of NPGD are also shown on Figure 17.5. Overall contract responsibility for supplying the NSSS is assigned to a Project Manager within NPGD.

Quality management within each PGG Division has stop work authority. The Quality Control Inspectors are located in each PGG Division. An organization chart for quality assurance within NPGD is included in Figure 17.5. Responsibility for QA activities is assigned to the Quality Assurance Manager. The manager of Quality Assurance reports directly to the Division Vice President. In response to our questions Figures 17.6 and 17.7 were supplied. These figures show similar QA organizational independence for the Nuclear Equipment Division and the Commercial Nuclear Fuel Plant.

Since the quality assurance organizations are independent of manufacturing organizations and since the Quality Assurance Managers are on the same organizational reporting level as those managers directly responsible for cost and schedule, we conclude that B&W

QA organizations have sufficient independence and authority to properly carry out their QA responsibilities without undue influence and pressures from those organizations directly responsible for project costs and schedules.

B&W's QA Program includes coverage of each of the 18 criteria of Appendix B to 10 CFR Part 50. The PSAR includes a table showing the B&W QA procedures as they address each criterion and a summary table of the QA procedures required of B&W suppliers of safety related structures, systems, and components. Based on our review of these tables and Section 17.1B of the PSAR, we conclude that an acceptable QA Program has been documented and that this program is in compliance with Appendix B of 10 CFR Part 50.

B&W uses three levels of organizational control to evaluate the QA Program. At the first level, suppliers and B&W divisions other than NPGD are required to have internal audit programs. At the second level, process audits are conducted by the Nuclear Power Generation Division and by other divisions to assure functional areas are adequately covered. The third level consists of quality system audit of involved B&W facilities by the QA organization of Group Operation Services of the Power Generation Group.

NPGD surveillance of suppliers during fabrication, inspection, test, and shipping of safety related structures, systems, and components is planned and conducted in accordance with Product Surveillance Plans which include inprocess hold points. The Product Surveillance Plans provide the field representatives with instructions on source surveillance, critical process verification, and final inspection. B&W holds the supplier responsible for inspection and testing. The field representative assures that it is done in accordance with preapproved procedures.

The organization structure and functional responsibility assignments are such that attainment of quality objectives is accomplished by individuals assigned responsibility for specifying quality or for performing work to specifications. We find that verification of conformance to established quality requirements is accomplished by those who do not have direct responsibility for specifying or for performing work to specifications. The QA Program of B&W contains those quality system elements necessary to provide assurance that systems and components important to safety meet applicable codes, standards, and regulatory requirements and the quality requirements of TVA.

17.3 Regulatory Operations

The Directorate of Regulatory Operations (RO) has examined the quality assurance program for this facility to determine its conformance with commitments in the application for a construction permit and with the requirements of Appendix B to 10 CFR 50. The RO examination included: (1) a detailed review of the application for the construction permit; (2) attendance at regulatory meetings with Tennessee Valley Authority, in which the proposed program was explained in detail, (3) in-depth examinations of the quality assurance program, organization and procedures for design and procurement at the applicant's Knoxville, Tennessee offices; (4) detailed discussions of the results of RO inspection findings with management of Tennessee Valley Authority, including the applicant's proposed corrective measures. Based on these actions, it is the opinion of the Directorate of Regulatory Operations that: (1) the applicant has provided and implemented a quality assurance program commensurate with the project status, and his program, including corrective commitments, conforms to 10 CFR 50, Appendix B, and the application commitments; (2) the applicant has implemented his quality assurance program through his Division of Engineering Design (DED) and nuclear steam system supplier design and procurement activities, and by conducting qualification and performance audits of major component suppliers in accordance with the provisions of Criterion XVIII of Appendix B, 10 CFR 50; and (3) the nuclear steam system supplier's quality assurance program has been made an integral part of the applicant's quality assurance program.

Regulatory Operations will accompany the applicant on selected audits of major contractors and suppliers, and perform additional inspections as necessary prior to initiation of construction activities to examine the current status of development and implementation of the overall quality assurance program for the Bellefonte nuclear project.

17.4 Conclusions

Based on our detailed review and evaluation of the QA program description of TVA and B&W described in the PSAR and an in-depth QA audit by our Directorate of Regulatory Operations, we conclude that the QA organizations provide QA personnel with sufficient authority and organization freedom to perform their crucial functions effectively and without reservation; and that the QA Programs describe adequate QA procedures, requirements, and controls demonstrating that quality related activities will be conducted in accordance with the requirements of Appendix B to 10 CFR Part 50.

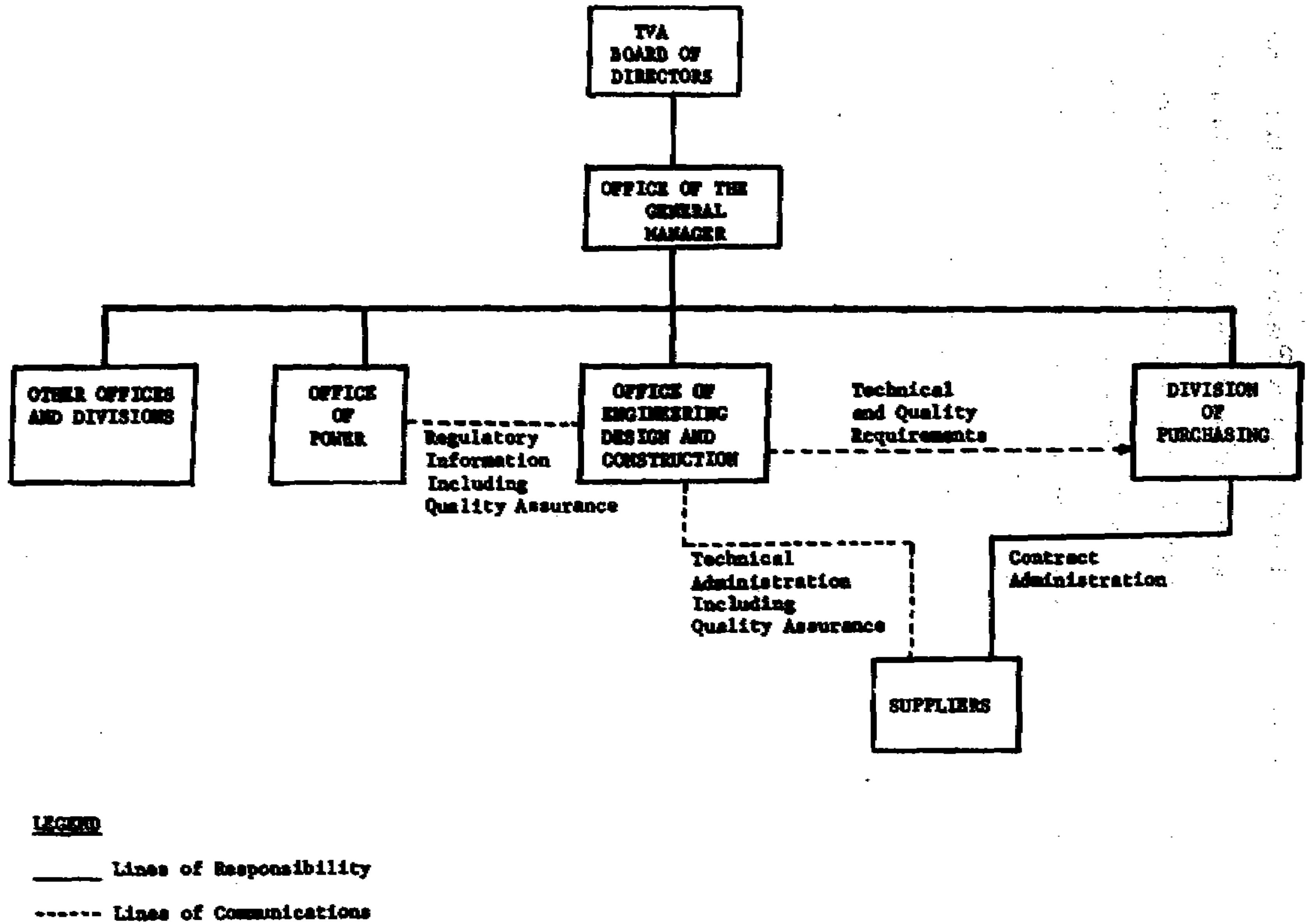
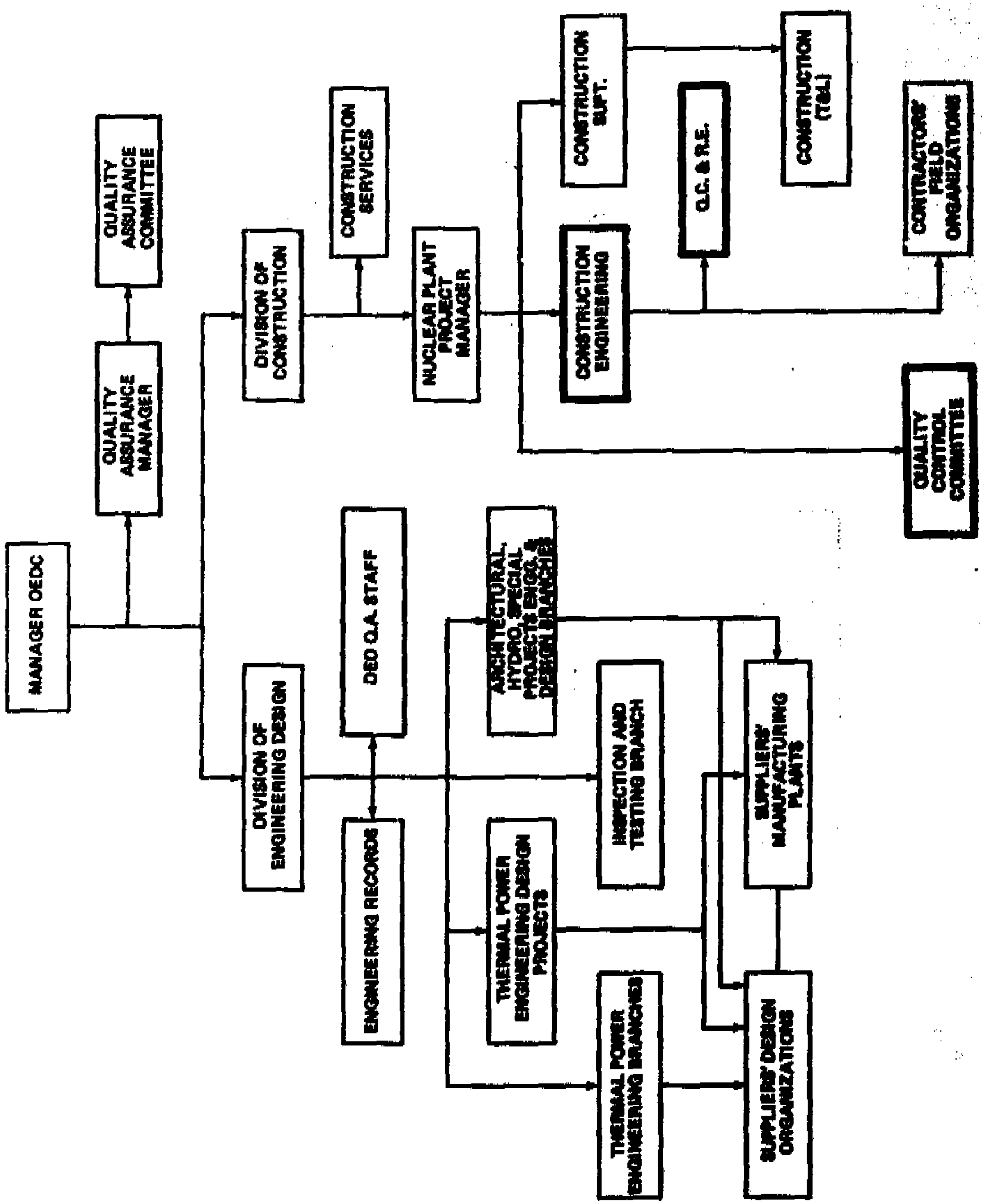


Figure. 17.1. TVA Organization



MANAGER OEDC
QUALITY ASSURANCE MANAGER
QUALITY ASSURANCE COMMITTEE
DIVISION OF CONSTRUCTION
CONSTRUCTION SERVICES
NUCLEAR PLANT PROJECT MANAGER
CONSTRUCTION SUPT.
CONSTRUCTION (TS&I)
Q.C. & R.E.
CONSTRUCTION ENGINEERING
CONTRACTORS' FIELD ORGANIZATIONS
QUALITY CONTROL COMMITTEE
DIVISION OF ENGINEERING DESIGN
OED Q.A. STAFF
ENGINEERING RECORDS
ARCHITECTURAL, HYDRO, SPECIAL PROJECTS ENGG. & DESIGN BRANCHES
INSPECTION AND TESTING BRANCH
THERMAL POWER ENGINEERING DESIGN PROJECTS
SUPPLIERS' MANUFACTURING PLANTS
THERMAL POWER ENGINEERING BRANCHES
SUPPLIERS' DESIGN ORGANIZATIONS

Figure 17.2

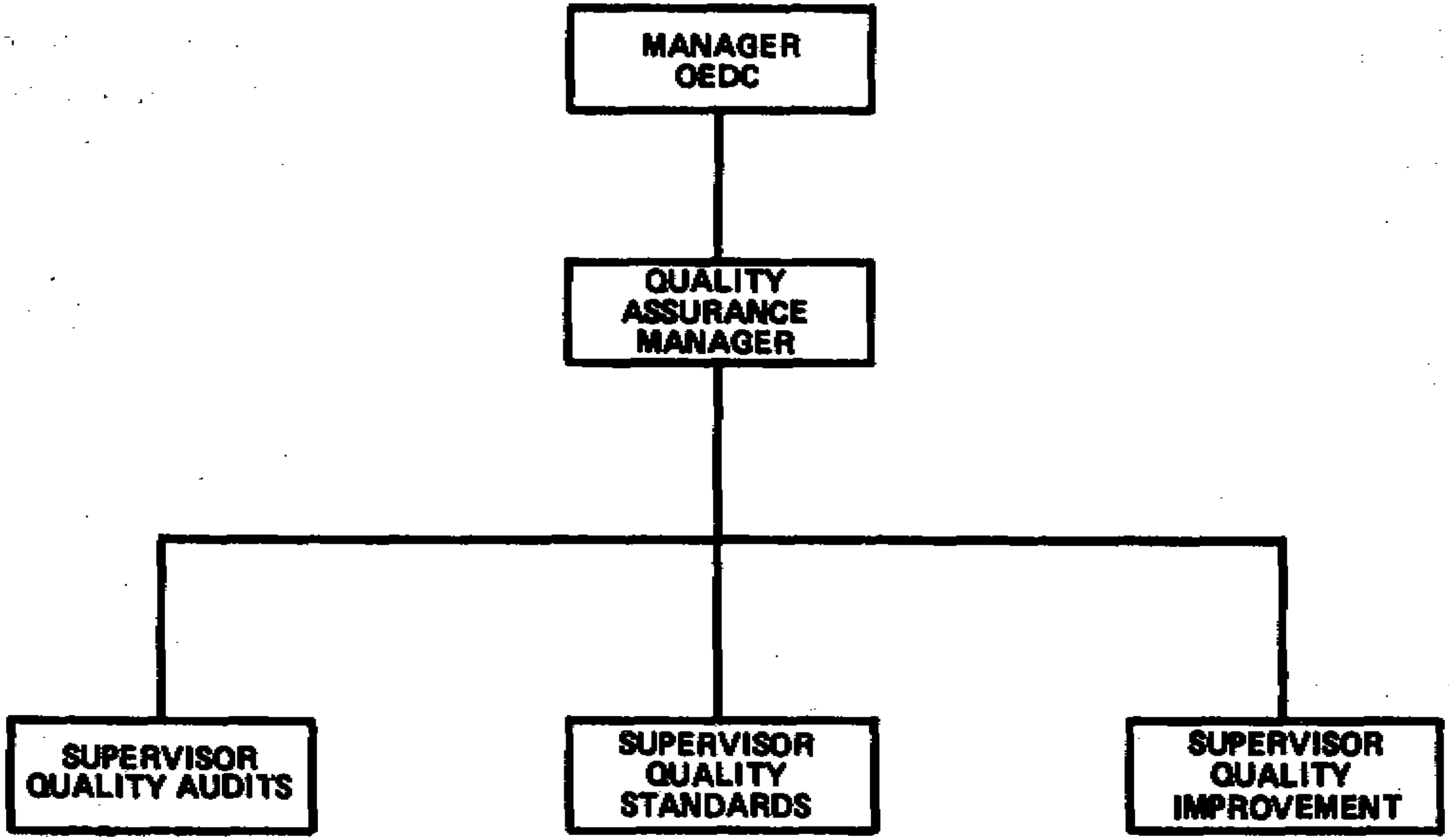


Figure. 17.3. OEDC ORGANIZATION

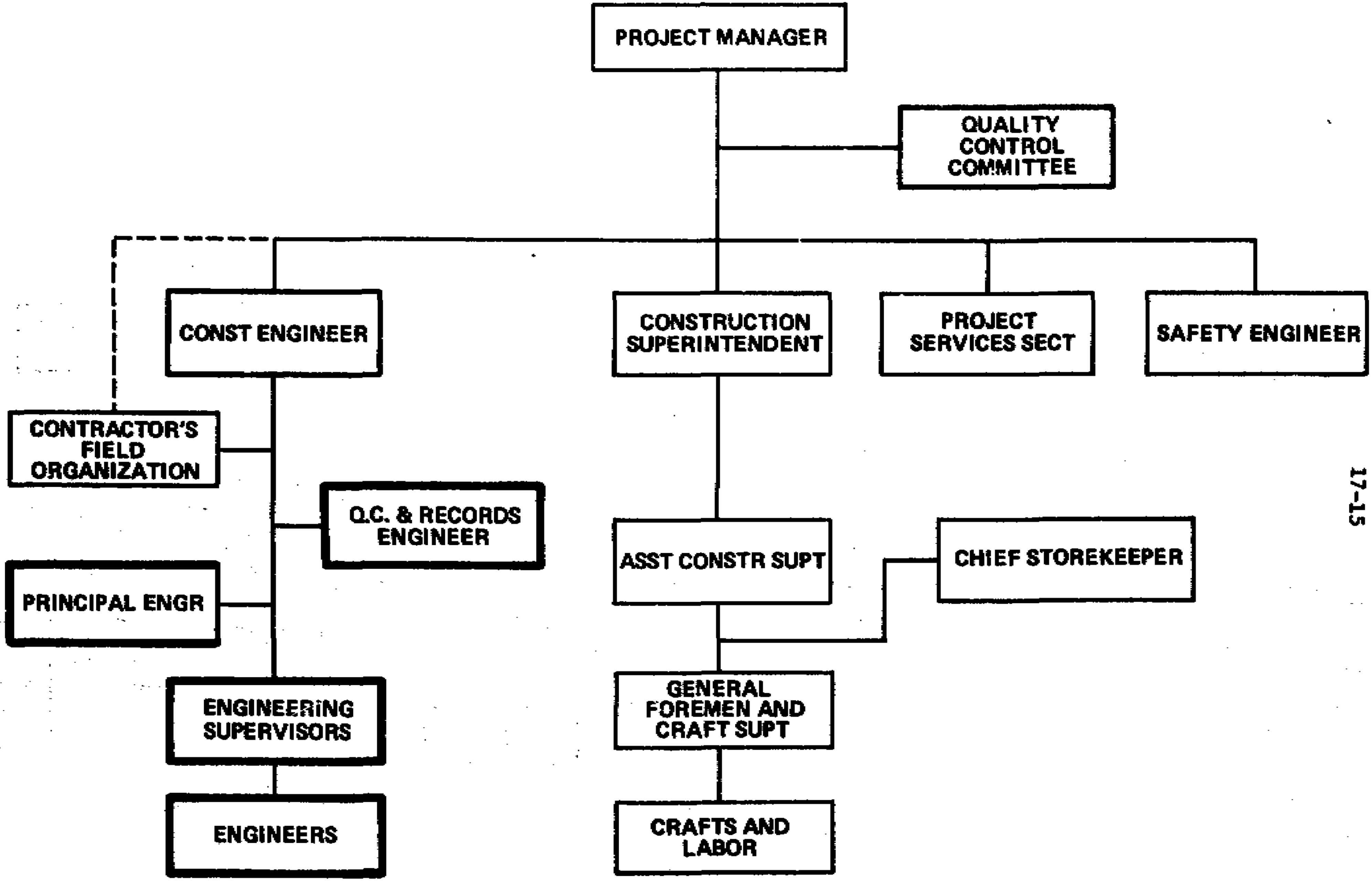


Figure 17.4 NUCLEAR PLANT CONSTRUCTION PROJECT ORGANIZATION

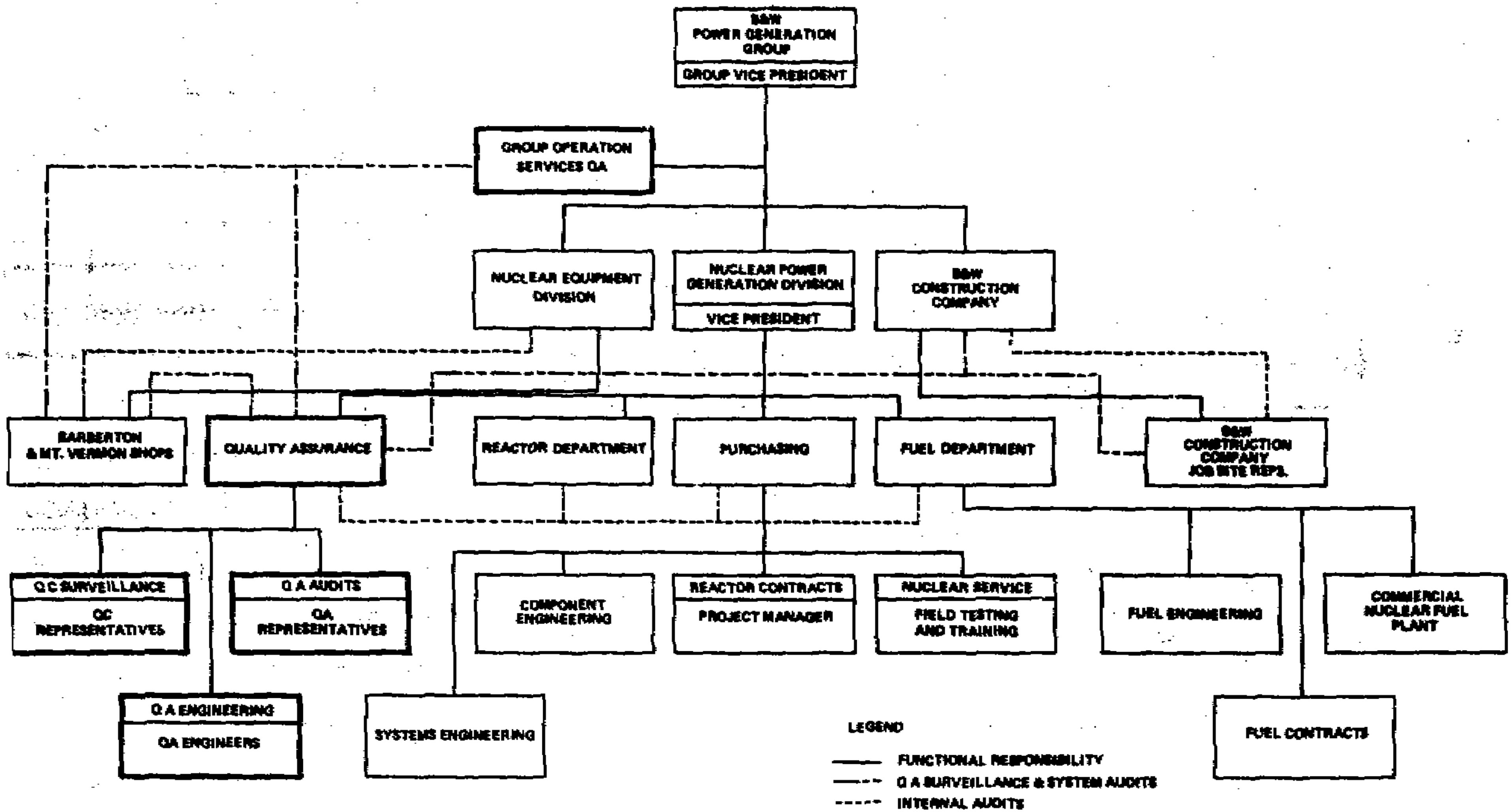


Figure 17.5 B&W POWER GENERATION GROUP

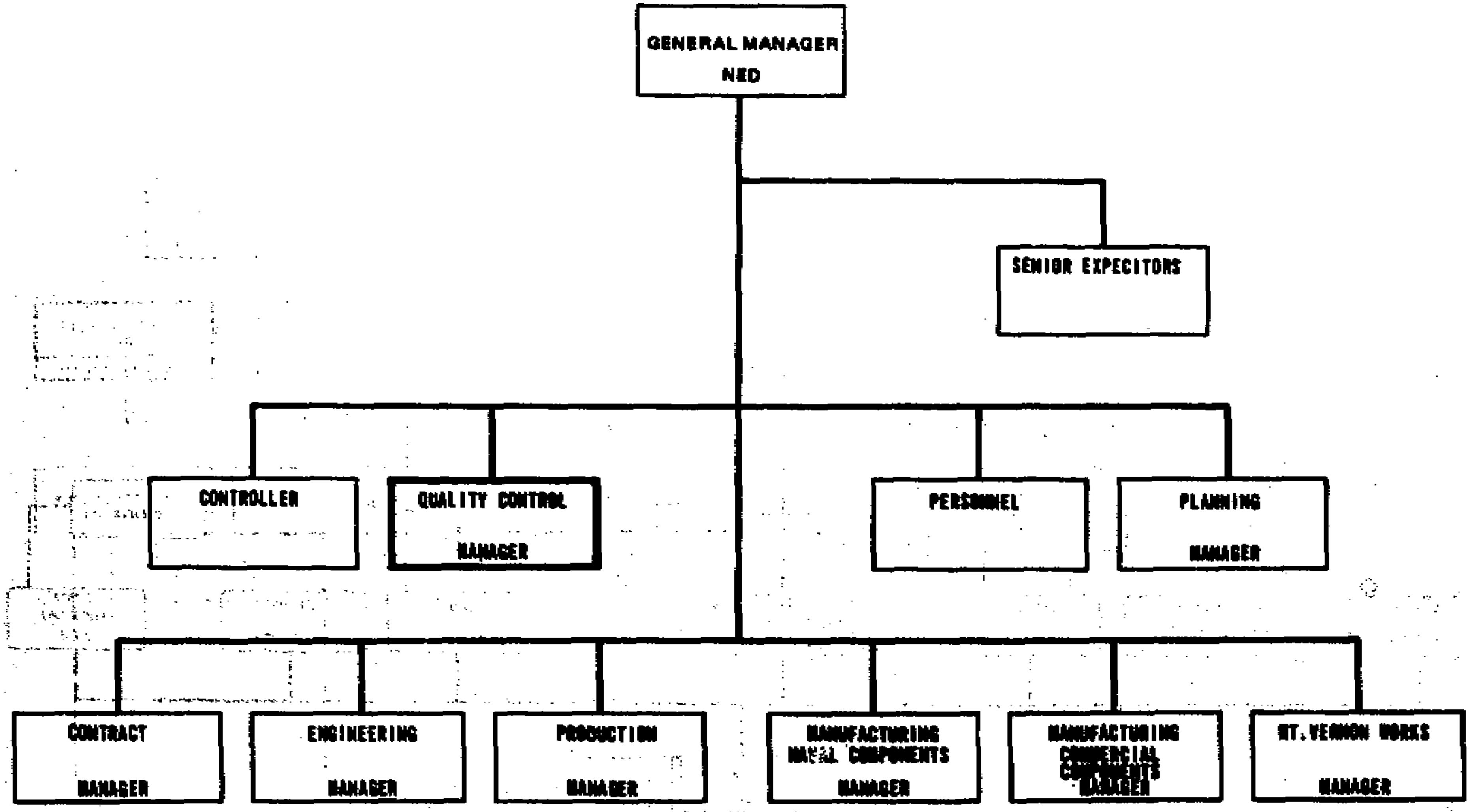


Figure. 17.6. NUCLEAR EQUIPMENT DIVISION

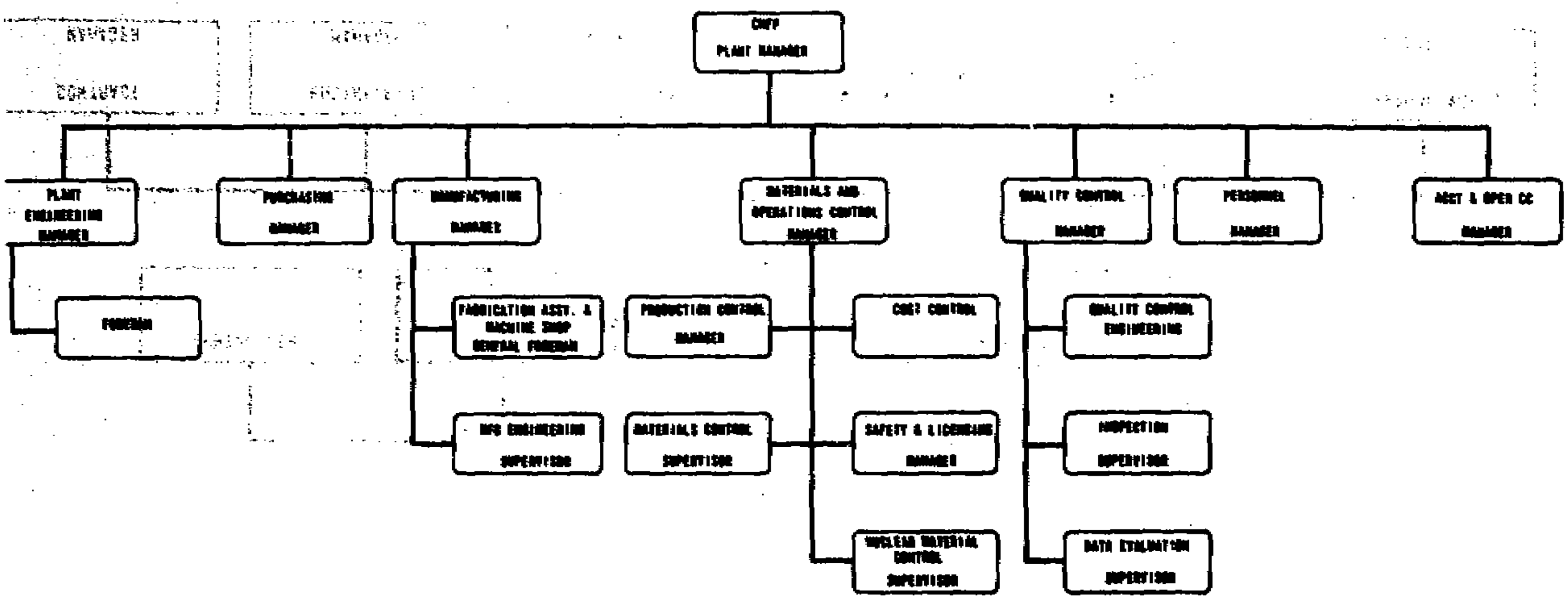


Figure 17.7. CNFP ORGANIZATION

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The application for the Bellefonte plant will be reviewed by the ACRS. We intend to issue a Supplement to this Safety Evaluation after the Committee's report to the Commission is available relative to this review. The Supplement will append a copy of the Committee's report, will address the significant comments made by the Committee, and will describe steps taken by the staff to resolve any issues raised as a result of the Committee's review.

19.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all the directors and principal officers of the applicant are citizens of the United States. TVA is a corporate agency of the Federal Government.

The applicant is not owned, dominated or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we find that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility construction permit are Paragraph 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. We have reviewed the financial information presented in the application and the amendments thereto regarding financial qualifications. Based on this review and consideration of financial data generally available to the financial analyst, we have concluded that the Tennessee Valley Authority is financially qualified to design and construct the proposed facility to be known as Bellefonte Nuclear Plant Units 1 and 2.

Our conclusion is based on the following facts and considerations:

1. Applicant estimates the cost of construction of the plant, including certain transmission facilities and other associated costs, and the initial reactor cores will total $804.8 million. The details of these estimates are summarized below:

	Unit 1 (millions)	Unit 2 (millions)
Total nuclear production plant costs	$350.0	$345.0
Transmission, distribution, and general plant costs	18.3	18.2
Nuclear fuel inventory cost for first core	38.6	34.7
Total	$406.9	$397.9

The Regulatory staff has reviewed the estimated construction costs of the nuclear production plant and has found them to be on the low side. Revised cost estimates are being prepared by the applicant. The staff intends to evaluate these estimates as part of its preparation of financial qualifications testimony to be included in a Supplement to the Safety Evaluation Report. The earliest and latest estimated dates for completion of the construction of each unit are stated below:

	Earliest Date	Latest Date
Unit 1	March 1, 1979	September 1, 1979
Unit 2	December 1, 1979	June 1, 1980

2. The proposed facility is a necessary part of the applicant's continuing expansion of facilities to provide the electric energy for new growth and for the increasing demands placed on its system from increased usage by its customers. According to the applicant, the greatest part of this growth has been and will be due to rising standards of living, industrial development with better job opportunities and incomes, and a rising population.
3. Construction of the plant will be financed as an integral part of the applicant's power facilities construction program. New power facilities are financed largely from borrowings made through the sale of power bonds and notes and in part from power revenues remaining

after covering operating and other costs. During the years 1974 through 1980, the latest year the construction of the subject facility is estimated to be completed, sources of funds for construction and nuclear fuel expenditures are estimated by the applicant to be internally generated funds (39%) and proceeds from the sale of power notes and bonds (61%). During the years 1968-1973, internally generated funds and proceeds from the sale of power notes and bonds accounted for 21% and 79%, respectively, of construction and nuclear fuel expenditures.

4. Section 15d of the TVA Act (Tennessee Valley Authority Act of 1933, as amended) authorizes TVA to issue bonds, notes, and other evidences of indebtedness up to a total of $5 billion outstanding at any one time (of which $2.3 billion is outstanding) to assist in financing its power program. Debt service on these obligations, which is payable solely from TVA's net power proceeds, has precedence over payments to the U.S. Treasury based on a return on the net appropriation investment in power facilities plus repayments of such investment. When the need for an increase in the bond ceiling arises, the applicant will submit its recommendation in accordance with the Office of Management and Budget Circular No. A-19, Revised, dated July 31, 1972.

Neither the TVA act nor the Basic Tennessee Valley Authority Power Bond Resolution (Indenture) requires that TVA meet specific interest coverage ratios, debt coverage ratios, or debt to equity ratios.

However, Section 15d(f) of the TVA Act requires that TVA maintain rates sufficiently high to meet its financial obligations, to protect its bondholders, and to protect the equity of the Federal government.

5. Information presented in TVA's "power annual report" for the fiscal year ended June 30, 1973 indicates that operating revenues from the power program totaled $749.4 million. Operating expenses were stated at $577.4 million, of which $89.5 million represented depreciation. Interest on long-term debt was earned 2.2 times. Net income totaled $106.4 million, of which $53.8 million was paid to the U.S. Treasury as a return on Federal government appropriations invested in the power system. The dividend is determined each year by applying the Federal government's average interest rate payable on marketable Treasury securities at the beginning of the fiscal year to the Federal government's net appropriation investment in TVA facilities at the same date. The return of $53.8 million was computed by applying the average interest rate of 5.099% to the net appropriation investment of $1,054.8 million. Financial ratios based on TVA's balance sheet for its entire operations as of June 30, 1973 indicate an adequate financial conditions, e.g., long-term debt to total capitalization - 41%, and to net utility plant - 39%; net plant to capitalization - 1.06; and investment and surplus to total assets - 50%. The record of TVA's power program during the

fiscal years 1971-73 shows that operating revenues increased from $598.0 million in fiscal year 1971 to $749.4 million in fiscal year 1973 and net investment in utility plant from $3,635.9 million to $4,567.6 million. However, net income declined from $119.0 million in fiscal year 1971 to $106.4 million in fiscal year 1973, and the number of times interest was earned declined from 4.0 to 2.2. Moody's Investors Service rates TVA's power bonds as Aaa (highest quality). More recent data included in TVA's Power Quarterly Report issued in February 1974 dicates that operating revenues from operations increased from $683.2 million in calendar year 1972 to $804.0 million in calendar year 1973. However, net income declined from $103.0 million to $81.4 million.

A summary analysis reflecting various ratios and other pertinent data is attached as Appendix B.

21.0 CONCLUSIONS

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

1. The applicant has described the proposed design of the facility including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;
2. Such further technical or design information as may be required to complete the safety analysis and which reasonably can be left for later consideration will be supplied in the final safety analysis report;
3. Safety features or components which require research and development have been described by the applicant and the applicant

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has identified, and there will be conducted, research and development programs reasonably designed to resolve safety questions associated with such features or components;

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

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

6. The applicant has estimated the costs reasonably and is qualified financially to design and construct the proposed facility; and

7. The issuance of permits for construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

BIBLIOGRAPHY

(Documents referenced in or used to prepare the Safety Evaluation Report for the Bellefonte Nuclear Plant)

1. Preliminary Safety Analysis Report with Amendments 1 through 10 for the Bellefonte Nuclear Plant, Units 1 and 2 (Docket Nos. 50-438 and 50-439).

2. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Rev. 1, October 1972.

3. United States Atomic Energy Commission Rules and Regulations, 10 CFR:

   Part 1, Statement of Organization and General Information

   Part 2, Rules of Practice

   Part 9, Public Record

   Part 20, Standards for Protection Against Radiation

   Part 50, Licensing of Production and Utilization Facilities

   Part 100, Reactor Site Criteria

4. United States Atomic Energy Commission Regulatory Guides.

5. National Environmental Policy Act of 1969 (NEPA).

6. Gross, E., "The National Air Pollution Potential Forecast Program," <u>ESSA Technical Memorandum WBTM NMC 47</u>, National Meteorological Center, Washington, D.C., 1970.

7. Holzworth, G. C., "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," <u>AP-101</u>, Environmental Protection Agency, Office of Air Programs, Research Triangle Park, North Carolina, 1972.

8. Huschke, R. E., "Glossary of Meteorology," American Meteorological Society, Boston, Massachusetts, 1959.

9. Korshover, J., "Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936-1965," <u>Public Health Service Publication No. 999-AP-34</u>, Cincinnati, Ohio, 1967.

10. List, R. J. (ed.), "Smithsonian Meteorological Tables, Smithsonian Institution," Washington, D.C., 1971.

11. Pasquill, F. and Smith, F. B., "The Physical and Meteorological Basis for the Estimation of Dispersion," 1970. Paper presented at the Second International Clean Air Congress of the International Union of Air Pollution Prevention Associations, Washington, D. C.

12. SELS Unit Staff, National Severe Storms Forecast Center, "Severe Local Storm Occurrences, 1955-1967," ESSA Technical Memorandum WBTM FCST 12, Office of Meteorological Operations, Silver Spring, Maryland, 1969.

13. Smith, M. E. (ed.), 1968: Recommended Guide for the Prediction of the Dispersion of Airborne Effluents. The American Society of Mechanical Engineers, New York, N.Y.

14. Slade, D. H. (ed.), "Meteorology and Atomic Energy - 1968"." TID-24190, National Technical Information Service, Springfield, Virginia, 1968.

15. Thom, H. C. S., "Tornado Probabilities," Monthly Weather Review, October-December 1963, pp. 730-737.

16. Thom, H. C. S., "New Distributions of Extreme Winds in the United States," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers - July 1968, pp. 1787-1801.

17. Turner, D. B., "Workbook of Atmospheric Dispersion Estimates," Public Health Service Publication No. 999-AP-26, Cincinnati, Ohio, 1970.

18. U.S. Department of Commerce, Environmental Data Service: "Local Climatological Data, Annual Summary with Comparative Data - Chattanooga, Tennessee." Published annually through 1972.

19. The Computer program used to calculate the X/Q values: Nuclear Power Station Evaluation Program (in FORTRAN code), J. R. Sagendorf, ARL/NOAA programmer. Program available at the USAEC, Directorate of Licensing, Bethesda, Md., or at the Air Resources Laboratory, NOAA, Field Research Office, Idaho Falls, Idaho.

20. "Wind Forces on Structures," Final Report of the Task Committee on Wind Forces of the Committee on Load and Stresses of the Structural Division, Transactions of the American Society of Civil Engineers, 345 East 47th Street, New York, N.Y. 10017, Paper No. 3269, Vol. 126, Part II, 1961, p. 1124-1198.

21. American Institute of Steel Construction, "Specification for Design, Fabrication & Erection of Structural Steel for Buildings," 101 Park Avenue, New York, New York, 10017, 1969.

22. American Concrete Institute, "Building Code Requirements for Reinforced Concrete (ACI 318-1971)," P. O. Box 4754, Redford Station, Detroit, Michigan 48219.

23. Proposed Standard Code for Concrete Reactor Vessels and Containments, Proposed Section III, Division 2; The American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N. Y. 10017.

24. A. Amirkian, "Design of Protective Structures," Bureau of Yards and Docks, Publication No. NAVDOCKS P-51, Department of the Navy, Washington, D. C., August 1950.

25. R. C. Gwaltney, "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants," USAEC Report ORNL-NSIC-22, September 1968.

26. Williamson and Alvy, "Impact Effects of Fragments Striking Structural Elements," NP-6515(1957), Holmes and Narver, Inc.

27. Revision 1 to Regulatory Guide 1.12, "Instrumentation for Earthquakes", Draft 3, dated January 10, 1974.

28. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code," Sections II, III, VIII and XI, with Addenda through Summer 1973, United Engineering Center, 345 East 47th Street, New York, N. Y. 10017.

29. American Society for Testing and Materials, 1916 Race Street, Philadelphia, Pa. 19103.

   - ASTM Specification, A-262-70, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Annual Book of ASTM Standards, Part 3, April 1973.

   - ASTM Specification E-185-73, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, July 1973.

   - ASTM Specification E-208-69, "Standard Method for Conducting Dropweight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, July 1973.

   - ASTM Specification E-23-72, "Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, Part 31, July 1973.

30. American National Standards Institute, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors," ANSI N45.4-1972, March 16, 1972, 1430 Broadway, New York, N. Y. 10018.

31. ASME Specification, SA-370-71b, "Methods and Definitions for Mechanical Testing of Steel Products," ASME Boiler and Pressure Vessel Code, Section II, Part A - Ferrous, 1971 Edition, Summer and Winter, 1972 Addenda.

32. AEC Guideline Document, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspections," January 21, 1969.

33. United States Atomic Energy Commission, "Final Environmental Statement Concerning Proposed Rule Making Action: Numerical Guides for Design Objectives and Limiting Conditions to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water Cooled Nuclear Power Reactor Effluents," WASH 1258, USAEC, Washington, D. C., July 1973.

34. BAW-10054, Topical Report (Proprietary) "Fuel Densification Report" January 1973.

35. "Technical Report on Densification of Babcock & Wilcox Reactor Fuels," USAEC Regulatory staff, July 6, 1973.

36. BAW-10001, Topical Report, "Incore Instrumentation Test Program, August 1969.

37. BAW-10010, Part 1, Topical Report, "Stability Margin for Xenon Oscillations Model Analysis," August 1969.

38. BAW-10010, Part 2, Topical Report, "Stability Margin for Xenon Oscillations - One Dimensional Digital Analysis," February 1970.

39. BAW-10010, Part 3, Topical Report, "Stability Margin for Xenon Oscillations - Two and Three Dimensional Analysis," April 1970.

40. Final Safety Analysis Report, Browns Ferry Nuclear Plant, Docket Nos. 50-259, 50-260 and 50-296.

41. USAEC "Guide for the Planning of Preoperational Testing Programs", December 7, 1970.

42. USAEC "Guide for the Planning of Initial Startup Programs", December 17, 1970.

43. ANSI N18.1-1971, "Standard for Selection and Training of Personnel for Nuclear Power Plants".

44. ANSI N18.7-1972 (ANS 3.2), "Standard for Administrative Controls for Nuclear Power Plants".

45. American National Standards Institute, 1430 Broadway, New York, N. Y. 10018.

- ANSI B31.1.0-1967, "Power Piping."

- ANSI B96.1-1967, Specification for Welded Aluminum - Alloy Field-Erected Storage Tanks.

46. American Petroleum Institute, 1801 K Street, N.W., Washington, D. C. 20006.

- API Standard 620, Fourth Edition, February 1970, "Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks."

- API Standard 650, Third Edition, July 1966, "Welded Steel Tanks for Oil Storage."

47. Tubular Exchanger Manufacturers Association, Standards, Fifth Edition, 1968, 331 Madison Avenue, New York, N. Y., 10017.

48. American Water Works Association, AWWA D100-67, "AWWA Standard for Steel Tanks-Standpipes, Reservoirs, and Elevated Tanks for Water Storage," Two Park Avenue, New York, N.Y. 10016.

49. L. C. Richardson, L. J. Finnegan, R. J. Wagner, and J. M. Waage, "CONTEMPT A Computer Program for Predicting the Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," IDO-17220, Phillips Petroleum Company, June 1967.

50. D. C. Slaughterbeck, "Comparison of Analytical Techniques Used to Determine Distribution of Mass and Energy in the Liquid and Vapor Regions of a PWR Containment Following a Loss-of-Coolant Accident," Special Interim Report, Idaho Nuclear Corporation, January 1970.

51. R. C. Schmitt, G. E. Bingham and J. A. Norberg, "Simulated Design Basis Accident Tests of the Carolina Virginia Tube Reactor Containment - Final Report," IN-1403, Idaho Nuclear Corporation, December 1970.

52. D. C. Slaughterbeck, "Review of Heat Transfer Coefficients for Condensing Steam in a Containment Building Following a Loss-of-Coolant Accident," IN-1388, Idaho Nuclear Corporation, September 1970.

53. T. Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1),"

(cont'd) 54. Prepared for the National Reactor Testing Station, February 28, 1966, (Unpublished work).

55. H. Uchida, A. Oyama and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," in Proceeding of the Third International Conference on the Peaceful Uses of Atomic Energy Held in Geneva, August 31-September 9, 1964, Volume 13, Session 3.9, New York, United Nations 1965, (A/Conf. 28/P/436) (May 1964) pp. 93-104.

56. "FLOOD/MOD002 - A Code to Determine the Core Reflood Rate for a PWR Plant with 2 Core Vessel Outlet Legs and 4 Core Vessel Inlet Legs," Interim Report, Aerojet Nuclear Company, November 2, 1972.

57. W. H. Rettig, G. A. Jayne, K. V. Moore, C. E. Slater, and M. L. Uptmor, "RELAP3 - A Computer Program for Reactor Blowdown Analysis," IN-1321, Idaho Nuclear Corporation, June 1970.

58. F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Vol. 87, Pg. 134, Journal of Heat Transfer, February 1965.

59. L. F. Parsly, "Design Considerations of Reactor Containment Spray Systems, Part VI, The Heating of Spray Drops in Air-Steam Atmospheres," USAEC Report ORNL-TM-2412, January 1970.

60. H. F. Coward, G. W. Jones, "Limits of Flammability of Gases and Vapors," Bureau of Mines Bulletin 503, 1952.

61. A. O. Allen, "The Radiation Chemistry of Water and Aqueous Solutions," D. Van Nostrand Co., Inc., 1961.

62. ANS Standard ANS-5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fuel Thermal Reactors (DRAFT)," American Nuclear Society, Hinsdale, Illinois, October 1971.

63. Institute of Electrical and Electronics Engineers, 345 East 47th Street, New York, N. Y., 10017.

- IEEE Std 279-1971 - "Criteria for Protection Systems for Nuclear Power Generating Stations."
- IEEE Std 308-1971 - "Class IE Electric Systems for Nuclear Power Generating Stations."
- IEEE Std 317-1971 - "IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations."
- IEEE Std 323-1971 - "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations."

- IEEE Std 336-1971 - "IEEE Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations."

- IEEE Std 338-1971 - "Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems."

- IEEE Std 344-1971 - "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations."

64. Sections 6, 7 and 8 of the PSAR for Rancho Seco Unit 1.

65. Safety Evaluation Report - Rancho Seco Unit 1.

66. "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2," The American Society of Mechanical Engineers, New York, 1969.

67. U S A Standard B 16.5-1968, "Steel Pipe Flanges and Flanged Fittings," The American Society of Mechanical Engineers, New York, 1968.

68. U S A Standard B 16.9-1964, "Wrought Steel Buttwelding Fittings," The American Society of Mechanical Engineers, New York, 1964.

69. U S A Standard B 31.1.0-1967, "Power Piping," The American Society of Mechanical Engineers, New York, 1967.

70. U S A Standard B 31.7-1969, "Nuclear Power Piping", The American Society of Mechanical Engineers, New York, 1969.

71. U S A Standard B 36.10-1959, "Wrought-Steel and Wrought-Iron Pipe," The American Society of Mechanical Engineers, New York, 1959.

72. MSS Standard Practice SP-58, "Pipe Hangers and Supports-Materials and Design," Manufacturers Standardization Society, Arlington, Virginia, 1959.

73. MSS Standard Practice SP-66, "Pressure-Temperature Ratings for Steel Butt-Welding End Valves," Manufacturers Standardization Society, New York, 1959.

74. American Nuclear Society Standard 20.2 (draft), "Design Basis for Protection Against Pipe Whip," June 1973.

75. BAW-10043, Supplement 1, "Overpressurization Protection for Babcock and Wilcox Pressurized Water Reactors with 205 Fuel Assemblies, December 1973.

76. BAW-10065, Supplement 1, "Multinode Analysis of B&W's 205-Fuel-Assembly (Mark C Design) Nuclear Plant During a Loss-of-Coolant Accident", August 1973.

77. BAW-10074, "Multinode Analysis of Small Breaks for B&W's 205 Fuel-Assembly Nuclear Plants with Internals vent valves", November 1973.

78. Bollinger, G. A., "Seismicity of the Southeastern United States," Bulletin of the Seismological Society of America, 1793, No. 63, p. 1785.

79. Dutton, C. E., "The Charleston Earthquake of August 31, 1886," United States Geological Survey Ninth Annual Report, 1887-1888.

APPENDIX A

Chronology of Safety Review

May 14, 1973	Application tendered for acceptance review.
June 14, 1973	Letter to applicant accepting application, advising to docket and requesting additional information. (Sections 2-7, 9-12, 16 & 17 of the PSAR).
June 21, 1973	Application and PSAR docketed.
July 5, 1973	Letter to applicant requesting additional information. Sections 2, 3, 5 & 6 of the PSAR).
July 20, 1973	Amendment #1 docketed.
August 3, 1973	Meeting with applicant to discuss containment pressure analysis.
August 7, 1973	Amendment #2 docketed.
September 12, 1973	Letter to applicant requesting additional information. (Sections 4-6, 11-14 & 16 of the PSAR).
September 14, 1973	Meeting with applicant to discuss high energy line breaks outside of containment.
September 14, 1973	Letter to applicant requesting additional information. (Sections 3-6 & 15 of the PSAR).
September 18, 1973	Letter to applicant requesting additional information. (Sections 2, 11 & 12 of the PSAR).
September 18, 1973	Amendment #3 docketed.
September 18, 1973	Meeting with applicant to discuss the structural design of containment.
September 19, 1973	Site visit and meeting with the applicant.
September 27, 1973	Letter to applicant requesting additional information. (Sections 7 & 8 of the PSAR).
October 2, 1973	Letters to applicant requesting additional information. (Sections 2, 6, 9 & 17 of the PSAR).
October 3, 1973	Letter to applicant requesting additional information. (Sections 2 of the PSAR).

October 4, 1973	Meeting with the applicant to discuss Regulatory Guides 1.46, 1.48 and 1.52.
October 12, 1973	Amendment #4 docketed.
October 15, 1973	Letter to applicant requesting additional information. (Section 6 of the PSAR).
October 23, 1973	Letter to applicant requesting additional information. (Sections 3, 9 & 10 of the PSAR).
October 31, 1973	Meeting with applicant to discuss Regulatory Guides 8.8 and 1.42.
November 6, 1973	Meeting with the applicant to discuss subcompartment differential pressure analysis.
November 9, 1973	Amendment #5 docketed.
November 19, 1973	Amendment #6 docketed.
November 20, 1973	Meeting with the applicant to discuss alternate intake designs.
December 3, 1973	Amendment #7 docketed.
December 4, 1973	Letter to the applicant requesting the degree of conformance with the staff document, "Physical Independence of Electric Systems".
December 7, 1973	Letter to the applicant requesting additional information. (Sections 7, 8 & 15 of the PSAR).
December 14, 1973	Letter to the applicant transmitting staff positions and requesting additional information. (Sections 2, 6, 9 & 12 of the PSAR).
December 27, 1973	Letter to the applicant transmitting staff positions and requesting additional information. (Section 2 of the PSAR).
January 4, 1974	Letter to the applicant transmitting staff positions and requesting additional information. (Sections 3-6, 13 & 15 of the PSAR).
January 11, 1974	Letter to the applicant transmitting staff positions and requesting additional information. (Sections 7, 8 & 17 of the PSAR).

January 16, 1974	Letter to the applicant requesting additional information. (Section 6 of the PSAR).
January 21, 1974	Amendment #8 docketed.
January 25, 1974	Letter to the applicant requesting additional information. (Sections 3, 8 & 9 of the PSAR).
January 28, 1974	Letter to the applicant requesting additional information. (Section 2 of the PSAR).
February 5, 1974	Letter to the applicant requesting financial information.
February 6, 1974	Meeting with the applicant to discuss slope stability of intake canal.
February 20, 1974	Amendment #9 docketed.
March 5, 1974	Amendment #10 docketed.
March 7, 1974	Letter to the applicant requesting additional information. (Section 7 of the PSAR).
March 8, 1974	Generic meeting with B&W to discuss containment pressure analysis.
March 15, 1974	Generic meeting with B&W to discuss the reactor protection system.
March 28, 1974	Letter to the applicant requesting additional information (Section 6 of the PSAR).
April 10, 1974	Letter to the applicant concerning the reactor protection system.
April 19, 1974	Meeting with the applicant to discuss QA organization.
April 26, 1974	Letter to the Applicant transmitting staff position (Section 17 of the PSAR).
May 2, 1974	Meeting with the applicant to discuss items unresolved for the Safety Evaluation Report.
May 3, 1974	Meeting with the applicant to discuss QA organization.
May 9, 1974	Letter to the applicant transmitting staff positions (Section 5 of the PSAR).

APPENDIX B
TENNESSEE VALLEY AUTHORITY
FINANCIAL ANALYSIS

DOCKET NOS. 50-438 and 50-439
(dollars in millions)

	Year Ended 6-30-73	Year Ended 6-30-72	Year Ended 6-30-71
Long-term debt	$1,775.0	$1,225.0	$ 675.0
Utility plant (net)	4,567.6	4,166.0	3,635.9
Ratio - debt to fixed plant	.39	.29	.19
Utility plant (net)	4,567.6	4,166.0	3,635.9
Capitalization	4,296.0	3,692.9	3,083.0
Ratio of net plant to capitalization	1.06	1.13	1.18
Investment and surplus	2,521.0	2,467.9	2,408.0
Total assets	5,051.9	4,583.1	3,993.8
Ratio of invest. and surplus to total assets	.50	.54	.60
Net income	106.4	112.1	119.0
Investment and surplus	2,521.0	2,467.9	2,408.0
Rate of earnings on invest. and surplus	4.2%	4.5%	4.9%
Net income before interest	245.8	212.5	196.7
Utility plant (net)	4,567.6	4,166.0	3,635.9
Rate of earnings on net plant	5.4%	5.1%	5.4%
Net income before interest	245.8	212.5	196.7
Interest on long-term debt	111.4	69.0	48.6
No. of times long-term interest earned	2.2	3.1	4.0
Net income	106.4	112.1	119.0
Total revenues	823.2	693.8	646.2
Net income ratio	.13	.16	.18
Total utility operating expenses	577.4	481.3	449.5
Total utility operating revenues	749.4	641.9	598.0
Operating ratio	.77	.75	.75
Utility plant (gross)	5,808.2	5,321.6	4,710.4
Utility operating revenues	749.4	641.9	598.0
Ratio of plant investment to revenues	7.8	8.3	7.9

	6-30-73		6-30-72	
	Amount	% of Total	Amount	% of Total
Capitalization:				
Long-term debt	$1,775.0	41.3%	$1,225.0	33.2%
Investment	1,697.3	39.5	1,696.9	45.9
Surplus	823.7	19.2	771.0	20.9
Total	$4,296.0	100.0%	$3,692.9	100.0%

NOTE: Amounts derived from balance sheets reflect all operations of TVA and those derived from income statements reflect TVA's power program only.